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Title: Proof of Concept of the use of Advanced Computational Accident Scenario Modeling for Safety Margin Economic Evaluation

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The goal of this project was to pave the way for more data-driven decision making when considering safety within Nuclear Engineering by proving the concept of new and innovative accident scenario modeling techniques for the analysis of the economics of nuclear safety margins. To do this, a simple, if extremely detailed, cost-benefit analysis of potential nuclear power plant upgrades related to safety was performed. In this analysis, the cost of the upgrade was the direct monetary cost of implementing the upgrade. The benefit of the upgrade was the Risk avoided by implementing it, where Risk is the probability the upgrade will prevent or mitigate a radionuclide release, multiplied by the economic consequences of the unprevented or unmitigated radionuclide release. Offsite economic consequences have been found to scale largely linearly with the magnitude of the radionuclide release. To find the probability of a

nuclear power plant upgrade preventing or mitigating a radionuclide release, Monte Carlo sampling of accident scenario stochastic parameters was used. By taking advantage of modern super computing capabilities to account for randomness within accident scenarios, a more in-depth and detailed view of safety was attained than is possible with older, more binary approaches. By mapping out the 'failure space' comprised by all possible combinations of stochastic parameters that lead to radionuclide release in an accident scenario, both with and without an upgrade, the average impact of the change was analyzed. Finally, comparing the costs and benefits of various potential power plant upgrades, the most cost-effective ways of improving nuclear safety were discerned.

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# Proof of Concept of the use of Advanced Computational Accident Scenario Modeling for Safety Margin Economic Evaluation

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Thomas H. Riley

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APPROVED:

Major Professor, representing Nuclear Engineering

Head of the School of Nuclear Science and Engineering

Dean of the Graduate School

I understand that my thesis will become part of the permanent collection of Oregon State University libraries. My signature below authorizes release of my thesis to any reader upon request.

Thomas H. Riley, Author

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## 1. Introduction

Idaho National Laboratory (INL) has developed a novel approach to Probabilistic Risk Assessment (PRA) analysis, using a Monte Carlo (MC) informed approach. RAVEN is a probability distribution agnostic and computer code agnostic platform for using MC style sampling of input parameters for Thermal Hydraulics (TH) codes for the purposes of a Risk Informed Safety Margin Characterized (RISMC) approach to PRA analysis (Alfonsi et al. 2013; Smith, Rabiti, and Martineau 2012). As well as the standard deterministic safety margins – often characterized as a ratio of the stress on a safety mechanism to its ability to withstand stress – RISMC allows for probabilistic safety margins – a probability distribution describing the probability of how sorely a safety mechanism will be taxed, or potentially overwhelmed. This allows for new, more detailed analysis of transients.

The aim of the research described here was to provide the proof of concept of this type of RISMC approach to PRA as a tool for the analysis and comparison of the cost-effectiveness of nuclear power plant severe accident mitigation safety upgrades. A critical part of this analysis was the use of a modeling code for severe accident scenario analysis, for which MELCOR 2.1 was chosen (Sandia National Laboratories 2011). An extended Station Blackout (SBO) accident scenario was chosen for a proof of concept case studied examined in this research. This accident was applied to a Boiling Water Reactor (BWR) and numerous initial conditions of the transient and basic physical parameters of the reactor were sampled using MC methods and known probability distributions. For each set of sampled parameters, MELCOR model was run to determine whether a radionuclide release would occur and, if so, the magnitude of the release. These models were executed with and without a series of nuclear power plant severe accident mitigation safety upgrades applied to enable a comparison of the results. Finally, the benefits of each upgrade configuration were compared to the cost of installing, maintaining, and operating the upgrade to compare for cost-effectiveness.

This thesis includes a Literature Review, in Section 2, discussing a survey of the existing body of work and the ways in which this work is grounded in existing knowledge and efforts. The Research Question that motivated the project and the goals of the project is also discussed, in Section 3. The means and methods by which these goals were pursued, and the research performed is discussed in Section 4. The results obtained in this project are presented and discussed in Section 5 and the conclusions that can be drawn from these results are discussed in Section 6. Finally, in Section 7, possible applications, extensions and further lines of inquiry related to this work are discussed.

## 2. Literature Review

### 2.1. *PRA development*

Probabilistic Risk Assessment has existed since the dawn of commercial Nuclear Power in the United States. In the early days of Nuclear Power generation (1957-1975), US Nuclear Safety Regulations were guided by WASH-740, 'Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants,' or 'The Brookhaven Report' for short (USAEC, 1957). The Brookhaven Report is an analysis of what is, in its writers' opinion, the maximum credible accident. They decided this was a major meltdown at a plant with no containment building, unfavorable weather conditions, and half of the reactor released into the atmosphere as a fine dust. Original estimates from these analyses and assumptions were 3400 deaths, 43,000 injuries, and property damage of roughly $60 billion (2013 dollars) of property damage. This was later increased to 45,000 deaths, 100,000 injuries, and roughly $130 billion in property damage when the Brookhaven Report was revised to account for newer, larger reactors (WASH-740). Assuming the worst-case conditions for all variables, however, was unrealistic – the assumptions made were overly conservative, and the radionuclide release data was based on fallout data from atomic bomb tests. These problems were later revised through further, more realistic analysis.

WASH-1400 (Rasmussen et al, 1975), titled “The Reactor Safety Study” but also called the “Rasmussen Report,” after its lead author, changed the approach to accident analysis from assuming the worst conditions in all cases to using fault trees and event trees that describe the evolution of an accident scenario and assess the probability of an accident progressing to core damage and the release of radionuclides to the environment. It also considered the consequences of such an accident. WASH-1400 concluded that the probability of a full core meltdown in a modern (at the time) LWR was roughly 1 per 20,000 years of reactor operation. The American Physical Society later criticized the report for only accounting for deaths in the first 24 hours after an accident, ignoring the potential of a high radiation dose to cause cancer deaths many years after the initiating event (APS, 1984), but WASH-1400 remains as the first attempt to apply modern fault tree methods to nuclear power plant accident analysis.

In response to the accident at Three Mile Island (TMI), NUREG-880 was published (USNRC, 1983), setting new public safety goals for Nuclear Power. These were mostly qualitative goals, but with one quantitative goal of having no more than one core melt per 10,000 reactor operating years. These goals were set to provide “an explicit policy statement on safety philosophy and the role of safety-cost tradeoffs” (NRC, 1983), give industry safety discussions a series of guidelines to direct safety related decisions, state the NRC’s views on the acceptable level of risk to public health, and to address increasing public concern in the wake of TMI.

Three years later the NRC revised 10CFR50 with 51FR30028, containing two qualitative goals and two quantitative goals (USNRC, 2002a). The Quantitative Goals contained in 51FR30028 are:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 0.1% of the sum of prompt fatality risks from other accidents to which members of the U.S. population are generally exposed; and

– The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 0.1% of the sum of cancer fatality risk resulting from all other causes.

The Qualitative Goals contained in 51FR30028 are:

– Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health; and

– Societal risk to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should bear no significant addition to other societal risks.

The Quantitative and Qualitative Goals of NUREG-880 are two ways to express the same objective – Nuclear Power must be safe. NUREG-880 also recommended that “the overall mean frequency of a large release of radioactive materials to the environment should be less than 1 in 1,000,000 years of reactor operation,” a guideline later codified into 10CFR50.109 (USNRC, 2003b) and used when evaluating facility changes and updates.

Since NUREG-880 was published, many studies of the consequences of Nuclear Power have been performed, including NUREG-1150 (USNRC, 1990) in 1990 and State-of-the-Art Reactor Consequence Analysis (SOARCA) in 2012 (USNRC, 2012). NUREG-1150 was an analysis of five nuclear power plants, using Accident Progression Event Trees, to quantify the progression of power plant accidents and the likelihood of safety systems being unable to properly withstand the accident. The plants analyzed were the Peach Bottom Atomic Power Station in Pennsylvania, Surry Power Station in Virginia, Grand Gulf Nuclear Generating Station in Mississippi, Zion Nuclear Power Station in Illinois, and Sequoyah Nuclear Generating Station in Tennessee. These plants were chosen to give a broad survey of both Boiling Water Reactors

(BWRs) and Pressurized Water Reactors (PWRs) and of multiple models of BWR and PWR and analyzed using the Accident Progression Event Tree approach.

An Accident Progression Event Tree (APET) looks at the way containment can fail or be bypassed, as well as the way severe accidents affect the mode and timing of containment failure and magnitude of radionuclide release to the environment (Hakobyan et al, 2008). In the older WASH-1400 event trees, accident progression was examined solely on the basis of whether a particular safety system succeeds or fails on demand, where the new Accident Progression Event Tree approach asks non-binary questions like ‘how long does battery power last’ and ‘how fast does the vessel leak water in a loss-of-coolant-accident (LOCA)’ and so forth. The questions all have at least two answers, creating multiple branches to follow after each branch point in the sequence of events. Generally, fault tree analysis is not used to generate branch probabilities for branch points in an APET. Branching probabilities are calculated using physical conditions from the severe accident scenario in comparison to the criteria for the different branches of a branch point (Hakobyan, 2006). Unfortunately, APETs do not give a deterministic pass/fail outcome for a scenario as they are inherently probabilistic, and uncertainty analysis is used to determine failure probability by performing accident progression calculations with different input assumptions.

