FRAMEWORK FOR ASSESSING INTEGRATED SITE RISK OF SMALL MODULAR REACTORS USING DYNAMIC PROBABILISTIC RISK ASSESSMENT SIMULATION

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The events at the Fukushima nuclear power station highlighted the need for consideration of risks from multiple nuclear reactors co-located at a site. Determining site risk will also be important for small modular reactor designs because of the number and proximity of reactor modules on a site. To gain an accurate view of a site's risk profile, the core damage frequency (CDF) for the site, rather than the unit, should be considered. There are many types of events that could create a dependency between multiple reactor units or modules from a risk perspective. In order to effectively account for these risks when looking to create a multi-module dynamic probabilistic risk assessment (MMDPRA), six commonality classifications have been established. These commonality classifications and dynamic PRA will be used to establish a system model which will be applied to two adjacent integral pressurized water reactors (iPWRs) with shared safety and support systems. To accomplish this, the previously developed dynamic simulator, ADS-IDAC, will be upgraded in three areas: 1) the thermal-hydraulic code responsible for modeling the system performance will be upgraded to the most current RELAP5 version, 2) ADS-IDAC will be incorporated with a commercial "executive platform" to allow parallel communication between two or more nuclear reactors, and 3) the hardware reliability model within ADS-IDAC will be improved to include system models which cover the dependency classes and capture dynamic hardware reliability. To date, a new module has been added to the code which will allow the ADS-IDAC operator control panel to interface with simulator-derived information from either RELAP-HD or other balance-of-plant simulation modules. Additionally, a system classification matrix has been developed for an iPWR to delineate support and front-line systems and iPWR attributes belonging to the six commonality classifications. Using this information, a RELAP-HD nodalization has been developed for an iPWR. The research described in this paper expands the system state-space to more than one reactor unit and evaluates initiating events and accident sequences composed of hardware failures and human errors on one or more reactor units. The development and demonstration of a novel methodology proposed here provides a framework for more realistic PRA analyses and assessment of the relative contribution of important core damage end states.

I. INTRODUCTION

Currently, multi-unit or multi-module site risk is not being formally nor adequately considered in either the regulatory or the commercial nuclear environment[1–3] despite the fact that the question of multiple reactor accidents is not one of possibility, but of probability. These types of multi-unit incidents do occur at operating nuclear power plants as evident through reporting to the U.S. Nuclear Regulatory Commission (NRC)[4] and need to be addressed[2]. Fleming, the International Atomic Energy Agency (IAEA), and Jung, et al. have recommended ideas to deal with different facets of a multi-unit probabilistic risk assessment (PRA)[2,5,6]; however, there are still no well-established, accepted methods for considering multi-unit or multi-module site dependencies when creating a PRA[7]. For a history of multi-unit nuclear power plant regulation in the U.S., the reader is directed to Schroer & Modarres (2013).

To gain an accurate view of a site's risk profile, the core damage frequency (CDF) for the site rather than the unit or module should be considered. There are many types of events that could create a dependency between multiple reactor units or modules from a risk perspective. In order to effectively account for these risks when looking to create a multi-module PRA (MMPRA), six main commonality classifications have been established: initiating events, shared connections, identical components, proximity dependencies, human dependencies, and organizational dependencies[4]. These commonality classifications and dynamic probabilistic risk assessment (DPRA) will be used to establish a framework for simulating the stochastic interaction of more than one nuclear reactor system which shares front-line and support systems. The research outlined in this paper is divided into three tasks: 1) multi-module risk methodology development, 2) DPRA simulation tool development, and 3) methodology and tool capability demonstration using a small modular reactor (SMR) example.

Research discussed here combines facets of multiple disciplines including computer science, nuclear engineering, and reliability engineering. Therefore, it is important to limit the scope of the project to achieve manageable goals and leverage existing expertise where available. To achieve the objectives listed previously, it is mandatory to impose some assumptions and limitations on the research and development proposed here as follows:

1) The proof-of-concept application example will be a multi-unit risk analysis of SMRs, more specifically the integral pressurized water reactor (iPWR) reactor design consisting of two adjacent reactor modules, front-line systems, and support systems.
2) System reliability modeling will be limited to the binary description of system success or failure based on system success criteria as defined in the ASME/ANS Level 1 PRA Standard[8]. Therefore, degraded system performance will not be considered.
3) Nuclear power plant risk analysis will be limited to Level 1 PRA and calculation of CDF as applicable. As such, no severe accident phenomena will be considered. A discussion of CDF definitions used among PRA analysts will be discussed; however, nuclear fuel temperature (greater than 2200 °F) calculated in RELAP5 will be used as a surrogate for core damage.
4) Analysis on proximity, human, and organizational dependencies will be predicated on the level of information available about plant operations and maintenance.
5) At least one initiating event will be chosen which has the possibility of impacting multiple reactors. Initiating events considered could include station blackout (SBO) and seismic events.

