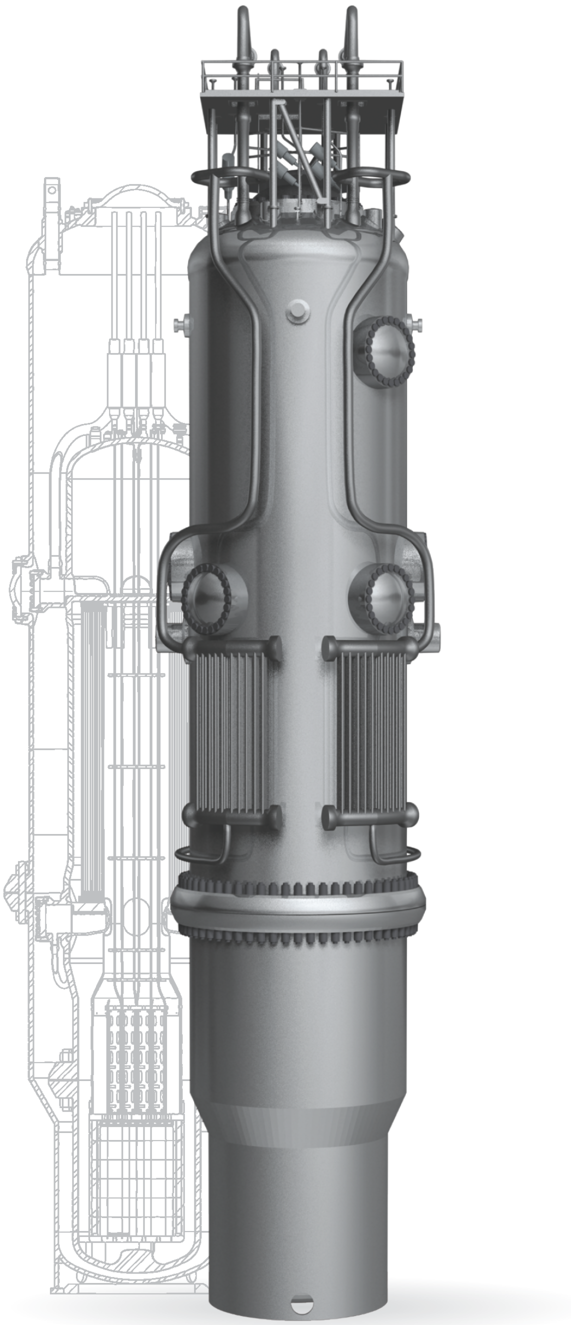

NuScale Standard Plant
Design Certification Application

Applicant's Environmental Report - Standard Design Certification

PART 3

Revision 0
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1.0 Introduction

The purpose of this report is to document the evaluation of severe accident mitigation design alternatives (SAMDAs) associated with the NuScale Power Plant design and the bases for not incorporating SAMDAs in the design to be certified. The SAMDA evaluation described in this report was performed in support of the NuScale Design Certification Application.

The U.S. Court of Appeals decision in Limerick Ecology Action versus NRC, 869 F.2d 719 (3rd Cir. 1989), has effectively required the NRC to include consideration of SAMDAs in the environmental impact review performed under Section 102(2)(c) of the National Environmental Policy Act of 1969 (NEPA). This report forms the basis of the NRC's environmental assessment of the NuScale Power Plant design, per 10 CFR 51.30(d), to assure compliance with Section 102(2)(c) of the NEPA of 1969. 10 CFR 52.47(b)(2) and 10 CFR 51.55(a) also require the submission of this environmental report for standard design certifications.

The SAMDA analysis is a cost-benefit analysis wherein the costs of modifying the nuclear power plant's design is weighed against the potential financial risk of consequences stemming from a possible severe accident. Nuclear Energy Institute (NEI) NEI 05-01 (Reference 8.1-1) and NUREG/BR-0184 (Reference 8.1-2) provide guidance for performing the SAMDA analysis. These guidance documents are cited throughout this report. The Environmental Standard Review Plan (NUREG-1555) Section 7.3 (Reference 8.1-29) provides guidance for reviewing severe accident mitigation alternatives (SAMAs). Note that the term SAMA is used when making reference to similar analyses performed by operating reactors and the terms SAMA and SAMDA will be used interchangeably at times in the context of this report, particularly in Appendix A.

2.0 Abbreviations and Definitions

Table 2-1: Acronyms and Abbreviations

Term	Definition
AAPS	auxiliary AC power source
AC	alternating current
AOC	averted offsite property damage costs
AOE	averted occupational exposure
AOSC	averted onsite costs
APD	avoided public dose
APE	averted public exposure
APL	actuation and priority logic
ATWS	anticipated transient without scram
BDG	backup diesel generator
BPSS	backup power supply system
BWR	boiling water reactor
CCDF	conditional core damage frequency
CCF	common cause failure
CDF	core damage frequency
CES	containment evacuation system
CFDS	containment flooding and drain system

Table 2-1: Acronyms and Abbreviations (Continued)

Term	Definition
CFWS	condensate and feedwater system
CIS	containment isolation signal
CLRF	conditional large release frequency
CNTS	containment system
CNV	containment vessel
COE	cost of enhancement
CPI	Consumer Price Index
CRDS	control rod drive system
CST	condensate storage tank
CTG	combustion turbine generator
DC	direct current
DHRS	decay heat removal system
DWS	demineralized water system
ECCS	emergency core cooling system
EEM	external events multiplier
EIM	equipment interface module
EOP	emergency operating procedure
ESFAS	engineered safety features actuation system
FGR	Federal Guidance Report
FWIV	feedwater isolation valve
HVAC	heating ventilation and air conditioning
IE	internal events
LOCA	loss of coolant accident
LOOP	loss of offsite power
LPSD	low power and shutdown
LRF	large release frequency
MPS	module protection system
MSIV	main steam isolation valve
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
PD	population dose
PPI	producer price index
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RC	release category
RCCWS	reactor component cooling water system
RCS	reactor coolant system
RHR	residual heat removal
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RSV	reactor safety valve
RTB	reactor trip breaker
RTS	reactor trip system
RVV	reactor vent valve
SAI	severe accident impact

Table 2-1: Acronyms and Abbreviations (Continued)

Term	Definition
SAMA	severe accident mitigation alternative
SAMDA	severe accident mitigation design alternative
SBO	station blackout
SGTF	steam generator tube failure
SMA	seismic margin analysis
SOARCA	State-of-the-Art Reactor Consequence Analysis
SOV	solenoid-operated valve
SRV	safety relief valve
SVM	scheduling and voting module
TCD	time to core damage
UHS	ultimate heat sink

Table 2-2: Definitions

Term	Definition
C	present value factor
D_{IO}	immediate occupational dose
D_{LTO}	long term occupational dose
F	core damage frequency
m	on-site cleanup period
Maximum Benefit	The benefit that a SAMDA could achieve if it eliminated all risk. Or in other words, the monetary value attributed to eliminating all risk from the NuScale design.
PV_{CD}	present value of the cost of cleanup and decontamination
PV_{RP}	net present value of replacement power for a single event
r	real discount rate (taken as some percent per year)
R	monetary value associated with unit of dose
t_f	years remaining until end of facility life
t_i	years before facility begins operation
U_{CD}	present value of the cleanup and decontamination costs over the remaining life of the facility
W_{IO}	monetary value of occupational health risk avoided due to "immediate" doses, after discounting
W_{LTO}	monetary value of long term dose risk avoided, accounting for present value factor

3.0 Approach and Methodology Overview

This report consists of identifying the potential causes and impacts of severe accidents, identifying SAMDAs to prevent or mitigate severe accidents, evaluating the costs and benefits associated with the SAMDAs, and documenting whether or not particular SAMDAs were judged cost-beneficial to incorporate into the NuScale Power Plant design. NuScale's SAMDA analysis methodology is based on the approaches suggested in NEI 05-01.

The process to identify SAMDA candidates for the NuScale design used generic industry SAMDAs found in NEI 05-01. Top cutsets and risk significant events from NuScale probabilistic

risk assessment (PRA) models were used to assist in the development of NuScale Power Plant specific SAMDAs. This identification process is described in Section 4.0.

Rather than begin by explicitly calculating the benefits associated with a particular SAMDA, a simpler conservative analysis approach is taken by calculating "the maximum benefit". The maximum benefit is the benefit that a SAMDA could achieve if it eliminated all risk. Or in other words, the maximum benefit is the monetary value attributed to eliminating all risk from the NuScale design. The derivation of the maximum benefit is described in Section 5.0 and relies on inputs from severe accident modeling calculations to assess offsite consequences of severe accidents. Key assumptions of these offsite consequence analyses are given in Appendix B. The calculated offsite effects are based on data associated with the Surry Power Station, which was seen as a reasonably representative site for the purposes of this report. A moderate response from the local population is modeled, per Section B.1.7 description of offsite consequence modeling. The calculation of the maximum benefit accounts for the effect of seismic events and multiple modules. A series of sensitivity studies on the maximum benefit (including the use of Peach Bottom instead of Surry data to assess offsite consequences) are described in Section 5.8.

The SAMDA candidates are then qualitatively screened into one of seven possible categories. The intent of the screening is to identify the SAMDA candidates that warrant a detailed cost-benefit evaluation or to provide a basis for why a particular SAMDA does not require further consideration. Part of the screening process involves comparing the estimated cost associated with a SAMDA to the maximum benefit. If the estimated cost associated with a SAMDA is higher than the maximum benefit, then the SAMDA is not cost-beneficial to incorporate into the NuScale design. The screening process is described in Section 6.0 and the results are shown in Appendix A.

A summary of results and conclusions are provided in Section 7.0 of this report.

4.0 SAMDA Candidate Identification

The first step in the SAMDA analysis process is to identify a robust list of possible candidate SAMDAs to analyze. This section describes the SAMDA identification process.

The list of industry standard SAMDAs for pressurized water reactors (PWRs) found in NEI 05-01 are identified as candidates and shown as SAMDA ID numbers 1 through 153 in Appendix A of this report. The list of industry standard SAMDAs for boiling water reactors (BWRs) found in NEI 05-01 were reviewed, but found to add no additional value compared to the list of PWR SAMDAs already considered. Therefore, the list of industry standard SAMDAs for BWRs is not formally dispositioned or listed in this report.

NuScale design-specific SAMDA candidates are identified through the use of PRA insights. The PRA, as discussed in Chapter 19 of the NuScale Final Safety Analysis Report, identifies various different combinations of events that could lead to radiological releases from the NuScale Power Plant and each unique combination of events is referred to as a cutset.

Section 5 of NEI 05-01 recommends the examination of dominant accident sequences, or cutsets, in addition to dominant equipment or human failures based on importance measures. This recommendation is accomplished by the following:

- Examination of basic events from the Level 1 and Level 2 NuScale PRAs deemed risk significant through comparison to the NuScale risk significance criteria to determine significant equipment and human failures, and
- Examination of Level 1 and Level 2 cutsets that contribute to more than one percent of the core damage frequency (CDF) or large release frequency (LRF) of each individual PRA, to determine significant equipment failures for that internal or external hazard. Cutsets that are above this one percent cutoff are referred to as "top cutsets" throughout this report.

Top cutsets and risk significant events from the following PRAs are used as the basis for SAMDA identification:

- internal events (during full power operation)
- internal fires
- internal flooding
- low power and shutdown (LPSD)
- external flooding
- high winds

Insights into the dominant contributors to seismic risk were obtained by calculating seismic CDF cutsets for a NuScale Power Module (NPM) located at the Surry and Peach Bottom sites with Surry as the base case and Peach Bottom as a sensitivity study. Seismic risk and potential SAMDAs are discussed in Section 4.1.13 and seismic CDFs are reported in Table B-20.

Risk significance criteria for the NuScale Power Plant design are presented in Table 4-1. These risk significance criteria are described in the NuScale topical report TR-0515-13952-NP-A (Reference 8.1-3).

Table 4-1: NuScale Risk-Significance Criteria

Parameter	Risk Significant Criteria	Notes
Component-level basic event	CCDF ≥ 3 x 10^{-6}/yr	Conditional CDF (CCDF) or conditional LRF (CLRF) that would result if the component always fails to operate in response to a plant upset
Component-level basic event	CLRF ≥ 3 x 10^{-7}/yr	
System-level basic event	CCDF ≥ 1 x 10^{-5}/yr	CCDF or CLRF that would result if the system level event always fails to operate in response to a plant upset
System-level basic event	CLRF ≥ 1 x 10^{-6}/yr	
Basic event / contributor	Total FV ≥ 0.20	Percent contribution to CDF or LRF; applies to individual components and human actions

NuScale design specific SAMDAs that are identified by examining the top cutsets and risk significant basic events are numbered from 154 to 203. These SAMDAs are shown in Appendix A.

The basic events from the NuScale PRAs that meet the Table 4-1 criteria or are in the PRA top cutsets are presented in Table 4-2. In many cases, the same basic event is risk significant or in the top cutsets of multiple PRAs. In these cases, the basic event is presented once in Table 4-2. Some basic events, such as human error probabilities (in contrast to the actual human failure events, which are included), are not included in this list as these events do not describe equipment or human failures but are modeling conveniences used in the cutset quantification. SAMDA numbers relevant to each basic event are listed. In some cases, general PWR SAMDAs were sufficient to address the basic event and no NuScale specific SAMDAs were added. However, in the majority of cases, new SAMDAs were developed. The process for creating these NuScale specific SAMDAs is described in Section 4.1.1 through Section 4.1.13.

The dominant contributor to LPSD risk was found to be reactor building crane failure and potential SAMDAs related to reactor building crane failure are discussed in Section 4.1.11. Reactor building crane failure is not shown in Table 4-2 because it is evaluated separately from the LPSD internal events model. No specific components are modeled in the module drop portion of the LPSD PRA. The module drop portion of the LPSD PRA is modeled as a crane failure initiating event combined with a probability of the module remaining upright following the drop. A reactor building crane specific PRA was performed to establish the frequency of the module drop event. Insights from this reactor building crane PRA are used to derive SAMDAs meant to reduce the probability of crane failure.

Table 4-2: NuScale Basic Events Used for SAMDA Generation

Basic Event Description	SAMDA ID	System	PRA	CCDF	CLRF	FVCDF	FVLRF	Top CDF Cutsets	Top LRF Cutsets
Common Cause Failure (CCF) of two of two CES containment isolation valves fail to close	SAMDA 183	CNTS	H-W	-	-	-	X	-	X
			E-Flood	-	-	-	X	-	X
CFDS isolation valve fails to open	SAMDA 156	CFDS	IE	-	-	-	-	-	X
			LPSD	-	-	-	-	-	X
Operator fails to initiate CFDS injection	SAMDA 157	CFDS	IE	-	-	-	X	X	X
			LPSD	-	-	-	X	-	X
CFDS containment isolation valve one fails to open	SAMDA 156	CNTS	IE	-	-	-	-	-	X
			LPSD	-	-	-	-	-	X
CFDS containment isolation valve two fails to open	SAMDA 156	CNTS	IE	-	-	-	-	-	X
			LPSD	-	-	-	-	-	X
Given actuation, at least three of 16 rods fail to insert	SAMDA 130 SAMDA 185	CRDS	IE	-	-	-	-	X	-
			I-Fires	-	-	-	-	X	-
			I-Flood	-	-	-	-	X	-
CVCS makeup combining valve fails to open	SAMDA 158	CVCS	IE	-	-	-	-	X	-
			I-Fires	-	-	-	-	X	-
			LPSD	-	-	-	-	X	-
Operator fails to initiate CVCS injection	SAMDA 159	CVCS	IE	-	-	-	-	X	X
			I-Fires	-	-	-	-	X	-
			LPSD	-	-	X	-	X	-
CVCS discharge containment isolation valve one fails to close	SAMDA 184	CNTS	IE	-	-	-	X	-	-
			I-Fires	-	-	-	X	-	-
CVCS discharge containment isolation valve two fails to close	SAMDA 184	CNTS	IE	-	-	-	X	-	-
			I-Fires	-	-	-	X	-	-
CCF of two of two CVCS discharge containment isolation valves fail to close	SAMDA 184	CNTS	IE	-	-	-	X	X	X
			I-Fires	-	-	-	X	-	X
			H-W	-	-	-	X	-	X
			E-Flood	-	-	-	X	-	X
			LPSD	-	-	-	X	-	X
CVCS LOCA does not initiate excess flow check valve	SAMDA 184	CNTS	IE	-	-	-	X	X	X
			LPSD	-	-	-	X	-	X

Table 4-2: NuScale Basic Events Used for SAMDA Generation (Continued)

Basic Event Description	SAMDA ID	System	PRA	CCDF	CLRF	FVCDF	FVLRF	Top CDF Cutsets	Top LRF Cutsets
Actuation valve A for DHRS train one fails to open	SAMDA 173 SAMDA 177 SAMDA 178	DHRS	I-Flood	-	-	X	-	-	-
Actuation valve B for DHRS train one fails to open	SAMDA 173 SAMDA 177 SAMDA 178	DHRS	I-Flood	-	-	X	-	-	-
CCF of two of two DHRS train one actuation valves fail to open	SAMDA 172 SAMDA 173 SAMDA 177 SAMDA 178	DHRS	I-Flood	-	-	X	-	-	-
CCF of four of four DHRS actuation valves fail to open	SAMDA 172 SAMDA 173 SAMDA 177 SAMDA 178	DHRS	IE	-	-	-	-	X	-
			I-Flood	-	-	X	-	X	-
			LPSD	-	-	-	-	X	-
Actuation valve A for DHRS train two fails to open	SAMDA 173 SAMDA 177 SAMDA 178	DHRS	I-Flood	-	-	X	-	-	-
Actuation valve B for DHRS train two fails to open	SAMDA 173 SAMDA 177 SAMDA 178	DHRS	I-Flood	-	-	X	-	-	-
CCF of two of two DHRS train two actuation valves fail to open	SAMDA 172 SAMDA 173 SAMDA 177 SAMDA 178	DHRS	I-Flood	-	-	X	-	-	-
CCF of two of two DHRS heat exchangers plugging	SAMDA 174 SAMDA 175 SAMDA 178	DHRS	I-Flood	-	-	-	-	X	-
DWS north reactor building CVCS pump isolation valve fails to open	SAMDA 160	CVCS	IE	-	-	-	-	X	-
			I-Fires	-	-	-	-	X	-
			LPSD	-	-	-	-	X	-

Table 4-2: NuScale Basic Events Used for SAMDA Generation (Continued)

Basic Event Description	SAMDA ID	System	PRA	CCDF	CLRF	FVCDF	FVLRF	Top CDF Cutsets	Top LRF Cutsets
ECCS reactor vent valve A passive actuation to open valve fails	SAMDA 171	ECCS	IE	-	-	X	-	-	-
			I-Fires	-	-	X	-	-	-
			H-W	-	-	X	X	-	-
			E-Flood	-	-	X	X	-	-
ECCS reactor vent valve A fails to open	SAMDA 171	ECCS	I-Fires	-	-	X	-	-	-
			E-Flood	-	-	X	X	-	-
ECCS reactor vent valve B passive actuation to open valve fails	SAMDA 171	ECCS	IE	-	-	X	-	-	-
			I-Fires	-	-	X	-	-	-
			H-W	-	-	X	X	-	-
			E-Flood	-	-	X	X	-	-
ECCS reactor vent valve B fails to open	SAMDA 171	ECCS	I-Fires	-	-	X	-	-	-
			E-Flood	-	-	X	X	-	-
ECCS reactor vent valve C passive actuation to open valve fails	SAMDA 171	ECCS	IE	-	-	X	-	-	-
			I-Fires	-	-	X	-	-	-
			H-W	-	-	X	X	-	-
			E-Flood	-	-	X	X	-	-
ECCS reactor vent valve C fails to open	SAMDA 171	ECCS	I-Fires	-	-	X	-	-	-
			E-Flood	-	-	X	X	-	-
ECCS reactor recirculation valve A passive actuation to open valve fails	SAMDA 171	ECCS	IE	-	-	X	-	-	-
			I-Fires	-	-	X	X	-	X
			I-Flood	-	-	X	-	X	-
			H-W	-	-	X	X	X	-
			E-Flood	-	-	X	X	X	X
ECCS reactor recirculation valve A fails to open	SAMDA 171	ECCS	IE	-	-	X	-	-	-
			I-Fires	-	-	X	X	-	-
			I-Flood	-	-	X	-	-	-
			H-W	-	-	X	X	-	-
			E-Flood	-	-	X	X	-	-

Table 4-2: NuScale Basic Events Used for SAMDA Generation (Continued)

Basic Event Description	SAMDA ID	System	PRA	CCDF	CLRF	FVCDF	FVLRF	Top CDF Cutsets	Top LRF Cutsets
ECCS reactor recirculation valve B passive actuation to open valve fails	SAMDA 171	ECCS	IE	-	-	X	-	-	-
			I-Fires	-	-	X	X	-	X
			I-Flood	-	-	X	-	X	-
			H-W	-	-	X	X	X	-
			E-Flood	-	-	X	X	X	X
ECCS reactor recirculation valve B fails to open	SAMDA 171	ECCS	IE	-	-	X	-	-	-
			I-Fires	-	-	X	X	-	-
			I-Flood	-	-	X	-	-	-
			H-W	-	-	X	X	-	-
			E-Flood	-	-	X	X	-	-
CCF of two of two ECCS reactor recirculation valves fail to open	SAMDA 163 SAMDA 164 SAMDA 171	ECCS	IE	-	-	X	-	X	-
			I-Fires	-	-	X	X	X	X
			I-Flood	-	-	X	-	X	-
			H-W	-	-	X	X	X	X
			E-Flood	-	-	X	X	X	X
			LPSD	-	-	X	-	X	-
CCF of three of three ECCS reactor vent valves fail to open	SAMDA 165 SAMDA 166 SAMDA 171	ECCS	IE	-	-	X	-	X	-
			I-Fires	-	-	X	-	X	X
			I-Flood	-	-	-	-	X	-
			H-W	-	-	X	X	X	X
			E-Flood	-	-	X	X	X	X
			LPSD	-	-	X	-	X	-
ECCS RVV trip valve A fails to open	SAMDA 168 SAMDA 196	ECCS	I-Fires	-	-	X	-	-	-
ECCS RVV trip valve A fails due to hot short	SAMDA 168 SAMDA 190 SAMDA 196	ECCS	I-Fires	-	-	X	-	X	-
ECCS RVV trip valve B fails due to hot short	SAMDA 168 SAMDA 190 SAMDA 196	ECCS	I-Fires	-	-	-	-	X	-

Table 4-2: NuScale Basic Events Used for SAMDA Generation (Continued)

Basic Event Description	SAMDA ID	System	PRA	CCDF	CLRF	FVCDF	FVLRF	Top CDF Cutsets	Top LRF Cutsets
CCF of two of two ECCS RRV trip valves fail to open	SAMDA 167 SAMDA 168 SAMDA 196	ECCS	I-Fires	-	-	-	-	-	X
			I-Flood	-	-	-	-	X	-
			H-W	-	-	-	-	X	-
			E-Flood	-	-	-	-	X	-
CCF of three of three ECCS RVV trip valves fail to open	SAMDA 167 SAMDA 168 SAMDA 196	ECCS	I-Fires	-	-	X	-	-	-
Heat transfer to ultimate heat sink fails	SAMDA 169 SAMDA 170	ECCS	IE	X	-	-	-	-	-
			I-Fires	X	X	-	-	-	-
			I-Flood	X	-	-	-	X	-
			H-W	X	-	-	-	-	-
			E-Flood	X	-	-	-	X	-
Initiator is CCF of four of four EDSS DC buses to operate	SAMDA 5	EDSS	IE	-	-	-	-	X	-
			LPSD	-	-	-		X	
Combustion turbine generator fails to run (hours two through 48)	SAMDA 13 SAMDA 154 SAMDA 155	BPSS	IE	-	-	X	-	-	-
Combustion turbine generator fails to start	SAMDA 13 SAMDA 154 SAMDA 155	BPSS	IE	-	-	X	-	-	-
Combustion turbine generator unavailable due to test and maintenance	SAMDA 13 SAMDA 154 SAMDA 155	BPSS	IE	-	-	X	-	-	-
Offsite power not restored before batteries deplete	SAMDA 13 SAMDA 15 SAMDA 154 SAMDA 155	EHVS	IE	-	-	X	-	-	-
			H-W	-	-	X	X	X	X
CCF of two of two CFWS containment isolation valves fail to close	SAMDA 179 SAMDA 180	DHRS	I-Fires	-	-	-	-	X	-
CCF of four of four APL modules in DHRS actuation valves	SAMDA 196	MPS	I-Flood	-	-	-	-	X	-

Table 4-2: NuScale Basic Events Used for SAMDA Generation (Continued)

Basic Event Description	SAMDA ID	System	PRA	CCDF	CLRF	FVCDF	FVLRF	Top CDF Cutsets	Top LRF Cutsets
CCF of two of two APL modules in CVCS letdown isolation valves	SAMDA 196	MPS	IE	-	-	-	-	-	X
			I-Fires	-	-	-	-	-	X
CCF of two of two APL modules in top branch reactor trip breakers	SAMDA 196	MPS	I-Fires	-	-	-	-	X	-
			I-Flood	-	-	-	-	X	-
CCF of two of two APL modules in bottom branch reactor trip breakers	SAMDA 196	MPS	I-Fires	-	-	-	-	X	-
			I-Flood	-	-	-	-	X	-
Division I ESFAS CFDS EIM one fails to operate	SAMDA 197	MPS	IE	-	-	-	-	-	X
			LPSD	-	-	-	-	-	X
Division I ESFAS CFDS EIM two fails to operate	SAMDA 197	MPS	IE	-	-	-	-	-	X
			LPSD	-	-	-	-	-	X
Division II ESFAS CFDS EIM one fails to operate	SAMDA 197	MPS	IE	-	-	-	-	-	X
Division II ESFAS CFDS EIM two fails to operate	SAMDA 197	MPS	IE	-	-	-	-	-	X
			LPSD	-	-	-	-	-	X
Manual division I CIS override switch fails to close	SAMDA 198	MPS	IE	-	-	-	-	-	X
Manual division I ESFAS nonsafety enable switch fails to close	SAMDA 198	MPS	IE	-	-	-	-	-	X
Manual division II CIS override switch fails to close	SAMDA 198	MPS	IE	-	-	-	-	-	X
Manual division II ESFAS nonsafety enable switch fails to close	SAMDA 198	MPS	IE	-	-	-	-	-	X
CCF of three of four reactor vessel level process logic elements	SAMDA 195	MPS	H-W	-	-	-	-	X	-
			E-Flood	-	-	-	-	X	X
CCF of two of three division I ESFAS scheduling and voting modules	SAMDA 199	MPS	I-Fires	-	-	-	-	X	-
CCF of two of three division II ESFAS scheduling and voting modules	SAMDA 199	MPS	I-Fires	-	-	-	-	X	-
CCF of two of two main steam system containment isolation valves fail to close	SAMDA 181 SAMDA 182	DHRS	I-Fires	-	-	-	-	X	-

Table 4-2: NuScale Basic Events Used for SAMDA Generation (Continued)

Basic Event Description	SAMDA ID	System	PRA	CCDF	CLRF	FVCDF	FVLRF	Top CDF Cutsets	Top LRF Cutsets
RCS reactor safety valve A fails to reclose	SAMDA 193	RCS	IE	-	-	X	-	X	-
			I-Fires	-	-	X	X	X	-
			I-Flood	-	-	X	-	X	-
			LPSD	-	-	-	-	X	-
RCS reactor safety valve A fails to open	SAMDA 193	RCS	IE	-	-	X	-		-
			I-Fires	-	-	X	X	-	-
			I-Flood	-	-	X	-	X	-
RCS reactor safety valve B fails to open	SAMDA 193	RCS	I-Fires	-	-	X	X	-	-
			I-Flood	-	-	X	-	-	-
CCF of two of two RCS reactor safety valves fail to open	SAMDA 191 SAMDA 192 SAMDA 193	RCS	IE	-	-	X	-	X	-
			I-Fires	-	-	X	X	X	X
			I-Flood	-	-	X	-	X	-
			LPSD	-	-	-	-	X	-
CCF of three of four RCS vessel level sensors fail to operate on demand	SAMDA 194	MPS	H-W	-	-	-	-	X	-
			E-Flood	-	-	-	-	X	X
CCF of two of two top branch reactor trip breakers fail to open	SAMDA 132 SAMDA 137 SAMDA 186 SAMDA 187	RTS	I-Flood	-	-	-	-	X	-
CCF of two of two bottom branch reactor trip breakers fail to open	SAMDA 132 SAMDA 137 SAMDA 186 SAMDA 187	RTS	I-Flood	-	-	-	-	X	-

Note: The terms "H-W", "E-Flood", "IE", "LPSD", "I-Fires", and "I-Flood" refer to high winds, external floods, internal events, low power and shut down, internal fires, and internal flood PRAs, respectively.

4.1 SAMDA Generation

Design alternatives are grouped based on the system that owns the affected component. For example, although a containment evacuation system (CES) containment isolation valve is on the CES line, the component is a part of the containment system (CNTS). The SAMDAs postulated for events and systems with failures identified in Section 4.0 are discussed in Section 4.1.1 through Section 4.1.13.

4.1.1 Emergency Core Cooling System

The emergency core cooling system (ECCS) consists of five main valves: three reactor vent valves (RVVs) and two reactor recirculation valves (RRVs). The RVVs are located on top of the reactor pressure vessel (RPV) above the pressurizer. The RRVs are located on opposite sides of the RPV, above the RPV flange. The five ECCS valves are closed during normal operation, and are part of the reactor coolant pressure boundary. Upon ECCS actuation, and when the differential pressure across the valves is small enough, the valves open to allow flow paths between the RPV and the containment vessel (CNV). ECCS steady state operation is shown in Figure 4-1.

The ECCS valve design consists of a trip valve, a reset valve, an arming valve, and a main valve. The trip and reset valves are located outside of the CNV and are connected to the arming and main valves, which are located inside the CNV and outside of the RPV, through a vent line. The ECCS valve design is shown in Figure 4-2.

The ECCS is actuated by removing power to the trip valves corresponding to each main valve. Upon opening the trip valve, the main valve control chamber discharges through the vent line.

There are 18 basic events related to ECCS that are considered risk significant or are in the top cutsets. These events include the common cause failure (CCF) of the reactor recirculation valves (RRVs), the CCF of the reactor vent valves (RVVs), the CCF of the RRV trip valves, the CCF of the RVV trip valves, failure of any RVV to passively actuate, failure of any RRV to passively actuate, and the failure of the ECCS heat transfer to the ultimate heat sink (UHS).

The diversification of the existing ECCS valves or the addition of another, diverse valve are proposed design changes for the CCFs of the RRVs or RVVs (SAMDA 163, SAMDA 164, SAMDA 165, and SAMDA 166). To mitigate the trip valve failures, SAMDA 168 was identified to implement a manual bypass of the ECCS trip valve to allow the actuation of ECCS when the trip valves fail to operate.

