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The Development of a Demonstration Passive System Reliability Assessment

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Abstract: In this paper, the details of the development of a demonstration problem to assess the reliability of a passive safety system are presented. An advanced small modular reactor (advSMR) design, which is a pool-type sodium fast reactor (SFR) coupled with a passive reactor cavity cooling system (RCCS) is described. The RELAP5-3D models of the advSMR and RCCS that will be used to simulate a long-term station blackout (SBO) accident scenario are presented. Proposed benchmarking techniques for both the reactor and the RCCS are discussed, which includes utilization of experimental results from the Natural convection Shutdown heat removal Test Facility (NSTF) at the Argonne National Laboratory. Details of how mechanistic methods, specifically the Reliability Method for Passive Systems (RMPS) approach, will be utilized to determine passive system reliability are presented. The results of this mechanistic analysis will ultimately be compared to results from dynamic methods in future work. This work is part of an ongoing project at Argonne to demonstrate methodologies for assessing passive system reliability.

Keywords: Passive System Reliability, PRA, advSMR, SFR, RMPS

1. INTRODUCTION

Recently, significant resources have been invested in the development of small modular reactors (SMRs) based on current light water reactor technology and on non-light water designs, known as advanced small modular reactors (advSMRs). These new designs offer both financial and technological advances over the current fleet of nuclear reactors. Relatively low capital costs due to compact designs, shorter construction timeframes, lower power output, and the ability to take advantage of passive safety systems can potentially increase the number of installations of SMRs globally. While these designs represent a step forward for the nuclear industry, they provide a challenge to the current regulatory framework due to the increased dependence on passive safety systems whose reliability may not be easily quantified using conventional reliability methods. This difficulty arises as a result of the nature of passive systems, which can fail functionally without a physical component failure.

In FY13, a review was conducted of the available techniques for modeling the reliability of a passive system using natural circulation [1]. The results of this survey returned three distinct classifications of passive system reliability methodologies. The first technique involved conservative bounding analyses and margin assessment. The second was the mechanistic methods of such projects as the Reliability Method for Passive Systems (RMPS) [2] developed in Europe. Lastly, simulation-based or dynamic techniques present a promising option for the most phenomenologically consistent analyses, but such techniques are still in their infancy.

The next step in this project is to demonstrate several of the surveyed methodologies by analyzing an example passive safety system. The system of choice is the reactor cavity cooling system (RCCS), which is a derivative of the reactor vessel auxiliary cooling system (RVACS) from such plant designs as the PRISM sodium fast reactor (SFR) [3]. An advanced small modular reactor (advSMR), which is a 100 MWe pool-type SFR design, will be analyzed to evaluate the reliability of the RCCS during a station blackout (SBO) scenario. The focus of this analysis is the use of reliability estimate techniques

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to address large uncertainties and the ability to provide a meaningful comparison of competing designs using risk-based information.

This paper details the goals of the analysis and the development of the demonstration problem including descriptions of the models and techniques utilized to analyze passive system reliability. A sister paper "Dynamic Methods for the Assessment of Passive System Reliability" [4] presents the specifics of the dynamic methodology, while the focus of this paper will be on the demonstration problem details and mechanistic modeling techniques.