II. MULTI-MODULE RISK ANALYSIS METHODOLOGY

This paper outlines a methodology to assess the relative risk posed by multi-module reactors, in particular SMR designs. The methodology should encompass the following steps:

1) Define taxonomy of connections within and between units in a nuclear plant site that affect performance and functionality of critical SSCs. For example, a proof-of-concept SSC in an iPWR would be a shared ultimate heat sink structure that is susceptible to seismic loads and is important to safety for multiple reactor modules. The taxonomy will be based on the previous University of Maryland (UMD) research using six main commonality classifications previously discussed [4].
2) Develop a dependency matrix for the reactor system, including support systems. This matrix can then be used to bin systems into one or more the commonality classifications stated above in step 1. The systems binned in each commonality classification would represent important candidate sequences to investigate using the DPRA simulator.
3) Rank base PRA accident sequences in order to develop a focused MMPRA.
4) Use traditional importance measures in the base PRA such as risk achievement worth (RAW) or Fussell-Vesely (FV) to determine the components and systems which may be risk significant and compare to a list of multi-module dependencies.
5) Establish a thermal-hydraulic (T-H) model of the nuclear reactor system, which is most likely already available as part of a safety system evaluation.
6) Expand fault trees to include common cause failures and system dependencies across adjacent units or modules.
7) Build ADS-IDAC simulator model including input files for crew, hardware reliability, indicators, scheduler, and system.
8) Prune accident sequences based on probability truncation, event time or end state conditions, such as achieving average fuel temperature equal to 2200 °F. An algorithm for improved control of accident sequence branches beyond simple probability truncation is being investigated.

The following sections will discuss in more detail the major aspects of the steps listed above including using the base PRA, developing the system dependency matrix and classification, and applying dynamic PRA to the multi-module reactor case, herein referred to as multi-module dynamic PRA (MMDPRA).

II.A. Base Probabilistic Risk Assessment

Instead of looking at the frequency of core damage per module, per year irrespective of the operating states of other modules, it is more useful to consider a site CDF. The site CDF or integrated site risk can be calculated as the frequency of exactly one core damage event occurring per site per year. That is, the probability of one module having a core damage event during a year, while the other modules do not. Another option is to calculate the frequency of one or more core damages occurring nearly simultaneously per site per year. That is, the frequency of one module experiencing a core damage event during a year, while another module is also experiencing a core damage event. All combinations of these cases lead to the definition of site CDF, which is at least one core damage per site per year. The CDF for exactly one core damage event occurring per site year would be, for example, the frequency of module 1 only experiencing a core damage event or only module 2 experiencing a core damage event. This would be either the dark or light circle in Fig. 1, respectively. Whereas, the frequency of multiple core damage events would be the intersection in Fig. 1. The sum of the areas would represent at least one core damage event [4].

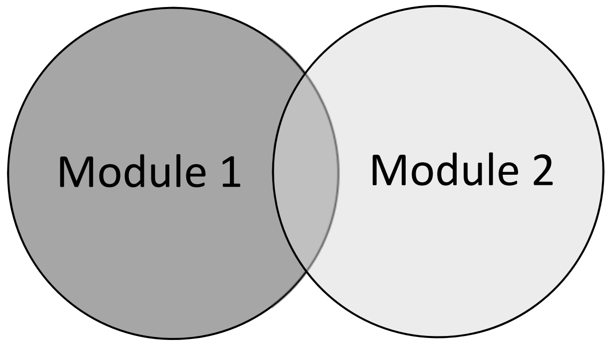

Fig. 1. Venn diagram of core damage frequency.

II.B. Classification Scheme and Dependency Matrix

Previous researchers and international experts have recommended conducting a feasibility study for MMPRA[4,9]. In order to accomplish this and apply the methodology developed in this research, the dependencies of all front-line systems are defined in a dependency matrix. This approach is already typically performed for single-module PRAs and includes only hard physical connections, such as a motor-operated valve needing to have power from a predefined source. These matrices allow the PRA model developer to know what to consider when creating the system fault tree.

Using the base PRA, the initiating events, shared connections, and identical components would be developed. Further research may likely allow similar development of proximity, human, and organizational dependencies. For proximity dependencies, room location, nearby doors and conduits, and the location with respect to other modules onsite would need to be noted. For human dependencies, the operator or the maintenance team member responsible for the action, as well as whether the action occurs before or after the event would need to be noted. For organizational dependencies, the department responsible for the system, the procedures used when operating the component, and the training that is given on the SSCs would need to be noted[4].

Support systems considered provide needed functions such as electrical power and cooling to front-line systems which may be safety or non-safety related. In either instance, it is important to recognize that either type of system (safety or non-safety) may be credited in the PRA. System sub-components and assumed support systems are assigned for each system which may be credited in the PRA to mitigate an accident.

Given the known mitigating systems and possible support systems listed above, these can now be provisionally assigned to the classification scheme outlined in step 1 at the beginning of the multi-module risk analysis methodology. **Error! Reference source not found.** shows what iPWR systems or events may belong to each of the six classifications.

TABLE I. iPWR system classifications.

