

Sensitivity and Uncertainty Quantification of the Fuel Salt Temperature during an Uncontrolled Control Rod Withdrawal Accident in the MSRE Using Modelica/TRANSFORM and RAVEN

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1. Introduction

Probabilistic risk assessment (PRA) of light water reactors (LWRs) has conventionally been performed within a static framework, in which risk is quantified through event trees and fault trees supported by operating data and conservative assumptions. Molten salt reactors (MSRs), as an advanced reactor concept, lack the accumulated operating experience on which static event trees and fault trees rely, and this shortage limits the direct application of static PRA to MSRs. A new approach to the safety assessment of MSRs is therefore required. Dynamic PRA (DPRA) meets this requirement, as it is a probabilistic safety assessment method that explicitly treats the temporal ordering of accident progression and the physical state of the system.

Several tools have been developed to implement DPRA by executing and managing external analysis codes while generating and controlling accident scenarios. These include ADAPT (Analysis of Dynamic Accident Progression Trees) [1], originally developed at Ohio State University (Aldemir et al.) and subsequently advanced at Sandia National Laboratories; RAPID (Risk Assessment with Plant Interactive Dynamics) [2], developed at the Japan Atomic Energy Agency; and RAVEN (Risk Analysis Virtual ENvironment) [3], developed at Idaho National Laboratory (INL).

Several preceding studies have moved toward the DPRA of MSRs. Williams et al. [4] performed uncertainty quantification of a generic MSR loop model in which COBRA-TF (CTF) was coupled with TRANSFORM. Fischer and Bureš [5] constructed a thermal-hydraulic system model of the MSRE with the TRANSFORM library and validated it against the operating data.

The Molten Salt Reactor Experiment (MSRE) was an experimental reactor built and operated at ORNL to demonstrate the operational feasibility of a liquid-fueled MSR. Its design data, property measurements, and operating and test records are publicly available in the form of technical reports, and the MSRE is therefore the liquid-fueled MSR for which the information required to construct a system-level model can be obtained. Uncontrolled control rod withdrawal (UCRW) is an accident explicitly addressed in the ORNL technical reports, which induces power and temperature transients caused by the withdrawal of the control rods [1].

In the present study, sensitivity and uncertainty quantification was carried out in a framework coupling RAVEN with the Modelica/TRANSFORM library. For this analysis, a goal function defined in terms of the fuel salt temperature was established. In addition, a set of uncertain parameters associated with the UCRW accident was selected on the basis of the preceding studies.

2. Methods

2.1. MSRE fuel salt system

The MSRE is a liquid-fueled reactor in which fission is induced as the fuel salt flows through the graphite grid of the primary loop. The heat released in the core is transported by the circulating fuel salt to the primary heat exchanger (PHX), where it is transferred to the coolant salt of the secondary loop. The overall specifications of the primary loop under steady-state conditions and the thermophysical property data of the fuel and coolant salts are summarized in Table 1 and Table 2, respectively.

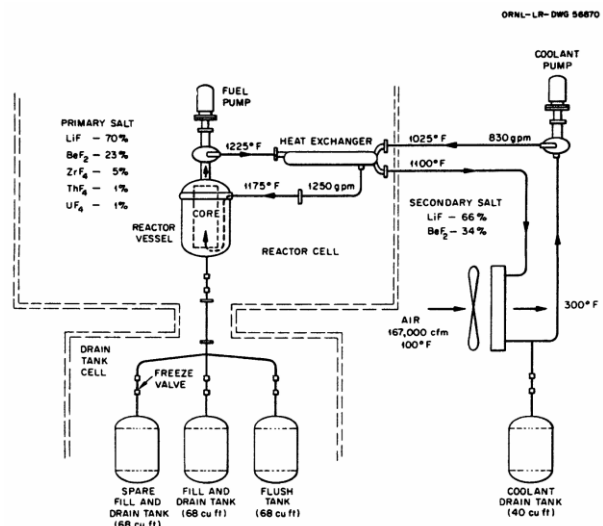


Fig. 1. Flow diagram of the MSRE [6].