SOARCA later built on NUREG-1150 by incorporating onsite and offsite actions that may prevent or mitigate accident consequences, as well as more detailed computer modeling to look at how severe accident conditions affect a reactor and how a radionuclide release could affect the public. The SOARCA team used core damage frequency to select scenarios, as the physical integrity of the fuel rods is the first barrier to major radionuclide release. The SOARCA team used older PRA models to identify scenarios with a high core damage frequency to focus on the most likely severe accident scenarios, as well as some lower probability accidents that, for various reasons, have high potential consequences. The team used a detailed MELCOR model to analyze the onsite accident progression and mitigation measures for both the BWR design at the

Peach Bottom Atomic Power Station and the PWR design of the Surry Power Station. Using MELCOR output data, MACCS2, a nuclear release consequence evaluation code, was used to model offsite release of radioactive material, as well as the emergency response to the release and the potential health effects of such a release (USNRC, 2012). The SOARCA's main findings suggested that existing resources and procedures can prevent or mitigate an accident and its impact to prevent it from affecting public health, that even wholly uncontrolled accidents take significantly longer to progress and release radionuclides than prior analyses concluded, and that the analyzed accidents would cause essentially zero immediate deaths and only an extremely small increase in the risk of long-term cancer deaths.

Modern PRA efforts are divided into three levels that correspond to important transition points in the progression of an accident scenario. Level 1 starts with an initiating event and ends at Core Damage, Level 2 starts with Core Damage and ends with Radionuclide release, and Level 3 starts with Radionuclide release and examines the consequences. The advantage of higher level PRA is that it provides more in depth and detailed analysis of the risks and repercussions of accident scenarios than lower level PRA. The disadvantage is that higher level PRA costs a great deal more than lower level PRA.

A Level 1 PRA is solely a calculation of the core damage frequency. It looks at accident progression in terms of accidents that lead to core damage to estimate the Core Damage Frequency (CDF), starting from an initiating event and branching out through safety system success or failure until core damage is reached. This is represented graphically with fault trees. Each of these fault trees is analyzed to provide a core damage frequency for that particular accident, then all the frequencies are added together to get a total CDF.

A Level 2 PRA begins at the end of a Level 1 PRA by examining the plant's response to the Level 1 events that lead to core damage and analyzing how the plant responds to this state.

Incidents that lead to core damage are typically called severe accidents. Level 2 PRA is analysis of the plant's severe accident response, and whether it is capable of keeping the severe accident consequences sealed within the containment building. This uses further fault trees and, rather than primarily looking at safety systems success/failure, looks at phenomenological events like "Steam Generator tube rupture" or "hydrogen explosions." Because different severe accident paths lead to different Plant Damage States (PDS) when Core Damage occurs, severe accident progression analysis is necessary for each PDS, making Level 2 PRA drastically more expensive and lengthy than Level 1 PRA.

A Level 3 PRA begins with loss of secondary containment. It estimates the consequences of a radionuclide release, and combined with levels 1 and 2, presents an overall estimate of the effect on the people living near the plant, and the potential for the plant to contaminate the surrounding environment with radioactive material. The consequences, both in terms of the health of the public and the quality of land, depend on multiple factors, though all of these factors affect both the health of the public and the land quality. For example, population density and evacuation readiness and conditions affect only the health effects white others, like weather conditions, geography, and the size of the radionuclide release affect both. A Level 3 PRA estimates the final measure of risk by combining the consequences and likelihood of a radionuclide release. However, it is rarely done because it requires a great deal of computational power and is very expensive, as the various paths to radionuclide release in an accident scenario affect the nature of the radionuclide release, and these differences need to be fully accounted for.

## 2.2. *Risk-Informed Safety Margin Characterization*

The Risk-Informed Safety Margin Characterization (RISMC) approach is the integration of probabilistic and deterministic safety analysis methods into one cohesive method that considers accident risk instead of only accident probability or accident consequences (Smith, Rabiti, Martineau, 2012). The interaction between mechanistic and probabilistic tools is shown in Figure 1.

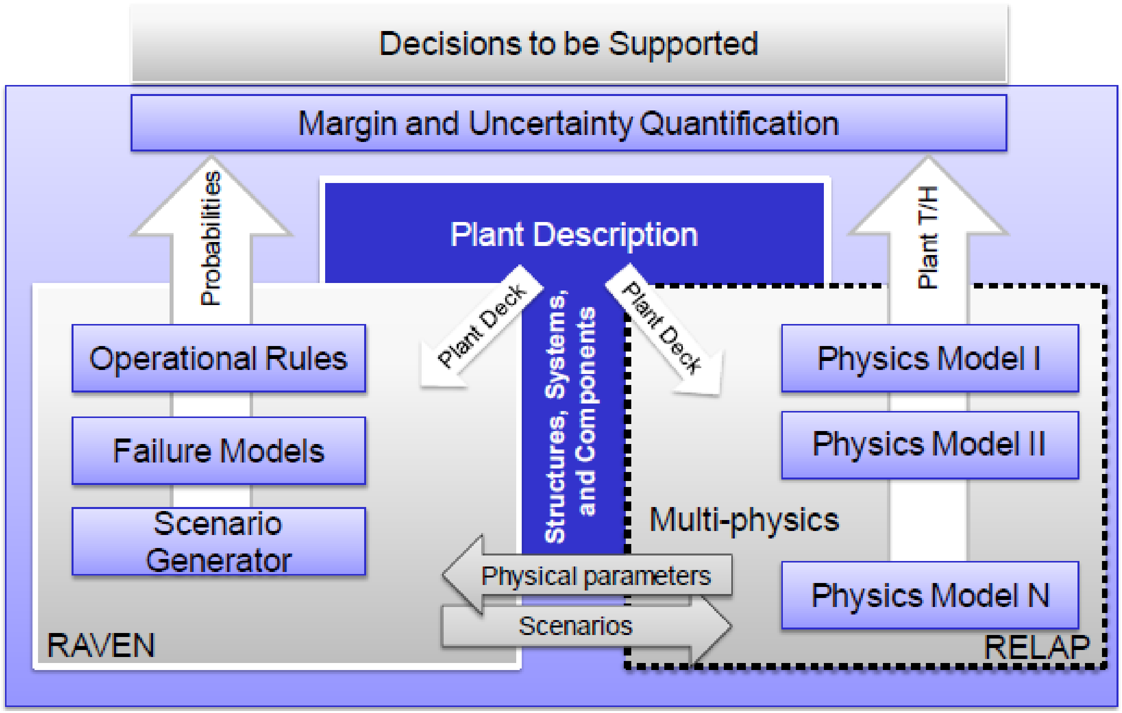


Figure 1. Probabilistic and Mechanistic Approach Interactions (Mandelli et al, 2013)

The probabilistic analysis is performed with standard risk assessment techniques, and the mechanistic approach is accomplished through plant physics calculations. By melding the two, uncertainties can be quantified, and safety margins characterized. This interaction between the two approaches is accomplished through alterations in the plant physics model parameters and accident scenarios stochastic parameters to account for unknown and unpredictable variables within an accident scenario. Older methods rely strictly on conservative assumptions to account for unknown variables, where risk-informed analyses consider both the potential values of these variables and the probability of the potential values occurring within an accident scenario.

The Plant Description informs both the mechanistic and probabilistic sides of the RISMC approach. On the probabilistic side, the Plant Description governs plant operational rules and component failure models, which in turn control the stochastic parameters that are important to the construction of RAVEN input files. These stochastic parameters represent important, but unpredictable aspects of an accident scenario, and accounting accurately for these so-called "unknowable unknowns" instead of simply assuming worst-case scenario values for them is one of the major improvements the RISMC approach represents over older methods. On the mechanistic side, the Plant Description forms the most fundamental basis for the physical specifications described within the multi-physics model used to analyze the physical progression of an NPP accident scenario or severe accident scenario. The operating temperature of a plant, the power generation, the decay heat, the volume of the reactor pressure vessel and various ECCS systems, and other important aspects of the physical structure of the power plant and the plant description are captured within the multi-physics model and reflected in the accident scenario progression and output data.

For the probabilistic parts of the RISMC approach here, RAVEN is used to sample input parameters to the physical model and for processing output data from the physical model. RAVEN requires the ability to access the inputs and outputs of a code but provides for an array of tools to do so that enable it to interact with almost any code in existence. With access to the inputs and outputs of a code, RAVEN is capable of generating any number of iterations of input decks for multi-physics modeling codes such as RELAP5, RELAP7, MELCOR, and others to represent the possible variations in accident scenarios. RAVEN allows for a variety of stochastic parameter probability distribution functions to be used – triangular, normal, exponential, and uniform to name a few, as well as multi-dimensional distributions that allow for dependences between parameters. RAVEN employs several different parameter sampling strategies, ranging from simplistic forward samplers like simple random sampling, grid sampling, and stratified sampling to smart adaptive samplers like limit surface search approaches and dynamic event tree sampling.

For the mechanistic portion of the RISMC approach discussed here, a multi-physics model of an NPP is used. Using input files created based on the plant description and accident scenario stochastic parameter samplings generated by RAVEN, the multi-physics code models the progression of the desired accident scenario, including heat and fluid transfer, material properties, and the possibility of core degradation. Some codes model up to core damage (notably RELAP5, RELAP5-3D, and RELAP7), while others are capable of modeling beyond core damage, up to and including radionuclide release. MELCOR is an example of these latter codes.

