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A MECHANISTIC RELIABILITY ASSESSMENT OF RVACS AND METAL FUEL INHERENT REACTIVITY FEEDBACK

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GE Hitachi Nuclear Energy (GEH) and Argonne National Laboratory (Argonne) participated in a two year collaboration to modernize and update the probabilistic risk assessment (PRA) for the PRISM sodium fast reactor. At a high level, the primary outcome of the project was the development of a next-generation PRA that is intended to enable risk-informed prioritization of safety- and reliability-focused research and development. A central Argonne task during this project was a reliability assessment of passive safety systems, which included the Reactor Vessel Auxiliary Cooling System (RVACS) and the inherent reactivity feedbacks of the metal fuel core. Both systems were examined utilizing a methodology derived from the Reliability Method for Passive Safety Functions (RMPS), with an emphasis on developing success criteria based on mechanistic system modeling while also maintaining consistency with the Fuel Damage Categories (FDCs) of the mechanistic source term assessment. This paper provides an overview of the reliability analyses of both systems, including highlights of the FMEAs, the construction of best-estimate models, uncertain parameter screening and propagation, and the quantification of system failure probability. In particular, special focus is given to the methodologies to perform the analysis of uncertainty propagation and the determination of the likelihood of violating FDC limits. Additionally, important lessons learned are also reviewed, such as optimal sampling methodologies for the discovery of low likelihood failure events and strategies for the combined treatment of aleatory and epistemic uncertainties.

I. INTRODUCTION

Beginning in 2015, a project was initiated to update and modernize the probabilistic risk assessment (PRA) of the GE Hitachi Nuclear Energy (GEH) PRISM sodium fast reactor (SFR)[1]. This project was a collaboration between GEH and Argonne National Laboratory (Argonne), and was funded in part by the U.S. Department of Energy. The goal of the project was the development of a next-generation PRA that will enable risk-informed prioritization of safety- and reliability-focused research and development, while also identifying gaps that may be resolved through additional research. Additionally, this effort was executed in accordance with guidance provided by the recently issued ASME/ANS Non-LWR PRA standard[2], which has been approved for trial use.

The current paper focuses on one of the main Argonne project tasks, which was the mechanistic assessment of passive safety system reliability. In a review of the PRISM Preliminary Safety Information Document (PSID) submitted in late 1980s, the U.S. Nuclear Regulatory Commission (NRC) was critical of the lack of mechanistic modeling of accident scenarios in the PRA, particularly in regard to the performance of passive systems[3]. Additionally, the ASME/ANS Non-LWR PRA standard now requires the mechanistic assessment of passive system performance as part of the success criteria (SC) determination process[2].

Two passive safety systems were examined during the project. The first system, the Reactor Vessel Auxiliary Cooling System (RVACS) utilizes natural circulation and the ambient environment to provide decay heat removal during scenarios where the normal heat removal pathway has been lost. In unprotected (without SCRAM) transients that result in increased core temperatures, inherent reactivity feedback (IRF) of the metal fuel core provides negative reactivity, which lowers reactor power without operator intervention. This paper begins with a review of the methodology utilized to assess the two passive safety systems, followed by a detailed review of each analysis. Additional information on the GEH/Argonne collaboration, included details on other Argonne tasks, can be found in[4-6].

I.A. Methodology

The methodology utilized for the assessment of passive safety system reliability and the determination of SC is detailed elsewhere[7], but an overview is provided here. The process follows the flow chart shown in Fig. 1, where the reliability assessment and SC determination process are conducted in parallel as there are many points of communication and feedback between the two analyses. The passive system reliability assessment is a modified version of the Reliability Method for Passive Safety Functions (RMPS)[8], which was developed through a collaboration of European organizations. The following two sections provide step-by-step details of the application of the methodology to the RVACS and IRFs.

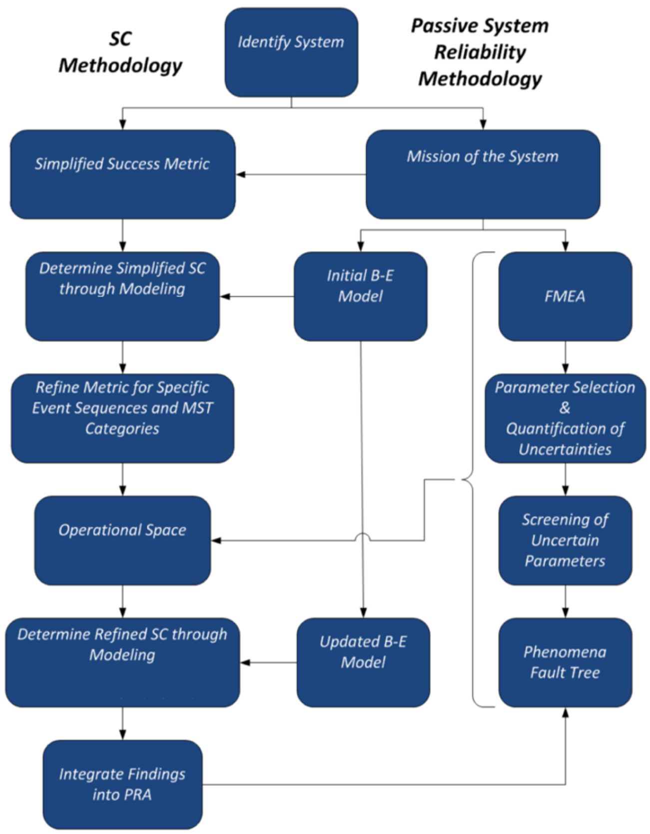

Fig. 1. Passive System Reliability and Success Criteria Methodology

II. RVACS ANALYSIS

The RVACS is a safety-grade heat removal system driven by natural circulation. A detailed description of the RVACS can be found in ref[1]. The RVACS analysis process began with the identification of the system and its boundaries. RVACS includes a total of eight air inlets that feed into four inlet stacks and one common inlet plenum. This air then flows down the reactor silo outside of the collector cylinder. After a 180° turn, the air then rises through a channel between the containment vessel and collector cylinder where it removes heat from the reactor vessel before entering a common outlet plenum. From there, the hot air splits into four hot air stacks and exits the system through four air outlets. The remainder of this section is divided into two subsections, one reviewing the passive system reliability analysis and one detailing the SC determination, as shown in the flow chart in Fig. 1.