Table 1. Steady-state specifications of the MSRE primary loop

Parameter	Value
Reactor thermal power	10 MW

Parameter	Value
(design)	
Reactor thermal power (operated)	8.0 MW
Core inlet temperature	635 °C
Core outlet temperature	663 °C
Fuel salt flow rate	0.0757 m ³ /s
Coolant salt flow rate	0.0536 m ³ /s
Fuel salt volume in primary loop	1.996 m ³

Table 2. Thermophysical properties of the fuel and coolant salts

Property	Fuel salt	Coolant salt
Composition (mol%)	LiF-BeF ₂ -ZrF ₄ -UF ₄ (65-29.1-5-0.9)	LiF-BeF ₂ (66-34)
Density (kg/m ³)	2242 (at 648.9°C)	2001 (at 571.1°C)
Specific heat (J/kg·K)	1982	2416
Thermal conductivity (W/m·K)	1.44	1.00
Dynamic viscosity (mPa·s)	7.73	9.91
Liquidus temperature (°C)	434	458

2.2. UCRW accident sequence

The withdrawal of a control rod inserts positive reactivity into the core and raises the power and the temperature of the fuel salt. The PHX is the first component that mitigates this temperature rise, since the elevated core outlet temperature increases the heat transferred to the coolant salt of the secondary loop. A reactor scram is subsequently demanded. In the MSRE, the control rods were designed to be dropped automatically upon a signal of short reactor period, high reactor power, or high fuel salt outlet temperature, and a manual scram performed by the operator after identification of the accident is credited as a backup in the event that the automatic actuation fails.

If the scram fails and the heat removal through the PHX is insufficient, the following sequence is initiated. As the fuel salt temperature continues to rise, freeze valve(FV-103) is thawed automatically by the temperature of the fuel salt, and the fuel salt passing through the valve is drained into the DTS(Drain Tank System) by gravity. DT-1(Drain Tank-1) is the first tank to receive the overheated fuel salt, and the decay heat of the fuel salt is removed by the DHRS(Decay Heat Removal System) of DT-1. If the DHRS fails to remove the decay heat, the fuel salt is transferred to DT-2. Failure to cool the fuel salt in DT-2 may result in failure of the structural boundary, leading to a salt spill accident [7].

2.3. TRANSFORM library

Modelica is a non-proprietary, object-oriented, equation-based modeling language in which a multi-

domain physical system is described acausally as a system of differential-algebraic equations (DAEs) rather than as a sequence of assignment statements, allowing components to be connected without predefining the input-output causality. Dymola is a Modelica-based simulation environment that provides a graphical user interface through which the components and systems of libraries written in the Modelica language can be utilized, and OpenModelica is another representative implementation environment of the language. The TRANSFORM library is an open-source developed at Oak Ridge National Laboratory (ORNL) in the Modelica language, and it is designed so that components can be freely customized by the user. TRANSFORM library is intended to be applicable to various advanced reactor concepts, and for MSRs in particular, it has been employed in the analysis of the transport of delayed neutron precursors (DNPs) and fission products. The library supports the Multiphysics analysis of MSRs by providing fluid and heat transfer components, a Point-kinetics model accounting for the transport of DNPs in circulating fuel, and molten salt property packages.

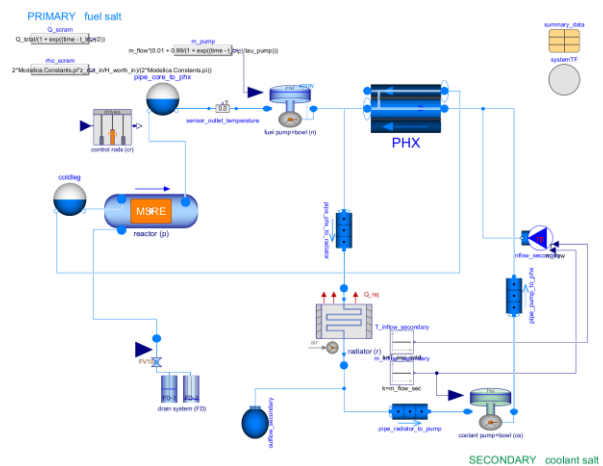


Fig. 2. MSRE fuel salt system model developed with the Dymola/TRANSFORM library and employed as the analysis tool of the present study.