As well as generating scenario-based inputs based on plant descriptions and stochastic parameter sampling, RAVEN is capable of data processing to handle the volume of data generated by huge numbers of multi-physics model runs. While output parsing needs to be programmed on a per-code basis, RAVEN is capable of parsing any plain-text output format and presenting the data in a comma separated value (CSV) file. RAVEN can also present key output data from a scenario alongside the probability of that particular combination of stochastic parameters being sampled, allowing for the probability and consequences of a scenario to readily be combined to produce a risk-based scenario evaluation. The aim of this research project is to use this capability for risk-based scenario evaluation to compare the risk posed by long-term SBO scenarios without NPP safety upgrades to the risk posed by the same scenarios with NPP upgrades safety in order to perform a risk-informed comparative cost-benefit analysis of the NPP safety upgrades. Risk-informed approaches have been used for a variety of applications previously. Dube et al. applied a RISMC approach to extended power uprate analysis for both BWRs and PWRs (Dube et al., 2014). For the PWR analysis, they examined a Loss of Main Feedwater scenario with loss of auxiliary feedwater, leading to the necessity of feed and bleed cooling to prevent core damage. For BWR analysis, the focus was on station blackout sequences that lead to core damage. Using a Monte

Carlo method to sample stochastic parameters for randomly generated cases, the scenarios were evaluated using the MAAP4 code. The study found that for small power uprates, the loss of safety margin was small, and that it was possible to recover the safety margin given moderate plant operation and design changes.

Liang et al. used the RISMC methodology to generate a spectrum of peak clad temperature for 14 potential variations of a Large Break Loss-of-Coolant Accident (LBLOCA) to analyze the peak clad temperature safety margin from a risk-informed perspective (Liang et al. 2016). They used RELAP5 to model each case. It was found that the peak clad temperature generally increased for sequences of decreasing likelihood (i.e. the accidents closer to causing core damage were less likely), and the risk-informed peak clad temperature safety margin ranged from 184.2K to 202.1K, greater than the margin found by prior deterministic methods.

Sherry, Gabor, and Hess combined a Monte Carlo method of sampling important parameters with the MAAP-4 code to perform a risk-informed analysis of a Loss of Feedwater (LOFW) transient at a PWR (Sherry, Gabor, and Hess 2013). They performed their analysis using 100 Latin Hypercube Sampled runs for each of 11 LOFW scenario variations that were various combinations of the number of Pilot Operated Relief Valves available, the number of trains of the High Pressure Safety Injection system and Centrifugal Charging Pumps available, and whether the Reactor Coolant Pump successfully tripped or not. They found the conditions typically accepted as success criteria in traditional PRA analysis to be potentially non-conservative. The typical success criteria of one Head Safety Injection System and two Pilot Operated Relief Valves, with no Centrifugal Charging Pumps operational, had a 20-50% conditional probability of core damage, though the success criteria of one Centrifugal Charging Pump and one Pilot Operated Relief Valve was found to be appropriately conservative.

Mandelli et al. used risk-informed analysis to examine the repercussions of a power uprate of a BWR during a station blackout scenario. They coupled RELAP5-3D to RAVEN to analyze a BWR with a Mk1 Containment (Mandelli et al. 2013). Using RAVEN to perform stochastic parameter sampling across twelve stochastic parameters at both 100% core power and at 120% core power, they found that a 20% power uprate doubled the conditional core damage probability during a station blackout, increasing it from 9.82E-3 to 1.96E-2. They found that the increased core decay heat reduced the time until the Automatic Depressurization System (ADS) would trigger, as well as decreasing the amount of time plant staff would have to secure alternate low-pressure sources of water after the ADS triggered. Additionally, they modeled the effects of the FLEX system of portable AC and DC emergency generators adopted in response to the Fukushima accident, and found that it decreased the core conditional damage probability, with a 20% power uprate from 1.96E-2 to 4.59E-3, also a decrease when compared to the 100% power conditional core damage probability without the FLEX system.

## 2.3. *Monte Carlo Applications*

A key piece of RAVEN's functionality is the use of Monte Carlo methods for the sampling of input parameters for deterministic code calculations. Monte Carlo methods are a category of numerical algorithms that rely on random sampling to find answers and are often useful when it is impossible to calculate an answer directly. At a simplistic level, they rely on the idea that if a person throws a sufficient number of darts at a dartboard, one can establish, with good confidence, the probability of a thrown dart landing in any designated area of the dartboard. A simple example of the use of a Monte Carlo method is the calculation of pi. If one randomly samples two variables, bounded from 0 to 1, adds them in quadrature – square them, sum the squares, and take the square root of the summed squares – geometry dictates that the fraction of samplings that will produce a result less than or equal to one is equal to pi divided by 4. With a sufficient number of sampled points, this method can produce accurate estimations of pi.

Monte Carlo methods are extremely well-established techniques for finding answers to problems with uncertain input parameters and problems too complex to be solved analytically or computationally. In this work, the former is of greater concern, as current thermal-hydraulics codes are more than capable of modeling and predicting plant behavior mechanistically. One of the greatest pitfalls of thermal-hydraulics codes is that one cannot precisely predict the initial conditions of a plant when a transient begins (for example, the temperature outside affects the state of the plant, and is impossible to predict precisely in advance). To circumvent this issue, Monte Carlo techniques are to be used to repeatedly run a thermal-hydraulics simulation of choice with a variety of initial conditions randomly sampled and imposed for each run. Using Monte Carlo techniques to randomly sample initial conditions for a mechanistic simulation is done in other fields to establish a good prediction of what conditions will cause a system to fail.

Monte Carlo techniques have been used to account for stochastic parameters in a variety of fields, including radiation transport, robotics, aerospace, microelectronics, and telecommunications applications. In Aerospace applications, Monte Carlo techniques were used to account for component failure in models of long-term manned spaceflight operations, with particular interest in a manned mission to Mars (Bavuso, 1997). By sampling for component failure of both active components breaking down during operations, as well as on demand failures from backup components, the researchers examined the overall reliability of a spaceflight system for a manned mission to Mars. In the past, Monte Carlo techniques were not used in aerospace applications for reasons similar to those why Monte Carlo techniques have not been previously used for systems level thermal hydraulics modelling of NPPs – it was excessively expensive due to the cost of the computer resources involved with using Monte Carlo techniques. However, advances in both supercomputing and in Monte Carlo techniques themselves have reduced the costs of Monte Carlo techniques greatly.

In microelectronics, Monte Carlo techniques are used to aid in modelling what are called 'single event effects' (Weller et al, 2010). These are when the fine semiconductors in microelectronics are disrupted by radiation. Due to the nature of radiation, the timing and location of a radiation event is inherently stochastic and cannot be known a priori. Previously, other methods worked sufficiently well to model single event rates and effects, but the increasingly minute size of computer semiconductors has necessitated the use of Monte Carlo techniques to model the effects of radiation on microelectronics.

In robotics, Monte Carlo techniques are used for autonomous robot localization (Wu et al, 2006). For the purposes of mapping out a robot's surroundings, it is impossible to know in advance where obstacles will be, or to evaluate whether an apparent path is viable. According to Wu et al, other, previous techniques for an autonomous robot mapping its surroundings are not suitable for unstructured outdoor environments. Using Monte Carlo techniques, these environments can be navigated. According to Wu et al, Monte Carlo techniques are also more computationally efficient than older methods of localization, on top of being more versatile.

In telecommunications, Monte Carlo techniques are used for the analysis of error rate performance in communications systems (Bononi et al, 2009). Bononi et al lay out a methodology for the use of Monte Carlo techniques in analyzing telecommunications systems. With their methodology, Monte Carlo techniques are adaptively used to improve Importance Sampling techniques – an older method for estimating the occurrence of events in the system – by making them self-adapting. This greatly lowers the prior knowledge one needs of the system and shortens the planning and construction phase of the model. Similarly, in NPP modeling, by using a reactive model and Monte Carlo techniques, it is possible to analyze the transient event sequence of a reactor without knowing a great deal, in advance, about how the plant will respond to the transient.

### 2.4. *Prior MELCOR modeling of BWRs*

MELCOR is the model being used with RAVEN for this research. Developed at Sandia National Laboratories (SNL), it is an engineering-level computer code used to model severe accident progression in nuclear power plants (Code Manual for MACC2s, Vol.1, 1998). MELCOR was chosen over RELAP5 because one of the goals of this project is to do severe accident consequence evaluation, and RELAP5 lacks the capability to do any kind of severe accident analysis. MELCOR is a widely used code for severe accident analysis and is an industry standard in nuclear power and research. It has been validated and assessed extensively on LWR, including being used for the State-of-the-Art Reactor Consequence Analyses (USNRC, 2012). Because of this extensive use, it is known that MELCOR is suitable for modeling severe accident scenario progression in LWRs.

Polo-Labarrios and Espinosa-Paredes performed a comparative study of hydrogen generation during short term SBOs using MELCOR and SCDAP/RELAP5 (Polo-Labarrios and Espinosa-Parades 2015). They examined a severe, unmitigated short term SBO in a BWR-5 with Mark-II containment. Specifically, the modeled sequence includes the failure of HPCI and RCIC, which normally function during an SBO. They found that MELCOR and SCDAP/RELAP5 produced similar thermal-hydraulic results up to the occurrence of core damage but diverged during and after core relocation. Regarding hydrogen production, MELCOR predicted approximately 20% greater hydrogen production than SCDAP/RELAP5.