II.A. RVACS – Passive System Reliability

II.A.1. RVACS – System Mission

The passive system reliability analysis began with the identification of the system mission. RVACS has several safety-related functions, including the central mission, which is to provide removal of all reactor decay and sensible heat following reactor shutdown when the normal heat rejection paths through the steam generator or through the Auxiliary Cooling System (ACS) are unavailable. The secondary mission is to maintain reactor structural temperatures below safe limits following reactor shutdown in the event that all other heat rejection paths are unavailable. In addition, RVACS should operate at all times, with no operator action required.

II.A.2. RVACS – Best-Estimate Model

A best-estimate model of RVACS was developed using models contained within the SFR systems analysis and severe accident computer code SAS4A/SASSYS-1[9]. The SAS4A/SASSYS-1 PRISM model contains the complete reactor system, including the core, the primary and intermediate circuits, the steam generator, and the RVACS. As described in past studies[10], the analysis of passive system reliability through the use of an integral model is essential for properly capturing dependencies and feedback effects. The SAS4A/SASSYS-1 model also includes the IRFs discussed in Section III.

II.A.3. RVACS – FMEA

A Failure Modes and Effects Analysis (FMEA) was performed as a joint effort between GEH and Argonne. This analysis considered 66 possible failure (or degradation) modes. In general, these failure modes can be groups into three fundamental categories, as shown in Table I. The first category relates to flow blockages. These airflow restrictions can occur at the stack inlets/outlets, within the stacks themselves, or in the flow channels within the reactor silo. The second class includes flow disruptions, which usually result in a change in the pressure drop across the system. Lastly, RVACS requires sufficient heat transfer from the primary sodium system to the ambient air. This requires heat conduction through the reactor vessel, radiation to the containment vessel, conduction through the containment vessel, and then radiation and convection to the ambient air. Any disruption along this pathway can result in inadequate heat transfer to the environment.

II.A.4. RVACS – Parameter Selection and Quantification of Uncertainties

Leveraging the results of the RVACS FMEA, an initial selection of parameters, whose uncertain characteristics may influence RVACS performance, was performed. The parameters are shown in Table II. The choice of parameters also reflects the specifics of the SAS4A/SASSYS-1 RVACS model. It is important to note that the ranges shown in the table do not represent the likely values the parameter may take in reality, but establish the range of possible values. As will be shown in Section II.B.2., the preliminary RVACS SC assessment reviews the complete range of values for the uncertain parameters to determine failure thresholds. Subsequent steps of the reliability assessment then assign a probability of exceeding that threshold.

TABLE I. RVACS - Fundamental Failure Modes

Failure Mode	Description	Examples
Flow Blockage	The air flow area is reduced to the point where the flow rate is no longer sufficient to remove the necessary amount of heat from the reactor system.	Air inlet blockages due to snow; water blockage at bottom of reactor silo; stack blockage due to maintenance plug
Flow Disruption	A change in the system pressure drop that causes a flow disruption due to insufficient buoyancy motive force.	High wind outside reactor building; breach between hot and cold plenum; stack collapse (without blockage); surface friction increase
Insufficient Heat Transfer	Inadequate heat transfer from the primary sodium system to the ambient air.	Containment vessel surface degradation (reduced emissivity); reactor vessel degradation (reduced conductivity); high ambient air temperature

TABLE II. RVACS – Uncertain Parameters

Component	Parameter	Value Range
Stacks/Air Inlets	Elevation[1] Flow Area Pressure Drop	16.2 to 3.24m 100% to 1% Up to 25 × nom.
Stacks/Air Outlets	Elevation[1] 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	28°C to 48°C

[1]Range of elevation heights represents possible breaches between the hot/cold ducts at various heights in the stacks, which would appear to the system as a reduction in stack height.

There are several benefits to conducting the RVACS reliability analysis in this fashion. First, by sampling the complete range of parameter values, then assigning a probability post-analysis, the probability distribution of the parameters can later be adjusted without the need for repeating the simulations. Second, it allows the same simulations to be used for multiple scenario analyses. For example, there may be scenarios where the probability distributions for parameters differ. Since the parameter probabilities are assigned after the simulations, additional simulations are not necessary. Lastly, if the failure threshold for a parameter falls far outside its probability distribution, it can be screened from further analyses (as will be shown in Section II.A.5.).

It should also be noted that the parameter uncertainties include both epistemic and aleatory uncertainties, with several variables having both epistemic and aleatory properties. For example, the environmental conditions, such as the ambient temperature, could likely be considered an aleatory uncertainty, as they are essentially random, although influenced by the location and season. Conversely, the flow area of the RVACS could be considered an epistemic uncertainty, as it could be measured or monitored during operation to reduce the uncertainty related to the state-of-knowledge. A variable such as the emissivity of the guard vessel could be considered to have both epistemic and aleatory characteristics, as the property of the material could be measured for additional knowledge, but the aleatory characteristics of the environment will also impact the emissivity at any given point in time.

