A LIMITED SCOPE LEVEL 1 PROBABILISTIC RISK ASSESSMENT OF A LOSS OF COOLANT EVENT FOR AN ADVANCED LIGHT WATER SMALL MODULAR REACTOR

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ABSTRACT

- In the nuclear industry, probabilistic risk assessment (PRA) is crucial in providing insights into the strengths and weaknesses of a plant's design and operation.
- A PRA is used to quantitatively estimate risk by determining what can go wrong, how likely it is to go wrong, and possible consequences [6].
- As computer modeling and simulation techniques continue to improve, increasingly comprehensive PRAs are performed.

INTRODUCTION

This research aims to adopt a generic Level 1 PRA model of a possible loss of coolant event for a light water small modular reactor (SMR) and provide safety improvements using qualitative and quantitative results.

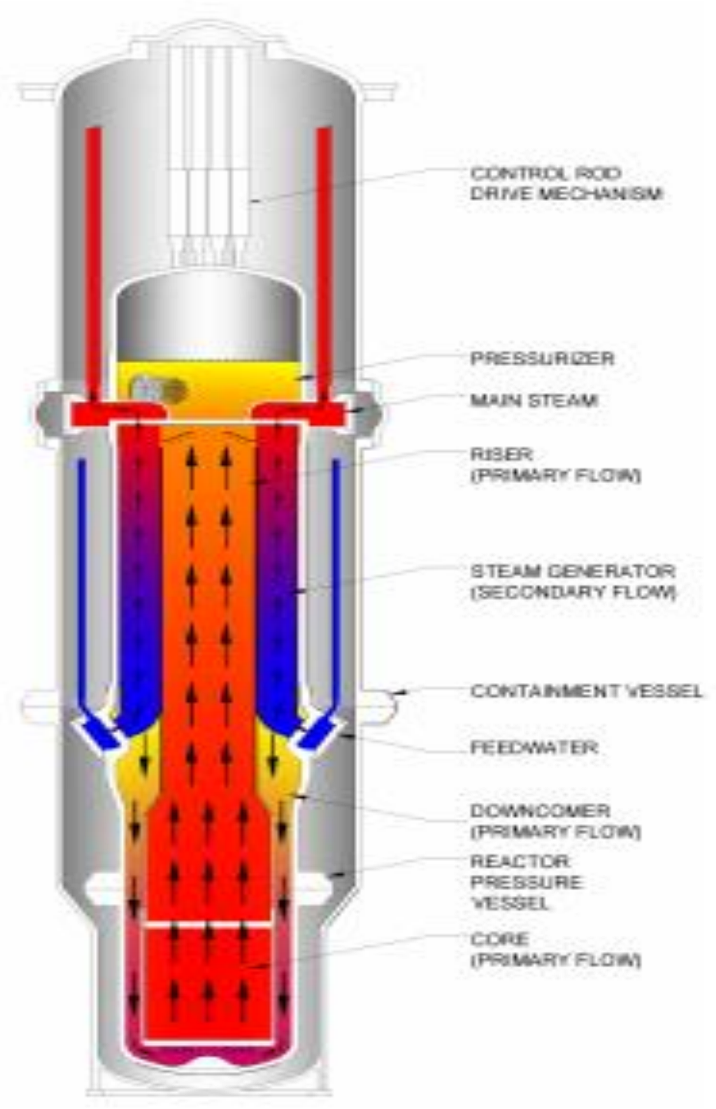

Figure 1: SMR from NuScale [7]

As classified by the Nuclear Regulatory Commission (NRC), light water reactors produce less than 300 megawatts of electric power (MWe). The light-water SMR (Figure 1) is described as portable, versatile, and scalable, making it an ideal prospect for substations and temporary power plants. As of January 2021, there has only been one SMR approved for power generation in the US with five other SMRs pending approval by the NRC [3].

METHODS

Initially, a hypothetical SMR will be modeled and analyzed to understand how limited scope level 1 PRA can be used to improve safety through implementation of mitigation strategies.