Figure 4-1: Schematic Depicting Steady State ECCS Conditions

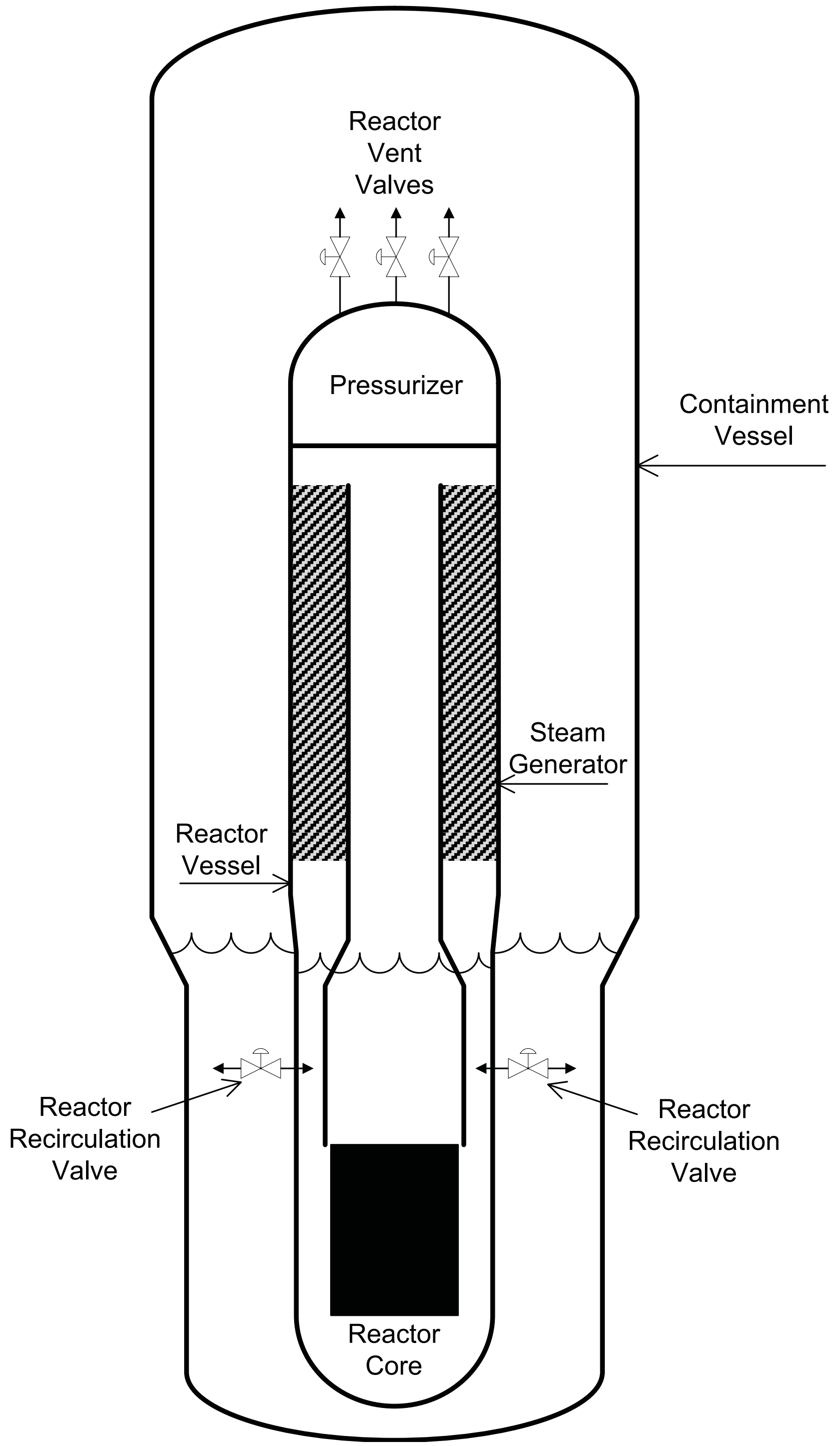
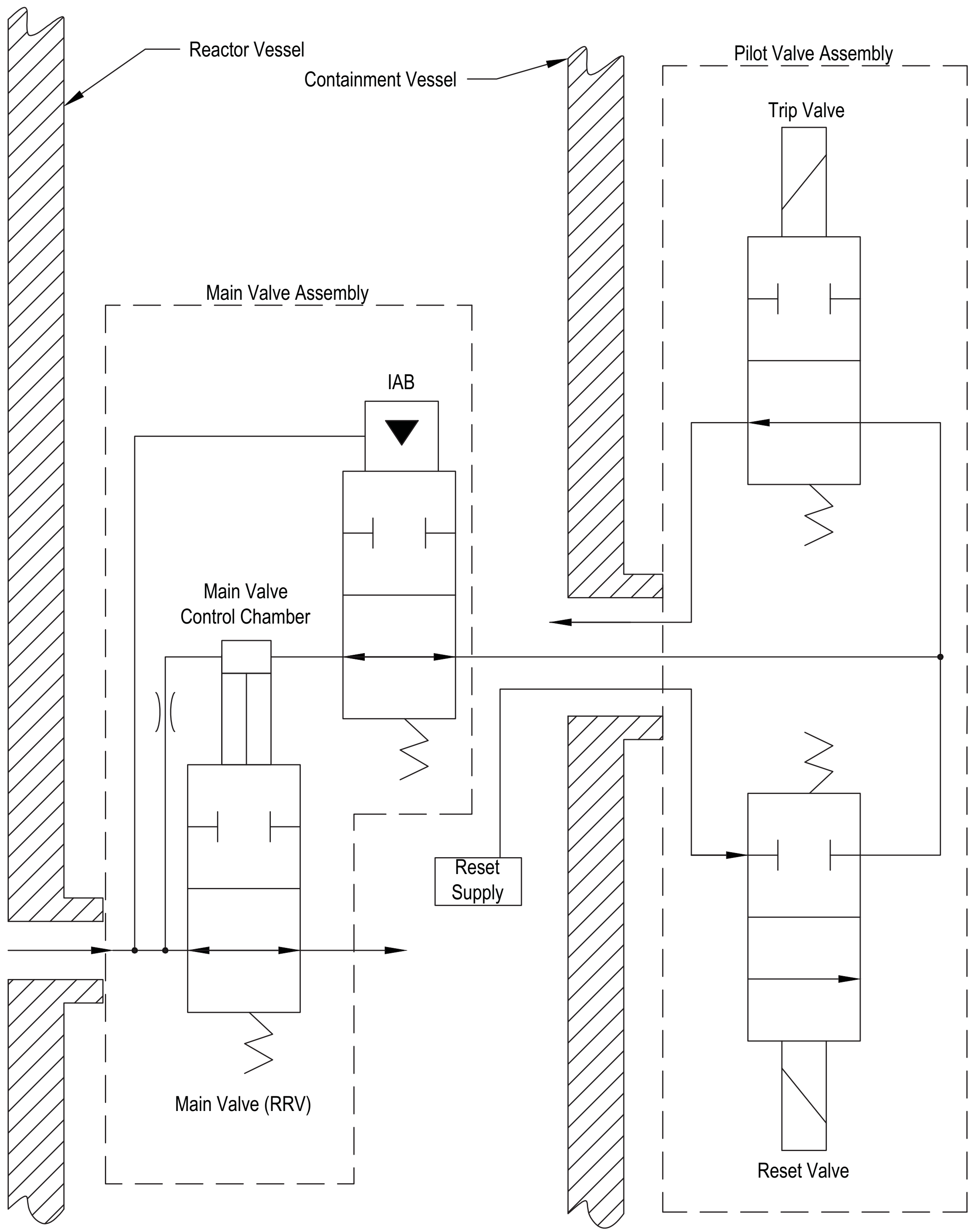


Figure 4-2: ECCS Valve Design



The context of these common cause ECCS valve failures is also important when considering the design alternatives. There are sequences that are on a success path and decay heat removal is being accomplished by other systems. Then, a partial actuation of ECCS due to loss of DC power occurs and leads to core damage. An ECCS partial actuation occurs when there is a CCF of three RVVs to open and the RRVs open successfully, or when there is a CCF of two RRVs to

open and the RVVs open successfully. To mitigate these events where ECCS is actuated due to failing open on loss of power when ECCS is not required, two SAMDAs are suggested. The first, SAMDA 3, suggests adding a battery charger or portable generator to the existing DC system. SAMDA 3 allows the ECCS valves to remain powered and not actuate. The second, SAMDA 167, suggests implementing block valves into the ECCS design to prevent the ECCS from actuating during a loss of DC power where the ECCS is not required to avoid core damage. These block valves would be normally open and manually closed to maintain pressure in the vent line upon recognition that ECCS is not required for core cooling prior to battery depletion and automatic actuation of ECCS upon loss of power is undesirable. If the ECCS were to completely actuate, rather than partially actuate, then core damage would be avoided. Suggested SAMDAs to actuate ECCS after a failure to open were considered in the previous paragraph.

Failure of heat transfer to the UHS could occur during a scenario where reactor coolant is lost by a flow path that bypasses the containment. When ECCS actuates, it is postulated that due to the low reactor coolant system (RCS) inventory, that even though reactor coolant is added to the CNV through the RRVs, the liquid level in containment does not reach the RRVs once the inventory in the RPV is depleted to the RRV elevation. This scenario would not allow natural circulation of the reactor coolant to the CNV and decay heat from the reactor core would not be removed through convection to the CNV, leading to core damage. To mitigate this failure mode, SAMDA 169 suggests lowering the RRV elevation to just above the top of the fuel. SAMDA 170 suggests redesigning the CNV to reduce the volume below the RRV level. These SAMDAs allow the ECCS recirculation to be effective with a smaller coolant inventory. SAMDA 157, discussed further in Section 4.1.8, would also mitigate this failure mode by adding coolant to the CNV through the containment flooding and drain system. Existing SAMDA 97 and SAMDA 98 also apply to this scenario, and suggest improvements to retain core debris in the event of a full core melt.

One additional design alternative, SAMDA 171, suggests improving testing and maintenance of the ECCS valves to improve the reliability of the system.

4.1.2 Containment Isolation

There are seven basic events related to containment isolation that are considered risk significant or are in the top cutsets. These basic events can be separated into two groups: the containment isolation valves fail to close or the containment isolation valves fail to open.

4.1.2.1 Containment Isolation Valves Fail to Close

There are five basic events that include or are affected by the containment isolation valves failing to close when demanded. These are the CCF of both CES containment isolation valves to close, the failure of either chemical and volume control system (CVCS) discharge line containment isolation valve to close, the CCF of both CVCS discharge line containment isolation valves to close, and the CVCS loss of coolant accident (LOCA) does not initiate the excess flow check valve.

SAMDA 183 suggests diversifying the existing CES containment isolation valves to reduce the probability of CCFs of these valves.

The CVCS containment isolation valve failures are dominated by the CCF of the discharge line containment isolation valves to close. SAMDA 184 suggests diversifying the existing CVCS

discharge line containment isolation valves to reduce the probability of CCFs of these valves. No SAMDAs are postulated specifically to mitigate the "CVCS LOCA does not initiate excess flow check valve" event. This event is not an equipment failure where the flow check valve does not function. The check valve is not demanded because the condition of the LOCA is such that the flow rate is too low to demand the check valve to close. Additionally, if the CVCS discharge line containment isolation valves close successfully then the check valve is not required to close, and therefore SAMDA 184 applies to this event.

4.1.2.2 Containment Isolation Valves Fail to Open

There are two basic events that include the containment isolation valves failing to open when demanded. These basic events include the failure of either containment flooding and drain system (CFDS) containment isolation valves to open. SAMDA 156 suggests improving the reliability of the CFDS containment isolation valves to reduce the probability of these failures to open.

4.1.3 Control Rod Drive System

One basic event related to the control rod drive system (CRDS) is considered risk-significant or is in the top cutsets, which is the failure of at least 3 of the 16 control rods to insert into the reactor core upon actuation of the CRDS. SAMDA 130 applies to this event and suggests adding an independent boron injection system to improve the availability of boron injection and control of reactivity during an anticipated transient without scram (ATWS). The reactor trip breakers that signal the CRDS are part of the reactor trip system and are discussed in Section 4.1.4.

One additional design alternative, SAMDA 185, suggests improving the testing and maintenance of the CRDS.

4.1.4 Reactor Trip System

Two basic events related to the reactor trip system (RTS) are considered risk significant or are in the top cutsets. These basic events include the CCF of the top branch of two reactor trip breakers (RTBs) to fail to open and the CCF of the bottom branch of two RTBs to fail to open.

SAMDA 132 applies to the failure of the RTS and suggests providing an additional control system for rod insertion to improve redundancy and reduce ATWS frequency. SAMDA 137 also applies to the RTS and suggests providing the capability to remove power from the bus powering the control rods to decrease the time required to insert control rods if the reactor trip breakers (RTBs) fail. The control rods are controlled by electromagnetic coils. When the power is removed from the coils, the control rods insert into the core. SAMDA 186 suggests providing an additional, diverse division of RTBs to reduce the probability of CCFs and improve the reliability of the RTS to insert the control rods when demanded.

One additional design alternative, SAMDA 187, suggests improving the testing and maintenance of the RTBs to improve the reliability of the RTS.

4.1.5 Decay Heat Removal System

Ten basic events related to the decay heat removal system (DHRS) are considered risk significant or are in the top cutsets. These basic events can be binned into three categories: the failure of the DHRS actuation valves to open, the failure of the DHRS due to the heat exchangers plugging, and the feedwater isolation valves (FWIV) or main steam isolation valves (MSIV) failing to close.

There are seven basic events that apply to the DHRS valves failing to open. These basic events include the failure of any of the four individual DHRS actuation valves to open (two valves for each of the two DHRS trains), the CCF of both DHRS actuation valves to open on train one of the DHRS, the CCF of both DHRS actuation valves to open on train two of the DHRS, and the CCF of all four DHRS actuation valves to open. These failures are dominated by the CCF of the DHRS valves to open (both the CCF of two valves for either division of DHRS and the CCF of all four DHRS valves in both divisions). SAMDA 172 suggests diversifying the DHRS actuation valves to improve the reliability of the DHRS and reduce the probability of CCFs of these valves to open.

The CCF of both trains of DHRS due to plugging is the one basic event that applies to the failure of the DHRS heat exchangers due to plugging. SAMDA 175 suggests adding removable screens to the DHRS, which would be cleaned every refueling cycle and reduce the probability that the DHRS fails due to plugging.

There are two basic events that apply to the failure of the FWIVs and MSIVs to close. These basic events are the CCF of both FWIVs to close (one on each DHRS train) and the CCF of both MSIVs to close (one on each DHRS train). If either the FWIVs or MSIVs fail to close, the DHRS inventory will be lost and the DHRS will not function. SAMDA 179 suggests providing a redundant, diverse FWIV on each DHRS train to reduce the probability of the CCF of these valves to close. SAMDA 181 suggests providing a redundant, diverse MSIV on each DHRS train to reduce the probability of the CCF of these valves to close.

SAMDA 178 suggests providing a backup means of removing decay heat to improve the availability of decay heat removal when the DHRS is not functional, which applies to all ten basic events.

Four additional design alternatives, SAMDA 173, SAMDA 174, SAMDA 180, and SAMDA 182 suggest procedural changes to improve the reliability of the DHRS.

4.1.6 Reactor Coolant System

There are four basic events related to the RCS that are considered risk significant or are in the top cutsets. These basic events include the failure of each reactor safety valve (RSV) to open, the CCF of both RSVs to open, and the failure of one RSV to reclose after opening.

The CCF of both RSVs to open contributes more to risk than the independent failure of each RSV to open individually. Therefore, two SAMDAs are postulated to reduce the probability of CCF of the RSVs to open. SAMDA 191 suggests diversifying the existing RSVs to reduce the probability of common cause failure. SAMDA 192 suggests providing an additional, diverse, RSV to provide further defense in depth and reduce the probability of common cause failure.

SAMDA 193 suggests improving the reliability of the RSV components, which would reduce the probability of failure for all four basic events.

One additional design alternative, SAMDA 161, suggests training reactor operators to open the CVCS discharge line containment isolation valves when the RSVs fail to open.

4.1.7 Backup Power Supply System

Three basic events related to the backup power supply system (BPSS), specifically the auxiliary AC power supply (AAPS, modeled in the PRA as a combustion turbine generator [CTG]), are considered risk significant or are in the top cutsets. These basic events include the failure of the CTG to start, the failure of the CTG to run during hours 2 through 48 of service, and the CTG being unavailable during an accident due to testing and maintenance.

SAMDA 13 suggests installing an additional, buried offsite power source to reduce the probability of a loss of offsite power and increase the availability of onsite AC power. SAMDA 14 suggests installing a gas turbine generator to increase the availability of onsite AC power. SAMDA 155 suggests providing a redundant AAPS to improve the availability of AC power in the event of a loss of offsite power. All three design alternatives reduce the probability of failure for all three basic events.

One additional design alternative, SAMDA 154, suggests improving AAPS testing and maintenance procedures.

4.1.8 Containment Flooding and Drain System

Two basic events related to the CFDS are considered risk significant or are in the top cutsets. The first basic event is the failure of CFDS containment isolation valve to open when demanded to allow the CFDS to supply coolant from the common (to six modules) system to the containment of one module. The second basic event is the failure of the operator to start containment flooding.

SAMDA 156 suggests improving the reliability of the CFDS containment isolation valves to improve the probability that the valves will function when demanded.

SAMDA 157 suggests improving operator training to start containment flooding.

4.1.9 Chemical and Volume Control System

Three basic events related to the CVCS are considered risk significant or are in the top cutsets. The first basic event is the failure of the CVCS makeup combining valve to open, which supplies reactor coolant to the CVCS. The second basic event is the failure of the valve that isolates the demineralized water system (DWS) from the CVCS. The final basic event is the failure of the operator to initiate CVCS injection.

SAMDA 158 suggests improving the reliability of the CVCS makeup combining valve, which reduces the probability of the failure of the CVCS makeup combining valve to open.

SAMDA 160 suggests incorporating an additional, diverse, parallel DWS CVCS pump isolation valve, which applies to the failure of the CVCS pump isolation valve to open. A parallel isolation

valve increases the probability that inventory is made available to the CVCS makeup pumps when demanded.

SAMDA 159 suggests improving operator training to start CVCS injection. SAMDA 162 suggests training operators to divert CVCS flow from one module to another through the module heat-up system. Both of these SAMDAs mitigate CVCS failures to operate. SAMDA 161 also applies to the CVCS, but is an operator action to prevent RPV overpressurization due to a failure of the RSVs to open, and was discussed in Section 4.1.6.

4.1.10 Module Protection System

There are 16 basic events that apply to the Module Protection System (MPS) that are risk significant or are in the top cutsets. These basic events can be separated into five categories: the CCF of actuation and priority logic (APL) modules to operate, the failure of equipment interface modules (EIMs) to operate, the failure of manual override switches to close, the CCF of scheduling and voting modules (SVMs) to operate, and the CCF of MPS sensors.

Four basic events apply to the CCF of APL modules to operate, which include the failure of APLs to the DHRS actuation valves, the CVCS discharge line isolation valves, the top branch RTBs, and the bottom branch RTBs. Four basic events apply to the failure of EIMs to operate, which include the failure of the CFDS isolation valve EIMs (i.e., the two EIMs for each CFDS isolation valve for a total of four). SAMDA 196 suggests improving redundancy in removing control power to equipment. SAMDA 196 applies to the EIM and the APL (a subcomponent of the EIM) basic events. When an EIM fails to operate on an actuation signal, control power is not removed from a component and the component does not actuate. Providing additional means to remove control power mitigates these failures. Additionally, SAMDA 197 suggests improving the reliability of EIM components, reducing the probability of failure for both categories.

Four basic events apply to the failure of the manual switches to operate. These basic events include the failure of both divisions of the containment isolation signal override switch to close and the failure of both divisions of the nonsafety enable override switch to close. SAMDA 198 suggests improving the reliability of the manual override switch components, reducing the probability of failure for all four basic events.

Two basic events apply to the failure of the SVMs to operate. These basic events include the CCF of two of three division I engineered safety feature actuation system (ESFAS) SVMs and the CCF of two of three division II ESFAS SVMs. SAMDA 199 suggests improving the reliability of the SVM components, reducing the probability of failure.

Three basic events apply to the CCFs of MPS. These basic events include the CCF of three of four reactor vessel level sensors to operate on demand, the CCF of three of four pressurizer level sensors to operate on demand, and the CCF of three of four reactor vessel level process logic elements. The sensors detect the physical conditions while the process logic elements convert the signals from the sensors to engineering units. SAMDA 194 suggests diversifying the RCS level sensors to reduce the probability of CCFs. SAMDA 195 suggests diversifying the process logic elements to reduce the probability of CCFs.

4.1.11 Reactor Building Crane Failures

As shown in Table 5-4, failures of the reactor building crane during module transport contribute the greatest proportion of any equipment failure to the maximum benefit of risk elimination in the NuScale design, contributing over 98 percent of the maximum benefit. Four SAMDAs are proposed to mitigate reactor building crane failures. The Reactor Building crane PRA showed that the hoist contributes over 95 percent to the probability of crane failure while the remainder of the probability is shared between the bridge and trolley. Therefore, reactor building crane SAMDAs focused on the hoist.

SAMDA 200 suggests improving the reliability of the Reactor Building crane (specifically the hoist components), which is already designed to be single failure proof. SAMDA 201 suggests improving the redundancy in the reactor building crane hoist components. SAMDA 202 suggests incorporating a railway system in the reactor pool design to reduce the reliance on the hoist system during module transport. SAMDA 203 suggests improving testing and maintenance of the reactor building crane.

4.1.12 Initiating Events

NuScale-specific design changes to preclude the possibility of initiating events are not postulated in this analysis. There are many generic PWR SAMDAs that suggest design changes to mitigate initiating events. Additionally the basic events in Table 4-2 occur in response to initiating events. Therefore, the SAMDAs generated in Section 4.1.1 through Section 4.1.11 also apply to the NuScale initiating events.

4.1.13 Fires, Floods, High Winds, External Floods, Seismic, and Low Power and Shutdown Events

One SAMDA is proposed to mitigate failures of plant equipment due to internal fires. SAMDA 190 suggests using three hour rated fire cable for safety related equipment. This SAMDA would prolong the time to cable damage and reduce the probability of spurious operation of plant equipment during an internal fire, allowing more time to suppress fires in the plant.

No NuScale specific SAMDAs are proposed to mitigate failures of plant equipment due to internal flooding. The internal flooding PRA shows that the DHRS, ECCS, and RSV components are significant to internal flooding risk in the NuScale design. Existing SAMDAs discussed in Section 4.1.1, Section 4.1.5, and Section 4.1.6 are considered adequate to address internal flood mitigation in the NuScale design.

High wind events are modeled in the PRA as causing a loss of offsite power. SAMDAs 1 through 24, 154, and 155 suggest design alternatives to mitigate a loss of offsite power, and therefore no additional SAMDAs are suggested to address high winds.

The external flooding PRA is similar to high winds, in that an external flood is modeled as causing an extended loss of offsite power. Damage to equipment is similar to that of internal flooding, therefore no new SAMDAs are postulated that apply to external flooding.

The dominant contributors to seismic risk are structural failures. Seismically induced equipment failures also appear in the top cutsets. SAMDA 140 suggests increasing the seismic ruggedness of plant components to reduce the probability of failure in a seismic event. Reactor building crane failures are considered in the evaluation of seismic risk in the NuScale design.

Seismically induced failures of the reactor building crane are captured by SAMDA 140. SAMDA 188 suggests base mat isolation to reduce the seismic stresses on equipment during seismic events. SAMDA 189 suggests improving the RPV support skirt to increase the seismic ruggedness of the RPV.

The low power and shutdown PRA is similar to the internal events PRA, but considers the reactor operating at low power instead of full power. Therefore, any SAMDAs that apply to equipment failures in the internal events PRA also apply to the low power and shutdown PRA. The low power and shutdown PRA also considers reactor building crane failures, which are discussed in Section 4.1.11. Therefore, any SAMDAs that apply to reactor crane failures also apply to the low power and shutdown PRA.

5.0 Maximum Benefit

A key equation for the initial cost-benefit assessment of implementing a particular SAMDA is as follows:

$$Net\ Value = APE + AOC + AOE + AOSC - COE \qquad \text{Equation 5-1}$$

where,

APE = present value of averted public exposure ($),

AOC = present value of averted offsite property damage costs ($),

AOE = present value of averted occupational exposure ($),

AOSC = present value of averted on-site costs ($), and

COE = cost of enhancement [or cost of implementing SAMDA] ($).

If the net value is positive, then the SAMDA would warrant further consideration for incorporation into the NuScale design. If the net value is negative, then the SAMDA would be unbeneficial to implement.

Any particular SAMDA would only reduce the probability of a small subset of severe accident scenarios. However, to be conservative (and to simplify the evaluation), the process of calculating the maximum benefit includes the summation of contributions from all severe accidents and assume that any particular SAMDA would reduce the probability of all severe accidents to zero.

The remainder of Section 5.0 establishes present values associated with severe accident risk and discusses the various calculational procedures implemented to determine the maximum benefit of SAMDAs in the NuScale design.

5.1 Present Value Factor

In order to account for the time value of money, a real discount rate is assigned in order to determine a present value factor (C). This time value of money is different than the update of

the economic data in the Surry site file from 2005 to 2016 mentioned in Section B.1.5.2 of this report. The updating of the economic data in the site file is to bring past values up to date.

The present value factor applies to the economic and dose consequences of an accident as calculated by MACCS code (Reference 8.1-7). The dose and economic costs calculated by MACCS are multiplied by the point estimate CDF, giving a dollar amount saved or a dose avoided per year. The dose avoided per year is converted to a dollar amount per year based on a dollar per rem factor suggested in NUREG-1530 (Reference 8.1-4, page 26). The present value factor is then used to account for the costs saved over the entire lifetime of the plant from avoiding an accident, and to express those dollars in present value.

As discussed in NUREG/BR-0184, Section B.2, the costs saved over a period of years must be summed. However, this summation cannot be done directly as money in the present is more valuable than the same amount at a future date. Discounting is used to compare amounts of money expended at different times. The result of discounting is called the present value. NUREG/BR-0184 recommends the use of continuous discounting in regulatory analyses in which the cost is weighed by an accident frequency over the remaining life of a facility, as is done throughout Section 5.0 of this report where appropriate. The accident frequency is a continuous variable (Reference 8.1-2, Section B.2.3). NUREG/BR-0184 gives the present value factor (C) as

$$C = (e^{-rt_i} - e^{-rt_f})/r \qquad \text{Equation 5-2}$$

where,

r = real discount rate (per year),

t_f = years remaining until end of facility life, and

t_i = years before facility begins operation.

This report will calculate the net present value relative to the first year of a plant's operation, which sets t_i=0 and simplifies the present value factor to

$$C = \frac{1 - e^{-rt_f}}{r} \qquad \text{Equation 5-3}$$

Taking r to be 7 percent per year, as recommended by NUREG/BR-0184, and t_f to be 60 years yields C=14 years.

5.2 Averted Public Exposure

The present value associated with averted public exposure (APE) is determined by multiplying the avoided public dose (APD) by a surrogate monetary value associated with each person-rem of dose avoided and multiplying that product by the present value factor.

$$APE = APD \times R \times C \qquad \text{Equation 5-4}$$

where,

APD = Avoided Public Dose (person-rem/year),

R = monetary value associated with unit of dose ($/(person-rem)),

C = present value factor (year), and

NUREG-1530 suggests a surrogate monetary value of R=$5,100 per person-rem (Reference 8.1-4, page 26).

The avoided public dose (before discounting) can be found by multiplying the population dose (PD) for each release category (RC) by the frequency of the release category and summing the result of this product across all the release categories.

$$APD = \sum_{\substack{Release \\ Categories}} [\textit{release category frequency} \times \textit{release category PD}] \qquad \text{Equation 5-5}$$

The population dose for each release category was calculated with MELCOR-derived release information and a MACCS model using site data from the Surry SOARCA Analysis with a 50-mile radius, which was developed to support this report. To simplify the analysis, the reactor building is not credited for radionuclide deposition. This simplification is bounding, as some amount of the release from the module will deposit in the reactor building before reaching the environment. The results of these calculations may be found in Table 5-1, which describes the population dose associated with each release category, the frequency associated with each release category, and the total APD of 1.3E-04 person-rem per year. For a more detailed description of MACCS modeling, see Appendix B Section B.1.

A short description of the NuScale Level 2 PRA-derived release categories is given here to aid interpretation of Table 5-1 and other release category discussions later in this report. For a more detailed description of release categories, see Appendix B Section B.2. Note that these release categories are different from the two release categories (core damage with intact containment and core damage with containment bypass) given in the NuScale Level 2 PRA as a more precise treatment was desired for the purposes of this report. Also note that release category associated information (and Table 5-1 results) is based on internal events, low-power and shutdown, internal flooding, internal fires, external floods, and high winds PRAs.

RC 1: CVCS LOCA Inside Containment

RC 2: LOCA Outside Containment, Isolated

RC 3: CVCS LOCA Outside Containment, Unisolated

RC 4: ECCS Spurious Actuation

RC 5: Steam Generator Tube Failure

RC 6: General Transient with RSV Stuck Open

RC 7: General Transient with No RSVs

RC 8: Dropped Module During Transport

Table 5-1: Frequency of Occurrence and Offsite Consequences for Each Release Category

RC	Release Frequency (per year)	Offsite Dose per Event (person-rem/event)	Offsite Dose per Year (person-rem/year)
1	2.2E-11	4.4E+01	9.5E-10
2	1.1E-12	3.9E+01	4.1E-11
3	1.7E-11	1.3E+06	2.2E-05
4	2.4E-09	3.0E+01	7.2E-08
5	1.4E-13	3.2E+05	4.3E-08
6	2.1E-10	2.8E+01	5.9E-09
7	2.7E-10	2.2E+01	5.9E-09
8	5.3E-07	2.0E+02	1.0E-04
Total	5.3E-07	-	1.3E-04

Utilizing these values and a present value factor of 14 yields an APE of $9.3.

$$APE = 1.3E-04\frac{person\text{-}rem}{year} \times \frac{\$5{,}100}{person\text{-}rem} \times 14\ year = \$9.3 \qquad \text{Equation 5-6}$$

5.3 Averted Offsite Costs

The averted offsite property damage costs (AOC) are calculated using an equation adapted from NUREG/BR-0184, Equation 5-7.

$$AOC = C \times offsite\ economic\ impact\left(\frac{\$}{year}\right) \qquad \text{Equation 5-7}$$

The offsite economic impact for each release category was calculated with MELCOR-derived release information and a MACCS model, using site data from the Surry SOARCA analysis with a 50-mile radius, which was developed to support this report. The results of these calculations may be found in Table 5-2, which describes the offsite economic impact per event associated with each release category, the frequency associated with each release category, and the annual monetary value of offsite economic impact associated with each release category. The total monetary value of offsite economic impact is determined by a summation of the offsite economic impact calculated for each release category and found to be 8.2E-02 dollars per year. For a more detailed description of MACCS calculation procedure, see Appendix B Section B.1.

Table 5-2: Frequency of Occurrence and Offsite Consequences for Each Release Category

RC	Release Frequency (per year)	Offsite Economic Impact per Event ($)	Offsite Economic Impact ($/year)
1	2.2E-11	8.4E-01	1.8E-11
2	1.1E-12	1.8E-01	2.0E-13
3	1.7E-11	4.7E+09	8.0E-02
4	2.4E-09	0.0E+00	0.0E+00
5	1.4E-13	1.9E+08	2.7E-05

Table 5-2: Frequency of Occurrence and Offsite Consequences for Each Release Category

RC	Release Frequency (per year)	Offsite Economic Impact per Event ($)	Offsite Economic Impact ($/year)
6	2.1E-10	0.0E+00	0.0E+00
7	2.7E-10	0.0E+00	0.0E+00
8	5.3E-07	3.7E+03	2.0E-03
Total	5.3E-07	-	8.2E-02

Multiplying by the present value factor (14 year) gives an AOC value of:

$$AOC = 14\,year \times 8.2E-02\left(\frac{\$}{year}\right) = \$1.1 \qquad \text{Equation 5-8}$$

5.4 Averted Occupational Exposure

The averted occupational exposure (AOE) is estimated by combining the estimated values associated with averting short-term and long-term dose to personnel in response to a severe accident. The methodology for performing these estimates is based on NUREG/BR-0184.

5.4.1 Short Term Dose to Personnel

The short term dose to personnel was estimated with Equation 5-9.

$$W_{IO} = R \times CDF \times D_{IO} \times C \qquad \text{Equation 5-9}$$

where,

W_{IO} = monetary value of occupational health risk avoided due to "immediate" doses, after discounting ($),

R = monetary value associated with unit of dose ($/(person-rem)),

CDF = Core damage frequency (events per year),

D_{IO} = immediate occupational dose, (person-rem per event), and

C = present value factor (year).

Using values associated with the NuScale analysis:

R = $5,100 per person rem (per Reference 8.1-4),

CDF = 5.3E-07 events per year,

D_{IO} = 3,300 person rem per event (best estimate value based on Three Mile Island and Chernobyl data per NUREG/BR-0184), and

C = 14 year.

Therefore,

$$W_{IO} = \frac{\$5{,}100}{person\text{-}rem} \times 5.3E-07\frac{events}{year} \times 3300\frac{person\text{-}rem}{event} \times 14\ year = \$120 \qquad \text{Equation 5-10}$$

5.4.2 Long Term Dose to Personnel

The long-term occupational dose associated with the cleanup and decontamination process following the severe accident is given by Equation 5-11 which was derived from NUREG/BR-0184 Section 5.7.3.1.

$$W_{LTO} = R \times CDF \times D_{LTO} \times C \times \frac{1-e^{-rm}}{rm} \qquad \text{Equation 5-11}$$

where,

W_{LTO} = monetary value of long term dose risk avoided, accounting for present value factor,

R = monetary value associated with unit of dose $\left(\frac{\$}{person\text{-}rem}\right)$,

CDF = core damage frequency (events per year),

D_{LTO} = long term occupational dose, (person-rem per event),

C = present value factor (year),

r = real discount rate (percent per year),

m = on-site cleanup period (years), and

Using values associated with the NuScale analysis:

R = $5,100 per person-rem (per Reference 8.1-4),

CDF = 5.3E-07 events per year,

D_{LTO} = 20,000 person-rem per event (per NUREG/BR-0184 best estimate),

C = 14 year,

r = 0.07 per year, and

m = 10 years (per NUREG/BR-0184).

Therefore,

$$W_{LTO} = \frac{\$5100}{person\text{-}rem} \times 5.3E-07\frac{events}{year} \times 20{,}000\frac{person\text{-}rem}{event} \times$$ Equation 5-12

$$14\ year \times \frac{1-e^{-\frac{0.07}{year} \times 10\ years}}{\frac{0.07}{\text{year}} \times 10\ years} = \$540$$

5.4.3 Total Averted Occupational Exposure

Combining the immediate and long-term on-site exposure costs results in a total on-site exposure cost, W_O, given by Equation 5-13:

$$W_O = W_{IO} + W_{LTO}$$ Equation 5-13

Using the values for W_{IO} and W_{LTO} calculated above:

- $W_{IO} = \$120$
- $W_{LTO} = \$540$

The total averted on-site exposure costs are calculated to be:

$$W_O = \$120 + \$540 = \$660$$ Equation 5-14

5.5 Averted Onsite Costs

The averted onsite costs (AOSC) associated with onsite property damage from a severe accident can be broken down into cleanup and decontamination costs, repair and refurbishment costs, and costs associated with replacement power over the remaining life of the facility. Since it is assumed that the module undergoing core damage will be irrecoverably damaged, the repair and refurbishment component of the AOSC will not be considered further. Decommissioning a nuclear facility at the end of the facility lifetime also has an associated cost, which is included in the cleanup and decontamination cost estimate recommended in NUREG/BR-0184.

The AOSC calculated in Section 5.5.1 through Section 5.5.3 applies to a single NuScale module. The AOSC for a 12 module plant is discussed in Section 5.7.

5.5.1 Cleanup and Decontamination Costs

A cleanup period of 10 years was assumed per NUREG/BR-0184. NUREG/BR-0184 estimated the value associated with total cleanup and decontamination costs as \$1.5E+9 in 1993. This value of \$1.5E+9 is the sum of the estimated cleanup and decontamination cost of \$1.1E+9 and the estimated decommissioning cost of \$4E+8 (Reference 8.1-2, page 5.42-5.43). This value is updated to 2016 dollars by multiplying by the ratio of the 2016 consumer price index (CPI) of

240.236 to the 1993 CPI of 144.5 (Reference 8.1-10, Table 24). The ratio of these two CPIs is 1.663. The net present value of a single event is determined through Equation 5-15:

$$PV_{CD} = \left(\frac{C_{CD}}{m}\right) \times \left[\frac{1 - e^{-rm}}{r}\right] \qquad \text{Equation 5-15}$$

where,

PV_{CD} = net present value of a single event ($),

C_{CD} = total cost of cleanup and decontamination = $1.5E+9 * 1.663 = $2.5E+09,

m = cleanup period (years) = 10 years, and

r = real discount rate (per year) = 0.07 per year.

