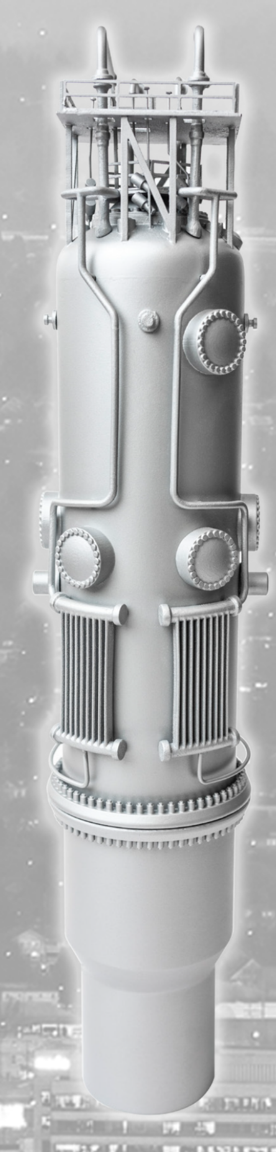
NUSCALE™
Power for all humankind

NuScale Standard Plant
Design Certification Application

Applicant's Environmental Report - Standard Design Certification



PART 3

Revision 2
October 2018

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES

LIST OF FIGURES

Applicant's Environmental Report - Standard Design Certification

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 off-site property damage costs
AOE	averted occupational exposure
AOSC	averted on-site 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
FSAR	Final Safety Analysis Report
FWIV	feedwater isolation valve
HVAC	heating ventilation and air conditioning
IE	internal events
LOCA	loss-of-coolant accident
LOOP	loss of off-site 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

Table 2-1: Acronyms and Abbreviations (Continued)

Term	Definition
SAI	severe accident impact
SAMA	severe accident mitigation alternative
SAMDA	severe accident mitigation design alternative
SBO	station blackout
SCWS	site cooling water system
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
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 severe accident risk. Or in other words, the monetary value attributed to eliminating all severe accident risk from the NuScale design.
PV_{CD}	present value of the cost of cleanup and decontamination
PV_{RP}	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 Reference 8.1-1.

The process to identify SAMDA candidates for the NuScale design used generic industry SAMDAs described in Reference 8.1-1. Top cutsets and candidate risk significant events from NuScale probabilistic risk assessment (PRA) models, described in Final Safety Analysis Report (FSAR) Section 19.1, 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 described in Section B.2 to assess off-site consequences of severe accidents. Key assumptions of these off-site consequence analyses are given in Appendix B. The calculated off-site 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 off-site consequence modeling. The calculation of the maximum benefit accounts for the effect of seismic events and multiple NuScale Power Modules (NPMs). A series of sensitivity studies on the maximum benefit (including the use of Peach Bottom instead of Surry data to assess off-site 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) described in Table 14 of Reference 8.1-1 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) described in Table 13 of Reference 8.1-1 were reviewed, but determined 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 for plants with 1 to 12 NPMs are identified through the use of PRA insights. The PRA, as discussed in Chapter 19 of the NuScale FSAR, 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 Reference 8.1-1 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 that meet the NuScale risk significance criteria
- 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 candidate risk significant events from the following PRAs are used as the basis for SAMDA identification:

- full power internal events (FSAR Tables 19.1-18, 19.1-20, 19.1-26, and 19.1-27)
- internal fires (FSAR Tables 19.1-45 and 19.1-64)
- internal flooding (FSAR Tables 19.1-53 and 19.1-64)
- low power and shutdown (LPSD) (FSAR Tables 19.1-69 and 19.1-70)
- external flooding (FSAR Tables 19.1-57 and 19.1-64)
- high winds (FSAR Tables 19.1-62, 19.1-63, and 19.1-64)

Insights into the dominant contributors to seismic risk were obtained by calculating seismic CDF cutsets for an 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-30.

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). Key insights from the Level 1 PRA and Level 2 PRA evaluations are provided in FSAR Tables 19.1-23 and 19.1-32, respectively.

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 candidate risk significant basic events are numbered from 154 to 203. These SAMDAs are shown in Appendix A.

Note that the terminology "risk significant basic events" is used throughout the text for simplicity whereas the PRA only identifies "candidate" risk significant basic events. Individual basic events are also tied to respective components and systems in Table 4-2.

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. 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 determined to be a dropped NPM event due to 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 NPM drop portion of the LPSD PRA. The NPM drop portion of the LPSD PRA is modeled as a crane failure initiating event combined with a probability of the NPM remaining upright following the drop. A Reactor Building crane evaluation was performed to establish the frequency of the NPM drop event, as discussed in FSAR Section 19.1.6.1.2. 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
Component: Containment Isolation Valves									
Common cause failure (CCF) of two of two containment evacuation system (CES) containment isolation valves fail to close	SAMDA 183	CNTS	H-W	-	-	-	X[9]	-	X[8]
			E-Flood	-	-	-	X[9]	-	X[7]
CCF of two of two chemical and volume control system (CVCS) discharge containment isolation valves fail to close	SAMDA 184	CNTS	IE	-	-	-	X[4]	X[1]	X[3]
			I-Fires	-	-	-	X[9]	-	X[5]
			H-W	-	-	-	X[9]	-	X[8]
			E-Flood	-	-	-	X[9]	-	X[7]
			LPSD	-	-	-	X[11]	-	X[10]
CVCS discharge containment isolation valve A fails to close	SAMDA 184	CNTS	IE	-	-	-	X[4]	-	-
			I-Fires	-	-	-	X[9]	-	-
CVCS discharge containment isolation valve B fails to close	SAMDA 184	CNTS	IE	-	-	-	X[4]	-	-
			I-Fires	-	-	-	X[9]	-	-
CCF of two of two condensate and feedwater system containment isolation valves fail to close	SAMDA 179, SAMDA 180	CNTS	I-Fires	-	-	-	-	X[5]	-
CCF of two of two main steam system containment isolation valves fail to close	SAMDA 181, SAMDA 182	CNTS	I-Fires	-	-	-	-	X[5]	-
Component: Makeup Valves									
CVCS makeup combining valve fails to open	SAMDA 158	CVCS	IE	-	-	-	-	X[1]	-
			I-Fires	-	-	-	-	X[5]	-
			LPSD	-	-	-	-	X[10]	-
Demineralized water system (DWS) north Reactor Building CVCS pump isolation valve fails to open	SAMDA 160	CVCS	IE	-	-	-	-	X[1]	-
			I-Fires	-	-	-	-	X[5]	-
			LPSD	-	-	-	-	X[10]	-
Containment flooding and drain system (CFDS) NPM makeup valve fails to open	SAMDA 156	CFDS	IE	-	-	-	-	-	X[3]
			LPSD	-	-	-	-	-	X[10]
CFDS containment isolation valve A fails to open	SAMDA 156	CNTS	IE	-	-	-	-	-	X[3]
			LPSD	-	-	-	-	-	X[10]

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
CFDS containment isolation valve B fails to open	SAMDA 156	CNTS	IE	-	-	-	-	-	X[3]
			LPSD	-	-	-	-	-	X[10]
Component: Control Rods									
Given actuation, at least three of 16 rods fail to insert	SAMDA 130 SAMDA 185	CRDS	IE	-	-	-	-	X[1]	-
			I-Fires	-	-	-	-	X[5]	-
			I-Flood	-	-	-	-	X[6]	-
Component: DHRS Actuation Valves									
CCF of four of four decay heat removal system (DHRS) actuation valves fail to open	SAMDA 172, SAMDA 173, SAMDA 177 SAMDA 178	DHRS	IE	-	-	-	-	X[1]	-
			I-Flood	-	-	X[9]	-	X[6]	-
			LPSD	-	-	-	-	X[10]	-
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[9]	-	-	-
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[9]	-	-	-
Actuation valve A for DHRS train one fails to open	SAMDA 173, SAMDA 177 SAMDA 178	DHRS	I-Flood	-	-	X[9]	-	-	-
Actuation valve B for DHRS train one fails to open	SAMDA 173, SAMDA 177 SAMDA 178	DHRS	I-Flood	-	-	X[9]	-	-	-
Actuation valve A for DHRS train two fails to open	SAMDA 173, SAMDA 177 SAMDA 178	DHRS	I-Flood	-	-	X[9]	-	-	-
Actuation valve B for DHRS train two fails to open	SAMDA 173, SAMDA 177 SAMDA 178	DHRS	I-Flood	-	-	X[9]	-	-	-

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
Component: DHRS Heat Exchangers									
CCF of two of two DHRS heat exchangers plugging	SAMDA 174, SAMDA 175 SAMDA 178	DHRS	I-Flood	-	-	-	-	X[6]	-
Component: ECCS Reactor Vent Valves									
CCF of three of three emergency core cooling system (ECCS) reactor vent valves (RVVs) fail to open	SAMDA 165, SAMDA 166, SAMDA 171	ECCS	IE	-	-	X[2]	-	X[1]	-
			I-Fires	-	-	X[9]	-	X[5]	X[5]
			I-Flood	-	-	-	-	X[6]	-
			H-W	-	-	X[9]	X[9]	X[8]	X[8]
			E-Flood	-	-	X[9]	X[9]	X[7]	X[7]
			LPSD	-	-	X[11]	-	X[10]	-
ECCS RVV A fails to open	SAMDA 171	ECCS	I-Fires	-	-	X[9]	-	-	-
			E-Flood	-	-	X[9]	X[9]	-	-
ECCS RVV A passive actuation to open valve fails	SAMDA 171	ECCS	IE	-	-	X[2]	-	-	-
			I-Fires	-	-	X[9]	-	-	-
			H-W	-	-	X[9]	X[9]	-	-
			E-Flood	-	-	X[9]	X[9]	-	-
ECCS RVV B fails to open	SAMDA 171	ECCS	I-Fires	-	-	X[9]	-	-	-
			E-Flood	-	-	X[9]	X[9]	-	-
ECCS RVV B passive actuation to open valve fails	SAMDA 171	ECCS	IE	-	-	X[2]	-	-	-
			I-Fires	-	-	X[9]	-	-	-
			H-W	-	-	X[9]	X[9]	-	-
			E-Flood	-	-	X[9]	X[9]	-	-
ECCS RVV C fails to open	SAMDA 171	ECCS	I-Fires	-	-	X[9]	-	-	-
			E-Flood	-	-	X[9]	X[9]	-	-

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 RVV C passive actuation to open valve fails	SAMDA 171	ECCS	IE	-	-	X[2]	-	-	-
			I-Fires	-	-	X[9]	-	-	-
			H-W	-	-	X[9]	X[9]	-	-
			E-Flood	-	-	X[9]	X[9]	-	-
Component: ECCS Reactor Recirculation Valves									
CCF of two of two ECCS reactor recirculation valves (RRVs) fail to open	SAMDA 163, SAMDA 164, SAMDA 171	ECCS	IE	-	-	X[2]	-	X[1]	-
			I-Fires	-	-	X[9]	X[9]	X[5]	X[5]
			I-Flood	-	-	X[9]	-	X[6]	-
			H-W	-	-	X[9]	X[9]	X[8]	X[8]
			E-Flood	-	-	X[9]	X[9]	X[7]	X[7]
			LPSD	-	-	X[11]	-	X[10]	-
ECCS RRV A fails to open	SAMDA 171	ECCS	IE	-	-	X[2]	-	-	-
			I-Fires	-	-	X[9]	X[9]	-	-
			I-Flood	-	-	X[9]	-	-	-
			H-W	-	-	X[9]	X[9]	-	-
			E-Flood	-	-	X[9]	X[9]	-	-
ECCS RRV A passive actuation to open valve fails	SAMDA 171	ECCS	IE	-	-	X[2]	-	-	-
			I-Fires	-	-	X[9]	X[9]	-	X[5]
			I-Flood	-	-	X[9]	-	X[6]	-
			H-W	-	-	X[9]	X[9]	X[8]	-
			E-Flood	-	-	X[9]	X[9]	X[7]	X[7]
ECCS RRV B fails to open	SAMDA 171	ECCS	IE	-	-	X[2]	-	-	-
			I-Fires	-	-	X[9]	X[9]	-	-
			I-Flood	-	-	X[9]	-	-	-
			H-W	-	-	X[9]	X[9]	-	-
			E-Flood	-	-	X[9]	X[9]	-	-

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 RRV B passive actuation to open valve fails	SAMDA 171	ECCS	IE	-	-	X[2]	-	-	-
			I-Fires	-	-	X[9]	X[9]	-	X[5]
			I-Flood	-	-	X[9]	-	X[6]	-
			H-W	-	-	X[9]	X[9]	X[8]	-
			E-Flood	-	-	X[9]	X[9]	X[7]	X[7]
Component: ECCS Reactor Vent Trip Valves									
CCF of three of three ECCS RVV trip valves fail to open	SAMDA 167, SAMDA 168, SAMDA 196	ECCS	I-Fires	-	-	X[9]	-	-	-
ECCS RVV trip valve A fails to open	SAMDA 168, SAMDA 196	ECCS	I-Fires	-	-	X[9]	-	-	-
ECCS RVV trip valve A fails due to hot short	SAMDA 168, SAMDA 190, SAMDA 196	ECCS	I-Fires	-	-	X[9]	-	X[5]	-
ECCS RVV trip valve B fails due to hot short	SAMDA 168, SAMDA 190, SAMDA 196	ECCS	I-Fires	-	-	-	-	X[5]	-
Component: ECCS Reactor Recirculation Trip Valves									
CCF of two of two ECCS RRV trip valves fail to open	SAMDA 167, SAMDA 168, SAMDA 196	ECCS	I-Fires	-	-	-	-	-	X[5]
			I-Flood	-	-	-	-	X[6]	-
			H-W	-	-	-	-	X[8]	-
			E-Flood	-	-	-	-	X[7]	-
Component: Auxiliary AC Power Supply									
Combustion turbine generator fails to start	SAMDA 13, SAMDA 154, SAMDA 155	BPSS	IE	-	-	X[2]	-	-	-
Combustion turbine generator fails to run (hours two through 48)	SAMDA 13, SAMDA 154, SAMDA 155	BPSS	IE	-	-	X[2]	-	-	-
Component: Actuation and Priority Logic									
CCF of four of four actuation and priority logic (APL) modules in DHRS actuation valves	SAMDA 196	MPS	I-Flood	-	-	-	-	X[6]	-

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 discharge isolation valves	SAMDA 196	MPS	IE	-	-	-	-	-	X[3]
			I-Fires	-	-	-	-	-	X[5]
CCF of two of two APL modules in top branch reactor trip breakers	SAMDA 196	MPS	I-Fires	-	-	-	-	X[5]	-
			I-Flood	-	-	-	-	X[6]	-
CCF of two of two APL modules in bottom branch reactor trip breakers	SAMDA 196	MPS	I-Fires	-	-	-	-	X[5]	-
			I-Flood	-	-	-	-	X[6]	-
Component: Equipment Interface Modules									
Division I engineered safety features actuation system (ESFAS) equipment interface module (EIM) one fails to operate	SAMDA 197	MPS	IE	-	-	-	-	-	X[3]
			LPSD	-	-	-	-	-	X[10]
Division I ESFAS EIM two fails to operate	SAMDA 197	MPS	IE	-	-	-	-	-	X[3]
			LPSD	-	-	-	-	-	X[10]
Division II ESFAS EIM one fails to operate	SAMDA 197	MPS	IE	-	-	-	-	-	X[3]
Division II ESFAS EIM two fails to operate	SAMDA 197	MPS	IE	-	-	-	-	-	X[3]
			LPSD	-	-	-	-	-	X[10]
Component: Manual Switches									
Manual division I containment isolation override switch fails to close	SAMDA 198	MPS	IE	-	-	-	-	-	X[3]
Manual division I ESFAS nonsafety enable switch fails to close	SAMDA 198	MPS	IE	-	-	-	-	-	X[3]
Manual division II containment isolation override switch fails to close	SAMDA 198	MPS	IE	-	-	-	-	-	X[3]
Manual division II ESFAS nonsafety enable switch fails to close	SAMDA 198	MPS	IE	-	-	-	-	-	X[3]
Component: Level Process Logic									
CCF of three of four reactor pressure vessel (RPV) level process logic elements	SAMDA 195	MPS	H-W	-	-	-	-	X[8]	-
			E-Flood	-	-	-	-	X[7]	X[7]
Component: Scheduling and Voting Module									
CCF of two of three division I ESFAS scheduling and voting modules	SAMDA 199	MPS	I-Fires	-	-	-	-	X[5]	-
CCF of two of three division II ESFAS scheduling and voting modules	SAMDA 199	MPS	I-Fires	-	-	-	-	X[5]	-

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
Component: Reactor Safety Valves									
CCF two of two reactor coolant system (RCS) reactor safety valves (RSVs) fail to open	SAMDA 191, SAMDA 192, SAMDA 193	RCS	IE	-	-	X[2]	-	X[1]	-
			I-Fires	-	-	X[9]	X[9]	X[5]	X[5]
			I-Flood	-	-	X[9]	-	X[6]	-
			LPSD	-	-	-	-	X[10]	-
RCS RSV A fails to open	SAMDA 193	RCS	IE	-	-	X[2]	-		-
			I-Fires	-	-	X[9]	X[9]	-	-
			I-Flood	-	-	X[9]	-	X[6]	-
RCS RSV B fails to open	SAMDA 193	RCS	I-Fires	-	-	X[9]	X[9]	-	-
RCS RSV A fails to reclose	SAMDA 193	RCS	IE	-	-	X[2]	-	X[1]	-
			I-Fires	-	-	X[9]	X[9]	X[5]	-
			I-Flood	-	-	X[9]	-	X[6]	-
			LPSD	-	-	-	-	X[10]	-
Component: Level Sensors									
CCF of three of four RPV level sensors fail to operate on demand	SAMDA 194	MPS	H-W	-	-	-	-	X[8]	-
			E-Flood	-	-	-	-	X[7]	X[7]
Component: Reactor Trip Breakers									
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[6]	-
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[6]	-
Human Actions									
Operator fails to initiate CFDS injection	SAMDA 157	CFDS	IE	-	-	-	X[4]	X[1]	X[3]
			LPSD	-	-	-	X[11]	-	X[10]

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
Operator fails to initiate CVCS injection	SAMDA 159	CVCS	IE	-	-	-	-	X[1]	X[3]
			I-Fires	-	-	-	-	X[5]	-
			LPSD	-	-	X[11]	-	X[10]	-
System: Highly Reliable DC Power									
Initiator is CCF of four of four highly reliable DC power system (EDSS) buses to operate	SAMDA 5	EDSS	IE	-	-	-	-	X[1]	-
			LPSD	-	-	-		X[10]	
System: Ultimate Heat Sink									
Heat transfer to reactor pool fails	SAMDA 169, SAMDA 170	UHS	IE	X[2]	-	-	-	-	-
			I-Fires	X[9]	X[9]	-	-	-	-
			I-Flood	X[9]	-	-	-	X[6]	-
			H-W	X[9]	-	-	-	-	-
			E-Flood	X[9]	-	-	-	X[7]	-

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.

[1] FSAR Table 19.1-18
[2] FSAR Table 19.1-20
[3] FSAR Table 19.1-26
[4] FSAR Table 19.1-27
[5] FSAR Table 19.1-45
[6] FSAR Table 19.1-53
[7] FSAR Table 19.1-57
[8] FSAR Tables 19.1-62 and 19.1-63
[9] FSAR Table 19.1-64
[10] FSAR Table 19.1-69
[11] FSAR Table 19.1-70

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. Table 4-2 provides a comprehensive list of the SAMDAs generated for the NuScale design, the PRA basic events that these SAMDAs mitigate, and the information used to identify the PRA basic events as significant to risk in the NuScale design.

4.1.1 Emergency Core Cooling System

The emergency core cooling system (ECCS) is described in FSAR Section 6.3. The 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 FSAR Figure 6.3-2, reprinted here as 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 FSAR Figure 6.3-3, reprinted here as 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.

As shown in Table 4-2, there are 18 basic events from Chapter 19 PRAs that are related to ECCS and considered risk significant or are in the top cutsets. These events include the common cause failure (CCF) of the RRVs, the CCF of the 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 (FSAR Figure 6.3-2)

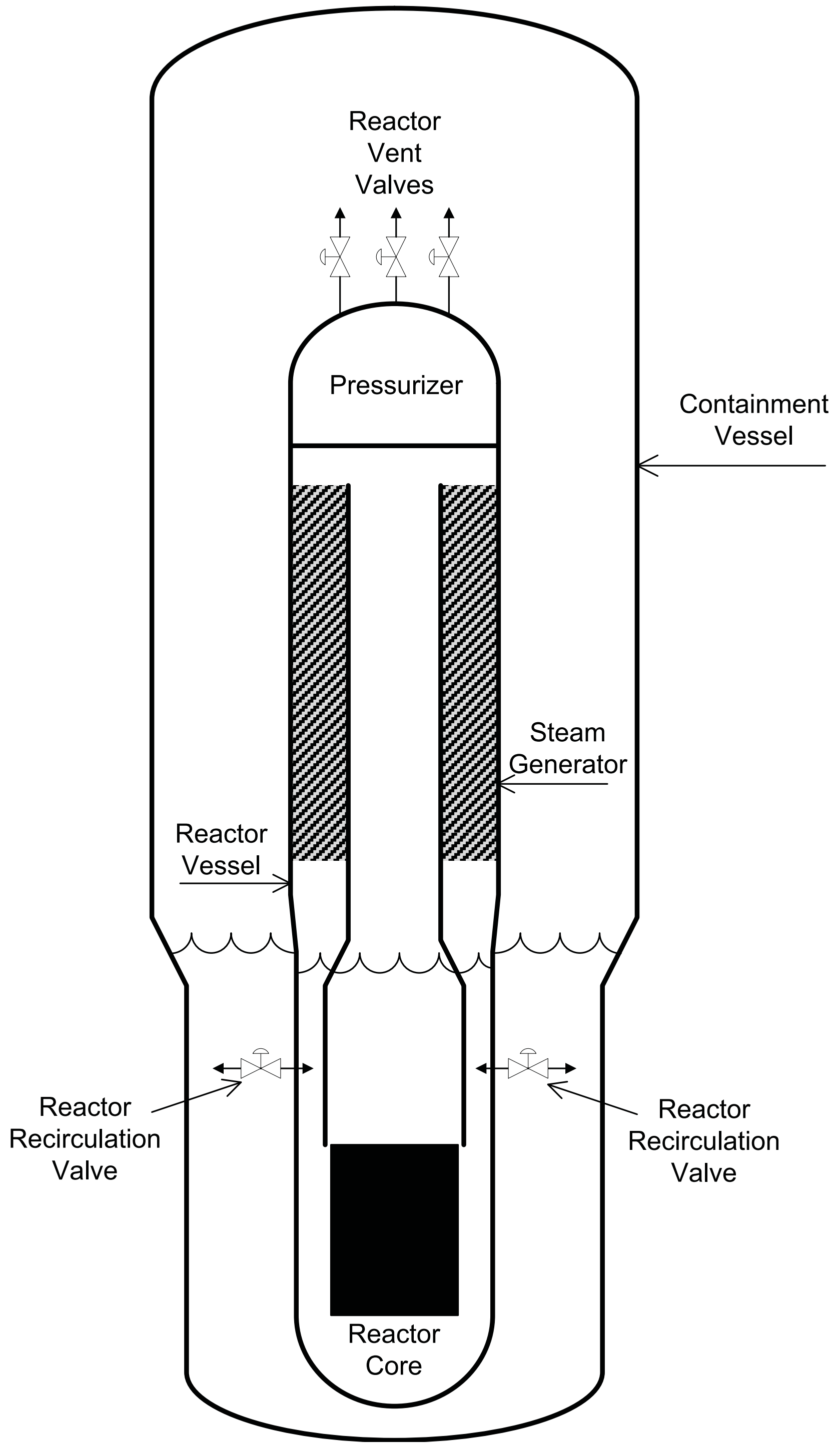
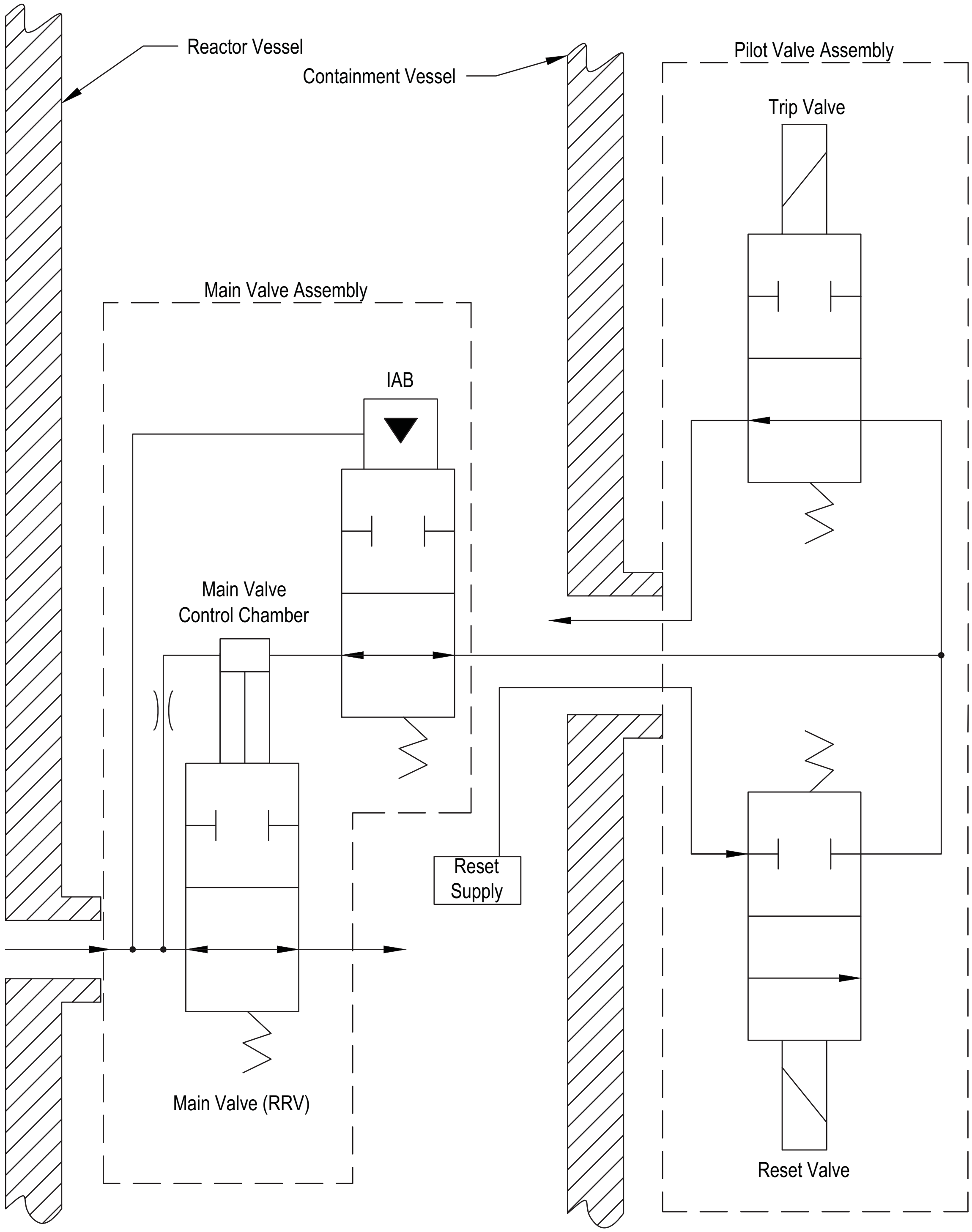


Figure 4-2: ECCS Valve Design (FSAR Figure 6.3-3)



The context of these common cause ECCS valve failures is also important when considering the design alternatives. There are loss of all AC power sequences that are on a success path and decay heat removal is being accomplished by other systems. Then, a partial actuation of ECCS 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 all AC power when ECCS is not required, one existing SAMDA is considered and one new SAMDA is 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

Containment isolation is described in FSAR Section 6.2.4. There are six basic events from the PRA related to containment isolation from Table 4-2 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 four PRA basic events shown in Table 4-2 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, and the CCF of both CVCS discharge line containment isolation valves to close.