Cardoni et al. used MELCOR to perform analysis of the SBO at the Fukushima Daiichi Unit 3 reactor (Cardoni et al. 2013). Using publicly available data concerning the accident scenario progression, they ran a series of four variations of a MELCOR model to attempt to account for uncertainty in available data regarding the exact progression of the accident scenario. Using MELCOR, they were able to largely reproduce the TEPCO data available and to make several predictions regarding the transport of $H_2$ from the containment vessel to the reactor containment building.

Wang, Wang, and Teng performed a comparative study of severe accidents using SCDAP/RELAP5, MAAP, and MELCOR (Wang, Wang, and Teng 2004). They modeled both a LBLOCA and a severe SBO for a BWR-6 with Mark-III containment. They found some timing differences between the codes but found good agreement regarding the overall sequence of events. For the LBLOCA they modeled an accident sequence where the ECCS was not functional, so the reactor core uncovered and reached core damage temperatures very quickly, leading to a great deal of hydrogen production in the core. They found good agreement for hydrogen production results between SCDAP/RELAP5, MAAP, and MELCOR.

Li et al. performed a MELCOR analysis of melt behavior in the lower head of a BWR during a LOCAs and SBOs (Li et al. 2016). In the LOCA case, the ECCS was disabled, making it similar to the LBLOCA with loss of water injection scenario we intend to model. They ran four cases to examine variations in melt behavior in LOCAs and SBOs, specifically to examine the impact of Instrument Guide Tube (IGT) failure on accident scenario progression. They found that the inclusion of IGT failure in lower head modeling made debris relocation occur multiple hours sooner than if IGT failure modeling is not included.

### 2.5. *Nuclear Power Plant Severe Accident Mitigation Upgrade Analysis*

During the disaster at the Fukushima Daiichi Nuclear Power Plant, conditions arose during the accident that damaged the containment venting system and rendered it inoperable in several of the Fukushima Daiichi units. The inability to operate the containment venting system contributed significantly to the buildup of pressure and hydrogen in the reactor building that lead to the hydrogen explosions that ruptured the reactor buildings of several of the reactors at the Fukushima Daiichi Nuclear Power Plant, leading to multiple large-scale releases of radioactive material to the environment. In response to the Fukushima Daiichi disaster, the United State Nuclear Regulatory Commission mandated that all operating BWRs with Mk-I and Mk-II containments upgrade their containment buildings with hardened venting systems rated to withstand conditions in containment that are challenging to the continued operation of containment venting systems (Nuclear Regulatory Commission 2013), with the intent that another accident scenario akin to Fukushima would not lead to such disastrous results if proper containment venting were available. Given the choice to mandate the use of this upgrade, we have selected it as a baseline for cost-effectiveness comparison with other upgrades.

A direct addition to the hardened venting system upgrade is to combine it with a passive filtered containment venting system (FCVS). The intent of a passive filtered containment venting system is to clean the steam being vented from the containment of radioactive contaminants as thoroughly as possible. Filters used for FCVSs come primarily in two varieties wet and dry scrubbing. One approach to wet scrubbing is to force contaminated gas through a body of water, causing water to become entrained in the gas flow, which removes both radioactive and non-radioactive aerosols from the gas. Other wet scrubbers use wetted packed fibers for their method of capture, and yet others spray liquid into the stream of gas, rather than forcing the gas through a pool of water (Morewitz 1988). Dry scrubbing can be accomplished by forcing the gas flow through beds of densely packed stainless-steel fibers, beds of sand or gravel, or HEPA filters, which employ randomly arranged fiberglass fibers of various diameters for filtration. The commercially available passive FCVS sold by Areva employs a combination of a wet scrubber that forces the vented gas through a pool of water for a wet scrubbing stage, followed by a metal fiber dry scrubbing stage, and is advertised as having upwards of 99.99% of aerosols and 99.5% of iodides during venting (AREVA 2011).

A second major threat to reactor building integrity is the buildup and uncontrolled ignition of hydrogen gas, leading to a deflagration or explosion that ruptures the reactor building pressure boundary, leading to uncontrolled radioactive release. One potential upgrade to prevent the buildup of hydrogen in the reactor building is the addition of a series of hydrogen igniters to the containment building. Hydrogen igniters generally come in the form of glow plug igniters and spark igniters, with glow plug igniters being more common (IAEA 2011). Glow plug igniters are simple electrical resistance heaters that produce a hot surface for the combustion of hydrogen. A drawback to glow plug igniters is that they have high power demands and can be very difficult to supply power to during a Station Blackout scenario. Spark igniters operate along the same principles as a gasoline car's spark plugs, generating an electric spark on demand – the extreme flammability of hydrogen makes it easy for a simple spark to ignite a confined hydrogen-oxygen gas mixture. A major advantage spark igniters that have over glow plug igniters is a reduced power demand, though electric interference from spark igniters is a point of concern. MELCOR is capable of modeling hydrogen igniters, using a simple model based on LeChatelier's formula to determine the gas fraction required for ignition, with the gas fraction limits required for ignition with igniters being relaxed – less hydrogen required, more carbon dioxide presence permitted – than the gas fraction limits required for ignition without igniters (SNL 2011).

Where hydrogen igniters require power and are thus greatly vulnerable to failure by loss of power – such as during a Station Blackout accident – other hydrogen mitigation solutions do not. Passive Autocatalytic Recombiners (PARs) use materials to act as a catalyst for the recombination of hydrogen and oxygen at conditions outside of the normal limits of hydrogen flammability. PARs are, at a basic level, just catalytic materials arranged in configurations conducive to high surface area and gas flow across all possible surfaces. The heat of the recombination drives a natural circulation convection current through the PAR, drawing in more gas from below (IAEA 2011). The greatest advantage of PARs is their passivity – they have no moving parts and do not demand power, making them invaluable in a Station Blackout scenario where hydrogen recombination is necessary, but power is unavailable, or other severe accident progressions where there are challenges making it difficult or impossible to provide power to hydrogen igniters. The downside to this passivity is that PARs cannot be "turned off" when hydrogen production is too rapid for them to keep up and the PARs can change from being a safe way to mitigate hydrogen production to being a potential hot surface and cause of a deflagration.

Dehjourian et al. investigated the effects of containment spray and PARs on hydrogen effects in the containment of a PWR using MELCOR during a severe LOCA accident sequence (Dehjourian et al. 2016). They compared accident sequences with and without containment sprays and with and without PARs and found that 55 PAR units installed throughout the containment provided a significant reduction in hydrogen accumulation within the containment, improving safety, and that containment sprays did not hamper the operation of the PAR units.

## 2.6. *Accident Scenario Consequence Evaluation*

An important part of a comparative cost-benefit study across power plant upgrades is evaluating the impact of the upgrades. We intend to use probabilistic reduction in release source term as the metric by which the impact of upgrades will be evaluated. When considering power plant accidents, potential accidents were broken into six categories:

Table 1. Qualitative Power Plant Accident Societal Economic Consequences

| Accident Category | Societal Economic Consequences |
|---|---|
| Less than Severe (No Core Damage) | Minimal – plant repairs, replace power. |
| Core Damage, RPV intact | Moderate – decommission plant, replace power, replace plant |
| Core Damage, RPV breached, Containment intact, no release | Moderate – decontaminate plant, decommission plant, replace power, replace plant |

| | |
|---|---|
| Core Damage, RPV breached, Containment intact, small release | Moderate – decontaminate plant, decommission plant, replace power, replace plant |
| Core Damage, RPV breached, Containment breached, significant release | Major – decontaminate plant, decommission plant, evacuation, land use restrictions, agricultural restrictions, land decontamination |
| Core Damage, RPV breached, Containment breached, massive release | Massive – decontaminate plant, decommission plant, evacuation, relocation, interdiction, condemnation, land use restrictions, agricultural restrictions, land decontamination |

In reviewing the literature on severe accidents, there were no cases found in which plant operations were successfully recovered after a severe accident. Because of this, for the purposes of this analysis, the cost of repairing or replacing the plant is not heavily considered, as it is an assumed loss across all examined power plant severe accidents analyzed. As such, and given the project's focus on severe accident mitigation, the primary information of interest lies in the differences in the offsite economic consequences of severe accidents with and without power plant upgrades.

For the evaluation of the impact of nuclear power plant upgrades for severe accident mitigation, we have chosen to use the source term of the radioactive release. Reviewing documentation for the MELCOR Accident Consequence Code System (Sandia National Laboratories 1998), the inputs that inform the offsite consequences of a severe accident can be, in general terms, lumped into three categories: release source term parameters, weather, and site information, with the last of these being something of a catchall that includes both the physical location of the site and the population and economic details of the surrounding area, and also emergency planning decisions and decisions regarding dose thresholds for sheltering, evacuation, relocation, interdiction, condemnation, agricultural restrictions, land use restrictions, and other responses to a nuclear release or the potential for a nuclear release (Sandia National Laboratories 1998). Information regarding source term parameters is dependent, in many ways, on the nature of the power plant accident and the results of the MELCOR analyses performed in the course of this research. Weather is, in its own right, a set of stochastic parameters that can be broken down, sampled, and examined for insights in the same manner many power plant accident parameters are being examined in this project (in fact, MACCS is capable of doing exactly this). Lastly, the site information catchall is a set of information generally in existence before an accident occurs. Performing a detailed Level 3 PRA analysis encompassing all of the stochastically sampled power plant accident analyses is beyond the scope of this research, and site-specific information is unnecessary for a generic, proof-of-concept study, so this research focuses on the economics of the release source term parameters and their impact on the offsite economic consequences of accident scenarios.