2. PLANT DESIGN

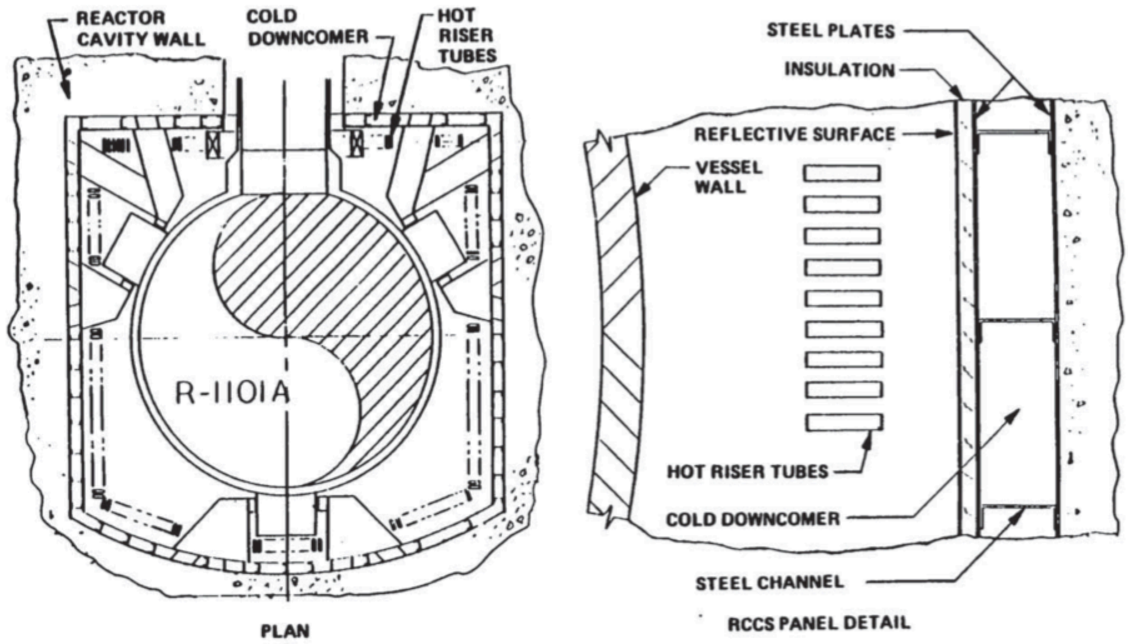
The focus of this analysis is on passive safety systems reliability assessment and not on a specific reactor. For that reason a generic advSMR SFR pool-type design will be utilized in this analysis. The design characteristics of the generic advSMR are listed in Table 1. In addition to these features, the reactor design has intermediate heat exchangers (IHX) which serve as the connection between the hot and cold sodium pools. During normal operation, the IHX is utilized to transfer heat from the hot sodium pool to the secondary side of the plant for electrical power generation. However, in an accident scenario, the IHX can be utilized (if AC power is available) to reject heat via a heat sink (such as the condenser) on the secondary side of the plant. The design also incorporates a guard vessel which surrounds the primary vessel and an RCCS which is based on the design of the RCCS of the General Atomics Modular High Temperature Gas cooled Reactor (GA-MHTGR) [5]. The design will, however, be modified to accommodate the physical size of the advSMR used in this analysis.

Table 1: Design characteristics of the generic advSMR.

Characteristics	
Power rating	250 MWth/100 MWe
Primary coolant	Sodium
Primary system type	Pool
Fuel type	Metallic
Primary coolant flow rate	~ 1270 kg/s
Coolant pump type	Electromagnetic
Number of coolant pumps	4
Primary vessel height	10 m
Core inlet temperature	~ 400°C
Core outlet temperature	~ 550°C

The RCCS uses natural convection to drive air from the outside environment through a cold downcomer and into a lower plenum where the air then flows through hot riser ducts that surround the reactor pressure vessel and line the concrete containment structure. The air in the hot riser ducts is heated through a combination of radiation and convection before being exhausted back to the outside environment. The system is designed to remove decay heat, but because it is completely passive (no baffle or damper operation is required) it also functions during normal reactor operation. A plan view of the RCCS from the GA-MHTGR is shown in Figure 1 [5].

Figure 1: Plan view of the GA-MHTGR cavity and the RCCS [5].