II.A.5. RVACS – Screening of Uncertain Parameters

The screening of influential parameters utilizes the findings of the RVACS simplified SC analysis, which is described in Section II.B.2. The simplified SC analysis investigated both separate and combined effects, and established failure thresholds. The location of the failure thresholds in parameter space allowed for an initial screening of uncertain parameters. If the failure threshold was located far outside the uncertainty distribution of the parameter, then the parameter could be removed from further analysis. Similarly, if the parameter does not appreciably affect the SC analysis when evaluating the combined effects (which is the most limiting set of conditions), then the parameter may be removed from further analyses by assigning a conservative value to it.

For example, the parameter ranges in Table II were compared to the unprotected loss of heat sink (ULOHS) simplified SC results (described in Section II.B.2.). The preliminary simplified SC analysis shows that emissivity values must be less than 20% of nominal before heat transfer becomes insufficient for RVACS to fulfill its mission (with all other values at nominal conditions). Conservative emissivity estimates for the reactor vessel and containment vessel based on past data are above 0.6, which is well above the failure threshold. While it might be possible to justify the complete screening of the reactor vessel emissivity based on these findings, the approach utilized here continued to include the emissivity parameters, but attempts to demonstrate the extremely low probability of relatively small emissivities.

Based on the findings of the simplified SC analysis (described in Section II.B.2.), the uncertain parameters were screened and reduced to those listed in Table III. As the table shows, several variables, such as flow area, represent parameters that could impact multiple components within the system.

TABLE III. RVACS – Screened Parameters

Uncertain Parameters
Emissivity
Flow Area
Stack Height
Friction Factor
Inlet Pressure Drop Coefficient
Ambient Air Temperature

II.A.6. RVACS – Phenomena Fault Tree

The phenomenological fault tree for RVACS, shown in Fig. 2, was developed based on the six simplified parameter groups discussed in the preceding section. Utilizing the results of the refined SC analysis that will be described in Section II.B.5., it is possible to quantify the fault tree for different event sequences. However, the analyst has the choice whether or not to utilize the phenomenological fault tree, as the results of the refined SC analysis could be directly inserted into the PRA event trees without first processing them using a fault tree representation.

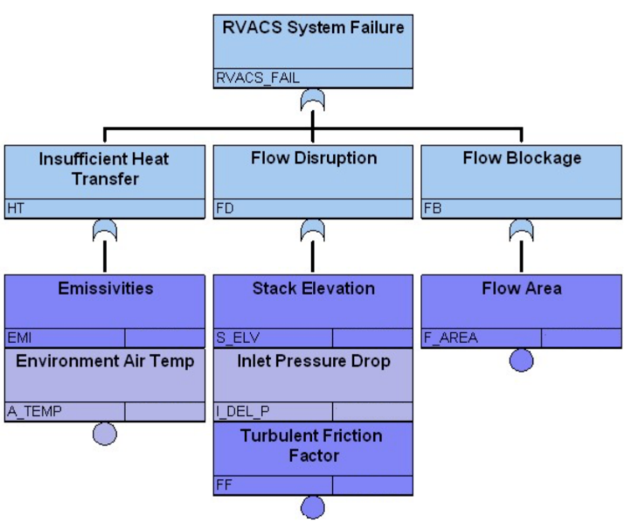

Fig. 2. RVACS – Phenomenological Fault Tree[a]

[a] While "OR" gates are shown in the fault tree, RVACS failure may be caused by degradation in multiple parameters simultaneously.

II.B. RVACS – Success Criteria Determination

II.B.1. RVACS – Simplified Success Metric

Based on the system mission (defined in Section II.A.1.) simplified success metrics were defined to establish the preliminary simplified SC. The metrics, shown in Table IV, relate to both the potential of core damage, which would imply failure of the RVACS primary mission, and damage to the reactor vessel/components, which would signify failure of the RVACS secondary mission.

TABLE IV. RVACS – Simplified SC Metrics

Failure Metric Description
PCT[1] > Minimum temperature for eutectic degradation of cladding to occur
Hot Pool > ASME Service Level D for equipment
PCT > Temperature for acceleration of eutectic penetration of cladding
PCT > Onset of rapid eutectic formation
Sodium Boiling (Possibility of fuel melting)

[1]PCT – Peak Cladding Temperature

II.B.2. RVACS – Determine Simplified SC

A series of transients were simulated, using the best-estimate SAS4A/SASSYS-1 model, to determine the failure thresholds of the parameters outlined in Table IV. These scenarios included loss-of-flow and loss-of-heat sink events, which were identified as important sequences during initial PRA development efforts. The detailed results are not presented here due to proprietary restrictions. However, the separate effect analysis, which examined the impact of the parameters individually, demonstrated that mission failure was very unlikely when only a single parameter was in a degraded state (i.e., extremely degraded parameter values were necessary to result in mission failure). Combined effects analysis was also examined through the use of a 729-simulation full factorial experiment. The full factorial design utilized three values for each of the six parameters of interest: a nominal value, a degraded value, and a highly degraded value. The full factorial design allowed for an analysis of the failure space that was created when multiple parameters were at degraded values simultaneously.

II.B.3. RVACS – Refine Success Metrics

As the project evolved, the success metrics were refined in conjunction with the development of the mechanistic source term (MST) analysis. Specifically, fuel damage categories (FDCs) were established, which are the connection between system SC analysis and the event sequence consequence analysis. For the transient sequences that included RVACS, six FDCs were established, shown in Table V, which were based on the findings of the simplified SC analysis. Most of the FDCs coincide with the predicted failure of different batches of fuel in the core. The core has four fuel batches plus the spent fuel that is located within the in-vessel storage. Since the fuel batches are at different burnup levels, they are at different internal pin pressures and will fail at different reactor condition levels. The most severe FDC is the "vessel integrity analysis," which is the result of prolonged elevated primary system temperatures. This FDC does not indicate that the reactor vessel has failed, but only that additional structural analyses are necessary.