There is little investigation in the literature of the comparative offsite economic consequences of various potential accident scenarios, as most regulation and research is focused on potential health effects for the public. The literature that has been identified, however supports the approach of using release source term as an estimator for the offsite economic consequences of a large radionuclide release. Silva, Ishiwatari, and Takahar performed a comparative examination of the cost of various severe accidents using the Off-Site Consequence Analysis code for Atmospheric Release in reactor accident (OSCAAR) code developed by the Japanese Atomic Energy Research Institute (Silva et al. 2014). Using OSCAAR, they performed a Level 3 PRA analysis to evaluate the economic consequences of a nuclear power plant severe accident, including radiation effects, psychological effects, relocation costs, evacuation costs, shelter costs for evacuees, decommissioning the plant, replacement electricity, and land decontamination. They found, among other results, that the relationship between the cost of each severe accident and its release source term was fairly linear ($R^2 = 0.98$). With this result in mind, it is believed that source term magnitude reduction or prevention will be an appropriate metric by which to evaluate the economic effectiveness of nuclear power plant safety upgrades.

## 3. Research Question

The goal of this project was to examine and, if possible, prove the concept of using the RAVEN code as a tool for the economic analysis of NPP safety upgrades. Severe accidents were examined for a generic BWR plant until they had been examined sufficiently to provide enough data for a proof of concept safety upgrade economic analysis. This was determined to be when sufficient data had been produced to provide for a reasonable comparison between accident scenarios with and without various safety upgrades, as this would allow for cost-effectiveness conclusions to be drawn. Additionally, it is believed that the data generated by this project may shed light on ideal ways in which to augment the safety of older Light Water Reactor NPPs, in support of the Light Water Reactor Sustainability program.

For the methodology to be considered a successful system for analyzing NPP safety upgrades using a RISMC economic approach, it needs to produce predictions regarding the cost-effectiveness of the tested safety upgrades. Additionally, the predictions need to be reasonable, and not obviously wrong. Where comparative data is not available, engineering judgement must be used to evaluate and examine the predictions made to see if they are reasonable and realistic.

## 4. Methodology

### 4.1. *The RISMC Approach*

The RISMC approach not only allows the frequency of undesirable transient outcomes to be determined, but also how severe the consequences of those outcomes are, and how probable different magnitudes of consequence are (Smith, Rabiti, Martineau 2012). To accomplish this, the RISMC approach uses coupled mechanistic and probabilistic analyses. The physical parameters of an NPP direct the development of the MELCOR mechanist model and the RAVEN probabilistic model. RAVEN handles elements like stochastic parameter probability distribution functions, which in turn represent operational rules of the plant and potential failure models, as well as being used to generate combinations of sampled stochastic parameter values to account for a variety of scenarios. These sampled stochastic parameter values are inserted into the MELCOR model, which models the physical and mechanistic elements of the plant. These include but are not limited to thermal-hydraulics, heat and mass transfer, material properties, and a variety of severe accident progression phenomena. The probabilistic analysis is performed with standard risk assessment techniques, and the mechanistic approach is accomplished through plant physics calculations. By melding the two, uncertainties can be quantified and safety margins characterized. This interaction between the two approaches is accomplished through alterations in the plant physics model parameters and accident scenarios. In the case studies presented here, a single plant was modeled for each case study and was chosen to be a generic BWR with a Mk1 containment. In one case study, a long term SBO (LT-SBO) potentially leading to core damage, containment failure, and large-scale radionuclide release was examined. In another case study, a LB-LOCA with loss of water injection likely leading to core damage, containment failure, and very large-scale radionuclide release was examined.

### 4.2. *Base BWR MELCOR Model*

In a BWR, during normal operations water is boiled directly in the core, and the resulting steam is directly utilized to drive a turbine, then is cooled in a condenser, turned back into liquid water, and pumped back into the core for a complete loop of the water. A high-level schematic of the BWR components involved in this process is shown in Figure 2.

Upon exiting the core of the BWR, the steam is passed through a series of heaters and steam dryers in the Upper Plenum of the Reactor Vessel to ensure that there is no liquid water vapor carried along by the steam. Because of the high rotational velocity and fine precision of a turbine blade, liquid water present in the steam can cause tremendous damage to the turbine and plant. After being heated and dried, the steam is piped out of the Reactor Pressure Vessel (RPV) and into the two-stage turbine. It first enters the high pressure stage of the turbine then enters the low pressure stage of the turbine. These provide mechanical energy to a generator, which converts it to electricity. A two-stage turbine is not required for a plant to operate, but significantly boosts the efficiency with which the turbine and generator can produce electricity from the heat the core produces.

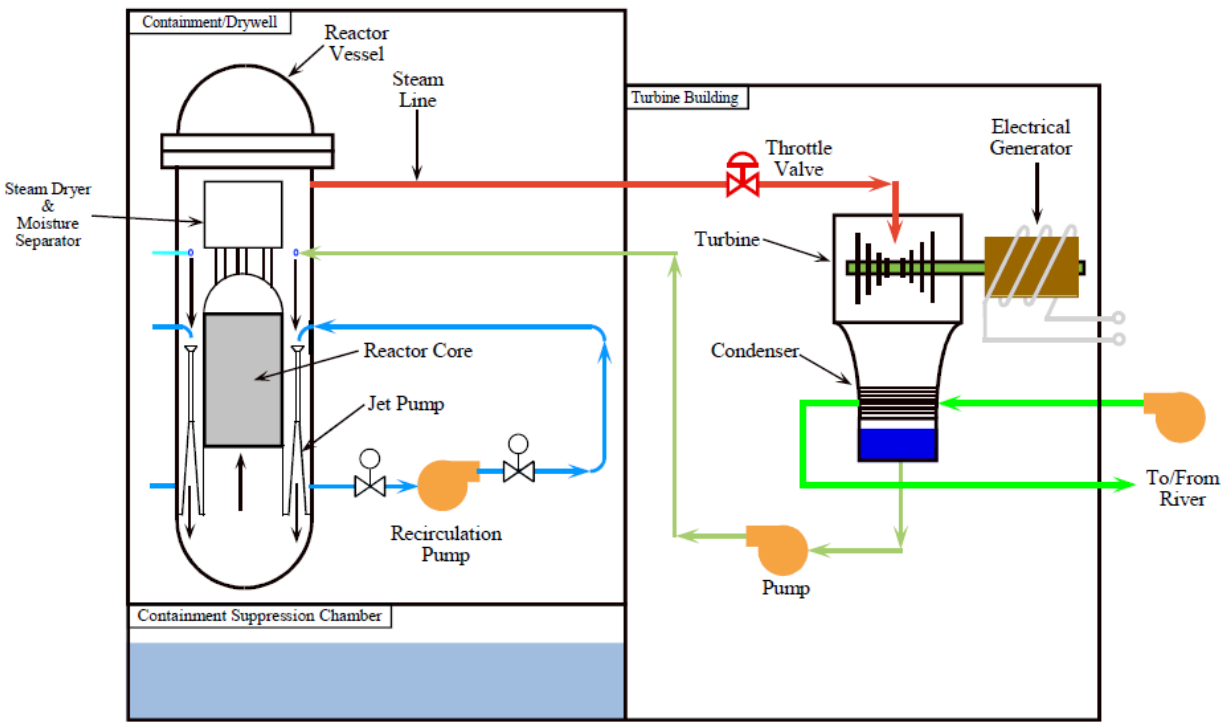


Figure 2. Schematic of BWR Main Loop (nrc.gov 2012)

After leaving the turbines, the steam enters the condenser, where it exchanges heat with a secondary loop of water that acts as an ultimate heat sink. This generally takes the form of either cooling towers that release the heated steam from the secondary side into the atmosphere, or a large body of water that acts as a nearly infinite heatsink. The steam on the primary side of the condenser is converted back into liquid water and pumped through water heaters and back into the core, where it is heated back into steam and the cycle begins again.

BWRs have a variety of redundant, diverse, and independent safety systems. These safety systems include the Reactor Protection System (RPS), the various components of the Emergency Core-Cooling System (ECCS), the Standby Liquid Control System (SLCS), and the Containment Building. The RPS is a computerized system designed to quickly and completely shut down the reactor and render it into a safe and stable configuration without human interference. The RPS will SCRAM the reactor, which is to quickly insert all of the control rods into the reactor, killing the nuclear chain reaction. Additionally, the RPS will activate ECCS subsystems as needed to provide emergency core cooling. SLCS is a safety system that injects a neutron poison into the reactor to shut down the chain reaction and acts as a backup to the RPS in case of failure to SCRAM the reactor.

The ECCS consists of numerous subsystems to provide makeup water to the core to keep the fuel covered and cooled, including the Automatic Depressurization System (ADS), the High-Pressure Coolant Injection system (HPCI), the Reactor Core Isolation Cooling system (HPCI), the Low-Pressure Core Spray system (LPCS), and the Low-Pressure Coolant Injection system (LPCI). With the exception of the ADS, all of these systems, at a basic level, add water to the core. The ADS is a system to vent much of the contents of the core, reducing the pressure and allowing the high volume, low pressure systems to be used. HPCI and RCIC are high pressure systems that can be used to add lesser amounts of water to the core at high pressure, while LPCS and LPCI are low pressure systems that can be used to inject huge amounts of water into the core once it is depressurized. In an SBO transient, LPCI and LPCS cannot be used because they require AC power, while HPCI, RCIC, and the ADS can be operated while battery power is available. Figures of the SLSC and all discussed ECCS subsystems can be found in Appendix A.