Therefore,

$$PV_{CD} = \frac{\$2.5E + 09}{10\ years} \times \left[\frac{1 - e^{-\frac{0.07}{year} \times 10\ years}}{\frac{0.07}{year}}\right] = \$1.8E + 09 \qquad \text{Equation 5-16}$$

The present value of the cleanup and decontamination costs over the remaining life of the facility (U_{CD}) is determined through Equation 5-17:

$$U_{CD} = PV_{CD} \times C \qquad \text{Equation 5-17}$$

where,

PV_{CD} = net present value of a single event ($) = $1.8E+09, and

C = present value factor (year) = 14 year.

Therefore,

$$U_{CD} = \$1.8E + 09 \times 14\ year = 2.5\text{E+10}\$ years \qquad \text{Equation 5-18}$$

5.5.2 Replacement Power Cost

Following a severe accident it is assumed the plant's owner would still be responsible for replacement power costs. Replacement power costs can be calculated per the guidance of NUREG/BR-0184, which is based on a reference 910 MWe plant that operates at roughly a 60 percent capacity factor in 1993 dollars.

To convert the 1993 dollar basis to modern day dollars, the Bureau of Labor Statistics' annual, non-seasonally adjusted Producer Price Index (PPI) for the commodity "Electric Power" is used. The PPI measures the average change over time in the selling prices received by producers for their respective outputs and is a widely accepted means for adjusting long-term pricing contracts. The 1993 PPI for electric power was 128.6 and the 2015 PPI for electric power was 203.9 (Reference 8.1-5). Multiplying the 1993 replacement power cost estimate by the ratio of the 2015 to 1993 PPI is a reasonable means of estimating the 2015 cost of replacement power associated with a severe accident. The 2015 PPI is the most recent annual PPI available at the time of this report's creation.

Note further that the value from NUREG/BR-0184 was based on old data that assumed a roughly 60 percent capacity factor for a nuclear power plant. The NuScale design is expected to achieve a capacity factor greater than 95 percent. Therefore, to adjust this estimate to account for the larger capacity factor, a simple multiplication of the previously determined replacement power costs by the ratio of the NuScale assumed capacity factor to the previously assumed 60 percent capacity factor is a reasonable estimate.

Equation 5-19 is derived from NUREG/BR-0184 and adjusts the reference 910 MWe plant to the 50 MWe NuScale Power Module. Equation 5-19 approximates the average value of net present value replacement power costs for a single event:

$$PV_{RP} = \left[\frac{\mathrm{B} \times \left(\frac{0.95}{0.60}\right) \times \left(\frac{2015\ PPI}{1993\ PPI}\right) \times \left(\frac{50\ MWe}{910\ MWe}\right)}{r}\right] \times [1 - e^{-rt_f}]^2 \qquad \text{Equation 5-19}$$

where,

PV_{RP} = net present value of replacement power for a single event,

B = a constant representing a string of replacement power costs that occur over the lifetime of a reactor after an event = 1.2E+08 $/year (1993 dollars, 910 MWe, 60 percent capacity factor),

r = real discount rate (per year) = 0.07 per year, and

t_f = years remaining until end of facility life = 60 years.

Therefore,

$$PV_{RP} = \left[\frac{1.2E+08\frac{\$}{year} \times \left(\frac{0.95}{0.60}\right) \times \left(\frac{203.9}{128.6}\right) \times \left(\frac{50\ MWe}{910\ MWe}\right)}{\frac{0.07}{year}}\right] \times \left(1 - e^{-\frac{0.07}{year} \times 60\ years}\right)^2 = \$2.3E+08 \qquad \text{Equation 5-20}$$

The net present value of replacement power over the life of the facility can be found with the following equation from NUREG/BR-0184 (Equation 5-21). Note that the squared term in Equation 5-21 is a proxy for the fact that the costs for events in future years decline due to the reduced number of years of possible power production left on the license for which replacement power would need to be procured.

$$U_{RP} = \frac{PV_{RP}}{r} \times (1 - e^{-rt_f})^2 \qquad \text{Equation 5-21}$$

Therefore,

$$U_{RP} = \frac{\$2.3E+08}{\frac{0.07}{year}} \times \left(1 - e^{-\frac{0.07}{year} \times 60\ years}\right) = 3.2E+09\$years \qquad \text{Equation 5-22}$$

5.5.3 Total Averted Onsite Costs

The total on-site economic costs, W_{OSC}, are calculated by summing the cleanup and decontamination costs and the replacement power costs, then multiplying this value by the CDF, as shown in Equation 5-23:

$$W_{OSC} = (U_{CD} + U_{RP}) \times CDF \qquad \text{Equation 5-23}$$

where,

U_{CD} = total cost of cleanup and decontamination over the analysis period ($-years) and

U_{RP} = net present value of replacement power over the life of facility ($-years).

Using the values calculated above and from the NuScale PRA:

- U_{CD} = 2.5E+10 $-years
- U_{RP} = 3.2E+09 $-years
- CDF = 5.3E-07 per year

The total on-site economic costs are calculated to be:

$$W_{OSC} = ((2.5E + 10\ \$years) + (3.2E + 09\ \$years)) \times \frac{5.3E - 07}{year} \qquad \text{Equation 5-24}$$

$$= \$15{,}000$$

5.6 Single Module Severe Accident Impact

An interim step in the calculation of the maximum benefit involves the summation of APE (Section 5.2), AOC (Section 5.3), AOE (Section 5.4), and AOSC (Section 5.5) to obtain a term referred to as the severe accident impact (SAI). Table 5-3 shows a summary of the values calculated in Section 5.2 through Section 5.5 separated by release category and summed together to calculate the SAI. The total single module SAI calculates the monetary value attributed to the consequences of an accident involving a single module at a 12 module plant, considering internal events, low power shutdown, internal floods, internal fires, external floods, high winds, and reactor crane failure as $16,000.

Table 5-3: Summary of Results Related to Calculating Single Module SAI

Release Category	APE (public exposure, $)	AOC (offsite cost, $)	AOE (onsite exposure, $)	AOSC (onsite cost, $)	SAI ($)	Percentage of total SAI
1	6.9E-05	2.5E-10	2.8E-02	6.2E-01	6.5E-01	4.0E-03
2	3.1E-06	2.8E-12	1.4E-03	3.1E-02	3.2E-02	2.0E-04
3	1.5E+00	1.1E+00	2.1E-02	4.8E-01	3.1E+00	1.9E-02
4	5.2E-03	0.0E+00	3.1E+00	6.8E+01	7.1E+01	4.4E-01
5	3.2E-03	3.8E-04	1.7E-04	3.9E-03	7.7E-03	4.8E-05
6	4.2E-04	0.0E+00	2.7E-01	5.9E+00	6.2E+00	3.9E-02
7	4.2E-04	0.0E+00	3.4E-01	7.6E+00	7.9E+00	4.9E-02
8	7.8E+00	2.8E-02	6.6E+02	1.5E+04	1.6E+04	9.9E+01
Total	9.3E+00	1.1E+00	6.6E+02	1.5E+04	1.6E+04	1.0E+02
Percentage of total SAI	5.8E-02	6.9E-03	4.1E+00	9.4E+01	1.0E+02	-

5.7 Maximum Benefit

In Section 5.6, the SAI for a single module due to internal events, low-power shutdown events, internal flooding events, internal fire events, external flooding events, high winds events, and reactor building crane failure events is calculated. However, the effect of seismic events and multiple modules co-located in the same reactor building must also be taken into account to calculate the maximum benefit.

Following the guidance in NEI 05-01, seismic events are accounted for using an external events multiplier (EEM) on the SAI calculated in Section 5.6. The seismic margin assessment (SMA) event tree sequences do not match the sequences of the other PRAs because the SMA contains additional failures (i.e., structural failures) that cannot be directly matched with the sequences of the other PRAs. These differences in sequences are accounted for by using an EEM. An EEM is calculated and applied to all release categories. The seismic CDF for the Surry site is 3.2E-8 per

year as shown in Table B-20 and is combined with the existing CDF of 5.3E-07 for a total CDF of 5.6E-07 per year. Therefore, the EEM is:

$$EEM = \frac{CDF_{seismic} + \sum CDF_{RCs\,1\text{-}8}}{\sum CDF_{RCs\,1\text{-}8}} = \qquad \text{Equation 5-25}$$

$$\frac{3.2E-08\ per\,year + 5.3E-07\ per\,year}{5.3E-07\ per\,year} = 1.1$$

The NuScale design is unique in that there are 12 operating modules housed in one plant. NuScale has developed a multi-module PRA, which quantifies the frequency per year that two or more modules experience simultaneous core damage events. Predicting the specific number of modules that experience a core damage event is outside the scope of that analysis and, therefore, the multi-module PRA is not used to estimate a multi-module multiplier for the purposes of calculating a maximum benefit for a 12 module NuScale Power Plant. To account for the total number of modules, an additional multiplier of 12 is simply applied to the SAI calculated in the previous section for release categories one through seven. The contribution to the maximum benefit for the Surry site from release categories one through seven is calculated as follows:

$$Maximum\ Benefit_{RCs\ 1\text{-}7} = \sum (APE + AOC + AOE + AOSC)_{i,\ per\ module} \qquad \text{Equation 5-26}$$

$$\times\ EEM \times 12\ modules = \$89 \times 1.1 \times 12 = \$1{,}200$$

The low power and shutdown PRA analysis has established that the maximum number of modules that can be damaged in a module drop accident is three. Therefore, a multiplier of three is applied to release category 8 to account for multiple modules, with the exception of the AOSC. This approach is conservative as it is unlikely that an event will impact multiple modules and cause core damage. For the AOSC calculated in release category 8, it is assumed that replacement power is required for the entire NuScale plant following an accident. Therefore, in calculating the contribution to the maximum benefit for the Surry site from release category 8, the APE, AOC, AOE, and cleanup and decontamination (U_{CD}) portion of the AOSC are all multiplied by three. However, the replacement power costs (U_{RP}) are multiplied by 12. The contribution to the maximum benefit for the Surry site from release category 8 is calculated as follows:

$$Maximum\ Benefit_{RC\ 8} = (3 \times [APE_{RC8} + AOC_{RC8} + AOE_{RC8}] + \qquad \text{Equation 5-27}$$

$$[12 \times U_{RP,RC8} + 3 \times U_{CD,RC8}] \times CDF) \times EEM =$$

$$\Big(3 \times [\$7.8 + \$2.8E\text{-}2 + \$660] +$$

$$[12 \times 3.2E+09\ \$years + 3 \times \$2.50E+10\ \$years] \times \frac{5.3E-07}{year}\Big) \times 1.1 = \$68{,}000$$

The maximum benefit for the Surry site is calculated in Equation 5-28 and is shown in Table 5-4.

$$\text{Maximum Benefit} = \text{Maximum Benefit}_{RCs\,1\text{-}7} + \text{Maximum Benefit}_{RC8} = \$1{,}200 + \$68{,}000 = \$69{,}200 = \$69{,}000 \qquad \text{Equation 5-28}$$

Table 5-4: 12 Module Maximum Benefit for the Surry Site

Release Category	APE (public exposure, $)	AOC (offsite cost, $)	AOE (onsite exposure, $)	AOSC (onsite cost, $)	Maximum Benefit ($)	Percentage of maximum benefit
1	9.1E-04	3.3E-09	3.7E-01	8.2E+00	8.6E+00	1.2E-02
2	4.1E-05	3.7E-11	1.8E-02	4.1E-01	4.3E-01	6.2E-04
3	2.0E+01	1.5E+01	2.8E-01	6.3E+00	4.2E+01	6.1E-02
4	6.9E-02	0.0E+00	4.1E+01	9.0E+02	9.4E+02	1.4E+00
5	4.2E-02	5.0E-03	2.2E-03	5.1E-02	1.0E-01	1.4E-04
6	5.5E-03	0.0E+00	3.6E+00	7.8E+01	8.2E+01	1.2E-01
7	5.5E-03	0.0E+00	4.5E+00	1.0E+02	1.0E+02	1.4E-01
8	2.6E+01	9.2E-02	2.2E+03	6.6E+04	6.8E+04	9.8E+01
Total	4.6E+01	1.5E+01	2.2E+03	6.7E+04	6.9E+04	1.0E+02
Percentage of maximum benefit	6.7E-02	2.2E-02	3.2E+00	9.7E+01	1.0E+02	-

Reactor building crane failures are significant risk contributors to the NuScale design with the module drop release category (RC 8) contributing approximately 99 percent of the SAI (Table 5-3) and approximately 98 percent of the total maximum benefit (Table 5-4).

5.8 Maximum Benefit Sensitivity Study

To examine the sensitivity of the maximum benefit to input parameters and MACCS modeling assumptions, several sensitivity calculations are performed. Maximum benefit sensitivity cases one through six are calculated using MACCS sensitivity results. Maximum benefit sensitivity cases seven through nine are not calculated using MACCS sensitivity results, but are sensitivities of suggested cost-benefit analysis values in NUREG/BR-0184 and the NUREG-1530 Revision 1 draft for public comment (Reference 8.1-4). Maximum benefit values are calculated for the following scenarios:

1) Radionuclide release occurs at the top of the reactor building (24.689 m), compared to 0 m in the base case.

2) Radionuclide release is buoyant with a heat content of 100,000 W per plume segment, compared to 0 W in the base case.

3) No radionuclide deposition in the CNV for intact containment sequences, compared to the CNV deposition calculated by MELCOR in the base case.

4) Radionuclide aerosol dry deposition velocity (0.01 m/s) from NUREG-1150 for all radionuclide classes, compared to the deposition velocities specified in Table 4-11 of Reference 8.1-11. This deposition velocity is only assessed for intact containment sequences.

5) High burnup core inventory, compared to the best estimate core inventory in the base case.

6) Radionuclide release occurs at the Peach Bottom site, compared to the Surry site in the base case.

7) High estimate of immediate and long-term occupational (onsite) dose, which is 14,000 person-rem immediate and 30,000 person-rem long term (Reference 8.1-2, pages 5.30, 5.31), compared to 3,300 person-rem immediate and 20,000 person-rem long term in the base case.

8) High estimate of dollar-per-person-rem value, which is $7,500 per person-rem (Reference 8.1-4, page 26), compared to $5,100 per person-rem in the base case.

9) Real discount rate of three percent per year, compared to seven percent per year in the base case.

The same process outlined in Section 5.0 through Section 5.7 is used to calculate the maximum benefit for the sensitivity cases outlined above. With the exception of case six, which uses a Peach Bottom site-specific seismic CDF, the CDF and release category frequencies remain unchanged. Peach Bottom is more seismically active than Surry and therefore has a larger seismic CDF of 2.0E-06. The external events multiplier for the Peach Bottom site is calculated below.

$$EEM_{Peach\ Bottom} = \frac{\frac{5.3E-07 + 2.0E-06}{year}}{\frac{5.3E-07}{year}} = 4.8 \qquad \text{Equation 5-29}$$

Intact containment MACCS simulations required additional assumptions to analyze offsite dose, compared to containment bypass simulations. These assumptions include plume buoyancy, particle size distribution/deposition velocity in the environment, and containment deposition. Thus, these sensitivities were not performed for release category three and release category five and instead the base case results were used to calculate the maximum benefit.

The results of the sensitivity cases are presented in Table 5-5.

Table 5-5: Maximum Benefit Sensitivity

Case	Description	Maximum Benefit
-	Base case	$69,000
1	Release height of 24.689m	$69,000
2	100,000 W per plume segment	$69,000
3	No CNV deposition	$69,000
4	Aerosol dry deposition velocity 0.01m/s	$69,000
5	High burnup core inventory	$69,000
6	Peach Bottom site	$300,000
7	High estimate of onsite dose	$72,000
8	High $/person-rem estimate	$70,000
9	Real discount rate of three percent	$180,000

There are a few key findings from these sensitivity cases. The Peach Bottom site sensitivity shows a noticeable increase in the maximum benefit compared to the base case Surry site. This increase is mostly due to the much larger seismic CDF of 2.0E-6 per year at the Peach Bottom site, compared to the seismic CDF of 3.2E-08 per year at the Surry site. Both the Peach Bottom and Surry SAIs are comparable at approximately $16,000. Parameters attributed to dose, the monetary value of dose, and MACCS modeling are shown to have a negligible effect on the end result. Because approximately 97 percent of the maximum benefit is attributed to onsite economic costs, as shown in Table 5-4, the value of the real discount rate used to account for the devaluing of money over time is significant.

6.0 Assessment of SAMDA Candidates

The candidate SAMDAs identified in Section 4.0 of this report are qualitatively screened into one of seven initial screening categories. The intent of the screening is to identify the candidates that warrant a detailed cost-benefit evaluation. These categories and the screening process itself were based on the "Phase I" analysis screening criteria found in NEI 05-01. These seven categories include "not applicable," "already implemented," "combined," "excessive implementation cost," "very low benefit," "not required for design certification," and "considered for further evaluation." These seven categories are described in greater detail in Section 6.1. Screening results are shown along with the list of candidate SAMDAs in Appendix A and summarized in Section 6.2.

6.1 Initial "Phase I" SAMDA Screening Categories

6.1.1 Not Applicable

SAMDA candidates that are not considered applicable to the NuScale design are those developed for systems specifically associated with boiling water reactors (BWRs) or with specific PWR equipment that is not in the NuScale design. For example, the NuScale design does not include reactor coolant pumps and, therefore, any SAMDA that deals with reactor coolant pumps would be screened as "not applicable" to the NuScale design.

6.1.2 Already Implemented

Candidate SAMDAs that are already included in the NuScale design or whose intent is already fulfilled by a different NuScale design feature are considered "already implemented" in the NuScale design. If a particular SAMDA has already been implemented in the NuScale design, it is not retained for further analysis. For example, the NuScale chemical and volume control system (CVCS) is capable of mitigating small loss of coolant accidents (LOCAs) under accident conditions. This capability satisfies SAMDA 37 which calls for an upgrade to the CVCS to mitigate small LOCAs.

6.1.3 Combined

The SAMDA candidates that are similar to one another are combined and evaluated in conjunction with each other. This combination of SAMDA candidates leads to a more comprehensive or plant-specific SAMDA candidate set. The combined SAMDA are then assessed against the remaining six screening categories. For example, SAMDA 19 that calls for the use of the fire water system as a backup source for diesel cooling and SAMDA 20 that calls

for the addition of a new backup source of diesel cooling have a similar intent and a similar effect on the overall plant risk. Therefore, these proposed SAMDAs were combined and evaluated in conjunction with each other.

6.1.4 Excessive Implementation Cost

The maximum benefit is $69,000, as described in Section 5.7, and is used to screen SAMDA candidates in this analysis. The maximum benefit is the monetary value attributed to eliminating all risk in the NuScale design. If a SAMDA requires extensive changes that would exceed the maximum benefit of $69,000 even without an implementation cost estimate, it is not retained for further analysis. For example, in SAMDA 130 the cost to design and implement an additional independent boron injection system is expected to be greater than $100,000, which exceeds the maximum benefit of $69,000 and therefore, is not retained for further analysis.

6.1.5 Very Low Benefit

If a proposed SAMDA is related to a system whose improved reliability would have a negligible impact on overall plant risk, then the SAMDA is considered to have a very low benefit for implementation and is not retained for further analysis. The Level 1 and Level 2 PRA importance lists from all NuScale PRAs, discussed in Chapter 19 of the NuScale Final Safety Analysis Report, are used to determine the risk-significance of systems and components in the NuScale design for the SAMDA screening process. The systems that are considered risk-significant to the NuScale design are shown in Table 6-1. If a system is not considered risk-significant to the NuScale design, then implementing a SAMDA related to that system is of very low benefit. The components that are considered risk-significant to the NuScale design are shown in Table 6-2. If a component is not considered risk-significant to the NuScale design, then implementing a SAMDA related to that component is of very low benefit.

Table 6-1: Systems Risk-Significant to NuScale design

System	Description
ECCS	emergency core cooling system
MPS	module protection system

Table 6-2: Components Risk-Significant to the NuScale design

Description
CES CNV isolation valve 1
CES CNV isolation valve 2
CVCS discharge line CNV isolation valve 1
CVCS discharge line CNV isolation valve 2
DHRS train 1 actuation valve A
DHRS train 1 actuation valve B
DHRS train 2 actuation valve A
DHRS train 2 actuation valve B
ECCS RVV main valve A
ECCS RVV main valve B
ECCS RVV main valve C
ECCS RRV main valve A

Table 6-2: Components Risk-Significant to the NuScale design (Continued)

Description
ECCS RRV main valve B
ECCS RVV trip valve A
Combustion turbine generator
Reactor safety valve A
Reactor safety valve B

NuScale design-specific SAMDAs are not screened in Phase I as very low benefit as these SAMDAs are generated in response to the risk-significant basic events and are, therefore, risk significant by definition. However, NuScale design-specific SAMDAs may still be screened by the other categories.

6.1.6 Not Required for Design Certification

SAMDA candidates related to potential procedural enhancements, surveillance action enhancements, multiple 12 module sites, or design elements that are to be finalized in a later stage of the design process are outside of the scope of this report and are categorized as "not required for design certification." For example, SAMDA 145 suggests increasing the fire brigade's awareness for the sake of decreasing the consequences of a fire, which is outside of the scope of the design certification application, and therefore, this report. SAMDAs that are screened using this category will be re-evaluated by COL applicants.

6.1.7 Considered for Further Evaluation

Any SAMDA candidate that did not screen into any of the previous six screening categories is subject to a more in-depth cost-benefit analysis.

6.2 Phase I Screening Results

A total of 203 SAMDA candidates developed from industry and NuScale documents were evaluated in this phase of the analysis. The screening of each SAMDA and the basis for the screening is shown in Appendix A.

- 47 SAMDA candidates were not applicable to the NuScale design.
- 18 SAMDA candidates already were implemented into the NuScale design either as suggested in the SAMDA or as an equivalent replacement that fulfilled the intent of the SAMDA.
- 13 SAMDA candidates were combined with another SAMDA because the candidates had the same intent.
- 37 SAMDA candidates were not required for design certification because the candidates were related to a procedural or surveillance action or were related to a multiple 12 module site.
- 32 SAMDA candidates were of very low benefit to reducing risk in the NuScale design.
- 56 SAMDA candidates were categorized as having an excessive implementation cost.
- None of the SAMDA candidates were retained for further evaluation.

6.3 Phase II SAMDA Screening

NEI 05-01 prescribes that any SAMDA candidates that screened into the "considered for further evaluation" category in the Phase I screening should be subjected to a more realistic cost-benefit analysis referred to as a "Phase II analysis". Phase II analysis entails subtracting the value of the severe accident risk associated with the design after the SAMDA has been incorporated in the design from the maximum benefit derived in Section 5.0 (which conservatively assumed that the implementation of any SAMDA would reduce the total plant risk to zero). However, no SAMDA candidates from Phase I are retained for further analysis and, therefore, Phase II analysis was not performed.

A sensitivity study was performed to determine if any of the SAMDAs would screen in for a Phase II analysis and considered cost beneficial. Re-screening the list of candidate SAMDAs using the case six "Peach Bottom site" sensitivity study maximum benefit of $300,000 instead of the base case maximum benefit of $69,000 shown in Table 5-5 resulted in several SAMDAs being retained for Phase II analysis. Of these SAMDAs that would no longer screen out in Phase I analysis, none applied to the reactor crane. Reactor crane failure is shown in Table 5-4, as release category eight, to contribute approximately 98 percent to the maximum benefit. Therefore, a conservative estimated maximum benefit for non-reactor crane SAMDAs for the case six sensitivity study is approximately two percent of the maximum benefit, or approximately $6,000. All SAMDAs that were retained for Phase II analysis in this sensitivity study had cost estimates of over $100,000 and therefore none of the SAMDAs were found to be cost beneficial in this sensitivity study.

7.0 Summary and Conclusions

A list of generic and NuScale-specific SAMDAs were identified and considered for implementation in the NuScale design. The maximum benefit that could be associated with implementing a SAMDA was established and sensitivity studies on this value were performed. A preliminary screening analysis was conducted, with the result that none of the SAMDAs were considered to be cost beneficial to implement. The NuScale design is robust and already mitigates the likelihood and the consequences of severe accidents such that any additional severe-accident mitigation features beyond those already incorporated in the NuScale design are difficult to justify from a cost-benefit perspective.

8.0 References

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Appendix A List of Candidate SAMDAs and Screening Results

Table A-1 lists the candidate SAMDAs considered in this analysis, provides a description of each SAMDA, details the screening disposition of the candidate SAMDA, and provides a basis for the assessment. Note that some of the SAMDAs in Table A-1 discuss a loss of offsite power (LOOP). A LOOP, as used in the PRA analysis, is a loss of normal AC power. The expected response to a LOOP is startup of the onsite CTG or a backup diesel generator (BDG). See Chapter 19 of the NuScale Final Safety Analysis Report for more information on LOOP in the context of the NuScale PRA.