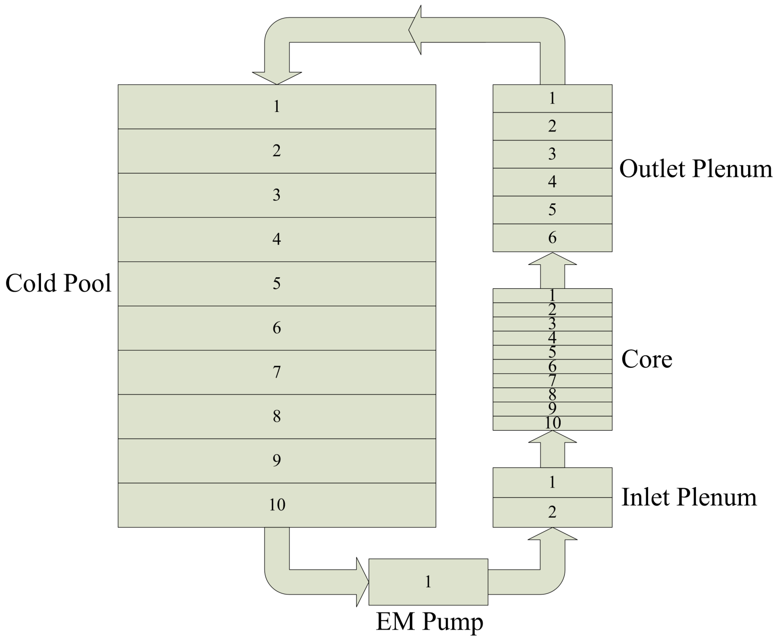


3. MODELING

The advSMR and the RCCS will be modeled together in RELAP5-3D [6]. This allows for the coupled treatment of the heat transfer effects of the natural circulation in the sodium pool and the natural circulation of air in the RCCS. Heat will transfer from cold sodium pool to the walls of the primary vessel. As the primary vessel temperature increases, heat will transfer (via radiation) to a guard vessel and then to the hot riser ducts of the RCCS that surround the primary vessel inside the containment wall. The heat is then convected from the hot riser duct walls to the air passing through each riser. The air that surrounds the vessel and the outer surfaces of the hot riser tubes will not be considered in the heat transfer calculation, as it is assumed that radiation from the guard vessel to the hot riser tubes will be the dominant mechanism of heat transfer. This assumption is consistent with previous analysis of the RCCS [7].

A basic representation of a RELAP5-3D nodalization diagram of an advSMR is shown in Figure 2. While the actual model is much more detailed, this diagram is provided as an overview of the major components that are included in the RELAP5-3D model. In this model, sodium is pumped via an electromagnetic (EM) pump from the cold pool to the inlet plenum before entering the core region. Sodium flows out of the core and into the outlet plenum before passing though the IHX system (not shown in the diagram) and back into the cold pool. This is the assumed flow pathway during normal operation and during accident scenarios where natural circulation flow will occur. Also not shown is an argon filled gap that exists between the outer wall of the primary vessel and a guard vessel. This gap will be modeled in the final RELAP5-3D model.

Figure 2: Simplified nodalization diagram of a generic pool-type advSMR RELAP5-3D model.