Accident Sequence Classifications	Definition	Classification Members
Initiating Events	Single events that have the capacity to affect multiple units	Loss of offsite power, loss of ultimate heat sink, seismic event (including seismically-induced tsunami), external fire, external flood, hurricane, high wind, extreme temperature
Shared Connections	Links that physically connect SSCs of multiple units	Reactor pool, chilled water system, BOP water system, spent fuel pool cooling system, circulating water system, reactor component cooling water system, high, medium and low voltage AC distribution systems
Identical Components	Components with same design, operations, or operating environment	Safety DC electrical and essential AC distribution system, reactor module bay, containment, decay heat removal system, emergency core cooling system, non-safety instrumentation and control, chemical volume and control system, power conversion system
Proximity Dependencies	A common environment has the potential to affect multiple units	Reactors, ultimate heat sink, containment, non-safety DC electrical and essential AC distribution system, control room HVAC
Human Dependencies	A person's interaction with a machine affects multiple units	Shared control room, operator staffing more than one reactor
Organizational Dependencies	Connection through multiple units typically by a logic error that permeates the organization	Same vendor for safety and non-safety system valves, consolidated utility ownership of multiple nuclear power plant sites, decision-maker overseeing more than one reactor or more than one operator

II.C. Dynamic Probabilistic Risk Assessment

Investigating the risk of more than one reactor on a site is not a novel concept and has been done previously using static PRA methods in the 1980's for the Seabrook Station[2]. Practically speaking, performing this type of PRA is an onerous process and still does not explicitly consider the timing of events as they occur in the accident sequence, especially when operator actions or conditional failures are included. Therefore, applying automated dynamic PRA methods to select scenarios with significant dynamic and coupled characteristics at the Level 1 PRA stage is a natural stepping stone to quantifying the integrated site risk. A dynamic PRA can be described as one that "couples the stochastic and phenomenological models of the plant to account for possible dependencies between events in which the need for and timing of branching is determined by conditions of the analysis rather than predetermined"[10].

In order to truly address a MMPRA, one first must be able to understand all of the avenues in which units could be coupled. The multi-module methodology proposed here defines a unit as a reactor core and its front-line systems and SSCs. That is, a unit at a traditional nuclear power plant would be everything inside of the primary containment and power generation buildings and supporting systems, and for SMRs, the "unit" terminology would be considered one module [4].

The first step in conducting a MMPRA using dynamic simulations involves analyzing the static PRA developed for a single module. Using the traditional static PRA as the base case for investigating accident scenarios which involve SSCs shared by more than module or could potentially impact the operation of another module limits the state space and reduces the computational coverage needed to analyze all possible accident scenarios. This type of base PRA is already developed as part of the Level 1 internal and external events PRAs used to support regulatory licensing actions such as design certification, risk-informed technical specifications or risk-informed in-service inspection[11]. The information gathered from the base PRA helps develop the dependency matrix system classification as was previously shown in **Error! Reference source not found.**.

Over the last 20 years, several dynamic PRA methods have been developed, mostly through academia. These methods fall into two categories: 1) discrete dynamic event trees (DDETs) and 2) continuous dynamic event trees (CDETs). The first method employs event trees that branch at fixed points, partially determined by the modeler. The second method employs event trees that branch at any point in time via Monte Carlo analysis and converge on average system behavior through iterative simulation [10,12]. Use of CDETs is limited because they are significantly more computationally intensive than DDETs, and algorithms are only applicable to the specific system being considered [13]. On the other hand, DDETs allow for a comprehensive and systematic coverage of possible accident sequence while relying on a time-dependent model, such as RELAP5 or MELCOR, to guide scenario generation[13]. Other DDET tools have been developed such as ADAPT and RAVEN[10,12,14], but to date none have attempted to perform a DPRA of more than one nuclear reactor. Similarly, UMD has developed the dynamic PRA tool ADS-IDAC which has emphasized modeling human performance in nuclear power plant operation, again, only in the domain of one nuclear reactor[15–17]. ADS-IDAC includes a T-H model based on RELAP5, the operator cognitive model IDAC, and a control panel which interfaces between the plant model (RELAP) and operator model (IDAC). Therefore, part of the methodology development in this research is to expand the capabilities of the ADS-IDAC computer code to analyze multiple reactors in a dynamic PRA framework.

III.DYNAMIC HARDWARE RELIABILITY

III.A. Current ADS-IDAC Hardware Reliability Model

The current ADS-IDAC hardware reliability module models time dependent failures and conditional demand failures (e.g., failure-to-start). Time dependent failures occur at a prescribed time during the simulation evolution and can be used to reflect hardware failures (e.g., failure of a pump or valve at time *t*) or accident initiating events (e.g., main steam line break at time *t*). Conditional failures occur when a component changes operating state to a pre-selected target value, thereby initiating the conditional failure of another system or component. The dynamic event tree branching is structured such that time dependent failures only generate a failure sequence branch; however, conditional failures generate both a success and failure branch.

These hardware failures must be specified in the "ControlPanel.txt" input file and the "SystemReliability.txt" input file which specifies the parameters of the component failure and recovery distributions. If the operator attempts to restart failed equipment, branches representing component recovery and permanent failure are generated. Thus each hardware failure event can result in three outcomes[15]:

1) the equipment does not fail,
2) the equipment initially fails but is later recovered, and
3) the equipment fails and is unrecoverable.