Table A-1: Screening of Proposed SAMDAs

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
Generic PWR Improvements Related to AC and DC Power				
SAMDA1	Provide additional DC battery capacity.	Extended DC power availability during an SBO.	Already implemented	The result of this generic SAMDA has been considered in previous SAMA analyses by operating reactors to extend battery capacity to 24 hours (Reference 8.1-21, page E.2-4; Reference 8.1-27, page E.2-7). The NuScale design provides for 24 hours of battery life to maintain ECCS valve closure, and 72 hours of post accident monitoring. Additionally, each DC separation group has redundant battery capacity.
SAMDA 2	Replace lead-acid batteries with fuel cells.	Extended DC power availability during an SBO.	Combined	Combined with SAMDA 1. The intent of this SAMDA is similar to SAMDA 1, which is already implemented by the NuScale design.
SAMDA 3	Add additional battery charger or portable diesel-driven battery charger to existing DC system.	Improved availability of DC power system.	Already implemented	The BDG switchgear is designed with a 480 V plug-in connection which will be able to accept offsite equipment . Additionally, each DC bus is supplied by two redundant battery chargers.
SAMDA 4	Improve DC bus load shedding.	Extended DC power availability during an SBO.	Already implemented	Manual load shedding is not required in the NuScale design. The MPS automatically deenergizes all loads that are not required for post accident monitoring or required to hold ECCS valves closed.
SAMDA 5	Provide DC bus cross-ties.	Improved availability of DC power system.	Already implemented	The NuScale design has four highly reliable DC power divisions. The redundancy and capacity of the battery divisions meets the intent of this SAMDA.
SAMDA 6	Provide additional DC power to the 120/240V vital AC system.	Increased availability of the 120 V vital AC bus.	Not applicable to the NuScale design	The NuScale design does not rely on AC power as a vital power source. Power is supplied to maintain components in the non-actuated position, and power is removed to actuate components. Safety related systems, such as the DHRS and ECCS, do not require electric power to operate.
SAMDA 7	Add an automatic feature to transfer the 120V vital AC bus from normal to standby power.	Increased availability of the 120 V vital AC bus.	Combined	Combined with SAMDA 6, as they have the same intent.
SAMDA 8	Increase training on response to loss of two 120 V AC buses, which causes inadvertent actuation signals.	Improved chances of successful response to loss of two 120 V AC buses.	Not applicable to the NuScale design	The NuScale design does not rely on AC power as a vital power source. The MPS is supplied directly by DC power. The safety function of MPS is to remove power from the actuator.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 9	Provide an additional diesel generator.	Increased availability of onsite emergency AC power.	Very low benefit	The backup diesel generators are not risk significant components (Table 6-2) and therefore implementing this SAMDA would be of very low benefit.
SAMDA 10	Revise procedure to allow bypass of diesel generator trips.	Extended diesel generator operation.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 11	Improve 4.16 kV bus cross-tie ability.	Increased availability of onsite AC power.	Already implemented	Cross tie capability available for all EMVS busses. Intent considered satisfied by existing design.
SAMDA 12	Create AC power cross-tie capability with other unit (multi-unit site).	Increased availability of onsite AC power.	Combined	Combined with SAMDA 11, which already implements this change in the multi-module NuScale plant.
SAMDA 13	Install an additional, buried offsite power source.	Reduced probability of loss of offsite power.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMA at Cooper Nuclear Station was estimated to be $2,485,000 by Nebraska Public Power District in 2008 (Reference 8.1-21, SAMDA 9, page E.2-33). This exceeds the maximum benefit of $69,000.
SAMDA 14	Install a gas turbine generator.	Increased availability of onsite AC power.	Already implemented	Auxiliary AC power source available in the event of a loss of AC power. BDGs are also available. Intent considered satisfied by existing design.
SAMDA 15	Install tornado protection on gas turbine generator.	Increased availability of onsite AC power.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 16	Improve uninterruptible power supplies.	Increased availability of power supplies supporting front-line equipment.	Already implemented	Approximately 72 hours of battery power are available following accident. Backup diesel generators and alternate AC power source are available in addition to batteries. Intent considered satisfied.
SAMDA 17	Create a cross-tie for diesel fuel oil (multiunit site).	Increased diesel generator availability.	Not required for the NuScale design certification application	While the NuScale design is technically a multiunit site, it is not a multiunit site in the context implied here. There will be two collocated BDGs that service all 12 modules. The intent of this SAMDA is to combine the fuel oil for diesel generators located at different locations. The design certification application is for a single unit site in the context of this SAMDA.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 18	Develop procedures for replenishing diesel fuel oil.	Increased diesel generator availability.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 19	Use fire water system as a backup source for diesel cooling.	Increased diesel generator availability.	Not applicable to the NuScale design	BDGs are independently cooled using either a radiator or heat exchanger driven by the BDG itself and will not rely on plant cooling or service water.
SAMDA 20	Add a new backup source of diesel cooling.	Increased diesel generator availability.	Combined	Combined with SAMDA 19.
SAMDA 21	Develop procedures to repair or replace failed 4 KV breakers.	Increased probability of recovery from failure of breakers that transfer 4.16 kV nonemergency buses from unit station service transformers.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 22	In training, emphasize steps in recovery of offsite power after an SBO.	Reduced human error probability during offsite power recovery.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 23	Develop a severe weather conditions procedure.	Improved offsite power recovery following external weather-related events.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 24	Bury offsite power lines.	Improved offsite power reliability during severe weather.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA by Ameren Missouri at Callaway Plant 1 in 2011 was estimated to be greater than $3,000,000 (Reference 8.1-20, SAMA 24, page F-88). This exceeds the maximum benefit of $69,000.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
Generic PWR Improvements Related to Core Cooling Systems				
SAMDA 25	Install an independent active or passive high pressure injection system.	Improved prevention of core melt sequences.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA at Cooper Nuclear Station was estimated to be $1,000,000 by Nebraska Public Power District in 2008 (Reference 8.1-21, SAMA 22, page E.2-35). The current high pressure injection system (CVCS) in the NuScale design is independent to each module. An additional high pressure injection system would also be module specific. Therefore, this cost estimate is considered an appropriate estimate for an additional 12 independent injection systems and associated equipment. This exceeds the maximum benefit of $69,000.
SAMDA 26	Provide an additional high-pressure injection pump with independent diesel.	Reduced frequency of core melt from small LOCA and SBO sequences.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA at Cooper Nuclear Station was estimated to be $1,000,000 by Nebraska Public Power District in 2008 (Reference 8.1-21, SAMA 23, page E.2-35). The CVCS system is module specific. An additional high pressure injection pump would be required for each module to implement this SAMDA. This exceeds the maximum benefit of $69,000.
SAMDA 27	Revise procedure to allow operators to inhibit automatic vessel depressurization in non-ATWS scenarios.	Extended HPCI and RCIC operation.	Not applicable to the NuScale design	The CVCS injection pumps have an operating range of 100 to 3000 psi. Therefore, slowing reactor depressurization will not likely extend CVCS operation as CVCS can operate over a wide range of pressures.
SAMDA 28	Add a diverse low pressure injection system.	Improved injection capability.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA at Callaway 1 was estimated to be greater than $1,000,000 by Ameren Missouri in 2005 (Reference 8.1-20, SAMA 28, page F-88). This exceeds the maximum benefit of $69,000.
SAMDA 29	Provide capability for alternate injection via diesel-driven fire pump.	Improved injection capability.	Combined	Combined with SAMDA 28.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 30	Improve ECCS suction strainers.	Enhanced reliability of ECCS suction.	Not applicable to the NuScale design	The NuScale design does not have insulation or other material that could come loose and interfere with the ECCS RRVs. The CNV is maintained in a partial vacuum, precluding the need for insulation. Therefore, the addition of strainers to the ECCS design would not improve reliability.
SAMDA 31	Add the ability to manually align emergency core cooling system recirculation.	Enhanced reliability of ECCS suction.	Not applicable to the NuScale design	The ECCS does not have both injection and recirculation modes of operation, only a recirculation mode. The ECCS does not include or require a safety grade source of water for injection. Therefore, manually realigning the ECCS is not applicable for ECCS operation and would not improve ECCS reliability.
SAMDA 32	Add the ability to automatically align emergency core cooling system to recirculation mode upon refueling water storage tank depletion.	Enhanced reliability of ECCS suction.	Combined	Combined with SAMDA 31, as both SAMDAs discuss manual realignment of the ECCS.
SAMDA 33	Provide hardware and procedure to refill the reactor water storage tank once it reaches a specified low level.	Extended reactor water storage tank capacity in the event of a steam generator tube rupture.	Not applicable to the NuScale design	NuScale design does not have a reactor water storage tank.
SAMDA 34	Provide an in-containment reactor water storage tank.	Continuous source of water to the safety injection pumps during a LOCA event, since water released from a breach of the primary system collects in the in-containment reactor water storage tank, and thereby eliminates the need to realign the safety injection pumps for long-term post-LOCA recirculation.	Not applicable to the NuScale design	The ECCS does not have both injection and recirculation modes of operation, only a recirculation mode. The ECCS does not include or require a safety grade source of water for injection. Therefore, realigning the ECCS is not applicable for ECCS operation. Additionally, reactor coolant collects in containment for LOCAs inside containment and the ECCS transfers coolant from the containment to the RPV upon actuation, which is the intent of this SAMDA.
SAMDA 35	Throttle low pressure injection pumps earlier in medium or large-break LOCAs to maintain reactor water storage tank inventory.	Extended reactor water storage tank capacity.	Not applicable to the NuScale design	The NuScale design does not contain large diameter primary coolant system piping, therefore a large break LOCA cannot physically be postulated for the NuScale design. Additionally, the ECCS does not have an injection mode, only a recirculation mode. The ECCS does not include or require a safety grade source of water for injection. Therefore, this SAMDA does not apply to the NuScale design.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 36	Emphasize timely recirculation alignment in operator training.	Reduced human error probability associated with recirculation failure.	Not applicable to the NuScale design	Reactor coolant recirculation is accomplished through ECCS, which is actuated automatically and does not require operator action.
SAMDA 37	Upgrade the chemical and volume control system to mitigate small LOCAs.	For a plant like the Westinghouse AP600, where the chemical and volume control system cannot mitigate a small LOCA, an upgrade would decrease the frequency of core damage.	Already implemented	The CVCS is capable of mitigating small LOCAs under accident conditions.
SAMDA 38	Change the in-containment reactor water storage tank suction from four check valves to two check and two air-operated valves.	Reduced common mode failure of injection paths.	Combined	Combined with SAMDA 34, as both SAMDAs discuss improving reliability of ECCS injection.
SAMDA 39	Replace two of the four electric safety injection pumps with diesel-powered pumps.	Reduced common cause failure of the safety injection system. This SAMA was originally intended for the Westinghouse-CE System 80+, which has four trains of safety injection. However, the intent of this SAMA is to provide diversity within the high- and low-pressure safety injection systems.	Not applicable to the NuScale design	The NuScale ECCS design consists of 5 main valves and does not include pumps. Therefore, reducing the CCF probability of ECCS pumps is not applicable to the NuScale design. Additionally, the ECCS does not include or require a safety grade source of water for injection and therefore diversifying injection systems is not applicable to the NuScale design.
SAMDA 40	Provide capability for remote, manual operation of secondary side pilot-operated relief valves in a station blackout.	Improved chance of successful operation during station blackout events in which high area temperatures may be encountered (no ventilation to main steam areas).	Not applicable to the NuScale design	There are no pilot-operated valves on the secondary side in the NuScale design. The NuScale design is capable of 100 percent turbine bypass steam dump to the condenser that aids in preventing overpressurization of the secondary side.
SAMDA 41	Create a reactor coolant depressurization system.	Allows low pressure emergency core cooling system injection in the event of small LOCA and high-pressure safety injection failure.	Not applicable to the NuScale design	ECCS components have design temperatures and pressures that bound the expected conditions in the RPV, CNV, and UHS for normal and post-accident environments. The ECCS does not include or require a safety grade source of water for injection. Therefore, implementing a SAMDA to allow low pressure ECCS injection is not applicable.
SAMDA 42	Make procedure changes for reactor coolant system depressurization.	Allows low pressure emergency core cooling system injection in the event of small LOCA and high-pressure safety injection failure.	Combined	Combined with SAMDA 41, as both SAMDAs produce the same result.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
Generic PWR Improvements Related to Cooling Water				
SAMDA 43	Add redundant DC control power for SW pumps.	Increased availability of SW.	Very low benefit	Site cooling water (SCW) is not risk significant (Table 6-1). Site cooling water only supports auxiliary systems.
SAMDA 44	Replace ECCS pump motors with air-cooled motors.	Elimination of ECCS dependency on component cooling system.	Not applicable to the NuScale design	The ECCS consists of valves only and does not interface with component cooling water. Intent considered satisfied.
SAMDA 45	Enhance procedural guidance for use of cross-tied component cooling or service water pumps.	Reduced frequency of loss of component cooling water and service water.	Not applicable to the NuScale design	Site cooling water pumps and reactor component cooling water system (RCCWS) pumps are not cross-tied. Site cooling water interfaces with RCCWS only through providing cooling water to the RCCWS heat exchangers.
SAMDA 46	Add a service water pump.	Increased availability of cooling water.	Very low benefit	Site cooling water is not risk significant (Table 6-1). Site cooling water only supports auxiliary systems.
SAMDA 47	Enhance the screen wash system.	Reduced potential for loss of SW due to clogging of screens.	Very low benefit	Site cooling water is not risk significant (Table 6-1). Site cooling water only supports auxiliary systems.
SAMDA 48	Cap downstream piping of normally closed component cooling water drain and vent valves.	Reduced frequency of loss of component cooling water initiating events, some of which can be attributed to catastrophic failure of one of the many single isolation valves.	Very low benefit	RCCWS is not a risk-significant system (Table 6-1). Reactor component cooling water does not support any systems that are required for safe shutdown or to maintain safe shutdown.
SAMDA 49	Enhance loss of component cooling water (or loss of service water) procedures to facilitate stopping the reactor coolant pumps.	Reduced potential for reactor coolant pump seal damage due to pump bearing failure.	Not applicable to the NuScale design	There are no reactor coolant pumps in the NuScale design.
SAMDA 50	Enhance loss of component cooling water procedure to underscore the desirability of cooling down the reactor coolant system prior to seal LOCA.	Reduced probability of reactor coolant pump seal failure.	Not applicable to the NuScale design	There are no reactor coolant pumps in the NuScale design.
SAMDA 51	Additional training on loss of component cooling water.	Improved success of operator actions after a loss of component cooling water.	Very low benefit	RCCWS is not risk significant (Table 6-1). Reactor component cooling water does not support any systems that are required for safe shutdown or to maintain safe shutdown.
SAMDA 52	Provide hardware connections to allow another essential raw cooling water system to cool charging pump seals.	Reduced effect of loss of component cooling water by providing a means to maintain the charging pump seal injection following a loss of normal cooling water.	Very low benefit	RCCWS is not risk significant (Table 6-1). Reactor component cooling water does not support any systems that are required for safe shutdown or to maintain safe shutdown.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 53	On loss of essential raw cooling water, proceduralize shedding component cooling water loads to extend the component cooling water heat-up time.	Increased time before loss of component cooling water (and reactor coolant pump seal failure) during loss of essential raw cooling water sequences.	Very low benefit	RCCWS is not risk significant (Table 6-1). Reactor component cooling water does not support any systems that are required for safe shutdown or to maintain safe shutdown.
SAMDA 54	Increase charging pump lube oil capacity.	Increased time before charging pump failure due to lube oil overheating in loss of cooling water sequences.	Very low benefit	RCCWS is not risk significant (Table 6-1). Reactor component cooling water does not support any systems that are required for safe shutdown or to maintain safe shutdown.
SAMDA 55	Install an independent reactor coolant pump seal injection system, with dedicated diesel.	Reduced frequency of core damage from loss of component cooling water, service water, or station blackout.	Not applicable to the NuScale design	There are no reactor coolant pumps in the NuScale design.
SAMDA 56	Install an independent reactor coolant pump seal injection system, without dedicated diesel.	Reduced frequency of core damage from loss of component cooling water or service water, but not a station blackout.	Not applicable to the NuScale design	There are no reactor coolant pumps in the NuScale design.
SAMDA 57	Use existing hydro test pump for reactor coolant pump seal injection.	Reduced frequency of core damage from loss of component cooling water or service water, but not a station blackout.	Not applicable to the NuScale design	There are no reactor coolant pumps in the NuScale design.
SAMDA 58	Install improved reactor coolant pump seals.	Reduced likelihood of reactor coolant pump seal LOCA.	Not applicable to the NuScale design	There are no reactor coolant pumps in the NuScale design.
SAMDA 59	Install an additional component cooling water pump.	Reduced likelihood of loss of component cooling water leading to a reactor coolant pump seal LOCA.	Not applicable to the NuScale design	There are no reactor coolant pumps in the NuScale design.
SAMDA 60	Prevent makeup pump flow diversion through the relief valves.	Reduced frequency of loss of reactor coolant pump seal cooling if spurious high pressure injection relief valve opening creates a flow diversion large enough to prevent reactor coolant pump seal injection.	Not applicable to the NuScale design	There are no reactor coolant pumps in the NuScale design.
SAMDA 61	Change procedures to isolate reactor coolant pump seal return flow on loss of component cooling water, and provide (or enhance) guidance on loss of injection during seal LOCA.	Reduced frequency of core damage due to loss of seal cooling.	Not applicable to the NuScale design	There are no reactor coolant pumps in the NuScale design.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 62	Implement procedures to stagger high pressure safety injection pump use after a loss of service water.	Extended high pressure injection prior to overheating following a loss of service water.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 63	Use fire prevention system pumps as a backup seal injection and high pressure makeup source.	Reduced frequency of reactor coolant pump seal LOCA.	Not applicable to the NuScale design	There are no reactor coolant pumps in the NuScale design.
SAMDA 64	Implement procedure and hardware modifications to allow manual alignment of the fire water system to the component cooling water system, or install a component cooling water header cross-tie.	Improved ability to cool residual heat removal heat exchangers.	Very low benefit	RCCWS is not risk significant (Table 6-1). Reactor component cooling water does not support any systems that are required for safe shutdown or to maintain safe shutdown.
Generic PWR Improvements Related to Feedwater and Condensate				
SAMDA 65	Install a digital feedwater upgrade.	Reduced chance of loss of main feedwater following a plant trip.	Very low benefit	The condensate and feedwater system (CFWS) is not risk significant (Table 6-1).
SAMDA 66	Create ability for emergency connection of existing or new water sources to feedwater and condensate systems.	Increased availability of feedwater.	Very low benefit	The CFWS is not risk significant (Table 6-1).
SAMDA 67	Install an independent diesel for the condensate storage tank makeup pumps.	Extended inventory in CST during an SBO.	Very low benefit	The CFWS is not risk significant (Table 6-1).
SAMDA 68	Add a motor-driven feedwater pump.	Increased availability of feedwater.	Very low benefit	The CFWS is not risk significant (Table 6-1).
SAMDA 69	Install manual isolation valves around auxiliary feedwater turbine-driven steam admission valves.	Reduced dual turbine-driven pump maintenance unavailability.	Not applicable to the NuScale design	The DHRS removes heat from the steam generators, and is analogous to an auxiliary feedwater system in the NuScale design. The DHRS does not contain pumps, but relies on natural circulation to remove decay heat. Therefore, this SAMDA does not apply to the NuScale design.
SAMDA 70	Install accumulators for turbine-driven auxiliary feedwater pump flow control valves.	Eliminates the need for local manual action to align nitrogen bottles for control air following a loss of offsite power.	Not applicable to the NuScale design	The DHRS removes heat from the steam generators, and is analogous to an auxiliary feedwater system in the NuScale design. The DHRS does not contain pumps, but relies on natural circulation to remove heat. No operator action or electrical power is required to actuate the DHRS valves. Therefore, this SAMDA does not apply to the NuScale design.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 71	Install a new condensate storage tank (auxiliary feedwater storage tank).	Increased availability of the auxiliary feedwater system.	Not applicable to the NuScale design	The DHRS maintains the appropriate liquid water level during standby in normal plant operations. When the DHRS is actuated, it relies on the existing inventory to remove decay heat to the UHS through a closed loop. Therefore, this SAMDA does not apply to the NuScale design.
SAMDA 72	Modify the turbine-driven auxiliary feedwater pump to be self-cooled.	Improved success probability during a station blackout.	Not applicable to the NuScale design	The DHRS removes heat from the steam generators, and is analogous to an auxiliary feedwater system in the NuScale design. The DHRS does not contain pumps, but relies on natural circulation to remove heat. Therefore, this SAMDA does not apply to the NuScale design.
SAMDA 73	Proceduralize local manual operation of auxiliary feedwater system when control power is lost.	Extended auxiliary feedwater availability during a station blackout. Also provides a success path should auxiliary feedwater control power be lost in non-station blackout sequences.	Not applicable to the NuScale design	The DHRS removes heat from the steam generators, and is analogous to an auxiliary feedwater system in the NuScale design. The DHRS is actuated by removing power to the actuation valves and relies on natural circulation to remove heat. Therefore, this SAMDA does not apply to the NuScale design.
SAMDA 74	Provide hookup for portable generators to power the turbine-driven auxiliary feedwater pump after station batteries are depleted.	Extended auxiliary feedwater availability.	Not applicable to the NuScale design	The DHRS removes heat from the steam generators, and is analogous to an auxiliary feedwater system in the NuScale design. The DHRS does not contain pumps, but relies on natural circulation to remove heat. Therefore, this SAMDA does not apply to the NuScale design.
SAMDA 75	Use fire water system as a backup for steam generator inventory.	Increased availability of steam generator water supply.	Very low benefit	The CFWS, which supplies the water for the steam generator tubes, is not risk significant (Table 6-1).
SAMDA 76	Change failure position of condenser makeup valve if the condenser makeup valve fails open on loss of air or power.	Allows greater inventory for the auxiliary feedwater pumps by preventing condensate storage tank flow diversion to the condenser.	Not applicable to the NuScale design	The DHRS removes heat from the steam generators, and is analogous to an auxiliary feedwater system in the NuScale design. The DHRS does not contain pumps, but relies on natural circulation to remove heat. Therefore, this SAMDA does not apply to the NuScale design.
SAMDA 77	Provide a passive, secondary-side heat rejection loop consisting of a condenser and heat sink.	Reduced potential for core damage due to loss-of-feedwater events.	Already implemented	The DHRS is a passive closed loop system that removes decay heat to the UHS using natural circulation.
SAMDA 78	Modify the startup feedwater pump so that it can be used as a backup to the emergency feedwater system, including during a station blackout scenario.	Increased reliability of decay heat removal.	Not applicable to the NuScale design	The feedwater system does not include startup pumps and the normal feedwater pumps provide for the full range of operations. The DHRS, which does not contain pumps, acts as a backup to the CFWS in the event CFWS is not available to remove heat.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 79	Replace existing pilot-operated relief valves with larger ones, such that only one is required for successful feed and bleed.	Increased probability of successful feed and bleed.	Not applicable to the NuScale design	In the NuScale design, successfully cycling one RSV is sufficient to cool the fuel by adding water to the CNV. This provides a heat transfer path to the UHS. Additionally, the NuScale design does not have a safety-related injection system for feed and bleed. Therefore, feed and bleed is not a relevant option.
Generic PWR Improvements Related to Heating, Ventilation, and Air Conditioning				
SAMDA 80	Provide a redundant train or means of ventilation.	Increased availability of components dependent on room cooling.	Very low benefit	HVAC system is not risk significant (Table 6-1).
SAMDA 81	Add a diesel building high temperature alarm or redundant louver and thermostat.	Improved diagnosis of a loss of diesel building HVAC.	Very low benefit	HVAC system is not risk significant (Table 6-1).
SAMDA 82	Stage backup fans in switchgear rooms.	Increased availability of ventilation in the event of a loss of switchgear ventilation.	Very low benefit	HVAC system is not risk significant (Table 6-1).
SAMDA 83	Add a switchgear room high temperature alarm.	Improved diagnosis of a loss of switchgear HVAC.	Very low benefit	HVAC system is not risk significant (Table 6-1).
SAMDA 84	Create ability to switch emergency feedwater room fan power supply to station batteries in a station blackout.	Continued fan operation in a station blackout.	Very low benefit	HVAC system is not risk significant (Table 6-1).
Generic PWR Improvements Related to Instrument Air and Nitrogen Supply				
SAMDA 85	Provide cross-unit connection of uninterruptible compressed air supply.	Increased ability to vent containment using the hardened vent.	Not required for the NuScale design certification application	The design certification application is for a single unit site in the context of this SAMDA. This SAMDA will be re-evaluated by the COL applicant for a multiple 12 module site.
SAMDA 86	Modify procedure to provide ability to align diesel power to more air compressors.	Increased availability of instrument air after a LOOP.	Very low benefit	Instrument air is not risk significant (Table 6-1).
SAMDA 87	Replace service and instrument air compressors with more reliable compressors which have self-contained air cooling by shaft driven fans.	Elimination of instrument air system dependence on service water cooling.	Very low benefit	Instrument air is not risk significant (Table 6-1).
SAMDA 88	Install nitrogen bottles as backup gas supply for safety relief valves.	Extended SRV operation time.	Not applicable to the NuScale design	NuScale RSVs are passive and open when pressure exceeds threshold. No air supply is required.
SAMDA 89	Improve SRV and MSIV pneumatic components.	Improved availability of SRVs and MSIVs.	Not applicable to the NuScale design	RSVs are passive and do not require pneumatic components. Primary MSIVs are not air operated.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
Generic PWR Improvements Related to Containment Phenomena				
SAMDA 90	Create a reactor cavity flooding system.	Enhanced debris coolability, reduced core concrete interaction, and increased fission product scrubbing.	Already implemented	The CNV floods from CFDS or ECCS and a natural circulation path from the core to the CNV is available through the ECCS RRVs. The CVCS is available to add RCS inventory. Intent of cooling ex-vessel debris is considered satisfied by existing design.
SAMDA 91	Install a passive containment spray system.	Improved containment spray capability.	Excessive implementation cost	The implementation of a containment spray system would require a redesign of the reactor module. The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 92	Use the fire water system as a backup source for the containment spray system.	Improved containment spray capability.	Combined	Combined with SAMDA 91, as a containment spray system would first have to be incorporated in the NuScale design.
SAMDA 93	Install an unfiltered, hardened containment vent.	Increased decay heat removal capability for non-ATWS events, without scrubbing released fission products.	Very low benefit	A containment vent would not reduce the risk for the NuScale design as containment overpressure failures are not a risk significant event.
SAMDA 94	Install a filtered containment vent to remove decay heat. Option 1: gravel bed filter; Option 2: multiple venturi scrubbers.	Increased decay heat removal capability for non-ATWS events, with scrubbing of released fission products.	Very low benefit	A containment vent would not reduce the risk for the NuScale design as containment overpressure failures are not a risk significant event.
SAMDA 95	Enhance fire protection system and standby gas treatment system hardware and procedures.	Improved fission product scrubbing in severe accidents.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 96	Provide post-accident containment inerting capability.	Reduced likelihood of hydrogen and carbon monoxide gas combustion.	Very low benefit	Hydrogen and carbon monoxide gas combustion is not a risk significant event. Containment overpressurization is also not a risk significant event.
SAMDA 97	Create a large concrete crucible with heat removal potential to contain molten core debris.	Increased cooling and containment of molten core debris. Molten core debris escaping from the vessel is contained within the crucible and a water cooling mechanism cools the molten core in the crucible, preventing melt-through of the basemat.	Very low benefit	The core debris will be retained successfully during all severe accidents with intact containment for the NuScale design.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 98	Create a core melt source reduction system.	Increased cooling and containment of molten core debris. Refractory material would be placed underneath the reactor vessel such that a molten core falling on the material would melt and combine with the material. Subsequent spreading and heat removal from the vitrified compound would be facilitated, and concrete attack would not occur.	Combined	Combined with SAMDA 97.
SAMDA 99	Strengthen primary/secondary containment (e.g., add ribbing to containment shell).	Reduced probability of containment overpressurization.	Very low benefit	The CNV is a second pressure vessel with a design pressure of 1000 psi. Containment overpressurization is not a risk significant event.
SAMDA 100	Increase depth of the concrete basemat or use an alternate concrete material to ensure melt-through does not occur.	Reduced probability of basemat melt-through.	Combined	Combined with SAMDA 98.
SAMDA 101	Provide a reactor vessel exterior cooling system.	Increased potential to cool a molten core before it causes vessel failure, by submerging the lower head in water.	Already implemented	The CNV is a second pressure vessel surrounding the RPV. The CNV floods, creating a heat transfer path from the RPV to the UHS (reactor pool).
SAMDA 102	Construct a building to be connected to primary/secondary containment and maintained at a vacuum.	Reduced probability of containment overpressurization.	Very low benefit	Containment overpressurization is not a risk significant event. Containment overpressurization is mitigated by a high design pressure of 1000 psi and cooling from the UHS on the exterior of the CNV and cooling from containment flooding on the interior of the CNV.
SAMDA 103	Institute simulator training for severe accident scenarios.	Improved arrest of core melt progress and prevention of containment failure.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 104	Improve leak detection procedures.	Increased piping surveillance to identify leaks prior to complete failure. Improved leak detection would reduce LOCA frequency.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 105	Delay containment spray actuation after a large LOCA.	Extended reactor water storage tank availability.	Not applicable to the NuScale design	There is no containment spray system in the NuScale design.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 106	Install automatic containment spray pump header throttle valves.	Extended time over which water remains in the reactor water storage tank, when full containment spray flow is not needed.	Not applicable to the NuScale design	There is no containment spray system in the NuScale design.
SAMDA 107	Install a redundant containment spray system.	Increased containment heat removal ability.	Very low benefit	Containment overpressurization is not a risk significant event in the NuScale design. Additionally, the CNV is submerged in the UHS and heat is removed to the UHS directly through conduction.
SAMDA 108	Install an independent power supply to the hydrogen control system using either new batteries, a non-safety grade portable generator, existing station batteries, or existing AC/DC independent power supplies, such as the security system diesel.	Reduced hydrogen detonation potential.	Very low benefit	Hydrogen detonation in the containment is not risk significant in the NuScale design.
SAMDA 109	Install a passive hydrogen control system.	Reduced hydrogen detonation potential.	Combined	Combined with SAMDA 108, as they have the same intent.
SAMDA 110	Erect a barrier that would provide enhanced protection of the containment walls (shell) from ejected core debris following a core melt scenario at high pressure.	Reduced probability of containment failure.	Very low benefit	High pressure melt ejection is not a threat to containment integrity in the NuScale design.
Generic PWR Improvements Related to Containment Bypass				
SAMDA 111	Install additional pressure or leak monitoring instruments for detection of ISLOCAs.	Reduced ISLOCA frequency.	Already implemented	The MPS monitors multiple different parameters to detect CNV bypass LOCAs. The use of additional monitoring instruments is considered satisfied.
SAMDA 112	Add redundant and diverse limit switches to each containment isolation valve.	Reduced frequency of containment isolation failure and ISLOCAs.	Excessive implementation cost	The cost of adding redundant and diverse limit switches to each containment isolation valve for all 12 modules is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 113	Increase leak testing of valves in ISLOCA paths.	Reduced ISLOCA frequency.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 114	Install self-actuating containment isolation valves.	Reduced frequency of isolation failure.	Already implemented	CNV isolation valves fail closed on loss of power. ESFAS automatically isolates containment.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 115	Locate residual heat removal (RHR) inside containment.	Reduced frequency of ISLOCA outside containment.	Not applicable to the NuScale design	DHRS depends on being located outside containment to remove decay heat to UHS. Additionally, the DHRS is designed to primary system operating pressure which reduces the risk of a leak path that bypasses containment.
SAMDA 116	Ensure ISLOCA releases are scrubbed. One method is to plug drains in potential break areas so that break point will be covered with water.	Scrubbed ISLOCA releases.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA at Callaway was estimated by Ameren Missouri to require more than $1,000,000 in 2011 (Reference 8.1-20, SAMA 116, page F-92). This exceeds the maximum benefit of $69,000.
SAMDA 117	Revise EOPs to improve ISLOCA identification.	Increased likelihood that LOCAs outside containment are identified as such. A plant had a scenario in which an RHR ISLOCA could direct initial leakage back to the pressurizer relief tank, giving indication that the LOCA was inside containment.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 118	Improve operator training on ISLOCA coping.	Decreased ISLOCA consequences.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 119	Institute a maintenance practice to perform a 100 percent inspection of steam generator tubes during each refueling outage.	Reduced frequency of steam generator tube ruptures.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 120	Replace steam generators with a new design.	Reduced frequency of steam generator tube ruptures.	Already implemented	The NuScale design uses helical coil steam generators, which differ from the steam generator design of a typical PWR. The NuScale steam generators have the high pressure primary coolant on the outside of the steam generator tubes, maintaining the tubes in a constant state of compression and minimizing the likelihood of tube failure.
SAMDA 121	Increase the pressure capacity of the secondary side so that a steam generator tube rupture would not cause the relief valves to lift.	Eliminates release pathway to the environment following a steam generator tube rupture.	Already implemented	Secondary side piping is rated for operating pressure.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 122	Install a redundant spray system to depressurize the primary system during a steam generator tube rupture.	Enhanced depressurization capabilities during steam generator tube rupture.	Excessive implementation cost	The cost of implementing this SAMDA includes the redesign of the RPV and CNV in addition to the design of a new spray system and the materials associated with the new system. The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 123	Proceduralize use of pressurizer vent valves during steam generator tube rupture sequences.	Backup method to using pressurizer sprays to reduce primary system pressure following a steam generator tube rupture.	Not applicable to the NuScale design	There are no pressurizer vent valves, in the context of the functionality envisioned by this SAMDA, in the NuScale design.
SAMDA 124	Provide improved instrumentation to detect steam generator tube ruptures, such as Nitrogen-16 monitors.	Improved mitigation of steam generator tube ruptures.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA at Kewaunee Power Station was estimated to be $149,746 by Dominion in 2008 (Reference 8.1-22, SAMA ID 124, page F-248). This exceeds the maximum benefit of $69,000.
SAMDA 125	Route the discharge from the main steam safety valves through a structure where a water spray would condense the steam and remove most of the fission products.	Reduced consequences of a steam generator tube rupture.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA at Kewaunee Power Station was estimated to be $2,700,000 by Dominion in 2008 (Reference 8.1-22, SAMA ID 125, page F-248). This exceeds the maximum benefit of $69,000.
SAMDA 126	Install a highly reliable (closed loop) steam generator shell-side heat removal system that relies on natural circulation and stored water sources.	Reduced consequences of a steam generator tube rupture.	Already implemented	DHRS is a closed loop steam generator shell-side heat removal system that relies on natural circulation and stored water sources.
SAMDA 127	Revise emergency operating procedures to direct isolation of a faulted steam generator.	Reduced consequences of a steam generator tube rupture.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 128	Direct steam generator flooding after a steam generator tube rupture, prior to core damage.	Improved scrubbing of steam generator tube rupture releases.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 129	Vent main steam safety valves in containment.	Reduced consequences of a steam generator tube rupture.	Not applicable to the NuScale design	The steam generator tubes are located inside the RPV. Main steam system piping inside containment is designed for full RCS pressure and a Steam Generator Tube Failure (SGTF) can be isolated by closing the MSIVs and FWIVs. A SGTF is detected with the same signal that trips containment isolation. The main steam safety valves are located downstream of the MSIVs and are not expected to lift following a reactor trip.
Generic PWR Improvements Related to ATWS				
SAMDA 130	Add an independent boron injection system.	Improved availability of boron injection during ATWS.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing this SAMDA at Arkansas Nuclear One Unit 2 was estimated to be over $300,000 in 2003 (Reference 8.1-19, SAMA AT-02, page E-37). This exceeds the maximum benefit of $69,000.
SAMDA 131	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.	Improved equipment availability after an ATWS.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA at Kewaunee Power Station was estimated to be $700,000 by Dominion in 2008 (Reference 8.1-22, SAMA ID 131, page F-249). This exceeds the maximum benefit of $69,000.
SAMDA 132	Provide an additional control system for rod insertion (e.g., AMSAC).	Improved redundancy and reduced ATWS frequency.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA at Shearon Harris was estimated by Progress Energy to require $400,000 in 2006 (Reference 8.1-23, SAMA 16, page E-108). This exceeds the maximum benefit of $69,000.
SAMDA 133	Install an ATWS sized filtered containment vent to remove decay heat.	Increased ability to remove reactor heat from ATWS events.	Very low benefit	Containment overpressurization is not a risk significant event in the NuScale design. Additionally, the CNV is submerged in the UHS and heat is removed to the UHS directly through conduction.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 134	Revise procedure to bypass MSIV isolation in turbine trip ATWS scenarios.	Affords operators more time to perform actions. Discharge of a substantial fraction of steam to the main condenser (i.e., as opposed to into the primary containment) affords the operator more time to perform actions (e.g., SLC injection, lower water level, depressurize RPV) than if the main condenser was unavailable, resulting in lower human error probabilities.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 135	Revise procedure to allow override of low pressure core injection during an ATWS event.	Allows immediate control of low pressure core injection. On failure of high pressure core injection and condensate, some plants direct reactor depressurization followed by five minutes of automatic low pressure core injection.	Not applicable to the NuScale design	Low pressure injection (CFDS) must be initiated by the operators under appropriate conditions and is not automatic.
SAMDA 136	Install motor generator set trip breakers in control room.	Reduced frequency of core damage due to an ATWS.	Combined	Combined with SAMDA 137. The intent of these SAMDAs is to remove power from the control rods, allowing rod insertion.
SAMDA 137	Provide capability to remove power from the bus powering the control rods.	Decreased time required to insert control rods if the reactor trip breakers fail (during a loss of feedwater ATWS which has rapid pressure excursion).	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
Generic PWR Improvements Related to Internal Flooding				
SAMDA 138	Improve inspection of rubber expansion joints on main condenser.	Reduced frequency of internal flooding due to failure of circulating water system expansion joints.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 139	Modify swing direction of doors separating turbine building basement from areas containing safeguards equipment.	Prevents flooding propagation.	Not applicable to the NuScale design	Safeguards equipment is not in danger of damage due to turbine building flooding.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
Generic PWR Improvements to Reduce Seismic Risk				
SAMDA 140	Increase seismic ruggedness of plant components.	Increased availability of necessary plant equipment during and after seismic events.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA at Kewaunee Power Station was estimated to exceed $5,089,322 by Dominion in 2008 (Reference 8.1-22, pages F-27 and F-228). This exceeds the maximum benefit of $69,000.
SAMDA 141	Provide additional restraints for CO2 tanks.	Increased availability of fire protection given a seismic event.	Not applicable to the NuScale design	Fire protection system does not use CO2 tanks.
Generic PWR Improvements to Reduce Fire Risk				
SAMDA 142	Replace mercury switches in fire protection system.	Decreased probability of spurious fire suppression system actuation.	Not applicable to the NuScale design	There are no mercury switches in the fire protection system.
SAMDA 143	Upgrade fire compartment barriers.	Decreased consequences of a fire.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMA at Crystal River was estimated as $150,000 by Duke Energy in 2008 (Reference 8.1-25, SAMA 49, page E.6-30). This exceeds the maximum benefit of $69,000.
SAMDA 144	Install additional transfer and isolation switches.	Reduced number of spurious actuations during a fire.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA at Columbia Generating Station was estimated as $2,000,000 by Energy Northwest in 2008 (Reference 8.1-26, SAMA ID FR-03, page E-213). This exceeds the maximum benefit of $69,000.
SAMDA 145	Enhance fire brigade awareness.	Decreased consequences of a fire.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 146	Enhance control of combustibles and ignition sources.	Decreased fire frequency and consequences.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
Generic PWR Other Improvements				
SAMDA 147	Install digital large break LOCA protection system.	Reduced probability of a large break LOCA (a leak before break).	Not applicable to the NuScale design	The NuScale design does not contain large diameter primary coolant system piping, therefore a large break LOCA cannot physically be postulated for the NuScale design.
SAMDA 148	Enhance procedures to mitigate large break LOCA.	Reduced consequences of a large break LOCA.	Not applicable to the NuScale design	The NuScale design does not contain large diameter primary coolant system piping, therefore a large break LOCA cannot physically be postulated for the NuScale design.
SAMDA 149	Install computer aided instrumentation system to assist the operator in assessing post-accident plant status.	Improved prevention of core melt sequences by making operator actions more reliable.	Already implemented	The control room uses computer aided digital monitoring.
SAMDA 150	Improve maintenance procedures.	Improved prevention of core melt sequences by increasing reliability of important equipment.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 151	Increase training and operating experience feedback to improve operator response.	Improved likelihood of success of operator actions taken in response to abnormal conditions.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 152	Develop procedures for transportation and nearby facility accidents.	Reduced consequences of transportation and nearby facility accidents.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 153	Install secondary side guard pipes up to the main steam isolation valves.	Prevents secondary side depressurization should a steam line break occur upstream of the main steam isolation valves. Also guards against or prevents consequential multiple steam generator tube ruptures following a main steam line break event.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the cost of implementing a similar SAMDA at Callaway 1 was estimated as being greater than $1,000,000 in 2011 (Reference 8.1-20, SAMA 153, page F-93). This exceeds the maximum benefit of $69,000.
NuScale Specific Improvements Related to AC and DC Power				
SAMDA 154	Improve auxiliary AC power supply testing and maintenance procedures.	Improved reliability of the AAPS in the event of a loss of AC power.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 155	Provide redundant AAPS.	Improved availability of the AAPS in the event of a Loss of AC power.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. For example, the estimated cost of implementation of a similar SAMDA at Vermont Yankee was estimated to be greater than $2,000,000 in 2006 (Reference 8.1-27, SAMA 31, page E.2-30). This exceeds the maximum benefit of $69,000.
NuScale Specific Improvements Related to Core Cooling Systems				
SAMDA 156	Improve reliability of CFDS isolation valves.	Reduced core damage frequency. Improved availability of reactor coolant makeup.	Excessive implementation cost	The cost of implementing this SAMDA for 12 modules is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 157	Improve operator training to start containment flooding.	Reduced core damage frequency.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 158	Improve the reliability of the CVCS makeup combining valve.	Improves reliability of CVCS makeup combining valve to open to DWS storage tank to reduce core damage frequency.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 159	Improve operator training to start CVCS injection.	Reduced core damage frequency.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 160	Install an additional, diverse, parallel DWS CVCS pump isolation valve.	Improved reliability of CVCS injection.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 161	Train operators to open CVCS letdown line containment isolation when RSVs fail to open.	RPV is protected against overpressurization in the event of RSV failure.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 162	Establish capability and train operators to divert CVCS flow from one module to another through the module heat-up system.	Improve CVCS injection availability when CVCS injection to a module fails.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 163	Diversify ECCS recirculation valves.	Improved reliability of ECCS. Common cause failure of these valves is risk significant. Only one recirculation valve is required.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 164	Provide additional, diverse ECCS RRV.	Improved reliability of ECCS. Common cause failure of these valves is risk significant. Only one recirculation valve is required.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 165	Diversify ECCS vent valves.	Improved reliability of ECCS. Common cause failure of these valves is risk significant. Only one vent valve is required.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 166	Provide additional, diverse ECCS RVV.	Improved reliability of ECCS. Common cause failure of these valves is risk significant. Only one vent valve is required.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 167	Add block valves to ECCS.	ECCS valves fail open on loss of DC power. In an event where ECCS valves are not required for successful mitigation of an accident, block valves would prevent the ECCS valves from opening.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 168	Add remote manual bypass of ECCS trip valve.	This would allow ECCS to actuate when the trip valve in the control portion of ECCS fails.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 169	Lower elevation of ECCS RRVs to just above the top of the reactor core.	Improved ability of the ECCS to remove decay heat in a scenario where the CNV liquid level is higher in elevation than the top of the fuel, but not high enough in elevation to recirculate to the RPV through the RRVs.	Excessive implementation cost	The implementation of this SAMDA would require a redesign of the RPV, lower CNV, and revisions to thermal hydraulic calculations of the NuScale module. The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 170	Reduce CNV volume below ECCS RRV elevation.	Improved availability of the ECCS to remove decay heat by required less coolant to support recirculation.	Excessive implementation cost	The implementation of this SAMDA would require a redesign of the lower CNV and revisions to thermal hydraulic calculations of the NuScale module. The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 171	Improve testing and maintenance on ECCS valves.	Improved reliability of ECCS.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 172	Diversify DHRS Actuation Valves.	Improved reliability of DHRS. Common cause failure of these valves is risk significant.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 173	Improve testing and maintenance of DHRS actuation valves.	Improved reliability of DHRS. Common cause failure of these valves is risk significant.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 174	Institute a maintenance practice to perform a 100 percent inspection of DHRS during each refueling outage.	Improved reliability of DHRS. Common cause failure of both trains of DHRS due to loose parts is risk significant.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 175	Add removable screens to DHRS.	Improved reliability of DHRS. Common cause failure of both trains of DHRS due to plugging is risk significant.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 176	Improve operator training to actuate DHRS.	Improve DHRS availability when DHRS fails to actuate automatically.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 177	Improve reliability of DHRS actuation valve components.	Reduced core damage frequency.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 178	Provide backup means of removing decay heat.	Improved availability of decay heat removal when DHRS is not functional.	Already implemented	ECCS actuation creates direct heat removal path to UHS.
SAMDA 179	Provide redundant, diverse FW isolation valve.	Reduces the probability of common cause failures of feedwater isolation. Improves DHRS availability.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 180	Improve testing and maintenance procedures on FWIVs.	Improved reliability of feedwater isolation. Improves DHRS availability.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 181	Provide redundant, diverse MS isolation valve.	Reduces the probability of common cause failures of main steam isolation. Improves DHRS availability.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 182	Improve testing and maintenance procedures on MSIVs.	Improved reliability of main steam isolation. Improves DHRS availability.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
NuScale Specific Improvements Related to Feedwater and Condensate				
	None identified.			
NuScale Specific Improvements Related to Containment Bypass				
SAMDA 183	Diversify CES containment isolation valves.	Improve reliability of CES containment isolation valves to close to prevent release outside of containment. Common cause failure of both CES isolation valves to close is risk significant.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 184	Diversify CVCS containment isolation valves.	Improve reliability of CVCS containment isolation valves to close to prevent release outside of containment. Common cause failure of both CVCS isolation valves to close is risk significant.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
NuScale Specific Improvements Related to ATWS				
SAMDA 185	Improve testing and maintenance of control rod system.	Improved reliability of control rods to insert. Failure of rods to insert upon demand is risk significant.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
SAMDA 186	Add a third, diverse division of reactor trip breakers.	Improved reliability of reactor trip system.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 187	Improve testing and maintenance of reactor trip breakers.	Improved reliability of reactor trip system.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.
NuScale Specific Improvements Related to Internal Flooding				
	None identified.			