TABLE V. RVACS – FDCs

Fuel Damage Categories (FDCs)
No Damage
Spent Fuel Damage (in-vessel storage)
SF[1] + Batch 4 Fuel Damage
SF + Batch 4,3 Fuel Damage
SF + Batch 4,3,2 Fuel Damage
Vessel Integrity Analysis

[1]SF – Spent fuel stored within the vessel

For FDC, a hot pool temperature threshold was determined. Since the transients involving RVACS include long-term loss-of-normal heat removal, SAS4A/SASSYS-1 simulations predict a near uniform temperature of the primary sodium and little difference in temperature between the primary sodium and fuel within the core. This is due to the high thermal conductivity of the metal fuel and metal coolant. Therefore, the hot pool temperature can be used as a surrogate for the fuel/cladding temperature. The hot pool temperatures for each FDC threshold were the 10,000-second exposure failure temperature for each batch of fuel. Based on the burnup level and internal pin pressure for each batch of fuel, the rate of eutectic penetration of the cladding that would cause the fuel pin to fail after a 10,000-seond exposure to that temperature was calculated based on past experiments and internal studies conducted by GEH. The 10,000-second (2.78 hour) exposure failure temperature was chosen as the FDC hot pool temperature threshold since the transients that included RVACS are long-term and occur over many hours. This simplified approach was utilized since the metal fuel failure models within SAS4A/SASSYS-1 are currently being improved[11].

II.B.4. RVACS – Operational Space

The operational space is the range of uncertainty values that must be examined during the refined SC analysis (Section II.B.5.). As mentioned in Section II.A.4., the method utilized for the RVACS analysis was to explore the entire range of parameter uncertainty values, regardless of probability. Once the analysis was complete, probabilities would be assigned to determine the likelihood of failure. Only the parameter uncertainties that were not screened (those shown in Table III) were included in the operational space.

II.B.5. RVACS – Determine Refined SC

With the refined SC metrics and the operational space determined, the refined SC could be established. Like the simplified SC, a 729-simulation full factorial experiment was utilized to explore the operational space, which separated each of the six uncertain parameters into three levels. The transients re-analyzed included loss-of-flow and loss-of-heat sink sequences, but sequences with additional sources of heat or additional heat sinks were also examined, as they were now part of the refined PRA event trees. These additional sequences included scenarios where the primary pumps failed to trip and add heat to the primary circuit until critical degradation, and scenarios where other reactor systems, such as the ACS, were used to remove decay heat from the primary system.

The determination of the RVACS SC for an event sequence utilized the following process. First, the 729-simulation full factorial would be conducted for the sequence of interest. The uncertainty parameters detailed in Table II.A.4. were varied for each of the 729 simulations. The result of each simulation would fall within one of the six FDCs in Table V. Each of the 729 simulations was then assigned a probability of occurrence based on the uncertain parameter values utilized. The probability of each uncertain parameter value was based on a probability distribution based on literature reviews, data analysis, and expert judgement.

For example, one of the uncertain parameters is the ambient air temperature. Temperatures at the selected reactor location can vary greatly between the seasons. Temperatures near 30°C have been experienced during the summer, with record lows around -5°C in the winter. Warmer temperatures degrade RVACS performance as the temperature difference between the guard vessel and the environment decreases. As the RVACS inlets are in close proximity to the RVACS outlets on the roof of the reactor building, it is assumed that the RVACS inlets experience a temperature approximately 15°C warmer than the ambient air[b]. Therefore, a probability distribution for ambient air temperatures was created with the available data and the 15°C increase assumption. This probability distribution results in the probabilities shown in Table VI for the three

[b] The air exiting the RVACS is at ~90°C during normal operation, with higher temperatures (>125°C) during transients with loss of normal heat removal.

ambient air values used in the 729-simulation full factorial. This process was repeated for each of the six uncertain parameters.

TABLE VI. RVACS – Ambient Air Probabilities

Ambient Air Value	Probability
28°C	0.99
38°C	9.9E-3
48°C	1.0E-4

Once all the uncertain parameter probability values were established, complimentary cumulative distribution (CCDF) functions could be created for each of the transient sequences analyzed. An example CCDF is shown in Fig. 3 for a protected (with SCRAM) loss-of-flow and loss-of-heat sink (PLOF). The shaded regions represent the FDC temperature thresholds. As shown in the figure, by using the results of the 729-simulation full factorial experiment, the likelihood of violating a FDC threshold could be quantitatively established. These values are then used directly within the PRA event trees to assign the branch probabilities or they can be used to quantify the phenomenological fault tree, as described in Section II.A.6. For the example shown, error bars were placed at ±25°C to account for model uncertainty, which was not directly addressed as part of the current project. Even when accounting for this additional uncertainty, the probability of exceeding a fuel damage FDC threshold was very low for the RVACS transients.

III. IRF ANALYSIS

The analysis of the IRFs followed a procedure similar to that of the RVACS analysis, but with several small modifications. The IRFs are intrinsic properties of the fuel and core that result in reactivity changes with variations in system temperature. The IRFs are a key feature of metal fuel, pool-type SFR designs, such as PRISM. Table VII lists the IRFs examined in the current analysis. The system identification of the IRFs included not only the intrinsic reactivity properties of the reactor system, but also the Gas Expansion Modules (GEMs), which are an engineered system developed to lower reactivity in loss-of-primary-flow transients.