In both case studies, a generic BWR power plant with a Mk1 containment was modeled. The plant physics calculations were performed using the MELCOR NPP modeling code (Sandia National Laboratories 2011) and RAVEN (Alfonsi et al. 2013). For the thermal-hydraulics simulations, the main structures examined were the Reactor Pressure Vessel (RPV) and primary containment – the Drywell, Wetwell (also called the Pressure Suppression Pool, or PSP), and Reactor circulation pumps. These are pictured in Figure 3. Additionally, the High Pressure Core Injection System (HPCI), Reactor Core Isolation Cooling System (RCIC), Safety Relief Valves (SRVs), and Automatic Depressurization System (ADS) were considered.

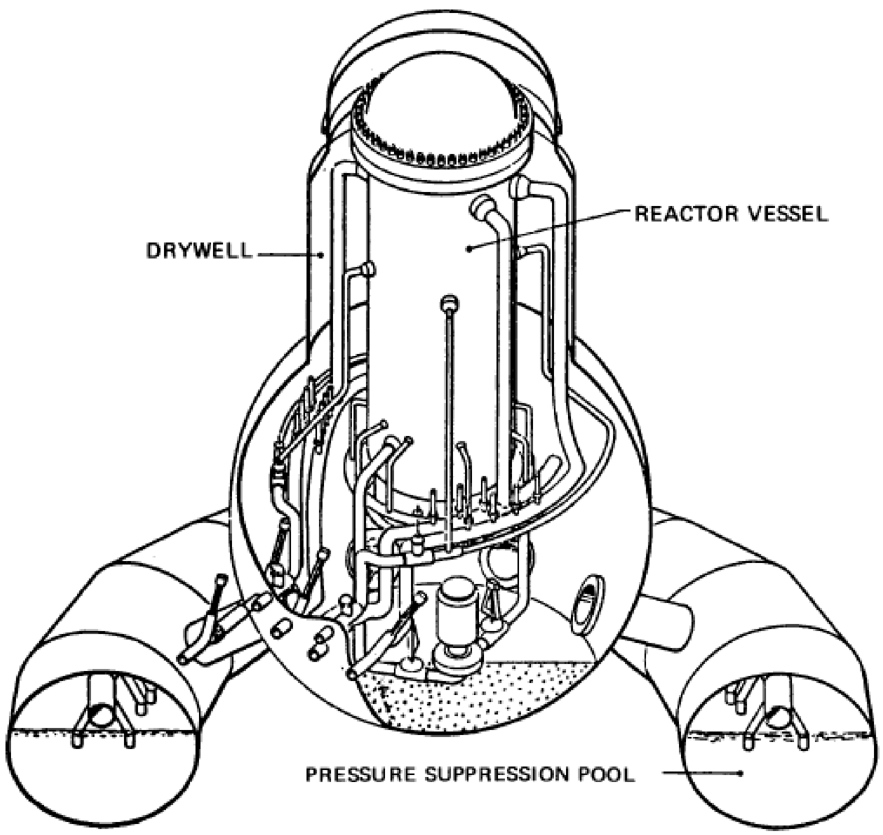


Figure 3. Cutaway View of the BWR Mk. I Primary Containment (Mandelli et al, 2013)

The HPCI injects water from the Condensate Storage Tank (CST) into the RPV at high pressure and at a high rate. The HPCI is powered by a turbine that siphons steam from the main steam line and vents it into the wetwell, providing for both HPCI and RCIC. The valves controlling the flow of steam to the turbine are powered by onsite batteries, and HPCI and RCIC lose power and shut off when the batteries are no longer able to control the valves that feed the turbine. RCIC is similar to HPCI but injects water at a lower rate. LPCI functions along the same general mechanisms as the HPCI and RCIC, but pumps a much greater volume of water, can only function at a much lower

RPV pressure, and cannot be used without AC power. The SRVs are spring loaded valves that control the RPV pressure and vent excess steam into the wetwell. The ADS is a separate set of battery operated relief valves that are capable of rapidly depressurizing the RPV. This rapid depressurization is known as an RPV blowdown. Further discussion and diagrams of these safety systems can be found in Appendix A.

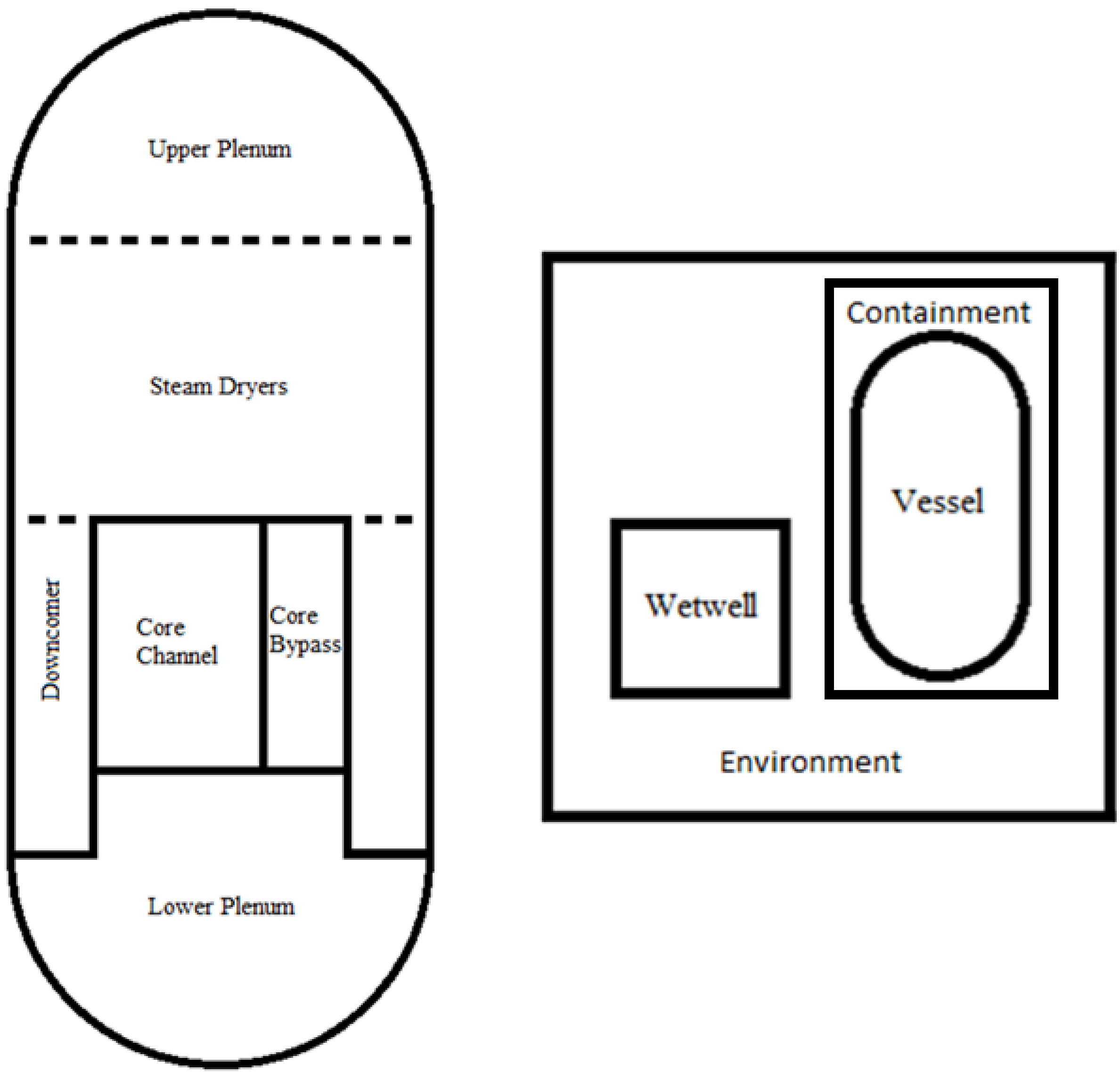


Figure 4. BWR MELCOR Model Nodalization

Figure 4 shows a diagram of the model nodalization used. It is worth noting that the simplicity of the model was a necessary choice made to allow for a model that would run fast enough that the thousands of runs of the MELCOR model to be executed in a practical amount of time. The model as executed typically produced a warp factor – the ratio of real world processor time used to execute the programmed model to the amount of problem time simulated – between 5 and 10, depending on the particulars of an individual run. With a warp factor between 10 and 20, a single run took anywhere from 7 to 14 hours. Each upgrade configuration examined was investigated by sampling and performing 200 separate MELCOR runs, demanding anywhere from 1,000 to 2,000 hours of processor time. With a goal of examining approximately thirty different potential upgrade configurations across three reactor power ratings, this meant that a total model runtime of between 90,000 and 180,000 CPU hours (or 3,750 to 7,500 CPU days) to was anticipated. Fortunately, with access to the INL High-Performance Computing (HPC) cluster, we were able to utilize close to 4,000 processing cores, allowing the modeling to be done in a reasonable amount of time – close to a week - rather than waiting several years for a humble 8-core personal computer to finish the task.