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
NuScale Specific Improvements to Reduce Seismic Risk				
SAMDA 188	Base mat isolation.	Reduced seismic stresses during seismic event.	Excessive implementation cost	This design change would require a redesign of the reactor building. The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 189	Improve RPV support skirt.	Increase the ruggedness of the reactor module in a seismic event.	Excessive implementation cost	This design change would require a redesign of the RPV, CNV, and possibly the reactor building crane for the change in reactor module weight. The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
NuScale Specific Improvements to Reduce Fire Risk				
SAMDA 190	Use 3 hour fire cable for safety related equipment.	Increased reliability of equipment during a fire.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
NuScale Specific Other Improvements				
SAMDA 191	Diversify existing RSVs.	Improve reliability of RSVs to prevent RPV overpressurization. Common cause failure of both RSVs to open is risk significant.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 192	Add a third, diverse RSV.	Improve reliability of RSVs to prevent RPV overpressurization. Common cause failure of both RSVs to open is risk significant.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 193	Improve reliability of RSV components.	Reduced core damage frequency.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 194	Diversify RCS level sensors.	Improve reliability of ECCS actuation due to low RPV level. The common cause failure of these sensors is risk significant.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 195	Diversify process logic elements.	Reduced core damage frequency.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.

Table A-1: Screening of Proposed SAMDAs (Continued)

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 196	Improve redundancy in removing control power to equipment.	Improved reliability of ECCS, DHRS, CVCS, and RTBs when APLs fail to de-energize control solenoids.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 197	Improve reliability of EIM components.	Improved reliability of component actuation.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 198	Improve reliability of manual switch components.	Improved reliability of component actuation.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 199	Improve reliability of SVM components.	Improved reliability of component actuation.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $100,000. This exceeds the maximum benefit of $69,000.
SAMDA 200	Improve reactor building crane hoist reliability.	Reduced core damage frequency from reactor building crane failures.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $450,000. This exceeds the maximum benefit of $69,000.
SAMDA 201	Improve redundancy in reactor building crane hoist components.	Reduced core damage frequency from reactor building crane failures.	Excessive implementation cost	The cost of implementing this SAMDA is expected to be greater than $450,000. This exceeds the maximum benefit of $69,000.
SAMDA 202	Provide a railway system on the reactor pool floor to assist in transporting the reactor module to the refueling area.	Reduced core damage frequency from reactor building crane failures.	Excessive implementation cost	This would require the redesign of the reactor building. The cost of implementing this SAMDA is expected to be greater than $450,000. This exceeds the maximum benefit of $69,000.
SAMDA 203	Improve testing and maintenance procedures for the reactor building crane.	Reduced core damage frequency from reactor building crane failures.	Not required for the NuScale design certification application	Training/procedural changes are not required for design certification.

Appendix B Offsite Consequence Calculation

The purpose of this appendix is to show key assumptions used in the calculation of the APD and the Monetary Value of Offsite Economic Impact. These two parameters are used in Section 5.2 and Section 5.3 to calculate the APE and AOC, respectively, as part of the calculation of the maximum benefit.

The MACCS 3.10.0 code in conjunction with Level 2 PRA results and Environmental Report-specific MELCOR 1.8.6 analyses were utilized. The MELCOR simulations used to support this report were based on a previous design of the NuScale plant and an older version of the MELCOR code (1.8.6). A portion of the release category simulations have since been reevaluated with a MELCOR version 2.1 model based on the NuScale design in the design certification application. These simulations show a smaller radionuclide release from the fuel with similar accident timing compared to the MELCOR simulations used in this report. A reduction in radionuclide release would be expected across all release categories if the MELCOR version 2.1 model based on the NuScale design submitted with the design certification application were used to update the results in this report. Because over 99 percent of the contribution to the maximum benefit calculated in Section 5.0 is from onsite consequences, and the onsite consequences are dependent upon recommended onsite dose, cleanup cost, and decontamination cost estimates from NUREG/BR-0184 and not the MELCOR results, the end result of the maximum benefit calculation and SAMDA evaluation would not be significantly affected by updating the MELCOR simulations.

B.1 Offsite Models

A goal of this design certification SAMDA analysis is to use realistic, representative site data for the evaluation of offsite consequences. Since there is no specific site associated with the NuScale design certification application, the Surry Power Station with population extrapolated to 2060 is used in this analysis. The Surry Power Station in 2060 is viewed as a realistic and representative site given the large amount of available data and research performed by the NRC State-of-the-Art Reactor Consequence Analysis (SOARCA) project on the Surry site. A sensitivity study using data representative of the Peach Bottom site is also provided.

NUREG-7110 Volumes 1 & 2 (Reference 8.1-18 & Reference 8.1-28) were used to form the basis of the inputs to the NuScale MACCS model.

B.1.1 Meteorological Modeling

The site and weather information is in 64 azimuthal sectors. Meteorological data is sampled in a stratified, random manner, taking readings for wind speed, wind direction, atmospheric stability category, and rain rate from the file every hour over the entire year. Using this sampling approach, start times are randomly sampled from equally-spaced intervals specified by the parameter NSMPLS, the number of sequences to be chosen from each day of the year. In these analyses NSMPLS is set to 24, and each hour of the year of the supplied meteorological data is used as the starting point for the atmospheric transport and dispersion calculations.

B.1.2 Dose Conversion Factors

The dose conversion factors are used from the "Fgr13dcf.inp" file included in the MACCS 3.10.0 installation package, which includes factors from Federal Guidance Report (FGR)-11 (for inhaled exposure), FGR-12 (for external exposure), and FGR-13 (for cancer risk coefficients), with the chronic inhalation dose factors for the 69 SOARCA nuclides updated with the mean values from NUREG/CR-7155 SOARCA Uncertainty Analysis Table 4.2-8 (Reference 8.1-1).

B.1.3 Economic Data Updates

The economic data is updated by multiplying the regional economic data block values corresponding to the total annual farm sales for the region, farmland property value for the region, and nonfarm property value for the region by a factor of 1.260. This value is calculated as shown below:

$$\frac{\textit{May 2016 Urban Consumer Price Index}}{\textit{January 2005 Urban Consumer Price Index}} = \frac{240.236}{190.7} = 1.260$$

The values for the CPI were used from the CPI Detailed Report for May 2016 (Reference 8.1-10, Table 24), which includes the historical CPI dating back to 1913 for each month of the year. The CPI for May 2016 is used as it is the most recent CPI available at the time of the creation of this report.

This economic projection approach is adequate because the same process is used in NUREG/CR-7009, "MACCS Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project" (Reference 8.1-11, Section 4.1.2). In the SOARCA project, the site file economic information is updated to the target year by first determining the ratio of the present CPI to the past CPI. Then, the economic information is multiplied by this ratio to perform the update. The regional CPIs for Virginia and Pennsylvania are also viable for updating economic parameters at the Surry and Peach Bottom sites, respectively. Using regional CPIs produces a similar ratio to the national CPI, thus national CPI is preferred for the ease of updating multiple U.S. sites.

B.1.4 Population Data Extrapolation

NEI 05-01 recommends a population distribution within 50 miles (80 km) of the site that is extrapolated to a predicted population for a year within the second half of the period of operation. The expected lifetime of a NuScale plant is 60 years with the first plant expected to be brought online in 2024. Extrapolation of the population from 2005 to 2060 satisfies this recommendation.

The population data was updated by multiplying the population data block values by a factor of 1.406. This value was calculated as shown below:

$$\frac{\textit{U.S. Population 2060}}{\textit{U.S. Population 2005}} = \frac{416{,}795{,}000}{296{,}410{,}404} = 1.406$$

The value for the estimated population of the U.S. in 2060 was used from Projections of the Size and Composition of the U.S. Population: 2014-2060 (Reference 8.1-12). The value for the

population of the U.S. in 2005 was used from Annual Estimates of the Population for the United States and States, and for Puerto Rico: April 1, 2000 to July 1, 2005 (Reference 8.1-13). The population is not estimated to a year later than 2060 because that is the latest date to which the U.S. national population was projected by the U.S. Census Bureau in a referenceable document at the time this report was constructed.

This population projection approach is adequate as the same process is used in NUREG/CR-7009. In the SOARCA project, the site file population information is updated to the target year by applying a ratio of a projected future national population to the previous national population at the time of site file creation. The SOARCA projects also used the U.S. Census for the total U.S. population data for the population projection. Available regional population projections were not significantly different than the national average population projections and therefore the choice of using national instead of regional data is not taken to be of consequence in the context of the scope of this report.

B.1.5 Surry Site

The Surry site is defined in the SOARCA Surry 2005 site file (Reference 8.1-14), with a surface roughness of 0.1m (suburban) and the population distribution defined in Section B.1.5.1. The SOARCA Surry site file has been updated from 2005 to 2016 economic information and the population extrapolated to 2060 (Table B-1) per Section B.1.3 and Section B.1.4 derived factors.

B.1.5.1 Surry Population Information

The extrapolated population distribution is shown in Table B-1.

Table B-1: Surry 2060 population distribution

Sector	0-10 mi	10-20 mi	20-30 mi	30-40 mi	40-50 mi	50 mi total
1 [N]	5115	1840	1450	1227	1549	11182
2	5115	1840	1450	1227	1549	11182
3	4366	2507	2439	2591	2116	14019
4	3613	3175	3428	3954	2683	16852
5 [NNE]	3613	3175	3428	3954	2683	16852
6	3613	3175	3428	3954	2683	16852
7	2250	3151	3000	2549	1341	12291
8	889	3128	2573	1143	0	7733
9 [NE]	889	3128	2573	1143	0	7733
10	889	3128	2573	1143	0	7733
11	1303	3606	1518	591	1019	8038
12	1720	4082	463	39	2037	8340
13 [ENE]	1720	4082	463	39	2037	8340
14	1720	4082	463	39	2037	8340
15	6241	9226	273	325	1346	17410
16	10763	14369	83	612	652	26479
17 [E]	10763	14369	83	612	652	26479
18	10763	14369	83	612	652	26479
19	10674	35953	19690	15976	16866	99160
20	10586	57536	39299	31343	33079	171843
21 [ESE]	10586	57536	39299	31343	33079	171843

Table B-1: Surry 2060 population distribution (Continued)

Sector	0-10 mi	10-20 mi	20-30 mi	30-40 mi	40-50 mi	50 mi total
22	10586	57536	39299	31343	33079	171843
23	5294	34298	47029	94153	50527	231301
24	0	11060	54761	156963	67974	290758
25 [SE]	0	11060	54761	156963	67974	290758
26	0	11060	54761	156963	67974	290758
27	96	8308	32661	83282	34835	159182
28	194	5558	10563	9600	1696	27611
29 [SSE]	194	5558	10563	9600	1696	27611
30	194	5558	10563	9600	1696	27611
31	346	3314	6740	7460	2070	19930
32	499	1067	2916	5320	2445	12248
33 [S]	499	1067	2916	5320	2445	12248
34	499	1067	2916	5320	2445	12248
35	321	761	2244	5597	1929	10852
36	143	453	1575	5874	1414	9460
37 [SSW]	143	453	1575	5874	1414	9460
38	143	453	1575	5874	1414	9460
39	180	426	1198	3303	1618	6725
40	219	404	823	730	1824	3999
41 [SW]	219	404	823	730	1824	3999
42	219	404	823	730	1824	3999
43	299	458	1062	1189	1651	4659
44	382	515	1303	1648	1479	5327
45 [WSW]	382	515	1303	1648	1479	5327
46	382	515	1303	1648	1479	5327
47	302	408	1208	12184	7823	21925
48	224	302	1112	22722	14167	38527
49 [W]	224	302	1112	22722	14167	38527
50	224	302	1112	22722	14167	38527
51	146	720	1375	19275	46319	67835
52	72	1142	1638	15827	78472	97150
53 [WNW]	72	1142	1638	15827	78472	97150
54	72	1142	1638	15827	78472	97150
55	1310	1004	1631	10551	60423	74919
56	2549	868	1623	5273	42375	52687
57 [NW]	2549	868	1623	5273	42375	52687
58	2549	868	1623	5273	42375	52687
59	3975	3197	2018	3020	22552	34762
60	5400	5526	2414	768	2730	16838
61 [NNW]	5400	5526	2414	768	2730	16838
62	5400	5526	2414	768	2730	16838
63	5257	3682	1932	998	2140	14009
64	5115	1840	1450	1227	1549	11182
Total	**169465**	**444089**	**504086**	**1052173**	**1018308**	**3188122**

B.1.5.2 Surry Economic Information

NEI 05-01 prescribes that economic data be expressed in current dollars. Extrapolation of the economic data from 2005 to 2016 satisfies this prescription. MACCS economic parameter values for the Surry site are shown in Table B-2. The depreciation rate and investment rate of return are unchanged.

Table B-2: Surry Economic Parameters

Variable	Description	Value
DPRATE	Property depreciation rate (per year)	0.2
DSRATE	Investment rate of return (per year)	0.12
EVACST	Daily cost for a person who has been evacuated ($/person-day)	217
POPCST	Population relocation cost ($/person)	15120
RELCST	Daily cost for a person who is relocated ($/person-day)	217
CDFRM	Cost of farm decontamination for various levels of decontamination ($/hectare)	1) 1675 2) 3730
CDNFRM	Cost of non-farm decontamination per resident for various levels of decontamination ($/person)	1) 8959 2) 23940
DLBCST	Average cost of decontamination labor ($/person-year)	105840
VALWF	Value of farm wealth ($/hectare)	8694
VALNWF	Value of non-farm wealth ($/person)	277200

B.1.5.3 Surry Weather Information

Meteorological data used in this evaluation is the same as that used in the Surry SOARCA analysis (Reference 8.1-15). The licensee of the Surry power plant provided two years of meteorological information to Sandia National Laboratories and the NRC. Different trends in the meteorology over the two years were estimated to have a relatively minor effect on end results (Reference 8.1-11, Section 4.3.5). The data recovery rate for the Surry 2004 meteorological data was approximately 99 percent, meaning that wind speed, wind direction, atmospheric stability, and precipitation were measured for a given hour for approximately 99 percent of the hours of the year.

B.1.6 Peach Bottom Site

The Peach Bottom site is defined in the SOARCA Peach Bottom 2005 site file (Reference 8.1-16) with a surface roughness of 0.1m (suburban) and the population distribution defined in Section B.1.6.1. The SOARCA Peach Bottom site file has been updated from 2005 to 2016 economic information and the population extrapolated to 2060 per Section B.1.3 and Section B.1.4 derived factors.

B.1.6.1 Peach Bottom Population Information

The extrapolated population distribution is shown in Table B-3.

Table B-3: Peach Bottom 2060 Population Distribution

Sector	0-10 mi	10-20 mi	20-30 mi	30-40 mi	40-50 mi	50 mi total
1 [N]	867	33495	39001	10891	20497	104751
2	867	33495	39001	10891	20497	104751

Table B-3: Peach Bottom 2060 Population Distribution (Continued)

Sector	0-10 mi	10-20 mi	20-30 mi	30-40 mi	40-50 mi	50 mi total
3	1047	20550	25462	14788	49313	111160
4	1227	7603	11923	18687	78130	117570
5 [NNE]	1227	7603	11923	18687	78130	117570
6	1227	7603	11923	18687	78130	117570
7	987	5479	12368	16432	59575	94841
8	744	3356	12815	14177	41020	72112
9 [NE]	744	3356	12815	14177	41020	72112
10	744	3356	12815	14177	41020	72112
11	716	2933	12148	32573	56218	104588
12	688	2511	11482	50969	71415	137065
13 [ENE]	688	2511	11482	50969	71415	137065
14	688	2511	11482	50969	71415	137065
15	888	4753	18337	74280	75274	173532
16	1091	6996	25195	97590	79134	210006
17 [E]	1091	6996	25195	97590	79134	210006
18	1091	6996	25195	97590	79134	210006
19	1080	6392	26086	67491	42683	143732
20	1073	5788	26976	37390	6231	77458
21 [ESE]	1073	5788	26976	37390	6231	77458
22	1073	5788	26976	37390	6231	77458
23	987	5395	15149	21207	7424	50162
24	906	5000	3324	5024	8615	22869
25 [SE]	906	5000	3324	5024	8615	22869
26	906	5000	3324	5024	8615	22869
27	894	8740	2095	4627	6075	22431
28	882	12481	866	4232	3537	21998
29 [SSE]	882	12481	866	4232	3537	21998
30	882	12481	866	4232	3537	21998
31	967	15522	8931	2759	3014	31193
32	1052	18564	16993	1286	2491	40386
33 [S]	1052	18564	16993	1286	2491	40386
34	1052	18564	16993	1286	2491	40386
35	1050	18970	32221	118637	58215	229093
36	1050	19376	47453	235987	113938	417804
37 [SSW]	1050	19376	47453	235987	113938	417804
38	1050	19376	47453	235987	113938	417804
39	1199	12627	36942	193490	103075	347333
40	1349	5876	26430	150992	92214	276861
41 [SW]	1349	5876	26430	150992	92214	276861
42	1349	5876	26430	150992	92214	276861
43	1133	4078	15632	82819	58118	161780
44	917	2283	4835	14646	24023	46704
45 [WSW]	917	2283	4835	14646	24023	46704
46	917	2283	4835	14646	24023	46704
47	1220	3065	6466	17642	19820	48213
48	1531	3846	8097	20640	15618	49732
49 [W]	1531	3846	8097	20640	15618	49732

Table B-3: Peach Bottom 2060 Population Distribution (Continued)

Sector	0-10 mi	10-20 mi	20-30 mi	30-40 mi	40-50 mi	50 mi total
50	1531	3846	8097	20640	15618	49732
51	1043	4160	25816	21827	13291	66137
52	558	4477	43533	23013	10965	82546
53 [WNW]	558	4477	43533	23013	10965	82546
54	558	4477	43533	23013	10965	82546
55	469	3896	30426	21727	52195	108713
56	380	3315	17318	20440	93423	134876
57 [NW]	380	3315	17318	20440	93423	134876
58	380	3315	17318	20440	93423	134876
59	667	6812	22256	18389	62826	110950
60	955	10310	27191	16338	32228	87022
61 [NNW]	955	10310	27191	16338	32228	87022
62	955	10310	27191	16338	32228	87022
63	910	21902	33096	13614	26363	95885
64	867	33495	39001	10891	20497	104751
Total	**61067**	**581105**	**1293727**	**2889208**	**2773916**	**7599023**

B.1.6.2 Peach Bottom Economic Information

NEI 05-01 prescribes that economic data be expressed in current dollars. Extrapolation of the economic data from 2005 to 2016 satisfies this prescription. MACCS economic parameter values for the Peach Bottom site are shown in Table B-4. The depreciation rate and the investment rate of return are unchanged.

Table B-4: Peach Bottom Economic Parameters

Variable	Description	Value
DPRATE	Property depreciation rate (per year)	0.2
DSRATE	Investment rate of return (per year)	0.12
EVACST	Daily cost for a person who has been evacuated ($/person-day)	217
POPCST	Population relocation cost ($/person)	15120
RELCST	Daily cost for a person who is relocated ($/person-day)	217
CDFRM	Cost of farm decontamination for various levels of decontamination ($/hectare)	1) 1675 2) 3730
CDNFRM	Cost of non-farm decontamination per resident for various levels of decontamination ($/person)	1) 8959 2) 23940
DLBCST	Average cost of decontamination labor ($/person-year)	105840
VALWF	Value of farm wealth ($/hectare)	11390
VALNWF	Value of non-farm wealth ($/person)	264600

B.1.6.3 Peach Bottom Weather Information

Meteorological data used in this evaluation is the same as that used in the Peach Bottom SOARCA analysis (Reference 8.1-17). The licensee of the Peach Bottom power plant provided two years of meteorological information to Sandia National Laboratories and the NRC. Different trends in the meteorology over the two years were estimated to have a relatively minor effect on end results (Reference 8.1-11, Section 4.3.5). The data recovery rate for the Peach Bottom 2006 meteorological data was approximately 99 percent, meaning that wind speed, wind

direction, atmospheric stability, and precipitation were measured for a given hour for approximately 99 percent of the hours of the year.

B.1.7 Emergency Sheltering and Evacuation Modeling Information

The Surry and Peach Bottom sites are modeled similar to the SOARCA studies. Emergency sheltering and evacuation are not modeled in the NuScale analysis, but relocation is modeled in the same manner as was done in the SOARCA studies. Early-phase and long-term relocation are modeled as part of the SAMDA economic model as relocation, interdiction, and condemnation costs are considered in the calculation of the overall cost of the radionuclide release. The economic parameters are shown in Table B-2 and Table B-4.

There is a subtle difference between the terms "evacuation" and "relocation" in the context of MACCS simulations. Evacuation refers to the ordered evacuation of the population from within the emergency planning zone during the emergency phase immediately following accident initiation with the goal of reducing plume exposure dose by removing the population from the pathway of the released plume. The emergency phase is modeled as having a duration of one week in the NuScale analysis. Relocation occurs in both the emergency phase and the long-term action phase, modeled as having a five-year duration. The long-term action phase begins immediately after the emergency phase and dose projections are used to determine if certain segments of the population should be temporarily relocated (or, if they were already relocated during the early phase then they continue to remain relocated), to avoid exceeding a habitability dose threshold. If a population segment was relocated in the early phase, then the long term dose projections are used to determine if that population segment should continue to be relocated or if they are allowed to return at the start of the long-term phase. The emergency phase habitability dose threshold in the NuScale analysis is the Environmental Protection Agency protective action guides of one- to five-rem whole body dose. The long-term habitability dose threshold in this analysis is four rem over the five-year action period. The early- and long-term habitability decision making and the relocation processes modeled by MACCS are discussed in Section 6.0 and Section 7.0, respectively, of the Code Manual for MACCS2 (Reference 8.1-7).

B.1.8 Core Inventory at Accident Initiation

Best estimate core radionuclide inventories are used. These best estimate radionuclide inventories are calculated at the end of the irradiation of the equilibrium fuel cycle for the NuScale reactor core, using SCALE 6.1.3.

Table B-5: Core Inventory

NuScale Best Estimate Core Inventory (Bq)					
Kr-85	2.52E+15	**I-135**	3.13E+17	**Ce-144**	2.18E+17
Kr-85m	4.54E+16	**Te-127**	1.40E+16	**Np-239**	2.95E+18
Kr-87	9.04E+16	**Te-127m**	2.22E+15	**Pu-238**	4.99E+14
Kr-88	1.19E+17	**Te-129**	4.07E+16	**Pu-239**	9.76E+13
Xe-133	3.31E+17	**Te-129m**	7.01E+15	**Pu-240**	1.08E+14
Xe-135	1.41E+17	**Te-131m**	2.89E+16	**Pu-241**	2.75E+16
Xe-135m	6.93E+16	**Te-132**	2.27E+17	**Zr-95**	2.79E+17
Cs-134	2.40E+16	**Te-131**	1.37E+17	**Zr-97**	2.76E+17
Cs-136	7.77E+15	**Rh-105**	1.47E+17	**Am-241**	4.52E+13

Table B-5: Core Inventory (Continued)

NuScale Best Estimate Core Inventory (Bq)					
Cs-137	2.52E+16	**Ru-103**	2.35E+17	**Cm-242**	8.29E+15
Rb-86	2.07E+14	**Ru-105**	1.53E+17	**Cm-244**	3.14E+14
Rb-88	1.21E+17	**Ru-106**	8.05E+16	**La-140**	2.92E+17
Ba-139	2.95E+17	**Rh-103m**	2.33E+17	**La-141**	2.69E+17
Ba-140	2.85E+17	**Rh-106**	8.60E+16	**La-142**	2.60E+17
Sr-89	1.66E+17	**Nb-95**	2.80E+17	**Nd-147**	2.86E+15
Sr-90	1.94E+16	**Co-58**	5.07E+12	**Pr-143**	1.07E+17
Sr-91	2.08E+17	**Co-60**	2.33E+13	**Y-90**	1.98E+16
Sr-92	2.22E+17	**Mo-99**	3.00E+17	**Y-91**	2.12E+17
Ba-137m	2.39E+16	**Tc-99m**	2.64E+17	**Y-92**	2.24E+17
I-131	1.58E+17	**Nb-97**	2.77E+17	**Y-93**	2.50E+17
I-132	2.31E+17	**Nb-97m**	2.62E+17	**Y-91m**	1.22E+17
I-133	3.30E+17	**Ce-141**	2.70E+17	**Pr-144**	2.19E+17
I-134	3.72E+17	**Ce-143**	2.55E+17	**Pr-144m**	2.58E+15

B.1.9 Chemical Groups

The following isotopes are associated with the corresponding chemical groups as shown.

"Xe" group: Kr-85, Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135, & Xe-135m.

"Cs" group: Cs-134, Cs-136, Cs-137, Rb-86, & Rb-88.

"Ba" group: Ba-139, Ba-140, Sr-89, Sr-90, Sr-91, Sr-92, & Ba-137m.

"I" group: I-131, I-132, I-133, I-134, & I-135.

"Te" group: Te-127, Te-127m, Te-129, Te-129m, Te-131m, Te-132, & Te-131.

"Ru" group: Rh-105, Ru-103, Ru-105, Ru-106, Rh-103m, & Rh-106.

"Mo" group: Nb-95, Co-58, Co-60, Mo-99, Tc-99m, Nb-97, & Nb-97m.

"Ce" group: Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu-240, Pu-241, Zr-95, & Zr-97.

"La" group: Am-241, Cm-242, Cm-244, La-140, La-141, La-142, Nd-147, Pr-143, Y-90, Y-91, Y-92, Y-93, Y-91m, Pr-144, & Pr-144m.

B.2 Release Categories

There are various different combinations of events that could lead to radiological releases from the NuScale plant (referred to as cutsets). Each of these cutsets may lead to different offsite radiological consequences, which will be referred to as public dose in this report, and each of these cutsets may occur at a different frequency than the other cutsets.

To simplify the analysis and reduce the time and number of simulations required to calculate the maximum benefit, the NuScale severe-accident cutsets are binned into eight release

categories based on initiating event and mitigating systems available. In this manner, scenarios with similar accident progressions are grouped together. Containment bypass sequences and intact containment sequences are not grouped together, even if the initiating event and mitigating systems available are similar, because the magnitude of the release is larger in containment bypass scenarios when compared to intact containment scenarios. Sequences that do not reach core damage are not included in a release category as core damage must occur to allow a significant release to the environment. A core damage event due to reactor building crane failure, where the module fails to remain upright and is completely submerged in the reactor pool, is a unique event and is considered separately as release category 8 (Section B.2.8).

NEI 05-01 recommends that sequences are grouped into release categories based on timing of release and radionuclide release fractions to the environment. When evacuation is considered, offsite dose to the public is sensitive to the timing of emergency-phase evacuation and radionuclide release. The NuScale SAMDA analysis is not considering emergency-phase evacuation of the public (as discussed in Section B.1.7) and the sensitivity to release timing due to core inventory decay is relatively minor. Therefore, accident timing was not considered in the grouping of sequences into release categories and solely the release fraction to the environment was considered.

Many sequences have different initiating events, coming from different event trees, but have similar accident progressions due to mitigating systems available and are binned together. For example, sequences where coolant is lost to containment due to an emergency core cooling system RVV spurious actuation in which no coolant makeup is available are expected to progress similarly, independent of how the ECCS is actuated (spurious ECCS, loss of power to ECCS control solenoids, etc.) and are grouped together in release category 4.

To calculate the release category frequencies, the following NuScale PRAs are used as inputs. These PRAs are discussed in more detail in Chapter 19 of the NuScale Final Safety Analysis Report. Each PRA calculates a CDF for each end state. The frequencies associated with each end state binned into a specific release category are summed together to establish the frequency associated with that specific release category, as shown in Table B-17.

- internal events PRA
- low-power and shutdown PRA
- internal flooding PRA
- internal fires PRA
- external floods PRA
- high winds PRA

The offsite consequence associated with each release category (shown in Table B-19) is taken to be the offsite consequence of the bounding (from a radionuclide release standpoint) constituent accident sequence out of all the accident sequences in the release category. The bounding scenario is selected through engineering judgment and is described in Section B.2.1 to Section B.2.8. In general, the bounding accident sequence for each release category is considered to be the sequence in which the fewest number of mitigating systems are available with the exception of ECCS failures. ECCS failures are handled separately for intact CNV sequences and CNV bypass sequences. Intact CNV sequences assume that the failure of ECCS is

a partial failure, with only the RVVs actuating successfully, which removes reactor coolant from the RPV to the CNV and does not allow core reflood and the possible termination of release from fuel. CNV bypass sequences assume complete ECCS failure, which removes the possibility of radionuclides depositing in the CNV, allowing the largest amount of radionuclides to bypass containment.

The eight release categories defined in Section B.2.1 to Section B.2.8 could be split into more release categories to more realistically represent the release magnitudes and frequencies of the NuScale design. For example, release category 3 (CVCS LOCA unisolated outside the CNV, Section B.2.3) could be further split into two release categories: CVCS LOCA unisolated outside CNV with successful ECCS actuation and CVCS LOCA unisolated outside CNV with unsuccessful ECCS. Successful ECCS would delay the onset of core damage and reduce the radionuclide release to the environment compared to unsuccessful ECCS. The offsite consequences for release category 3 are estimated for a CVCS LOCA unisolated outside containment with unsuccessful ECCS and, therefore, the currently defined release category 3 overestimates the offsite consequences in the interest of simplifying the analysis. None of the release categories take credit for the reactor building with respect to fission product retention.

B.2.1 Release Category 1: CVCS LOCA Inside Containment

The first release category consists of intact containment LOCAs involving the CVCS. ECCS and RSV spurious actuations, which challenge nearly the same systems and equipment as the CVCS intact containment LOCAs, are handled separately in release category 4. Release category 1 is an intact containment scenario.

The bounding (from a radionuclide release standpoint) scenario is judged to be a high-break CVCS injection line LOCA inside containment. ECCS failure is a partial actuation with the RRVs failing to open and CVCS injection through the pressurizer spray line fails to operate, which produces a scenario where reactor coolant is lost to containment through the CVCS line break and the RVVs and coolant is not available either through RRV recirculation or through CVCS pressurizer spray injection. The core is uncovered and unable to be cooled, leading to total core support failure and core relocation to the RPV lower plenum. The plume release information for release category 1 that was used as input for MACCS calculations is shown in Table B-6. The particle size distribution for release category 1 was assumed to be the same as the particle size distribution for release category 5 (see Table B-11).