III.A. IRF – Passive System Reliability

III.A.1. IRF – System Mission

The IRFs provide a mechanism to lower reactor power during transients when the reactor shutdown system fails. For the PRISM design, during transients that include a loss-

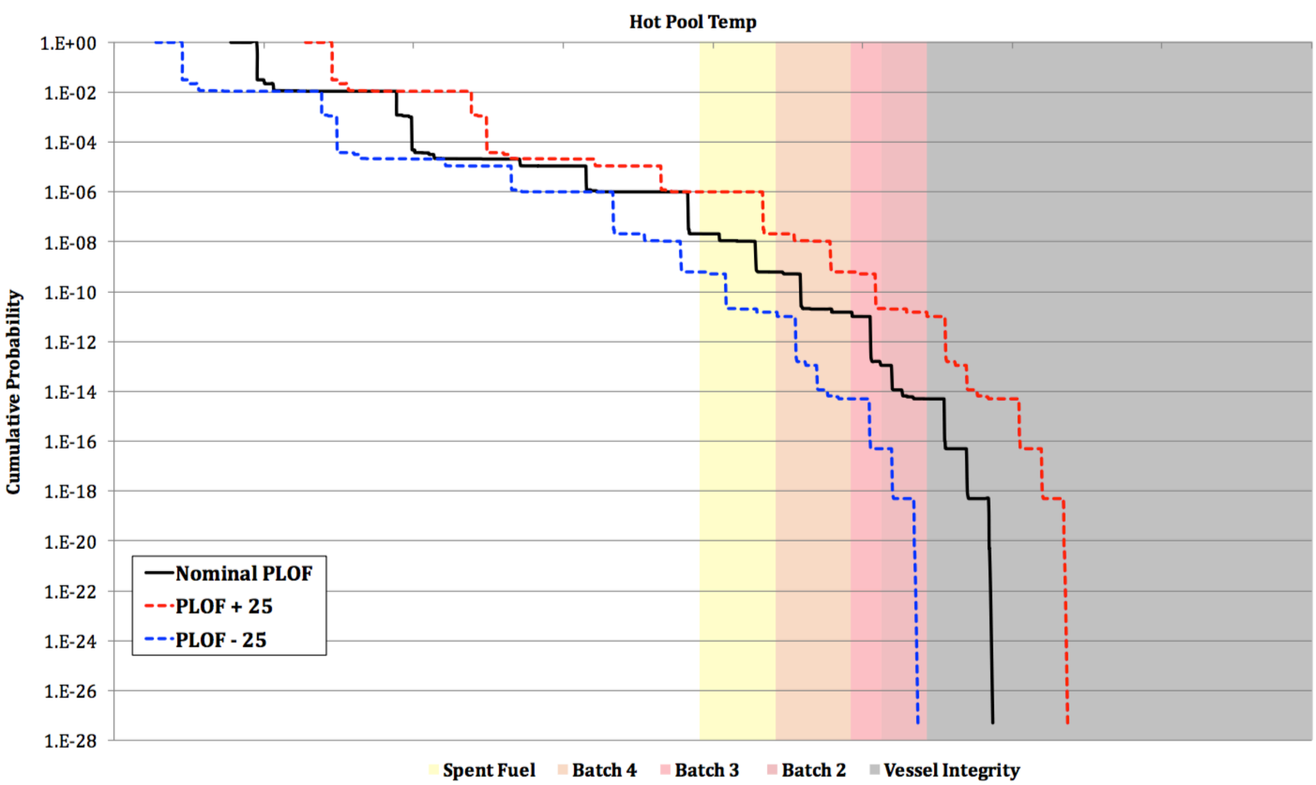

Fig. 3. RVACS – PLOF CCDF[c]

[c] Temperature values removed due to proprietary restrictions.

of-primary-flow, the IRFs work in tandem with the GEMs to lower reactor power. For unprotected transients involving the reactivity feedbacks, the system mission is to bring the reactor to a new steady-state condition while still critical, but at a lower power. This power level should be safely sustainable until ultimate shutdown can be achieved.

TABLE VII. IRF – Mechanisms Considered

Reactivity Feedback Mechanism	**Description**
Doppler	Doppler broadening with temperature increase affects neutron cross sections.
Sodium Density	Change in the density of the primary sodium with temperature influences the amount of moderation the sodium provides.
Fuel Axial Expansion/Contraction	The expansion/contraction of the metallic fuel rod with temperature can influence power through changes in fuel density and neutron leakage.
Net Radial Expansion/Contraction	Core load pad and restraint ring expansion/contraction with temperature influences power through changes in fuel density and neutron leakage.
Control Rod Expansion/Contraction	Control rod expansion/contraction with temperature can influence reactivity within the core through neutron absorption. Expansion/contraction of the vessel also influences the distance between the core and control rod.
Gas Expansion Modules (GEMs)	An engineered system that introduces negative reactivity with a loss-of-primary-flow.

For the analysis conducted here, it is assumed that ultimate shutdown must be achieved within the first 12 hours following the initiation of the transient. Therefore, the IRFs must maintain the reactor at an acceptable power level and temperature for 12 hours.

III.A.1. IRF – Best-Estimate Model

SAS4A/SASSYS-1 was utilized as the best-estimate model of the system, as it contains internal models for all of the IRFs under consideration. In addition, as mentioned in Section II.A.2., the use of SAS4A/SASSYS-1 allowed the integral analysis of the complete reactor system, rather than reduced, separate effect tests.