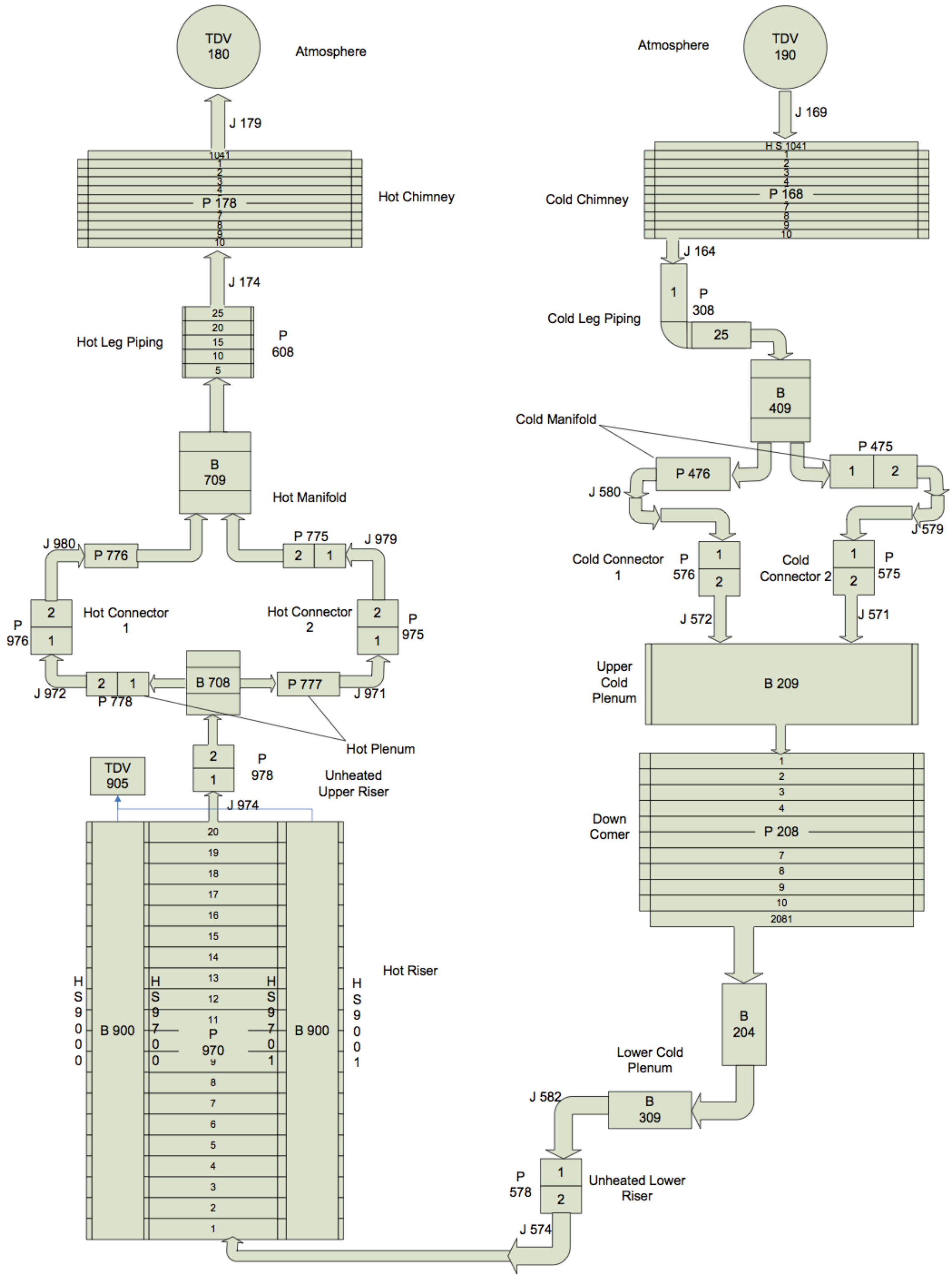


During steady-state simulations, 250 MW of power will be added to the system in the core region. Approximately the same amount of power will be removed from the system via heat structures connected to the cold pool that will be used to simulate the IHX system. The remaining power will be removed from the cold pool via heat structures which represent the guard vessel, the hot riser tubes, and the inside wall of the containment.

During the SBO scenario progression, core power will either be reduced over time to simulate generation of decay heat for protected accidents, or it will remain at operating power until negative reactivity feedback, modeled via integral reactivity feedback coefficients, ultimately reduces core power for unprotected accidents. In the SBO scenario, it is assumed that AC power from the grid will not be regained within 72 hours of the initiating event due to a massive grid failure. However, during the initial 24 hour timeframe, onsite AC power may be restored utilizing FLEX equipment, which is portable backup equipment stored both onsite and offsite as outlined in the FLEX strategy [8].

An existing RELAP5-3D model of the RCCS will be utilized in this analysis. The existing model, of which a nodalization diagram is shown in Figure 3, is from previous GA-MHTGR analyses [7]. This model will be modified appropriately for the physical size difference between the GA-MHTGR vessel and the advSMR vessel utilized in this analysis. The diameters of the primary vessels of the GA-MHTGR and of the advSMR are similar such that it is assumed that the view factors utilized in the analyses of the GA-MHTGR for radiative heat transfer will not require recalculation. However, the height of the GA-MHTGR primary vessel is taller (approximately 22 m) than that of the advSMR (10 m), and therefore the height of the heated section of the hot riser tubes will be modified accordingly. It is also important to note that radiative heat transfer between an axial node of the guard vessel and an axial node of the hot riser duct will only be calculated for nodes at the same elevation. This treatment of the radiative heat transfer is consistent with previous analysis of the RCCS [7] and greatly simplifies the calculation of view factors that are manually input into RELAP5-3D and also reduces the length of time required for each simulation to be performed.

Figure 3: Nodalization diagram of the RCCS RELAP5-3D model [7].