Table B-6: Release Category 1 Hourly Release Information

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)								
				Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
1	2.00E+04	3750	0	2.42E-05	5.76E-06	4.51E-08	4.35E-06	4.40E-06	1.34E-08	1.23E-06	4.11E-13	4.11E-13
2	2.38E+04	3500	0	4.91E-05	5.82E-07	3.14E-09	5.35E-07	6.63E-07	1.16E-08	1.26E-07	2.26E-13	2.26E-13
3	2.73E+04	3500	0	5.68E-05	2.45E-07	4.50E-09	3.75E-07	3.96E-07	1.17E-08	4.78E-08	3.22E-13	3.24E-13
4	3.08E+04	3750	0	6.75E-05	1.99E-07	2.40E-08	4.47E-07	1.95E-06	9.36E-10	3.49E-08	4.50E-13	5.28E-13
5	3.45E+04	3500	0	6.19E-05	1.21E-07	1.05E-08	2.88E-07	7.01E-07	4.42E-10	2.18E-08	2.89E-13	3.15E-13
6	3.80E+04	3250	0	6.47E-05	3.82E-07	2.73E-09	2.83E-07	3.20E-07	3.48E-09	8.33E-08	2.13E-13	2.14E-13
7	4.13E+04	3250	0	6.81E-05	7.20E-08	2.86E-09	4.65E-07	5.94E-07	5.68E-10	8.06E-09	1.92E-14	1.91E-14
8	4.45E+04	4000	0	7.94E-05	4.83E-08	2.31E-09	2.81E-07	4.63E-07	3.24E-10	5.60E-09	1.15E-14	1.13E-14
9	4.85E+04	3750	0	7.21E-05	2.20E-08	1.01E-09	1.23E-07	2.03E-07	1.36E-10	2.60E-09	4.86E-15	4.79E-15
10	5.23E+04	3500	0	6.62E-05	1.03E-08	4.50E-10	5.56E-08	9.01E-08	5.98E-11	1.23E-09	2.13E-15	2.10E-15
11	5.58E+04	3500	0	6.54E-05	5.18E-09	2.15E-10	2.78E-08	4.30E-08	2.84E-11	6.19E-10	1.01E-15	1.00E-15
12	5.93E+04	3750	0	6.82E-05	2.72E-09	1.07E-10	1.51E-08	2.14E-08	1.41E-11	3.19E-10	5.04E-16	4.96E-16
13	6.30E+04	3500	0	6.28E-05	1.38E-09	4.97E-11	8.15E-09	9.98E-09	6.58E-12	1.53E-10	2.35E-16	2.32E-16
14	6.65E+04	3750	0	6.45E-05	6.27E-09	2.51E-11	1.87E-08	4.71E-09	3.31E-12	6.52E-10	1.14E-16	1.12E-16
15	7.03E+04	3500	0	5.95E-05	1.18E-08	2.04E-11	3.35E-08	3.36E-09	2.47E-12	1.25E-09	7.87E-17	7.73E-17
16	7.38E+04	3500	0	5.94E-05	7.25E-09	1.24E-11	2.12E-08	2.03E-09	1.46E-12	7.60E-10	4.61E-17	4.52E-17
17	7.73E+04	3750	0	6.37E-05	4.35E-09	8.81E-12	1.37E-08	1.54E-09	8.85E-13	4.55E-10	2.77E-17	2.72E-17
18	8.10E+04	3500	0	5.94E-05	2.45E-09	1.04E-11	8.57E-09	2.05E-09	5.24E-13	2.53E-10	1.57E-17	1.54E-17
19	8.45E+04	3750	0	6.34E-05	1.69E-09	1.43E-11	6.68E-09	1.60E-09	3.75E-13	1.69E-10	1.04E-17	1.02E-17
20	8.83E+04	3500	0	5.91E-05	1.01E-09	9.63E-12	4.73E-09	1.02E-09	2.32E-13	9.80E-11	6.09E-18	6.00E-18
21	9.18E+04	3500	0	5.91E-05	6.53E-10	6.48E-12	3.78E-09	6.67E-10	1.48E-13	6.09E-11	3.76E-18	3.70E-18
22	9.53E+04	3750	0	6.32E-05	4.28E-10	4.31E-12	3.34E-09	4.37E-10	9.49E-14	3.85E-11	2.35E-18	2.32E-18
23	9.90E+04	3500	0	5.89E-05	2.64E-10	2.70E-12	2.75E-09	2.71E-10	5.76E-14	2.24E-11	1.38E-18	1.36E-18
24	1.03E+05	3750	0	6.31E-05	1.76E-10	1.82E-12	2.68E-09	1.80E-10	3.81E-14	1.42E-11	8.92E-19	8.80E-19
25	1.06E+05	3500	0	5.88E-05	1.26E-10	1.34E-12	2.39E-09	1.27E-10	2.76E-14	8.89E-12	6.08E-19	6.01E-19
26	1.10E+05	3500	0	5.88E-05	1.07E-10	1.15E-12	2.33E-09	1.03E-10	2.38E-14	6.27E-12	4.93E-19	4.88E-19
27	1.13E+05	3750	0	6.29E-05	9.00E-11	9.72E-13	2.43E-09	8.15E-11	2.06E-14	4.47E-12	4.06E-19	4.02E-19
28	1.17E+05	3500	0	5.87E-05	7.48E-11	7.92E-13	2.24E-09	6.26E-11	1.78E-14	3.04E-12	3.36E-19	3.33E-19
29	1.21E+05	3750	0	6.28E-05	6.21E-11	6.37E-13	2.36E-09	4.92E-11	1.55E-14	2.17E-12	2.85E-19	2.82E-19
30	1.24E+05	3500	0	5.86E-05	5.86E-11	5.38E-13	2.19E-09	4.13E-11	1.53E-14	1.60E-12	2.72E-19	2.70E-19
31	1.28E+05	3500	0	5.86E-05	5.44E-11	4.52E-13	2.18E-09	3.53E-11	1.51E-14	1.22E-12	2.63E-19	2.61E-19

Table B-6: Release Category 1 Hourly Release Information (Continued)

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)								
				Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
32	1.31E+05	3750	0	6.27E-05	4.92E-11	3.69E-13	2.31E-09	2.95E-11	1.42E-14	9.29E-13	2.45E-19	2.43E-19
33	1.35E+05	3500	0	5.85E-05	4.60E-11	2.98E-13	2.15E-09	2.49E-11	1.42E-14	7.00E-13	2.41E-19	2.39E-19
34	1.39E+05	3750	0	6.26E-05	4.24E-11	2.44E-13	2.30E-09	2.13E-11	1.38E-14	5.43E-13	2.33E-19	2.31E-19
35	1.42E+05	3500	0	5.84E-05	3.49E-11	1.78E-13	2.14E-09	1.63E-11	1.20E-14	3.80E-13	2.00E-19	1.99E-19
36	1.46E+05	3500	0	5.84E-05	3.17E-11	1.41E-13	2.13E-09	1.38E-11	1.14E-14	2.91E-13	1.90E-19	1.89E-19
37	1.49E+05	3750	0	6.25E-05	2.88E-11	1.12E-13	2.28E-09	1.17E-11	1.09E-14	2.25E-13	1.80E-19	1.79E-19
38	1.53E+05	3500	0	5.83E-05	2.32E-11	7.92E-14	2.12E-09	8.82E-12	9.07E-15	1.56E-13	1.50E-19	1.49E-19
39	1.57E+05	3750	0	6.24E-05	2.20E-11	6.38E-14	2.27E-09	7.70E-12	8.84E-15	1.24E-13	1.45E-19	1.44E-19
40	1.60E+05	3500	0	5.82E-05	1.84E-11	4.61E-14	2.12E-09	6.06E-12	7.69E-15	8.99E-14	1.26E-19	1.25E-19
41	1.64E+05	3500	0	5.82E-05	1.56E-11	3.44E-14	2.12E-09	4.89E-12	6.73E-15	6.76E-14	1.10E-19	1.09E-19
42	1.67E+05	3750	0	6.23E-05	1.35E-11	2.64E-14	2.27E-09	4.05E-12	6.02E-15	5.24E-14	9.82E-20	9.76E-20
43	1.71E+05	3500	0	5.81E-05	1.15E-11	1.90E-14	2.11E-09	3.23E-12	5.22E-15	3.91E-14	8.49E-20	8.42E-20
44	1.75E+05	3750	0	6.23E-05	1.12E-11	1.55E-14	2.26E-09	2.96E-12	5.23E-15	3.38E-14	8.50E-20	8.39E-20
45	1.78E+05	3500	0	5.81E-05	8.90E-12	1.08E-14	2.11E-09	2.26E-12	4.26E-15	2.48E-14	6.91E-20	6.85E-20
46	1.82E+05	3500	0	5.81E-05	7.39E-12	7.89E-15	2.11E-09	1.81E-12	3.61E-15	1.91E-14	5.84E-20	5.80E-20
47	1.85E+05	3750	0	6.22E-05	7.02E-12	6.29E-15	2.26E-09	1.63E-12	3.46E-15	1.66E-14	5.59E-20	5.57E-20
48	1.89E+05	3500	0	5.80E-05	5.85E-12	4.46E-15	2.11E-09	1.29E-12	2.90E-15	1.29E-14	4.69E-20	4.67E-20
49	1.93E+05	3750	0	6.21E-05	5.49E-12	3.53E-15	2.26E-09	1.15E-12	2.75E-15	1.13E-14	4.42E-20	4.41E-20
50	1.96E+05	3500	0	5.80E-05	4.79E-12	2.57E-15	2.11E-09	9.56E-13	2.41E-15	9.35E-15	3.87E-20	3.86E-20
51	2.00E+05	3500	0	5.79E-05	4.05E-12	1.90E-15	2.11E-09	7.81E-13	2.04E-15	7.60E-15	3.29E-20	3.28E-20
52	2.03E+05	3750	0	6.20E-05	3.94E-12	1.54E-15	2.26E-09	7.23E-13	1.98E-15	7.07E-15	3.19E-20	3.17E-20
53	2.07E+05	3500	0	5.79E-05	3.18E-12	1.08E-15	2.11E-09	5.62E-13	1.60E-15	5.51E-15	2.57E-20	2.48E-20
54	2.11E+05	3750	0	6.20E-05	2.99E-12	8.64E-16	2.26E-09	5.04E-13	1.49E-15	4.98E-15	2.40E-20	2.27E-20
55	2.14E+05	3500	0	5.78E-05	2.65E-12	6.46E-16	2.11E-09	4.26E-13	1.31E-15	4.28E-15	2.09E-20	1.99E-20
56	2.18E+05	3500	0	5.78E-05	2.41E-12	5.10E-16	2.11E-09	3.74E-13	1.19E-15	3.83E-15	1.87E-20	1.80E-20
57	2.21E+05	3750	0	6.19E-05	2.59E-12	4.58E-16	2.26E-09	3.83E-13	1.26E-15	4.04E-15	1.97E-20	1.93E-20
58	2.25E+05	3500	0	5.78E-05	2.13E-12	3.39E-16	2.11E-09	3.08E-13	1.05E-15	3.33E-15	1.63E-20	1.60E-20
59	2.29E+05	3750	0	6.19E-05	2.06E-12	2.92E-16	2.26E-09	2.90E-13	1.02E-15	3.21E-15	1.58E-20	1.57E-20
60	2.32E+05	3500	0	5.77E-05	2.25E-12	4.02E-14	2.11E-09	4.33E-12	2.83E-15	2.12E-14	4.57E-20	4.56E-20
61	2.36E+05	3500	0	5.77E-05	2.44E-12	5.17E-14	2.11E-09	5.54E-12	3.76E-15	2.61E-14	6.09E-20	6.07E-20
62	2.39E+05	3750	0	6.18E-05	2.36E-12	4.89E-14	2.26E-09	5.28E-12	3.75E-15	2.45E-14	6.07E-20	6.04E-20
63	2.43E+05	3500	0	5.76E-05	2.54E-12	4.60E-14	2.11E-09	5.03E-12	3.84E-15	2.31E-14	6.22E-20	6.15E-20

Table B-6: Release Category 1 Hourly Release Information (Continued)

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)								
				Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
64	2.47E+05	3750	0	6.17E-05	2.63E-12	4.27E-14	2.26E-09	4.76E-12	3.99E-15	2.19E-14	6.46E-20	6.39E-20
65	2.50E+05	3500	0	5.75E-05	2.28E-12	3.22E-14	2.11E-09	3.67E-12	3.42E-15	1.71E-14	5.54E-20	5.49E-20
66	2.54E+05	3500	0	5.75E-05	2.71E-12	3.01E-14	2.11E-09	3.61E-12	4.14E-15	1.79E-14	6.72E-20	6.66E-20
67	2.57E+05	1950	0	3.20E-05	1.89E-12	1.70E-14	1.18E-09	2.17E-12	3.02E-15	1.20E-14	4.90E-20	4.87E-20
68-98	2.59E+05	3600	0	5.91E-05	3.50E-12	3.15E-14	2.17E-09	4.01E-12	5.58E-15	2.21E-14	9.05E-20	8.99E-20
Total	2.00E+04	350800	0	5.85E-03	7.49E-06	9.71E-08	7.53E-06	9.87E-06	4.27E-08	1.57E-06	1.95E-12	2.06E-12

B.2.2 Release Category 2: LOCA Outside Containment, Isolated

The second release category consists of LOCAs that occur either through the CVCS charging line, CVCS letdown line, or steam generator tubes and are subsequently isolated with the failure of the RSVs to open. Scenarios where the RSVs are demanded and an RSV sticks open are included in release category 6 in Section B.2.6. Release category 2 is an intact containment scenario.

The bounding (from a radionuclide release standpoint) scenario is judged to be a CVCS injection line LOCA outside containment that is isolated, with a complete failure of DHRS and the RSVs, which creates an RPV overpressure scenario where the RPV fails at the pressurizer heater inspection port and the RVVs actuate on low RPV level after sufficient leakage to containment. The RRVs fail to open upon low RPV level actuation and the main valve spring fails to force the ECCS valve open at low differential pressure. Due to partial ECCS actuation, reactor coolant completely relocates to the CNV and is unable to recirculate to the RPV. The core is uncovered and unable to be cooled, leading to total core support failure and core relocation to the RPV lower plenum. The plume release information for release category 2 that was used as input for MACCS calculations is shown in Table B-7. The particle size distribution for release category 2 was assumed to be the same as the particle size distribution for release category 5 (see Table B-11).

Table B-7: Release Category 2 Hourly Release Information

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)								
				Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
1	35750	3750	0	7.89E-07	6.38E-07	1.29E-07	6.34E-07	6.48E-07	1.70E-11	1.39E-09	2.72E-16	2.72E-16
2	39000	3500	0	1.05E-05	1.56E-06	5.65E-09	1.12E-06	1.14E-06	1.86E-09	3.41E-07	9.54E-14	9.54E-14
3	43000	3750	0	5.41E-05	4.32E-06	2.15E-08	3.60E-06	3.75E-06	7.06E-08	9.40E-07	1.42E-12	1.42E-12
4	46750	3500	0	6.20E-05	3.47E-08	4.68E-10	2.47E-07	6.59E-08	5.54E-10	3.64E-09	3.66E-14	4.01E-14
5	50250	3750	0	6.75E-05	1.81E-08	1.72E-09	8.41E-09	6.10E-08	1.67E-10	4.02E-09	3.50E-14	3.08E-14
6	54000	3500	0	6.33E-05	1.17E-08	1.01E-10	7.52E-09	5.71E-09	3.81E-11	2.61E-09	1.01E-14	1.07E-14
7	57500	3500	0	6.30E-05	4.72E-09	3.47E-11	5.46E-09	2.46E-09	1.05E-11	1.02E-09	4.79E-15	5.29E-15
8	61000	3750	0	6.53E-05	9.98E-10	6.37E-12	3.85E-09	4.45E-10	1.96E-12	2.02E-10	8.86E-16	9.78E-16
9	64750	3500	0	7.01E-05	2.98E-08	2.89E-11	5.27E-08	4.13E-09	1.71E-11	5.28E-09	7.78E-15	8.67E-15
10	68250	3750	0	7.49E-05	1.77E-08	4.87E-11	3.31E-08	9.13E-09	3.06E-12	3.49E-09	1.31E-14	1.51E-14
11	72000	3500	0	6.52E-05	1.08E-08	4.46E-11	8.09E-08	8.40E-09	5.23E-13	8.90E-10	1.21E-15	1.25E-15
12	75500	3500	0	6.32E-05	1.55E-08	3.15E-11	8.17E-08	1.08E-08	4.85E-12	1.60E-09	1.33E-15	1.35E-15
13	79000	3750	0	6.71E-05	9.53E-09	1.36E-11	3.51E-08	4.52E-09	9.56E-12	1.32E-09	6.70E-16	6.78E-16
14	82750	3500	0	6.19E-05	1.71E-09	1.38E-12	6.14E-09	4.31E-10	2.86E-12	3.13E-10	8.57E-17	8.63E-17
15	86250	3750	0	6.61E-05	7.82E-10	3.22E-13	4.27E-09	1.29E-10	1.16E-12	1.60E-10	2.46E-17	2.47E-17
16	90000	3500	0	6.15E-05	3.25E-10	9.30E-14	3.64E-09	6.45E-11	4.38E-13	6.74E-11	7.98E-18	8.00E-18
17	93500	3500	0	6.03E-05	1.48E-10	3.78E-14	3.50E-09	4.01E-11	1.98E-13	2.97E-11	3.40E-18	3.41E-18
18	97000	3750	0	6.35E-05	6.99E-11	2.36E-14	3.67E-09	2.44E-11	1.34E-13	1.29E-11	2.22E-18	2.22E-18
19	100750	3500	0	5.86E-05	5.03E-09	4.23E-12	9.42E-09	2.70E-10	2.65E-12	9.94E-10	1.23E-16	1.24E-16
20	104250	3750	0	6.25E-05	3.20E-09	2.64E-12	7.44E-09	1.68E-10	1.66E-12	6.32E-10	7.72E-17	7.77E-17
21	108000	3500	0	5.82E-05	1.77E-09	1.43E-12	5.47E-09	9.14E-11	9.03E-13	3.48E-10	4.19E-17	4.22E-17
22	111500	3500	0	5.81E-05	6.27E-10	4.99E-13	4.09E-09	3.19E-11	3.16E-13	1.22E-10	1.46E-17	1.47E-17
23	115000	3750	0	6.21E-05	5.52E-11	4.19E-14	3.63E-09	2.70E-12	2.67E-14	1.04E-11	1.23E-18	1.24E-18
24	118750	3500	0	5.79E-05	3.73E-11	2.72E-14	3.37E-09	1.76E-12	1.74E-14	6.74E-12	8.02E-19	8.08E-19
25	122250	3750	0	6.19E-05	3.09E-11	2.16E-14	3.59E-09	1.41E-12	1.39E-14	5.32E-12	6.38E-19	6.42E-19
26	126000	3500	0	5.77E-05	1.90E-11	1.26E-14	3.33E-09	8.34E-13	8.20E-15	3.06E-12	3.74E-19	3.77E-19
27	129500	3500	0	5.76E-05	1.58E-11	9.91E-15	3.33E-09	6.61E-13	6.48E-15	2.35E-12	2.94E-19	2.96E-19
28	133000	3750	0	6.17E-05	1.37E-11	7.92E-15	3.55E-09	5.36E-13	5.24E-15	1.82E-12	2.36E-19	2.37E-19
29	136750	3500	0	5.75E-05	9.38E-12	4.86E-15	3.31E-09	3.36E-13	3.28E-15	1.07E-12	1.46E-19	1.46E-19
30	140250	3750	0	6.15E-05	7.17E-12	3.38E-15	3.54E-09	2.38E-13	2.32E-15	6.99E-13	1.02E-19	1.02E-19
31	144250	3500	0	5.73E-05	4.59E-12	1.96E-15	3.29E-09	1.41E-13	1.38E-15	3.75E-13	5.95E-20	5.97E-20
32	147500	3500	0	5.72E-05	3.52E-12	1.36E-15	3.29E-09	1.00E-13	9.82E-16	2.41E-13	4.15E-20	4.17E-20
33	151000	3750	0	6.13E-05	2.88E-12	9.90E-16	3.52E-09	7.53E-14	7.40E-16	1.59E-13	3.05E-20	3.07E-20

Table B-7: Release Category 2 Hourly Release Information (Continued)

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)								
				Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
34	154750	3500	0	5.71E-05	1.76E-12	5.90E-16	3.28E-09	4.65E-14	4.60E-16	8.48E-14	1.84E-20	1.85E-20
35	158250	3750	0	6.12E-05	1.56E-12	4.48E-16	3.51E-09	3.66E-14	3.65E-16	5.74E-14	1.41E-20	1.42E-20
36	162000	3500	0	5.70E-05	1.18E-12	2.80E-16	3.27E-09	2.39E-14	2.42E-16	3.09E-14	9.00E-21	9.05E-21
37	165500	3500	0	5.70E-05	9.76E-13	1.94E-16	3.27E-09	1.74E-14	1.79E-16	1.83E-14	6.37E-21	6.40E-21
38	169000	3750	0	6.10E-05	8.15E-13	1.46E-16	3.50E-09	1.37E-14	1.43E-16	1.16E-14	4.87E-21	4.77E-21
39	172750	3500	0	5.69E-05	5.51E-13	8.63E-17	3.27E-09	8.51E-15	9.06E-17	5.61E-15	2.95E-21	2.85E-21
40	176250	3750	0	6.09E-05	4.83E-13	6.63E-17	3.50E-09	6.87E-15	7.46E-17	3.57E-15	2.26E-21	2.16E-21
41	180000	3500	0	5.68E-05	3.33E-13	4.16E-17	3.26E-09	4.52E-15	5.01E-17	1.82E-15	1.42E-21	1.34E-21
42	183500	3500	0	5.68E-05	2.42E-13	2.84E-17	3.26E-09	3.22E-15	3.63E-17	1.02E-15	9.74E-22	9.41E-22
43	187000	3750	0	6.08E-05	1.85E-13	2.04E-17	3.49E-09	2.41E-15	2.76E-17	6.04E-16	7.27E-22	6.84E-22
44	190750	3500	0	5.67E-05	2.53E-13	2.54E-17	3.26E-09	3.12E-15	3.64E-17	6.41E-16	6.80E-22	5.48E-22
45	194250	3750	0	6.08E-05	4.05E-13	3.84E-17	3.49E-09	4.85E-15	5.73E-17	8.63E-16	1.01E-21	8.05E-22
46	198000	3500	0	5.67E-05	4.25E-13	3.68E-17	3.25E-09	4.81E-15	5.78E-17	7.33E-16	9.64E-22	6.68E-22
47	201500	3500	0	5.67E-05	4.37E-13	3.40E-17	3.25E-09	4.63E-15	5.67E-17	6.05E-16	9.24E-22	4.93E-22
48	205000	3750	0	6.07E-05	4.57E-13	3.19E-17	3.48E-09	4.54E-15	5.68E-17	5.10E-16	8.71E-22	1.07E-22
49	208750	3500	0	5.66E-05	3.73E-13	2.51E-17	3.25E-09	3.77E-15	4.84E-17	3.61E-16	6.48E-22	3.51E-23
50	212250	3750	0	6.06E-05	3.96E-13	2.54E-17	3.48E-09	4.13E-15	5.52E-17	3.44E-16	8.17E-22	1.24E-23
51	216000	3500	0	5.65E-05	3.82E-13	2.17E-17	3.25E-09	3.94E-15	5.54E-17	3.09E-16	8.41E-22	3.52E-24
52	219500	3500	0	5.65E-05	3.98E-13	1.96E-17	3.24E-09	3.95E-15	5.81E-17	3.08E-16	9.26E-22	2.07E-24
53	223000	3750	0	6.05E-05	4.29E-13	1.83E-17	3.47E-09	4.05E-15	6.22E-17	3.26E-16	9.91E-22	9.67E-25
54	226750	3500	0	5.64E-05	3.76E-13	1.40E-17	3.24E-09	3.36E-15	5.33E-17	2.83E-16	8.00E-22	6.42E-25
55	230250	3750	0	6.04E-05	3.44E-13	3.07E-17	3.47E-09	7.08E-15	2.32E-17	1.37E-14	1.21E-21	1.20E-21
56	234000	3500	0	5.64E-05	3.21E-13	1.42E-16	3.24E-09	3.28E-14	9.75E-17	5.38E-14	5.65E-21	5.73E-21
57	237500	3500	0	5.63E-05	3.53E-13	2.15E-16	3.23E-09	4.95E-14	1.47E-16	5.97E-14	8.53E-21	8.65E-21
58	241000	3750	0	6.03E-05	2.97E-13	1.52E-16	3.46E-09	3.50E-14	1.04E-16	5.00E-14	6.03E-21	6.11E-21
59	244750	3500	0	5.63E-05	2.05E-13	6.70E-17	3.23E-09	1.54E-14	4.59E-17	3.40E-14	2.65E-21	2.68E-21
60	248250	3750	0	6.03E-05	1.77E-13	3.25E-17	3.46E-09	7.49E-15	2.24E-17	2.84E-14	1.28E-21	1.28E-21
61	252000	3500	0	5.62E-05	1.44E-13	1.41E-17	3.23E-09	3.25E-15	9.87E-18	2.12E-14	5.46E-22	5.37E-22
62	255500	3500	0	5.62E-05	1.35E-13	8.21E-18	3.23E-09	1.90E-15	6.03E-18	1.68E-14	3.14E-22	2.94E-22
63	259000	200	0	3.21E-06	7.53E-15	3.34E-19	1.84E-10	7.76E-17	2.78E-19	7.07E-16	1.33E-23	8.79E-24
64-98	259200	3600	0	5.78E-05	1.35E-13	5.91E-18	3.32E-09	1.37E-15	4.96E-18	1.25E-14	2.34E-22	1.36E-22
Total	35750	227050	0	3.68E-03	6.69E-06	1.59E-07	6.10E-06	5.71E-06	7.33E-08	1.31E-06	1.63E-12	1.63E-12

B.2.3 Release Category 3: CVCS LOCA Outside Containment, Unisolated

The third release category consists of LOCAs that bypass containment through the CVCS charging or letdown lines and are not isolated, allowing a direct release path to the reactor building. Deposition in the CVCS pipe is not credited for mitigation. The total release from the module is considered to be directly to the environment. Release category 3 is a bypassed containment scenario.

The bounding (from a radionuclide release standpoint) scenario is judged to be a CVCS injection line LOCA outside containment that remains unisolated with the complete failure of all mitigating systems. The reactor coolant bypasses containment through the CVCS line break, completely uncovering the core leading to total core support failure and core relocation to the RPV lower plenum. Nearly the complete core inventory of Xe, Cs, I, and Te groups are released from the fuel with a majority releasing to the environment due to the lack of crediting CVCS pipe deposition and reactor building retention. The particle size distribution for release category 3 is presented in Table B-8. The plume release information for release category 3 that was used as input for MACCS calculations is shown in Table B-9.

Table B-8: Particle Size Distribution for Release Category 3

Particle Size	Chemical Group								
	Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
0.15 µm	1.00E-01	2.86E-03	4.87E-03	2.59E-03	9.45E-04	1.10E-03	2.65E-03	9.11E-04	9.04E-04
0.29 µm	1.00E-01	1.40E-02	1.82E-02	1.04E-02	5.36E-03	1.09E-02	1.35E-02	8.86E-03	8.78E-03
0.53 µm	1.00E-01	6.83E-02	4.89E-02	4.92E-02	3.68E-02	6.97E-02	6.83E-02	6.88E-02	6.85E-02
0.99 µm	1.00E-01	2.50E-01	1.86E-01	2.19E-01	1.95E-01	2.53E-01	2.50E-01	2.59E-01	2.59E-01
1.84 µm	1.00E-01	3.67E-01	4.00E-01	3.79E-01	3.79E-01	3.93E-01	3.65E-01	3.81E-01	3.82E-01
3.43 µm	1.00E-01	1.94E-01	2.57E-01	2.22E-01	2.40E-01	2.04E-01	1.93E-01	1.98E-01	1.98E-01
6.38 µm	1.00E-01	6.51E-02	6.79E-02	7.75E-02	8.74E-02	4.92E-02	6.61E-02	5.33E-02	5.33E-02
11.9 µm	1.00E-01	2.32E-02	1.25E-02	2.67E-02	3.28E-02	1.19E-02	2.41E-02	1.61E-02	1.61E-02
22.1 µm	1.00E-01	9.31E-03	3.22E-03	9.98E-03	1.41E-02	4.27E-03	9.77E-03	6.91E-03	6.92E-03
41.2 µm	1.00E-01	6.15E-03	1.96E-03	4.71E-03	9.29E-03	2.67E-03	6.58E-03	7.03E-03	7.03E-03

Table B-9: Release Category 3 Hourly Release Information

Plume Segment	Start of Release (s)	Release Duration (s)	Flow Rate (kg/s)	Gas Density (kg/m^3)	Chemical Group (fraction of initial core inventory)								
					Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
1	11000	3659	1.66E-01	1.73E-01	7.39E-01	6.56E-01	9.35E-03	5.58E-01	5.93E-01	5.17E-03	1.69E-01	1.13E-07	1.13E-07
2	14660	3591	1.29E-01	1.36E-01	1.68E-01	1.38E-01	1.17E-03	2.29E-01	1.60E-01	3.85E-03	3.50E-02	8.94E-08	8.95E-08
3	18250	3750	3.33E-01	1.51E-01	1.70E-02	8.43E-03	3.64E-04	1.12E-02	3.83E-02	3.75E-04	2.13E-03	1.18E-08	1.18E-08
4	22000	3500	7.97E-03	1.74E-01	1.62E-03	1.26E-03	4.00E-06	7.84E-04	1.09E-03	1.57E-06	2.67E-04	6.93E-11	6.94E-11
5	25500	3750	3.63E-06	2.98E-02	2.23E-05	1.52E-05	8.01E-08	2.21E-05	9.78E-06	8.29E-08	3.53E-06	1.21E-11	1.22E-11
6	29250	3500	1.08E-06	2.17E-02	2.85E-05	1.19E-05	7.64E-08	2.03E-05	9.54E-06	7.73E-08	3.04E-06	1.59E-11	1.60E-11
7	32750	3500	1.00E-06	2.15E-02	3.15E-05	1.12E-05	9.97E-08	2.37E-05	1.16E-05	9.31E-08	2.76E-06	1.58E-11	1.60E-11
8	36250	3750	4.21E-06	2.43E-02	2.47E-04	1.05E-04	6.31E-07	2.30E-04	6.35E-05	1.78E-06	2.55E-05	1.78E-10	1.75E-10
9	40000	3500	1.12E-06	3.57E-02	1.59E-04	8.43E-05	1.91E-07	7.87E-05	2.34E-05	8.67E-07	2.29E-05	9.11E-11	9.31E-11
10	43500	3750	1.11E-06	3.02E-02	4.50E-04	1.18E-04	6.70E-07	1.66E-04	3.61E-05	3.88E-06	3.11E-05	2.99E-10	3.00E-10
11	47250	3500	1.07E-06	2.64E-02	5.15E-04	5.82E-05	1.14E-06	2.41E-04	3.14E-05	6.47E-06	1.24E-05	4.84E-10	4.87E-10
12	50750	3500	1.00E-06	2.39E-02	2.86E-04	1.27E-05	6.91E-07	1.02E-04	1.57E-05	2.93E-06	1.68E-06	2.60E-10	2.65E-10
13	54250	3750	1.00E-06	2.23E-02	1.04E-04	6.74E-06	3.66E-07	8.11E-05	1.10E-05	1.24E-06	2.24E-07	1.14E-10	1.12E-10
14	58000	3500	1.00E-06	2.28E-02	5.67E-05	6.91E-06	2.62E-07	8.51E-05	1.19E-05	9.35E-07	1.64E-07	8.98E-11	8.96E-11
15	61500	3750	1.00E-06	2.39E-02	8.17E-06	1.85E-06	3.63E-08	2.25E-05	4.11E-06	1.28E-07	4.47E-08	8.95E-12	8.97E-12
16	65250	3500	1.00E-06	3.61E-02	5.78E-06	2.09E-06	2.89E-08	2.50E-05	5.78E-06	4.38E-08	4.47E-08	2.98E-13	2.42E-13
17	68750	3500	1.00E-06	3.96E-02	5.53E-05	2.28E-05	3.68E-07	2.73E-04	7.30E-05	1.77E-07	5.96E-07	2.56E-13	2.98E-13
18	72250	3750	1.00E-06	4.37E-02	3.61E-05	1.68E-05	2.95E-07	1.97E-04	6.06E-05	2.05E-07	5.51E-07	1.42E-14	1.42E-14
19	76000	3500	1.00E-06	4.73E-02	6.46E-05	3.36E-05	6.36E-07	3.84E-04	6.16E-05	7.35E-07	1.30E-06	0.00E+00	0.00E+00
20	86750	3500	1.00E-06	5.92E-02	1.13E-06	6.56E-07	1.68E-08	8.40E-06	3.58E-07	1.02E-08	0.00E+00	1.28E-13	1.71E-13
21	94000	3500	1.00E-06	8.56E-02	6.74E-06	3.64E-06	1.21E-07	4.77E-05	2.03E-06	1.12E-08	2.98E-08	1.19E-12	6.11E-13
22	97500	3750	1.00E-06	9.21E-02	8.29E-06	4.41E-06	1.58E-07	5.83E-05	2.44E-06	5.59E-09	0.00E+00	1.02E-12	0.00E+00
23	101250	3500	1.00E-06	1.01E-01	2.18E-05	1.13E-05	4.38E-07	1.50E-04	5.84E-06	4.66E-09	0.00E+00	0.00E+00	0.00E+00
24	104750	3500	1.93E-06	1.08E-01	3.37E-05	1.70E-05	6.95E-07	2.26E-04	8.46E-06	4.66E-09	1.49E-08	0.00E+00	0.00E+00
25	108250	3750	1.00E-06	1.11E-01	7.03E-06	3.52E-06	1.48E-07	4.67E-05	1.73E-06	9.31E-10	2.98E-08	0.00E+00	0.00E+00
26	180250	3750	1.00E-06	1.71E-01	2.98E-07	0.00E+00	2.79E-09	5.96E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	11000	93750	-	-	9.27E-01	8.04E-01	1.09E-02	8.02E-01	7.93E-01	9.42E-03	2.07E-01	2.16E-07	2.16E-07

B.2.4 Release Category 4: ECCS Spurious Actuation

The fourth release category consists of loss of RPV inventory into containment due to the spurious opening of an ECCS or RSV valve. DHRS success does not always preclude core damage; therefore, sequences that contain DHRS success and sequences that contain DHRS failure are both considered in this release category. Release category 4 is an intact containment scenario.