III.A.2. IRF – FMEA

Unlike the RVACS analysis, an FMEA was not performed for the IRFs. This step was excluded from the project as there are a multitude of possible factors that can influence the performance of the IRFs. Many of the factors require separate, detailed calculations to properly quantify the associated uncertainties[d]. Such effort was outside the scope of the current project. Instead, past analyses were used as the basis for the quantification of the associated IRF parameters and their uncertainties, as described in the following section.

III.A.3. IRF – Parameter Selection and Quantification of Uncertainties

Table VIII provides an overview of the uncertainties associated with each of the IRFs, other than GEMs, included in the current analysis. These uncertainty values were determined during a past Argonne study as part of the Integral Fast Reactor (IFR) project[13]. As can be seen, several of the IRFs have large assigned uncertainty ranges. For example, radial expansion has a 1σ uncertainty of 50% at power-to-flow ratios less than 0.8.

TABLE VIII. IRF – Parameter Uncertainties[13]

Reactivity Feedback Mechanism	**1σ Uncertainty**
Doppler	20%
Sodium Density	20
Fuel Axial Expansion/Contraction	30
Neutronic	20
Thermo-mechanical	20
Net Radial Expansion[a] (P/F > 0.8)	20
Neutronic	15
Thermal Hydraulic	10
Structural	10
(P/F < 0.8)	50
Neutronic	15
Thermal Hydraulic	15
Structural	50
Control Rod Expansion	20
Neutronic	10
Thermal Hydraulic	<20

[a]Includes bowing

Due to the constraints of the project, it was not feasible to use all of the separate effect uncertainties for the analysis detailed here. Instead, the integral uncertainties were used for fuel axial expansion and net radial expansion. Table IX summarizes the parameters used for the SAS4A/SASSYS-1 simulation, along with their uncertainty ranges. For net radial expansion the larger uncertainty range recommended for power-to-flow ratios below 0.8 was chosen. This choice

[d] For example, the analysis of core bowing (a phenomenon that affects radial feedback) includes structural calculations with computer codes such as ANSYS and NUBOW-3D[12].

was based on the reasoning provided in ref[13]. If the core temperatures and temperature gradients are close to those at full power and flow, then the uncertainty range is reduced, as the load pad contact state is unchanged. However, a deviation from these conditions results in large uncertainty in bowing behavior. As it was not known *a priori* what conditions would be encountered for each IRF simulation, the larger uncertainty range was conservatively chosen.

TABLE IX. IRF – Selected Uncertainty Ranges

Reactivity Feedback Mechanisms	**1σ Uncertainty[a]**
Doppler	20%
Sodium Density	20%
Fuel Axial Expansion/Contraction	30%
Net Radial Expansion	50%
Control Rod Expansion (Neutronic)	10%
Control Rod Expansion (Thermal Hydraulic)	20%

[a]Normal distribution assumed for uncertainties

For GEMs, a conservative uncertainty range was assigned in place of a detailed analysis. The design basis for GEMs is a negative reactivity insertion of -$0.60 due to a loss-of-primary flow. A one-sided normal distribution with a standard deviation of $0.116 and a mean of -$0.60 was used, which resulted in 99% of the cumulative distribution falling between -$0.30 and -$0.60. The remaining 1% of cumulative probability was assumed to be uniformly distributed between -$0.30 and $0.0. This distribution is shown in Fig. 4.

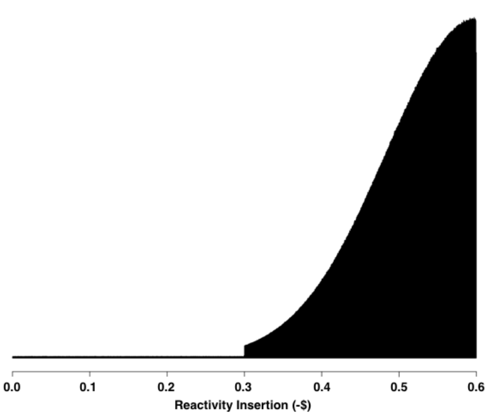

Fig. 4. IRF – GEMs Probability Distribution

The shape of the GEMs probability distribution was chosen based on several factors. First, it was assumed that the most likely degraded GEMs scenarios would be the slight underperformance of GEMs, compared to nominal, due to calibration errors or unforeseen uncertainties related to the response of GEMs with flow changes. This possibility accounts for the normal distribution section of the probability density function. However, the other degraded GEMs scenario is the possibility of common-cause failure of all GEMs. This scenario is less likely, but could result in substantial degradation in the negative reactivity response of GEMSs. Therefore, the uniform probability distribution between -$0.30 and $0.0, with 1% overall cumulative probability, was taken to conservatively represent this scenario.

III.A.4. IRF – Screening of Uncertain Parameters

As there were relatively few uncertain IRF parameters and since the SAS4A/SASSYS-1 simulations of IRF sequences were comparatively fast running, screening of parameters was not necessary for the IRFs. Instead, the uncertainty range of each parameter was investigated using multiple sampling techniques, as will be described in Section III.B.5. The relative importance of the individual IRFs and their correlation to the metrics of interest was also investigated.

III.A.5. IRF – Phenomena Fault Tree

As described with the RVACS analysis, it is possible to create a phenomenological fault tree for the IRFs based on the six mechanisms outlined in Table VII. However, for the IRF analysis, the results of the refined SC analysis (described in Section III.B.5.) were utilized directly within the PRA event trees without the additional step of fault tree creation.

III.B. IRF – Success Criteria Determination

III.B.1. IRF – Simplified Success Metric

Based on the system mission (defined in Section III.A.1.) simplified success metrics were defined to establish the preliminary simplified SC. The metrics, shown in Table X, include simple metrics for fuel failure, fuel melting, and component integrity, which relate to the mission of the IRF described in Section III.A.1. The "pin failure" metric assesses the likelihood of cladding failure that is a result of exceedance of a stress threshold due to a combination of hoop stress and cladding degradation from eutectic penetration.

