A Dynamic Coupled-Code Assessment of Mitigation Actions in an Interfacing System Loss of Coolant Accident

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Abstract: Containment bypass scenarios in nuclear power plants can lead to large early release of radionuclides. A residual heat removal (RHR) system interfacing system loss of coolant accident (ISLOCA) has the potential to cause a hazardous environment in the auxiliary building, a loss of coolant from the primary system, a pathway for early release of radionuclides, and the failure of a system important to safely shutting down the plant. Prevention of this accident sequence relies on active systems that may be vulnerable to cyber failures in new or retrofitted plants with digital instrumentation and control systems. RHR ISLOCA in a hypothetical pressurized water reactor is analyzed in a dynamic framework to evaluate the time-dependent effects of various uncertainties on the state of the nuclear fuel, the auxiliary building environment, and the release of radionuclides. The ADAPT dynamic event tree code is used to drive both the MELCOR severe accident analysis code and the RADTRAD dose calculation code to track the progression of the accident from the initiating event to its end states. The resulting data set is then mined for insights into key events and their impacts on the final state of the plant and radionuclide releases.

Keywords: Dynamic PRA, ADAPT, ISLOCA, DET.

1. INTRODUCTION

Probabilistic Risk Assessment (PRA)[1] is a powerful tool for identifying the events and sequences of events that have an impact in Nuclear Power Plant (NPP) accidents. The results of a PRA can be mined for insights that may be used toward such diverse goals as developing procedures for abnormal or emergency conditions, requesting license amendments, or scheduling maintenance and repair. Sequences with the potential for a Large Early Release (LER) of radionuclides beyond the site are of particular concern with regard to public dose when compared to similarly-composed releases that occur later in the accident [1].

This paper examines a Light Water Reactor (LWR) Interfacing System Loss of Coolant Accident (ISLOCA) which has the potential for early releases [2]. The high pressure Reactor Coolant System (RCS) and lower pressure systems such as Residual Heat Removal (RHR) must intermittently interface in an NPP. For example, flow must be established through both systems during shutdown while the RHR system is isolated during at-power operation. Inadvertent communication between the systems is generally prevented with check valves or active Motor Operated Valves (MOVs) (or a combination of both types of valves) depending on the requirements of the interface. If inadvertent communication does occur, it is possible for the lower-pressure RHR system to become overpressurized and for components to rupture [3]. Such a rupture may, in turn, lead to a loss of inventory that may lead to damage to the fuel and release of radionuclides through the pathway opened by the ruptured component [4].

Due to the potential for a LER, system interfaces that can lead to an ISLOCA are typically designed so that a single failure cannot open the pathway. The isolation scheme has been assessed in existing plant configurations to greatly reduce the probability of an ISLOCA [4]. Due to their high potential

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[1] No distinction is made between PRA and Probabilistic Safety Assessment (PSA) in this paper.

consequences, ISLOCAs are often included in risk analyses despite their low probability [1, 5]. The particular interface examined in this paper, the RHR suction from the RCS (see Section 2.2), is typically isolated from the RCS using two independently-powered and controlled MOVs in series. It is assumed that both MOVs at the hypothetical plant have been upgraded to digital control and that a common controller fault causes both to open simultaneously during at-power operation. The cause and frequency [6, 7, 8] of the initiating event are not examined in this paper. The analysis and associated insights should be taken as being conditional upon the initiating event.

This paper examines the impacts of uncertainties of events that follow the initiating event through potential hardware failures and mitigative operator actions. Various uncertainties such as component capacities and timings of operator actions are of particular interest. These uncertainties are generally difficult to represent using traditional PRA and thus a Dynamic Probabilistic Risk Assessment (DPRA) approach is used to capture their impacts [9]. The progression of the accident is tracked in response to each uncertainty by generating a Dynamic Event Tree (DET) to be mined for insights into impactful parameters. The ADAPT[2] DET driver has been used for previous analyses along this line of research [10].

