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Passive Safety System Reliability Analysis: Methodology, Application, and Results

prepared by

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PASSIVE SAFETY SYSTEM RELIABILITY ANALYSIS: METHODOLOGY, APPLICATION, AND RESULTS

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1. INTRODUCTION

Many advanced reactor designs rely on passive systems to provide assurance that certain safety functions, such as reactivity control and reactor heat removal, are maintained during transient scenarios. The performance of these systems is sensitive to boundary and environmental conditions and can lead to functional failure, which is the inability of the system to perform as intended even without any physical failures. Additionally, conditions can cause the passive systems to operate in an intermediate or degraded state. These operational modes of passive systems and their overall reliability can be difficult to capture in traditional probabilistic frameworks where discrete operating modes are assumed or time-dependent boundary conditions are not accounted for. Novel strategies and methods are needed to be able to capture the reliability of these passive systems and their impact on facility safety. The current abstract provides a summary of such a methodology and examples of its use in a case study, the GE Hitachi (GEH) PRISM probabilistic safety assessment (PSA), and the U.S. Department of Energy (USDOE) Versatile Test Reactor (VTR) PSA.

2. METHODOLOGY

While passive systems are designed to achieve high reliability through reliance on natural phenomena rather than engineered, active support systems, properly assessing their reliability during transient scenarios is a necessity for an accurate PSA. This includes the potential for system failures through traditional causes, such as physical component failures, but also functional failures, which are the inability of the passive system to satisfy its mission due to a deviation in expected conditions. Physical component failures can usually be addressed using traditional system reliability analysis approaches and fault tree development. However, functional failures typically necessitate the use of mechanistic system models. The use of mechanistic methods for the assessment of passive safety system reliability and the possibility of functional failure is a requirement of the ASME/ANS Non-Light Water Reactor PSA Standard [1].

The methodology used to assess functional failure in the analyses described in this extended abstract is a variation of the Reliability Method for Passive System (RMPS) [2], with a simplified overview of the approach provided in Figure 1. The RMPS was selected as a starting point because it provides a structured, rigorous approach to assess the reliability of passive systems, however, the method was modified to include integrated mechanistic and simulation-based uncertainty propagation. The methodology provided in Figure 1 was utilized for a case study of the reliability of a passive reactor cavity cooling system (RCCS) during an extreme external event, the results of which are discussed in the next section. The methodology was later expanded to include system PSA success criteria analysis integrated directly within steps in the passive system reliability analysis. Figure 2 provides an overview of the integrated methodology which was then utilized to perform analysis of the reactor vessels auxiliary cooling system (RVACS) of the PRISM and VTR designs.