TABLE X. IRF – Simplified Success Metrics

Failure Metric
SAS4A/SASSYS-1 predicts significant cladding degradation (<0.4mm remaining)
SAS4A/SASSYS-1 predicts pin (cladding) failure
Sodium Boiling (Possibility of fuel melting)
Hot Pool > ASME Service Level D for equipment

III.B.2. IRF – Determine Simplified SC

Three transients were simulated with SAS4A/SASSYS-1 to determine simplified SC. These included an unprotected loss-of-flow (ULOF), an unprotected loss-of-heat-sink (ULOHS), and an unprotected transient overpower (UTOP). For this initial analysis, 2500 Monte Carlo simulations were performed for each of the three transients.

The detailed results are not presented here due to proprietary restrictions, but the preliminary results demonstrated very low fuel damage probabilities for each of the transients assessed. In general, the failure thresholds were only surpassed when the parameter uncertainty values were representative of a very degraded state. However, as will be described in more detail in Section III.B.5, it was difficult to properly capture the low likelihood events using Monte Carlo sampling. Therefore, alternative sampling approaches were utilized for the detailed SC assessment.

III.B.3. IRF – Refine Success Metrics

As with the RVACS analysis, the success metrics were refined in conjunction with the development of the MST analysis. Specifically, the FDCs shown in Table XI were developed based on the findings of the simplified SC analysis. Due to the shorter problem time of the IRF transients relative to the RVACS transients, failure of the fuel batch in the spent fuel rack was not considered, as hot pool temperatures generally do not reach elevated levels for an extended period of time.

TABLE XI. IRF – FDCs

FDCs	Description
Batch 4	Clad failure of batch 4 (no fuel melt)
Batch 4,3	Clad failure of batch 4,3 (no fuel melt)
Batch 4,3,2	Clad failure of batch 4,3,2 (no fuel melt)
Batch 4,3,2,1	Clad failure of batch 4,3,2,1 (no fuel melt)
Boil/Melt	Boiling and/or fuel melting

III.B.4. IRF – Operational Space

There was no refinement of the operational space for the IRF analysis, since, as mentioned in Section III.A.4, the SAS4A/SASSYS-1 simulations were fast running and the number of uncertainties was limited.

III.B.5. IRF – Determine Refined SC

The detailed SC assessement once again examined the ULOF, ULOHS, and UTOP transient sequences, but utlized the IRF FDC thresholds and modified sampling schemes.

First, 5,000 Monte Carlo simulations were performed to propagate the uncertainties for each of the three transient sequences. Example results for the ULOF transient sequence are shown in Fig. 6., which illustrates the change in peak fuel temperatures with the selected value for each of the IRF uncertainties. As can be seen, since the transient involves a loss-of-primary flow, the GEMs are particularly influential on the response of the reactor system.

As with the RVACS analysis, it was possible to construct CCDF curves illustrating IRF performance for different event sequences, with an example CCDF curve for the UTOP sequence in Fig. 5. As the figure shows, one downside of utilizing Monte Carlo (MC) sampling is that the lowest probability event assessed is limited by the number of simulations conducted. For example, with 5,000 simulations, the probability of occurrence for each simulation is 0.0002. Therefore, the lowest probability resolution of the CCDF curve is 0.0002. The only way to explore the regions of the operational space with much lower probability values using an MC sampling scheme is to drastically increase the number of simulations conducted. This was not feasible with the current computational capabilities available to the project.

Due to the limitations with the MC sampling technique, the detailed IRF SC analysis was repeated using a full factorial method similar to that used for RVACS. In this way, several points were selected from each IRF parameter uncertainty distribution, heavily biased toward values that would degrade system performance. This process was more difficult for the IRF analysis, as an increase in a parameter value may be detrimental in some event sequences and beneficial in others. Therefore, sensitivity analyses were first performed to determine the uncertainty value direction that would degrade system performance for each transient sequence. From there, a 4096-simulation full factorial experiment was conducted utilizing the selected values. Fig. 5 compares the full factorial and MC sampling experiments for a UTOP. As the results show, even though fewer simulations were performed with the full factorial experiment, the results provide greater detail at lower probabilities (while also agreeing with the Monte Carlo results within an order of magnitude at high probability values).

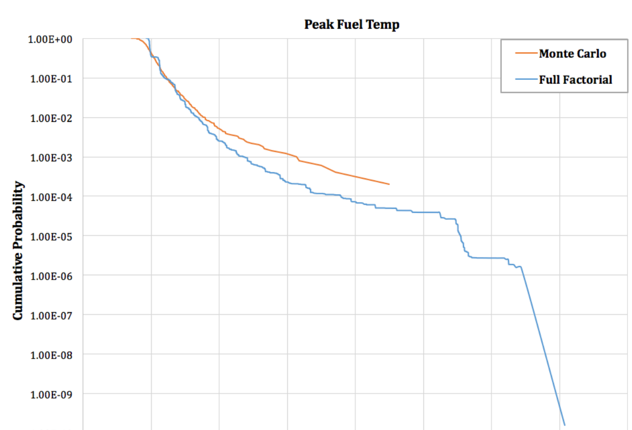

Fig. 5. IRF – UTOP CCDF[e]

[e] Temperature values removed due to proprietary restrictions.