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

The NuScale PRA demonstrated that 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.

4.1.2.2 Containment Isolation Valves Fail to Open

There are two PRA basic events shown in Table 4-2 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

The control rod drive system (CRDS) is described in FSAR Section 4.6. There is one PRA basic event shown in Table 4-2 related to the CRDS that 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 (in addition to the CVCS) 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

The reactor trip system (RTS) is described in FSAR Section 7.0.4.1.2. There are two PRA basic events shown in Table 4-2 related to the RTS that 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 failure of the RTS to de-energize the CRDS 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 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

The decay heat removal system (DHRS) is described in FSAR Section 5.4.3.2. There are 10 PRA basic events shown in Table 4-2 related to the DHRS that 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 (FWIVs) or main steam isolation valves (MSIVs) 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

The RCS is described in FSAR Section 5.1. There are four PRA basic events shown in Table 4-2 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 RSVs are described in FSAR Section 5.2.2.4.1.

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

The backup power supply system (BPSS) is described in FSAR Section 8.3.1.1.2. There are two PRA basic events shown in Table 4-2 related to the 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, and the failure of the CTG to run during hours 2 through 48 of service.

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

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

4.1.8 Containment Flooding and Drain System

The CFDS is described in FSAR Section 9.3.6.2.2. There are two PRA basic events shown in Table 4-2 related to the CFDS that are considered risk significant or are in the top cutsets. The first basic event is the failure of CFDS NPM makeup valve to open when demanded to allow the CFDS to supply coolant from the common (to six NPMs) system to the containment of one NPM. 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

The CVCS is described in FSAR Section 9.3.4.2. There are three PRA basic events shown in Table 4-2 related to the CVCS that 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 providing an alternate source of inventory for the CVCS, which reduces the impact of the failure of the CVCS makeup combining valve to open to the DWS, limiting the inventory available to CVCS.

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 NPM 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

The module protection system (MPS) is described in FSAR Section 7.0.4.1. There are 16 PRA basic events shown in Table 4-2 that apply to the 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 features 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 RPV 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 RPV 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 RPV 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 NPM transport, leading to a dropped NPM, contribute the greatest proportion of any equipment failure to the maximum benefit of risk elimination in the NuScale design, contributing over 99 percent of the maximum benefit. Four SAMDAs are proposed to mitigate Reactor Building crane failures. The Reactor Building crane evaluation, discussed in FSAR Section 19.1.6.1.2, 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 NPM 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

The internal fire PRA is described in FSAR Section 19.1.5.2. 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.

The internal flooding PRA is described in FSAR Section 19.1.5.3. 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.

The high-wind PRA is described in FSAR Section 19.1.5.5. High wind events are modeled in the PRA as causing a loss of off-site power. SAMDAs 1 through 24, 154, and 155 suggest design alternatives to mitigate a loss of off-site power, and therefore no additional SAMDAs are suggested to address high winds.

The external flooding PRA is described in FSAR Section 19.1.5.4. The external flooding PRA is similar to high winds, in that an external flood is modeled as causing an extended loss of off-site power. Damage to equipment is similar to that of internal flooding, therefore no new SAMDAs are postulated that apply to external flooding.

The seismic risk evaluation, discussed in FSAR Section 19.1.5.1, determined that 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 LPSD PRA, described in FSAR Section 19.1.6, 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 LPSD PRA. The LPSD PRA also considers Reactor Building crane failures, which are discussed in Section 4.1.11. Therefore, any SAMDAs that apply to Reactor Building crane failures also apply to the LPSD PRA.

5.0 Maximum Benefit

The base case calculation is discussed throughout this section. Sensitivity cases are described in Section 5.8. A key equation for the initial cost-benefit assessment of implementing a particular SAMDA is adapted from Section 4 of Reference 8.1-1 as follows:

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

where,

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

AOC = present value of averted off-site 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), per Section B.2 of Reference 8.1-2. This time value of money is different than the update of the economic data in the Surry site file from 2005 to 2017 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. The dose and economic costs 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 accounts for the costs saved over the entire lifetime of the plant from avoiding an accident, and expresses those dollars in present value.

As discussed in Section B.2 of Reference 8.1-2, 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. Reference 8.1-2 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). Reference 8.1-2 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 Reference 8.1-2, and t_f to be 60 years yields C=14.1 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)) = \$5,100 per person-rem (per Reference 8.1-4, Table 3)

C = present value factor (years) = 14.1 years

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 calculated by multiplying the population dose (PD) for each release category (RC) by the frequency of the RC and summing the result of this product across all the RCs.

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

The population dose for each RC is 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 are documented in Table 5-1, which describes the population dose associated with each RC, the frequency associated with each RC, and the total APD of 2.42E-04 person-rem per year. For a more detailed description of MACCS modeling, see Appendix B Section B.1 and Section B.2.

A short description of the SAMDA-specific RCs 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 RCs, see Appendix B Section B.2. Note that these RCs 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, described in FSAR Section 19.1.4.2.1.4, as a more precise treatment is desired for the purposes of this report. Also note that RC associated information (and Table 5-1 results) is based on internal events (FSAR Section 19.1.4), LPSD (FSAR Section 19.1.6), internal flooding (FSAR Section 19.1.5.3), internal fires (FSAR Section 19.1.5.2), external floods (FSAR Section 19.1.5.4), and high winds PRAs (FSAR Section 19.1.5.5). The frequencies corresponding to each RC are shown in Table B-25.

RC 1: CVCS loss-of-coolant accident (LOCA) Inside Containment

RC 2: Pipe Break Outside Containment, Isolated

RC 3: CVCS Pipe Break 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 NPM During Transport

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

RC	Release Frequency (per year)	Off-site Dose per Event (person-rem/event)	Off-site Dose per Year (person-rem/year)
1	2.15E-11	2.62E+01	5.64E-10
2	1.06E-12	2.09E+01	2.22E-11
3	1.79E-11	1.02E+06	1.82E-05
4	2.44E-09	1.87E+01	4.57E-08
5	1.39E-13	2.15E+05	2.98E-08
6	2.06E-10	1.35E+01	2.78E-09
7	2.70E-10	2.01E+01	5.42E-09
8	1.05E-06	2.13E+02	2.24E-04
Total	1.05E-06	-	2.42E-04

Utilizing these values and a present value factor of 14.1 years yields an APE of $17.40.

$$APE = 2.42E-04\frac{person\text{-}rem}{year} \times \frac{\$5{,}100}{person\text{-}rem} \times 14.1\ years = \$17.4 \qquad \text{Equation 5-6}$$

5.3 Averted Off-Site Costs

The averted off-site property damage costs (AOC) are calculated using an equation adapted from Reference 8.1-2, Equation 5-7.

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

The off-site economic impact for each RC is calculated with MELCOR-derived release information and a MACCS model (see Appendix B), using site data from the Surry SOARCA analysis (Reference 8.1-28) with a 50-mile radius, which was developed to support this report. The results of these calculations are documented in Table 5-2, which describes the off-site economic impact per event associated with each RC, the frequency associated with each RC, and the annual monetary value of off-site economic impact associated with each RC. The total monetary value of off-site economic impact is determined by a summation of the off-site economic impact calculated for each RC and calculated to be 5.37E-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 Off-site Consequences for Each Release Category

RC	Release Frequency (per year)	Off-site Economic Impact per Event ($)	Off-site Economic Impact ($/year)
1	2.15E-11	0.00E+00	0.00E+00
2	1.06E-12	0.00E+00	0.00E+00
3	1.79E-11	2.72E+09	4.87E-02
4	2.44E-09	0.0E+00	0.0E+00
5	1.39E-13	8.42E+07	1.17E-05
6	2.06E-10	0.0E+00	0.0E+00
7	2.70E-10	0.0E+00	0.0E+00
8	1.05E-06	4.77E+03	5.01E-03
Total	1.05E-06	-	5.37E-02

Multiplying by the present value factor (14.1 years) gives an AOC value of:

$$AOC = 14.1\ years \times 5.37E-02\left(\frac{\$}{year}\right) = \$0.755 \qquad \text{Equation 5-8}$$

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 Reference 8.1-2.

5.4.1 Short-Term Dose to Personnel

The short-term dose to personnel is 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)) = $5,100 per person-rem (per Reference 8.1-4, Table 3)

CDF = Core damage frequency (events per year) = 1.05E-06 events per year

D_{IO} = immediate occupational dose, (person-rem per event) = 3,300 person-rem per event (best estimate value based on Three Mile Island and Chernobyl data per Section 5.7.3.1 of Reference 8.1-2)

C = present value factor (years) = 14.1 years

Therefore,

$$W_{IO} = \frac{\$5{,}100}{\textit{person-rem}} \times 1.05\text{E-}06\frac{\textit{events}}{\textit{year}} \times 3300\frac{\textit{person-rem}}{\textit{event}} \times 14.1\ \textit{years} = \$249 \qquad \text{Equation 5-10}$$

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 is derived from Reference 8.1-2 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 $\left(\frac{\$}{\textit{person-rem}}\right)$ = \$5,100 per person-rem (per Reference 8.1-4, Table 3)

CDF = core damage frequency (events per year) = 1.05E-06 events per year

D_{LTO} = long term occupational dose, (person-rem per event) = 20,000 person-rem per event (per Reference 8.1-2 best estimate)

C = present value factor (years) = 14.1 years

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

m = on-site cleanup period (years) = 10 years (per Reference 8.1-2)

Therefore,

$$W_{LTO} = \frac{\$5100}{\textit{person-rem}} \times 1.05\text{E-}06\frac{\textit{events}}{\textit{year}} \times 20{,}000\frac{\textit{person-rem}}{\textit{event}} \times \qquad \text{Equation 5-12}$$

$$14.1\ \textit{years} \times \frac{1 - e^{-\frac{0.07}{\textit{year}} \times 10\ \textit{years}}}{\frac{0.07}{\text{year}} \times 10\ \textit{years}} = \$1{,}090$$

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:

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

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

- $W_{IO} = \$249$
- $W_{LTO} = \$1.090$

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

$$W_O = \$249 + \$1{,}090 = \$1{,}339 = \$1{,}340 \qquad \text{Equation 5-14}$$

5.5 Averted On-Site Costs

The averted on-site costs (AOSC) associated with on-site 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 NPM 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 Reference 8.1-2.

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

5.5.1 Cleanup and Decontamination Costs

A cleanup period of 10 years is assumed per Reference 8.1-2. Reference 8.1-2 estimated the value associated with total cleanup and decontamination costs as \$1.5E+09 in 1993. This value of \$1.5E+09 is the sum of the estimated cleanup and decontamination cost of \$1.1E+09 and the estimated decommissioning cost of \$4E+08 (Reference 8.1-2, page 5.42-5.43). This value is updated to 2017 dollars by multiplying by the ratio of the September 2017 consumer price index (CPI) of 246.819 to the January 1993 CPI of 142.6 (Reference 8.1-10, Table 24). The net present value of a single event is determined through Equation 5-15:

$$PV_{CD} = \left[\frac{C_{CD} \times \left(\frac{(2017\ CPI)}{(1993\ CPI)}\right)}{m}\right] \times \left[\frac{1 - e^{-rm}}{r}\right] \qquad \text{Equation 5-15}$$

where,

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

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

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

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

Therefore,

$$PV_{CD} = \frac{\$1.5E{+}09 \times \left(\frac{246.819}{142.6}\right)}{10\ years} \times \left[\frac{1 - e^{-\frac{0.07}{year} \times 10\ years}}{\frac{0.07}{year}}\right] = \$1.87E + 09 \qquad \text{Equation 5-16}$$

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 ($) = $1.87E+09, and

C = present value factor (years) = 14.1 years.

Therefore,

$$U_{CD} = \$1.87E + 09 \times 14.1\ years = 2.63E{+}10\$ - years \qquad \text{Equation 5-18}$$

5.5.2 Replacement Power Cost

Following a severe accident it is assumed the plant's owner is responsible for replacement power costs. Replacement power costs are calculated per the guidance of Reference 8.1-2, 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 is 128.6 and the preliminary 2017 PPI for electric power is 210.9 (Reference 8.1-5). Multiplying the 1993 replacement power cost estimate by the ratio of the 2017 to 1993 PPI is a reasonable means of estimating the 2017 cost of replacement power associated with a severe accident. The 2017 PPI is the most recent annual PPI available at the time of this report's creation.

Note further that the value from Reference 8.1-2 is based on old data that assumes a roughly 60 percent capacity factor for a nuclear power plant. The NuScale design is assumed to achieve a capacity factor of 100 percent, to bound the expected range of capacity factors. 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 ($CF_{NuScale}$) to the previously assumed 60 percent capacity factor (CF_{0184}) is a reasonable estimate.

Equation 5-19 is derived from Reference 8.1-2 and adjusts the reference 910 MWe plant (MWe_{0184}) to the 50 MWe NPM (MWe). Equation 5-19 approximates the average value of net present value replacement power costs for a single event:

$$PV_{RP} = \left[\frac{B \times \left(\frac{CF_{NuScale}}{CF_{0184}}\right) \times \left(\frac{2017\ PPI}{1993\ PPI}\right) \times \left(\frac{MWe_{NuScale}}{MWe_{0184}}\right)}{r}\right] \times [1 - e^{-rt_f}]^2 \qquad \text{Equation 5-19}$$

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,

$$PV_{RP} = \left[\frac{1.2E + 08\frac{\$}{year} \times \left(\frac{1.0}{0.60}\right) \times \left(\frac{210.9}{128.6}\right) \times \left(\frac{50\ MWe}{910\ MWe}\right)}{\frac{0.07}{year}}\right] \times \qquad \text{Equation 5-20}$$

$$\left(1 - e^{-\frac{0.07}{year} \times 60\ years}\right)^2 = \$2.50E + 08$$

The net present value of replacement power over the life of the facility can be determined with the following equation from Reference 8.1-2 (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,

$$U_{RP} = \frac{\$2.50E+08}{\frac{0.07}{year}} \times \left(1 - e^{-\frac{0.07}{year} \times 60\ years}\right)^2 = 3.46E+09\$\text{-}years \qquad \text{Equation 5-22}$$

5.5.3 Total Averted On-Site 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} = 2.63E+10 $-years
- U_{RP} = 3.46E+09 $-years
- CDF = 1.05E-06 per year

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

$$W_{OSC} = ((2.63E+10\ \$years) + (3.46E+09\ \$years)) \times \frac{1.05E-06}{year} \qquad \text{Equation 5-24}$$
$$= \$31,300$$

5.6 Single NuScale Power Module Severe Accident Impact

Calculations throughout Section 5 are performed for all RCs individually and then summed together. 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 RC and summed together to calculate the SAI for the base case. The total single NPM SAI calculates the monetary value attributed to the consequences of an accident involving a single NPM at a 12 NPM plant, considering internal events, low power shutdown, internal floods, internal fires, external floods, high winds, and dropped NPM events as $32,700.

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

RC	APE (public exposure, $)	AOC (off-site cost, $)	AOE (on-site exposure, $)	AOSC (on-site cost, $)	SAI ($)	Percent of total SAI
1	4.05E-05	0.00E+00	2.73E-02	6.40E-01	6.67E-01	2.04E-03
2	1.59E-06	0.00E+00	1.35E-03	3.16E-02	3.29E-02	1.01E-04
3	1.31E+00	6.85E-01	2.27E-02	5.32E-01	2.55E+00	7.80E-03
4	3.28E-03	0.00E+00	3.10E+00	7.27E+01	7.58E+01	2.32E-01
5	2.14E-03	1.64E-04	1.76E-04	4.13E-03	6.61E-03	2.02E-05
6	2.00E-04	0.00E+00	2.62E-01	6.13E+00	6.39E+00	1.96E-02
7	3.90E-04	0.00E+00	3.43E-01	8.03E+00	8.38E+00	2.56E-02
8	1.61E+01	7.05E-02	1.33E+03	3.12E+04	3.26E+04	9.97E+01
Total	1.74E+01	7.55E-01	1.34E+03	3.13E+04	3.27E+04	1.00E+02
Percent of total SAI	5.32E-02	2.31E-03	4.09E+00	9.59E+01	1.00E+02	-

5.7 Maximum Benefit

In Section 5.6, the SAI for a single NPM due to internal events, LPSD events, internal flooding events, internal fire events, external flooding events, high winds events, and dropped NPM events is calculated. However, the effect of seismic events and multiple NPMs co-located in the same Reactor Building must also be taken into account to calculate the maximum benefit. The maximum benefit calculated in this section is for a 12 NPM plant, which is bounding for a NuScale plant with a smaller number of NPMs.

Following the guidance in Reference 8.1-1, 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 3.17E-08 per year as shown in Table B-30 and is combined with the summed NuScale RC CDF of 1.05E-06 for a total CDF of 1.08E-06 per year. Therefore, the EEM is:

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

$$\frac{3.17E-08\ per\ year + 1.05E-06\ per\ year}{1.05E-06\ per\ year} = 1.03$$

The NuScale design is unique in that there are 12 operating NPMs housed in one Reactor Building. To conservatively account for the total number of NPMs, an additional multiplier of 12 is simply applied to the SAI calculated in the previous section for RCs 1 through 7. The contribution to the maximum benefit for the Surry site from RCs 1 through 7 is calculated as follows:

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

$$\times EEM \times 12\ NPMs = \$93.8 \times 1.03 \times 12 = \$1{,}160$$

As described in FSAR Section 19.1.7.4, the low power and shutdown PRA analysis has established that the maximum number of NPMs that can be damaged in an NPM drop accident is three. Therefore, a multiplier of three is applied to RC 8 to account for multiple NPMs, with the exception of the AOSC. This approach is conservative as it is unlikely that an event will impact multiple NPMs and cause core damage. For the AOSC calculated in RC 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 RC 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 RC 8 is calculated as follows:

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

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

$$\Big(3 \times [\$16.1 + \$7.05\text{E-}2 + \$1{,}330] +$$

$$[12 \times 3.46E + 09\ \$years + 3 \times \$2.63E + 10\ \$years] \times \frac{1.05E-06}{year}\Big) \times 1.03 = \$134{,}000$$

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

$$Maximum\ Benefit = Maximum\ Benefit_{RCs\,1\text{-}7} + Maximum\ Benefit_{RC8} = \qquad \text{Equation 5-28}$$
$$\$1{,}160 + \$134{,}000 = \$135{,}600 = \$136{,}000$$

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

RC	APE (public exposure, $)	AOC (off-site cost, $)	AOE (on-site exposure, $)	AOSC (on-site cost, $)	Maximum Benefit ($)	Percent of maximum benefit
1	5.00E-04	0.00E+00	3.38E-01	7.91E+00	8.25E+00	6.09E-03
2	1.97E-05	0.00E+00	1.67E-02	3.90E-01	4.07E-01	3.00E-04
3	1.62E+01	8.46E+00	2.81E-01	6.58E+00	3.15E+01	2.32E-02
4	4.05E-02	0.00E+00	3.83E+01	8.98E+02	9.37E+02	6.91E-01
5	2.65E-02	2.03E-03	2.18E-03	5.10E-02	8.17E-02	6.03E-05
6	2.47E-03	0.00E+00	3.23E+00	7.58E+01	7.90E+01	5.83E-02
7	4.82E-03	0.00E+00	4.24E+00	9.93E+01	1.04E+02	7.64E-02
8	4.96E+01	2.18E-01	4.12E+03	1.30E+05	1.34E+05	9.91E+01
Total	6.59E+01	8.68E+00	4.16E+03	1.31E+05	1.36E+05	1.00E+02
Percent of maximum benefit	4.86E-02	6.41E-03	3.07E+00	9.69E+01	1.00E+02	-

Dropped NPM events are significant risk contributors to the NuScale design with the NPM drop RC (RC 8) contributing approximately 99.7 percent of the SAI (Table 5-3) and approximately 99.1 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. MACCS sensitivity analyses are discussed in detail in Section B.3 of this report. Maximum benefit sensitivity cases 1 through 11 are calculated using MACCS sensitivity results. Maximum benefit sensitivity cases 12 through 14 are not calculated using MACCS sensitivity results, but are sensitivities of suggested cost-benefit analysis values in Reference 8.1-2 and the NUREG-1530 Revision 1 draft for public comment (Reference 8.1-4). Maximum benefit values are calculated for the following scenarios:

1) A 10 mile EPZ evacuation is modeled, compared to a site boundary EPZ in the base case.

2) The site population and economic information is based on SecPOP 4.3 (2010 census data and 2012 economic data), compared to the previous SecPOP version (2000 census data and 2002 economic data) used in the base case.

3) Radionuclide release occurs at the Peach Bottom site (see Section B.1.6), compared to the Surry site (see Section B.1.5) in the base case.

4) Radionuclide release occurs at the top of the Reactor Building (24.689 m above grade, see FSAR Section 1.2.2.1), compared to 0 m in the base case.

5) High burnup core inventory, shown in Table B-6, compared to the best estimate core inventory in the base case shown in Table B-5.

6) The release occurs immediately at transient initiation, compared to the time to first gap release predicted by MELCOR in the base case.

7) The maximum non-farmland decontamination cost and duration allowed by MACCS is used for all decontamination levels, compared to the CPI scaled values from the SOARCA analysis used in the base case.

8) The value of economic parameters listed in Table B-2, except DPRATE and DSRATE, are multiplied by two compared to the base case.

9) All radionuclides are available for release from containment, compared to only the airborne fraction calculated by MELCOR in the base case.

10) Radionuclide aerosol dry deposition velocity (0.01 m/s) from NUREG-1150 (Reference 8.1-34) for all radionuclide classes, compared to the base case deposition velocities calculated by MELMACCS (see Appendix B) using the methodology in NUREG/CR-7161 (Reference 8.1-6, Section 3.0).

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

12) High estimate of immediate and long-term occupational (on-site) 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.

13) 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.

14) Real discount rate of three percent per year, compared to seven percent per year in the base case (Reference 8.1-2, Section B.2.1).

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 three, which uses a Peach Bottom site-specific seismic CDF (see Section B.4), the CDF and release category frequencies remain unchanged. Peach Bottom is more seismically active than Surry and therefore has a larger seismic CDF of 2.01E-06, per Table B-30. The external events multiplier for the Peach Bottom site is calculated below.

$$EEM_{Peach\ Bottom} = \frac{\frac{1.05E-06 + 2.01E-06}{year}}{\frac{1.05E-06}{year}} = 2.91 \qquad \text{Equation 5-29}$$

Intact containment MACCS simulations required additional assumptions to analyze off-site 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 are 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	$136,000
1	10 mile EPZ	$151,000
2	2010 population and 2012 economic site data	$136,000
3	Peach Bottom site	$383,000
4	24.689 m release height	$136,000
5	High burnup inventory	$136,000
6	Immediate Release	$136,000
7	Maximum TIMDEC and CDNFRM	$136,000
8	Table B-2, except DPRATE and DSRATE, multiplied by two	$136,000
9	Airborne and deposited release from CNV	$136,000
10	0.01m/s deposition velocity	$136,000
11	100000 W plume heat content	$136,000
12	High on-site dose estimate	$140,000
13	High $/person-rem estimate	$138,000
14	Three percent real discount rate	$341,000

There are five general insights that can be extracted from the results of the sensitivities in Table 5-5.

1) MACCS modeling of the release, and off-site dose and economic consequences are not important contributors to the maximum benefit.

2) On-site dose consequences are not important contributors to the maximum benefit.

3) The monetary value attributed to dose is not an important contributor to the maximum benefit.

4) The site at which the release occurs is an important contributor to the maximum benefit.

5) The inherent value of money is an important contributor to the maximum benefit.

Sensitivity cases 1 through 11 are all related to off-site modeling assumptions and off-site consequences. Of these 11 cases, only case 1 (10 mile EPZ) and case 3 (Peach Bottom site) show any deviation from the base case results.

The 10 mile EPZ sensitivity (case 1) shows an increase of $15,000 over the base case maximum benefit. Evacuation within 10 miles of the release is costly, effectively creating a minimum off-site economic impact of 3.45E+08 per event for all RCs. Even considering that five of eight RCs have no off-site economic impacts per event in the base case (Table 5-2), there is only a slight increase to the maximum benefit value when 10 mile EPZ evacuation is considered. This small impact on the maximum benefit from the large increase in economic consequences is because the economic consequences are frequency weighted, and the RC frequencies are exceedingly small.