2.4. Goal function

In sensitivity and uncertainty analysis, a single metric is required to quantitatively judge the results of the multiple simulations performed over the uncertain parameters. This metric is referred to as the goal function, and it serves as the criterion for determining whether the safety function of the system is maintained. In LWRs, core damage is established as the end state of severe accidents, whereas no corresponding criterion has yet been established for MSRs, in which the fuel salt circulates. The MSR phenomena identification and ranking table (PIRT) treats the salt spill accident, in which the fuel salt is released outside the system, as an accident directly linked to the release of radioactive material to the environment, and in the present study the salt spill accident was regarded as the end state

corresponding to core damage in LWRs. The goal function was therefore defined in terms of the thermal conditions that may lead to a salt spill, namely the temperature at which the integrity of the structural material containing the fuel salt is lost. In a UCRW accident, the fuel salt temperature may rise rapidly owing to the inserted reactivity. The primary loop and the drain tanks are both constructed of INOR-8, and the limiting temperature of this alloy was extrapolated in an ORNL evaluation from both the maximum allowable design stress and the ultimate tensile strength to lie between 900 and 1000°C [7]. In that evaluation, 955°C, which is 260°C above the normal fuel salt hot-leg temperature, was taken as a conservative estimate of the temperature that would produce a failure of the structural integrity of the drain tank. The temperature target of the present study, T_{limit} , was set to 955°C on this basis, and the value was adopted for the primary loop as well because the two share the same structural material. Because structural integrity is governed by stress and strength as well as by temperature, T_{limit} is used as a conservative temperature target for the goal function rather than as a strict failure criterion. The goal function of the present study is therefore defined as follows:

$$g(T_{fuel,peak}) = T_{limit} - T_{fuel,peak} \quad (1)$$

$$G(T_{fuel,peak}) = \begin{cases} +1, & (g > 0) \\ -1, & (g \leq 0) \end{cases} \quad (2)$$

In Eq. (1), g denotes the margin function, $T_{fuel,peak}$ is the peak fuel salt temperature reached over the course of each simulation, and T_{limit} is the temperature target of 955°C introduced above. Equation (2) maps this margin onto the binary indicator G , which takes the value +1 when the peak temperature remains below the target ($g > 0$) and -1 when the target is reached or exceeded ($g \leq 0$). Each simulation is thereby classified as a success for $G = +1$ and as a failure for $G = -1$. This ± 1 form is the convention in which a goal function is evaluated in RAVEN, where the resulting classification defines the limit surface, the boundary that separates system success from system failure.

2.5. Uncertain parameters

The PIRT classifies the UCRW accident into Scenario Category II, dynamic system thermal/hydraulic/power response, and presents the associated material properties and other key parameters as follows [8]:

- Parameters associated with the plant's operating state at the start of the transient or accident
- Material properties listed in Scenario Category I, including fouling and corrosion properties
- Reactivity feedback coefficients
- Reactivity worth of control mechanisms (dk/k)

Among the presented key parameters, the material properties listed in Scenario Category I, including the fouling and corrosion properties, cannot be calculated in

the present model and were therefore excluded from the uncertain parameters. To reflect the parameters associated with the plant's operating state at the start of the transient or accident in the analysis, the initial power, and the fuel salt flow rate were selected as uncertain parameters. The reactivity feedback coefficients are a collective term for the coefficients that represent the relations through which changes in the system state variables are fed back into the reactivity; among them, the temperature feedback coefficients (α_T) and feedback reference temperature (T_{ref}) determine the reactivity change induced by the temperature change of each medium. In the modified point-kinetics model of TRANSFORM, the reactor power is governed by Eq. (3), and the temperature feedback is implemented in the form of Eq. (4) and summed into the reactivity term $\rho(t)$ of Eq. (3) [9].

$$\frac{dP(t)}{dt} = \frac{\rho(t) - \beta + \Delta\rho_{fp}}{\Lambda} P(t) + \frac{w_f}{\Lambda v} \sum_k \lambda_k C_k(t) + \frac{w_f}{\Lambda v} S_{ext} \quad (3)$$

$$\Delta\rho_T = \alpha_T (T - T_{ref}) \quad (4)$$

In Eq. (3), P is the reactor power, $\rho(t)$ is the reactivity, β is the total delayed neutron fraction, $\Delta\rho_{fp}$ is the reactivity contribution of the fission products, Λ is the neutron generation time, w_f is the energy released per fission, v is the number of neutrons produced per fission, λ_k and C_k are the decay constant and the concentration of the k -th DNP group, respectively, and S_{ext} is the external neutron source. In Eq. (4), $\Delta\rho_T$ is the reactivity change due to the temperature feedback, α_T is the temperature feedback coefficient, T is the fluid temperature, and T_{ref} is the nominal reference temperature of each fluid.