This work presented in this paper leverages new features of ADAPT to expand the potential scope of the DET and to complete it faster than would have been possible before [11]. Many potential operator actions in the hypothesized accident must be taken outside of the control room within the auxiliary building. Due to the nature of an ISLOCA outside containment, the auxiliary building may reach a hazardous environmental condition. In order to represent both the plant systems and the potential contamination of the auxiliary building, two simulators are used under ADAPT. The MELCOR severe accident code [12] is used to model the response of the core and thermal-hydraulic plant systems to the initiating event and mitigative actions. Radionuclide Transport, Removal, and Dose Estimation (RADTRAD) [13] is used to estimate the dose rate in different rooms throughout the auxiliary building given the profile of contamination from an RHR component burst.

By combining MELCOR and RADTRAD under a unified analysis, a richer set of data is generated. Insights are drawn from this set to assess the importance of system reliability and operator actions in an RHR ISLOCA. Previous analyses involving this initiating event evaluated the operator actions assuming operators were able to move through the building [14, 10]. By combining MELCOR with RADTRAD, an expected dose is determined for each action and a corresponding determination made for whether the action can proceed without incurring an unacceptable dose. The plant system and transient are detailed in Section 2. Section 3 develops the multi-simulator ADAPT model of the accident. Results are presented in Section 4 and the impact of the analysis is summarized in Section 5.

2. PLANT TRANSIENT

This section details the plant configuration and nominal accident sequence for the hypothesized RHR ISLOCA in Sections 2.1 and 2.2, respectively. The models used to represent the system are briefly described in Section 2.3.

2.1. Plant Configuration

The plant considered in this work is a hypothetical three loop Pressurized Water Reactor (PWR) with a primary RCS pressure of 15.5 MPa. The Low Pressure Safety Injection (LPSI) and RHR systems

[2] ADAPT is no longer an acronym.

share pumps, valves, and a significant amount of piping[3]. RHR provides cooling of the RCS water during shutdown operations. LPSI provides up to 3750 gpm of water makeup during accidents where the RCS is at a sufficiently low pressure. The two pumps and two Heat Exchangers (HXs) of the RHR system are assumed to be located outside containment but within the auxiliary building as is the case in several operating plants [15]. The RHR system has a maximum operating pressure of 2 MPa but individual components may have been rated for a higher pressure. Water enters the RHR system from a hot leg of the RCS during shutdown. During operation, this pathway is isolated by two MOVs in series. When the valves are open, water passes through the RHR pumps to the RHR HXs which are cooled by Component Cooling Water (CCW). CCW is also required for cooling of the RHR pumps and High Pressure Safety Injection (HPSI) pumps. Cooled water returns to a cold leg of the RCS. The direction of flow at the RHR outlet is controlled using check valves.

Figure 1 shows the lower level of the hypothetical plant's auxiliary building where the RHR pumps and HXs reside. The three HPSI pumps, which provide high-head injection to the core in the event of an accident, are also located on this floor. HPSI operates up to 12 MPa and may provide flow up to 375 gpm. HPSI operates at a lower pressure than the 15.5 MPa RCS but the pressure differential is small enough that damage is not expected for inadvertent communication at full RCS pressure. Both LPSI and HPSI primarily draw water from the Refueling Water Storage Tank (RWST) which is outside the auxiliary building. LPSI may also draw from the containment sump if sufficient water exists in the sump [16]. Auxiliary Feedwater (AFW), which provides water to the steam generators when the main feedwater system is inoperative, draws water from the Condensate Storage Tank (CST) which is outside of the building. Each room on this floor that houses pumps or HXs is assumed to have two 3 inch diameter floor drains. There are assumed to be 10 similar drains in the main hallway of the floor. If an RHR component ruptures, the rupture may lead to a flood on this floor that may damage equipment and prove hazardous to operators due to radiation and steam. Flood water and contamination may be contained by room doors if they are latched and are sealing properly [17].