The Peach Bottom site sensitivity shows an increase of $247,000 in the maximum benefit compared to the base case Surry site. This is almost entirely due to the much larger seismic CDF of 2.01E-06 per year at the Peach Bottom site, compared to the seismic CDF of 3.17E-08 per year at the Surry site. Before applying the external events multiplier (which accounts for seismic risk) to the single NPM SAI, the Peach Bottom SAI and the Surry SAI are both calculated to be $32,700.

Based on the discussion above and the results of sensitivity cases 1 through 11, it can be concluded that radionuclide releases are not important contributors to the maximum benefit value, unless their frequency of occurrence is increased dramatically. However, site-specific hazards, such as seismic events, are an important contributor to the maximum benefit.

Sensitivity cases 12 and 13 are related to the on-site dose consequences and dollar per person-rem estimate, respectively. Cases 12 and 13 show an increase of $4,000 and $2,000, respectively, in the maximum benefit compared to the base case. This is expected, because the dose consequences (the APE and AOE) contribute less than five percent of the base case maximum benefit.

Sensitivity case 14 is related to the rate at which the value of money decreases over time. This case shows an increase in the maximum benefit of nearly a factor of three over the base case. Therefore, the rate at which the value of money decreases over time is an important factor in estimating the current dollar value of the maximum benefit.

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 described in Reference 8.1-1. 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 PWR equipment that is not in 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.

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.

6.1.4 Excessive Implementation Cost

The maximum benefit is $136,000, 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 $136,000 even without an implementation cost estimate, it is not retained for further analysis.

6.1.5 Very Low Benefit

If a generic 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 listings of candidate risk significant SSC from all NuScale PRAs, discussed in FSAR Tables 19.1-20, 19.1-27, and 19.1-64, are used to inform 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: Candidate Risk-Significant Systems for the NuScale Design

System	Description
CNTS	containment system
ECCS	emergency core cooling system
MPS	module protection system
UHS	ultimate heat sink

Table 6-2: Candidate Risk-Significant Components for 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

Table 6-2: Candidate Risk-Significant Components for the NuScale Design (Continued)

Description
ECCS RVV main valve C
ECCS RRV main valve A
ECCS RRV main valve B
ECCS RVV trip valve X
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 NPM 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."

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.

- 45 SAMDA candidates are not applicable to the NuScale design.
- 18 SAMDA candidates are already 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 are combined with another SAMDA because the candidates had the same intent.
- 37 SAMDA candidates are not required for design certification because the candidates were related to a procedural or surveillance action or were related to a multiple plant site.
- 34 SAMDA candidates are of very low benefit to reducing risk in the NuScale design.
- 3 SAMDA candidates are categorized as having an excessive implementation cost.
- 53 SAMDA candidates are retained for further evaluation.

6.3 Phase II SAMDA Screening

Reference 8.1-1 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). The 53 SAMDAs that are considered for further evaluation in the Phase I screening are evaluated in this section.

None of the SAMDAs from Phase I that are considered for further analysis in Phase II involve changes to the Reactor Building crane design. To provide a more realistic estimate of the benefit of implementing SAMDAs that do not involve the Reactor Building crane, all contributions to the maximum benefit from dropped NPM risk are subtracted in Phase II. Therefore, an upper bound estimate of the benefit of implementing all non-crane SAMDAs is $1,160 (Equation 5-26), which considers the contributions from RCs 1 through 7 and does not consider the contributions from RC 8.

All SAMDAs are evaluated assuming a $100,000 minimum cost of implementation. This assumption is in alignment with previous evaluations accepted by the NRC, which use an estimated cost range of $100,000 to $1,000,000 for hardware modifications in place of a detailed cost estimate for industry standard SAMDAs. This cost estimate range is used in analysis supporting many license renewal applications, including the Cooper Nuclear Station (Reference 8.1-21, page E.2-3) and Vermont Yankee Nuclear Power Station (Reference 8.1-27, page E.2-4) applications. As the $100,000 estimate is on the lower end of the range, it is considered a conservative estimate for the cost of hardware modifications to a 12 NPM plant.

Using the conservative estimate of the benefit of implementing any of the 53 SAMDAs and the minimum cost of implementation, none of the SAMDAs are determined to be cost beneficial to implement.

6.4 Screening Sensitivity

A sensitivity study was performed to determine if any of the SAMDAs would screen in for a Phase II analysis and considered cost beneficial when using different assumptions to calculate the maximum benefit. Re-screening the list of candidate SAMDAs using either the case 3 "Peach Bottom site" sensitivity study maximum benefit of $383,000 or the case 14 "three percent real discount rate" sensitivity study maximum benefit of $341,000 instead of the base case maximum benefit of $136,000 shown in Table 5-5 resulted in 53 SAMDAs being retained for Phase II analysis. Of these 53 SAMDAs that would no longer screen out in Phase I analysis, none applied to the Reactor Building crane. Similar to the base case evaluation, the maximum benefit of implementing non-crane SAMDAs is estimated to be the total benefit attributed to non-crane risk in the NuScale design for each case. This approach yields a benefit of $3,320 for case 3 and a benefit of $2,800 for case 14. However, because Peach Bottom (case 3) is more seismically active than Surry and the Peach Bottom seismic risk is a significant contributor to the maximum benefit, seismic SAMDAs are assumed to have a benefit equal to the maximum benefit weighted by the fraction of the total CDF attributed to seismic CDF, for a benefit of $252,000 for seismic SAMDAs.

Three SAMDAs are determined to be potentially cost beneficial for the Peach Bottom site (case 3) in the sensitivity screening. These potentially cost beneficial SAMDAs are SAMDA 140, SAMDA 188, and SAMDA 189, which are all related to seismic improvements. This shows that design changes to improve the seismic ruggedness of the NuScale design may be beneficial to implement at sites that are seismically active. However, it is likely that with a more detailed cost estimate for implementing these SAMDAs, the SAMDAs would not be cost beneficial. The cost of implementing SAMDA 140 is estimated to be greater than $5,000,000 in the Kewaunee license renewal analysis (Reference 8.1-22, pages F-27 and F-228); therefore, the cost of implementing a similar SAMDA in the NuScale design would exceed the Peach Bottom maximum benefit of $383,000. SAMDA 188 and SAMDA 189 are potential design changes to the NPM and the Reactor Building, respectively. These SAMDAs would require extensive redesigns of the NPM and Reactor Building; therefore, the cost of implementing these SAMDAs would exceed the Peach Bottom maximum benefit of $383,000.

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. An analysis was conducted, with the result that none of the SAMDAs are 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

8.1 Referenced Documents

8.1-1 Nuclear Energy Institute, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document," NEI 05-01, Revision A, November 2005.

8.1-2 U.S. Nuclear Regulatory Commission, "Regulatory Analysis Technical Evaluation Handbook," NUREG/BR-0184, January 1997.

8.1-3 NuScale Topical Report, "Risk Significance Determination," TR-0515-13952-NP-A, Revision 0, October 2016, ADAMS Accession Number ML16284A016.

8.1-4 U.S. Nuclear Regulatory Commission, "Reassessment of NRC's Dollar Per Person-Rem Conversion Factor Policy" (Draft Report), NUREG-1530, Revision 1, August 2015.

8.1-5 U.S. Department of Labor, Bureau of Labor Statistics, Producer Price Indexes, Accessed January 15, 2018, http://www.bls.gov/ppi/, Series ID WPU054, Electric Power.

8.1-6 U.S. Nuclear Regulatory Commission, "Synthesis of Distributions Representing Important Non-Site-Specific Parameters in Off-Site Consequence Analyses," NUREG/CR-7161, SAND2010-3380P, April 2013.

8.1-7 U.S. Nuclear Regulatory Commission, "Code Manual for MACCS2: Volume 1, User's Guide," NUREG/CR-6613, SAND97-0594, May 1998.

8.1-8 U.S. Environmental Protection Agency, "Manual of Protective Action Guides and Protective Actions for Nuclear Accidents," EPA-400-R-92-001, May 1992.

8.1-9 U.S. Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station," NUREG/CR-7155, May 2016.

8.1-10 U.S. Department of Labor Bureau of Labor Statistics, Archived Consumer Price Index Supplemental Files, Accessed January 10, 2018, https://www.bls.gov/cpi/tables/supplemental-files/home.htm.

8.1-11 U.S. Nuclear Regulatory Commission, "MACCS Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project," NUREG/CR-7009, August 2014.

8.1-12 U.S. Census Bureau, "Projections of the Size and Composition of the U.S. Population: 2014-2060," P25-1143, March 2015.

8.1-13 U.S. Census Bureau, "Annual Estimates of the Population for the United States and States, and for Puerto Rico: April 1, 2000 to July 1, 2005," NST-EST2005-01, December 2005.

8.1-14 U.S. Nuclear Regulatory Commission, "SOARCA Surry 2005 Site File," April 2015, ADAMS Accession Number ML15097A078.

8.1-15 U.S. Nuclear Regulatory Commission, "SOARCA Surry 2004 Meteorological File," April 2015, ADAMS Accession Number ML15097A117.

8.1-16 U.S. Nuclear Regulatory Commission, "SOARCA Peach Bottom 2005 Site File," April 2015, ADAMS Accession Number ML15097A076.

8.1-17 U.S. Nuclear Regulatory Commission, "SOARCA Peach Bottom 2006 Meteorological File," April 2015, ADAMS Accession Number ML15097A114.

8.1-18 U.S. Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequences Analyses Project, Volume 1: Peach Bottom Integrated Analysis," NUREG/CR-7110, Revision 1, May 2013.

8.1-19 Reserved.

8.1-20 Reserved.

8.1-21 Nebraska Public Power District, Attachment E, Severe Accident Mitigation Alternatives Analysis Submittal Related to Licensing Renewal for the Cooper Nuclear Station, September 2008, ADAMS Accession Number ML083030246.

8.1-22 Dominion, Attachment F, Severe Accident Mitigation Alternatives Analysis Submittal Related to License Renewal for Kewaunee Power Station, August 2008, ADAMS Accession Number ML082341039.

8.1-23 Reserved.

8.1-24 Reserved.

8.1-25 Reserved.

8.1-26 Reserved.

8.1-27 Entergy Nuclear, Appendix E-Attachment E, Severe Accident Mitigation Alternatives Analysis Submittal Related to Licensing Renewal for the Vermont Yankee Nuclear Power Station, January 2006, ADAMS Accession Number ML060300086.

8.1-28 U.S. Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses Project, Volume 2: Surry Integrated Analysis," NUREG/CR-7110, Revision 1, August 2013.

8.1-29 U.S. Nuclear Regulatory Commission, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan," NUREG-1555, Section 7.3, "Severe Accident Mitigation Alternatives," Revision 1, July 2007.

8.1-30 U.S. Nuclear Regulatory Commission, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Guide 1.183, July 2000.

8.1-31 U.S. Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," NUREG-1935, November 2012.

8.1-32 U.S. Nuclear Regulatory Commission, "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, MACCS Input," NUREG/CR-4551, Volume 2, Revision 1, Part 7, December 1990.

8.1-33 U.S. Nuclear Regulatory Commission, Memorandum and Order, CLI-16-07, May 2016.

8.1-34 U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

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 off-site 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 on-site 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). As described in FSAR Table 8.3-5, 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 to accept portable equipment (see FSAR Section 8.3.1.1.2). Additionally, each DC bus is supplied by two redundant battery chargers (see FSAR Section 8.3.2.1.1).
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 (see FSAR Section 7.0.4.1.4).
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 (see FSAR Section 8.3.2.1.1). 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 (see FSAR Section 8.3). Power is supplied to maintain components in the non-actuated position, and power is removed to actuate components (see FSAR Section 7.0.4.1.3). Safety related systems, such as the DHRS (see FSAR Section 5.4.3.1) and ECCS (see FSAR Section 6.3), do not require electric power to perform their safety functions.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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 (see FSAR Section 8.3). The MPS is supplied directly by DC power (see FSAR Section 7.0.4.1.4). The safety function of MPS is to remove power from the actuator (see FSAR Section 7.0.4.1.3).
SAMDA 9	Provide an additional diesel generator.	Increased availability of on-site 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. Additionally, a connection for a portable diesel generator is available, as discussed in SAMDA 3.
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 on-site AC power.	Already implemented	Cross tie capability available for all EMVS busses, as described in FSAR Section 8.3.1.1.1. Intent considered satisfied by existing design.
SAMDA 12	Create AC power cross-tie capability with other unit (multi-unit site).	Increased availability of on-site AC power.	Combined	Combined with SAMDA 11, which already implements this change in the multi-module NuScale plant.
SAMDA 13	Install an additional, buried off-site power source.	Reduced probability of loss of off-site power.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 14	Install a gas turbine generator.	Increased availability of on-site AC power.	Already implemented	As described in FSAR Section 8.3.1.1.2, the auxiliary AC power source is 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 on-site AC power.	Considered for further evaluation	Evaluated in Phase II.
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, as described in FSAR Section 8.3.2. Backup diesel generators and alternate AC power source are available in addition to batteries, as described in FSAR Section 8.3.1.1.2. Intent considered satisfied.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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. As described in FSAR Section 8.3.1.1.2, there will be two collocated BDGs that service all 12 NPMs. 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.
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	Each BDG has an independent cooling system and does not rely on external water sources (see FSAR Section 8.3.1.1.2).
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 off-site power after an SBO.	Reduced human error probability during off-site 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 off-site 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 off-site power lines.	Improved off-site power reliability during severe weather.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 26	Provide an additional high-pressure injection pump with independent diesel.	Reduced frequency of core melt from small LOCA and SBO sequences.	Considered for further evaluation	Evaluated in Phase II.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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 can operate at low RCS pressures (see FSAR Section 9.3.4 for a description of the CVCS). Therefore, slowing reactor depressurization will not likely extend CVCS operation.
SAMDA 28	Add a diverse low pressure injection system.	Improved injection capability.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 29	Provide capability for alternate injection via diesel-driven fire pump.	Improved injection capability.	Combined	Combined with SAMDA 28.
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 (see FSAR Section 6.3.2.5). The CNV is maintained in a partial vacuum, precluding the need for insulation, as described in FSAR Section 6.2.1.1.2. Therefore, the addition of strainers to the ECCS design would not improve reliability (see FSAR Section 6.3.2.5).
SAMDA 31	Add the ability to manually align emergency core cooling system recirculation.	Enhanced reliability of ECCS suction.	Not applicable to the NuScale design	As described in FSAR Section 6.3, 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 (see FSAR Section 6.3.2.5).
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.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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 (see FSAR Section 6.3.2), 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 (see FSAR Section 15.6.5). 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.
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 (see FSAR Section 6.3.2.8).
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 preventing core damage in sequences involving pipe breaks outside of containment that are not isolated and SGTFs that are not isolated (see top event CVCS-T01 in FSAR Table 19.1-7).
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.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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 (see FSAR Section 6.3.2). 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, as described in FSAR Section 10.4.4.1.
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, as described in FSAR Section 6.3.2.4. 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.
Generic PWR Improvements Related to Cooling Water				
SAMDA 43	Add redundant DC control power for SW pumps.	Increased availability of SW.	Very low benefit	The site cooling water system (SCWS) is not risk significant (Table 6-1). The SCWS only supports auxiliary systems, as described in FSAR Section 9.2.7.
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 the reactor component cooling water system (RCCWS) (see FSAR Section 9.2.2). 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	The SCWS pumps and RCCWS pumps are not cross-tied. The SCWS interfaces with RCCWS only through providing cooling water to the RCCWS heat exchangers (see FSAR Section 9.2.2).

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 46	Add a service water pump.	Increased availability of cooling water.	Very low benefit	The SCWS is not risk significant (Table 6-1). The SCWS only supports auxiliary systems, as described in FSAR Section 9.2.7.
SAMDA 47	Enhance the screen wash system.	Reduced potential for loss of SW due to clogging of screens.	Very low benefit	The SCWS is not risk significant (Table 6-1). The SCWS only supports auxiliary systems, as described in FSAR Section 9.2.7.
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	The RCCWS is not a risk-significant system (Table 6-1). The RCCWS does not support any systems that are required for safe shutdown or to maintain safe shutdown (see FSAR Section 9.2.2.1).
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, as described in FSAR Section 5.1.
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, as described in FSAR Section 5.1.
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	The RCCWS is not risk significant Table 6-1). The RCCWS does not support any systems that are required for safe shutdown or to maintain safe shutdown (see FSAR Section 9.2.2.1).
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	The RCCWS is not risk significant (Table 6-1. The RCCWS does not support any systems that are required for safe shutdown or to maintain safe shutdown (see FSAR Section 9.2.2.1).
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	The RCCWS is not risk significant (Table 6-1). The RCCWS does not support any systems that are required for safe shutdown or to maintain safe shutdown (see FSAR Section 9.2.2.1).
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	The RCCWS is not risk significant (Table 6-1). The RCCWS does not support any systems that are required for safe shutdown or to maintain safe shutdown (see FSAR Section 9.2.2.1).

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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, as described in FSAR Section 5.1.
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, as described in FSAR Section 5.1.
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, as described in FSAR Section 5.1.
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 as described in FSAR Section 5.1.
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, as described in FSAR Section 5.1.
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, as described in FSAR Section 5.1.
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, as described in FSAR Section 5.1.
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, as described in FSAR Section 5.1.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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	The RCCWS is not risk significant (Table 6-1). The RCCWS does not support any systems that are required for safe shutdown or to maintain safe shutdown (see FSAR Section 9.2.2.1).
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	As discussed in FSAR Section 5.4.3, 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 off-site power.	Not applicable to the NuScale design	As discussed in FSAR Section 5.4.3, 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.
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	As discussed in FSAR Section 5.4.3.2, 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.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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	As discussed in FSAR Section 5.4.3, 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	As discussed in FSAR Section 5.4.3, 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	As discussed in FSAR Section 5.4.3, 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	As discussed in FSAR Section 5.4.3, 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	As discussed in FSAR Section 5.4.3, 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 CFWS does not include startup pumps and the normal feedwater pumps provide for the full range of operations (see FSAR Section 10.4.7.2.3). As discussed in FSAR Section 5.4.3, 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 (see top event RCS-T02 in FSAR Table 19.1-7). 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 site that contains multiple plants.
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 (see FSAR Section 5.2.2.4.1).

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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 (see FSAR Section 5.2.2.4.1). Primary MSIVs are not air operated (see FSAR Section 6.2.4).
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. Refer to FSAR Sections 6.2.1, 6.3, 9.3.6, and 9.3.4 for descriptions of the containment system, ECCS, CFDS, and CVCS, respectively.
SAMDA 91	Install a passive containment spray system.	Improved containment spray capability.	Considered for further evaluation	Evaluated in Phase II.
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 (see FSAR Section 19.2.4).
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 (see FSAR Section 19.2.4).
SAMDA 95	Enhance fire protection system and standby gas treatment system hardware and procedures.	Improved fission product scrubbing in severe accidents.	Considered for further evaluation	Evaluated in Phase II.
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 (see FSAR Section 19.2.3.3.2). Containment overpressurization is also not a risk significant event (see FSAR Section 19.2.4).

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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 (see FSAR Section 19.2.3.2.1).
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. See FSAR Section 6.2.1 for a description of the containment system. Containment overpressurization is not a risk significant event (see FSAR Section 19.2.4).
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). See FSAR Section 6.2.1 for a description of the containment system.
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 (see FSAR Section 19.2.4). 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.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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. See FSAR Section 6.2.1 for a description of the containment system.
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. See FSAR Section 6.2.1 for a description of the containment system.
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 (see FSAR Section 19.2.4). Additionally, the CNV is submerged in the UHS and heat is removed to the UHS directly through conduction. See FSAR Section 6.2.1 for a description of the containment system.
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 (see FSAR Section 19.2.3.3.2).
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 (see FSAR Section 19.2.3.3.4).

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 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 that may be used to detect CNV bypass LOCAs (see FSAR Table 7.1-2). 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.	Considered for further evaluation	Evaluated in Phase II.
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, as described in FSAR Section 6.2.4.2.1. ESFAS monitors multiple parameters for automatic containment isolation.
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. See FSAR Section 5.4.3 for a description of the DHRS.
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.	Considered for further evaluation	Evaluated in Phase II.
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.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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. See FSAR Section 5.4.1 for a description of the steam generator system.
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 that interfaces with the RCS is rated for operating pressure (see FSAR Table 10.3-1).
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.	Considered for further evaluation	Evaluated in Phase II.
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. See FSAR Section 5.4.5 for a description of the pressurizer. The RSVs are available for primary system overpressure protection. See FSAR Section 5.2.2 for a description of the reactor coolant system overpressure protection features 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.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
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. See FSAR Section 5.4.3 for a description of the DHRS.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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.
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 (see FSAR Section 5.4.1). Main steam system piping inside containment is designed for full RCS pressure (see FSAR Section 10.3-1) and a Steam Generator Tube Failure (SGTF) can be isolated by closing the MSIVs and FWIVs. MPS signals that detect a SGTF also isolate containment (see FSAR Section 15.6.3). The main steam safety valves are located downstream of the MSIVs and are not expected to lift following a reactor trip (see FSAR Section 10.3).
Generic PWR Improvements Related to ATWS				
SAMDA 130	Add an independent boron injection system.	Improved availability of boron injection during ATWS.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 131	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.	Improved equipment availability after an ATWS.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 132	Provide an additional control system for rod insertion (e.g., AMSAC).	Improved redundancy and reduced ATWS frequency.	Considered for further evaluation	Evaluated in Phase II.
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 (see FSAR Section 19.2.4). Additionally, the CNV is submerged in the UHS and heat is removed to the UHS directly through conduction. See FSAR Section 6.2 for a description of the containment system.

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. See FSAR Section 9.3.6 for a description of the CFDS.
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).	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 141	Provide additional restraints for CO2 tanks.	Increased availability of fire protection given a seismic event.	Very low benefit	Seismically induced fires are not risk significant in the NuScale design (see Task 13 of FSAR Section 19.1.5.2.1).
Generic PWR Improvements to Reduce Fire Risk				
SAMDA 142	Replace mercury switches in fire protection system.	Decreased probability of spurious fire suppression system actuation.	Very low benefit	NuScale safe shutdown equipment is designed to withstand the effects of flooding events, including spurious actuations of the fire protection system (see FSAR Section 3.4.1).
SAMDA 143	Upgrade fire compartment barriers.	Decreased consequences of a fire.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 144	Install additional transfer and isolation switches.	Reduced number of spurious actuations during a fire.	Considered for further evaluation	Evaluated in Phase II.
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.
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 (see FSAR Section 15.6.5).
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 (see FSAR Section 15.6.5).
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. See FSAR Section 7.2.13 for a description of the displays and monitoring.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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.	Considered for further evaluation	Evaluated in Phase II.
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.
SAMDA 155	Provide redundant AAPS.	Improved availability of the AAPS in the event of a Loss of AC power.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
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.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 158	Provide an alternate source of inventory for the CVCS.	Allows coolant injection when the CVCS combining valve fails to open to the DWS, limiting the inventory to CVCS.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 161	Train operators to open CVCS discharge 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 NPM to another through the module heat-up system.	Improve CVCS injection availability when CVCS injection to a NPM 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.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
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.
SAMDA 172	Diversify DHRS Actuation Valves.	Improved reliability of DHRS. Common cause failure of these valves is risk significant.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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.	Considered for further evaluation	Evaluated in Phase II.
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. See FSAR Section 6.3 for a description of the ECCS.
SAMDA 179	Provide redundant, diverse FW isolation valve.	Reduces the probability of common cause failures of feedwater isolation. Improves DHRS availability.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
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.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
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.			
NuScale Specific Improvements to Reduce Seismic Risk				
SAMDA 188	Base mat isolation.	Reduced seismic stresses during seismic event.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 189	Improve RPV support skirt.	Increase the ruggedness of the NPM in a seismic event.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.

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

SAMDA ID	Potential Enhancement	Result of Potential Enhancement	Screening Category	Basis
SAMDA 193	Improve reliability of RSV components.	Reduced core damage frequency.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 195	Diversify process logic elements.	Reduced core damage frequency.	Considered for further evaluation	Evaluated in Phase II.
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.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 197	Improve reliability of EIM components.	Improved reliability of component actuation.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 198	Improve reliability of manual switch components.	Improved reliability of component actuation.	Considered for further evaluation	Evaluated in Phase II.
SAMDA 199	Improve reliability of SVM components.	Improved reliability of component actuation.	Considered for further evaluation	Evaluated in Phase II.
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 $136,000.
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 $136,000.
SAMDA 202	Provide a railway system on the reactor pool floor to assist in transporting the NPM 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 $136,000.
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 Off-site Consequence Calculation

The purpose of this appendix is to show key assumptions used in the calculation of the APD and the Monetary Value of Off-site 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 3.10.1.2 code in conjunction with Level 2 PRA results and Environmental Report-specific MELCOR 2.2.9541 analyses were utilized. MELMACCS 2.0.1 is used to extract the required release information from the MELCOR containment bypass simulation results.

B.1 Off-site Models

A goal of this design certification SAMDA analysis is to use realistic, representative site data for the evaluation of off-site 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. As noted in NUREG-1935 (Reference 8.1-31), the SOARCA study was intended to realistically evaluate the off-site emergency response and radiological health consequences at the Surry and Peach Bottom sites. Therefore, adaptation of site modeling used in the SOARCA analysis is considered representative for off-site dose consequence evaluation. It is important to note, however, that the SOARCA study was not intended to evaluate off-site economic consequences of severe accidents at the Surry and Peach Bottom sites. Sensitivity studies on off-site economic inputs are performed to evaluate the impact of SOARCA economic consequence inputs in the calculation of off-site economic impact. These sensitivities are discussed in Section B.3.

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 3.10.1.2 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 8.1-9).