The reactivity worth of control mechanisms is a key parameter of the accident analysis presented by the PIRT, and the control rod worth was selected as an uncertain parameter in the present study. The uncertain parameters selected in the present study and their sampling ranges are summarized in Table 3.

Table 3. Uncertain parameters and their sampling ranges

Parameter	Lower limit	Upper limit
Initial reactor power Q_0 [MW]	9.0	11.0
Feedback reference temperature T_{ref} [K]	903.4	941.04
Fuel salt flow rate [gpm]	1140	1260

Fuel temperature feedback coefficient α_{fuel} [K ⁻¹]	-7.128×10 ⁻⁵	-4.752×10 ⁻⁵
Graphite temperature feedback coefficient α_{graphite} [K ⁻¹]	-9.324×10 ⁻⁵	-3.996×10 ⁻⁵
Control rod worth w_{rod}	0.0201843	0.0246697

The data for each parameter listed in Table 3 were established on the basis of the design values of the MSRE and the comparisons between the measured and calculated results reported in the ORNL technical reports. Since the available information is not sufficient to specify the shape of the distribution of each value, a uniform distribution was assumed for all parameters. The selected parameters are defined as the sampling variables of the multiple simulations within the RAVEN input deck, and RAVEN generates the Dymola model inputs corresponding to each sample and performs the repeated simulations.

2.6. RAVEN framework

RAVEN is an open-source uncertainty quantification and PRA framework developed at Idaho National Laboratory(INL). It does not perform the physical calculations itself; instead, it generates and manages multiple simulations of an external system code. RAVEN is a platform that is coupled with the system code through an input deck in the XML file format and controls the system code through the Python interface of the RAVEN library. The framework supports coupling with various analysis codes such as RELAP5, MAAP5, Modelica, and Dymola. In the present study, RAVEN is employed as follows: (1) the uncertain parameters are assigned probability distributions, and (2) the uncertain parameters are randomly sampled within the distributions and substituted into the conditions of the system code; (3) the system code performs the physical simulation under the sampled conditions, and (4) the PostProcessor of RAVEN post-processes the results of the system code according to the purpose of the study.

RAVEN provides calculation models for the Pearson coefficient and the Spearman coefficient. The two coefficients are used to examine the relations between the perturbed variables and the figures of merit. The Pearson coefficient measures the degree of the linear relation between two variables and is defined as Eq. (5).

$$r(X, Y) = \frac{\text{Cov}(X, Y)}{\sigma_X \sigma_Y} \quad (5)$$

It takes a value between -1 and +1, where +1 indicates a perfect positive linear relation, -1 a negative linear relation, and 0 the absence of a linear relation. The Spearman coefficient is obtained by applying the Pearson model to the sample data arranged by their ranks rather than by their actual values, as defined in Eq. (6).

$$r_s(X, Y) = \frac{\text{Cov}(R(X), R(Y))}{\sigma_{R(X)} \sigma_{R(Y)}} \quad (6)$$

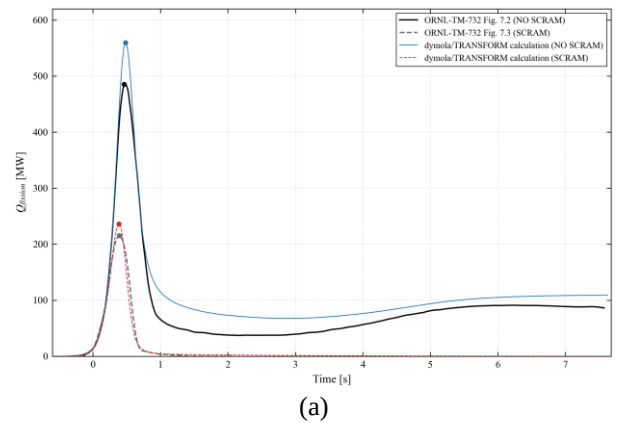
In Eq. (5) and Eq. (6), $\text{Cov}(X, Y)$ is the covariance of the variables X and Y , σ_X and σ_Y are the standard deviations of X and Y , and $R(X)$ and $R(Y)$ denote the ranks of X and Y , respectively. Since the sample data are not used directly, the Spearman coefficient shows a value close to +1 even for data exhibiting a nonlinear monotonically increasing relation.