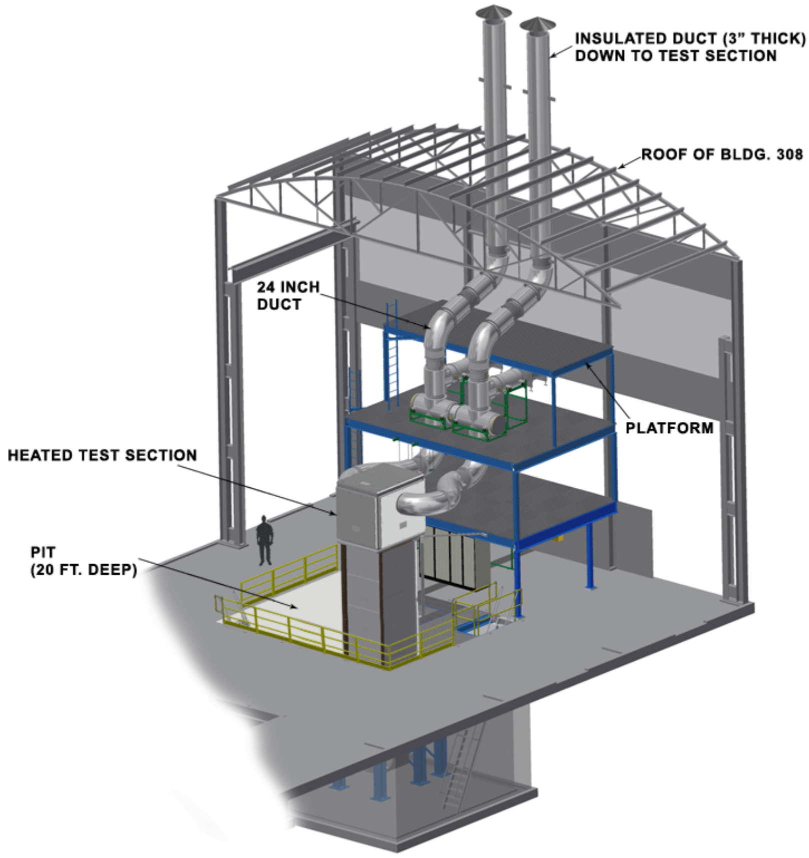


4. BENCHMARKING/VALIDATION

To benchmark the conditions in the RELAP5-3D model of the reactor during steady-state operation and long-term SBO scenarios, SAS4A/SASSYS-1 [9] will be utilized. Ideally, validation would be preferred, but benchmarking is the best available option for this analysis as experimental facilities are not available. SAS4A/SASSYS-1 has been historically utilized for SFR analyses and various aspects of the code have been validated [10,11,12,13]. The ability of SAS4A/SASSYS-1 to model both steady-state and severe accident conditions make it a viable option for benchmarking in this analysis.

To validate the conditions in the RCCS, data from experimental tests conducted at the Natural convection Shutdown heat removal Test Facility (NSTF), located at the Argonne National Laboratory, will be utilized. Originally, the NSTF was constructed to provide full scale simulation of the reactor vessel auxiliary cooling system (RVACS) for the PRISM design [3], but recently the facility has undergone renovation. The current geometry, shown in Figure 4 [14], represents a half-scale model of the RCCS from the GA-MHTGR design. In this test facility, air is drawn into the lower plenum (located in the pit) due to the chimney effect. The air then passes through hot riser ducts located in the heated test section and into the upper plenum. Two chimneys are connected to the upper plenum which allows the heated air to be released to the environment.

Figure 4: Layout of the NSTF [14].



5. RELIABILITY METHODS & UNCERTAINTY ANALYSIS

This section describes two methods that will be utilized to analyze the reliability of the passive RCCS. Section 5.1 describes the first method which is mechanistic and also describes the SBO event tree that will be utilized in the analyzed scenario. Section 5.2 will briefly describe the use of a dynamic analysis which is presented in more detail in a sister paper, titled “Dynamic Methods for the Assessment of Passive System Reliability” [4]. Finally, a description and characterization of the uncertainties associated with both the advSMR and the RCCS are provided in Section 5.3.

5.1. Mechanistic Methods

The mechanistic methods used in this analysis are based on the RMPS methodology [2]. This methodology provides a systematic approach for assessing the reliability of passive safety systems. The RMPS methodology can be divided into three main phases: a preprocessing and model development phase, a simulation and propagation phase, and an analysis/post-processing phase [15]. Table 2 provides details on the main steps of each phase.

Table 2: Three phases of the RMPS methodology.

Phase 1	• Identify a system, its intended mission, and possible failure modes • Select or develop a best-estimate code to perform the analysis • Identify important system variables and characterize their associated uncertainty
Phase 2	• Conduct sensitivity analysis • Determine failure probability of the system using best-estimate code • Assess the impact of the uncertainties of the important system variables
Phase 3	• Incorporate the failure probability into the PRA