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 1.294. This value is calculated as shown below:

$$\frac{\textit{September 2017 Urban Consumer Price Index}}{\textit{January 2005 Urban Consumer Price Index}} = \frac{246.819}{190.7} = 1.294$$

The values for the CPI were used from the Archived Consumer Price Index Files Historical CPI-U for November 2017 (Reference 8.1-10, Table 24), which includes the historical CPI dating back to 1913 for each month of the year. The CPI from January 2005 is used because this is the lowest CPI value for 2005, and therefore generates the largest possible CPI scaling factor for bringing values from 2005 up to date. The CPI for September 2017 is used as it is the largest 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

Reference 8.1-1 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. 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 Reference 8.1-11. 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 2017 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	11181
2	5115	1840	1450	1227	1549	11181
3	4365	2507	2439	2591	2116	14018
4	3614	3174	3428	3954	2683	16853
5 [NNE]	3614	3174	3428	3954	2683	16853
6	3614	3174	3428	3954	2683	16853
7	2249	3151	3001	2549	1341	12291
8	889	3129	2573	1143	0	7734
9 [NE]	889	3129	2573	1143	0	7734
10	889	3129	2573	1143	0	7734
11	1303	3607	1519	591	1019	8039
12	1719	4082	463	39	2037	8340
13 [ENE]	1719	4082	463	39	2037	8340
14	1719	4082	463	39	2037	8340
15	6241	9226	273	325	1346	17411
16	10762	14370	83	612	652	26479
17 [E]	10762	14370	83	612	652	26479
18	10762	14370	83	612	652	26479
19	10675	35953	19689	15976	16866	99159
20	10586	57536	39299	31343	33079	171843
21 [ESE]	10586	57536	39299	31343	33079	171843
22	10586	57536	39299	31343	33079	171843
23	5294	34297	47029	94153	50527	231300
24	0	11060	54761	156963	67974	290758

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
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	3313	6741	7460	2070	19930
32	498	1068	2916	5320	2445	12247
33 [S]	498	1068	2916	5320	2445	12247
34	498	1068	2916	5320	2445	12247
35	320	760	2244	5597	1929	10850
36	144	453	1575	5874	1414	9460
37 [SSW]	144	453	1575	5874	1414	9460
38	144	453	1575	5874	1414	9460
39	180	426	1198	3303	1618	6725
40	219	404	823	730	1824	4000
41 [SW]	219	404	823	730	1824	4000
42	219	404	823	730	1824	4000
43	299	458	1061	1189	1651	4658
44	382	515	1303	1648	1479	5327
45 [WSW]	382	515	1303	1648	1479	5327
46	382	515	1303	1648	1479	5327
47	301	407	1208	12184	7823	21923
48	223	302	1112	22722	14167	38526
49 [W]	223	302	1112	22722	14167	38526
50	223	302	1112	22722	14167	38526
51	147	720	1375	19275	46319	67836
52	72	1141	1638	15827	78472	97150
53 [WNW]	72	1141	1638	15827	78472	97150
54	72	1141	1638	15827	78472	97150
55	1310	1004	1631	10551	60423	74919
56	2550	868	1622	5273	42375	52688
57 [NW]	2550	868	1622	5273	42375	52688
58	2550	868	1622	5273	42375	52688
59	3974	3197	2018	3020	22552	34761
60	5401	5526	2414	768	2730	16839
61 [NNW]	5401	5526	2414	768	2730	16839
62	5401	5526	2414	768	2730	16839
63	5257	3683	1932	998	2140	14010
64	5115	1840	1450	1227	1549	11181
Total	**169461**	**444095**	**504088**	**1052173**	**1018303**	**3188120**

B.1.5.2 Surry Economic Information

Reference 8.1-1 prescribes that economic data be expressed in current dollars. Extrapolation of the economic data from 2005 to 2017 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 ($/person-day)	223
POPCST	Population relocation cost ($/person)	15531
RELCST	Daily cost for a person who is relocated ($/person-day)	223
CDFRM	Cost of farm decontamination for various levels of decontamination ($/hectare)	1) 1721 2) 3831
CDNFRM	Cost of non-farm decontamination per resident for various levels of decontamination ($/person)	1) 9202 2) 24951
DLBCST	Average cost of decontamination labor ($/person-year)	108719
VALWF	Value of farm wealth ($/hectare)	8931
VALNWF	Value of non-farm wealth ($/person)	284741

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 2017 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
3	1047	20550	25462	14788	49313	111160

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
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
50	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
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

Reference 8.1-1 prescribes that economic data be expressed in current dollars. Extrapolation of the economic data from 2005 to 2017 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 ($/person-day)	223
POPCST	Population relocation cost ($/person)	15531
RELCST	Daily cost for a person who is relocated ($/person-day)	223
CDFRM	Cost of farm decontamination for various levels of decontamination ($/hectare)	1) 1721 2) 3831
CDNFRM	Cost of non-farm decontamination per resident for various levels of decontamination ($/person)	1) 9202 2) 24951
DLBCST	Average cost of decontamination labor ($/person-year)	108719
VALWF	Value of farm wealth ($/hectare)	11700
VALNWF	Value of non-farm wealth ($/person)	271799

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 base case 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 in the base case, presented in Table B-5. 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. A high burnup core inventory was also calculated at 102 percent core power and is presented in Table B-6. The high burnup core inventory is used in a sensitivity analysis described in Section B.3.5.

Table B-5: Best Estimate 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
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**	1.07E+17
Sr-90	1.94E+16	**Co-58**	5.07E+12	**Pr-143**	2.49E+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

Table B-6: High Burnup Core Inventory

NuScale High Burnup Core Inventory (Bq)					
Kr-85	4.96E+15	**I-135**	3.16E+17	**Ce-144**	2.17E+17
Kr-85m	3.57E+16	**Te-127**	1.94E+16	**Np-239**	4.59E+18
Kr-87	6.89E+16	**Te-127m**	3.20E+15	**Pu-238**	3.06E+15
Kr-88	8.98E+16	**Te-129**	5.41E+16	**Pu-239**	1.27E+14
Xe-133	3.31E+17	**Te-129m**	9.24E+15	**Pu-240**	2.56E+14
Xe-135	1.31E+17	**Te-131m**	3.66E+16	**Pu-241**	6.34E+16
Xe-135m	7.73E+16	**Te-132**	2.37E+17	**Zr-95**	2.51E+17
Cs-134	9.88E+16	**Te-131**	1.44E+17	**Zr-97**	2.65E+17
Cs-136	2.18E+16	**Rh-105**	2.51E+17	**Am-241**	1.59E+14
Cs-137	6.05E+16	**Ru-103**	3.27E+17	**Cm-242**	3.91E+16
Rb-86	5.90E+14	**Ru-105**	2.67E+17	**Cm-244**	1.63E+16
Rb-88	9.21E+16	**Ru-106**	2.11E+17	**La-140**	2.89E+17
Ba-139	2.86E+17	**Rh-103m**	3.24E+17	**La-141**	2.58E+17
Ba-140	2.74E+17	**Rh-106**	2.27E+17	**La-142**	2.46E+17
Sr-89	1.24E+17	**Nb-95**	2.52E+17	**Nd-147**	1.06E+17
Sr-90	4.18E+16	**Co-58**	6.22E+12	**Pr-143**	2.31E+17
Sr-91	1.63E+17	**Co-60**	6.88E+13	**Y-90**	4.28E+16
Sr-92	1.81E+17	**Mo-99**	3.00E+17	**Y-91**	1.67E+17
Ba-137m	5.76E+16	**Tc-99m**	2.66E+17	**Y-92**	1.83E+17
I-131	1.72E+17	**Nb-97**	2.67E+17	**Y-93**	2.13E+17
I-132	2.45E+17	**Nb-97m**	2.52E+17	**Y-91m**	9.61E+16
I-133	3.30E+17	**Ce-141**	2.60E+17	**Pr-144**	2.18E+17
I-134	3.66E+17	**Ce-143**	2.35E+17	**Pr-144m**	2.62E+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 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 off-site 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. In the NuScale PRA (FSAR Chapter 19), these cutsets are grouped into sequences based on the systems affected by the specific equipment failures in a cutset. Since there are numerous sequences in the NuScale PRA, to make the calculation more manageable many of these sequences can be conservatively grouped into general RCs for the SAMDA analysis. In this manner, only a small number of RCs need to be analyzed and described.

The NuScale severe-accident sequences are binned into eight RCs based on initiating event and mitigating system availability. In this manner, sequences 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 sequences when compared to intact containment sequences. Sequences that do not reach core damage are not included in a RC, 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 NPM fails to remain upright and is completely submerged in the reactor pool, is a unique event and is considered separately as RC 8 (Section B.2.8).

Initiating events for many sequences may have different causes (e.g. a reactor trip could be caused by a random equipment failure or a fire), but have similar accident progressions due to mitigating system availability and are binned together. For example, sequences where coolant is lost to containment due to an ECCS RVV spurious actuation in which no coolant makeup is available, are expected to progress similarly, independent of how the ECCS is actuated (e.g., spurious ECCS, loss of power to ECCS control solenoids) and are grouped together in RC 4. The constituent sequences of each RC are identified by Level 1 PRA event tree (FSAR Chapter 19.1) and the numbered end state in that event tree, as shown in Section B.2.1 through Section B.2.7.

Reference 8.1-1 recommends that sequences are grouped into release categories based on timing of release and radionuclide release fractions to the environment. Both the timing and magnitude of release to the environment are expected to be dependent upon the mitigating systems available.

To calculate the RC frequencies, the following NuScale PRAs are used as inputs. These PRAs are discussed in more detail in Chapter 19 of the NuScale FSAR. Each PRA calculates a CDF for each sequence. The frequencies associated with sequence binned into a specific RC are summed together to establish the frequency associated with that specific RC, as shown in Table B-25.

- internal events PRA
- LPSD PRA
- internal flooding PRA
- internal fires PRA
- external floods PRA
- high winds PRA

The release associated with each RC (shown in Table B-26) is selected to be the sequence that is completely unmitigated, with the exception of ECCS failures, and is described in Section B.2.1 to Section B.2.8. 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. To simplify the analysis, the Reactor Building is not credited for radionuclide deposition, with the exception of the Reactor Building pool in RC 8. This simplification is bounding, as some amount of the release from the module will deposit or be held up in the Reactor Building before reaching the environment.

B.2.1 Release Category 1: CVCS LOCA Inside Containment

The first RC 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 RC 4. The PRA sequences that have a significant contribution to RC 1 are shown in Table B-7. RC 1 is an intact containment scenario.

The conservative release sequence is 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 core support failure and core relocation to the RPV lower plenum. Radionuclides released from the core relocate to the CNV, where the majority of aerosols deposit in the reactor coolant present in the CNV and on the CNV walls. In the MELCOR simulations, the CNV is modeled as sealed with no leakage. Therefore, the radionuclide release to the environment is calculated separately based on the fraction of the core inventory airborne in containment and the design basis containment leak rate of 0.2 weight percent per day (see FSAR Table 6.5-1).

The plume release information for RC 1 that is used as input for MACCS calculations is shown in Table B-8. As a result of modeling the CNV as sealed, assumptions relating to the particle size distribution and plume buoyancy are required as inputs to MACCS. The particle size distribution for RC 1 was assumed to be the same as the particle size distribution for RC 3 (see Table B-11). RC category 3 is a containment bypass accident, and the particle size distribution of a containment bypass accident is expected to cause increased deposition in the environment as compared to the actual particle size distribution for an intact containment sequence. The plume buoyancy is simulated using the heat model with an assumed plume heat content of zero watts for each plume segment. The result of this assumption is that the plume does not rise upon release, which maximizes the ground level radionuclide concentrations. Sensitivity studies evaluating the impact of the aerosol deposition in the CNV predicted by MELCOR, the assumed particle size distribution, and the non-buoyant plume are discussed in Section B.3.

Table B-7: PRA Sequences that Contribute Significantly to Release Category 1

Event Tree	Sequence	Description	FSAR Reference
CVCS--ALOCA-CIC-ET	3	Reactor trip, ECCS fails, CVCS injection fails	Figure 19.1-4

Table B-8: 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	13440	3600	0	4.31E-06	6.75E-07	4.88E-08	4.96E-07	5.31E-07	1.04E-09	1.27E-07	2.36E-14	2.36E-14
2	17040	3600	0	2.28E-05	9.86E-07	3.19E-09	5.75E-07	6.04E-07	4.24E-09	2.76E-07	7.63E-14	7.63E-14
3	20640	3600	0	2.87E-05	1.37E-07	2.70E-10	1.50E-07	8.37E-08	1.05E-09	3.65E-08	2.53E-14	2.53E-14
4	24240	3600	0	3.19E-05	6.27E-08	8.18E-10	8.26E-08	1.35E-07	8.20E-10	1.65E-08	1.91E-14	1.91E-14
5	27840	3600	0	3.40E-05	4.49E-08	3.33E-10	4.06E-08	4.87E-08	1.06E-09	1.24E-08	2.04E-14	2.04E-14
6	31440	3600	0	3.88E-05	8.27E-08	6.77E-10	8.14E-08	9.03E-08	1.54E-09	2.13E-08	4.16E-14	4.20E-14
7	35040	3600	0	4.19E-05	7.80E-08	2.61E-09	2.00E-07	2.40E-07	1.16E-10	1.73E-08	1.45E-14	1.66E-14
8	38640	3600	0	4.23E-05	3.91E-08	8.57E-10	1.56E-07	8.08E-08	2.89E-11	6.85E-09	6.40E-15	7.33E-15
9	42240	3600	0	4.08E-05	1.23E-08	2.14E-10	4.55E-08	2.32E-08	7.21E-12	2.09E-09	1.74E-15	1.96E-15
10	45840	3600	0	4.05E-05	4.44E-09	7.41E-11	1.82E-08	8.16E-09	2.56E-12	7.64E-10	6.15E-16	6.91E-16
11	49440	3600	0	4.05E-05	2.31E-09	3.25E-11	1.15E-08	3.77E-09	1.43E-12	3.96E-10	2.99E-16	3.32E-16
12	53040	3600	0	4.05E-05	1.45E-09	1.57E-11	8.95E-09	1.99E-09	1.05E-12	2.31E-10	1.70E-16	1.86E-16
13	56640	3600	0	4.04E-05	1.64E-09	7.43E-12	9.71E-09	1.24E-09	1.15E-12	1.49E-10	1.21E-16	1.28E-16
14	60240	3600	0	3.95E-05	2.09E-09	4.93E-12	1.18E-08	1.25E-09	1.16E-12	1.30E-10	1.11E-16	1.16E-16
15	63840	3600	0	3.92E-05	1.30E-09	2.64E-12	9.32E-09	7.87E-10	6.37E-13	8.52E-11	6.98E-17	7.32E-17
16	67440	3600	0	3.90E-05	8.20E-10	1.59E-12	7.05E-09	4.11E-10	7.24E-13	8.55E-11	4.45E-17	4.60E-17
17	71040	3600	0	3.87E-05	1.47E-09	2.86E-12	9.59E-09	4.28E-10	2.41E-12	2.70E-10	7.10E-17	7.14E-17
18	74640	3600	0	3.85E-05	2.36E-09	5.27E-12	1.75E-08	8.91E-10	4.52E-12	3.79E-10	1.54E-16	1.56E-16
19	78240	3600	0	3.82E-05	1.63E-09	3.64E-12	1.39E-08	8.05E-10	2.79E-12	2.58E-10	1.38E-16	1.40E-16
20	81840	3600	0	3.81E-05	1.01E-09	2.31E-12	1.04E-08	6.24E-10	1.54E-12	1.64E-10	1.06E-16	1.08E-16
21	85440	3660	0	3.88E-05	9.35E-10	2.17E-12	1.03E-08	7.32E-10	1.08E-12	1.61E-10	1.30E-16	1.32E-16
22	89100	3600	0	3.83E-05	1.40E-09	3.29E-12	1.38E-08	1.20E-09	1.02E-12	2.59E-10	2.46E-16	2.48E-16
23	92700	3600	0	3.84E-05	1.07E-09	2.57E-12	1.34E-08	9.46E-10	6.73E-13	2.02E-10	2.02E-16	2.03E-16
24	96300	3600	0	3.87E-05	6.45E-10	1.60E-12	1.52E-08	5.97E-10	3.68E-13	1.23E-10	1.30E-16	1.30E-16
25	99900	3600	0	3.93E-05	4.88E-10	1.30E-12	2.34E-08	5.21E-10	2.52E-13	9.47E-11	1.08E-16	1.08E-16
26	103500	3600	0	3.94E-05	3.22E-10	9.18E-13	2.51E-08	3.63E-10	1.70E-13	6.29E-11	7.80E-17	7.82E-17
27	107100	3600	0	3.95E-05	1.13E-10	3.59E-13	2.64E-08	1.36E-10	6.60E-14	2.24E-11	3.17E-17	3.17E-17
28	110700	3600	0	3.96E-05	2.97E-11	1.02E-13	2.81E-08	4.03E-11	1.87E-14	5.88E-12	9.22E-18	9.23E-18
29	114300	3600	0	3.98E-05	1.89E-11	6.84E-14	3.10E-08	2.74E-11	1.33E-14	3.73E-12	6.37E-18	6.37E-18
30	117900	3600	0	4.03E-05	4.47E-12	2.06E-14	3.66E-08	8.34E-12	4.72E-15	8.58E-13	2.16E-18	2.17E-18
31	121500	3600	0	4.09E-05	2.47E-12	1.41E-14	4.35E-08	5.35E-12	3.77E-15	4.63E-13	1.89E-18	1.91E-18

Table B-8: 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	125100	3600	0	4.15E-05	1.31E-12	8.74E-15	5.04E-08	3.04E-12	2.53E-15	2.44E-13	1.31E-18	1.30E-18
33	128700	3600	0	4.20E-05	6.17E-13	5.21E-15	5.73E-08	1.61E-12	1.66E-15	1.14E-13	7.91E-19	7.82E-19
34	132300	3600	0	4.26E-05	4.81E-13	5.37E-15	6.46E-08	1.50E-12	2.07E-15	8.62E-14	8.08E-19	7.98E-19
35	135900	3600	0	4.32E-05	3.50E-13	4.97E-15	7.17E-08	1.37E-12	2.57E-15	6.14E-14	7.32E-19	7.22E-19
36	139500	3600	0	4.43E-05	6.01E-12	1.03E-13	8.61E-08	3.29E-11	7.54E-14	1.07E-12	1.40E-17	1.37E-17
37	143100	3600	0	4.71E-05	3.86E-11	6.89E-13	1.25E-07	2.25E-10	5.76E-13	7.04E-12	8.57E-17	8.37E-17
38	146700	3600	0	4.93E-05	5.46E-11	1.01E-12	1.54E-07	3.62E-10	8.92E-13	1.05E-11	1.07E-16	1.05E-16
39	150300	3600	0	5.06E-05	6.73E-11	1.24E-12	1.74E-07	5.22E-10	1.02E-12	1.38E-11	1.10E-16	1.09E-16
40	153900	3600	0	5.15E-05	9.41E-11	1.66E-12	1.90E-07	7.75E-10	1.19E-12	2.04E-11	1.24E-16	1.23E-16
41	157500	3600	0	5.23E-05	1.15E-10	1.94E-12	2.03E-07	8.56E-10	1.16E-12	2.61E-11	1.21E-16	1.21E-16
42	161100	3600	0	5.29E-05	1.26E-10	2.13E-12	2.14E-07	8.64E-10	1.01E-12	2.93E-11	1.07E-16	1.08E-16
43	164700	3600	0	5.34E-05	1.01E-10	1.98E-12	2.24E-07	7.35E-10	6.81E-13	2.39E-11	7.51E-17	7.63E-17
44	168300	3600	0	5.39E-05	6.94E-11	1.78E-12	2.33E-07	6.06E-10	4.05E-13	1.63E-11	4.75E-17	4.91E-17
45	171900	3600	0	5.43E-05	4.35E-11	1.61E-12	2.42E-07	5.06E-10	2.22E-13	1.00E-11	2.88E-17	3.05E-17
46	175500	3600	0	5.47E-05	3.11E-11	1.70E-12	2.50E-07	5.02E-10	1.36E-13	6.99E-12	2.05E-17	2.25E-17
47	179100	3600	0	5.51E-05	2.41E-11	1.88E-12	2.58E-07	5.35E-10	8.74E-14	5.27E-12	1.62E-17	1.85E-17
48	182700	3600	0	5.55E-05	1.70E-11	1.83E-12	2.66E-07	5.08E-10	4.78E-14	3.61E-12	1.18E-17	1.40E-17
49	186300	3600	0	5.59E-05	1.33E-11	1.84E-12	2.73E-07	5.05E-10	2.71E-14	2.74E-12	9.55E-18	1.16E-17
50	189900	3600	0	5.62E-05	1.11E-11	1.86E-12	2.79E-07	5.07E-10	1.54E-14	2.24E-12	8.20E-18	1.02E-17
51	193500	3600	0	5.65E-05	9.68E-12	1.86E-12	2.85E-07	5.04E-10	8.87E-15	1.93E-12	7.30E-18	9.28E-18
52	197100	3600	0	5.68E-05	8.92E-12	1.88E-12	2.91E-07	5.08E-10	5.29E-15	1.77E-12	6.80E-18	8.72E-18
53	200700	3600	0	5.70E-05	8.15E-12	1.82E-12	2.96E-07	4.93E-10	3.16E-15	1.62E-12	6.23E-18	8.04E-18
54	204300	3600	0	5.72E-05	7.35E-12	1.71E-12	3.01E-07	4.63E-10	1.91E-15	1.46E-12	5.58E-18	7.22E-18
55	207900	3600	0	5.74E-05	6.59E-12	1.57E-12	3.05E-07	4.26E-10	1.21E-15	1.32E-12	4.92E-18	6.39E-18
56	211500	3600	0	5.76E-05	5.71E-12	1.39E-12	3.09E-07	3.76E-10	8.01E-16	1.14E-12	4.18E-18	5.42E-18
57	215100	3600	0	5.78E-05	4.98E-12	1.23E-12	3.13E-07	3.33E-10	5.81E-16	9.98E-13	3.55E-18	4.61E-18
58	218700	3600	0	5.79E-05	4.50E-12	1.13E-12	3.16E-07	3.06E-10	4.67E-16	9.01E-13	3.13E-18	4.05E-18
59	222300	3600	0	5.81E-05	4.11E-12	1.05E-12	3.19E-07	2.85E-10	4.00E-16	8.25E-13	2.79E-18	3.60E-18
60	225900	3600	0	5.82E-05	3.84E-12	9.99E-13	3.22E-07	2.70E-10	3.63E-16	7.70E-13	2.52E-18	3.26E-18
61	229500	3600	0	5.83E-05	3.54E-12	9.37E-13	3.24E-07	2.53E-10	3.35E-16	7.10E-13	2.26E-18	2.91E-18
62	233100	3600	0	5.84E-05	3.25E-12	8.74E-13	3.26E-07	2.34E-10	3.10E-16	6.50E-13	2.01E-18	2.59E-18
63	236700	3600	0	5.85E-05	2.87E-12	7.86E-13	3.28E-07	2.07E-10	2.77E-16	5.73E-13	1.73E-18	2.22E-18

Table B-8: 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	240300	3600	0	5.86E-05	2.58E-12	7.21E-13	3.30E-07	1.87E-10	2.52E-16	5.15E-13	1.51E-18	1.94E-18
65	243900	3600	0	5.86E-05	2.27E-12	6.47E-13	3.31E-07	1.65E-10	2.24E-16	4.54E-13	1.30E-18	1.67E-18
66	247500	3600	0	5.87E-05	1.90E-12	5.53E-13	3.33E-07	1.39E-10	1.91E-16	3.83E-13	1.11E-18	1.42E-18
67	251100	3600	0	5.87E-05	1.55E-12	4.53E-13	3.34E-07	1.13E-10	3.06E-16	3.12E-13	9.75E-19	1.24E-18
68	254700	3600	0	5.87E-05	1.11E-12	3.12E-13	3.35E-07	7.88E-11	1.28E-15	2.27E-13	8.25E-19	1.02E-18
69	258300	3600	0	5.87E-05	8.42E-13	2.19E-13	3.35E-07	5.71E-11	1.89E-15	1.73E-13	7.01E-19	8.43E-19
70-96	261900	3600	0	5.87E-05	8.35E-13	2.16E-13	3.35E-07	5.65E-11	1.90E-15	1.72E-13	6.97E-19	8.37E-19
Total	13440	345660	0	4.80E-03	2.14E-06	5.80E-08	2.07E-05	1.88E-06	9.93E-09	5.19E-07	2.33E-13	2.37E-13

B.2.2 Release Category 2: Pipe Break Outside Containment, Isolated

The second RC consists of pipe breaks that occur either through the CVCS injection line, CVCS discharge 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 RC 6 in Section B.2.6. The PRA sequences that have a significant contribution to RC 2 are shown in Table B-9. RC 2 is an intact containment scenario.

The conservative release sequence is a CVCS injection line pipe break 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 containment remains intact. 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 between the RPV and CNV. Due to partial ECCS actuation, reactor coolant completely relocates to the CNV and is unable to recirculate back 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. Radionuclides released from the core relocate to the CNV, where the majority of aerosols deposit in the reactor coolant present in the CNV and on the CNV walls. In the MELCOR simulations, the CNV is modeled as sealed with no leakage. Therefore, the radionuclide release to the environment is calculated separately based on the fraction of the core inventory airborne in containment and the design basis containment leak rate of 0.2 weight percent per day (see FSAR Table 6.5-1).

The plume release information for RC 2 that is used as input for MACCS calculations is shown in Table B-10. As a result of modeling the CNV as sealed, assumptions relating to the particle size distribution and plume buoyancy are required inputs to MACCS. The particle size distribution for RC 2 is assumed to be the same as the particle size distribution for RC 3 (see Table B-11). RC 3 is a containment bypass accident, and the particle size distribution of a containment bypass accident is expected to cause increased deposition in the environment as compared to the actual particle size distribution for an intact containment sequence. The plume buoyancy is simulated using the heat model with an assumed plume heat content of zero watts for each plume segment. The result of this assumption is that the plume does not rise upon release, which maximizes the ground level radionuclide concentrations. Sensitivity studies evaluating the impact of the aerosol deposition in the CNV predicted by MELCOR, the assumed particle size distribution, and the non-buoyant plume are discussed in Section B.3.