In addition to the actual physical specifications of the BWR, extensive control logic was implemented to accurately reflect when and how various safety systems of the BWR would engage. This included logic to reflect the technical specifications that dictated when the SRVs, ADS, HPCI, RCIC, and LPCI would actuate. The SRVs were set to begin opening when the RPV pressure exceeded 710 kPa, fully opening at 780 kPa. Once open, the SRVs began to close when the RPV pressure dropped below 720 kPa, fully closing at 630 kPa. HPCI and RCIC were set to engage when the core was below 12.1m above the bottom of the RPV, and to shut off when the water level reached 15m above the bottom of the RPV, or to turn on if the Drywell pressure exceeded 115 kPa. Additionally, HPCI was set to turn off 600 seconds into the transient and to remain off unless the RPV water level dropped below 11m above the bottom of the RPV, and both the HPCI and RCIC were programmed to shut off if the steam pressure in the RPV dropped below 115 kPa to reflect the reality that neither system can operate without sufficient pressure inside the RPV.

As well as these specifications, a series of relationships between the RPV pressure, wetwell temperature, pressure, and level, and the drywell temperature and level that control if and when the ADS depressurizes, and whether HPCI and RCIC draw from the CST or switch to drawing water from the PSP to prevent overfilling the RPV and PSP. Should the water level in the PSP increase too much, or the RPV pressure increase too much, the HPCI and RCIC will switch to drawing from the PSP. If the drywell temperature rises significantly, to the point that what water is in the wetwell is not sufficient to condense excess steam, blowdown will occur before the wetwell becomes incapable of controlling steam during the depressurization of the reactor. Similarly, in case of significant LOCA and drywell flooding, the reactor will depressurize to allow low pressure systems to engage and reflood the core. Lastly, if the wetwell level rises drastically without the wetwell pressurizing, blowdown will occur.

As well as physically modeling the RPV and Containment, their responses during a severe accident were modeled. Hydrogen production and possible ignition were modeled, as well as interactions between the RPV lower head and molten corium and interactions between molten corium and concrete. Additionally, lower head breach and corium mass transfer from the RPV to the containment was modeled, as well as potential aerosolization of radioactive contaminants and the possibility of containment rupturing, whether by hydrogen detonation or simple steam buildup overpressure, and releasing radionuclides to the environment.

To ensure that the model produces results that are a reasonable approximation of more detailed, realistic models, the developed model was benchmarked against the study performed at the Technical Research Centre of Finland by Tuomo Sevon (Sevon 2015). While there were differences in the results between the model, they showed good agreement for many of the in-core phenomena and the overall release mass. Model adjustment was necessary to account for inherent differences between the models and are discussed alongside the results of the benchmarking efforts.

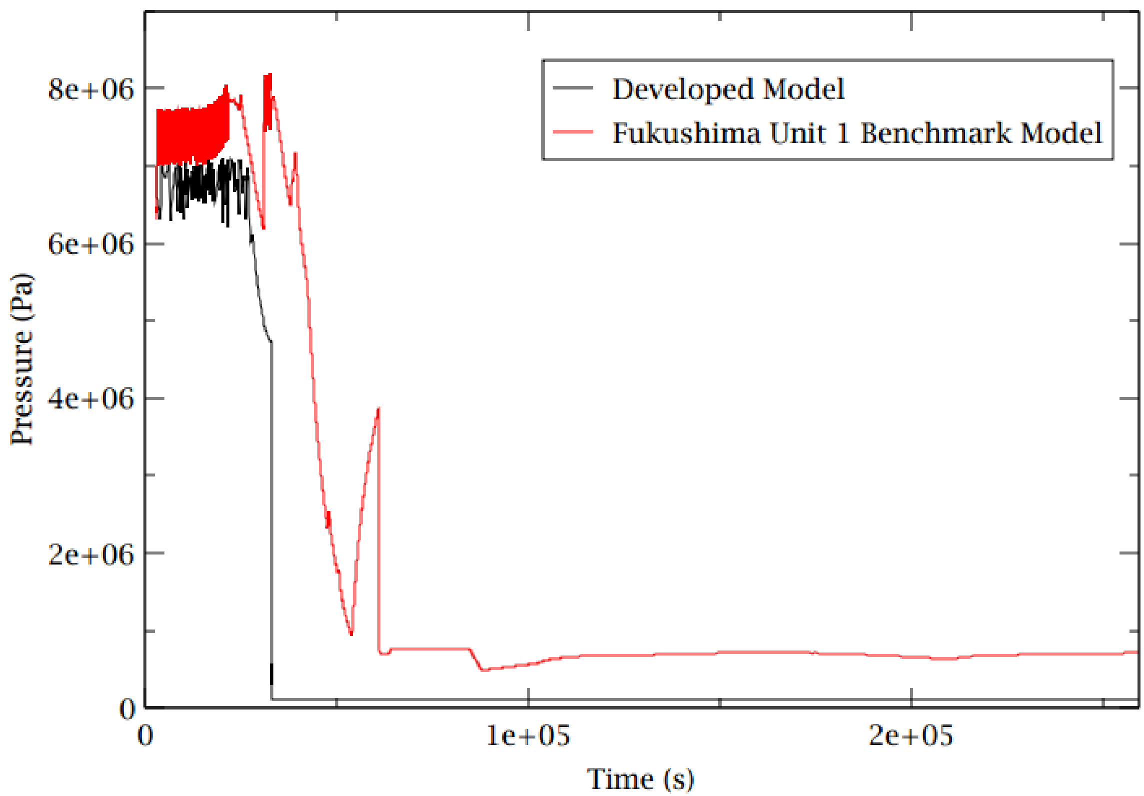


Figure 5 – RAVEN-MELCOR Model Benchmark against Sevon Model, RPV Pressure

The RAVEN-MELCOR model used different SRV setpoints for the RPV pressure (our model began opening the SRVs at 710 kPa and fully opened them by 780 kPa, the Sevon model did not begin to open the SRVs until 764 kPa and fully opened them by 800 kPa) but show reasonable agreement in the accident progression, otherwise, up to the point of core damage. In the Sevon model, the water drops below Top of Active Fuel (TAF) at 2 hours and 42 minutes and gap release begins at 3 hours and 54 minutes. In the RAVEN-MELCOR model benchmark, the water drops below TAF at 2 hours and 57 minutes, and gap release begins at 4 hours and 21 minutes. The only model adjustments that played a role in these results were to reduce the core power in the developed model – 1112 MWe is typical of an American BWR, but Fukushima Unit 1 had an electric power output of 460 MWe, and it would render the benchmarking exercise pointless if such a major difference in the plants being modeled were not accounted for.

Post core damage the results show significant divergence, due to the highly simplistic containment model used in this project. The Sevon model included modeling of a containment sump and pedestal that significantly affected the lower head failure speed, gas inerting in several volumes in the containment model, and modeling of the reactor building. These modeling differences caused divergence in the release timing, but there was reasonable agreement in the release mass at 72h after LOOP conditions.