RAVEN provides the Pearson/Spearman coefficient PostProcessor model, which computes the Pearson/Spearman coefficients of the output data of the system code [3].

3. Results and Discussion

3.1. UCRW accident analysis

The model was verified against the UCRW accident simulation reported in ORNL-TM-732 [6]. The UCRW simulation begins with the entire system isothermal at 1200°F and the reactivity held at criticality. The three control rods are withdrawn simultaneously at 0.5 in./s, and when the power of the MSRE reaches 15 MW, two of the three rods are dropped while the one that cannot be controlled continues to be withdrawn. The NO SCRAM case does not consider the control rod drop. In addition, to compare the UCRW results with the ORNL-TM-732 data under the same conditions, heat removal by the PHX was excluded entirely from dymola code.



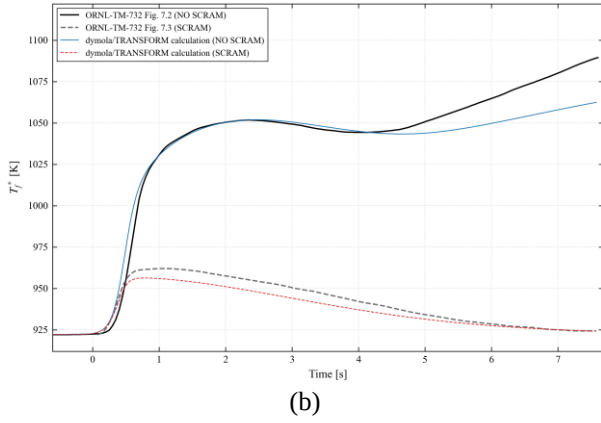
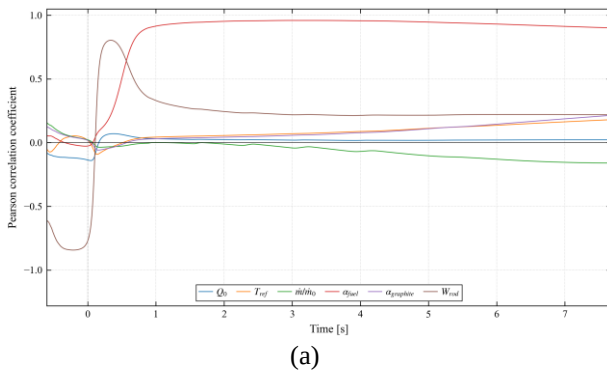


Fig. 3. Comparison of the Dymola UCRW accident simulation results with the ORNL-TM-732 data: (a) fission power (Q_{fission}) and (b) average fuel salt temperature (T_f^*).

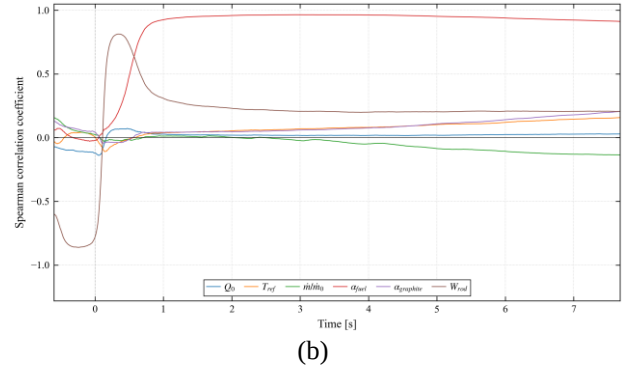
Fig. 3-(a) shows the fission power for the two accident paths, and Fig. 3-(b) shows the average fuel salt temperature T_f^* obtained from the same calculations. In both panels, the time axis is set to zero at the moment the power crosses 15 MW. Without the SCRAM, the control rod withdrawal raises the reactivity sharply, and the instantaneous thermal power rises accordingly. The sharp decrease in power at the thermal power peak is produced by the fuel and graphite temperature feedback coefficients. The power thereafter does not fall to zero but passes through a minimum and rises again. The continued withdrawal of the control rods accounts for this recovery. In the SCRAM case, the power converges to zero once it reaches 15 MW and the two control rods drop.