To capture the stochastic nature of failure-on-demand, the ADS-IDAC system reliability module uses the beta distribution to model both the probability of hardware failure and recovery by a Monte Carlo sample of the failure distribution. Therefore, the current hardware reliability model is limited in its ability to capture environmental factors which may accelerate the hardware failure probability. A contribution of the research is to establish a method of linking simulated environmental factors, such as pressure and temperature, with a probabilistic hardware failure model to update the time-dependent failure probability of a given component as the accident sequence evolves.

III.B. Accelerated Hardware Failure Methodology

iPWRs, much like currently operating pressurized water reactor (PWR) nuclear power plants, have pressure control equipment in and above the pressurizer tank to maintain the primary system at its normal operating pressure. The equipment includes heaters in the tank to increase pressure, water sprays in the steam volume to decrease pressure, power-operated relief valves (PORVs) above the tank that permit flow from the tank to decrease pressure, and safety valves. Like the PORVs, pressurizer safety valves are designed to permit flow from the pressurizer upper dome to mitigate pressure surges and maintain pressure below a maximum allowable system pressure. Pressurizer safety valves represent the final line of defense in protecting the primary system against overpressure[18]. Because they also represent a relatively simple mechanical hardware component, they were chosen as the representative hardware component to develop a time-dependent accelerated failure probability. The methodology presented in this section could be applied to other hardware components modeled in the MMDPRA.

III.B.1. Safety Vent Valve Description

The reactor primary system is designed to have a subcooled water state except for the steam volume at the top of the pressurizer. In case a plant transient causes the system pressure to exceed the design pressure, the safety valves are designed to automatically open to release the fluid from the pressurizer while keeping the system pressure below an allowable design pressure, then to reclose when the pressure drops to a specified value. Although one valve could be designed to pass all required flow rate, the ASME code requires that systems be protected by at least two safety or relief valves. In some plants, three or four safety valves are required to control the system pressure within allowable limits. The safety valves may all be set at the design pressure or their set points may be staggered, with the first valve opening the other opening 15 to 30 psia higher [18]. In addition to the opening and peak pressure requirements, the ASME code has specific operating rules for safety valves. First, spring-loaded safety valves and pilot-operated safety valves must open and close automatically in response to the fluid pressure at the valve inlet and outlet. No external power supply can be used to open or close the valve[18]. This simplifies the derivation of the accelerated failure probability equation by eliminating some typical failure modes that might otherwise affect a safety related pressure relief valve, such as supply of electrical power or solenoid reliability. An example of the safety vent valve (SVV) considered here is show below in Fig. 2.

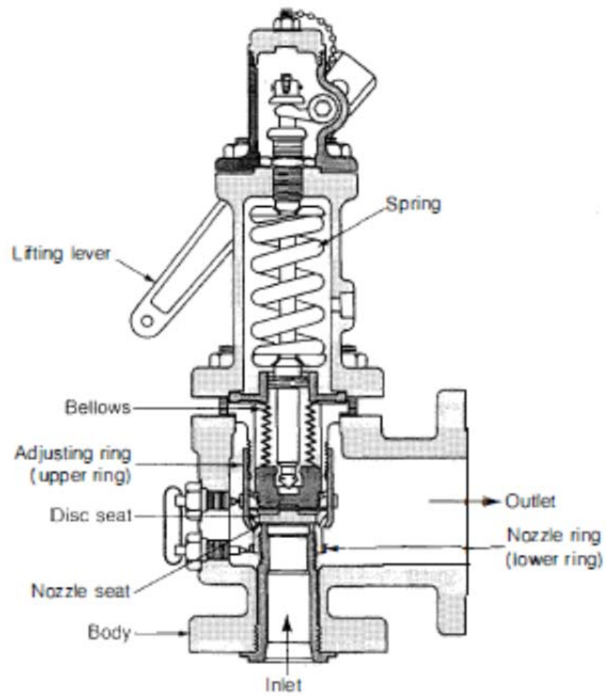

Fig. 2. Typical Crosby pressurizer safety vent valve model HB-BP-86[18].

III.B.2. Safety Vent Valve Failure Data and Equation

First, the specific type of safety relief valve that may be used in iPWRs needs to be determined. Two options are represented in the literature as a safety relief valve (SRV) which is encoded specifically to BWRs, or a safety vent valve (SVV) which is encoded specifically for PWRs[19]. The pressurizer SVVs are totally enclosed pop-open-type valves (similar to the main steam SVV). The valves are spring-loaded, self-actuating, and have backpressure compensation designed to prevent the reactor coolant system pressure from exceeding the design pressure by more than 10 percent[20]. Using data provided from the United States Nuclear Regulatory Commission (US NRC) in a 2010 update to NUREG/CR-6928, *Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants*, the base valve failure rate is shown below in

TABLE II[21]. The SVVs selected from the various valves available for PWRs are specifically used in the primary reactor coolant system (RCS) and therefore have a different failure distribution from those high pressure relief valves located on the main steam system (MSS).