Table B-9: PRA Sequences that Contribute Significantly to Release Category 2

Event Tree	Sequence	Description	FSAR Reference
CVCS--ALOCA-COC-ET	5	Reactor trip, LOCA isolated, DHRS fails, RSVs fail to open	Figure 19.1-2
	12	ATWS, LOCA isolated, RSVs fail to open	
CVCS--ALOCA-LOC-ET	6	Reactor trip, LOCA isolated, DHRS fails, RSVs fail to open	Figure 19.1-3
	16	ATWS, LOCA isolated, RSVs fail to open	
MSS---ALOCA-SG--ET	6	Reactor trip, LOCA isolated, DHRS fails, RSVs fail to open	Figure 19.1-7
	16	ATWS, LOCA isolated, RSVs fail to open	

Table B-10: 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	29400	3600	0	2.86E-06	3.23E-07	4.38E-08	2.79E-07	2.91E-07	9.81E-11	3.25E-08	5.20E-15	5.20E-15
2	33000	3600	0	1.73E-05	1.58E-06	6.50E-09	9.59E-07	9.84E-07	5.49E-09	4.40E-07	1.02E-13	1.02E-13
3	36600	3600	0	2.66E-05	9.29E-08	1.35E-10	6.37E-08	5.83E-08	8.85E-10	2.51E-08	2.07E-14	2.07E-14
4	40200	3600	0	2.88E-05	1.04E-07	4.64E-10	2.15E-07	1.25E-07	7.42E-10	2.53E-08	1.93E-14	1.92E-14
5	43800	3600	0	3.03E-05	4.20E-08	1.34E-09	6.14E-08	1.22E-07	4.05E-10	1.07E-08	1.11E-14	1.11E-14
6	47400	3600	0	3.07E-05	1.07E-07	4.76E-10	8.15E-08	8.34E-08	1.30E-09	2.97E-08	3.04E-14	3.07E-14
7	51000	3600	0	3.69E-05	6.17E-08	4.77E-10	6.47E-08	7.85E-08	2.22E-10	1.41E-08	7.42E-15	7.59E-15
8	54600	3600	0	3.70E-05	8.83E-09	4.12E-10	4.00E-08	3.46E-08	9.03E-12	1.34E-09	9.96E-16	9.02E-16
9	58200	3600	0	3.73E-05	3.16E-09	1.29E-10	1.57E-08	1.01E-08	2.42E-12	4.43E-10	3.11E-16	2.70E-16
10	61800	3600	0	3.74E-05	1.12E-09	3.90E-11	7.17E-09	3.04E-09	6.88E-13	1.46E-10	9.53E-17	8.15E-17
11	65400	3600	0	3.74E-05	4.36E-10	1.28E-11	4.43E-09	1.02E-09	2.22E-13	5.21E-11	3.17E-17	2.69E-17
12	69000	3600	0	3.72E-05	1.85E-10	4.49E-12	3.44E-09	3.71E-10	8.03E-14	1.95E-11	1.13E-17	9.53E-18
13	72600	3600	0	3.70E-05	8.93E-11	1.73E-12	3.06E-09	1.48E-10	3.19E-14	7.76E-12	4.36E-18	3.67E-18
14	76200	3600	0	3.67E-05	4.91E-11	6.95E-13	2.89E-09	6.16E-11	1.32E-14	3.19E-12	1.76E-18	1.48E-18
15	79800	3600	0	3.65E-05	3.19E-11	2.89E-13	2.81E-09	2.66E-11	5.57E-15	1.34E-12	7.30E-19	6.14E-19
16	83400	3600	0	3.63E-05	4.05E-11	1.56E-13	2.79E-09	1.52E-11	3.05E-15	7.30E-13	3.93E-19	3.30E-19
17	87000	3600	0	3.60E-05	1.29E-10	8.08E-14	2.87E-09	9.47E-12	1.63E-15	3.81E-13	2.04E-19	1.71E-19
18	90600	3600	0	3.56E-05	3.24E-10	2.10E-14	3.04E-09	4.15E-12	4.46E-16	9.83E-14	5.30E-20	4.45E-20
19	94200	3600	0	3.53E-05	3.83E-10	3.97E-15	3.16E-09	1.95E-12	1.28E-16	1.94E-14	1.06E-20	9.00E-21
20	97800	3600	0	3.50E-05	1.44E-10	1.48E-15	3.25E-09	2.13E-12	7.41E-17	9.12E-15	4.31E-21	3.65E-21
21	101400	3600	0	3.47E-05	7.87E-11	3.84E-16	3.50E-09	3.67E-12	2.09E-17	4.09E-15	9.70E-22	6.98E-22
22	105000	3600	0	3.46E-05	9.45E-11	1.51E-16	3.94E-09	6.06E-12	7.78E-18	2.48E-15	3.11E-22	3.22E-23
23	108600	3600	0	3.51E-05	2.98E-10	1.40E-16	6.84E-09	2.13E-11	4.24E-18	3.64E-15	1.76E-22	1.09E-22
24	112200	3600	0	3.48E-05	2.66E-10	6.76E-16	6.42E-09	1.99E-11	1.42E-17	4.58E-15	1.50E-21	1.15E-21
25	115800	3600	0	3.43E-05	2.46E-10	1.72E-16	6.12E-09	1.92E-11	3.63E-18	1.17E-15	2.45E-22	6.40E-23
26	119400	3600	0	3.42E-05	3.41E-10	3.25E-17	7.44E-09	3.02E-11	6.88E-19	5.15E-16	5.53E-25	2.50E-29
27	123000	3600	0	3.40E-05	1.58E-10	4.98E-18	4.90E-09	4.50E-11	1.05E-19	5.10E-16	6.55E-32	3.02E-32
28	126600	3600	0	3.39E-05	2.70E-11	6.59E-19	3.09E-09	5.10E-11	1.34E-20	5.11E-16	0.00E+00	0.00E+00
29	130200	3600	0	3.37E-05	4.44E-12	8.53E-20	2.77E-09	5.31E-11	1.11E-21	4.12E-16	0.00E+00	0.00E+00
30	133800	3600	0	3.36E-05	1.02E-12	1.06E-20	2.71E-09	5.18E-11	0.00E+00	1.40E-16	0.00E+00	0.00E+00
31	137400	3600	0	3.38E-05	5.40E-13	5.91E-22	2.72E-09	8.45E-11	8.67E-30	2.51E-16	4.05E-35	1.87E-35

Table B-10: 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
32	141000	3600	0	3.35E-05	4.35E-13	3.12E-28	2.70E-09	6.38E-11	4.49E-30	2.10E-16	2.10E-35	9.68E-36
33	144600	3600	0	3.34E-05	2.60E-13	2.22E-28	2.69E-09	4.55E-11	3.19E-30	6.28E-17	1.49E-35	6.89E-36
34	148200	3600	0	3.33E-05	1.99E-13	3.27E-29	2.68E-09	4.72E-11	4.70E-31	2.97E-17	2.20E-36	1.01E-36
35	151800	3600	0	3.32E-05	3.31E-11	7.69E-14	2.78E-09	3.10E-11	5.18E-14	5.12E-12	1.99E-19	1.85E-19
36	155400	3600	0	3.31E-05	6.33E-11	5.34E-14	2.86E-09	6.45E-12	3.60E-14	9.80E-12	1.38E-19	1.28E-19
37	159000	3600	0	3.30E-05	3.61E-11	2.10E-14	2.77E-09	2.88E-12	1.42E-14	5.58E-12	5.45E-20	5.02E-20
38	162600	3600	0	3.30E-05	1.94E-11	1.01E-14	2.71E-09	1.85E-12	6.78E-15	2.98E-12	2.59E-20	2.38E-20
39	166200	3600	0	3.29E-05	1.10E-11	5.48E-15	2.68E-09	1.36E-12	3.69E-15	1.69E-12	1.40E-20	1.26E-20
40	169800	3600	0	3.29E-05	6.70E-12	3.24E-15	2.66E-09	1.07E-12	2.19E-15	1.02E-12	8.24E-21	6.92E-21
41	173400	3600	0	3.28E-05	4.24E-12	1.99E-15	2.65E-09	9.14E-13	1.34E-15	6.43E-13	4.80E-21	4.06E-21
42	177000	3600	0	3.28E-05	2.77E-12	1.25E-15	2.65E-09	8.17E-13	8.44E-16	4.17E-13	2.90E-21	2.24E-21
43	180600	3600	0	3.27E-05	1.84E-12	7.92E-16	2.64E-09	7.16E-13	5.34E-16	2.75E-13	1.72E-21	1.09E-21
44	184200	3600	0	3.27E-05	1.23E-12	4.99E-16	2.64E-09	6.36E-13	3.36E-16	1.83E-13	9.38E-22	5.52E-22
45	187800	3600	0	3.27E-05	9.06E-13	3.41E-16	2.63E-09	6.20E-13	2.30E-16	1.34E-13	5.24E-22	2.48E-22
46	191400	3600	0	3.26E-05	6.30E-13	2.18E-16	2.63E-09	5.35E-13	1.47E-16	9.23E-14	2.88E-22	5.56E-23
47	195000	3600	0	3.26E-05	4.70E-13	1.49E-16	2.63E-09	5.47E-13	1.00E-16	6.85E-14	1.20E-22	3.97E-23
48	198600	3600	0	3.26E-05	3.53E-13	9.98E-17	2.62E-09	5.14E-13	6.73E-17	5.11E-14	3.69E-23	1.90E-23
49	202200	3600	0	3.26E-05	2.64E-13	6.62E-17	2.62E-09	5.09E-13	4.46E-17	3.76E-14	2.45E-23	7.15E-24
50	205800	3600	0	3.25E-05	2.03E-13	4.45E-17	2.62E-09	4.53E-13	3.00E-17	2.86E-14	1.17E-23	7.41E-25
51	209400	3600	0	3.25E-05	1.61E-13	3.09E-17	2.62E-09	4.97E-13	2.08E-17	2.23E-14	4.99E-24	2.12E-25
52	213000	3600	0	3.25E-05	1.19E-13	1.98E-17	2.62E-09	4.08E-13	1.33E-17	1.63E-14	1.36E-24	1.28E-25
53	216600	3600	0	3.25E-05	1.01E-13	1.44E-17	2.61E-09	4.10E-13	9.69E-18	1.37E-14	4.39E-25	1.05E-25
54	220200	3600	0	3.25E-05	8.17E-14	9.93E-18	2.61E-09	4.66E-13	6.69E-18	1.07E-14	8.50E-26	5.61E-28
55	223800	3600	0	3.24E-05	6.21E-14	6.46E-18	2.61E-09	3.91E-13	4.36E-18	8.03E-15	3.63E-26	3.73E-28
56	227400	3600	0	3.24E-05	4.86E-14	4.34E-18	2.61E-09	3.22E-13	2.93E-18	6.21E-15	3.22E-28	2.39E-28
57	231000	3600	0	3.24E-05	3.87E-14	2.97E-18	2.61E-09	2.72E-13	2.00E-18	4.89E-15	2.16E-28	1.11E-28
58	234600	3600	0	3.24E-05	3.80E-14	2.40E-18	2.61E-09	4.27E-13	1.62E-18	4.60E-15	1.21E-28	2.22E-38
59	238200	3600	0	3.24E-05	2.95E-14	1.57E-18	2.61E-09	4.31E-13	1.05E-18	3.43E-15	2.31E-31	2.51E-39
60	241800	3600	0	3.24E-05	2.36E-14	1.06E-18	2.60E-09	3.69E-13	7.13E-19	2.68E-15	1.73E-38	2.80E-40
61	245400	3600	0	3.23E-05	1.92E-14	7.29E-19	2.60E-09	3.17E-13	4.91E-19	2.14E-15	1.91E-39	3.09E-41
62	249000	3600	0	3.23E-05	1.57E-14	5.05E-19	2.60E-09	2.80E-13	3.40E-19	1.73E-15	2.06E-40	3.33E-42
63	252600	3600	0	3.23E-05	1.29E-14	3.51E-19	2.60E-09	2.53E-13	2.37E-19	1.40E-15	2.15E-41	3.48E-43

Table B-10: 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
64	256200	3600	0	3.23E-05	1.19E-14	2.68E-19	2.60E-09	2.98E-13	1.81E-19	1.26E-15	2.21E-42	3.64E-44
65-96	259800	3600	0	3.23E-05	1.34E-14	2.78E-19	2.60E-09	4.22E-13	1.88E-19	1.38E-15	9.04E-43	1.40E-44
Total	29400	345600	0	3.14E-03	2.33E-06	5.38E-08	2.04E-06	1.79E-06	9.16E-09	5.79E-07	1.97E-13	1.97E-13

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

The third RC consists of pipe breaks that bypass containment through the CVCS injection or discharge lines and are not isolated, allowing a direct release path to the Reactor Building, and the containment flooding and drain system (CFDS) is unavailable to provide coolant to the CNV. Deposition in the CVCS pipe is not credited for mitigation. The total release from the NPM is considered to be directly to the environment. The PRA sequences that have a significant contribution to RC 3 are shown in Table B-12. RC 3 is a bypassed containment scenario.

The conservative release sequence is a CVCS injection line pipe break 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 core support failure and core relocation to the RPV lower plenum. A significant portion of the core inventory of the Xe, Cs, I, and Te groups is released from the fuel and to the environment, due to the lack of crediting CVCS pipe deposition and Reactor Building retention. Because there is a direct release path to the environment modeled in MELCOR, all release information is calculated directly from MELCOR for RC 3. The particle size distribution for RC 3 is presented in Table B-11. The plume release information for RC 3 that is used as input for MACCS calculations is shown in Table B-13. The flow rate and density plume buoyancy model is used for RC 3 because hydrogen is generated during the accident and the flow rate and density model in MACCS better represents the plume buoyancy when releases contain hydrogen (Section 4.3.2 of Reference 8.1-11).

Table B-11: 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	1.06E-02	1.14E-02	1.50E-02	6.49E-03	6.36E-03	6.73E-03	6.53E-03	6.53E-03
0.29 µm	1.00E-01	1.65E-02	3.60E-03	1.11E-02	9.78E-03	2.65E-02	1.68E-02	2.57E-02	2.57E-02
0.53 µm	1.00E-01	9.58E-02	2.87E-02	6.73E-02	6.54E-02	1.63E-01	1.02E-01	1.59E-01	1.59E-01
0.99 µm	1.00E-01	2.71E-01	1.35E-01	2.64E-01	2.10E-01	3.38E-01	2.81E-01	3.28E-01	3.28E-01
1.84 µm	1.00E-01	3.16E-01	3.31E-01	3.44E-01	2.96E-01	2.55E-01	3.14E-01	2.53E-01	2.53E-01
3.43 µm	1.00E-01	1.71E-01	2.62E-01	1.77E-01	2.02E-01	1.13E-01	1.66E-01	1.18E-01	1.18E-01
6.38 µm	1.00E-01	6.74E-02	1.12E-01	6.69E-02	1.02E-01	5.01E-02	6.56E-02	5.48E-02	5.48E-02
11.9 µm	1.00E-01	2.80E-02	5.55E-02	2.89E-02	5.44E-02	2.56E-02	2.66E-02	2.88E-02	2.88E-02
22.1 µm	1.00E-01	1.26E-02	3.11E-02	1.38E-02	2.89E-02	1.30E-02	1.16E-02	1.49E-02	1.49E-02
41.2 µm	1.00E-01	1.09E-02	2.90E-02	1.16E-02	2.56E-02	9.18E-03	1.02E-02	1.20E-02	1.20E-02

Table B-12: PRA Sequences that Contribute Significantly to Release Category 3

Event Tree	Sequence	Description	FSAR Reference
CVCS--ALOCA-COC-ET	7	Reactor trip, LOCA unisolated, ECCS success, CFDS fails	Figure 19.1-2
	13	ATWS, LOCA unisolated, ECCS fails	
CVCS--ALOCA-LOC-ET	9	Reactor trip, LOCA unisolated, ECCS success, CVCS injection fails, CFDS fails	Figure 19.1-3

Table B-13: Release Category 3 Hourly Release Information

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Flow Rate (kg/s)	Plume Density (kg/m^3)	Chemical Group (fraction of initial core inventory)								
					Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
1	8640	3600	2.31E-01	2.06E-01	3.57E-01	2.81E-01	9.06E-03	1.62E-01	1.89E-01	1.05E-03	6.80E-02	2.07E-08	2.07E-08
2	12240	3600	1.25E-01	1.87E-01	7.67E-02	6.44E-02	5.59E-04	1.12E-01	9.05E-02	9.12E-04	1.65E-02	1.67E-08	1.67E-08
3	15840	3600	2.30E-01	1.64E-01	4.67E-02	3.70E-02	2.53E-04	4.12E-02	4.46E-02	6.87E-04	9.77E-03	1.19E-08	1.19E-08
4	19440	3600	3.93E-02	1.70E-01	2.75E-03	9.60E-04	3.68E-05	4.35E-03	3.06E-03	4.52E-06	8.87E-05	1.12E-10	1.11E-10
5	23040	3600	1.00E-06	2.66E-02	8.61E-06	5.19E-06	1.86E-08	2.71E-06	2.00E-06	6.89E-08	1.24E-06	1.68E-12	1.67E-12
6	26640	3600	2.05E-05	5.15E-02	8.07E-04	2.45E-04	9.97E-08	3.31E-05	9.98E-06	6.17E-06	7.14E-05	1.16E-10	1.16E-10
7	30240	3600	1.85E-05	8.89E-02	3.03E-04	2.00E-05	9.31E-09	5.60E-06	2.98E-08	3.56E-07	5.43E-06	6.69E-12	6.68E-12
8	33840	3660	9.22E-06	1.08E-01	9.84E-05	2.35E-06	9.31E-10	2.80E-06	5.96E-08	3.05E-08	5.59E-07	5.72E-13	5.86E-13
9	37500	3600	1.00E-06	2.28E-01	1.49E-07	5.96E-08	9.31E-10	2.68E-07	2.09E-07	0.00E+00	2.24E-08	5.68E-14	6.75E-14
10	41100	3600	1.00E-06	3.41E-01	5.96E-08	5.96E-08	9.31E-10	1.19E-07	5.96E-08	0.00E+00	1.49E-08	7.11E-15	1.07E-14
11	44700	3600	3.35E-05	3.43E-01	2.07E-05	6.08E-06	2.29E-07	2.91E-05	4.18E-05	0.00E+00	1.07E-06	1.89E-12	1.37E-12
12	48300	3600	3.66E-05	3.52E-01	3.56E-05	7.39E-06	5.53E-07	3.00E-05	1.61E-04	4.66E-10	1.28E-06	2.43E-12	3.10E-12
13	51900	3600	9.03E-06	3.54E-01	8.85E-06	1.07E-06	8.20E-08	3.70E-06	1.21E-05	0.00E+00	1.56E-07	3.06E-13	3.55E-13
14	55500	3280	6.92E-04	2.59E-01	6.97E-04	7.12E-05	3.62E-06	3.49E-04	3.21E-04	1.16E-09	1.25E-05	3.09E-12	3.23E-12
Total	8640	50140	-	-	4.85E-01	3.84E-01	9.92E-03	3.21E-01	3.28E-01	2.66E-03	9.44E-02	4.96E-08	4.96E-08

B.2.4 Release Category 4: ECCS Spurious Actuation

The fourth RC consists of loss of RPV inventory into containment due to the spurious opening of an RSV or ECCS valve. A spurious opening of an RSV is similar to RC 6, where an RSV opens and remains open; therefore, ECCS spurious actuation is considered in the representative RC 4 sequence. 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 RC. The PRA sequences that contribute to RC 4 are shown in Table B-14. RC 4 is an intact containment scenario.

The conservative release sequence for this RC is a spurious opening of an ECCS RVV, with a partial actuation of ECCS (only the remaining RVVs open) 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 core support failure and core relocation to the RPV lower plenum. Radionuclides released from the core relocate to the CNV, where the majority of aerosols deposit in the reactor coolant present in the CNV and on the CNV walls. In the MELCOR simulations, the CNV is modeled as sealed with no leakage. Therefore, the radionuclide release to the environment is calculated separately based on the fraction of the core inventory airborne in containment and the design basis containment leak rate of 0.2 weight percent per day (see FSAR Table 6.5-1).

The plume release information for RC 4 that was used as input for MACCS simulations is shown in Table B-15. As a result of modeling the CNV as sealed, assumptions relating to the particle size distribution and plume buoyancy are required inputs to MACCS. The particle size distribution for RC 4 was assumed to be the same as the particle size distribution for RC3 (see Table B-11). RC 3 is a containment bypass accident, and the particle size distribution of a containment bypass accident is expected to cause increased deposition in the environment as compared to the actual particle size distribution for an intact containment sequence. The plume buoyancy is simulated using the heat model with an assumed plume heat content of zero watts for each plume segment. The result of this assumption is that the plume does not rise upon release, which maximizes the ground level radionuclide concentrations. Sensitivity studies evaluating the impact of the aerosol deposition in the CNV predicted by MELCOR, the assumed particle size distribution, and the non-buoyant plume are discussed in Section B.3.

Table B-14: PRA Sequences that Contribute Significantly to Release Category 4

<table>
<tr><th>Event Tree</th><th>Sequence</th><th>Description</th><th>FSAR Reference</th></tr>
<tr><td>ECCS--ALOCA-RV1-ET</td><td>3</td><td>Reactor trip, ECCS fails, CVCS injection fails</td><td>Figure 19.1-6</td></tr>
<tr><td>RCS---ALOCA-IC--ET</td><td>3</td><td>Reactor trip, ECCS fails, CVCS injection fails</td><td>Figure 19.1-5</td></tr>
<tr><td>EDSS--LODC-----ET</td><td>3</td><td>Reactor trip, DHRS success, ECCS fails, CVCS injection fails</td><td>Figure 19.1-10</td></tr>
<tr><td rowspan="2">EHVS--LOOP-----ET</td><td>5</td><td>CTG fails, BDG fails, Reactor trip, DHRS success, Off-site power not recovered, ECCS fails</td><td rowspan="2">Figure 19.1-9</td></tr>
<tr><td>8</td><td>CTG fails, BDG fails, Reactor trip, DHRS success, RSV opens, RSV cycles, Off-site power not recovered, ECCS fails</td></tr>
</table>

Table B-15: 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	26160	3600	0	2.84E-06	3.27E-07	4.73E-08	2.85E-07	2.99E-07	6.71E-11	2.91E-08	4.52E-15	4.51E-15
2	29760	3600	0	1.74E-05	1.34E-06	6.61E-09	6.94E-07	8.01E-07	4.24E-09	3.74E-07	8.11E-14	8.11E-14
3	33360	3600	0	2.48E-05	9.94E-08	1.21E-10	1.20E-07	5.62E-08	9.38E-10	2.58E-08	2.02E-14	2.02E-14
4	36960	3600	0	2.66E-05	8.60E-08	5.37E-10	1.82E-07	1.29E-07	9.62E-10	2.14E-08	2.16E-14	2.16E-14
5	40560	3600	0	2.84E-05	6.91E-08	9.77E-10	8.91E-08	1.33E-07	6.24E-10	1.77E-08	1.51E-14	1.51E-14
6	44160	3600	0	3.16E-05	3.85E-08	1.48E-10	2.74E-08	3.51E-08	4.66E-10	1.07E-08	9.40E-15	9.40E-15
7	47760	3600	0	3.62E-05	7.34E-08	3.49E-10	1.12E-07	5.25E-08	4.90E-10	1.72E-08	2.47E-14	2.48E-14
8	51360	3600	0	3.98E-05	2.17E-08	1.27E-09	3.64E-08	5.29E-08	1.00E-12	5.26E-09	2.53E-15	2.99E-15
9	54960	3600	0	3.91E-05	1.16E-08	5.52E-10	2.39E-08	2.32E-08	3.51E-13	2.71E-09	1.20E-15	1.42E-15
10	58560	3600	0	3.81E-05	4.84E-09	1.82E-10	1.26E-08	1.01E-08	1.22E-13	1.04E-09	4.11E-16	4.85E-16
11	62160	3600	0	3.79E-05	1.77E-09	5.70E-11	7.06E-09	3.38E-09	3.53E-14	3.35E-10	1.17E-16	1.37E-16
12	65760	3600	0	3.78E-05	6.76E-10	2.10E-11	4.84E-09	1.13E-09	2.75E-14	1.09E-10	3.40E-17	3.85E-17
13	69360	3600	0	3.76E-05	2.74E-10	6.77E-12	3.89E-09	3.93E-10	2.80E-14	3.43E-11	9.43E-18	1.04E-17
14	72960	3600	0	3.76E-05	1.28E-10	1.98E-12	3.47E-09	1.43E-10	1.50E-14	1.06E-11	2.56E-18	2.80E-18
15	76560	3600	0	3.75E-05	6.78E-11	5.75E-13	3.27E-09	5.37E-11	6.27E-15	3.30E-12	7.13E-19	7.72E-19
16	80160	3600	0	3.74E-05	3.88E-11	1.71E-13	3.17E-09	2.04E-11	2.38E-15	1.05E-12	2.04E-19	2.20E-19
17	83760	3840	0	3.94E-05	4.94E-11	6.30E-14	3.33E-09	1.22E-11	1.27E-15	4.43E-13	7.03E-20	7.46E-20
18	87600	3600	0	3.70E-05	9.65E-11	2.56E-14	3.18E-09	1.58E-11	7.92E-16	2.32E-13	2.61E-20	2.71E-20
19	91200	3600	0	3.68E-05	5.62E-11	9.09E-15	3.12E-09	9.10E-12	3.37E-16	9.51E-14	8.81E-21	8.93E-21
20	94800	3600	0	3.65E-05	3.32E-11	2.64E-15	3.05E-09	3.38E-12	1.04E-16	2.95E-14	2.42E-21	2.27E-21
21	98400	3600	0	3.63E-05	3.99E-11	8.99E-16	3.04E-09	2.14E-12	4.21E-17	1.21E-14	6.82E-22	4.83E-22
22	102000	3600	0	3.63E-05	6.79E-11	3.58E-16	3.05E-09	2.31E-12	2.25E-17	7.10E-15	1.39E-22	5.65E-23
23	105600	3600	0	3.61E-05	6.00E-11	1.28E-16	3.03E-09	1.62E-12	1.02E-17	3.59E-15	1.44E-23	1.03E-28
24	109200	3600	0	3.60E-05	9.10E-11	5.26E-17	3.05E-09	1.67E-12	6.21E-18	2.27E-15	4.35E-30	0.00E+00
25	112800	3600	0	3.59E-05	1.43E-10	2.11E-17	3.08E-09	1.85E-12	3.92E-18	1.85E-15	0.00E+00	0.00E+00
26	116400	3600	0	3.57E-05	1.88E-10	7.44E-18	3.11E-09	1.76E-12	1.70E-18	1.73E-15	0.00E+00	0.00E+00
27	120000	3600	0	3.56E-05	2.33E-10	2.42E-18	3.14E-09	1.54E-12	5.93E-19	9.95E-16	0.00E+00	0.00E+00
28	123600	3600	0	3.55E-05	3.12E-10	7.55E-19	3.20E-09	1.43E-12	1.91E-19	4.81E-16	0.00E+00	0.00E+00
29	127200	3600	0	3.53E-05	2.60E-10	2.23E-19	3.30E-09	1.37E-12	5.63E-20	2.64E-16	0.00E+00	0.00E+00
30	130800	3600	0	3.52E-05	1.37E-10	6.27E-20	3.40E-09	1.26E-12	1.55E-20	1.67E-16	0.00E+00	0.00E+00
31	134400	3600	0	3.50E-05	7.81E-11	1.76E-20	3.34E-09	9.74E-13	4.24E-21	1.00E-16	0.00E+00	0.00E+00