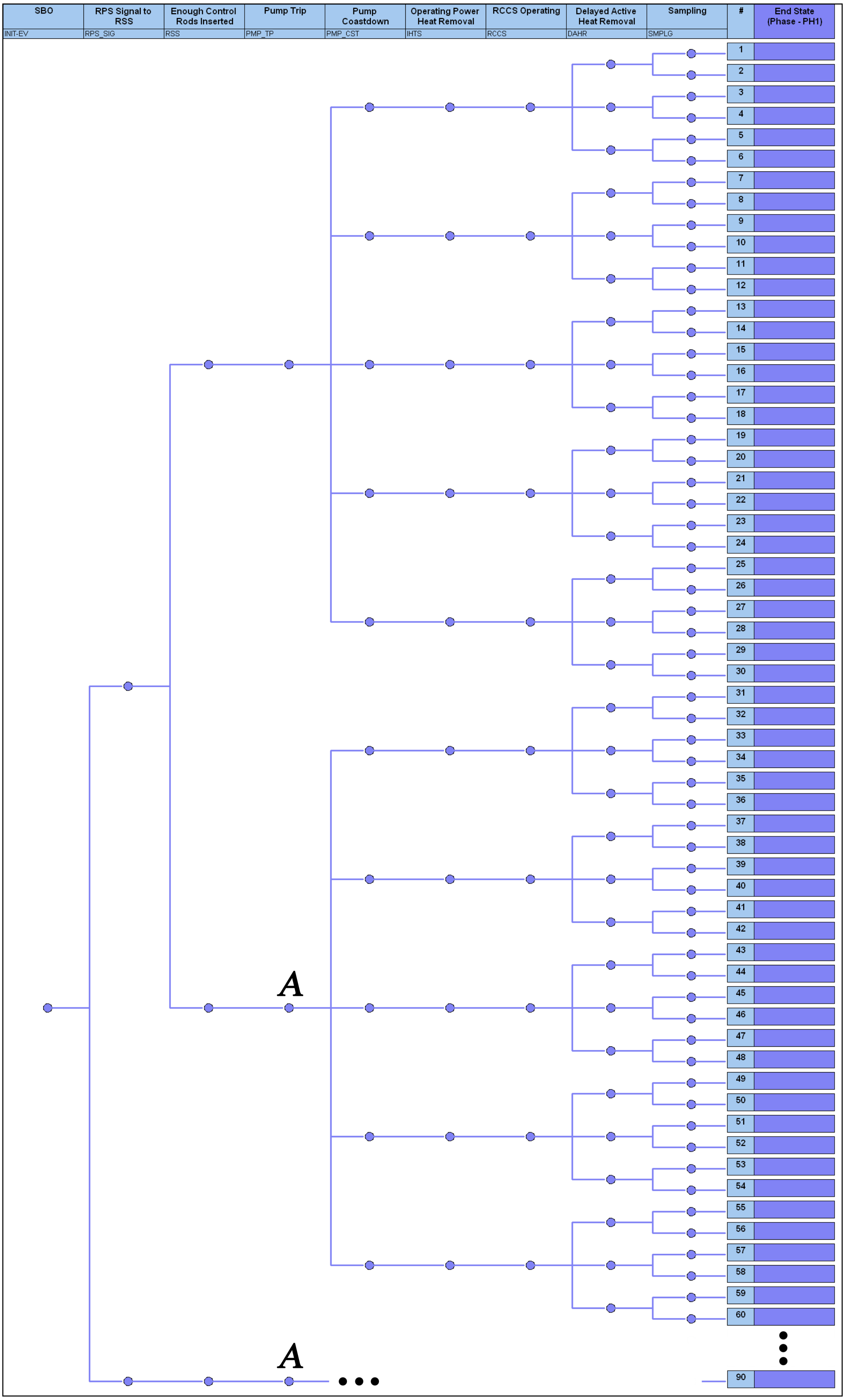
For this analysis, the results from the RMPS methodology will be represented in an event tree. An example SBO event tree, generated in SAPHIRE [16], is shown in Figure 5, which is based on the SBO event tree from the PRISM Preliminary Safety Information Document (PSID) [17]. For this initiating event, it is assumed that both off-site and on-site power (the PRISM design did not include a safety related diesel generator) are lost. For top events whose frequencies in Figure 5 are not unity, (these events are discussed later) the event frequency will be determined using data from in the NRC's "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," NUREG/CR-6928 [18] and the PSID for PRISM [17].

In the SBO event tree, seen in Figure 5, the first top event relates to the reactor protection system (RPS) sending a signal to the Reactor Shutdown System (RSS) to SCRAM the reactor. The signal transmission is either received successfully or not. If the transmission of the signal is successful, the RSS inserts control rods into the core region. This event is labelled in the event tree as "Enough Control Rods Inserted." The PRISM PSID defines "enough" as the successful insertion of at least one out of six control rods. Because a generic reactor/core design is being utilized in this work, a complete reactivity analysis is not being performed and it is assumed that all control rods are either inserted successfully (a protected scenario) or unsuccessfully (an unprotected scenario).

Note that the event tree has two points labelled as *A*. In both branches containing point *A*, a sufficient number of control rods are not inserted to successfully SCRAM the reactor. The only difference in the two branches relates to the signal transmission from the RPS to the RSS. In the upper branch, the signal transmission to SCRAM the reactor was successful, but the RSS failed to insert enough control rods. In the lower branch the signal transmission was unsuccessful; therefore the control rods were not inserted. From the point where the branches are labelled *A*, they are identical and the lower branch has been removed from the figure for simplicity. The probabilities of success and failure for the two branches are identical from point *A* onward, however the end state probabilities will differ due to actions occurring prior to point *A*.