TABLE II. Base Failure Probability for Reactor Coolant System (RCS) High Pressure Relief Valve Failure Modes

Failure Mode	**Mean Failure Probability**	**Distribution Type**	**α**	**β**
Failure to Open (FTO)	7.32E-04	Beta	1.50	2.05E+03
Failure to Close (FTC)	7.32E-04	Beta	1.50	2.05E+03
Stick Open (SO)	6.96E-08	Gamma	9.50	1.37E+08

The total valve assembly failure rate, λ_{VA}, in failures/million operations for poppet-type safety valve is shown in Eq. (1) and represents the summation of the individual failure modes of valve components. Further discussion on each failure mode is provided in subsequent sections.

$$\lambda_{VA} = \lambda_{PO} + \lambda_{SE} + \lambda_{SP} + \lambda_{SO} + \lambda_{HO} \tag{1}$$

where
λ_{PO} = failure rate of poppet assembly in failures per million operations
λ_{SE} = failure rate of seals in failure per million operations
λ_{SP} = failure rate of spring in failure per million operations
λ_{SO} = failure rate of solenoid in failures per million operations
λ_{HO} = failure rate of valve housing

III.B.3. Poppet Assembly Failure Rate, λ_{PO}

As discussed above, the failure rate for a poppet style valve is given as the summation of the contribution from various failure modes. A poppet valve is one in which the valve element travels perpendicular to the seating surface and is typically used for safety related pressure control. The failure rate for a poppet valve assembly can be further subdivided into the environmental factors which may impact the base failure rate. Thus, the base failure rate, $\lambda_{PO,B}$, can be multiplied by the various influencing factors. This modified failure rate, λ_{PO}, is given below in Eq. (2)[22].

$$\begin{aligned}\lambda_{PO} &= failure\ rate\ of\ poppet\ assembly\ in\ failures\ per\ million\ operations \\ &= \lambda_{PO,B}\ x\ C_P\ xC_Q\ x\ C_F\ x\ C_v\ x\ C_N\ x\ C_S\ x\ C_{DT}\ x\ C_{SW}\ x\ C_W\end{aligned} \tag{2}$$

where
$\lambda_{PO,B}$ = Base failure rate of the poppet valve assembly
C_P = Multiplying factor which considers the effect of **fluid pressure** on the base failure rate
C_Q = Multiplying factor which considers the effect of **allowable leakage** on the base failure rate
C_F = Multiplying factor which considers the effect of **surface finish** on base failure rate
C_v = Multiplying factor which considers the effect of **fluid viscosity and temperature** on the base failure rate
C_N = Multiplying factor which considers the effect of **contaminants** on the base failure rate
C_S = Multiplying factor which considers the effect of the **seat stress** on the base failure rate
C_{DT} = Multiplying factor which considers the effect of the **seat diameter** on the base failure rate
C_{SW} = Multiplying factor which considers the effect of the **seat land width** on the base failure rate
C_W = Multiplying factor which considers the effect of **flow rate** on the base failure rate

Not all contributors affecting the base failure rate are time-dependent. Multiplying factors that will affect the cycling SVVs used in an integral pressurized water reactor (iPWR) are those factors that vary with time. These include the listed factors in TABLE III. As seen from the table, only three multiplying factors are considered to be time-dependent, and therefore the remaining six constant multiplying factors can be subsumed into the base failure rate.

TABLE III. Poppet valve base failure rate multiplying factor categorized as time-dependent or constant.

Time-dependent multiplying factor		Constant multiplying factor	
C_P	Fluid pressure across valve, psi	C_Q	Allowable leakage, in^3/min
C_v	Ratio of initial viscosity to viscosity at given temperature for water	C_F	Surface finish, microinches
C_W	Ratio of actual flow rate to manufacturers rating	C_N	Contaminant flow, filter size, type and number of particles
		C_S	Poppet seat stress, lbs/in^2
		C_{DT}	Seat diameter, inches
		C_{SW}	Land width, inches

III.B.4. Seal Failure Rate, λ_{SE}

The base failure rate, $\lambda_{SE,B}$, for the valve seals can be multiplied by the various influencing factors. This modified failure rate, λ_{SE}, is given below in Eq. (3)[22].

$$\lambda_{SE} = failure\ rate\ of\ seals\ in\ failures\ per\ million\ operations \\ = \lambda_{SE,B}\ x\ C_P\ xC_Q\ x\ C_{DL}\ x\ C_H\ x\ C_F\ x\ C_v\ x\ C_T\ x\ C_N \tag{3}$$

where

$\lambda_{SE,B}$ = Base failure rate of the seals
C_P = Multiplying factor which considers the effect of **fluid pressure** on the base failure rate
C_Q = Multiplying factor which considers the effect of **allowable leakage** on the base failure rate
C_{DL} = Multiplying factor which considers the effect of **seal size** on base failure rate
C_H = Multiplying factor which considers the effect of **contact stress and seal hardness** on the base failure rate
C_F = Multiplying factor which considers the effect of the **seat smoothness** on the base failure rate
C_v = Multiplying factor which considers the effect of **fluid viscosity** on the base failure rate
C_T = Multiplying factor which considers the effect of the **temperature** on the base failure rate
C_N = Multiplying factor which considers the effect of **containmants** on the base failure rate

Not all contributors affecting the base failure rate are time-dependent. Multiplying factors that will affect the cycling SVVs used in an integral pressurized water reactor (iPWR) are those factors that vary with time. These include the listed factors in TABLE IV. As seen from the table, only three multiplying factors are considered to be time-dependent, and therefore the remaining five constant multiplying factors can be subsumed into the base failure rate.