Table B-15: 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
32	138000	3600	0	3.49E-05	5.61E-11	5.37E-21	3.29E-09	7.61E-13	9.27E-22	6.04E-17	0.00E+00	0.00E+00
33	141600	3600	0	3.48E-05	4.82E-11	1.32E-21	3.27E-09	6.95E-13	1.33E-22	4.20E-17	0.00E+00	0.00E+00
34	145200	3600	0	3.47E-05	4.03E-11	5.31E-22	3.21E-09	6.58E-13	1.93E-24	2.97E-17	0.00E+00	0.00E+00
35	148800	3600	0	3.46E-05	3.45E-11	2.85E-22	3.17E-09	7.15E-13	0.00E+00	3.74E-17	0.00E+00	0.00E+00
36	152400	3600	0	3.46E-05	5.30E-11	2.12E-14	3.18E-09	1.83E-12	1.48E-14	5.23E-12	1.53E-19	1.50E-19
37	156000	3600	0	3.45E-05	1.07E-10	4.89E-14	3.10E-09	2.68E-12	3.41E-14	2.22E-11	3.53E-19	3.48E-19
38	159600	3600	0	3.45E-05	8.89E-11	4.47E-14	3.02E-09	2.47E-12	3.12E-14	1.95E-11	3.23E-19	3.18E-19
39	163200	3600	0	3.45E-05	6.38E-11	3.29E-14	2.97E-09	1.84E-12	2.29E-14	1.41E-11	2.37E-19	2.34E-19
40	166800	3600	0	3.44E-05	4.86E-11	2.51E-14	2.94E-09	1.43E-12	1.75E-14	1.07E-11	1.81E-19	1.78E-19
41	170400	3600	0	3.44E-05	3.84E-11	1.97E-14	2.92E-09	1.14E-12	1.37E-14	8.36E-12	1.42E-19	1.40E-19
42	174000	3600	0	3.44E-05	3.16E-11	1.60E-14	2.91E-09	9.46E-13	1.12E-14	6.78E-12	1.16E-19	1.14E-19
43	177600	3600	0	3.43E-05	2.63E-11	1.32E-14	2.90E-09	7.95E-13	9.19E-15	5.56E-12	9.50E-20	9.35E-20
44	181200	3600	0	3.43E-05	2.21E-11	1.09E-14	2.89E-09	6.74E-13	7.62E-15	4.60E-12	7.87E-20	7.74E-20
45	184800	3600	0	3.43E-05	1.86E-11	9.03E-15	2.89E-09	5.70E-13	6.30E-15	3.79E-12	6.51E-20	6.40E-20
46	188400	3600	0	3.43E-05	1.75E-11	8.40E-15	2.88E-09	5.45E-13	5.86E-15	3.51E-12	6.06E-20	5.96E-20
47	192000	3600	0	3.42E-05	1.48E-11	7.05E-15	2.88E-09	4.70E-13	4.92E-15	2.94E-12	5.08E-20	5.00E-20
48	195600	3600	0	3.42E-05	1.28E-11	6.04E-15	2.87E-09	4.15E-13	4.22E-15	2.51E-12	4.35E-20	4.28E-20
49	199200	3600	0	3.42E-05	1.13E-11	5.32E-15	2.87E-09	3.77E-13	3.71E-15	2.20E-12	3.83E-20	3.77E-20
50	202800	3600	0	3.42E-05	9.88E-12	4.60E-15	2.86E-09	3.38E-13	3.21E-15	1.90E-12	3.32E-20	3.27E-20
51	206400	3600	0	3.41E-05	8.34E-12	3.86E-15	2.86E-09	2.95E-13	2.69E-15	1.59E-12	2.78E-20	2.74E-20
52	210000	3600	0	3.41E-05	7.19E-12	3.31E-15	2.86E-09	2.63E-13	2.31E-15	1.36E-12	2.38E-20	2.35E-20
53	213600	3600	0	3.41E-05	6.08E-12	2.78E-15	2.85E-09	2.31E-13	1.94E-15	1.14E-12	2.00E-20	1.97E-20
54	217200	3600	0	3.41E-05	5.55E-12	2.52E-15	2.85E-09	2.21E-13	1.76E-15	1.03E-12	1.82E-20	1.79E-20
55	220800	3600	0	3.41E-05	5.23E-12	2.35E-15	2.85E-09	2.20E-13	1.64E-15	9.65E-13	1.70E-20	1.67E-20
56	224400	3600	0	3.40E-05	4.36E-12	1.94E-15	2.85E-09	1.94E-13	1.36E-15	7.96E-13	1.40E-20	1.38E-20
57	228000	3600	0	3.40E-05	3.52E-12	1.56E-15	2.84E-09	1.65E-13	1.09E-15	6.39E-13	1.12E-20	1.10E-20
58	231600	3600	0	3.40E-05	3.14E-12	1.38E-15	2.84E-09	1.55E-13	9.61E-16	5.65E-13	9.93E-21	9.76E-21
59	235200	3600	0	3.40E-05	3.10E-12	1.35E-15	2.84E-09	1.67E-13	9.39E-16	5.53E-13	9.68E-21	9.51E-21
60	238800	3600	0	3.40E-05	2.57E-12	1.10E-15	2.84E-09	1.48E-13	7.70E-16	4.55E-13	7.93E-21	7.77E-21
61	242400	3600	0	3.39E-05	2.09E-12	8.90E-16	2.84E-09	1.27E-13	6.21E-16	3.68E-13	6.38E-21	6.23E-21
62	246000	3600	0	3.39E-05	2.20E-12	9.18E-16	2.83E-09	1.49E-13	6.41E-16	3.82E-13	6.59E-21	6.43E-21
63	249600	3600	0	3.39E-05	1.46E-12	6.04E-16	2.83E-09	1.03E-13	4.21E-16	2.52E-13	4.29E-21	4.17E-21

Table B-15: 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
64	253200	3600	0	3.39E-05	4.38E-12	1.72E-15	2.83E-09	4.49E-13	1.20E-15	7.33E-13	1.23E-20	1.20E-20
65	256800	3600	0	3.39E-05	7.37E-12	2.68E-15	2.84E-09	1.10E-12	1.87E-15	1.18E-12	1.93E-20	1.88E-20
66-96	260400	3600	0	3.39E-05	8.10E-12	2.88E-15	2.84E-09	1.32E-12	2.01E-15	1.28E-12	2.07E-20	2.02E-20
Total	26160	345840	0	3.27E-03	2.08E-06	5.81E-08	1.84E-06	1.60E-06	7.79E-09	5.05E-07	1.81E-13	1.82E-13

B.2.5 Release Category 5: Steam Generator Tube Failure

The fifth RC consists of SGTFs that are unisolated for which both CVCS injection and CFDS injection fail to supply additional coolant 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 RC. The PRA sequences that contribute to RC 5 are shown in Table B-16. RC 5 is a bypassed containment scenario.

The conservative release sequence for this RC is an SGTF with the complete failure of all mitigating systems, including failure of all five ECCS valves to open. A secondary line break, which is not a failure mode in the RC 5 PRA sequences, is also modeled for convenience to allow a direct release path from the NPM. The reactor coolant bypasses containment through the SGTF and secondary line break, completely uncovering the core leading to core support failure and core relocation to the RPV lower plenum. A significant portion of the core inventory of Xe, Cs, I, and Te groups is released from the fuel and to the environment due to the lack of crediting Reactor Building retention. Because there is a direct release path to the environment modeled in MELCOR, all release information is calculated directly from MELCOR for RC 5. The particle size distribution for RC 5 is presented in Table B-17. The plume release information for RC 5 that is used as input for MACCS calculations is shown in Table B-18. The flow rate and density plume buoyancy model in MACCS is used for RC 5 due to the hydrogen generated during the accident. The flow rate and density model better represents the plume buoyancy when releases contain hydrogen (Section 4.3.2 of Reference 8.1-11).

Table B-16: PRA Sequences that Contribute Significantly to Release Category 5

Event Tree	Sequence	Description	FSAR Reference
MSS---ALOCA-SG--ET	9	Reactor trip. LOCA unisolated, ECCS success, CVCS injection fails, CFDS fails	Figure 19.1-7
	18	ATWS, LOCA unisolated, ECCS fails, CVCS injection fails	

Table B-17: 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	1.93E-01	4.75E-02	1.80E-01	2.65E-01	4.28E-01	2.34E-01	3.91E-01	3.89E-01
0.29 µm	1.00E-01	5.49E-02	1.52E-02	3.87E-02	8.93E-02	7.42E-02	6.62E-02	6.77E-02	6.72E-02
0.53 µm	1.00E-01	8.18E-02	3.70E-02	1.15E-01	6.47E-02	4.49E-02	7.65E-02	4.64E-02	4.52E-02
0.99 µm	1.00E-01	3.12E-01	3.42E-01	3.60E-01	2.78E-01	2.24E-01	2.91E-01	2.62E-01	2.65E-01
1.84 µm	1.00E-01	3.10E-01	4.94E-01	2.76E-01	2.66E-01	2.00E-01	2.82E-01	2.04E-01	2.05E-01
3.43 µm	1.00E-01	4.70E-02	6.38E-02	3.01E-02	3.62E-02	2.67E-02	4.81E-02	2.62E-02	2.61E-02
6.38 µm	1.00E-01	1.49E-03	8.10E-04	9.11E-04	1.30E-03	1.69E-03	1.76E-03	1.56E-03	1.55E-03
11.9 µm	1.00E-01	1.28E-04	1.93E-04	1.59E-04	3.53E-04	4.11E-04	1.43E-04	4.30E-04	4.37E-04
22.1 µm	1.00E-01	9.75E-06	1.37E-05	2.33E-05	4.08E-05	2.62E-05	9.78E-06	4.51E-05	4.79E-05
41.2 µm	1.00E-01	1.34E-06	1.07E-06	1.73E-06	3.23E-06	3.61E-06	1.53E-06	3.88E-06	3.97E-06

Table B-18: Release Category 5 Hourly Release Information

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Flow Rate (kg/s)	Plume Density (kg/m^3)	Chemical Group (fraction of initial core inventory)								
					Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
1	44340	3660	3.48E-01	3.89E-01	1.90E-01	1.63E-02	1.28E-03	9.84E-03	9.41E-03	1.88E-05	2.68E-03	4.05E-10	4.04E-10
2	48000	3540	1.78E-01	3.57E-01	1.87E-01	9.51E-03	1.29E-05	9.20E-03	4.65E-03	5.42E-05	2.58E-03	9.80E-10	9.79E-10
3	51540	3660	2.13E-01	4.67E-01	4.89E-02	3.20E-03	3.80E-05	1.51E-03	7.00E-03	2.81E-05	9.04E-04	5.52E-10	5.52E-10
4	55200	3580	1.37E-01	4.03E-01	1.18E-02	6.47E-04	4.75E-06	8.80E-04	7.00E-04	1.88E-05	1.68E-04	3.72E-10	3.71E-10
5	58780	3600	8.36E-02	4.92E-01	1.42E-02	2.01E-04	1.56E-06	1.12E-04	1.97E-04	5.60E-06	5.65E-05	1.06E-10	1.06E-10
6	62380	3600	5.37E-01	5.30E-01	6.02E-02	7.71E-04	1.82E-06	6.55E-04	5.11E-04	1.12E-05	2.17E-04	2.16E-10	2.16E-10
7	65980	3600	2.57E-01	4.53E-01	8.56E-03	4.78E-04	2.45E-06	5.03E-04	2.46E-04	1.28E-06	1.21E-04	4.16E-11	4.24E-11
8	69580	3600	7.47E-02	3.71E-01	2.12E-03	3.37E-04	1.05E-06	1.84E-04	2.44E-04	6.25E-07	9.32E-05	2.96E-11	2.94E-11
9	73180	3600	2.53E-02	5.12E-01	1.66E-03	5.58E-05	5.95E-08	3.82E-05	7.34E-06	2.15E-07	1.56E-05	6.96E-12	7.05E-12
10	76780	3600	3.99E-02	5.44E-01	3.02E-03	1.18E-04	7.54E-08	4.62E-05	1.16E-05	7.27E-07	3.38E-05	1.07E-11	1.07E-11
11	80380	3600	3.66E-02	5.50E-01	1.80E-03	8.34E-05	4.13E-08	3.29E-05	1.01E-05	7.92E-07	2.39E-05	4.10E-12	4.07E-12
12	83980	3620	2.74E-02	5.35E-01	3.16E-04	1.64E-07	1.16E-10	7.10E-07	3.35E-08	2.02E-09	4.70E-08	1.04E-14	9.77E-15
13	87600	3600	2.45E-02	4.89E-01	3.15E-04	1.49E-08	0.00E+00	7.28E-07	3.73E-09	5.09E-10	5.59E-09	2.22E-15	2.22E-15
14	91200	3600	1.05E-01	4.32E-01	2.50E-02	6.24E-04	3.66E-05	4.34E-03	3.12E-03	2.70E-06	9.66E-05	3.70E-10	4.06E-10
15	94800	3600	1.32E-01	3.46E-01	1.38E-03	1.03E-04	6.90E-07	2.05E-04	1.28E-05	7.52E-09	2.09E-05	1.09E-11	1.17E-11
16	98400	3600	2.49E-02	3.42E-01	5.03E-05	8.33E-06	1.35E-08	4.10E-05	3.88E-05	4.09E-09	1.17E-06	3.15E-13	3.77E-13
17	102000	3600	2.54E-02	6.62E-01	2.75E-03	3.71E-05	1.21E-06	1.84E-04	1.26E-04	1.01E-07	6.26E-06	1.07E-11	1.17E-11
18	105600	3600	4.37E-03	7.34E-01	4.92E-04	3.83E-05	8.52E-08	5.54E-05	7.88E-06	9.78E-09	7.98E-06	1.31E-11	8.44E-13
19	109200	3600	4.60E-03	5.81E-01	3.05E-04	6.73E-05	1.32E-07	5.03E-05	7.27E-06	7.59E-08	1.68E-05	2.53E-12	2.33E-12
20	112800	3600	4.08E-03	6.20E-01	2.39E-04	4.03E-05	1.06E-07	2.55E-05	2.95E-06	2.41E-08	1.06E-05	7.82E-13	7.70E-13
21	116400	3600	3.82E-03	5.57E-01	9.34E-05	7.65E-05	6.71E-08	6.96E-05	1.45E-06	1.41E-08	8.91E-06	4.22E-13	4.28E-13
22	120000	3600	7.33E-03	5.10E-01	7.36E-05	2.13E-04	1.87E-07	2.58E-04	5.57E-06	5.45E-08	4.96E-05	1.66E-12	1.69E-12
23	123600	3600	1.22E-02	4.59E-01	2.28E-05	6.72E-04	1.13E-06	9.88E-04	8.21E-05	4.59E-07	1.60E-04	1.53E-11	1.55E-11
24	127200	3600	1.37E-02	4.28E-01	5.90E-06	7.82E-04	1.33E-06	1.27E-03	1.42E-04	6.69E-07	1.82E-04	2.26E-11	2.29E-11
25	130800	3600	1.39E-02	4.19E-01	1.85E-06	7.89E-04	6.85E-07	1.22E-03	1.45E-04	5.62E-07	1.80E-04	1.87E-11	1.90E-11
26	134400	3600	1.40E-02	4.18E-01	8.34E-07	7.27E-04	1.76E-07	1.08E-03	1.69E-04	2.80E-07	1.61E-04	1.02E-11	1.03E-11
27	138000	3600	1.41E-02	4.14E-01	2.98E-07	7.26E-04	9.46E-07	8.20E-04	1.71E-04	1.01E-06	1.78E-04	3.31E-11	3.36E-11
28	141600	3600	1.58E-02	4.11E-01	1.19E-07	6.07E-04	6.02E-07	7.77E-04	2.78E-04	6.53E-07	1.37E-04	2.16E-11	2.19E-11
29	145200	3600	1.44E-02	4.10E-01	0.00E+00	1.77E-04	2.78E-08	3.79E-04	3.18E-04	2.47E-08	2.50E-05	8.19E-13	8.32E-13
30	148800	3600	1.10E-02	4.23E-01	0.00E+00	6.53E-05	5.03E-08	1.04E-04	2.12E-04	3.10E-08	4.89E-06	8.92E-13	9.06E-13

Table B-18: Release Category 5 Hourly Release Information (Continued)

Plume Segment	Start of Release (s)	Release Duration (s)	Plume Flow Rate (kg/s)	Plume Density (kg/m^3)	Chemical Group (fraction of initial core inventory)								
					Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La
31	152400	3600	6.62E-03	4.58E-01	0.00E+00	3.71E-05	2.03E-08	4.64E-05	2.23E-04	1.31E-08	5.31E-06	3.73E-13	3.79E-13
32	156000	3600	2.12E-03	5.22E-01	1.79E-07	3.07E-06	5.82E-10	4.38E-06	8.44E-05	3.78E-10	7.45E-08	1.04E-14	1.07E-14
33	159600	3600	1.49E-03	7.39E-01	1.07E-04	2.16E-07	6.98E-10	1.71E-06	3.74E-07	1.46E-10	1.21E-08	4.22E-15	4.00E-15
34	163200	3600	4.22E-03	5.19E-01	3.34E-05	3.96E-05	1.16E-06	4.54E-05	1.90E-07	5.38E-10	2.24E-08	1.53E-14	1.60E-14
35	166800	3600	3.96E-03	5.20E-01	7.51E-06	5.71E-06	6.54E-07	1.25E-05	5.59E-08	1.46E-11	9.31E-10	4.44E-16	0.00E+00
36	170400	3600	9.29E-04	5.43E-01	2.56E-06	2.73E-06	4.66E-10	1.95E-06	1.30E-08	0.00E+00	9.31E-10	0.00E+00	0.00E+00
37	174000	3600	2.28E-02	5.54E-01	1.25E-04	9.93E-06	1.98E-09	1.50E-05	6.33E-08	8.73E-11	4.19E-08	2.00E-15	2.22E-15
38	177600	3600	4.46E-04	5.04E-01	2.62E-06	1.68E-07	1.16E-10	1.42E-07	7.45E-09	0.00E+00	9.31E-10	2.22E-16	4.44E-16
39	181200	3600	5.92E-03	5.98E-01	2.05E-04	1.69E-06	6.98E-10	4.11E-06	1.86E-08	5.82E-11	1.68E-08	1.33E-15	1.33E-15
40	184800	3600	1.02E-02	5.06E-01	6.04E-05	2.85E-06	0.00E+00	4.59E-06	9.31E-09	0.00E+00	2.61E-08	0.00E+00	0.00E+00
41	188400	3600	2.38E-03	5.40E-01	2.92E-06	2.98E-07	0.00E+00	5.18E-07	0.00E+00	0.00E+00	9.31E-10	2.22E-16	0.00E+00
42	195600	3600	6.48E-04	8.28E-01	3.62E-05	3.73E-09	1.16E-10	3.84E-07	1.86E-09	1.46E-11	0.00E+00	0.00E+00	4.44E-16
43	199200	3600	8.51E-03	5.17E-01	1.56E-04	2.65E-07	0.00E+00	2.69E-06	1.86E-09	0.00E+00	4.28E-08	0.00E+00	0.00E+00
44	202800	3600	5.86E-04	5.45E-01	7.33E-06	1.12E-08	0.00E+00	1.08E-07	0.00E+00	0.00E+00	9.31E-10	2.22E-16	0.00E+00
45	206400	3600	8.47E-04	7.91E-01	4.01E-05	3.73E-09	1.16E-10	3.99E-07	1.86E-09	2.91E-11	9.31E-10	2.22E-16	4.44E-16
46	210000	3600	2.60E-03	6.45E-01	4.27E-05	2.98E-08	4.66E-10	5.33E-07	1.12E-08	5.82E-11	3.73E-09	1.33E-15	1.11E-15
47	213600	3600	2.45E-03	6.12E-01	1.65E-05	3.73E-09	1.16E-10	1.71E-07	1.86E-09	1.46E-11	9.31E-10	2.22E-16	2.22E-16
48	217200	3600	3.76E-04	6.03E-01	2.38E-06	0.00E+00	0.00E+00	2.61E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
49	220800	3600	3.23E-03	6.14E-01	7.46E-05	1.01E-07	5.82E-10	7.45E-07	1.30E-08	5.82E-11	2.70E-08	1.55E-15	1.33E-15
50	224400	3600	3.09E-03	5.54E-01	1.58E-05	7.08E-08	0.00E+00	1.68E-07	0.00E+00	0.00E+00	2.05E-08	0.00E+00	0.00E+00
51	228000	3600	1.80E-03	5.68E-01	8.40E-06	3.73E-09	0.00E+00	8.57E-08	1.86E-09	1.46E-11	9.31E-10	2.22E-16	4.44E-16
52	231600	3600	6.09E-04	5.90E-01	4.35E-06	0.00E+00	0.00E+00	4.10E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
53	235200	3600	1.29E-03	7.04E-01	4.84E-05	3.58E-07	3.49E-10	4.77E-07	7.45E-09	4.37E-11	1.04E-07	8.88E-16	6.66E-16
54	238800	3600	8.23E-04	8.05E-01	4.74E-05	1.49E-08	1.16E-10	4.69E-07	1.86E-09	1.46E-11	3.73E-09	4.44E-16	4.44E-16
55	242400	3600	9.73E-03	6.53E-01	1.60E-04	2.12E-07	6.98E-10	4.91E-06	1.68E-08	7.28E-11	6.24E-08	1.78E-15	2.00E-15
56	246000	3600	1.08E-03	6.54E-01	4.79E-05	1.12E-08	2.33E-10	5.96E-07	5.59E-09	4.37E-11	1.86E-09	6.66E-16	6.66E-16
57	249600	3600	9.95E-04	7.04E-01	4.43E-05	5.22E-08	2.33E-10	5.33E-07	3.73E-09	1.46E-11	1.40E-08	6.66E-16	6.66E-16
58	253200	3600	1.18E-03	8.00E-01	1.19E-04	2.61E-08	2.33E-10	1.44E-06	5.59E-09	4.37E-11	7.45E-09	6.66E-16	4.44E-16
59	256800	2400	1.63E-03	7.35E-01	1.76E-04	1.12E-08	2.33E-10	2.34E-06	1.86E-09	1.46E-11	1.86E-09	4.44E-16	8.88E-16
Total	44340	211260	-	-	5.63E-01	3.76E-02	1.39E-03	3.50E-02	2.81E-02	1.47E-04	8.14E-03	3.26E-09	3.28E-09

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

The sixth RC 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 RC. The PRA sequences that contribute to RC 6 are shown in Table B-19. RC 6 is an intact containment scenario.

The conservative release sequence for this RC is 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 core support failure and core relocation to the RPV lower plenum. Radionuclides released from the core relocate to the CNV, where the majority of aerosols deposit in the reactor coolant present in the CNV and on the CNV walls. In the MELCOR simulations, the CNV is modeled as sealed with no leakage. Therefore, the radionuclide release to the environment is calculated separately based on the fraction of the core inventory airborne in containment and the design basis containment leak rate of 0.2 weight percent per day (see FSAR Table 6.5-1).

The plume release information for RC 6 that was used as input for MACCS simulations is shown in Table B-20. As a result of modeling the CNV as sealed, assumptions relating to the particle size distribution and plume buoyancy are required inputs to MACCS. The particle size distribution for RC 6 was assumed to be the same as the particle size distribution for RC 3 (see Table B-11). RC 3 is a containment bypass accident, and the particle size distribution of a containment bypass accident is expected to cause increased deposition in the environment as compared to the actual particle size distribution for an intact containment sequence. The plume buoyancy is simulated using the heat model with an assumed plume heat content of zero watts for each plume segment. The result of this assumption is that the plume does not rise upon release, which maximizes the ground level radionuclide concentrations. Sensitivity studies evaluating the impact of the aerosol deposition in the CNV predicted by MELCOR, the assumed particle size distribution, and the non-buoyant plume are discussed in Section B.3.