Following the "Enough Control Rods Inserted" event is the "Pump Trip" event. The EM pumps in the primary system require AC power for operation; therefore, when power is lost, the probability of pump trip is unity. The next top event is pump coastdown. EM pumps have almost no inertial force. If power is lost to the pump, the flow rate decreases so rapidly that essentially no pump coastdown occurs. To generate pump coastdown behavior, a separate unit consisting of a flywheel and synchronous machine are utilized. During normal operation, the flywheel and synchronous machine operate in parallel to the EM pump. If power is lost, the inertial energy of the flywheel is converted to electrical energy by the synchronous machine which is used to power the EM pump. This enables the EM pump to generate coastdown behavior similar to that of a mechanical pump. For this top event, five possibilities will be considered, because there are four pumps. Either four out of four pumps coastdown normally, or three out four pumps coastdown normally, and so on until the final possibility which is zero out of four pumps coastdown normally is considered. Successful pump coastdown is modeled in this fashion because the success criterion for sufficient pump coastdown is unknown due the generic reactor design utilized in this analysis.

Figure 5: Example SBO event tree.



The "Operating Power Heat Removal" failure probability is unity because the initiating event is an SBO. The probability of the RCCS operating is unity because the system is passive and an SBO would not necessarily affect the structures associated with RCCS. However, in FY15, the loss and repair of single or multiple RCCS chimneys due to extreme external events, such as seismic or wind, will be considered. The next event is "Delayed Active Heat Removal" in which three possible scenarios will be investigated. The timeframe over which these scenarios occur is between 4 hours to 24 hours after the initiating event, which is roughly based on the anticipated NRC rulemaking as a response to Fukushima [19]. It is assumed that any active heat removal during this time frame will be powered by FLEX equipment as AC power from the power grid is assumed to take longer than 24 hours to restore in the SBO scenario. The first possibility is that cooling via the IHX system is not activated within 24 hours of the SBO. This could be due to unavailability or failure of the FLEX equipment, or the utility opting to allow only the passive system to operate even though FLEX equipment is available. The second and third possibilities are that heat removal via the IHX is activated through the use of FLEX equipment relatively early (9 hours after the initiating event) or relatively late (19 hours after the initiating event) in the simulation. The selection of these two representative time frames will be discussed in more detail in Section 5.3.

The final top event in the tree is labelled "Sampling". In this branch of the analysis, the success and failure probabilities resulting from the propagation of input uncertainties are assessed. For example, the uncertainty of the value for primary vessel emissivity is represented by some distribution (based on existing literature and engineering judgment). Monte Carlo random sampling will be used to create samples from the uncertainty distribution. Each sample value will be simulated in the best-estimate code (RELAP5-3D) and the scenario progression will be prescribed by a certain path through the event tree. This simulation will either result in a safe plant state or a failed plant state (which would depend on the definition of those states). If this process were repeated N times, then the sampling success probability would be the number of simulations that lead to a success (as defined by the scenario success criteria) divided by N and the sampling failure probability would be one minus the sampling success probability. This method of incorporating the simulation results from the sensitivity analysis into the event tree was utilized in the pilot application of Risk Informed Safety Margin Characterization (RISMC) to a total loss of feedwater event [20].

5.2. Dynamic Methods

The most important drawback of a traditional event tree analysis is the inability to explicitly account for time within the event tree. This can potentially lead to phenomenological inconsistencies, particularly when uncertainties are considered that can change the order of events. Another drawback of a traditional event tree analysis is that it relies heavily on the PRA analysts' judgment or on expert elicitation to determine the timing and ordering of events in the event tree. Recent developments in dynamic methods, such as dynamic event trees (DETs) [21,22,23], allow for time to be treated explicitly and provide a way to perform a consistent and mechanistic uncertainty analysis.

In this analysis, the time dependent variable (delayed active heat removal activation time) will be treated utilizing DETs. The methodology for performing this analysis is outlined in a sister paper, titled "Dynamic Methods for the Assessment of Passive System Reliability" [4].

5.3. Uncertainty Analysis

The uncertainty analysis performed as part of this work will include uncertainties that pertain to the plant as well as uncertainties that pertain to the RCCS. Table 3 contains a list of uncertainties that have been identified for inclusion in this research. Engineering judgment along with information from the PRISM PSID [17] was utilized to characterize the distributions that represent the uncertainties associated with each variable. These values represent a starting point for the analysis and may change based on new information or data from experiments conducted at NSTF.