TABLE IV. Poppet valve seal base failure rate multiplying factor categorized as time-dependent or constant.

Time-dependent multiplying factor		Constant multiplying factor	
C_P	Fluid pressure across valve, psi	C_Q	Allowable leakage, in^3/min
C_v	Ratio of initial viscosity to viscosity at given temperature for water	C_F	Surface finish, microinches
C_T	Delta between rated seal temperature and operating temperature	C_N	Contaminant flow (microns/hour), filter size, type and number of particles
		C_H	Ratio of Meyer hardness (lbs/in^2) and contact pressure (lbs/in^2)
		C_{DL}	Seal diameter, inches

III.B.5. Spring Failure Rate, λ_{SP}

The base failure rate, $\lambda_{SP,B}$, for the valve spring can be multiplied by the various influencing factors. This modified failure rate, λ_{SP}, is given below in Eq. (4**Error! Reference source not found.**[22].

$$\lambda_{SP} = failure\ rate\ of\ spring\ in\ failures\ per\ million\ operations \\ = \lambda_{SP,B}\ x\ C_G\ xC_{DW}\ x\ C_{DC}\ x\ C_N\ x\ C_Y\ x\ C_L\ x\ C_K\ x\ C_{CS}\ x\ C_R\ x\ C_M \tag{4}$$

where

$\lambda_{PO,B}$ = Base failure rate of the poppet valve assembly
C_G = Multiplying factor which considers the effect of **material rigidity modulous** on the base failure rate
C_{DW} = Multiplying factor which considers the effect of **wire diameter** on the base failure rate
C_{DC} = Multiplying factor which considers the effect of **coil diameter** on base failure rate
C_N = Multiplying factor which considers the effect of **number of active coils** on the base failure rate
C_Y = Multiplying factor which considers the effect of **material tensile strength** on the base failure rate
C_L = Multiplying factor which considers the effect of the **spring deflection** on the base failure rate
C_K = Multiplying factor which considers the effect of the **spring concentration** on the base failure rate
C_{CS} = Multiplying factor which considers the effect of the **spring cycle rate** on the base failure rate
C_R = Multiplying factor which considers the effect of **a corrosive environment** on the base failure rate
C_M = Multiplying factor which considers the effect of **manufacturting proces** on the base failure rate

Not all contributors affecting the base failure rate are time-dependent. Multiplying factors that will affect the cycling SVVs used in an integral pressurized water reactor (iPWR) are those factors that vary with time. These include the listed factors in TABLE IV. As seen from the table, only one multiplying factor is considered to be time-dependent, and therefore the remaining nine constant multiplying factors can be subsumed into the base failure rate.

TABLE V. Poppet valve spring base failure rate multiplying factor categorized as time-dependent or constant.

Time-dependent multiplying factor		**Constant multiplying factor**	
C_{CS}	Cycles/minute	C_G	Modulus of rigidity, lbs/in^{2}x10^6
		C_{DW}	Wire diameter, inches
		C_{DC}	Coil diameter, inches
		C_N	Number of active coils
		C_Y	Tensile strength, lbs/in^{2}x10^3
		C_L	Delta spring deflection, inches
		C_K	Ratio of coil and wire diameter, inches
		C_R	1.0 using corrosion control
		C_M	1.0 when quality control applied

III.B.6. Solenoid and Valve Housing Failure Rate, λ_{SO} and λ_{HO}

Solenoid failure is not applicable to Crosby-style PWR safety valves because these are manual action valves and therefore the solenoid failure mode does not contribute to modifying the base failure rate. Additionally, the probability of a cracked housing is minimal and extremely unlikely given the valve is more likely to fail-to-close and continue to relieve pressure than experience a catastrophic failure of the valve housing. Therefore, the solenoid and valve housing failure mode are assumed to not contribute to the overall valve failure rate. Eq. (1) then can be reduced to only the accelerated degradation of the poppet assembly, the valve seal, and the valve spring as shown below in Eq. (5).

$$\lambda_{VA} = \lambda_{PO} + \lambda_{SE} + \lambda_{SP} \tag{5}$$

III.B.7.

The failure rate for the safety vent valve, λ_{VA}, is given as the contribution from the poppet, the seal and the spring. Reducing Equations (2), (3), and (4) to only the time-dependent factors accelerating the failure rate and substituting into Eq. (5) yields Eq. (6).

$$\lambda_{VA} = \left(\lambda_{PO,B}\ x\ C_{P,PO}\ x\ C_{v,PO}\ x\ C_{W,PO}\right) + \left(\lambda_{SE,B}\ x\ C_{P,SE}\ x\ C_{v,SE}\ x\ C_{T,SE}\right) + \left(\lambda_{SP,B}\ x\ C_{CS,SP}\right) \tag{6}$$

In order to apply the SVV failure probability published in NUREG/CR-6928[19], the base failure rate for each failure mode was normalized to determine the fractional contribution of each failure mode to the overall base failure rate. This normalization for the poppet, seal, and spring failure represent the fractional contribution for each of the multiplying factors modifying the base failure rate and is herein referred to as an α-factor. The α-factor was determined for each failure mode (poppet, seal and spring) by dividing each failure mode base failure rate by the total, as shown in Eq. (7), where $i = \lambda_{PO,B}$, $\lambda_{SE,B}$, $\lambda_{SP,B}$. Applying the α-factor to Eq. (6) gives Eq. (8) which represents the time-dependent environmental accelerating factors and their contribution to the overall valve base failure probability, $\lambda_{VA,B}$. The base failure rate can be found for a generic poppet assembly, a gasket or static seal, and a helical compression spring and is shown below in
TABLE VI along with the α-factor for each failure mode[22].

$$\alpha_i = \frac{\lambda_{i,B}}{\lambda_{PO,B} + \lambda_{SE,B} + \lambda_{SP,B}} \tag{7}$$

TABLE VI. Base failure rate and α-factor for poppet, seal and spring failure modes.