A pressurizer resides on one of the hot legs of the RCS and has two Pilot-Operated Relief Valves (PORVs) which may be used in an accident to regulate pressure and inventory in the RCS. The PORVs lead to a tank within containment that, when filled, will burst and drain into the containment sump. The PORVs may be opened automatically at a higher-than-normal RCS pressure or manually by operators. Manual opening has the potential to be helpful in an ISLOCA because RCS inventory and radionuclides may be blown into containment rather than through a leak path outside of containment [5]. Water that passes through the PORVs may eventually reach the containment sump and be available for later injection.

2.2. Accident Sequence

The initiating event for this accident is an opening of both RHR isolation valves simultaneously at full power and RCS pressure. Communication with the 15.5 MPa RCS will raise the pressure of the RHR system possibly leading to component failures. Three potential failures are modeled in this sequence. First, the RHR suction line between the isolation valves and the pump may fail. This line corresponds to the leak location *BRK-1* in Figure 1. The hypothesized break near the RHR pumps may lead to a large loss of coolant and release of radionuclides outside containment.

The system availability impacts of *BRK-1* are that RHR suction from the hot leg and LPSI suction from the RWST are disabled for the duration of the accident and establishing flow using water from the

[3] In this paper, the shared portions of the LPSI/RHR system are referred to generally as RHR. Discussion of the injection function specifically uses the term LPSI.

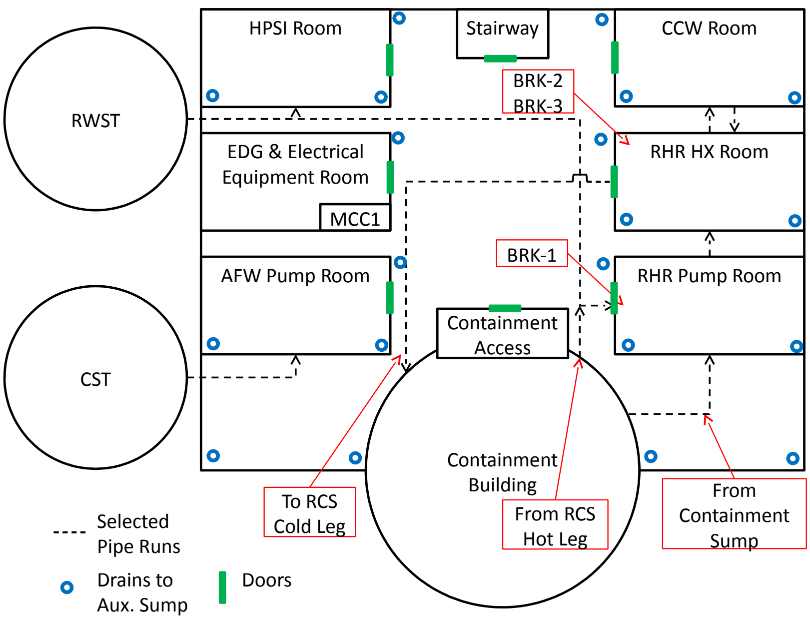

Figure 1: Auxiliary Building Lower Level Layout

containment sump will require manual operator action in the RHR pump room (see Figure 1). RCS inventory will spill into the room until the break is isolated which requires manual operator action at an MOV controller. Additionally, the normally-open pathway from the RWST [4] will drain water into the RHR pump room until closed from the control room. Operators will not attempt actions until the water level in the room with the required action (see Figure 1) is under 3 inches. This decision rule captures uncertainty in whether operators can safely and effectively perform the task in such an environment.

The remaining potential ruptures are on the RHR HXs which are assumed for this paper to behave identically. The two coupled failure modes considered are tube rupture and subsequent rupture of the HX shell. A similar loss of isolation at full power has occurred and led to the rupture of an RHR HX [18]. The tube rupture and shell rupture leak paths are *BRK-2* and *BRK-3*, respectively, in Figure 1. Either rupture will cause the HXs to be disabled for the duration of the accident.