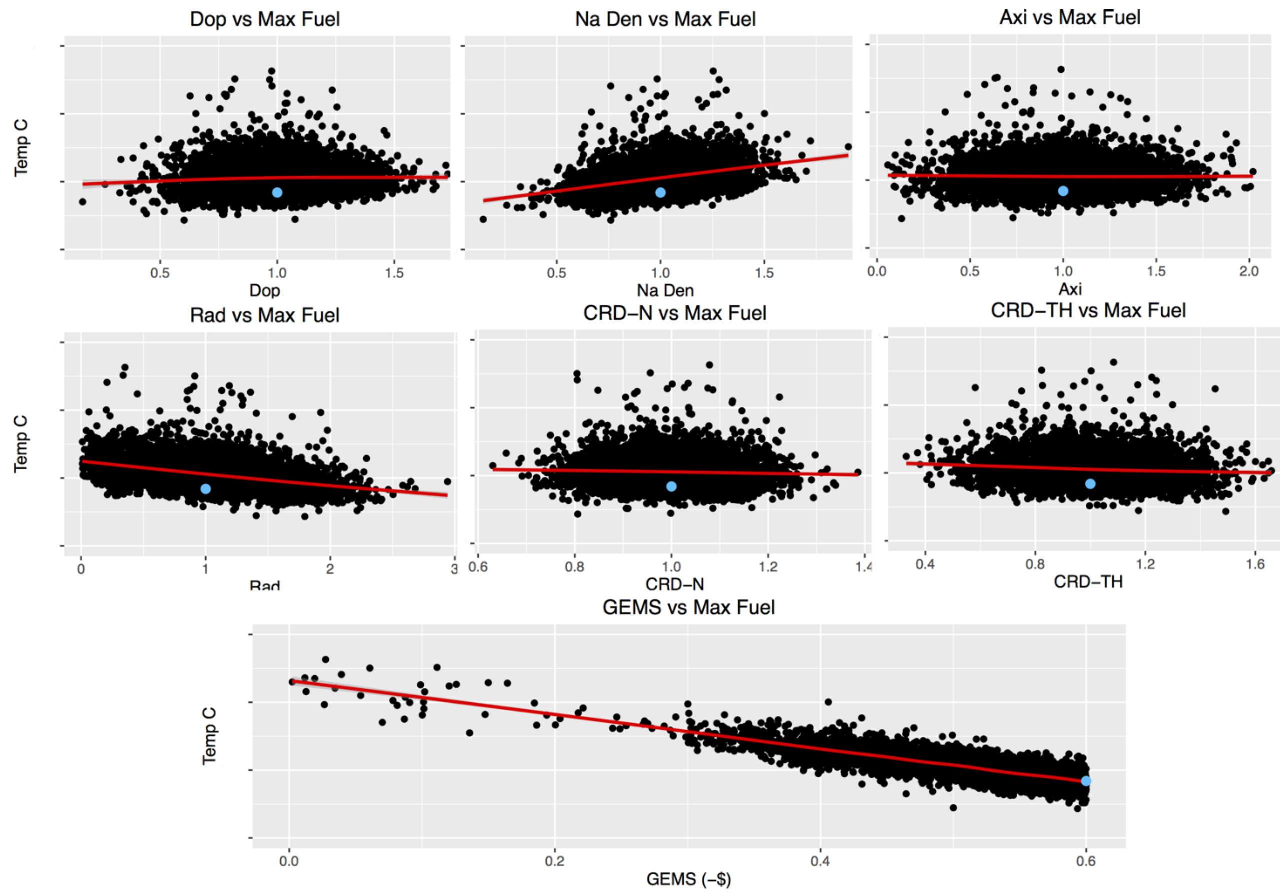

Fig. 6. IRF – ULOF Max Fuel Temperature Results by Uncertainty[f]

The results of the refined SC analysis were then used directly within the PRA event trees to populate the branching probabilities for each FDC. Since the FDC thresholds do not rely only on fuel/hot pool temperature, as they did with the RVACS, the FDC thresholds are not shown in Fig. 5, but were determined by examining the output of each simulation via an automated process. In general, the IRF analysis resulted in low probabilities of insufficient IRF response to prevent the exceedance of a fuel damage FDC threshold for the transient sequences assessed.

IV. CONCLUSION

The passive system reliability and SC analysis methodology, outlined in Section I.A., provided a straightforward approach for determining passive system reliability while ensuring consistency with the PRA event sequences and MST categories. The method also addressed a direct concern of the NRC during previous licensing interactions and satisfies the newly developed ASME/ANS Non-LWR PRA standard. Additionally, the breakdown of the passive system assessment into stages (preliminary and detailed) allowed for initial PRA calculations to be performed that provided additional information regarding the importance of particular event sequences to overall system risk. This permitted those event sequences deemed important to be analyzed in greater detail with refined SC. The methodology was also able to examine two passive systems with unique phenomenological characteristics (engineered safety system and inherent material properties).

Perhaps the greatest difficulty in the analysis process was determining a method to propagate uncertainties that could capture extremely low probability events. Passive systems are attractive due to their high reliability, but this quality can prove challenging to properly represent when using MC sampling techniques. As the RVACS and IRF results show, modified sampling schemes, such as full factorials, can help address this difficulty, but great care must be taken when constructing such an experiment as analyst knowledge (and opinion) can influence the selection of sampling points and probabilities and introduces bias in the final result. In contrast, one of the

[f] Temperature values have been removed from the figure due to proprietary restrictions. Blue dots indicate simulation result utilizing nominal values. One fuel pin failure case has been excluded from the plots. X-axis values for IRFs are in comparison to nominal value.

benefits of the sampling approaches chosen here, which examined the entire operational uncertainty space in search of failure thresholds, was that parameter uncertainty distributions could be modified post-analysis without the need for additional simulations.

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