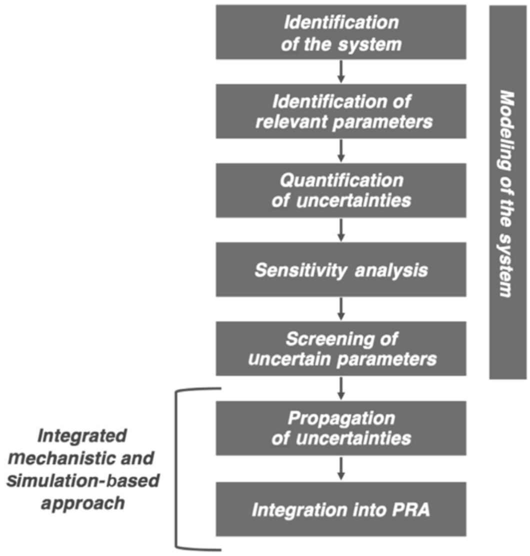

Fig. 1. Overview of a Variation of the RMPS Methodology

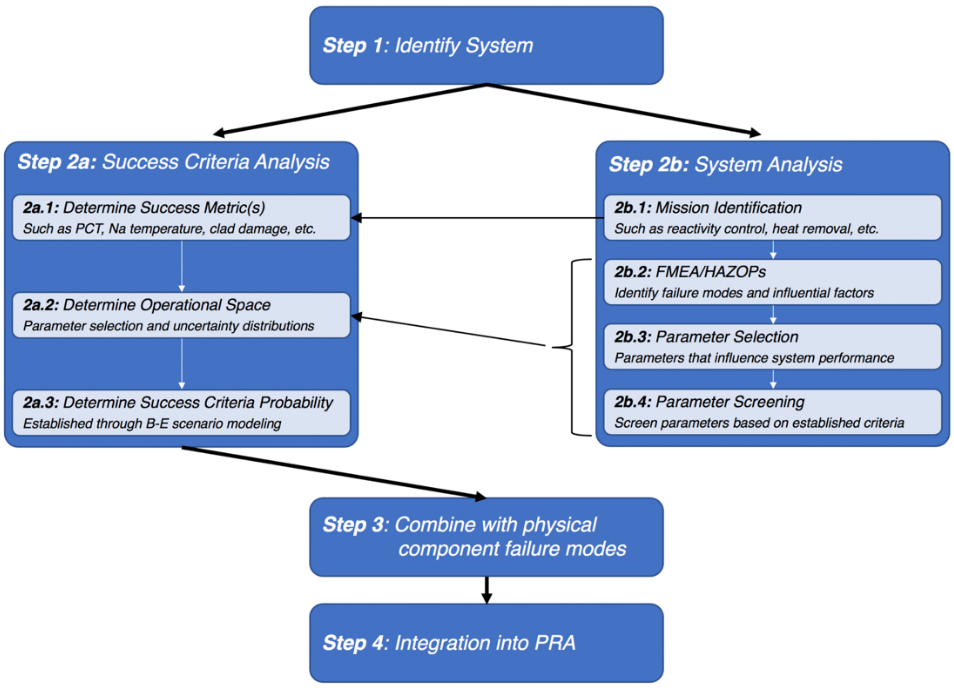

Fig. 2. Passive System Reliability and Success Criteria Methodology

The methodology in Figure 2 begins with identification of the passive system to be analyzed and then bifurcates into determination of success criteria and system analysis. The system analysis process includes identification of the mission of the system, such as providing decay heat removal, and then moves on to failure modes and important parameters identification/screening. In the parallel success criteria analysis, information from the system analysis is utilized to determine success metrics, the operational space over which the parameters will be evaluated, and the determination of the success criteria probability through the use of best-estimate (B-E) simulation tools and uncertainty propagation techniques. Next the results of the functional failure analysis are combined with the results of the traditional physical component failure analysis and then integrated into the PSA.

It is also important to highlight that the success criteria established through the process shown in Figure 2 also ensures consistency with mechanistic source term analysis. Specifically, success metrics related to core damage and radionuclide transport feed directly into the analysis of radionuclide release, including specification of the associated release categories.

3. SUMMARY OF APPLICATION AND RESULTS

3.1. Reactor Cavity Cooling System (RCCS) Case Study

The RCCS design utilized in the case study was a passive decay heat removal system that was originally proposed as part of the design of the General Atomics Modular High Temperature Gas cooled Reactor (GA-MHTGR) [3]. The system uses natural convection to drive air from the environment through cold downcomers and into a lower plenum. The air then flows through hot riser tubes surrounding the reactor guard vessel that line the inner wall of a concrete containment vessel. Heat from the guard vessel is transferred to the air in the hot riser tubes through a combination of radiation and convection before ultimately being rejected to the environment. For this study, the RCCS system was assumed to be the decay heat removal system of a pool-type sodium cooled fast reactor (SFR). Both the reactor and RCCS were modeled using RELAP5-3D [4].

The accident scenario analyzed was large earthquake with the possibility of a subsequent tsunami/flooding that resulted in a station blackout condition. The reactor in the case study was assumed to SCRAM following the earthquake, but due to the station blackout condition, the primary sodium coolant pumps trip off and the operating power heat rejection system is inoperable. Depending on the size of the earthquake, which was varied in the analysis, the integrity of the RCCS air pathway either remained fully intact or a short circuit was established between the cold air downcomer and hot risers allowing for a bypass airflow pathway. The subsequent tsunami could result in flooding of the site that would lead to back up of water in the RCCS due to failure of the RCCS sump system. The flood waters in the RCCS could result in an air flow blockage leading to a loss of RCCS heat rejection. The scenario did include the possibility of restoring functionality of the sump system to remove water from the RCCS as well as recovery of an active heat removal system later in the accident progression. The ability to restore these systems depended on the magnitude of the earthquake in the analysis.