In order to establish flow from the containment sump it will be necessary to isolate the ruptured HXs via manual action in the RHR HX room (see Figure 1). A tube rupture will overpressurize the CCW system [19] putting systems (including HPSI and LPSI) that depend on it out of operation until it is isolated by manual operator action from the RHR HX room. An HX shell rupture will result in an additional leak from CCW into the RHR HX room which will render CCW inoperable until isolated from the CCW room.

A recognized operator mitigation action for an ISLOCA is the opening of the PORVs to blow down the RCS inventory into containment [5]. This action is intended to quickly reduce RCS pressure by diverting inventory that would otherwise be blown outside containment. The action can be taken from the control room. The first operator action credited in the State-of-the-Art Reactor Consequence Analyses (SOARCA) ISLOCA, isolation of a LPSI pump, was completed 6 minutes after the initiating event [5]. All operator actions in this work that are not assigned a sampled delay (see Table 1) are assumed to occur, at a minimum, 6 minutes after initiation. In an ISLOCA there is potential for both radioactive contamination and a steamy environment in the auxiliary building beyond safe levels for operators to perform actions outside the control room.

2.3. Accident Modeling

The MELCOR severe accident analysis code [12] is used to represent most aspects of this accident. Its flexibility allows for the inclusion of logic such as the loss and time-delayed recovery of systems. MELCOR is responsible for modeling the thermal-hydraulic and core degradation phenomena in this scenario as well as radionuclide transport to the boundary between containment and the auxiliary building. Outputs of interest from MELCOR include the state of the fuel and the nature of radioactive releases outside of containment both due to the leak pathways from Section 2.2 and any others that may open in the course of the accident.

Once radionuclides enter the auxiliary building, their behavior is treated using RADTRAD [13]. The modeled building comprises three floors (with similar layouts to the one in Figure 1) which the operators may be required to move through to perform corrective actions [20]. RADTRAD is used to calculate the dose rate in each room of interest while operators are present. The dose rate is integrated and sent back to MELCOR as a total expected dose for the action. The action may be considered successful or failed depending on the assigned dose tolerance. This outcome will, in turn, influence the progression of the accident.

3. DYNAMIC MODELING

ADAPT is used to generate a DET by applying branching conditions to an accident progression model. Relative to a traditional event tree, a key feature of a DET is the treatment of the problem using a simulator. The timing of a branching event may affect the future growth of the tree. Rather than being set by expert judgment or a few representative simulation runs, the order of events is determined by the time-dependent accident model and a set of branching rules. To begin the DET, a single instance of the model is executed at an early stable state and the initiating event boundary conditions are applied. After each event, the model is duplicated and modified to the new parameters for each child branch. Each child branch is added to a queue to be run when computational resources are available. ADAPT has the capability to link multiple full-scale simulators to generate a single DET and this capability is leveraged to evaluate the ISLOCA using both MELCOR and RADTRAD [21]. This dynamic case was run under ADAPT at Sandia National Laboratories (SNL). A new capability allowed ADAPT to use SNL's High Performance Computing (HPC) resources which resulted in a twenty-fold increase in the concurrent computation rate [11].

Branching conditions considered in this work are presented in Table 1. Further discussion on the development of the MELCOR-related branching conditions is given in Ref. [14]. A general human error probability of 0.03 is used for actions taken by plant personnel both before the accident and during attempts at mitigation (*Initial Door Status* and *Mitigation Action Success* in Table 1) [1]. For the uncertainty in the timing of operator actions outside of the control room (uncertainty type *Mitigation Action Timing* in Table 1), a distribution was used from Ref. [22]. This distribution was originally used for the manual isolation of AFW which requires a similar level of travel through the auxiliary building and manual work to achieve as the isolation actions considered in this work (see Section 2.2). The distribution was sampled at its 5th, 50th, and 95th percentile values which correspond to the *393 s*, *608 s*, and *1050 s* values seen for actions (e.g., *RHR HX Tube Isolation*) in Table 1.