	Base Failure Rate[22] ($\lambda_{i,B}$) (failures/hour)	$\boldsymbol{\alpha_i}$
Poppet (PO)	1.40E-06	0.051
Seal (SE)	2.40E-06	0.087
Spring (SP)	2.38E-05	0.862

Total All Modes	2.76E-05	

$$\lambda_{VA} = \lambda_{VA,B}\left[\left(\alpha_{PO}\ x\ C_{P,PO}\ x\ C_{v,PO}\ x\ C_{W,PO}\right) + \left(\alpha_{SE}\ x\ C_{P,SE}\ x\ C_{v,SE}\ x\ C_{T,SE}\right) + \left(\alpha_{SP}\ x\ C_{CS,SP}\right)\right] \tag{8}$$

where the multiplying factors are defined below in Eqs. (9)-(15) from Ref. 22:

$$C_{P,PO} = \left(\frac{P_1 - P_2}{3000}\right)^2 \tag{9}$$

$$C_{v,PO} = \frac{2x10^{-8}\ \frac{lbf - min}{in^2}}{\nu} \tag{10}$$

$$C_{W,PO} = 1 + \left(\frac{Flow\ Rate\ \left(\frac{kg}{s}\right)}{Manufacturer\ Flow\ Rate\left(\frac{kg}{s}\right)}\right)^2 \tag{11}$$

$$C_{P,SE} = \begin{cases} P_1 - P_2 \leq 1500\ psi, & 0.25 \\ P_1 - P_2 > 1500\ psi, & \left(\frac{P_1 - P_2}{3000}\right)^2 \end{cases} \tag{12}$$

$$C_{v,SE} = \frac{2x10^{-8}\ \frac{lbf - min}{in^2}}{\nu} \tag{13}$$

$$C_{T,SE} = \begin{cases} T_R - T_o \leq 40°\text{F}, & 1/2^{\frac{T_R - T_o}{18}} \\ T_R - T_o > 40°\text{F}, & 0.21 \end{cases} \tag{14}$$

$$C_{CS,SP} = \begin{cases} CR \leq 30\frac{cycles}{min}, & 0.100 \\ 30\frac{cycles}{min} < CR \leq 300\frac{cycles}{min}, & \frac{CR}{300} \\ CR > 300\frac{cycles}{min}, & \left(\frac{CR}{300}\right)^3 \end{cases} \tag{15}$$

Substituting the α-factor values into Eq. (8) and modifying the variable λ to more accurately reflect the binomial nature of safety valve failure-on-demand, the SVV failure probability becomes a function of the time-dependent pressure, temperature, viscosity and valve cycles show in Eq. (16).

$$p_{VA} = p_{VA,B}\left[\left(0.051\ x\ C_{P,PO}\ x\ C_{v,PO}\ x\ C_{W,PO}\right) + \left(0.087\ x\ C_{P,SE}\ x\ C_{v,SE}\ x\ C_{T,SE}\right) + \left(0.862\ x\ C_{CS,SP}\right)\right] \tag{16}$$

IV. MULTIPLE REACTOR SIMULATION TOOL DEVELOPMENT

Through a mutual research and development project between UMD and GSE Systems, Inc. (GSE) ADS-IDAC is being integrated with the GSE nuclear power plant simulators. The GSE simulators use certified best estimate engineering modeling tools (also referred to as codes) such as RELAP5-3D, Simulate3 and MAAP along with the GSE-developed balance-of-plant (BOP) code JTopMeret and thermal-hydraulic (T-H) code RETACT to realistically mimic an operational nuclear power plant. GSE's experience with synchronizing multiple codes and their distributed computational architecture provides a capability to introduce DPRA into a framework which can simulate more than one reactor.

IV.A. Thermal-Hydraulic Model

GSE obtained RELAP5-3D from Idaho National Laboratory (INL) with the objective of using RELAP5-3D for real time nuclear power plant simulations. The resulting implementation of RELAP, under GSE's SimExec platform, is RELAP5-HD where the HD stands for high definition. It is considered high definition because users can view and modify the RELAP storage array parameters, referred to as a fast array (FA), during live runs, resulting in an implementation of RELAP5-3D that is much easier to debug, tune and modify compared to the standard batch method of problem submissions. By being placed under the SimExec platform, and using the HD 'client-server' relationship, RELAP5-HD is synchronized with all other codes mentioned previously including ADS-IDAC[23].