Table 3: Uncertainty characterizations.

Uncertainty	Characterization	Comments
RCCS		
Ambient temperature	$U(-30.0, 45.0)$	Assume conservative bounds, °C.
Primary vessel emissivity	$N(0.77, 0.035)$	Mean and bounding percentiles from [17].
Primary vessel thermal conductivity	$N(1.0, 0.0125)$	Scaling factor, varies up to ±2.5% of mean.
Guard vessel emissivity	$N(0.77, 0.035)$	Mean and bounding percentiles from [17].
Guard vessel thermal conductivity	$N(1.0, 0.0125)$	Scaling factor, varies up to ±2.5% of mean.
Duct surface roughness	$\ln N(3.45, 0.70)$	Assume large range of uncertainty due to weathering, bounding values of 10 μm, 100 μm.
Plant		
Initial power level	$N(1.0, 0.025)$	Scaling factor, varies up to ±5% of mean.
Decay heat curve	$N(1.0, 0.025)$	Scaling factor, varies up to ±5% of mean.
Active heat removal actuation time	$U(4.0, 24.0)$	Time frame of interest for this analysis, hours.
Pump coastdown failure	$P(1/4)$ = 1.000 E-4 $P(2/4)$ = 3.267 E-6 $P(3/4)$ = 1.108 E-6 $P(4/4)$ = 8.420 E-7	Failure per pump considered to be several orders of magnitude larger than value utilized in [17]. Failure of multiple pumps treated using alpha factor model with staggered testing for common cause failures.

The uncertainty associated with the ambient temperature is characterized by a uniform distribution whose bounds are -30°C and 45°C. This wide range of temperatures accounts for various climates as well as seasonal changes. The mean values for primary vessel and guard vessel emissivities are consistent with values specified in the PRISM PSID [17]. The PSID also provided a conservative value for vessel emissivity that was assumed to a 95th percentile value in this analysis. These two values were utilized to characterize normal distributions which represent the uncertainty associated with the emissivities of the vessels. The thermal conductivity of the primary vessel and guard vessel are entered into RELAP5-3D as temperature dependent functions. The uncertainty associated with the thermal conductivities is assumed to be due to normal variations in manufacturing of the vessels and would not vary greatly over the lifetime of the material. These uncertainties are treated with normally distributed scaling factors, assuming that the temperature dependent functions represent mean values of normal distributions and that the scaling factors can deviate by as much as ±2.5% of the mean values. The normal distributions representing these uncertainties are characterized by the 5th percentile value being 97.5% of the mean value and the 95th percentile being 102.5% of the mean value. A scaling factor will be sampled prior to each simulation and utilized to adjust the temperature dependent thermal conductivity of each vessel. The uncertainty associated with surface roughness of the air ducts of the RCCS are also analyzed as part of this work. The surface roughness of these ducts may vary greatly over the lifetime of the facility due to accumulation of particulate material. For this analysis, the uncertainty associated with the surface roughness of the ducts is represented with a lognormal distribution with a 5th percentile vale of 10 μm and a 95th percentile value of 100 μm. These values may be adjusted to reflect experimental results from the NSTF.

The uncertainty in initial power level and decay heat will be treated with scaling factors assuming that the values may vary ±5% from the mean values. The normal distributions representing these uncertainties are characterized by the 5th percentile value being 95% of the mean value and the 95th percentile being 105% of the mean value. The active heat removal actuation time uncertainty is characterized by a uniform distribution from 4 hours to 24 hours, which is the time frame of interest for this analysis. The probabilities for failure of a certain number of pumps to coastdown normally are provided in Table 3. These probabilities were determined using the alpha factor model with staggered testing for common cause failures [24]. As was the case in the PRISM PSID, common cause failures (those not explicitly modeled in the analysis) are assumed to dominate the likelihood of failure. Unlike the PRISM PSID, a much higher value (1.0E-4 per demand) for failure of a single pump to coastdown normally is utilized in this work, as the analysis performed in the PSID was criticized for having unjustifiably (due to lack of data and details) low failure probabilities [25].

6. CONCLUSION

This paper described the preliminary development of a demonstration problem to access the reliability of a passive system utilized to reject decay heat during an SBO scenario for an advSMR. Preliminary RELAP5-3D models of both the advSMR and the RCCS were presented along with a brief description of the codes and facilities that will be used to benchmark and validate the models. Details on the uncertainty analysis were provided along with a description of the RMPS methodology. An example SBO event tree was provided which incorporates a sampling branch in which the percentages of simulations that lead to an undesirable plant state are reported.

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