The action *RWST Isolation from RHR* is taken from the control room and is assumed to be completed quickly if it is successful (see Section 2.2). Distributions of the pressure capacities for the RHR suction line and the HX tube and shell for uncertainty type *Break Location* in Table 1 were taken from Refs. [23] and [24], respectively. Each distribution was sampled at its 5th, 50th, and 95th percentile

Table 1: Summary of ISLOCA Branching Conditions

Uncertainty Type	Branching Condition	Value	Probability
Break Location	RHR Suction Pipe Capacity	4.2 MPa	0.1
		8.9 MPa	0.8
		16 MPa	0.1
	RHR HX Tube Capacity	7.8 MPa	0.1
		11 MPa	0.8
		16 MPa	0.1
	RHR HX Shell Capacity	6.1 MPa	0.1
		9.4 MPa	0.8
		15 MPa	0.1
Initial Door Status	RHR Pump Room Door Status	Closed	0.97
		Open	0.03
	RHR HX Room Door Status	Closed	0.97
		Open	0.03
Mitigation Action Success	PORV Blowdown	Success	0.97
		Failure	0.03
	RHR Pump Suction Isolation	Success	0.97
		Failure	0.03
	RWST Isolation from RHR	Success	0.97
		Failure	0.03
	RHR HX Tube Isolation	Success	0.97
		Failure	0.03
	RHR HX Shell Isolation	Success	0.97
		Failure	0.03
Mitigation Action Timing	RHR Pump Suction Isolation Timing	393 s	0.1
		608 s	0.8
		1050 s	0.1
	RHR HX Tube Isolation Timing	393 s	0.1
		608 s	0.8
		1050 s	0.1
	RHR HX Shell Isolation Timing	393 s	0.1
		608 s	0.8
		1050 s	0.1
Dose Tolerance	Operator Dose Tolerance	5 rem	1/3
		25 rem	1/3
		No limit	1/3
Ending Conditions	10% of Fuel Melted	End	N/A
	24 hour Simulation Time Reached	End	N/A

values with probabilities as shown in Table 1.

After the expected dose for a mitigative action is calculated (see Section 2.3) with RADTRAD, control is passed back to MELCOR. During this passing of control, a *Dose Tolerance* uncertainty (see Table 1) is assigned for the action. This uncertainty recognizes that the plant personnel's understanding of the accident may shift as it progresses and the dose tolerance of crews may vary with that understanding. Five rem *Dose Tolerance* in Table 1 represents the yearly whole body dose equivalent limit for radiation workers while 25 rem represents the limit for emergency workers [25]. The determination of when an emergency is recognized is beyond the scope of this work and so an equal weighting was used for each sample including no dose limit.

While the *Ending Conditions* in Table 1 are not uncertainties, they do each generate a new branch under ADAPT. The new branch does not change any input parameters but serves as an ending point to the sequence.

4. RESULTS

A DET was created using the branching conditions in Section 3. Currently, 1,448,618 branches have been identified (representing 781,763 sequences[4]) and 697,663 branches have finished running. Although the DET experiment has not completed, sufficient progress has been made to infer the influential parameters and the unique features of this coupled-code analysis. To date, 24,849 of the identified sequences have reached a final state such as the 10% fuel damage or 1 day simulation time code stops. Sequences corresponding to 54% of the probability space of the DET have reached a final state.

The RCS pressure and Reactor Pressure Vessel (RPV) water level are shown in Figure 2 and Figure 3, respectively. Sequences with no break in the RHR system are identifiable by their pressure which slowly decreases due to leakage into containment through the RHR relief valves until a scram for low pressure is reached at approximately 1,500 seconds. These sequences quickly recover in pressure due to HPSI and the water level does not approach the top of active fuel (6.7 meters in Figure 3).