Table B-19: PRA Sequences that Contribute to Significantly Release Category 6

Event Tree	Sequence	Description	FSAR Reference
EHVS--LOOP-----ET	1-5	Reactor trip, DHRS success, RSV opens, RSV stays open, ECCS fails, CVCS injection fails	Figure 19.1-9
	2-5	CTG fails, Reactor trip, DHRS success, RSV opens, RSV stays open, ECCS fails, CVCS injection fails	
	10	CTG fails, BDG fails, Reactor trip, DHRS success, RSV opens, RSV stays open, ECCS fails	
TRN---TRAN--NPC-ET	5	Reactor trip, DHRS success, RSV opens, RSV stays open, ECCS fails, CVCS injection fails	Figure 19.1-11
	9	Reactor trip, DHRS fails, RSV opens, RSV stays open, ECCS fails, CVCS injection fails	
TGS---TRAN--NSS-ET	4	Reactor trip, DHRS success, RSV opens, RSV stays open, ECCS fails	Figure 19.1-12

Table B-20: Release Category 6 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	36600	3600	0	2.42E-06	2.99E-07	3.68E-08	2.45E-07	2.55E-07	1.00E-10	3.29E-08	5.00E-15	5.00E-15
2	40200	3600	0	1.04E-05	4.74E-07	3.44E-09	2.93E-07	2.84E-07	1.38E-09	1.30E-07	2.56E-14	2.56E-14
3	43800	3600	0	2.57E-05	4.18E-07	5.82E-10	2.22E-07	2.92E-07	2.93E-09	1.18E-07	5.38E-14	5.39E-14
4	47400	3600	0	2.89E-05	5.04E-08	1.99E-10	1.71E-07	4.92E-08	6.16E-10	1.11E-08	1.43E-14	1.44E-14
5	51000	3600	0	2.99E-05	4.52E-08	1.07E-09	8.48E-08	1.45E-07	6.27E-10	1.11E-08	1.65E-14	1.64E-14
6	54600	3600	0	3.10E-05	4.19E-08	3.20E-10	5.82E-08	4.87E-08	2.80E-10	1.09E-08	1.25E-14	1.25E-14
7	58200	3600	0	3.19E-05	3.47E-08	1.73E-10	3.87E-08	2.34E-08	1.71E-10	9.28E-09	9.00E-15	9.03E-15
8	61800	3600	0	3.38E-05	6.04E-08	2.73E-10	9.64E-08	4.61E-08	2.19E-10	1.46E-08	1.86E-14	1.87E-14
9	65400	3600	0	3.61E-05	7.74E-09	1.52E-10	1.51E-08	3.08E-08	1.85E-13	1.82E-09	2.55E-16	2.55E-16
10	69000	3600	0	3.58E-05	1.83E-08	5.60E-10	5.04E-08	4.81E-08	1.37E-13	4.00E-09	4.70E-16	4.70E-16
11	72600	3600	0	3.45E-05	1.78E-08	3.55E-10	1.09E-07	1.70E-08	8.12E-14	2.86E-09	3.23E-16	3.22E-16
12	76200	3600	0	3.40E-05	7.82E-09	9.53E-11	6.23E-08	6.38E-09	4.20E-14	9.31E-10	1.00E-16	9.95E-17
13	79800	3600	0	3.41E-05	2.30E-09	2.40E-11	1.88E-08	1.78E-09	3.24E-14	2.64E-10	2.91E-17	2.87E-17
14	83400	3600	0	3.44E-05	7.09E-10	6.39E-12	7.12E-09	5.13E-10	2.33E-14	7.89E-11	8.93E-18	8.71E-18
15	87000	3600	0	3.44E-05	2.24E-10	1.80E-12	3.95E-09	1.62E-10	1.20E-14	2.41E-11	2.72E-18	2.64E-18
16	90600	3600	0	3.43E-05	8.06E-11	5.42E-13	3.06E-09	5.69E-11	5.20E-15	7.77E-12	8.74E-19	8.42E-19
17	94200	3600	0	3.42E-05	3.05E-11	1.66E-13	2.77E-09	2.02E-11	1.96E-15	2.49E-12	2.78E-19	2.67E-19
18	97800	3600	0	3.39E-05	1.33E-11	5.36E-14	2.66E-09	7.67E-12	7.52E-16	8.43E-13	9.36E-20	8.93E-20
19	101400	3600	0	3.39E-05	5.85E-12	1.78E-14	2.63E-09	2.83E-12	2.78E-16	2.95E-13	3.18E-20	3.03E-20
20	105000	3600	0	3.38E-05	3.35E-12	6.54E-15	2.61E-09	1.19E-12	1.14E-16	1.22E-13	1.19E-20	1.09E-20
21	108600	3600	0	3.37E-05	2.46E-12	2.80E-15	2.60E-09	5.87E-13	5.39E-17	6.83E-14	4.95E-21	4.09E-21
22	112200	3600	0	3.41E-05	1.15E-11	1.27E-14	2.65E-09	3.01E-12	6.21E-17	1.29E-12	4.86E-21	3.91E-21
23	115800	3600	0	3.40E-05	1.80E-11	2.22E-14	2.66E-09	5.35E-12	5.69E-17	2.47E-12	4.15E-21	3.36E-21
24	119400	3600	0	3.38E-05	1.50E-11	1.92E-14	2.66E-09	5.19E-12	2.00E-17	2.24E-12	5.26E-21	4.68E-21
25	123000	3600	0	3.36E-05	1.03E-11	1.31E-14	2.64E-09	3.93E-12	7.34E-18	1.57E-12	1.66E-19	1.65E-19
26	126600	3600	0	3.35E-05	1.02E-11	1.33E-14	2.64E-09	4.67E-12	4.06E-18	1.64E-12	5.68E-19	5.68E-19
27	130200	3600	0	3.35E-05	1.27E-11	1.75E-14	2.63E-09	6.81E-12	4.00E-18	2.20E-12	1.05E-18	1.05E-18
28	133800	3600	0	3.34E-05	1.53E-11	2.20E-14	2.63E-09	8.93E-12	4.85E-18	2.78E-12	1.45E-18	1.45E-18
29	137400	3600	0	3.33E-05	1.70E-11	2.46E-14	2.63E-09	1.01E-11	5.55E-18	3.10E-12	1.69E-18	1.69E-18
30	141000	3600	0	3.33E-05	1.93E-11	2.67E-14	2.63E-09	1.11E-11	6.18E-18	3.35E-12	1.89E-18	1.89E-18
31	144600	3600	0	3.31E-05	4.68E-11	5.54E-14	2.64E-09	2.31E-11	1.28E-17	6.82E-12	3.97E-18	3.97E-18

Table B-20: Release Category 6 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	148200	3600	0	3.30E-05	9.13E-11	9.73E-14	2.67E-09	4.14E-11	2.12E-17	1.18E-11	6.59E-18	6.59E-18
33	151800	3600	0	3.29E-05	1.24E-10	1.23E-13	2.69E-09	5.32E-11	2.56E-17	1.48E-11	7.80E-18	7.80E-18
34	155400	3600	0	3.28E-05	1.49E-10	1.33E-13	2.71E-09	5.82E-11	2.77E-17	1.57E-11	8.29E-18	8.29E-18
35	159000	3600	0	3.27E-05	1.46E-10	1.12E-13	2.70E-09	4.87E-11	2.36E-17	1.27E-11	7.08E-18	7.08E-18
36	162600	3600	0	3.26E-05	1.49E-10	8.85E-14	2.70E-09	3.81E-11	1.90E-17	9.09E-12	5.83E-18	5.83E-18
37	166200	3600	0	3.25E-05	1.48E-10	7.04E-14	2.69E-09	3.02E-11	1.51E-17	6.37E-12	4.78E-18	4.78E-18
38	169800	3600	0	3.24E-05	1.32E-10	5.53E-14	2.67E-09	2.38E-11	1.06E-17	4.65E-12	3.44E-18	3.44E-18
39	173400	3600	0	3.23E-05	1.66E-10	5.75E-14	2.73E-09	7.21E-12	3.05E-14	2.61E-11	1.03E-18	1.03E-18
40	177000	3600	0	3.23E-05	1.58E-10	2.62E-14	2.73E-09	1.56E-12	1.69E-14	3.34E-11	2.93E-19	2.91E-19
41	180600	3600	0	3.22E-05	1.31E-10	2.06E-14	2.70E-09	1.28E-12	1.32E-14	2.95E-11	2.27E-19	2.26E-19
42	184200	3600	0	3.22E-05	1.09E-10	1.66E-14	2.67E-09	1.11E-12	1.05E-14	2.50E-11	1.81E-19	1.79E-19
43	187800	3600	0	3.21E-05	9.03E-11	1.36E-14	2.65E-09	9.83E-13	8.49E-15	2.09E-11	1.46E-19	1.44E-19
44	191400	3600	0	3.21E-05	7.53E-11	1.13E-14	2.64E-09	8.92E-13	6.94E-15	1.74E-11	1.19E-19	1.18E-19
45	195000	3600	0	3.20E-05	6.31E-11	9.43E-15	2.62E-09	8.19E-13	5.71E-15	1.45E-11	9.78E-20	9.69E-20
46	198600	3600	0	3.20E-05	5.30E-11	7.91E-15	2.61E-09	7.51E-13	4.71E-15	1.20E-11	8.04E-20	7.98E-20
47	202200	3600	0	3.20E-05	4.43E-11	6.63E-15	2.60E-09	6.85E-13	3.87E-15	9.91E-12	6.60E-20	6.55E-20
48	205800	3600	0	3.19E-05	3.70E-11	5.54E-15	2.59E-09	6.20E-13	3.17E-15	8.15E-12	5.41E-20	5.37E-20
49	209400	3600	0	3.19E-05	3.30E-11	4.93E-15	2.59E-09	6.07E-13	2.76E-15	7.10E-12	4.71E-20	4.67E-20
50	213000	3600	0	3.19E-05	3.04E-11	4.51E-15	2.59E-09	6.34E-13	2.44E-15	6.27E-12	4.16E-20	4.12E-20
51	216600	3600	0	3.19E-05	2.54E-11	3.75E-15	2.59E-09	5.87E-13	1.97E-15	5.04E-12	3.35E-20	3.32E-20
52	220200	3600	0	3.19E-05	2.08E-11	3.06E-15	2.58E-09	5.19E-13	1.56E-15	4.00E-12	2.66E-20	2.64E-20
53	223800	3600	0	3.19E-05	1.71E-11	2.50E-15	2.58E-09	4.53E-13	1.26E-15	3.20E-12	2.14E-20	2.12E-20
54	227400	3600	0	3.19E-05	1.41E-11	2.06E-15	2.57E-09	3.92E-13	1.02E-15	2.58E-12	1.73E-20	1.72E-20
55	231000	3600	0	3.18E-05	1.17E-11	1.70E-15	2.57E-09	3.36E-13	8.28E-16	2.09E-12	1.41E-20	1.40E-20
56	234600	3600	0	3.18E-05	9.70E-12	1.41E-15	2.56E-09	2.87E-13	6.78E-16	1.71E-12	1.16E-20	1.13E-20
57	238200	3600	0	3.18E-05	8.11E-12	1.17E-15	2.55E-09	2.46E-13	5.62E-16	1.41E-12	9.58E-21	8.92E-21
58	241800	3600	0	3.18E-05	6.81E-12	9.86E-16	2.55E-09	2.12E-13	4.69E-16	1.17E-12	7.96E-21	7.18E-21
59	245400	3600	0	3.17E-05	5.74E-12	8.32E-16	2.55E-09	1.83E-13	3.93E-16	9.77E-13	6.48E-21	5.96E-21
60	249000	3600	0	3.17E-05	4.86E-12	7.05E-16	2.54E-09	1.58E-13	3.32E-16	8.22E-13	5.30E-21	5.03E-21
61	252600	3600	0	3.17E-05	4.13E-12	6.00E-16	2.54E-09	1.38E-13	2.80E-16	6.94E-13	4.43E-21	4.29E-21
62	256200	3600	0	3.17E-05	3.53E-12	5.13E-16	2.54E-09	1.20E-13	2.39E-16	5.91E-13	3.77E-21	3.69E-21

Table B-20: Release Category 6 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
63-96	259800	3600	0	3.17E-05	3.36E-12	4.88E-16	2.54E-09	1.15E-13	2.27E-16	5.61E-13	3.58E-21	3.51E-21
Total	36600	345600	0	3.05E-03	1.48E-06	4.40E-08	1.69E-06	1.25E-06	6.32E-09	3.47E-07	1.57E-13	1.57E-13

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

The seventh RC consists of general transients where an RSV is demanded and subsequently fails to open. In scenarios where the DHRS or CFDS could mitigate core damage, these systems are assumed to be unavailable. The PRA sequences that contribute to RC 7 are shown in Table B-21. RC 7 is an intact containment scenario.

The conservative release sequence for this release category is a general transient where the RSVs fail to open, leading to RPV overpressure failure at the pressurizer heater inspection port. The containment remains intact. 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. Radionuclides released from the core relocate to the CNV, where the majority of aerosols deposit in the reactor coolant present in the CNV and on the CNV walls. In the MELCOR simulations, the CNV is modeled as sealed with no leakage. Therefore, the radionuclide release to the environment is calculated separately based on the fraction of the core inventory airborne in containment and the design basis containment leak rate of 0.2 weight percent per day (see FSAR Table 6.5-1).

The plume release information for RC 7 that is used as input for MACCS simulations is shown in Table B-22. As a result of modeling the CNV as sealed, assumptions relating to the particle size distribution and plume buoyancy are required inputs to MACCS. The particle size distribution for RC 7 is assumed to be the same as the particle size distribution for RC 3 (see Table B-11). RC 3 is a containment bypass accident, and the particle size distribution of a containment bypass accident is expected to cause increased deposition in the environment as compared to the actual particle size distribution for an intact containment sequence. The plume buoyancy is simulated using the heat model with an assumed plume heat content of zero watts for each plume segment. The result of this assumption is that the plume does not rise upon release, which maximizes the ground level radionuclide concentrations. Sensitivity studies evaluating the impact of the aerosol deposition in the CNV predicted by MELCOR, the assumed particle size distribution, and the non-buoyant plume are discussed in Section B.3.

Table B-21: PRA Sequences that Contribute Significantly to Release Category 7

Event Tree	Sequence	Description	FSAR Reference
EHVS--LOOP-----ET	2-17	CTG fails, ATWS, DHRS success, RSVs fail to open, CVCS injection fails	Figure 19.1-9
	16	CTG fails, BDG fails, Reactor trip, DHRS fails, RSVs fail to open	
	22	CTG fails, BDG fails, ATWS, RSVs fail to open	
TGS---TRAN--NPC-ET	11	Reactor trip, DHRS fails, RSVs fail to open, CFDS fails	Figure 19.1-11
	17	ATWS, DHRS success, RSV fail to open, CVCS injection fails	
	22	ATWS, DHRS fails, RSVs fail to open	
TGS---TRAN--NSS-ET	8	Reactor trip, DHRS fails, RSVs fail to open	Figure 19.1-12
	12	ATWS, DHRS fails, RSVs fail to open	

Table B-22: 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	26160	3600	0	2.92E-06	3.48E-07	4.89E-08	3.03E-07	3.16E-07	8.90E-11	3.22E-08	5.10E-15	5.09E-15
2	29760	3600	0	1.75E-05	1.46E-06	7.18E-09	8.53E-07	9.17E-07	4.90E-09	4.04E-07	9.22E-14	9.22E-14
3	33360	3600	0	2.48E-05	9.55E-08	1.09E-10	7.62E-08	4.93E-08	7.24E-10	2.51E-08	1.79E-14	1.79E-14
4	36960	3600	0	2.66E-05	7.74E-08	2.97E-10	1.83E-07	7.65E-08	7.19E-10	1.88E-08	1.71E-14	1.72E-14
5	40560	3600	0	2.85E-05	6.43E-08	8.25E-10	7.88E-08	1.17E-07	5.92E-10	1.70E-08	1.42E-14	1.42E-14
6	44160	3600	0	3.16E-05	1.38E-07	5.60E-10	1.67E-07	1.04E-07	1.08E-09	3.53E-08	3.00E-14	3.00E-14
7	47760	3600	0	3.46E-05	2.62E-08	1.04E-09	4.36E-08	7.27E-08	2.96E-12	5.57E-09	1.36E-15	1.38E-15
8	51360	3600	0	3.39E-05	1.70E-08	5.58E-10	2.20E-08	4.38E-08	7.34E-13	3.56E-09	7.10E-16	7.17E-16
9	54960	3600	0	3.37E-05	6.33E-09	1.73E-10	1.00E-08	1.43E-08	1.95E-13	1.18E-09	2.21E-16	2.22E-16
10	58560	3600	0	3.37E-05	2.09E-09	4.54E-11	4.46E-09	3.94E-09	7.98E-14	3.42E-10	6.12E-17	6.14E-17
11	62160	3600	0	3.35E-05	7.45E-10	1.25E-11	2.75E-09	1.19E-09	4.80E-14	1.02E-10	1.76E-17	1.75E-17
12	65760	3600	0	3.33E-05	2.91E-10	3.59E-12	2.15E-09	3.80E-10	2.13E-14	3.11E-11	5.19E-18	5.11E-18
13	69360	3600	0	3.31E-05	1.86E-10	1.05E-12	1.93E-09	1.26E-10	8.27E-15	9.50E-12	1.54E-18	1.51E-18
14	72960	3600	0	3.29E-05	1.74E-10	3.14E-13	1.85E-09	4.35E-11	3.08E-15	2.98E-12	4.70E-19	4.58E-19
15	76560	3600	0	3.26E-05	1.11E-10	1.01E-13	1.78E-09	1.60E-11	1.19E-15	1.00E-12	1.54E-19	1.50E-19
16	80160	3600	0	3.25E-05	4.63E-11	3.91E-14	1.74E-09	7.14E-12	5.54E-16	4.05E-13	6.05E-20	5.85E-20
17	83760	3840	0	3.45E-05	2.40E-11	1.69E-14	1.83E-09	3.51E-12	2.70E-16	1.81E-13	2.65E-20	2.53E-20
18	87600	3600	0	3.22E-05	1.30E-11	6.36E-15	1.71E-09	1.43E-12	1.08E-16	6.94E-14	9.87E-21	9.00E-21
19	91200	3600	0	3.21E-05	9.28E-12	2.96E-15	1.70E-09	6.97E-13	5.23E-17	3.26E-14	4.39E-21	4.05E-21
20	94800	3600	0	3.20E-05	7.20E-12	1.48E-15	1.69E-09	3.60E-13	2.68E-17	1.64E-14	2.19E-21	2.04E-21
21	98400	3600	0	3.19E-05	5.78E-12	7.86E-16	1.68E-09	1.95E-13	1.44E-17	8.72E-15	1.14E-21	1.05E-21
22	102000	3600	0	3.18E-05	6.96E-12	5.38E-16	1.68E-09	1.40E-13	9.91E-18	5.96E-15	7.49E-22	3.82E-22
23	105600	3600	0	3.19E-05	1.35E-11	4.88E-16	1.69E-09	1.61E-13	9.02E-18	5.39E-15	7.15E-22	1.85E-22
24	109200	3600	0	3.18E-05	1.90E-11	3.30E-16	1.69E-09	1.58E-13	6.11E-18	3.63E-15	4.90E-22	1.14E-23
25	112800	3600	0	3.18E-05	2.35E-11	1.93E-16	1.70E-09	1.44E-13	3.57E-18	2.09E-15	2.70E-22	5.38E-25
26	116400	3600	0	3.17E-05	2.81E-11	1.04E-16	1.70E-09	1.29E-13	1.93E-18	1.12E-15	6.38E-23	9.32E-28
27	120000	3600	0	3.16E-05	4.33E-11	5.73E-17	1.71E-09	1.41E-13	1.06E-18	6.07E-16	1.98E-23	1.52E-28
28	123600	3600	0	3.18E-05	2.15E-10	3.23E-17	1.87E-09	4.12E-13	5.92E-19	3.46E-16	0.00E+00	0.00E+00
29	127200	3600	0	3.13E-05	1.41E-10	1.07E-17	1.79E-09	2.46E-13	1.95E-19	1.16E-16	0.00E+00	0.00E+00
30	130800	3600	0	3.12E-05	1.34E-10	3.67E-18	1.77E-09	2.16E-13	6.59E-20	4.14E-17	0.00E+00	0.00E+00
31	134400	3600	0	3.11E-05	2.09E-10	1.28E-18	1.83E-09	3.83E-13	2.21E-20	2.09E-17	0.00E+00	0.00E+00

Table B-22: 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
32	138000	3600	0	3.09E-05	3.22E-10	4.20E-19	1.95E-09	7.35E-13	6.46E-21	2.59E-17	0.00E+00	0.00E+00
33	141600	3600	0	3.07E-05	3.08E-10	1.27E-19	2.09E-09	1.15E-12	1.04E-21	3.09E-17	0.00E+00	0.00E+00
34	145200	3600	0	3.06E-05	2.01E-10	3.65E-20	2.19E-09	1.55E-12	7.77E-23	3.23E-17	0.00E+00	0.00E+00
35	148800	3600	0	3.05E-05	1.06E-10	9.86E-21	2.29E-09	1.99E-12	0.00E+00	3.43E-17	0.00E+00	0.00E+00
36	152400	3600	0	3.04E-05	7.61E-11	2.10E-21	2.40E-09	2.48E-12	0.00E+00	3.51E-17	0.00E+00	0.00E+00
37	156000	3600	0	3.03E-05	7.41E-11	3.45E-22	2.54E-09	3.10E-12	0.00E+00	3.75E-17	0.00E+00	0.00E+00
38	159600	3600	0	3.03E-05	6.38E-11	0.00E+00	2.46E-09	2.93E-12	0.00E+00	3.26E-17	0.00E+00	0.00E+00
39	163200	3600	0	3.02E-05	5.73E-11	0.00E+00	2.39E-09	2.80E-12	0.00E+00	2.95E-17	0.00E+00	0.00E+00
40	166800	3600	0	3.02E-05	5.97E-11	0.00E+00	2.43E-09	3.08E-12	0.00E+00	3.10E-17	0.00E+00	0.00E+00
41	170400	3600	0	3.01E-05	6.01E-11	0.00E+00	2.44E-09	3.21E-12	0.00E+00	3.15E-17	0.00E+00	0.00E+00
42	174000	3600	0	3.01E-05	6.01E-11	0.00E+00	2.45E-09	3.27E-12	0.00E+00	3.19E-17	0.00E+00	0.00E+00
43	177600	3600	0	3.00E-05	6.24E-11	0.00E+00	2.48E-09	3.44E-12	0.00E+00	3.35E-17	0.00E+00	0.00E+00
44	181200	3600	0	3.00E-05	6.49E-11	5.51E-15	2.43E-09	2.02E-12	3.87E-15	1.08E-12	2.38E-20	2.38E-20
45	184800	3600	0	2.99E-05	5.41E-11	8.25E-15	2.12E-09	7.95E-13	5.79E-15	3.12E-12	3.57E-20	3.56E-20
46	188400	3600	0	2.98E-05	3.94E-11	6.90E-15	1.92E-09	6.01E-13	4.84E-15	3.00E-12	2.98E-20	2.97E-20
47	192000	3600	0	2.98E-05	2.75E-11	5.11E-15	1.81E-09	4.35E-13	3.59E-15	2.33E-12	2.21E-20	2.20E-20
48	195600	3600	0	2.97E-05	1.89E-11	3.63E-15	1.74E-09	3.13E-13	2.55E-15	1.70E-12	1.57E-20	1.56E-20
49	199200	3600	0	2.97E-05	1.30E-11	2.55E-15	1.69E-09	2.26E-13	1.79E-15	1.21E-12	1.10E-20	1.10E-20
50	202800	3600	0	2.96E-05	9.08E-12	1.79E-15	1.67E-09	1.66E-13	1.25E-15	8.53E-13	7.69E-21	7.68E-21
51	206400	3600	0	2.96E-05	6.42E-12	1.26E-15	1.65E-09	1.24E-13	8.86E-16	6.06E-13	5.43E-21	5.43E-21
52	210000	3600	0	2.96E-05	4.61E-12	9.05E-16	1.64E-09	9.52E-14	6.35E-16	4.36E-13	3.88E-21	3.88E-21
53	213600	3600	0	2.96E-05	3.36E-12	6.55E-16	1.63E-09	7.47E-14	4.59E-16	3.16E-13	2.81E-21	2.80E-21
54	217200	3600	0	2.95E-05	2.49E-12	4.80E-16	1.62E-09	6.00E-14	3.37E-16	2.33E-13	2.05E-21	2.05E-21
55	220800	3600	0	2.95E-05	1.87E-12	3.56E-16	1.62E-09	4.94E-14	2.50E-16	1.73E-13	1.52E-21	1.51E-21
56	224400	3600	0	2.95E-05	1.42E-12	2.66E-16	1.62E-09	4.16E-14	1.87E-16	1.31E-13	1.13E-21	8.08E-22
57	228000	3600	0	2.95E-05	1.10E-12	2.01E-16	1.61E-09	3.59E-14	1.41E-16	9.97E-14	8.54E-22	3.84E-22
58	231600	3600	0	2.95E-05	8.57E-13	1.53E-16	1.61E-09	3.15E-14	1.07E-16	7.68E-14	5.75E-22	2.52E-22
59	235200	3600	0	2.95E-05	6.76E-13	1.17E-16	1.61E-09	2.82E-14	8.21E-17	5.97E-14	2.69E-22	1.89E-22
60	238800	3600	0	2.94E-05	5.38E-13	8.99E-17	1.61E-09	2.55E-14	6.31E-17	4.68E-14	1.64E-22	1.48E-22
61	242400	3600	0	2.94E-05	4.32E-13	6.93E-17	1.61E-09	2.35E-14	4.86E-17	3.71E-14	1.21E-22	1.17E-22
62	246000	3600	0	2.94E-05	4.03E-13	5.88E-17	1.60E-09	3.57E-14	4.13E-17	3.28E-14	1.10E-22	1.03E-22
63	249600	3600	0	2.94E-05	3.83E-13	4.68E-17	1.60E-09	5.89E-14	3.28E-17	2.80E-14	9.73E-23	9.02E-23

Table B-22: 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
64	253200	3600	0	2.94E-05	3.22E-13	3.37E-17	1.60E-09	5.89E-14	2.36E-17	2.20E-14	7.79E-23	6.50E-23
65	256800	3600	0	2.94E-05	2.68E-13	2.47E-17	1.60E-09	5.25E-14	1.73E-17	1.76E-14	6.07E-23	4.31E-23
66-96	260400	3600	0	2.94E-05	2.58E-13	2.31E-17	1.60E-09	5.09E-14	1.62E-17	1.67E-14	5.74E-23	3.91E-23
Total	26160	345840	0	2.87E-03	2.24E-06	5.97E-08	1.90E-06	1.72E-06	8.10E-09	5.43E-07	1.79E-13	1.79E-13

B.2.8 Release Category 8: Dropped NuScale Power Module During Transport

The final RC consists of core damage and environmental release due to a dropped NPM during refueling transport. The NPM transport process begins 48 hours after reactor shutdown. In this event, the Reactor Building crane fails to hold the NPM during transport, causing the NPM to rest horizontally on the reactor pool floor. The NPM does not rest perfectly horizontally on the reactor pool floor, and the NPM is not completely flooded; therefore, the fuel is partially uncovered when the NPM is in this orientation. 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 NPM tips over, the NPM may strike other NPMs. The NPM drop induced CDF is estimated as 8.75E-08 per module critical year in the LPSD PRA. Consistent with a NuScale 12 NPM plant, that frequency is multiplied by 12 for a NPM drop CDF of 1.05E-06 per year. The PRA sequences considered in RC 8 are shown in FSAR Figures 19.1-30, 19.1-32, 19.1-33, 19.1-34, 19.1-35, 19.1-36, and 19.1-37. RC 8 is conservatively treated as a bypassed containment scenario.