In the cases calculated with RAVEN coupled to Dymola, the PHX was deliberately included. The PHX is the first heat sink to respond during a UCRW accident. However, the analysis confirmed that the PHX has a limited mitigating effect because its heat removal capacity is approximately 10 MW, which is insufficient to counteract a short-duration power excursion reaching several hundred megawatts.

3.2. Sensitivity and uncertainty estimation: No SCRAM



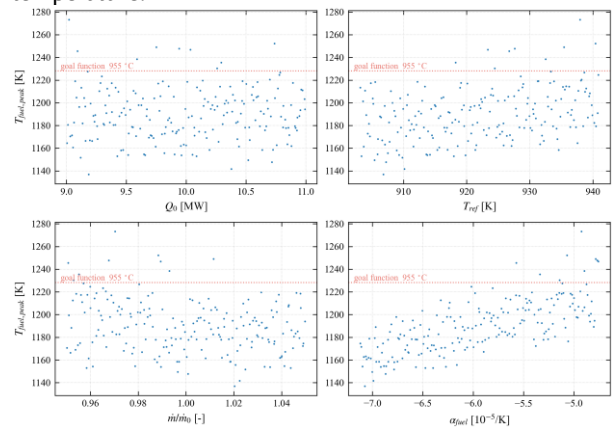
(a)



(b)

Fig. 4. Comparison of the Pearson and Spearman coefficients for the maximum fuel salt temperature (No SCRAM): (a) Pearson coefficient and (b) Spearman coefficient.

Fig. 4-(a) shows the Pearson coefficients for the maximum fuel salt temperature as a function of time, and Fig. 4-(b) shows the Spearman coefficients for the same quantity in the case of No SCRAM. Both coefficients evolve in almost the same manner. When the UCRW accident occurs, the coefficients change abruptly, and as the thermal power stabilizes, the coefficients of all the uncertain parameters converge. The correlation coefficient of the fuel temperature feedback coefficient rises sharply after the accident and approaches a quasi-steady value close to unity. This near-unity correlation indicates that the fuel temperature feedback coefficient is linearly proportional to the maximum fuel salt temperature immediately after the UCRW and exerts a dominant influence on that temperature. The coefficient of the control rod worth is initially large and negative, and it increases when the rods are withdrawn at the onset of the accident. After reaching its local maximum, the coefficient decreases sharply and then converges to a positive value. This transition shows that the control rod worth exhibits a strong linear proportionality to the maximum fuel salt temperature near the peak of the thermal power, but that the correlation is no longer valid once the thermal power stabilizes. The remaining uncertain parameters are confirmed to have no significant correlation with the maximum fuel salt temperature.



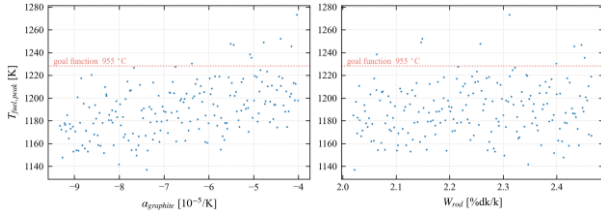


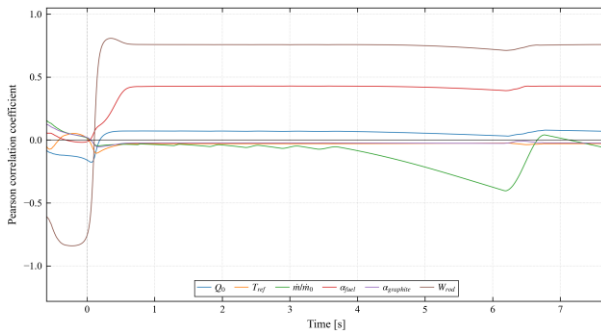
Fig. 5. Scatter plots of the peak fuel salt temperature $T_{\text{fuel,peak}}$ obtained from the multiple simulations against each uncertain parameter, with the 955°C goal-function upper limit marked by the red dotted line.