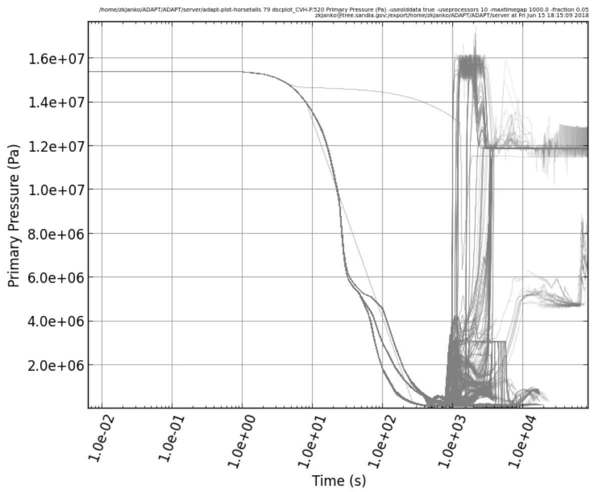

Figure 2: Primary Pressure

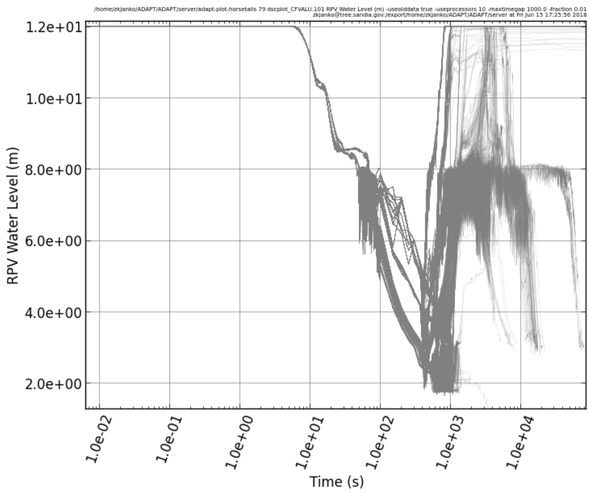

Figure 3: Reactor Pressure Vessel Water Level

Sequences where an RHR break occurs will quickly blow down the RCS regardless of which hypothesized break or breaks occur (see Section 2.2) at a simulation time of approximately 1 second to 300 seconds in Figure 2. This loss of pressure is accompanied by a loss of water level as seen in Figure 3 over the same time period. The PORV blowdown operator action was not modeled to occur until 6 minutes (360 seconds) into the ISLOCA to allow for the time required by operators to diagnose the condition and determine appropriate action (see Section 2.2). It can be seen from Figure 2 that by this time the RCS has already blown down substantially into the auxiliary building. Note that a complete break is assumed in the case of a pipe break and a flow area equal to a complete pipe break

[4] Sequence is used to refer to an ordered set of branches spanning from the initial branch of an event tree to a unique end state.

is assumed in the case of each RHR HX break. Given this type of ISLOCA and the assumptions used, it appears unlikely that PORV blowdown will substantially affect the inventory retained within containment.

When fuel damage is defined as a sequence having a final fuel intact mass fraction less than 1.0 (see Ref. [12]), the conditional core damage probability of the currently-finished sequences is 0.32. Note that this value may change significantly as additional sequences finish. The 22 terabytes (and growing) of simulator outputs give flexibility in interpreting the results of the DET beyond traditional PRA measures but also present challenges in reducing and mining the data. In Section 4.1, Dynamic Importances (DYIs) are used to identify the quantitative relationships between input parameters and consequence measures.

4.1. Dynamic Importance Measures

DYIs are recently-developed analysis tools in ADAPT that allow input parameters to be evaluated for their impact on consequences of interest [26, 27]. Due to the complexity of the sample DET in both timing and order of events, DYIs are used in an attempt to discover relationships and potential risk insights. The measures presented in Table 2 were applied to a selected set of input parameters and consequence measures to demonstrate the range of insights that may be gleaned from a multi-simulator ADAPT case. The selection of consequences was limited to a set that are believed to be indicative of the state of the system. These indicators include the final core intact fraction, the peak containment pressure, the final amount of hydrogen produced, and the final cesium release to the environment[5].