The bounding (from a radionuclide release standpoint) accident for this release category is judged to be a spurious opening of an ECCS RVV, with a partial actuation of ECCS (only the remaining RVVs) and a failure of all other mitigating systems. This accident produces a scenario where reactor coolant is lost to containment through the RVVs, and coolant is not available either through RRV recirculation or through CVCS injection. The core is uncovered and unable to be cooled, leading to total core support failure and core relocation to the RPV lower plenum. The plume release information for release category 4 that was used as input for MACCS simulations is shown in Table B-10. The particle size distribution for release category 4 was assumed to be the same as the particle size distribution for release category 5 (see Table B-11).

Table B-10: Release Category 4 Hourly Release Information

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)								
				Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
1	32500	3750	0	1.24E-05	2.38E-06	2.80E-08	1.72E-06	1.67E-06	2.41E-09	5.00E-07	1.30E-13	1.30E-13
2	36250	3500	0	4.54E-05	9.42E-07	5.82E-09	1.10E-06	9.58E-07	1.84E-08	2.00E-07	3.49E-13	3.49E-13
3	39750	3500	0	5.63E-05	8.55E-08	1.05E-09	7.43E-07	1.22E-07	1.12E-09	5.20E-09	8.70E-14	9.51E-14
4	43250	3750	0	5.78E-05	1.04E-07	4.67E-09	2.47E-07	4.85E-07	1.68E-09	1.75E-08	2.32E-13	2.26E-13
5	47000	3500	0	5.83E-05	1.64E-07	7.67E-09	1.61E-07	3.74E-07	1.77E-09	3.43E-08	2.32E-13	2.20E-13
6	50500	3500	0	5.85E-05	1.33E-07	3.88E-09	1.10E-07	2.30E-07	1.06E-09	2.84E-08	1.48E-13	1.45E-13
7	54000	3750	0	6.27E-05	1.17E-07	8.91E-10	9.91E-08	3.96E-08	2.32E-10	2.36E-08	9.33E-14	8.32E-14
8	57750	3500	0	7.22E-05	9.24E-07	1.76E-09	1.13E-06	2.39E-07	6.11E-09	1.75E-07	1.99E-13	2.01E-13
9	61250	3750	0	7.00E-05	3.46E-08	7.71E-10	2.98E-07	1.12E-07	7.63E-12	2.64E-09	1.13E-15	1.30E-15
10	65000	3500	0	6.86E-05	2.66E-08	6.55E-10	2.11E-07	8.23E-08	5.12E-12	2.32E-09	8.23E-16	9.54E-16
11	68500	3500	0	6.63E-05	1.92E-08	4.53E-10	1.35E-07	5.38E-08	3.17E-12	1.97E-09	5.31E-16	6.17E-16
12	72000	3750	0	6.40E-05	1.18E-08	2.62E-10	7.71E-08	3.08E-08	1.78E-12	1.32E-09	3.02E-16	3.51E-16
13	75750	3500	0	6.36E-05	7.38E-09	1.60E-10	4.81E-08	1.89E-08	1.09E-12	8.37E-10	1.85E-16	2.15E-16
14	79250	3750	0	6.28E-05	4.46E-09	9.49E-11	2.94E-08	1.12E-08	6.44E-13	5.10E-10	1.09E-16	1.27E-16
15	83000	3500	0	6.22E-05	2.77E-09	5.79E-11	1.88E-08	6.80E-09	3.92E-13	3.15E-10	6.65E-17	7.74E-17
16	86500	3500	0	6.20E-05	1.73E-09	3.56E-11	1.25E-08	4.19E-09	2.41E-13	1.95E-10	4.09E-17	4.76E-17
17	90000	3750	0	6.18E-05	1.06E-09	2.15E-11	8.48E-09	2.53E-09	1.46E-13	1.17E-10	2.47E-17	2.87E-17
18	93750	3500	0	6.15E-05	6.85E-10	1.37E-11	6.27E-09	1.61E-09	9.27E-14	7.40E-11	1.57E-17	1.83E-17
19	97250	3750	0	6.10E-05	4.37E-10	8.48E-12	4.80E-09	9.97E-10	5.76E-14	4.53E-11	9.75E-18	1.13E-17
20	101000	3500	0	6.02E-05	2.87E-10	5.44E-12	3.92E-09	6.39E-10	3.70E-14	2.86E-11	6.25E-18	7.27E-18
21	104500	3500	0	5.95E-05	1.92E-10	3.48E-12	3.35E-09	4.09E-10	2.42E-14	1.79E-11	4.01E-18	4.66E-18
22	108000	3750	0	5.87E-05	2.07E-09	4.85E-13	7.48E-09	4.33E-11	3.05E-14	2.92E-10	1.31E-18	1.37E-18
23	111750	3500	0	5.70E-05	1.35E-08	1.44E-12	3.41E-08	1.33E-10	1.99E-13	1.70E-09	6.92E-18	6.95E-18
24	115250	3750	0	5.68E-05	9.13E-09	9.66E-13	2.49E-08	1.44E-10	1.33E-13	1.11E-09	4.57E-18	4.60E-18
25	119000	3500	0	5.66E-05	6.34E-09	8.51E-13	1.94E-08	1.77E-10	9.08E-14	7.40E-10	3.10E-18	3.11E-18
26	122500	3500	0	5.65E-05	4.39E-09	9.36E-13	1.54E-08	1.98E-10	6.20E-14	4.90E-10	2.09E-18	2.10E-18
27	126000	3750	0	5.64E-05	2.94E-09	1.02E-12	1.19E-08	2.00E-10	4.12E-14	3.14E-10	1.37E-18	1.37E-18
28	129750	3500	0	5.62E-05	2.16E-09	2.09E-12	1.05E-08	3.62E-10	3.17E-14	2.08E-10	9.77E-19	9.80E-19
29	133250	3750	0	5.61E-05	1.44E-09	1.95E-12	7.95E-09	3.32E-10	2.20E-14	1.33E-10	6.49E-19	6.50E-19
30	137000	3500	0	5.61E-05	9.58E-10	1.41E-12	6.09E-09	2.39E-10	1.47E-14	8.71E-11	4.28E-19	4.29E-19
31	140500	3500	0	5.61E-05	6.36E-10	9.92E-13	4.84E-09	1.68E-10	9.84E-15	5.73E-11	2.83E-19	2.83E-19
32	144000	3750	0	5.60E-05	4.48E-10	1.15E-12	4.14E-09	1.93E-10	7.90E-15	3.72E-11	2.06E-19	2.06E-19
33	147750	3500	0	5.60E-05	3.08E-10	9.51E-13	3.59E-09	1.58E-10	5.76E-15	2.46E-11	1.44E-19	1.44E-19

Table B-10: Release Category 4 Hourly Release Information (Continued)

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)								
				Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
34	151250	3750	0	5.60E-05	2.09E-10	8.07E-13	3.19E-09	1.33E-10	4.27E-15	1.59E-11	1.01E-19	1.01E-19
35	155000	3500	0	5.59E-05	1.49E-10	7.26E-13	2.95E-09	1.18E-10	3.39E-15	1.07E-11	7.54E-20	7.55E-20
36	158500	3500	0	5.59E-05	1.07E-10	6.56E-13	2.78E-09	1.05E-10	2.76E-15	7.15E-12	5.80E-20	5.80E-20
37	162000	3750	0	5.59E-05	7.66E-11	5.90E-13	2.65E-09	9.33E-11	2.28E-15	4.68E-12	4.51E-20	4.51E-20
38	165750	3500	0	5.58E-05	5.65E-11	5.22E-13	2.57E-09	8.14E-11	1.89E-15	3.16E-12	3.60E-20	3.58E-20
39	169250	3750	0	5.58E-05	4.17E-11	4.65E-13	2.50E-09	7.15E-11	1.60E-15	2.08E-12	2.92E-20	2.83E-20
40	173000	3500	0	5.58E-05	3.19E-11	4.17E-13	2.46E-09	6.33E-11	1.38E-15	1.41E-12	2.43E-20	2.35E-20
41	176500	3500	0	5.57E-05	2.52E-11	3.83E-13	2.43E-09	5.75E-11	1.24E-15	9.61E-13	2.09E-20	2.04E-20
42	180000	3750	0	5.57E-05	1.96E-11	3.41E-13	2.41E-09	4.97E-11	1.08E-15	6.37E-13	1.78E-20	1.75E-20
43	183750	3500	0	5.57E-05	1.62E-11	3.24E-13	2.39E-09	4.35E-11	1.01E-15	4.37E-13	1.64E-20	1.62E-20
44	187250	3750	0	5.57E-05	1.30E-11	2.93E-13	2.38E-09	3.55E-11	9.04E-16	2.90E-13	1.45E-20	1.44E-20
45	191000	3500	0	5.56E-05	1.12E-11	2.87E-13	2.37E-09	3.05E-11	8.81E-16	2.00E-13	1.41E-20	1.40E-20
46	194500	3500	0	5.56E-05	9.56E-12	2.68E-13	2.37E-09	2.56E-11	8.30E-16	1.37E-13	1.32E-20	1.31E-20
47	198000	3750	0	5.56E-05	8.41E-12	2.55E-13	2.36E-09	2.18E-11	8.13E-16	9.23E-14	1.29E-20	1.28E-20
48	201750	3500	0	5.55E-05	7.29E-12	2.32E-13	2.36E-09	1.83E-11	7.63E-16	6.35E-14	1.20E-20	1.20E-20
49	205250	3750	0	5.55E-05	6.42E-12	2.12E-13	2.36E-09	1.55E-11	7.25E-16	4.29E-14	1.14E-20	1.13E-20
50	209000	3500	0	5.55E-05	5.66E-12	1.93E-13	2.36E-09	1.32E-11	6.78E-16	2.98E-14	1.05E-20	1.05E-20
51	212500	3500	0	5.55E-05	4.89E-12	1.70E-13	2.35E-09	1.10E-11	6.14E-16	2.07E-14	9.35E-21	9.35E-21
52	216000	3750	0	5.54E-05	4.38E-12	1.51E-13	2.35E-09	9.45E-12	5.74E-16	1.43E-14	8.60E-21	8.60E-21
53	219750	3500	0	5.54E-05	3.87E-12	1.25E-13	2.35E-09	8.07E-12	5.20E-16	1.02E-14	7.77E-21	7.76E-21
54	223250	3750	0	5.54E-05	3.32E-12	9.83E-14	2.35E-09	6.71E-12	4.55E-16	7.13E-15	6.82E-21	6.82E-21
55	227000	3500	0	5.54E-05	2.90E-12	7.73E-14	2.35E-09	5.69E-12	4.03E-16	5.18E-15	6.08E-21	6.07E-21
56	230500	3500	0	5.53E-05	2.53E-12	5.99E-14	2.35E-09	4.81E-12	3.55E-16	3.81E-15	5.34E-21	5.34E-21
57	234000	3750	0	5.53E-05	2.19E-12	4.52E-14	2.35E-09	4.04E-12	3.09E-16	2.80E-15	4.64E-21	4.64E-21
58	237750	3500	0	5.53E-05	1.85E-12	3.52E-14	2.35E-09	3.52E-12	2.79E-16	2.16E-15	4.16E-21	4.16E-21
59	241250	3750	0	5.53E-05	1.51E-12	2.72E-14	2.35E-09	3.12E-12	2.56E-16	1.69E-15	3.83E-21	3.83E-21
60	245000	3500	0	5.53E-05	1.22E-12	2.15E-14	2.35E-09	2.78E-12	2.35E-16	1.37E-15	3.56E-21	3.55E-21
61	248500	3500	0	5.53E-05	9.58E-13	1.68E-14	2.35E-09	2.44E-12	2.11E-16	1.12E-15	3.23E-21	3.22E-21
62	252000	3750	0	5.52E-05	7.50E-13	1.33E-14	2.35E-09	2.17E-12	1.93E-16	9.43E-16	2.99E-21	2.98E-21
63	255750	3450	0	5.52E-05	5.87E-13	1.06E-14	2.35E-09	1.90E-12	1.73E-16	8.03E-16	2.68E-21	2.66E-21
64-97	259200	3600	0	5.52E-05	5.87E-13	1.06E-14	2.35E-09	1.90E-12	1.73E-16	8.03E-16	2.68E-21	2.66E-21
Total	32500	349100	0	5.47E-03	5.01E-06	5.63E-08	6.47E-06	4.45E-06	3.28E-08	1.00E-06	1.47E-12	1.45E-12

B.2.5 Release Category 5: Steam Generator Tube Failure

The fifth release category consists of SGTFs that are unisolated for which CVCS injection fails to supply additional core inventory. ECCS success does not always preclude core damage in a SGTF scenario; therefore, sequences that include ECCS success and sequences that include ECCS failure are both considered in this release category. Release category 5 is a bypassed containment scenario.

The bounding (from a radionuclide release standpoint) scenario for this release category is judged to be an SGTF combined with a secondary line break immediately outside containment to allow a direct release path to the reactor building with the complete failure of all mitigating systems. The reactor coolant bypasses containment through the SGTF and secondary line break, completely uncovering the core leading to total core support failure and core relocation to the RPV lower plenum. Nearly the complete core inventory of Xe, Cs, I, and Te groups are released from the fuel with a large fraction releasing to the environment due to the lack of crediting reactor building retention. The particle size distribution for release category 5 is presented in Table B-11. The plume release information for release category 5 that was used as input for MACCS calculations is shown in Table B-12.

Table B-11: Particle Size Distribution for Release Category 5

Particle Size	Chemical Group								
	Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
0.15 µm	1.00E-01	2.78E-01	1.46E-01	3.89E-01	1.95E-01	1.89E-02	8.99E-02	2.24E-02	2.17E-02
0.29 µm	1.00E-01	1.39E-01	1.52E-02	2.00E-01	6.83E-02	5.80E-03	2.47E-02	6.66E-03	6.58E-03
0.53 µm	1.00E-01	8.19E-02	8.70E-02	1.04E-01	5.76E-02	2.47E-02	5.32E-02	2.70E-02	2.70E-02
0.99 µm	1.00E-01	1.78E-01	3.42E-01	1.59E-01	1.89E-01	2.86E-01	2.20E-01	2.95E-01	2.97E-01
1.84 µm	1.00E-01	2.74E-01	3.59E-01	1.33E-01	3.98E-01	5.90E-01	5.06E-01	5.72E-01	5.72E-01
3.43 µm	1.00E-01	4.75E-02	4.85E-02	1.52E-02	8.68E-02	7.33E-02	1.02E-01	7.42E-02	7.35E-02
6.38 µm	1.00E-01	1.99E-03	1.71E-03	4.47E-04	4.62E-03	1.96E-03	4.62E-03	2.67E-03	2.65E-03
11.9 µm	1.00E-01	2.75E-05	1.59E-05	6.99E-06	5.97E-05	4.99E-05	6.39E-05	6.51E-05	6.44E-05
22.1 µm	1.00E-01	1.40E-07	8.64E-08	5.91E-08	2.29E-07	1.04E-06	3.07E-07	1.04E-06	1.03E-06
41.2 µm	1.00E-01	3.17E-11	9.11E-11	1.70E-11	1.12E-10	1.59E-09	6.70E-11	1.41E-09	1.40E-09

Table B-12: Release Category 5 Hourly Release Information

Plume Segment	Start of Release (s)	Plume Duration (s)	Flow Rate (kg/s)	Gas Density (kg/m^3)	Chemical Group (fraction of initial core inventory)								
					Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
1	5.90E+04	3.72E+03	1.83E-01	3.14E-01	6.20E-01	1.00E-02	1.60E-04	6.33E-03	5.86E-03	4.96E-05	2.47E-03	1.15E-09	1.15E-09
2	6.27E+04	3.53E+03	1.38E-01	3.57E-01	1.86E-01	2.21E-03	3.29E-05	7.10E-03	4.24E-03	7.10E-05	4.79E-04	1.78E-09	1.81E-09
3	6.63E+04	3.75E+03	9.83E-02	3.55E-01	2.09E-02	1.76E-03	1.25E-05	1.26E-03	2.70E-03	1.14E-05	4.78E-04	5.14E-10	5.44E-10
4	7.00E+04	3.46E+03	2.87E-01	3.16E-01	8.91E-02	1.36E-02	6.20E-04	6.38E-02	4.81E-02	4.21E-05	1.24E-03	1.10E-09	1.11E-09
5	7.35E+04	3.79E+03	2.16E-01	2.63E-01	4.79E-02	2.47E-02	1.88E-04	1.38E-01	2.82E-02	3.03E-06	3.56E-04	1.31E-10	1.25E-10
6	7.73E+04	3.50E+03	9.02E-02	5.14E-01	1.49E-02	6.21E-03	5.30E-06	4.84E-02	3.82E-03	7.95E-07	3.61E-05	2.08E-11	2.10E-11
7	8.08E+04	3.50E+03	6.91E-02	5.82E-01	7.39E-03	2.06E-03	3.59E-06	1.89E-02	1.10E-03	5.75E-07	2.01E-05	1.51E-11	1.53E-11
8	8.43E+04	3.75E+03	5.11E-02	5.90E-01	5.49E-03	1.46E-03	2.56E-06	1.58E-02	5.79E-04	4.12E-07	1.39E-05	1.09E-11	1.10E-11
9	8.80E+04	3.50E+03	5.39E-02	5.95E-01	5.76E-03	1.12E-03	2.36E-06	1.25E-02	4.43E-04	3.80E-07	1.30E-05	1.01E-11	1.02E-11
10	9.15E+04	3.75E+03	5.76E-02	5.98E-01	7.45E-03	5.18E-04	2.54E-06	4.87E-03	4.42E-04	4.10E-07	1.48E-05	1.09E-11	1.10E-11
11	9.53E+04	3.50E+03	5.79E-02	5.96E-01	0.00E+00	2.94E-04	2.25E-06	2.26E-03	4.22E-04	3.64E-07	1.45E-05	9.98E-12	1.01E-11
12	9.88E+04	3.50E+03	6.13E-02	5.95E-01	0.00E+00	2.34E-04	2.25E-06	1.50E-03	5.20E-04	3.64E-07	1.51E-05	1.02E-11	1.03E-11
13	1.02E+05	3.75E+03	7.15E-02	5.93E-01	0.00E+00	2.46E-04	2.69E-06	1.39E-03	4.30E-04	4.36E-07	1.76E-05	1.19E-11	1.21E-11
14	1.06E+05	3.50E+03	7.95E-02	5.93E-01	0.00E+00	2.31E-04	2.66E-06	1.23E-03	3.71E-04	4.31E-07	1.69E-05	1.14E-11	1.16E-11
15	1.10E+05	3.50E+03	8.33E-02	5.94E-01	0.00E+00	2.30E-04	2.64E-06	1.20E-03	3.54E-04	4.31E-07	1.69E-05	1.14E-11	1.15E-11
16	1.13E+05	3.75E+03	8.73E-02	5.97E-01	0.00E+00	2.48E-04	2.81E-06	1.29E-03	3.70E-04	4.59E-07	1.82E-05	1.21E-11	1.22E-11
17	1.17E+05	3.50E+03	9.01E-02	6.03E-01	0.00E+00	2.30E-04	2.54E-06	1.19E-03	3.36E-04	4.18E-07	1.68E-05	1.10E-11	1.12E-11
18	1.20E+05	3.75E+03	8.99E-02	6.06E-01	0.00E+00	2.37E-04	2.54E-06	1.22E-03	3.43E-04	4.24E-07	1.72E-05	1.12E-11	1.13E-11
19	1.24E+05	3.50E+03	9.02E-02	6.10E-01	0.00E+00	2.14E-04	2.23E-06	1.11E-03	3.07E-04	3.77E-07	1.54E-05	9.92E-12	1.00E-11
20	1.28E+05	3.75E+03	9.17E-02	6.12E-01	0.00E+00	2.25E-04	2.28E-06	1.16E-03	3.20E-04	3.90E-07	1.61E-05	1.03E-11	1.04E-11
21	1.31E+05	3.50E+03	9.19E-02	6.15E-01	0.00E+00	2.02E-04	2.00E-06	1.05E-03	2.85E-04	3.46E-07	1.44E-05	9.11E-12	9.22E-12
22	1.35E+05	3.50E+03	9.18E-02	6.18E-01	0.00E+00	1.95E-04	1.90E-06	1.01E-03	2.72E-04	3.30E-07	1.38E-05	8.68E-12	8.78E-12
23	1.38E+05	3.75E+03	9.19E-02	6.20E-01	0.00E+00	2.01E-04	1.93E-06	1.05E-03	2.81E-04	3.38E-07	1.43E-05	8.88E-12	8.99E-12
24	1.42E+05	3.50E+03	9.21E-02	6.22E-01	0.00E+00	1.82E-04	1.72E-06	9.45E-04	2.53E-04	3.03E-07	1.28E-05	7.93E-12	8.03E-12
25	1.46E+05	3.50E+03	9.30E-02	6.24E-01	0.00E+00	1.77E-04	1.66E-06	9.22E-04	2.45E-04	2.93E-07	1.25E-05	7.67E-12	7.77E-12
26	1.49E+05	3.75E+03	9.29E-02	6.25E-01	0.00E+00	1.83E-04	1.70E-06	9.53E-04	2.52E-04	2.99E-07	1.29E-05	7.85E-12	7.94E-12
27	1.53E+05	3.50E+03	9.30E-02	6.28E-01	0.00E+00	1.65E-04	1.52E-06	8.59E-04	2.27E-04	2.67E-07	1.16E-05	7.01E-12	7.09E-12
28	1.56E+05	3.75E+03	9.30E-02	6.29E-01	0.00E+00	1.70E-04	1.56E-06	8.89E-04	2.34E-04	2.74E-07	1.20E-05	7.18E-12	7.27E-12
29	1.60E+05	3.50E+03	9.32E-02	6.30E-01	0.00E+00	1.54E-04	1.40E-06	8.03E-04	2.12E-04	2.45E-07	1.09E-05	6.42E-12	6.50E-12
30	1.64E+05	3.75E+03	9.33E-02	6.30E-01	0.00E+00	1.59E-04	1.44E-06	8.31E-04	2.19E-04	2.51E-07	1.13E-05	6.59E-12	6.67E-12
31	1.67E+05	3.50E+03	9.31E-02	6.30E-01	0.00E+00	1.43E-04	1.29E-06	7.48E-04	1.97E-04	2.24E-07	1.01E-05	5.87E-12	5.95E-12

Table B-12: Release Category 5 Hourly Release Information (Continued)

Plume Segment	Start of Release (s)	Plume Duration (s)	Flow Rate (kg/s)	Gas Density (kg/m^3)	Chemical Group (fraction of initial core inventory)								
					Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
32	1.71E+05	3.50E+03	9.30E-02	6.31E-01	0.00E+00	1.38E-04	1.24E-06	7.23E-04	1.91E-04	2.15E-07	9.79E-06	5.63E-12	5.70E-12
33	1.74E+05	3.75E+03	9.38E-02	6.31E-01	0.00E+00	1.44E-04	1.29E-06	7.55E-04	1.99E-04	2.23E-07	1.02E-05	5.82E-12	5.90E-12
34	1.78E+05	3.50E+03	9.33E-02	6.32E-01	0.00E+00	1.29E-04	1.15E-06	6.78E-04	1.78E-04	1.98E-07	9.17E-06	5.17E-12	5.24E-12
35	1.82E+05	3.75E+03	9.36E-02	6.32E-01	0.00E+00	1.34E-04	1.18E-06	7.05E-04	1.85E-04	2.04E-07	9.52E-06	5.33E-12	5.39E-12
36	1.85E+05	3.50E+03	9.34E-02	6.31E-01	0.00E+00	1.20E-04	1.06E-06	6.35E-04	1.67E-04	1.82E-07	8.56E-06	4.75E-12	4.81E-12
37	1.89E+05	3.50E+03	9.35E-02	6.32E-01	0.00E+00	1.16E-04	1.02E-06	6.16E-04	1.61E-04	1.75E-07	8.29E-06	4.56E-12	4.62E-12
38	1.92E+05	3.75E+03	9.33E-02	6.31E-01	0.00E+00	1.20E-04	1.05E-06	6.37E-04	1.66E-04	1.79E-07	8.56E-06	4.67E-12	4.73E-12
39	1.96E+05	3.50E+03	9.26E-02	6.30E-01	0.00E+00	1.07E-04	9.37E-07	5.72E-04	1.49E-04	1.59E-07	7.66E-06	4.15E-12	4.20E-12
40	2.00E+05	3.75E+03	9.25E-02	6.30E-01	0.00E+00	1.11E-04	9.65E-07	5.93E-04	1.54E-04	1.63E-07	7.93E-06	4.26E-12	4.31E-12
41	2.03E+05	3.50E+03	9.26E-02	6.29E-01	0.00E+00	9.99E-05	8.66E-07	5.37E-04	1.39E-04	1.46E-07	7.15E-06	3.82E-12	3.86E-12
42	2.07E+05	3.50E+03	9.13E-02	6.28E-01	0.00E+00	9.51E-05	8.22E-07	5.14E-04	1.33E-04	1.38E-07	6.82E-06	3.61E-12	3.65E-12
43	2.10E+05	3.75E+03	9.10E-02	6.28E-01	0.00E+00	9.81E-05	8.43E-07	5.32E-04	1.36E-04	1.41E-07	7.04E-06	3.70E-12	3.74E-12
44	2.14E+05	3.50E+03	9.07E-02	6.26E-01	0.00E+00	8.82E-05	7.54E-07	4.80E-04	1.22E-04	1.26E-07	6.34E-06	3.30E-12	3.34E-12
45	2.18E+05	3.50E+03	9.00E-02	6.28E-01	0.00E+00	8.46E-05	7.19E-07	4.63E-04	1.17E-04	1.20E-07	6.08E-06	3.15E-12	3.19E-12
46	2.21E+05	3.75E+03	9.01E-02	6.27E-01	0.00E+00	8.77E-05	7.44E-07	4.82E-04	1.21E-04	1.24E-07	6.32E-06	3.25E-12	3.29E-12
47	2.25E+05	3.50E+03	9.03E-02	6.27E-01	0.00E+00	7.93E-05	6.69E-07	4.38E-04	1.10E-04	1.11E-07	5.71E-06	2.92E-12	2.96E-12
48	2.28E+05	3.75E+03	9.02E-02	6.28E-01	0.00E+00	8.19E-05	6.88E-07	4.55E-04	1.14E-04	1.14E-07	5.90E-06	3.00E-12	3.04E-12
49	2.32E+05	3.50E+03	9.02E-02	6.28E-01	0.00E+00	7.39E-05	6.18E-07	4.13E-04	1.03E-04	1.02E-07	5.33E-06	2.70E-12	2.73E-12
50	2.36E+05	3.75E+03	9.07E-02	6.28E-01	0.00E+00	7.70E-05	6.41E-07	4.32E-04	1.07E-04	1.06E-07	5.56E-06	2.79E-12	2.83E-12
51	2.39E+05	3.50E+03	9.01E-02	6.29E-01	0.00E+00	6.89E-05	5.71E-07	3.89E-04	9.58E-05	9.46E-08	4.97E-06	2.49E-12	2.52E-12
52	2.43E+05	3.50E+03	9.08E-02	6.29E-01	0.00E+00	6.74E-05	5.58E-07	3.83E-04	9.36E-05	9.21E-08	4.87E-06	2.42E-12	2.45E-12
53	2.46E+05	3.75E+03	9.03E-02	6.29E-01	0.00E+00	6.94E-05	5.71E-07	3.97E-04	9.64E-05	9.43E-08	5.01E-06	2.48E-12	2.51E-12
54	2.50E+05	3.50E+03	9.09E-02	6.31E-01	0.00E+00	6.30E-05	5.17E-07	3.63E-04	8.76E-05	8.54E-08	4.56E-06	2.25E-12	2.27E-12
55	2.54E+05	3.50E+03	9.04E-02	6.32E-01	0.00E+00	6.07E-05	4.97E-07	3.51E-04	8.42E-05	8.18E-08	4.39E-06	2.15E-12	2.18E-12
56	2.57E+05	2.20E+03	9.02E-02	6.33E-01	0.00E+00	3.75E-05	3.06E-07	2.19E-04	5.20E-05	5.05E-08	2.72E-06	1.33E-12	1.34E-12
Total	5.90E+04	2.00E+05	-	-	1.00E+00	7.03E-02	1.09E-03	3.54E-01	1.05E-01	1.90E-04	5.59E-03	5.03E-09	5.09E-09

B.2.6 Release Category 6: General Transient With RSV Stuck Open

The sixth release category consists of general transients where an RSV is demanded and subsequently sticks open with the failure of ECCS and CVCS injection (when available). DHRS success does not always preclude core damage; therefore, sequences that contain DHRS success and sequences that contain DHRS failure are both considered in this release category. Release category 6 is an intact containment scenario.

The bounding (from a radionuclide release standpoint) accident for this release category is judged to be a transient where pressure builds in the RPV until an RSV is demanded to relieve primary system pressure. The RSV sticks open, followed by a partial actuation of ECCS with the RVVs opening and a failure of all other mitigating systems, which produces a scenario where reactor coolant is lost to containment through the RVVs and coolant is not available either through RRV recirculation or through CVCS injection. The core is uncovered and unable to be cooled, leading to total core support failure and core relocation to the RPV lower plenum. The plume release information for release category 6 that was used as input for MACCS simulations is shown in Table B-13. The particle size distribution for release category 6 was assumed to be the same as the particle size distribution for release category 5 (see Table B-11).