Uncertainties associated with parameters provided in Table 1 were represented by distributions with assumed parameters. These distributions would be sampled before each simulation to determine the input values for each uncertain parameter in the RELAP5-3D input deck. The parameters are those that were identified as being relevant following parameter sensitivity analysis and screening.

Table 1. Uncertain Parameters in RCCS Analysis

Uncertain Parameters	**Distribution**	**Comment**
Ambient temperature	U(-30.0, 45.0)	Conservative bounds (°C)
Primary vessel emissivity	N(0.77, 0.035)	Mean/bounding percentiles from [5]
Primary vessel thermal conductivity	N(1.0, 0.0125)	Scaling factor with limits ±2.5% of mean
Guard vessel emissivity	N(0.77, 0.035)	Mean/bounding percentiles from [5]
Guard vessel thermal conductivity	N(1.0, 0.0125)	Scaling factor with limits ±2.5% of mean
Hot riser duct emissivity	N(0.77, 0.035)	Mean/bounding percentiles from [5]
Hot riser duct thermal conductivity	N(1.0, 0.0125)	Scaling factor with limits ±2.5% of mean
Steel liner emissivity	N(0.77, 0.035)	Mean/bounding percentiles from [5]
Duct surface roughness	lnN(3.45, 0.70)	Uncertainty due to weathering (μm)
Initial power level	N(1.0, 0.025)	Scaling factor with limits ±5% of mean
Decay heat curve	N(1.0, 0.025)	Scaling factor with limits ±5% of mean
RCCS break area	U(0.5, 5.50)	Breach size (m^2)
Sump restore	U(1.0,24.0)	Time (h)
Active heat removal activation time		Time (h)
0.06-0.2g earthquake activation time	U(6.0, 12.0)	
0.2-0.4g earthquake activation time	U(12.0, 24.0)	
0.4-0.8g earthquake activation time	U(18.0, 36.0)	
0.8-2.0g earthquake Activation time	48	

100 RELAP5-3D simulations were performed using sampled values for the uncertain parameters in Table 1. Two metrics were used as indicators that core damage would be likely to occur in each scenario: the onset of sodium boiling and fuel pin cladding failure predicted by a simple eutectic penetration model. In additional to the combination of those two metrics indicating that core damage had occurred, sodium boiling alone was used as

metric for comparison. The central purpose of the analysis was the development, demonstration, and refinement of the passive system reliability assessment approach and the reactor model, assessed scenario, and subsequent results which are reported in [6] should not be interpreted as representative of any current SFR designs.

3.2. PRISM RVACS Analysis

Based on the findings and lessons learned from the analysis described in Section 2.1, the reliability analysis approach was further refined and utilized to perform a reliability assessment of an RVACS design, which is the passive safety-grade decay heat removal system of the GEH PRISM SFR design [7] that uses the natural circulation of air to remove heat from the reactor containment vessel. The analysis utilized the expanded methodology provided in Figure 2 and a SAS4A/SASSYS-1 [8] complete reactor system model of PRISM that included the core, the primary and intermediate coolant circuits, the steam generator, and the RVACS. An FMEA was conducted for the RVACS and the identified failures were sorted into 3 categories: air flow blockage failures, air flow disruption failures, and failures associated insufficient heat transfer from the primary system to the air passing through the RVACS. Uncertain parameters were identified and quantified as part of the analysis. Those parameters were then screened down to those provided in Table 2.