Table 2: DYI Dynamic Importance Measures

Importance Measure	Description
$DYI1 = \frac{R(x=1)}{R(x=0)}$	Consequence ratio of occurrence to non-occurrence
$DYI2(i) = \frac{R(x=1_i)}{R(x=0)}$	Consequence ratio of occurrence value $x = 1_i$ to non-occurrence
$DYI3(i) = \frac{R(x=1_i)}{R(x=1)}$	Consequence ratio of occurrence value $x = 1_i$ to average of occurrence $x = 1$

DYI values for various consequences associated with event occurrence versus non-occurrence are given in Table 3. An example interpretation is that RHR HX tube isolation success sequences have an expected (mean) environmental cesium release fraction 0.32 times that of RHR HX tube isolation failure sequences[6]. The core intact fraction and peak containment pressure were found to be insensitive to the input parameters. Two unexpected insights were identified by the DYIs in Table 3. The first relates to the status of doors in the auxiliary building and the second relates to the PORV blowdown mitigative action.

Interestingly, if the RHR pump room door was closed at the beginning of the scenario, the expected hydrogen generation and Cs release fraction are 3.2 times and 11 times, respectively, their expected values in sequences where the door was left open. Due to the assumed layout of the auxiliary building (see Figure 1), a flooded room will drain faster if it has access to the hallway. By keeping flood water contained in the room, actions in the RHR pump room were delayed (see Section 2.2), which in turn delayed the mitigation of the ISLOCA and resulted in negative consequences.

[5] In this ISLOCA beyond containment, environment is used to refer to the space not enclosed by the containment building or the auxiliary building.
[6] A value of 1.0 indicates no change in the expected value between occurrence and non-occurrence.

Table 3: DYI1 Values for Binary Events

Branching Parameter	Core Intact Fraction	Hydrogen Generation	Peak Containment Pressure	Environmental Cs Release Fraction
RHR HX Room Door Closed	1.0	0.99	1.0	1.2
RHR Pump Room Door Closed	1.0	3.2	1.0	11
RHR HX Tube Isolation	1.0	0.98	1.0	0.32
RHR HX Shell Isolation	1.0	0.69	1.0	0.56
RWST Isolation	1.0	0.097	1.0	0.074
PORV Blowdown	1.0	3.2	1.0	3.7

A modeling choice was made that once the operators open the PORVs they will not close them again during the accident. The motivation for this decision is to prevent repressurization of the RCS in case the PORVs fail. ISLOCA procedures emphasize depressurization to minimize break flow and the SOARCA study found it to be an effective action in reducing off-site releases for a small break ISLOCA [5]. In this study, however, PORV blowdown leads to increases both in hydrogen generation (3.2 times the no-depressurization case) and release of cesium to the environment (3.7 times the no-depressurization case). Due to the unintuitive PORV blowdown results for hydrogen and Cs, Table 4 presents DYIs calculated for PORV blowdown with other consequence measures. It can be seen that the action increases cesium release from the fuel to the RPV (4.0 times the no-depressurization case), to the containment building (3.1 times the no-depressurization case), and to the auxiliary building (3.0 times the no-depressurization case). These results suggest that the action leads to an increase in oxidation and failure of the fuel. The PORV blowdown mitigation action may be inappropriate or may require modification for a large ISLOCA if the size of the ISLOCA can be diagnosed.