1. Perform a level 1 PRA on initial design for quantitative results
2. Introduce new risk mitigation strategies
3. Perform level 1 PRA on mitigated design for quantitative results
4. Compare frequencies of probable failure between designs

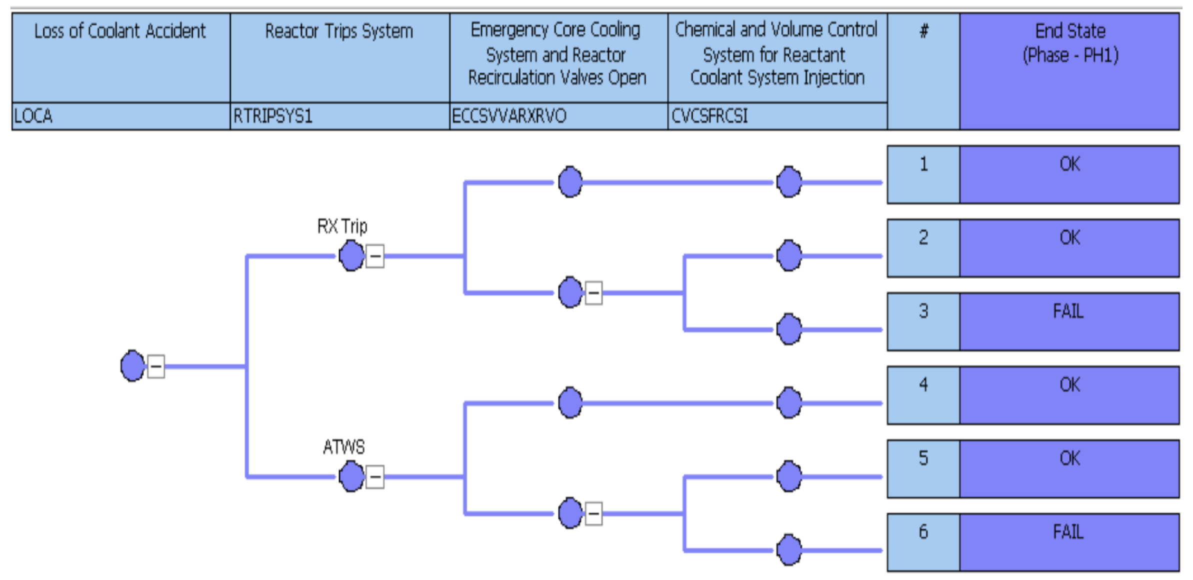

Figure 2: Event Tree for Original Design [4]

An event tree, seen in Figure 2, outlines potential event sequences with various desired and undesired outcomes, called end states, and enable calculating the annual probability frequency of occurrence.

- Figure 2 models a mitigation sequence for loss of coolant accident inside containment vessel (LOCA-IC)
- Event and fault tree models were created using SAPHIRE 8

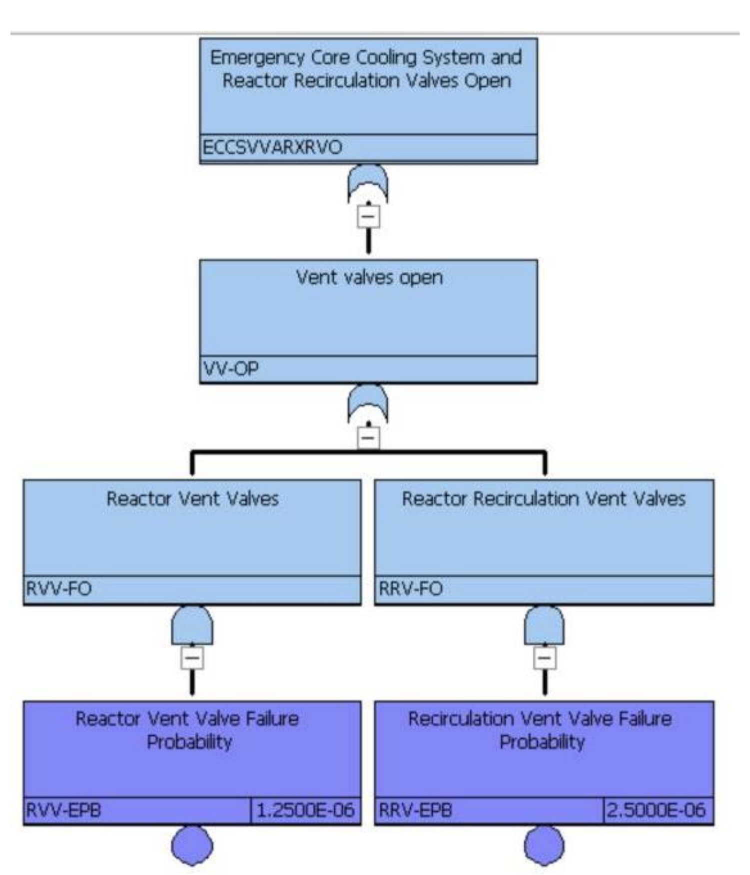

Figure 3: Fault Tree of Emergency Core Cooling System

- Fault trees graphically represent failure modes and their logical sequence and combination
- Frequency of annual occurrence for top level event depends on quantified failure mode probabilities

RESULTS

In order to reduce the probability of failure for the system in question, an operator inspection event will be added after the ECCS. In addition to the operator inspection, a manual override for the vent's valves will be added.