As currently developed, ADS-IDAC interacts with RELAP5 through control functions and interactive variables. The "RELAP5_channels.txt" input file establishes the communication channels necessary to pass information between RELAP thermal hydraulic simulation and the ADS-IDAC control panel. Seven types of communication channels can be used: hydraulic volume, hydraulic junction, variable trip, logical trip, heat structure, control variable, and interactive control. Each of these channels permits only one-way communication, except for the interactive control. In general, the channels provide a communication from the RELAP thermal hydraulic model to the ADS-IDAC control panel. Only the interactive control channel provides a communication link from the control panel to the thermal hydraulic model[15]. Using RELAP5 in batch mode as is done with the current version of ADS-IDAC, the user relies on RELAP5 restart files to stop and start a scenario at particular point in time. GSE's RELAP5-HD however, takes a complete memory dump of all the internal variables. When a snapshot of initial conditions is taken, all the database variables are snapped into memory, and when an initial condition is reset, all of those variables are restored to their values of the snapped initial condition. This eliminates the RELAP5 initialization and round off errors, therefore, making RELAP5-HD exactly repeatable[23].

The "RELAP5 channels" pass information such as temperature and pressure to ADS-IDAC defined variables. This communication is established through a FORTRAN file called "R5PAR" which passes information from RELAP5 to C++ in ADS-IDAC. Instead of using RELAP5/MOD3.2 as currently implement in ADS-IDAC, GSE's SimExec controlling RELAP5-HD provides the required information.

In order to conduct a multi-module simulation-based dynamic PRA, a T-H model must exist for each module, and those models must be able to communicate between one another. Fig. 3 shows the structure being proposed and investigated to couple multiple T-H models together to support ADS-IDAC. The Simulator Host Executive System addresses this issue and develops a database of RELAP5 memory whereby variables that interface between two RELAP5 HD models must pass through the client side of the simulator[23]. This architecture is beneficial because it allows the use of parallel processing using the ADSClient to spawn new simulations on separate processors while still allowing information sharing and communication of pertinent state variables and controls between the T-H model and ADS-IDAC.

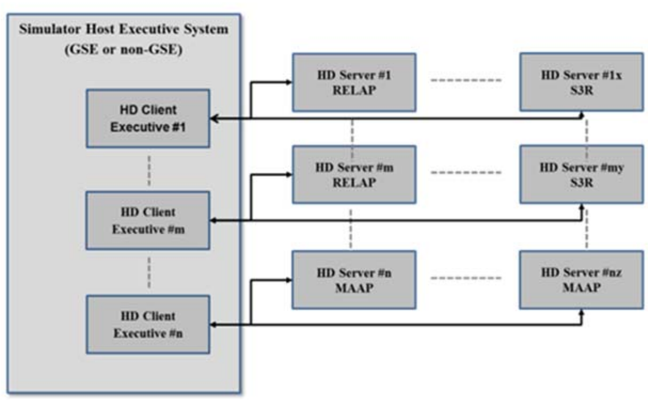

Fig. 3. Multiple client-server architecture[23].

IV.B. Multi-Module Event Tree Analysis

Two options are being explored to generate and control an event tree which is expected to capture the success and failure of two reactors. The first option uses ADS-IDAC as the scheduling mechanism to control generation of a new pair of server simulations for each pivotal event, effectively making a large event tree capturing the event progression of both modules, referred to as a conditional event tree. A second option would be to focus on only one reactor and develop a marginal event tree wherein the influence of an adjacent reactor is considered through symmetry. Both of these options are subjects of our near-term research. The framework, while applied in the proof-of-concept base case to a two module SMR, would be applicable to modeling m-out-of-n SMRs wherein a MMDPRA analysis for hypothetical reactor designs could account for many modules at a site. Example applications of such designs include those that have up to 12 modules within a common reactor complex. Future research would be needed to address the parallel processing capabilities required to simulate the interaction of more than two co-located SMR modules.

V. CONCLUSIONS

New nuclear power plants, especially SMRs, may be required in future reactor licensing and technical assessment to consider MMPRA. With this in mind, tools must be developed at this early stage that can assist in advancing the state-of-practice for current PRA to address questions about integrated site risk from multiple reactors. Therefore, a simulation-based technique is required to manage the proliferation of system information of these multi-unit sites. A new module has been added to the code which will allow the ADS-IDAC operator control panel to interface with simulator-derived information from either RELAP-HD or other balance-of-plant simulation modules. Additionally, a system classification matrix has been developed for an iPWR to delineate support and front-line systems and iPWR attributes belonging to the six commonality classifications. Using this information, a RELAP-HD iPWR nodalization has been developed to assess methodology and tool capability. Finally, a method was developed to ascertain the SVV failure probability evolution as a function of the time-dependent pressure, temperature, viscosity and valve cycles. In the future, the ADS-IDAC computer code will be extended to analyze multiple reactors in a dynamic PRA framework by implementing further enhancements to event tree modeling for two modules.

The research described in this paper is expected to develop and demonstrate a novel methodology that provides a framework for a more realistic MMPRA analysis and assessment of the relative contribution of important core damage end states. These end states, extracted from a static, single-module PRA, expand the system state-space to more than one reactor module and evaluate initiating events and accident sequences composed of hardware failures and human errors of the reactor modules.

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