Table 4: PORV Blowdown DYI1 Values

Consequence Measure	DYI Value
RPV Cs Release Fraction	4.0
Containment Building Cs Release Fraction	3.1
Auxiliary Building Cs Release Fraction	3.0
Peak Auxiliary Building Pressure	1.1
Final RCS Pressure	0.62

Table 5 shows DYI measures for mitigative actions for several consequence measures. The first two consequences (hydrogen generation and release fraction of Cs to the environment) generally indicate a less desirable condition with higher values. A higher peak containment pressure is not necessarily undesirable as one of the functions of containment is to allow blowdown of the RCS during an accident which will tend to raise the containment pressure. The earliest values of timing lead to significantly better outcomes for hydrogen generation and the Cs release fraction. For example, the earliest HX shell isolation timing (393.0 s) has a Cs release fraction $2.3 * 10^{-19}$ times the expected release fraction of all sequences while the next sampled time (608.0 s) has a release fraction $2.9 * 10^{-3}$ times that of all sequences. The peak containment pressure has a non-monotonic response to tube isolation timing reaching its highest value ($6.4 * 10^{4}$ times that of all sequences) for a timing of 608.0 s. It is possible, if only the HX tube failure occurred (see Section 2.2), that this isolation action closes the only open pathway to the auxiliary building from the RCS past containment. By contrast, the shell failure only occurs in conjunction with a tube failure and does not show such a strong response for an isolation timing of 608.0 s.

Table 5: DYI3 Values for Mitigation Action Timing

Branching Condition	Value	Hydrogen Generation	Environmental Cs Release Fraction	Peak Containment Pressure
RHR HX Tube Isolation Timing	393.0 s	$1.0*10^{-4}$	$1.9*10^{-4}$	$2.1*10^{-3}$
	608.0 s	$1.3*10^{-2}$	$4.9*10^{-2}$	$6.4*10^{4}$
	1050.0 s	0.16	$8.4*10^{-2}$	$3.0*10^{-2}$
	10^{20} s	$4.6*10^{9}$	$1.4*10^{10}$	$2.6*10^{-4}$
RHR HX Shell Isolation Timing	393.0 s	$5.6*10^{-4}$	$2.3*10^{-19}$	$2.2*10^{-3}$
	608.0 s	$6.4*10^{-2}$	$2.9*10^{-3}$	$2.3*10^{-2}$
	1050.0 s	$6.9*10^{-2}$	$5.4*10^{-2}$	$3.2*10^{-2}$
	10^{20} s	$1.2*10^{10}$	$4.4*10^{10}$	$2.4*10^{10}$

5. CONCLUSION

This paper explored uncertainties in the initiation and mitigation of an ISLOCA using a coupled-code DET analysis. An ISLOCA is a complex accident in which the states of the containment building and auxiliary building have a great impact on the success or failure of mitigative actions. The progression of the accident was tracked using MELCOR. Hypothesized operator actions were evaluated against the possibility of excessive radiation dose using the RADTRAD code with source terms generated by MELCOR. The success or failure of actions according to the dose returned by RADTRAD in turn influenced the state of the plant in MELCOR. In future work, the relationship between operator dose tolerance and various figures of merit will be explored to determine the extent to which radiation safety concerns may influence the mitigation of an ISLOCA.

The results of the coupled-code DET were analyzed in an attempt to uncover insights into mitigation of the accident. Hypothesized actions were evaluated for the impact of their success on measures of consequence such as degradation of the fuel or the release of cesium to the environment. The majority of hypothesized actions (e.g., isolation of the RWST from the break to preserve inventory for later injection) had a desirable result on offsite releases. For actions with uncertain timing, the same consequences were compared to discover time dependencies. For most action-consequence pairs, an earlier completion time led to a more desirable outcome, although there was a non-monotonic relationship between RHR HX tube isolation timing and the peak containment pressure.

The status of equipment room doors in the auxiliary building was found to have an interesting relationship with various figures of merit. It appears that the rooms (and drains) as modeled in this case are not sufficient to quickly drain the water from a large ISLOCA. The buildup of water in a critical room may prevent operators from performing time-sensitive mitigative actions. One hypothesized action used in the SOARCA ISLOCA analysis, which differs significantly from this one, is PORV blowdown of the RCS [5]. For this ISLOCA, it appears unlikely that PORV blowdown may be implemented in time to perform its intended function of preferentially blowing RCS inventory into containment during the initial depressurization. Keeping the RCS open to containment in this large ISLOCA appears to lead to less desirable outcomes.

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