Table B-13: Release Category 6 Hourly Release Information

Plume Segment	Start of Release (s)	Plume Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)								
				Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
1	44250	3750	0	1.11E-05	2.09E-06	2.03E-08	1.50E-06	1.21E-06	1.76E-09	4.38E-07	9.39E-14	9.39E-14
2	48000	3500	0	4.52E-05	1.51E-06	6.11E-09	1.25E-06	1.47E-06	1.97E-08	3.30E-07	3.80E-13	3.80E-13
3	51500	3500	0	5.46E-05	1.17E-07	1.33E-09	6.89E-07	1.45E-07	3.33E-09	1.44E-08	1.24E-13	1.28E-13
4	55000	3750	0	5.92E-05	1.73E-07	2.05E-08	2.52E-07	1.74E-06	1.80E-09	3.17E-08	3.41E-13	3.29E-13
5	58750	3500	0	5.56E-05	1.29E-07	9.06E-09	1.33E-07	6.84E-07	1.03E-09	2.59E-08	1.88E-13	1.85E-13
6	62250	3500	0	5.64E-05	1.24E-07	5.20E-09	1.19E-07	3.94E-07	1.21E-09	2.47E-08	1.74E-13	1.76E-13
7	65750	3750	0	6.05E-05	1.10E-07	3.19E-09	1.28E-07	2.33E-07	1.08E-09	2.21E-08	1.81E-13	1.85E-13
8	69500	4000	0	7.13E-05	4.46E-07	1.98E-09	3.94E-07	3.23E-07	1.18E-09	9.56E-08	1.50E-13	1.51E-13
9	73500	3500	0	7.35E-05	3.28E-08	4.74E-10	1.88E-07	9.10E-08	1.45E-11	4.14E-09	2.33E-15	2.52E-15
10	77000	3500	0	7.20E-05	2.95E-08	4.63E-10	1.39E-07	8.29E-08	8.99E-12	4.30E-09	1.63E-15	1.75E-15
11	80500	3500	0	6.97E-05	2.19E-08	3.36E-10	9.54E-08	6.11E-08	5.65E-12	3.35E-09	1.07E-15	1.14E-15
12	84000	3750	0	7.15E-05	1.53E-08	2.21E-10	6.20E-08	4.11E-08	3.48E-12	2.43E-09	6.69E-16	7.15E-16
13	87750	3500	0	6.63E-05	8.94E-09	1.28E-10	3.66E-08	2.38E-08	2.00E-12	1.43E-09	3.84E-16	4.11E-16
14	91250	3750	0	6.98E-05	5.85E-09	8.19E-11	2.45E-08	1.53E-08	1.27E-12	9.40E-10	2.45E-16	2.62E-16
15	95000	3500	0	6.45E-05	3.46E-09	4.76E-11	1.52E-08	8.93E-09	7.47E-13	5.56E-10	1.42E-16	1.52E-16
16	98500	3500	0	6.41E-05	2.20E-09	2.96E-11	1.04E-08	5.57E-09	4.92E-13	3.50E-10	8.90E-17	9.50E-17
17	102000	3750	0	6.83E-05	1.45E-09	1.91E-11	7.80E-09	3.60E-09	3.69E-13	2.27E-10	5.82E-17	6.21E-17
18	105750	3500	0	6.33E-05	8.60E-10	1.11E-11	5.49E-09	2.09E-09	2.81E-13	1.32E-10	3.49E-17	3.71E-17
19	109250	3750	0	6.75E-05	5.72E-10	7.22E-12	4.63E-09	1.36E-09	2.75E-13	8.53E-11	2.41E-17	2.56E-17
20	113000	3500	0	6.25E-05	3.43E-10	4.23E-12	3.60E-09	7.96E-10	2.70E-13	4.99E-11	1.59E-17	1.67E-17
21	116500	3500	0	6.24E-05	2.24E-10	2.67E-12	3.16E-09	5.00E-10	2.97E-13	3.17E-11	1.20E-17	1.25E-17
22	120000	3750	0	6.58E-05	1.53E-10	1.73E-12	3.02E-09	3.21E-10	3.71E-13	2.10E-11	1.06E-17	1.10E-17
23	123750	3500	0	6.07E-05	9.68E-11	1.00E-12	2.62E-09	1.80E-10	5.35E-13	1.29E-11	1.12E-17	1.14E-17
24	127250	3750	0	6.50E-05	7.42E-11	7.00E-13	2.69E-09	1.15E-10	8.56E-13	9.94E-12	1.55E-17	1.56E-17
25	131000	3500	0	6.02E-05	6.10E-11	5.55E-13	2.44E-09	7.00E-11	1.60E-12	8.93E-12	2.69E-17	2.69E-17
26	134500	3500	0	5.95E-05	5.63E-11	5.19E-13	2.37E-09	5.01E-11	2.18E-12	8.99E-12	3.60E-17	3.61E-17
27	138000	3750	0	6.27E-05	6.99E-11	7.08E-13	2.49E-09	5.29E-11	3.69E-12	1.27E-11	6.23E-17	6.23E-17
28	141750	3500	0	5.80E-05	9.68E-11	1.11E-12	2.31E-09	7.25E-11	5.63E-12	1.98E-11	1.07E-16	1.07E-16
29	145250	3750	0	6.19E-05	1.37E-10	1.67E-12	2.48E-09	1.01E-10	6.36E-12	2.90E-11	1.67E-16	1.67E-16
30	149000	3500	0	5.76E-05	1.16E-10	1.35E-12	2.32E-09	7.94E-11	4.47E-12	2.40E-11	1.34E-16	1.34E-16
31	152500	3500	0	5.76E-05	1.04E-10	1.22E-12	2.39E-09	6.27E-11	2.80E-12	1.92E-11	8.51E-17	8.51E-17

Table B-13: Release Category 6 Hourly Release Information (Continued)

Plume Segment	Start of Release (s)	Plume Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)								
				Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
32	156000	3750	0	6.17E-05	9.14E-11	1.18E-12	2.57E-09	5.93E-11	1.84E-12	1.56E-11	5.69E-17	5.69E-17
33	159750	3500	0	5.76E-05	7.17E-11	1.04E-12	2.39E-09	5.48E-11	1.10E-12	1.15E-11	3.51E-17	3.51E-17
34	163250	3750	0	6.17E-05	6.36E-11	1.04E-12	2.54E-09	5.44E-11	7.51E-13	9.67E-12	2.48E-17	2.48E-17
35	167000	3500	0	5.76E-05	5.02E-11	9.19E-13	2.36E-09	4.57E-11	4.69E-13	7.28E-12	1.62E-17	1.62E-17
36	170500	3500	0	5.76E-05	4.29E-11	8.74E-13	2.34E-09	4.06E-11	3.22E-13	5.96E-12	1.18E-17	1.18E-17
37	174000	3750	0	6.17E-05	3.92E-11	8.66E-13	2.49E-09	3.79E-11	2.37E-13	5.21E-12	9.37E-18	9.37E-18
38	177750	3500	0	5.76E-05	3.18E-11	7.34E-13	2.31E-09	3.10E-11	1.60E-13	4.06E-12	6.91E-18	6.91E-18
39	181250	3750	0	6.18E-05	2.94E-11	6.97E-13	2.46E-09	2.87E-11	1.25E-13	3.61E-12	6.02E-18	6.02E-18
40	185000	3500	0	5.77E-05	2.39E-11	5.69E-13	2.29E-09	2.31E-11	8.93E-14	2.82E-12	4.77E-18	4.77E-18
41	188500	3500	0	5.77E-05	2.08E-11	4.94E-13	2.28E-09	1.99E-11	6.99E-14	2.36E-12	4.16E-18	4.16E-18
42	192000	3750	0	6.18E-05	1.93E-11	4.53E-13	2.43E-09	1.81E-11	5.93E-14	2.09E-12	3.96E-18	3.97E-18
43	195750	3500	0	5.77E-05	1.58E-11	3.65E-13	2.26E-09	1.45E-11	4.57E-14	1.65E-12	3.39E-18	3.39E-18
44	199250	3750	0	6.18E-05	1.48E-11	3.33E-13	2.42E-09	1.32E-11	4.09E-14	1.48E-12	3.38E-18	3.38E-18
45	203000	3500	0	5.77E-05	1.23E-11	2.68E-13	2.25E-09	1.06E-11	3.32E-14	1.17E-12	3.00E-18	3.00E-18
46	206500	3500	0	5.77E-05	1.10E-11	2.31E-13	2.25E-09	9.20E-12	2.96E-14	1.01E-12	2.90E-18	2.90E-18
47	210000	3750	0	6.19E-05	1.05E-11	2.11E-13	2.40E-09	8.44E-12	2.86E-14	9.18E-13	3.01E-18	3.01E-18
48	213750	3500	0	5.77E-05	8.78E-12	1.70E-13	2.24E-09	6.84E-12	2.48E-14	7.43E-13	2.76E-18	2.76E-18
49	217250	3750	0	6.19E-05	8.46E-12	1.56E-13	2.40E-09	6.33E-12	2.52E-14	6.91E-13	2.92E-18	2.92E-18
50	221000	3500	0	5.78E-05	7.19E-12	1.26E-13	2.24E-09	5.18E-12	2.27E-14	5.71E-13	2.72E-18	2.72E-18
51	224500	3500	0	5.78E-05	6.57E-12	1.10E-13	2.24E-09	4.57E-12	2.23E-14	5.11E-13	2.73E-18	2.73E-18
52	228000	3750	0	6.19E-05	6.45E-12	1.03E-13	2.39E-09	4.32E-12	2.38E-14	4.94E-13	2.95E-18	2.94E-18
53	231750	3500	0	5.78E-05	5.61E-12	8.53E-14	2.23E-09	3.65E-12	2.27E-14	4.30E-13	2.81E-18	2.81E-18
54	235250	3750	0	6.20E-05	5.61E-12	8.08E-14	2.39E-09	3.57E-12	2.53E-14	4.36E-13	3.09E-18	3.09E-18
55	239000	3500	0	5.78E-05	4.93E-12	6.72E-14	2.23E-09	3.10E-12	2.47E-14	3.93E-13	2.96E-18	2.96E-18
56	242500	3500	0	5.79E-05	4.69E-12	6.00E-14	2.23E-09	2.94E-12	2.61E-14	3.86E-13	3.05E-18	3.05E-18
57	246000	3750	0	6.20E-05	4.78E-12	5.73E-14	2.40E-09	3.04E-12	2.98E-14	4.11E-13	3.38E-18	3.38E-18
58	249750	3500	0	5.79E-05	4.22E-12	4.78E-14	2.24E-09	2.76E-12	2.89E-14	3.78E-13	3.20E-18	3.20E-18
59	253250	3750	0	6.21E-05	4.29E-12	4.66E-14	2.40E-09	2.90E-12	3.21E-14	4.01E-13	3.45E-18	3.45E-18
60	257000	2200	0	3.64E-05	2.45E-12	2.62E-14	1.41E-09	1.70E-12	1.89E-14	2.34E-13	2.04E-18	2.04E-18
61-96	259200	3600	0	5.96E-05	4.00E-12	4.29E-14	2.30E-09	2.78E-12	3.09E-14	3.83E-13	3.33E-18	3.33E-18
Total	44250	344550	0	5.75E-03	4.82E-06	6.95E-08	5.24E-06	6.54E-06	3.12E-08	1.00E-06	1.64E-12	1.64E-12

B.2.7 Release Category 7: General Transient With No RSVs

The seventh release category consists of general transients where an RSV is demanded and subsequently fails to open. In scenarios where the DHRS or containment flooding and drain system (CFDS) could mitigate core damage, these systems are assumed to be unavailable. Release category 7 is an intact containment scenario.

The bounding (from a radionuclide release standpoint) accident for this release category is judged to be a general transient where the RSVs fail to open, leading to RPV overpressure failure at the pressurizer heater inspection port. The DHRS and CFDS fail to operate. The ECCS partially actuates upon high CNV level, with the RVVs opening. Due to partial ECCS actuation, reactor coolant completely relocates to the CNV and is unable to recirculate to the RPV. The core is uncovered and unable to be cooled, leading to core support failure and core relocation to the RPV lower plenum. The plume release information for release category 7 that was used as input for MACCS simulations is shown in Table B-14. The particle size distribution for release category 7 was assumed to be the same as the particle size distribution for release category 5 (see Table B-11).

Table B-14: Release Category 7 Hourly Release Information

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)								
				Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
1	33000	3750	0	1.60E-05	2.19E-06	7.38E-09	1.60E-06	1.66E-06	3.20E-09	4.80E-07	1.49E-13	1.49E-13
2	36750	3750	0	4.67E-05	1.14E-06	1.03E-08	1.61E-06	1.39E-06	2.90E-08	2.38E-07	5.53E-13	5.53E-13
3	40500	3500	0	5.28E-05	3.51E-08	4.57E-10	3.10E-07	4.00E-08	7.81E-10	2.58E-09	4.55E-14	4.81E-14
4	44000	3500	0	5.41E-05	3.06E-08	1.60E-09	5.74E-08	3.30E-07	4.36E-10	5.88E-09	6.69E-14	6.87E-14
5	47500	4000	0	6.31E-05	1.96E-08	5.08E-10	1.04E-08	2.26E-08	7.74E-11	4.20E-09	1.18E-14	1.18E-14
6	51500	3500	0	5.60E-05	1.50E-08	2.17E-10	1.14E-08	2.36E-08	4.95E-11	3.28E-09	1.25E-14	1.33E-14
7	55000	3870	0	6.83E-05	1.15E-07	4.19E-10	1.08E-07	8.18E-08	1.35E-10	2.47E-08	3.47E-14	3.41E-14
8	58870	3630	0	6.88E-05	8.80E-08	2.52E-09	2.98E-07	1.16E-07	1.19E-11	1.51E-08	3.72E-14	4.32E-14
9	62500	3500	0	6.55E-05	1.14E-08	6.03E-11	2.90E-08	5.88E-09	1.51E-12	2.13E-09	1.89E-15	1.87E-15
10	66000	3500	0	6.16E-05	8.28E-09	5.32E-11	7.41E-09	5.98E-09	3.01E-12	1.64E-09	9.25E-16	8.24E-16
11	69500	3500	0	6.09E-05	3.06E-09	7.97E-12	3.26E-09	8.14E-10	5.16E-12	5.69E-10	1.98E-16	1.84E-16
12	73000	3500	0	6.03E-05	2.23E-09	4.18E-12	2.75E-09	4.03E-10	5.52E-12	4.41E-10	1.42E-16	1.35E-16
13	76500	3500	0	6.01E-05	1.32E-09	2.16E-12	2.49E-09	1.99E-10	4.16E-12	2.64E-10	9.10E-17	8.80E-17
14	80000	3500	0	5.97E-05	7.29E-10	1.14E-12	2.35E-09	1.01E-10	2.83E-12	1.47E-10	5.65E-17	5.51E-17
15	83500	4000	0	6.77E-05	4.22E-10	6.55E-13	2.59E-09	5.56E-11	2.04E-12	8.57E-11	3.80E-17	3.74E-17
16	87500	3500	0	5.76E-05	2.00E-10	3.21E-13	2.18E-09	2.64E-11	1.18E-12	4.14E-11	2.10E-17	2.08E-17
17	91000	3500	0	5.65E-05	2.82E-09	2.65E-12	7.69E-09	1.38E-10	1.17E-12	5.05E-10	3.74E-17	3.80E-17
18	94500	3500	0	5.63E-05	3.18E-09	2.51E-12	8.47E-09	1.30E-10	1.09E-12	5.67E-10	3.52E-17	3.57E-17
19	98000	3500	0	5.61E-05	2.46E-09	1.85E-12	7.04E-09	9.63E-11	8.10E-13	4.38E-10	2.60E-17	2.64E-17
20	101500	4000	0	6.39E-05	1.91E-09	1.39E-12	6.21E-09	7.25E-11	6.13E-13	3.37E-10	1.96E-17	1.99E-17
21	105500	3500	0	5.58E-05	1.15E-09	8.25E-13	4.39E-09	4.30E-11	3.66E-13	2.03E-10	1.17E-17	1.18E-17
22	109000	3500	0	5.57E-05	7.82E-10	5.52E-13	3.64E-09	2.88E-11	2.46E-13	1.37E-10	7.82E-18	7.93E-18
23	112500	3500	0	5.56E-05	5.28E-10	3.68E-13	3.12E-09	1.92E-11	1.65E-13	9.15E-11	5.23E-18	5.31E-18
24	116000	3500	0	5.55E-05	3.56E-10	2.45E-13	2.77E-09	1.28E-11	1.11E-13	6.12E-11	3.50E-18	3.55E-18
25	119500	4000	0	6.33E-05	2.60E-10	1.77E-13	2.87E-09	9.27E-12	8.09E-14	4.43E-11	2.53E-18	2.57E-18
26	123500	3500	0	5.53E-05	1.55E-10	1.04E-13	2.36E-09	5.46E-12	4.80E-14	2.61E-11	1.50E-18	1.52E-18
27	127000	3500	0	5.52E-05	1.06E-10	7.03E-14	2.26E-09	3.70E-12	3.27E-14	1.76E-11	1.02E-18	1.03E-18
28	130500	3500	0	5.51E-05	7.29E-11	4.77E-14	2.20E-09	2.51E-12	2.25E-14	1.20E-11	6.94E-19	7.03E-19
29	134000	3500	0	5.51E-05	5.23E-11	3.34E-14	2.15E-09	1.77E-12	1.61E-14	8.34E-12	4.91E-19	4.97E-19
30	137500	4000	0	6.29E-05	4.62E-11	2.69E-14	2.43E-09	1.44E-12	1.43E-14	6.61E-12	4.15E-19	4.20E-19
31	141500	3500	0	5.49E-05	3.33E-11	1.73E-14	2.10E-09	9.36E-13	1.05E-14	4.15E-12	2.86E-19	2.89E-19
32	145000	3500	0	5.48E-05	2.76E-11	1.26E-14	2.09E-09	6.94E-13	8.91E-15	2.94E-12	2.27E-19	2.29E-19
33	148500	3500	0	5.48E-05	2.32E-11	9.30E-15	2.07E-09	5.21E-13	7.66E-15	2.11E-12	1.84E-19	1.85E-19

Table B-14: Release Category 7 Hourly Release Information (Continued)

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)								
				Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
34	152000	3500	0	5.47E-05	1.93E-11	6.81E-15	2.06E-09	3.89E-13	6.53E-15	1.50E-12	1.48E-19	1.49E-19
35	155500	4000	0	6.24E-05	1.78E-11	5.44E-15	2.35E-09	3.17E-13	6.19E-15	1.15E-12	1.33E-19	1.33E-19
36	159500	3500	0	5.45E-05	1.29E-11	3.48E-15	2.05E-09	2.08E-13	4.60E-15	7.10E-13	9.45E-20	9.47E-20
37	163000	3500	0	5.45E-05	1.07E-11	2.56E-15	2.04E-09	1.56E-13	3.91E-15	5.00E-13	7.73E-20	7.73E-20
38	166500	3500	0	5.44E-05	8.90E-12	1.90E-15	2.04E-09	1.18E-13	3.33E-15	3.53E-13	6.36E-20	6.35E-20
39	170000	3500	0	5.43E-05	7.41E-12	1.41E-15	2.03E-09	8.99E-14	2.84E-15	2.49E-13	5.27E-20	5.25E-20
40	173500	4000	0	6.20E-05	6.90E-12	1.17E-15	2.32E-09	7.63E-14	2.72E-15	1.92E-13	4.90E-20	4.88E-20
41	177500	3500	0	5.42E-05	5.08E-12	7.76E-16	2.03E-09	5.20E-14	2.05E-15	1.19E-13	3.61E-20	3.59E-20
42	181000	3500	0	5.42E-05	3.46E-12	4.79E-16	2.02E-09	3.29E-14	1.43E-15	6.83E-14	2.46E-20	2.45E-20
43	184500	3500	0	5.41E-05	3.98E-13	3.77E-17	2.02E-09	2.89E-15	1.78E-16	3.16E-15	2.85E-21	2.79E-21
44	188000	3500	0	5.41E-05	3.06E-13	2.28E-17	2.02E-09	1.90E-15	1.44E-16	7.03E-16	2.21E-21	2.05E-21
45	191500	4000	0	6.18E-05	3.01E-13	2.16E-17	2.30E-09	1.82E-15	1.46E-16	3.57E-16	2.21E-21	2.03E-21
46	195500	3500	0	5.40E-05	2.35E-13	1.70E-17	2.01E-09	1.42E-15	1.17E-16	2.16E-16	1.75E-21	1.61E-21
47	199000	3500	0	5.40E-05	2.13E-13	1.56E-17	2.01E-09	1.29E-15	1.08E-16	1.66E-16	1.61E-21	1.48E-21
48	202500	3500	0	5.39E-05	1.92E-13	1.43E-17	2.01E-09	1.17E-15	1.00E-16	1.35E-16	1.47E-21	1.36E-21
49	206000	3500	0	5.39E-05	1.77E-13	1.34E-17	2.01E-09	1.09E-15	9.43E-17	1.17E-16	1.38E-21	1.27E-21
50	209500	4000	0	6.15E-05	1.82E-13	1.41E-17	2.29E-09	1.13E-15	9.96E-17	1.15E-16	1.44E-21	1.34E-21
51	213500	3500	0	5.38E-05	1.46E-13	1.15E-17	2.01E-09	9.16E-16	8.16E-17	9.07E-17	1.18E-21	1.09E-21
52	217000	3500	0	5.38E-05	1.35E-13	1.08E-17	2.00E-09	8.55E-16	7.69E-17	8.31E-17	1.10E-21	1.02E-21
53	220500	3500	0	5.37E-05	1.22E-13	1.00E-17	2.00E-09	7.85E-16	7.13E-17	7.54E-17	1.01E-21	9.46E-22
54	224000	3500	0	5.37E-05	1.08E-13	9.04E-18	2.00E-09	7.04E-16	6.44E-17	6.70E-17	8.95E-22	8.55E-22
55	227500	4000	0	6.13E-05	1.06E-13	9.12E-18	2.29E-09	7.05E-16	6.51E-17	6.67E-17	8.75E-22	4.59E-22
56	231500	3500	0	5.36E-05	8.32E-14	7.27E-18	2.00E-09	5.60E-16	5.20E-17	5.27E-17	7.08E-22	3.68E-22
57	235000	3500	0	5.36E-05	7.71E-14	6.87E-18	2.00E-09	5.26E-16	4.91E-17	4.93E-17	6.39E-22	3.79E-22
58	238500	3500	0	5.36E-05	1.29E-13	1.08E-17	2.00E-09	8.32E-16	7.71E-17	8.21E-17	9.83E-22	4.63E-22
59	242000	3500	0	5.36E-05	2.01E-13	1.67E-17	2.00E-09	1.28E-15	1.19E-16	1.28E-16	1.50E-21	2.72E-22
60	245500	4000	0	6.12E-05	2.89E-13	2.46E-17	2.28E-09	1.88E-15	1.76E-16	1.86E-16	1.80E-21	3.34E-22
61	249500	3500	0	5.35E-05	2.77E-13	2.42E-17	1.99E-09	1.85E-15	1.73E-16	1.81E-16	1.64E-21	3.11E-22
62	253000	3500	0	5.35E-05	2.89E-13	2.60E-17	1.99E-09	1.97E-15	1.86E-16	1.91E-16	1.69E-21	8.68E-23
63	256500	2700	0	4.12E-05	2.26E-13	2.07E-17	1.54E-09	1.56E-15	1.48E-16	1.50E-16	1.31E-21	6.19E-23
64-97	259200	3600	0	5.50E-05	3.01E-13	2.75E-17	2.05E-09	2.09E-15	1.97E-16	2.00E-16	1.75E-21	8.26E-23
Total	33000	348600	0	5.41E-03	3.68E-06	2.35E-08	4.25E-06	3.68E-06	3.37E-08	7.82E-07	9.14E-13	9.25E-13

B.2.8 Release Category 8: Dropped Module During Transport

The final release category consists of core damage and environmental release due to a dropped module during refueling transport. In this group of scenarios, the reactor building crane fails to hold the module during transport and the module impacts the reactor pool floor. The module may impact the floor off center, causing the module to rest horizontally on the pool floor, which affects the ability of natural circulation to maintain core cooling, leading to core damage. The containment can become damaged during impact with the pool floor or subsequent horizontal relocation, creating a release path to the environment. Depending on the angle at which the module tips over, the module may strike other modules. The module drop induced CDF was estimated as 8.8E-08 per refueling in the LPSD PRA. There are six refueling outages per year in a NuScale 12 module plant; therefore, that frequency is multiplied by six for a module drop CDF of 5.3E-07 per year. This accounts for the probability that a single module could be dropped, per year, at a 12 module plant. Release category 8 is conservatively treated as a bypassed containment scenario.

The bounding (from a radionuclide release standpoint) accident for this release category is judged to be a module drop accident where the containment is damaged during the drop and the CNV fails with an opening of 0.5 cm^2 on the bottom of the module (the side of the module facing the pool floor). The module pressurizes, forcing the reactor coolant from the module while precluding any reactor pool water from entering the module. The accident progresses to severe core damage, with approximately 60 percent of the core inventory of iodine releasing to the reactor pool. However, the radionuclide aerosols released from the module are scrubbed by the pool, and the majority of the release to the environment is in the form of noble gases and methyl iodide, which greatly reduces the impact of the release. Reactor pool scrubbing of the release follows the approach outlined in Appendix B of Regulatory Guide 1.183 (Reference 8.1-30). The initial core inventory of organic iodide is assumed to comprise 0.15 percent of the total initial core inventory of iodine. All radionuclide groups, except for Xe and iodine, are assumed to be completely scrubbed by the pool. Xe and organic iodide are assumed not to be scrubbed by the reactor pool. Elemental iodine is assumed to have a decontamination factor of 500 due to reactor pool scrubbing. The particle size distribution for release category 8 is shown in Table B-15. The plume release information for release category 8 that was used as input for MACCS simulations is shown in Table B-16.

Table B-15: Particle Size Distribution for Release Category 8

Particle Size	Chemical Group (fraction of initial core inventory)		
	Xe	Elemental Iodine	Organic Iodide
0.15 μm	1.00E-01	9.78E-02	1.00E-01
0.29 μm	1.00E-01	5.97E-01	1.00E-01
0.53 μm	1.00E-01	3.02E-01	1.00E-01
0.99 μm	1.00E-01	2.61E-03	1.00E-01
1.84 μm	1.00E-01	3.28E-09	1.00E-01
3.43 μm	1.00E-01	0.00E+00	1.00E-01
6.38 μm	1.00E-01	0.00E+00	1.00E-01
11.9 μm	1.00E-01	0.00E+00	1.00E-01
22.1 μm	1.00E-01	0.00E+00	1.00E-01
41.2 μm	1.00E-01	0.00E+00	1.00E-01

Table B-16: Release Category 8 Hourly Release Information

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)		
				Xe	Elemental Iodine	Organic Iodide
1	191610	2790	0	0.00E+00	5.18E-08	0.00E+00
2	194400	3600	0	0.00E+00	1.89E-07	0.00E+00
3	198000	3600	0	0.00E+00	3.65E-07	0.00E+00
4	201600	3600	0	0.00E+00	5.86E-07	0.00E+00
5	205200	3600	0	0.00E+00	7.65E-07	0.00E+00
6	208800	3600	0	0.00E+00	9.16E-07	0.00E+00
7	212400	3600	0	0.00E+00	9.87E-07	0.00E+00
8	216000	3600	0	0.00E+00	1.05E-06	0.00E+00
9	219600	3600	0	0.00E+00	1.32E-06	0.00E+00
10	223200	3600	0	0.00E+00	1.82E-06	0.00E+00
11	226800	3600	0	0.00E+00	2.89E-06	0.00E+00
12	230400	3600	0	0.00E+00	4.98E-06	0.00E+00
13	234000	3600	0	0.00E+00	8.49E-06	0.00E+00
14	237600	3600	0	0.00E+00	1.28E-05	0.00E+00
15	241200	3600	0	0.00E+00	1.62E-05	0.00E+00
16	244800	3600	0	0.00E+00	1.95E-05	0.00E+00
17	248400	3600	0	0.00E+00	2.41E-05	0.00E+00
18	252000	3600	0	0.00E+00	2.78E-05	0.00E+00
19	255600	3600	0	0.00E+00	2.98E-05	0.00E+00
20	259200	3600	0	0.00E+00	3.45E-05	0.00E+00
21	262800	3600	0	0.00E+00	4.05E-05	0.00E+00
22	266400	3600	0	0.00E+00	4.67E-05	0.00E+00
23	270000	3600	0	0.00E+00	5.15E-05	0.00E+00
24	273600	3600	0	0.00E+00	5.57E-05	0.00E+00
25	277200	3600	0	0.00E+00	6.19E-05	0.00E+00
26	280800	3600	0	0.00E+00	6.97E-05	0.00E+00
27	284400	3600	0	0.00E+00	7.65E-05	0.00E+00
28	288000	3600	0	0.00E+00	8.24E-05	0.00E+00
29	291600	3600	0	0.00E+00	8.62E-05	0.00E+00
30	295200	3600	0	0.00E+00	8.83E-05	0.00E+00
31	298800	3600	0	0.00E+00	8.84E-05	0.00E+00
32	302400	3600	0	0.00E+00	8.66E-05	0.00E+00
33	306000	3600	0	0.00E+00	8.33E-05	0.00E+00
34	309600	3600	0	1.46E-01	5.99E-05	1.62E-01
35	313200	3600	0	2.79E-01	1.78E-05	3.06E-01
36	316800	3600	0	1.60E-01	1.01E-05	1.72E-01
37	320400	3600	0	5.13E-02	3.25E-06	5.48E-02
38	324000	3600	0	3.17E-03	2.23E-07	3.37E-03
39	338400	3600	0	0.00E+00	1.90E-08	0.00E+00
40	342000	3600	0	0.00E+00	5.28E-08	0.00E+00
41	345600	3600	0	0.00E+00	3.60E-09	0.00E+00
42	349200	3600	0	0.00E+00	4.02E-08	0.00E+00
43	352800	3600	0	0.00E+00	8.10E-08	0.00E+00
44	356400	3600	0	0.00E+00	1.11E-07	0.00E+00

Table B-16: Release Category 8 Hourly Release Information (Continued)

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Heat Content (W)	Chemical Group (fraction of initial core inventory)		
				Xe	Elemental Iodine	Organic Iodide
45	360000	3600	0	0.00E+00	1.48E-07	0.00E+00
46	363600	3600	0	0.00E+00	1.93E-07	0.00E+00
47	367200	3600	0	0.00E+00	1.86E-07	0.00E+00
48	370800	3600	0	0.00E+00	4.28E-08	0.00E+00
49	374400	3600	0	0.00E+00	2.12E-08	0.00E+00
50	378000	3600	0	0.00E+00	2.04E-08	0.00E+00
51	381600	3600	0	0.00E+00	2.98E-08	0.00E+00
52	385200	3600	0	0.00E+00	2.92E-08	0.00E+00
53	388800	3600	0	0.00E+00	3.70E-08	0.00E+00
54	392400	3600	0	0.00E+00	4.22E-08	0.00E+00
55	396000	3600	0	0.00E+00	7.90E-08	0.00E+00
56	399600	3600	0	0.00E+00	1.28E-07	0.00E+00
57	403200	3600	0	0.00E+00	1.70E-07	0.00E+00
58	406800	3600	0	0.00E+00	2.01E-07	0.00E+00
59	410400	2880	0	0.00E+00	1.72E-07	0.00E+00
Total	191610	210870	0	6.39E-01	1.20E-03	6.98E-01

B.2.9 Summary of Release Category Results

The CDFs for each sequence of each PRA in a release category are summed to give the total release category frequency. Engineering judgment is used to select the sequence in a release category as the bounding accident to represent that release category's offsite consequences. The eight release categories are presented in Table B-17.

Table B-17: Release Categories and Associated Module Core Damage Frequency (per module-year)

Hazard	Release Category								
	1	2	3	4	5	6	7	8	Total
IE	7.0E-12	1.1E-12	1.7E-11	1.6E-10	1.4E-13	4.0E-11	4.6E-11	-	2.7E-10
LPSD	3.2E-15	<1E-15	2.3E-14	3.6E-13	<1E-15	2.1E-14	4.4E-14	-	4.5E-13
I-Fires	1.5E-11	-	-	4.9E-10	-	1.4E-10	1.9E-10	-	8.4E-10
I-Flood	-	-	-	-	-	3.0E-11	3.0E-11	-	6.0E-11
E-Flood	-	-	-	9.4E-10	-	9.0E-13	5.4E-13	-	9.4E-10
High Winds	-	-	-	8.6E-10	-	3.0E-12	2.6E-12	-	8.7E-10
Reactor Building Crane Failure	-	-	-	-	-	-	-	5.3E-07	5.3E-07
RC Total	2.2E-11	1.1E-12	1.7E-11	2.4E-09	1.4E-13	2.1E-10	2.7E-10	5.3E-07	5.3E-07
% of Total	4.1E-03	2.1E-04	3.2E-03	4.5E-01	2.6E-05	3.9E-02	5.1E-02	9.9E+01	100.00%

An entry of "<1E-15" is used to signify that the PRA contains cutsets that would contribute to the release category, but the cutsets fall below the PRA truncation value of 1E-15.

The release fraction to the environment, time to core damage (TCD), and release duration for each release category are shown in Table B-18. The contribution from Xe to offsite consequences is small. This is because Xe does not experience wet or dry deposition in the

environment. Therefore, a conservative assumption is made as a modeling convenience in the release category relating to steam generator tube failures (release category 5) that 100 percent of the Xe is released to the environment.

Table B-18: Release Fraction to the Environment, Time to Core Damage, and Release Duration for each Release Category Calculated by MELCOR

Group	Release Category							
	1	2	3	4	5	6	7	8
Xe	5.9E-03	3.7E-03	9.3E-01	5.4E-03	1.0E+00	5.5E-05	5.7E-05	6.4E-01
Cs	3.1E-03	6.7E-06	8.0E-01	2.0E-03	7.0E-02	2.0E-05	1.9E-05	0.0E+00
Ba	7.5E-05	1.6E-07	1.1E-02	6.5E-05	1.1E-03	6.3E-07	5.3E-07	0.0E+00
I	4.2E-03	6.1E-06	8.0E-01	2.7E-03	3.5E-01	1.9E-05	1.5E-05	2.4E-03
Te	3.4E-03	5.7E-06	7.9E-01	2.0E-03	1.1E-01	1.9E-05	1.8E-05	0.0E+00
Ru	2.0E-05	7.3E-08	9.4E-03	1.6E-05	1.9E-04	1.4E-07	1.1E-07	0.0E+00
Mo	5.7E-04	1.3E-06	2.1E-01	3.6E-04	5.6E-03	3.7E-06	3.6E-06	0.0E+00
Ce	6.1E-10	1.6E-12	2.2E-07	5.1E-10	5.0E-09	5.1E-12	4.1E-12	0.0E+00
La	6.3E-10	1.6E-12	2.2E-07	5.1E-10	5.1E-09	5.2E-12	4.1E-12	0.0E+00
TCD (hr)	5.8	10	3.3	9.4	17	13	9.4	53
Duration (hr)	96	96	26	96	56	96	96	59

Table B-19 summarizes the results for the Surry site for each release category discussed in Section B.2.1 to Section B.2.8 for a single module. The offsite consequences are mean MACCS results over a year of weather trials and the release category frequencies are point estimates. The offsite consequences per event are converted to per year by multiplying the values by the respective release category frequency. Summation of all release categories yields an estimated offsite dose risk of 1.3E-04 person-rem whole body dose per year for the NuScale design (utilized in Section 5.2). Summation of all release categories yields an estimated offsite economic risk (excluding the dollar value of public dose accrued) of 8.2E-02 dollars per year for the NuScale design (utilized in Section 5.3).

Table B-19: Frequency of Occurrence and Offsite Consequences for Each Release Category

Doses are whole body effective dose (ICRP60ED)

RC	Release Frequency (per year)	Offsite Dose per Event (person-rem/event)	Offsite Dose per Year (person-rem/year)	Offsite Economic Impact per Event ($)	Offsite Economic Impact ($/year)
1	2.2E-11	4.4E+01	9.7E-10	8.4E-01	1.8E-11
2	1.1E-12	3.9E+01	4.3E-11	1.8E-01	2.0E-13
3	1.7E-11	1.3E+06	2.2E-05	4.7E+09	8.0E-02
4	2.4E-09	3.0E+01	7.2E-08	0.0E+00	0.0E+00
5	1.4E-13	3.2E+05	4.5E-08	1.9E+08	2.7E-05
6	2.1E-10	2.8E+01	5.9E-09	0.0E+00	0.0E+00
7	2.7E-10	2.2E+01	5.9E-09	0.0E+00	0.0E+00
8	5.3E-07	2.0E+02	1.1E-04	3.7E+03	2.0E-03
Total	5.3E-07	-	1.3E-04	-	8.2E-02

B.3 Seismic

The seismic CDF is considered in Section 5.7 through an external events multiplier to the SAI as part of the calculation of the maximum benefit. NuScale is not preparing a seismic PRA for the design certification application as there is not a standard seismic hazard curve associated with the standard design process. NuScale is instead performing a seismic margins analysis which generates conditional core damage probabilities and not CDFs. In order to quantify a representative seismic risk for the purposes of this report, the seismic hazard curves for Surry and Peach Bottom are combined with the conditional core damage probabilities from the seismic margins analysis to generate CDFs for a NuScale plant located at those sites. However, the seismic core damage sequences do not match one-to-one with the sequences in the internal events, low-power and shutdown, internal flooding, internal fires, external floods, and high winds PRAs, and so the seismic risk is accounted for through an external events multiplier to the SAI using the total seismic CDF shown in Table B-20 in the interest of simplifying the analysis. This simplification is not expected to drastically affect the results. This seismic CDF analysis used a preliminary NuScale PRA model, however in the context of the results of this report the use of the preliminary PRA model compared to the final PRA model to establish seismic CDF produces a more conservative result and shouldn't significantly affect the identification of SAMDAs.

Table B-20: Seismic Core Damage Frequencies for Peach Bottom and Surry Sites

Site	Peach Bottom	Surry
Core Damage Frequency	2.0E-6 per year	3.2E-8 per year

The difference of nearly two orders of magnitude between the seismic CDFs is a consequence of their hazard curves, because results are proportional to initiator frequencies. The fragilities used in the NuScale seismic margin assessment use enveloping soil profiles that are independent of local geology. A site-specific PRA would reduce the seismic core damage frequencies for both sites.