Among the multiple simulation results, the maximum fuel salt temperature exceeded the 955°C specified as the temperature target of the goal function in a number of cases. Nine such cases were found in total, and the combinations of the uncertain parameter values sampled for each of them are listed in Table 4.

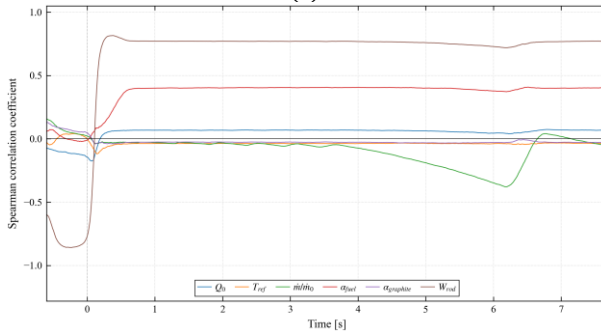
Table 4. Sampled uncertain parameter combinations, normalized to a range of 0 to 1, for the nine cases in which the maximum fuel salt temperature exceeded the 955°C goal-function upper limit.

Sample	Q_0	T_{ref}	$m/\dot{m}_0$	α_{fuel}	α_{graphite}	W_{rod}
7	0.52	0.53	0.39	1.00	0.73	0.95
8	0.87	0.98	0.38	0.87	0.92	0.29
19	0.65	0.40	0.05	0.75	0.80	0.97
82	0.30	0.78	0.43	0.94	0.80	0.10
95	0.47	0.65	0.18	0.99	0.72	0.53
107	0.01	0.92	0.20	0.93	0.99	0.65
115	0.63	0.56	0.03	0.84	0.55	0.85
121	0.38	0.85	0.62	0.99	0.85	0.29
167	0.05	0.62	0.01	0.66	0.97	0.92

3.3. Sensitivity and uncertainty estimation: SCRAM



(a)



(b)

Fig. 6. Comparison of the Pearson and Spearman coefficients for the maximum fuel salt temperature (SCRAM): (a) Pearson coefficient and (b) Spearman coefficient.

In the cases in which the accident was mitigated by a SCRAM, the control rod worth was observed to have the largest coefficients. Because the control rods are inserted by the SCRAM immediately after the accident and are held in that position, both the Pearson and Spearman coefficients of the control rod worth converge to approximately 0.75. The fuel temperature feedback coefficient, by contrast, exhibits a weaker correlation with the maximum fuel salt temperature than in the NO SCRAM case, because the thermal power has already been reduced by the SCRAM. For the remaining uncertain parameters, no appreciable correlation with the maximum fuel salt temperature is found.

The present study nevertheless presents several limitations. In the ORNL analysis data, the maximum salt temperature was defined as the temperature of the hottest of the individual salt channels passing through the graphite grid. The graphite grid is not modeled separately in the present study, and the maximum temperature is instead taken as the highest bulk temperature of the fuel salt. For this reason, the maximum salt temperature obtained from the simulations may be somewhat underestimated. A further limitation is that the temperature limits specified for the goal function were taken directly from material property data, so that their conservatism in a safety assessment is not guaranteed when they are applied directly to a PRA.

4. Conclusion

The present study has performed a DPRA-oriented sensitivity and uncertainty analysis of the MSRE UCRW accident by coupling RAVEN with a fuel salt loop model built with Dymola and the TRANSFORM library. Probability distributions were assigned to the uncertain parameters selected from the MSR PIRT, 200 samples were drawn with the LHS scheme, and the goal function was set by the 955°C conservative temperature target of INOR-8. The model was verified against the UCRW simulation reported in ORNL-TM-732 for both the NO SCRAM and the SCRAM paths.

Without a SCRAM, the fuel temperature feedback coefficient showed a near-unity correlation with the maximum fuel salt temperature, whereas the control rod worth correlated strongly with that temperature only near the thermal power peak and lost the correlation once the power stabilized. Nine cases exceeded the 955°C temperature target. With a SCRAM, the control rod worth exhibited the largest coefficients, converging to approximately 0.75, and the fuel temperature feedback correlation weakened. Two limitations qualify these results: the maximum salt temperature was taken as the highest bulk temperature of the fuel salt, and the goal-function limits came directly from material property data without guaranteed conservatism. Refining both remains

as future work toward the application of the present approach to PRA.

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