Table 2. Uncertain Parameters in RVACS Analysis

Component	Parameter	Value Range
Stacks/air inlets	Elevation Flow area Pressure drop	16.2 to 3.24m 100% to 1% Up to 25 x nominal
Stacks/air outlets	Elevation Flow area	18.6 to 3.72m 100% to 10%
Hot/cold ducts	Flow area Friction factor	100% to 10% 100% to 300%
Cold air downcomer	Flow area Friction factor	100% to 10% 100% to 300%
Hot air riser	Flow area Friction factor	100% to 10% 100% to 300%
Containment vessel	Emissivity	0.9 – 0.15
Reactor vessel	Friction factor Emissivity	100% to 300% 0.9 – 0.01
Ambient conditions	Temperature	28C to 48C

Success criteria analysis was performed to identify simplified success metrics that were related to peak clad temperature thresholds, hot pool temperature, and sodium boiling. Those metrics were later refined to be more relevant to mechanistic source term evaluations by converting them to fuel damage categories (FDCs) which are provided in Table 3. For the FDCs, the core was assumed to have four batches of fuel that differed in fuel burnup levels and therefore had different internal fuel pin pressures. It was also assumed that spent fuel assemblies were stored in the primary vessel that could fail if sodium pool temperatures were high enough and at very elevated pool temperatures primary vessel structural integrity analysis would be warranted.

Table 3. Fuel Damage Categories (FDCs) for RVACS Analysis

FDCs
No damage
Spent fuel in-vessel
Spent fuel in-vessel + fuel batch 4
Spent fuel in-vessel + fuel batch 3 and 4
Spent fuel in-vessel + fuel batch 2, 3, and 4
Primary vessel integrity analysis warranted

With the uncertain parameters and FDCs defined, a 729-simulation full factorial experiment was conducted for each related transient scenario. Based on the values utilized for the specific simulation, the probability of the simulation was weighted according to the uncertainty distribution associated with the factors outlined in Table 2. Example complementary cumulative distribution function (CCDF) results are provided in Figure 3 for a protected loss-of-flow and loss-of-heat sink (PLOF) scenario. Error ranges of ±25C were utilized as a preliminary assessment of model uncertainty based on engineering judgement. The shaded regions represent the temperature thresholds for the various FDCs. Specific temperature values are not provided here due to proprietary information restrictions. The probabilities of failure for each of the FDCs were then utilized in the PRISM PSA event trees.

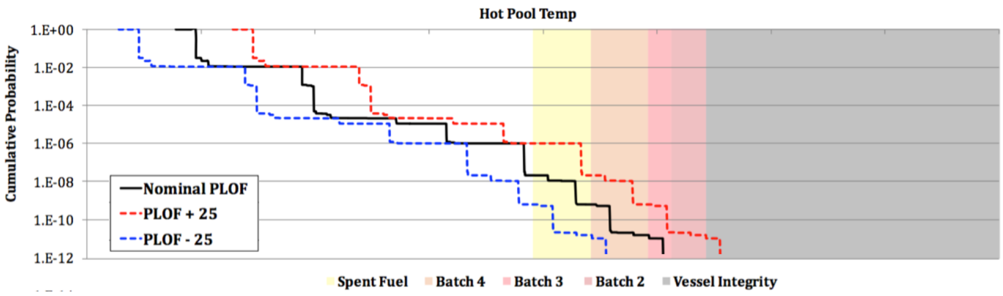

Fig. 3. RVACS PLOF CCDF[1]

3.3. VTR PSA Analysis and Regulatory Approval of the Reliability Methodology

The methodology provided in Figure 2 was utilized to perform a similar analysis as that described in Section 2.2 to access the reliability of passive system heat removal capabilities for the VTR, which is a sodium-cooled fast-neutron spectrum test reactor [9]. Like the PRISM design, the VTR proposed to use an RVACS as the passive safety-grade decay heat removal system. Before the analysis was performed, the methodology was documented in the mandated VTR PSA Plan and subsequently approved by the USDOE which was the authorizing (regulatory) body for VTR. Results from the reliability analysis were utilized in conceptual VTR PSA activities and documented in the VTR Conceptual Safety Design Report (CSDR), which was also approved by the VTR authorizing body (USDOE) [10]. While the results of the reliability analysis for the VTR are not provided here, approval by a regulatory body of the reliability assessment methodology and results utilizing that methodology are important to note for future reactor regulatory interactions.

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