- Operator inspects after ECCS
- Failure of ECCS leads to operator interference and manual opening of valves

#	Prob/Freq	Total %	Cut Sets	
	3.030E-11	100	Displaying 4 of 4 Cut Sets.	
1	2.000E-11	66.01	LOCAIC : 3	
	2.000E-3		LOCA	Loss of Coolant Accident
	4.000E-3		HUMANEPROB1	Human Error Probability
	2.500E-6		RRV-EPB	Recirculation Vent Valve Error Probability
2	1.000E-11	33	LOCAIC : 3	
	2.000E-3		LOCA	Loss of Coolant Accident
	4.000E-3		HUMANEPROB1	Human Error Probability
	1.250E-6		RVV-EPB	Reactor Vent Valve Error Probability
3	2.000E-13	0.66	LOCAIC : 6	
	2.000E-3		LOCA	Loss of Coolant Accident
	1.000E-2		GRXT-1EP	General Reactor Trip Error Probability
	4.000E-3		HUMANEPROB1	Human Error Probability
	2.500E-6		RRV-EPB	Recirculation Vent Valve Error Probability
4	1.000E-13	0.33	LOCAIC : 6	
	2.000E-3		LOCA	Loss of Coolant Accident
	1.000E-2		GRXT-1EP	General Reactor Trip Error Probability
	4.000E-3		HUMANEPROB1	Human Error Probability

Figure 4: Cut Sets for Original Design

#	Prob/Freq	Total %	Cut Sets	
	2.433E-13	100	Displaying 12 of 12 Cut Sets.	
1	8.000E-14	32.88	LOCAIC : 04	
	2.000E-3		LOCA	Loss of Coolant Accident
	4.000E-3		HUMANEPROB1	Human Error Probability
	4.000E-3		OP-ER	Operator error
	2.500E-6		RRV-EPB	Recirculation Vent Valve Error Probability
2	8.000E-14	32.88	LOCAIC : 06	
	2.000E-3		LOCA	Loss of Coolant Accident
	4.000E-3		HUMANEPROB1	Human Error Probability
	4.000E-3		ISP-FL	Inspector fails
	2.500E-6		RRV-EPB	Recirculation Vent Valve Error Probability
3	4.000E-14	16.44	LOCAIC : 04	
	2.000E-3		LOCA	Loss of Coolant Accident
	4.000E-3		HUMANEPROB1	Human Error Probability
	4.000E-3		OP-ER	Operator error
	1.250E-6		RVV-EPB	Reactor Vent Valve Error Probability
4	4.000E-14	16.44	LOCA,HUMANEPROB1,ISP-FL,RVV-EPB	
5	8.000E-16	0.33	LOCA,GRXT-1EP,HUMANEPROB1,ISP-FL,RRV-EPB	

Figure 5: Cut Sets for Mitigated Design

- Figure 4 shows the total annual probability of event occurrence is on the order of $3.03 * 10^{-11}$
- New mitigation factors include operator inspection and manual valve override of the ECCS
- New model shows probability of a LOCA occurring is reduced to $2.4 * 10^{-13}$

Though the frequency of annual occurrence decreased by a magnitude of 2, the original frequency was already extremely low. With NRC goals of a CDF of less than 1×10^{-4}, it is clear that the current mitigation strategies in place would have been sufficient [4].

CONCLUSIONS AND RECOMMENDATIONS

With the CDF already well below the goals set by the NRC, it would be somewhat impractical to implement this mitigation sequence into the NuScale design. This mitigation strategy would not be necessary for this system; however, this research helps show the importance of redundancy of mitigation events associated with potential failure modes inside of the reactor system.

In addition, operators can make errors of commission, meaning that they take erroneous actions thinking that they are the correct ones. Though the value for the frequency of human error per reactor year was the same in both models, one might consider errors of commission to lead to higher frequencies of failure as more humans are introduced as a mitigation technique.

Finally, uncertainty factors were not taken into account for the mitigated PRA, which would have resulted in different values for human error probabilities. In order to properly address this form of epistemic uncertainty, uncertainty analysis would need to be performed and re-integrated into the mitigated design.

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