The conservative release sequence for this RC is an NPM 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 NPM (i.e., the side of the NPM facing the reactor pool floor). The NPM pressurizes, forcing the reactor coolant from the NPM while precluding any reactor pool water from entering the NPM. The accident progresses to core damage, with approximately 60 percent of the core inventory of iodine releasing to the reactor pool. However, the radionuclide aerosols released from the NPM 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 off-site 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 RC 8 is shown in Table B-23. The plume release information for RC 8 that is used as input for MACCS simulations is shown in Table B-24.

Table B-23: Particle Size Distribution for Release Category 8

Particle Size	Chemical Group		
	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-24: 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	191580	3600	0	0.00E+00	8.16E-08	0.00E+00
2	195180	3600	0	0.00E+00	2.24E-07	0.00E+00
3	198780	3600	0	0.00E+00	4.16E-07	0.00E+00
4	202380	3600	0	0.00E+00	6.30E-07	0.00E+00
5	205980	3600	0	0.00E+00	8.00E-07	0.00E+00
6	209580	3600	0	0.00E+00	9.42E-07	0.00E+00
7	213180	3600	0	0.00E+00	1.00E-06	0.00E+00
8	216780	3600	0	0.00E+00	1.09E-06	0.00E+00
9	220380	3600	0	0.00E+00	1.42E-06	0.00E+00
10	223980	3600	0	0.00E+00	1.97E-06	0.00E+00
11	227580	3600	0	0.00E+00	3.30E-06	0.00E+00
12	231180	3600	0	0.00E+00	5.60E-06	0.00E+00
13	234780	3600	0	0.00E+00	9.58E-06	0.00E+00
14	238380	3600	0	0.00E+00	1.36E-05	0.00E+00
15	241980	3600	0	0.00E+00	1.69E-05	0.00E+00
16	245580	3600	0	0.00E+00	2.04E-05	0.00E+00
17	249180	3600	0	0.00E+00	2.54E-05	0.00E+00
18	252780	3600	0	0.00E+00	2.80E-05	0.00E+00
19	256380	3600	0	0.00E+00	3.08E-05	0.00E+00
20	259980	3600	0	0.00E+00	3.56E-05	0.00E+00
21	263580	3600	0	0.00E+00	4.20E-05	0.00E+00
22	267180	3600	0	0.00E+00	4.80E-05	0.00E+00
23	270780	3600	0	0.00E+00	5.24E-05	0.00E+00
24	274380	3600	0	0.00E+00	5.70E-05	0.00E+00
25	277980	3600	0	0.00E+00	6.36E-05	0.00E+00
26	281580	3600	0	0.00E+00	7.14E-05	0.00E+00
27	285180	3600	0	0.00E+00	7.80E-05	0.00E+00
28	288780	3600	0	0.00E+00	8.34E-05	0.00E+00
29	292380	3600	0	0.00E+00	8.68E-05	0.00E+00
30	295980	3600	0	0.00E+00	8.86E-05	0.00E+00
31	299580	3540	0	0.00E+00	9.40E-05	0.00E+00

Table B-24: 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
32	303120	3600	0	0.00E+00	8.58E-05	0.00E+00
33	306720	3600	0	0.00E+00	8.22E-05	0.00E+00
34	310320	3600	0	2.32E-01	4.10E-05	2.56E-01
35	313920	3600	0	2.56E-01	1.63E-05	2.79E-01
36	317520	3600	0	1.22E-01	7.72E-06	1.31E-01
37	321120	3600	0	2.78E-02	1.76E-06	2.97E-02
38	324720	3600	0	2.14E-03	1.51E-07	2.27E-03
39	328320	3600	0	0.00E+00	0.00E+00	0.00E+00
40	331920	3600	0	0.00E+00	0.00E+00	0.00E+00
41	335520	3600	0	0.00E+00	0.00E+00	0.00E+00
42	339120	3600	0	0.00E+00	6.92E-08	0.00E+00
43	342720	3600	0	0.00E+00	9.40E-09	0.00E+00
44	346320	3600	0	0.00E+00	1.16E-08	0.00E+00
45	349920	3600	0	0.00E+00	5.26E-08	0.00E+00
46	353520	3600	0	0.00E+00	9.08E-08	0.00E+00
47	357120	3600	0	0.00E+00	1.21E-07	0.00E+00
48	360720	3600	0	0.00E+00	1.59E-07	0.00E+00
49	364320	3600	0	0.00E+00	2.08E-07	0.00E+00
50	367920	3600	0	0.00E+00	1.39E-07	0.00E+00
51	371520	3600	0	0.00E+00	2.86E-08	0.00E+00
52	375120	3600	0	0.00E+00	2.10E-08	0.00E+00
53	378720	3600	0	0.00E+00	2.10E-08	0.00E+00
54	382320	3600	0	0.00E+00	3.18E-08	0.00E+00
55	385920	3600	0	0.00E+00	3.50E-08	0.00E+00
56	389520	3600	0	0.00E+00	3.60E-08	0.00E+00
57	393120	3600	0	0.00E+00	4.62E-08	0.00E+00
58	396720	3600	0	0.00E+00	9.36E-08	0.00E+00
59	400320	3600	0	0.00E+00	1.43E-07	0.00E+00
60	403920	3600	0	0.00E+00	1.81E-07	0.00E+00
61	407520	3600	0	0.00E+00	2.06E-07	0.00E+00
62	411120	3600	0	0.00E+00	2.34E-07	0.00E+00
63-96	414720	3600	0	0.00E+00	2.46E-07	0.00E+00
Total	191580	345540	0	6.40E-01	1.21E-03	6.99E-01

B.2.9 Summary of Base Case Release Category Results

The CDFs for each sequence of each PRA in a RC are summed to give the total RC frequency. Engineering judgment is used to select the conservative release sequence for each RC. The eight RCs are presented in Table B-25. The release fraction to the environment, start of release, and release duration for each RC are shown in Table B-26.

Table B-25: Release Categories and Associated NuScale Power Module Core Damage Frequency (per module-year[1])

Hazard	Release Category								
	1	2	3	4	5	6	7	8	Total
IE	7.02E-12	1.06E-12	1.79E-11	1.60E-10	1.39E-13	4.06E-11	4.52E-11	-	2.72E-10
LPSD	3.21E-15	-	2.31E-14	3.91E-13	-	2.13E-14	4.37E-14	-	4.82E-13
I-Fires	1.45E-11	-	-	4.85E-10	-	1.33E-10	1.92E-10	-	8.25E-10
I-Flood	-	-	-	-	-	2.96E-11	3.03E-11	-	5.98E-11
E-Flood	-	-	-	9.30E-10	-	4.16E-15	3.06E-13	-	9.30E-10
High Winds	-	-	-	8.68E-10	-	2.92E-12	2.62E-12	-	8.73E-10
Dropped NPM	-	-	-	-	-	-	-	1.05E-06	1.05E-06
RC Total	2.15E-11	1.06E-12	1.79E-11	2.44E-09	1.39E-13	2.06E-10	2.70E-10	1.05E-06	1.05E-06
% of Total	2.04E-03	1.01E-04	1.70E-03	2.32E-01	1.32E-05	1.96E-02	2.57E-02	9.97E+01	1.00E+02

Note:

[1]The dropped NPM frequency is "per plant year" rather than "per module year".

Table B-26: Release Fraction to the Environment, Start of Release, and Release Duration for each Release Category Calculated by MELCOR

Chemical Group	Release Category							
	1	2	3	4	5	6	7	8
Xe	4.80E-03	3.14E-03	4.85E-01	3.27E-03	5.63E-01	3.05E-03	2.87E-03	6.40E-01
Cs	2.14E-06	2.33E-06	3.84E-01	2.08E-06	3.76E-02	1.48E-06	2.24E-06	0.00E+00
Ba	5.80E-08	5.38E-08	9.92E-03	5.81E-08	1.39E-03	4.40E-08	5.97E-08	0.00E+00
I	2.07E-05	2.04E-06	3.21E-01	1.84E-06	3.50E-02	1.69E-06	1.90E-06	2.26E-03
Te	1.88E-06	1.79E-06	3.28E-01	1.60E-06	2.81E-02	1.25E-06	1.72E-06	0.00E+00
Ru	9.93E-09	9.16E-09	2.66E-03	7.79E-09	1.47E-04	6.32E-09	8.10E-09	0.00E+00
Mo	5.19E-07	5.79E-07	9.44E-02	5.05E-07	8.14E-03	3.47E-07	5.43E-07	0.00E+00
Ce	2.33E-13	1.97E-13	4.96E-08	1.81E-13	3.26E-09	1.57E-13	1.79E-13	0.00E+00
La	2.37E-13	1.97E-13	4.96E-08	1.82E-13	3.28E-09	1.57E-13	1.79E-13	0.00E+00
Start of Release (hr)	3.73	8.17	2.40	7.27	12.3	10.2	7.27	53.2
Release Duration (hr)	96.0	96.0	13.9	96.1	58.7	96.0	96.1	96.0

Table B-27 summarizes the results for the Surry site for each RC discussed in Section B.2.1 to Section B.2.8 for a single NPM. The off-site consequences are mean MACCS results over a year of weather trials and the RC frequencies are point estimates. The off-site consequences per event are converted to per year by multiplying the values by the respective RC frequency. Summation of all RCs yields an estimated off-site dose risk of 2.42E-04 person-rem whole body dose per year for the NuScale design (utilized in Section 5.2). Summation of all RCs yields an estimated off-site economic risk (excluding the dollar value of public dose accrued) of 5.37E-02 dollars per year for the NuScale design (utilized in Section 5.3).

Table B-27: Frequency of Occurrence and Off-site Consequences for each Release Category

RC	Release Frequency (per year)	Off-site Dose per Event (person-rem/event)	Off-site Dose per Year (person-rem/year)	Off-site Economic Impact per Event ($)	Off-site Economic Impact ($/year)
1	2.15E-11	2.62E+01	5.64E-10	0.00E+00	0.00E+00
2	1.06E-12	2.09E+01	2.22E-11	0.00E+00	0.00E+00
3	1.79E-11	1.02E+06	1.82E-05	2.72E+09	4.87E-02
4	2.44E-09	1.87E+01	4.57E-08	0.00E+00	0.00E+00
5	1.39E-13	2.15E+05	2.98E-08	8.42E+07	1.17E-05
6	2.06E-10	1.35E+01	2.78E-09	0.00E+00	0.00E+00
7	2.70E-10	2.01E+01	5.42E-09	0.00E+00	0.00E+00
8	1.05E-06	2.13E+02	2.24E-04	4.77E+03	5.01E-03
Total	1.05E-06	-	2.42E-04	-	5.37E-02

B.3 MACCS Sensitivity Analyses

Sensitivity studies that evaluate the impact of MACCS modeling assumptions made in the calculation of off-site consequences are discussed in this section.

B.3.1 Sensitivity Case 1: Standard Evacuation

The base case MACCS results used in the calculation of maximum benefit in Section 5.0 of this report assumed a site boundary emergency planning zone, where the public would only relocate if the EPA protective action guides are exceeded at a given location. In this sensitivity case a typical 10 mile emergency planning zone evacuation is modeled to investigate the effect of evacuation modeling on the calculation of the maximum benefit.

B.3.2 Sensitivity Case 2: Updated Site Data File

The base case MACCS results used in the calculation of the maximum benefit in Section 5.0 of this report uses population distribution and economic information for the Surry site based on 2005 population data and 2005 economic data to estimate the population distribution in 2060 and the economic data to 2017, as described in Section B.1.3 and Section B.1.4. To evaluate the effect of the initial population and economic information used to extrapolate to a 2060 population and 2017 dollars on the off-site consequences, in this sensitivity case 2010 U.S. Census and 2012 U.S. Agricultural Survey information is used as the starting point to estimate the population distribution in 2060 and the economic information in 2017.

B.3.3 Sensitivity Case 3: Release Site

The base case MACCS results used in the calculation of the maximum benefit in Section 5.0 of this report uses the Surry site as the location of the radionuclide release, because there is no standard site for a design certification application. To evaluate the impact of the site of release on the off-site consequences, in this sensitivity case the release is assumed to occur at the Peach Bottom site.

B.3.4 Sensitivity Case 4: Height of Release

The base case MACCS results used in the calculation of the maximum benefit in Section 5.0 of this report assume the radionuclide release to the environment occurs at ground level (zero m) because a release height of zero meters maximizes the ground level radionuclide concentration. To evaluate the impact of the release height on the off-site consequences, in this sensitivity case the release is assumed to occur at the height of the Reactor Building (81 ft or 24.689 m, per FSAR Section 1.2.2.1).

B.3.5 Sensitivity Case 5: Initial Core Inventory

The base case MACCS results used in the calculation of the maximum benefit in Section 5.0 of this report use the best estimate core inventory as input (Table B-5), which assumes 100 percent power operations. To evaluate the impact of the initial core inventory on the off-site consequences, in this sensitivity case a high burnup core inventory is used (Table B-6), which is derived by calculating the fuel assembly with the highest radioisotope inventory, assuming

102 percent power operation, and assuming every assembly in the core has that radioisotope inventory.

B.3.6 Sensitivity Case 6: Start of Release

The base case MACCS results used in the calculation of the maximum benefit in Section 5.0 of this report release radionuclides to the environment at the start of gap release predicted by MELCOR. However, because each RC represents multiple sequences, it is possible that a release to the environment may begin a different time when different mitigating systems are available. To evaluate the impact of the start of release on the off-site consequences, in this sensitivity case the release is assumed to begin immediately at transient initiation.

B.3.7 Sensitivity Case 7: Off-site Decontamination

The base case MACCS results used in the calculation of the maximum benefit in Section 5.0 of this report assume the time of decontamination (TIMDEC) and cost of decontamination (CDNFRM) for non-farmland from Reference 8.1-28, which are updated values based on Reference 8.1-34. However, as noted in Reference 8.1-33, there are not traceable bases for the TIMDEC and CDNFRM values from Reference 8.1-34. Therefore, consistent with the commission order issued in Reference 8.1-33, in this sensitivity case the TIMDEC and CDNFRM values are set to the maximum allowed MACCS input values for both decontamination levels modeled.

B.3.8 Sensitivity Case 8: Other Economic Parameters

The base case MACCS results used in the calculation of the maximum benefit in Section 5.0 of this report assume economic inputs from Reference 8.1-28 (shown in Table B-2). However, it is recognized on page C-14 of Reference 8.1-31 that the objective of the SOARCA analysis was to realistically evaluate the off-site dose consequences, not the economic impacts, of a severe accident. Therefore, in this sensitivity case the values of the economic parameters in Table B-2 that are expressed in terms of dollar amounts are increased by a factor of two to investigate the sensitivity of the maximum benefit calculation to these economic parameters.

B.3.9 Sensitivity Case 9: Containment Deposition

The base case MACCS results for radionuclide releases that do not bypass containment (release categories 1, 2, 4, 6, and 7) used in the calculation of the maximum benefit in Section 5.0 of this report assume that some fraction of the radionuclides deposit within the CNV (as calculated by MELCOR) and are unavailable for release to the environment. This sensitivity case unrealistically assumes that both the airborne and deposited radionuclides present in the containment are available for release to show the extent to which more conservative MELCOR radionuclide deposition rate calculation models could affect the maximum benefit calculation.

B.3.10 Sensitivity Case 10: Aerosol Dry Deposition in the Environment

The base case MACCS results for radionuclide releases that do not bypass containment (RCs 1, 2, 4, 6, and 7) or the reactor pool (RC 8) used in the calculation of the maximum benefit in Section 5.0 of this report assume aerosol particle size distributions for the release, which affects the aerosol dry deposition in the environment. The intact containment cases assume the particle size distribution from RC 3 is representative of intact containment releases. RC 8 assumes that the particle size distribution calculated from the flow path connecting the containment to the reactor pool is unaffected by reactor pool scrubbing. To evaluate the impact of these assumed particle size distributions on the off-site consequences, in this sensitivity case all aerosols present in the release are assumed to deposit at a velocity of 1 cm/s, consistent with the MACCS inputs on page A-5 of Reference 8.1-32. The particle size distribution that corresponds to the base case deposition velocities are calculated directly from MELCOR for RCs 3 and 5, and therefore this sensitivity does not apply to RCs 3 and 5.

B.3.11 Sensitivity Case 11: Plume Buoyancy

The base case MACCS results for radionuclide releases that do not bypass containment (RCs 1, 2, 4, 6, and 7) or the reactor pool (RC 8) used in the calculation of the maximum benefit in Section 5.0 of this report assume the radionuclide release to the environment occurs as a cold plume (0 watts) and therefore does not rise. The release energy of the plume segments from an intact containment or from beneath the reactor pool is uncertain. Therefore, a release energy of zero watts is used in the base case to maximize the ground level radionuclide concentration. To evaluate the impact of the release energy on the off-site consequences, in this sensitivity case the release is assumed to occur with an internal energy content of 1E+05 W in each plume segment, which corresponds to the energetic portion of a containment bypass release and will cause the plume to rise upon release. The plume buoyancy parameters are directly calculated from MELCOR for RCs 3 and 5, which bypass the containment, and therefore this sensitivity does not apply to RCs 3 and 5.

B.3.12 Sensitivity Results

The off-site consequence results for all sensitivity cases are shown in Table B-28 for population dose and Table B-29 for economic consequences.

Table B-28: Population Dose Results

Case	L-ICRP60ED Population Dose (rem) 50 mile							
	RC 1	RC 2	RC 3	RC 4	RC 5	RC 6	RC 7	RC 8
Base Case	2.62E+01	2.09E+01	1.02E+06	1.87E+01	2.15E+05	1.35E+01	2.01E+01	2.13E+02
Case 1	2.62E+01	2.09E+01	1.02E+06	1.87E+01	2.15E+05	1.35E+01	2.01E+01	2.13E+02
Case 2	2.40E+01	2.00E+01	1.01E+06	1.79E+01	2.11E+03	1.29E+01	1.92E+01	1.79E+02
Case 3	3.92E+01	2.42E+01	8.53E+05	2.16E+01	2.01E+05	1.58E+01	2.32E+01	4.42E+02
Case 4	2.79E+01	2.25E+01	1.05E+06	2.01E+01	2.22E+05	1.45E+01	2.16E+01	2.14E+02
Case 5	6.08E+01	5.79E+01	1.47E+06	5.17E+01	4.72E+05	3.71E+01	5.56E+01	2.31E+02
Case 6	2.63E+01	2.10E+01	1.02E+06	1.87E+01	2.15E+05	1.35E+01	2.01E+01	2.17E+02
Case 7	2.62E+01	2.09E+01	1.02E+06	1.87E+01	2.13E+05	1.35E+01	2.01E+01	2.13E+02
Case 8	2.62E+01	2.09E+01	1.02E+06	1.87E+01	2.15E+05	1.35E+01	2.01E+01	2.13E+02
Case 9	1.79E+04	1.74E+04	-	1.74E+04	-	1.43E+04	1.73E+04	-
Case 10	3.96E+01	3.24E+01	-	2.89E+01	-	2.09E+01	3.10E+01	4.74E+02
Case 11	2.65E+01	2.12E+01	-	1.90E+01	-	1.37E+01	2.03E+01	2.13E+02

Table B-29: Economic Consequence Results

Case	Economic Cost Measures ($) 50 mile							
	RC 1	RC 2	RC 3	RC 4	RC 5	RC 6	RC 7	RC 8
Base Case	0.00E+00	0.00E+00	2.72E+09	0.00E+00	8.42E+07	0.00E+00	0.00E+00	4.77E+03
Case 1	3.45E+08	3.45E+08	2.82E+09	3.45E+08	3.72E+08	3.45E+08	3.45E+08	3.45E+08
Case 2	0.00E+00	0.00E+00	2.70E+09	0.00E+00	8.28E+07	0.00E+00	0.00E+00	2.87E+03

Table B-29: Economic Consequence Results (Continued)

Case	Economic Cost Measures ($) 50 mile							
	RC 1	**RC 2**	**RC 3**	**RC 4**	**RC 5**	**RC 6**	**RC 7**	**RC 8**
Case 3	2.68E+02	6.57E+00	8.19E+09	4.17E+00	2.85E+08	5.17E-01	5.40E+00	4.84E+04
Case 4	0.00E+00	0.00E+00	2.90E+09	0.00E+00	8.96E+07	0.00E+00	0.00E+00	2.42E+03
Case 5	2.25E+01	4.92E+01	7.26E+09	3.21E+01	4.83E+08	2.85E+00	4.06E+01	5.33E+03
Case 6	0.00E+00	0.00E+00	2.73E+09	0.00E+00	8.64E+07	0.00E+00	0.00E+00	4.90E+03
Case 7	0.00E+00	0.00E+00	1.01E+10	0.00E+00	3.52E+08	0.00E+00	0.00E+00	4.77E+03
Case 8	0.00E+00	0.00E+00	5.43E+09	0.00E+00	1.66E+08	0.00E+00	0.00E+00	7.15E+03
Case 9	9.86E+04	9.55E+04	-	9.53E+04	-	7.57E+04	9.60E+04	-
Case 10	9.62E+01	2.27E+01	-	1.53E+01	-	5.43E-01	1.81E+01	6.46E+04
Case 11	0.00E+00	0.00E+00	-	0.00E+00	-	0.00E+00	0.00E+00	3.97E+03

The results of Case 10 show that while the aerosol dry deposition modeling does noticeably impact the off-site consequences, the aerosol dry deposition modeling is reasonable for the base case analysis. The off-site consequences, for all RCs where the particle distribution is assumed, remain small in magnitude regardless of the inputs used. RC 8 is most impacted by changes to aerosol dry deposition modeling. This is expected, because this release consists mostly of gases that do not deposit and small particles that deposit more slowly using the Reference 8.1-6 methodology than the 0.01 m/s velocity used in Case 10, and therefore the release is more likely to deposit within 50 miles of the release than the base case.

The value of economic inputs (Case 7 and Case 8) has a noticeable impact on economic consequences when protective actions are performed, as expected. The increase is not consistent across RCs 3, 5, and 8, which are the only RCs with non-zero economic costs. In RCs 3 and 5 the off-site economic consequences are largely influenced by population dependent decontamination and interdiction, and the increase of the decontamination cost and duration in Case 7 significantly impacts those two results and therefore the economic consequences increase. However, RC 8 does not experience any population dependent decontamination or interdiction, and therefore is not sensitive to changes in population dependent decontamination inputs.

Evacuation within the 10 mile EPZ (Case 1) has a large impact on the off-site consequences, as expected. The impact is the largest among release categories that contain small releases to the environment. RCs 1, 2, 4, 6, 7, and 8 all have small releases that do not demand emergency phase protective actions in the base case where evacuation is not modeled.

The use of updated site population and economic information (Case 2) has a small, but noticeable impact on the off-site consequences as expected. Therefore, the base case information is acceptable.

The results show that assumptions related to plume buoyancy (Case 11) and release height (Case 4) have an insignificant impact on both the dose and economic results, with the exception of the economic costs in RC 8. The economic costs for RC 8 show a significant reduction in case 4 and case 11, compared to the base case. Therefore, the assumptions used in the base case are acceptable.

Assumptions related to the magnitude of radionuclides released to the environment (Case 9 and Case 5) have a noticeable impact on the off-site consequences. Case 9, which assumes all radionuclides present in containment (airborne and deposited) are available for release to the environment, produces release fractions of the core inventory three orders of magnitude larger than the base case releases. The base case releases experience significant radionuclide deposition in containment. Accordingly, the off-site dose and economic consequences of Case 9 releases show a significant increase compared to the base case. However, it is important to note that assuming all radionuclides present in containment, which contains a large amount of water due to ECCS actuation, are available for release is unrealistically conservative. Case 5, which uses the high burnup core inventory (Table B-6) in place of the best estimate core inventory (Table B-5), shows a small increase in off-site consequences compared to the base case results.

The choice of site at which releases are evaluated has a noticeable impact on the off-site consequences, as expected. Postulating that radionuclide releases occur at the Peach Bottom site (Case 3) instead of the Surry site in the base case, generally results in an increase in off-site consequences across release categories. This is expected, as the population within 50 miles of the release at Peach Bottom is approximately twice as large as the population at Surry. However, the off-site dose consequences at the Peach Bottom site are reduced from the base case for the large releases in RCs 3 and 5. The Peach Bottom site has smaller long-term dose thresholds compared to Surry, and therefore a larger portion of the population is permanently relocated from the Peach Bottom site, reducing the overall population dose compared to the Surry site.

The timing of the start of release (Case 6) has a small impact on the off-site consequences, as expected. This is because the change in release timing is generally on the order of hours, whereas the exposure time in MACCS is 50 years, and therefore the reduced radioactive decay before release is insignificant compared to the exposure duration.

B.4 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, LPSD, 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-30 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-30: Seismic Core Damage Frequencies for Peach Bottom and Surry Sites

Site	Peach Bottom	Surry
Core Damage Frequency	2.01E-6 per year	3.17E-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.