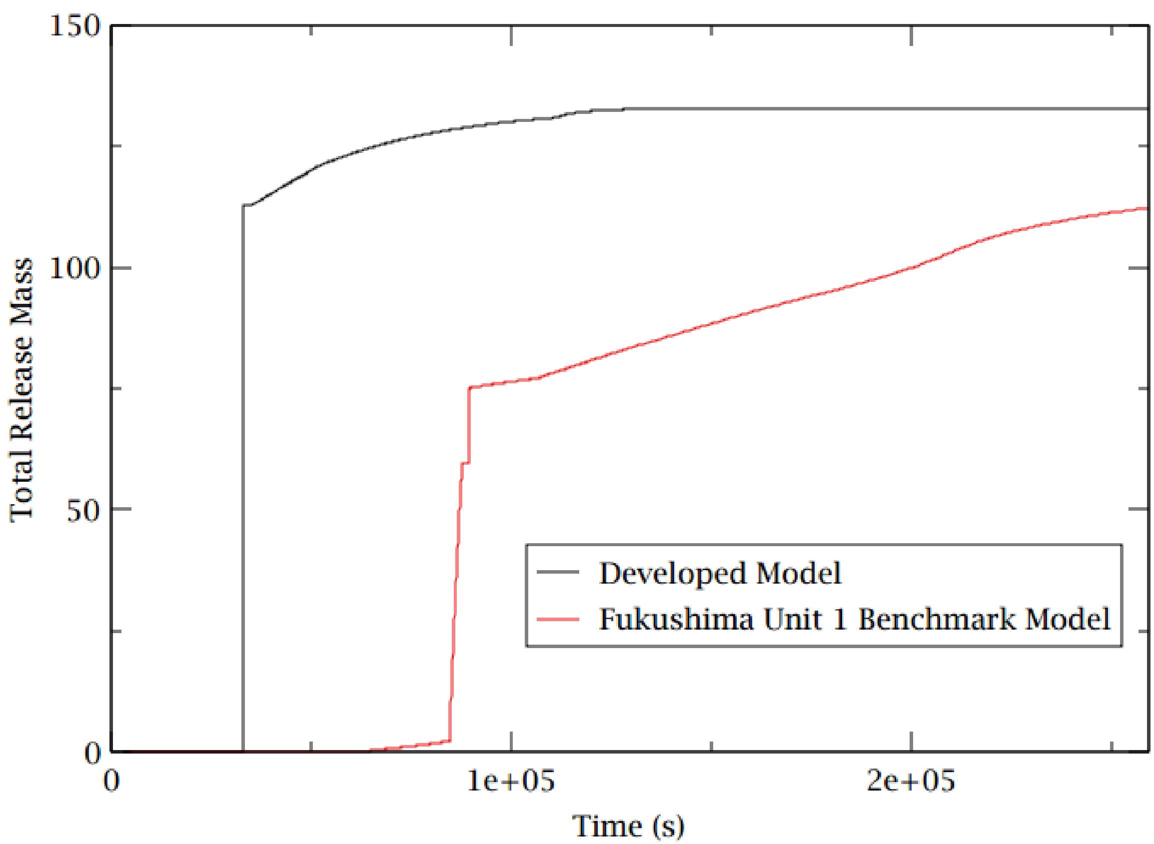


Figure 6 – RAVEN-MELCOR Model Benchmark against Sevon Model, Release Mass

In the Sevon model, the lower head failure time was at 14 hours, 6 minutes. In the developed model, the lower head failure time was at 10 hours. The more complex lower head model used in the Sevon model was less overly conservative with regards to heat transfer from the lower head to surrounding materials, resulting in a significant delay in lower head failure compared to the developed model. Additionally, the Sevon model used a heavily gas inerted wetwell and drywell, while the developed model did not assume such protections, resulting in a much earlier hydrogen deflagration time and earlier containment failure time in the developed model than the Sevon model. In the Sevon model, the first hydrogen deflagration was at 24 hours, 50 minutes after the initial LOOP conditions, and in the developed model the first hydrogen deflagration was at 10 hours, 8 minutes. Similarly, the containment ruptured at these times. Despite the major difference in release timing due to differing modeling assumptions, the release mass at 72 hours for the developed model was 132 kg, and the release mass at 72 hours for the Sevon model was 112 kg, showing reasonable agreement, to within 20%.

Showing that the MELCOR model developed is reasonably representative of a BWR's response to an LT-SBO is an important step in the research project. By demonstrating that the model responds in a manner that holds reasonable agreement with more complex, well accepted models that closely match real life accident scenarios, we demonstrate support for the idea that the model will respond to alterations, whether via power plant upgrades or stochastic parameter perturbations, in a manner that will also hold reasonable agreement with more highly realistic models. With a model that will appropriately respond to changes in the accident scenario in a manner reflective of reality, we can use stochastic parameter analysis with the model to explore the impact of the upgrades being examined.

### 4.3. *NPP Upgrade Modeling*

NPP upgrades to be modeled fell into two general categories – pressure mitigation and hydrogen mitigation. During a severe accident, hydrogen generation leading to a deflagration or detonation can pose a catastrophic threat to containment integrity. To explore options to combat this threat, the impact of hydrogen igniters and PARs on severe accident progression was assessed by modeling accident scenarios with and without hydrogen igniters and PARs, both by themselves and in combination with other NPP upgrades chosen for examination. As well as hydrogen buildup, pressure from non-condensable gases and uncondensed steam can build up and rupture containment. To evaluate potential upgrades for preventing this possibility, severe accident progression was modeled with and without hardened ventilation systems and filtered hardened ventilation systems.

To model Hardened Ventilation Systems and Hardened Passive Filtration Ventilation Systems, a flowpath of appropriate size was added connecting the containment control volume to the exterior environment boundary condition volume. A radionuclide filter is applied to this flowpath to model the potential filtration upgrade that can be applied to the containment ventilation system. The flowpath is programmed to open when the pressure in containment begins to build up beyond 25 kPa above the outside environment to model operational and automatic responses to pressure buildup. The flowpath closes if pressure in containment drops below 25 kPa above the outside environment. Without the Hardened Ventilation System or Hardened Passive Filtration Ventilation System, the flowpath simply never opens, as it is assumed that the hot and wet conditions in containment will cause it to fail, as shown possible by the Fukushima Daiichi disaster. To model containment response to a failure to mitigate pressure buildup, a second flowpath was added to the model connecting the containment control volume to the exterior boundary condition volume that is programmed to open if the pressure in containment exceeds the exterior environment pressure by 125 kPa or more, representing a breached containment. This flowpath is not filtered and, if opened, never closes, representing the conditions of a breach in containment.

To model Hydrogen Igniters and PARs for the prevention of hydrogen buildup and uncontrolled ignition leading to a deflagration or detonation, MELCOR's internal models were used. MELCOR models Hydrogen Igniters by lowering the $H_2$ mole fraction required for ignition in any volumes marked as having operational Hydrogen Igniters. Lowering the $H_2$ mole fraction required for ignition reduces the amount of hydrogen that can build up, reducing the size of any potential hydrogen burns that are possible. In reducing the size of the burn, the speed of the burn and the pressure increase created as a result of the burn are similarly reduced, relieving stress placed on the containment vessel and protecting its integrity as a pressure boundary and barrier to radionuclide release. To model the Hydrogen Igniters power needs, the igniters are programmed to fail to function if power is not available, like in an SBO scenario. If power is restored, Hydrogen Igniter functionality is also restored.

The internal PAR model is simple and based on the stoichiometry of the hydrogen-oxygen reaction present. The reaction rate for a single PAR unit is based on the hydrogen density of gas entering the PAR, the total gas volumetric flow rate through the PAR unit, the reaction efficiency of the PAR unit. The total gas volumetric flow rate through the PAR unit depends on the mole fraction of hydrogen and pre-programmed multiplier and exponent constants that depend on PAR unit design parameters. The design parameters used are those of a type of PAR unit developed by the NIS Company (Sandia National Laboratories 2011). Additionally, the number of PARs installed can be adjusted, and the containment vessel was modeled as having 55 PAR units installed. As PARs do not require power or operator action to function, there is no programmed logic to disable them due to operator inaction or loss of electric power.

A limitation of the model, with regards to hydrogen production, relocation, accumulation, and ignition is that the wetwell and drywell models are both, in a word, simplistic. Using one volume for the drywell and one for the wetwell means that the hydrogen concentration in either volume is homogenous, when both volumes are large enough for variations in hydrogen concentration – particularly near the top of the drywell – to exist and impact the progression of the accident with regards to hydrogen. A more detailed model would include multiple volumes and appropriate flowpaths to represent these volumes and to allow for hydrogen's lighter-than-air density to cause it to rise and accumulate at the top of these volumes.

To model a system akin to the AP1000 In-Containment Refueling Water Storage Tank (IRWST), we chose to examine a tank of water inside containment that was of comparable size to the AP1000 IRWST. The AP1000 IRWST was reported to be 590,000 gallons in capacity (USNRC 2007), or 2,233 $m^3$. The AP1000 IRWST equivalent upgrade was modeled as equivalent to a cylinder with diameter equal to its height, giving it a height of just over 14 meters, with a 16" diameter pipe leading from the IRWST to the upper head of the RPV. To give sufficient pressure head for gravity to drive water into the RPV, the bottom of the tank was set to 20 meters above grade, 5 meters above the top of the RPV. The IRWST was modeled at 305 K and the valves between the IRWST and the upper head of the RPV were set to open when the pressure inside the RPV dropped below three atmospheres.

Finally, during the course of testing the model, it became apparent that sometimes the ADS would not actuate automatically until after DC power had failed. This led to scenarios where the depressurization of the RPV when the lower head failed would eject enough hydrogen into the containment to overwhelm the PARs and for a hydrogen deflagration to occur and rupture the containment despite the presence of the PARs. With this in mind, we chose to model a procedural upgrade in which the ADS was manually actuated 4 hours into a Station Blackout, ensuring, when this upgrade was implemented, that the reactor would be depressurized before the potential loss of DC power.

The Accident Tolerant Fuels implementation was to alter the minimum hydrogen generation temperature sensitivity coefficient within MELCOR. Many of the unique characteristics of Zircalloy cladding are integrated into MELCOR's fuel degradation and relocation models and could not be altered to create a more real-world appropriate representation of the full range of impacts of the ATF upgrade, so a best effort attempt at representing the impact of the ATF upgrade on hydrogen generation was made with this sensitivity coefficient.

### 4.4. *LT-SBO Case Study*

The expected progression of a LT-SBO transient is that when Loss of Offsite Power (LOOP) condition occurs, numerous things will immediately happen. In no particular order, the Reactor Protection System will SCRAM the reactor, inserting the control rods and halting the chain reaction, the Main Steam Isolation Valves (MSIVs) will close, isolating the turbine from the reactor, and the EDGs will engage and maintain plant AC power. Using the plant AC power, the Residual Heat Removal (RHR) system will remove core decay heat and keep the core at a stable and safe configuration. Meanwhile, it is expected that the grid owner will begin attempts to repair offsite AC power.

If the EDGs do not fail before offsite AC power is repaired, then the LOOP condition ends without ever transitioning into a Station Blackout (SBO) conditions and, barring other malfunctions, plant safety will be maintained. However, it is assumed that at some point, the EDGs will fail and the plant will enter SBO conditions. At this point, AC power is lost, and the plant will switch over to HPCI and RCIC to maintain core cooling, as these are operable using only battery power and steam generated by the core. The Safety Relief Valves are also operable using battery power and will be used to maintain RPV pressure at a safe level by opening when the RPV pressure rises too far and closing after it has decreased back to the closure pressure setpoint. It is also expected that plant operators will immediately begin efforts to recover EDG function. If either plant operators recover EDG function or the grid owner recovers offsite AC power capabilities before core damage occurs, it is expected that the scenario can be recovered safely through the use of AC powered safety systems. The scenario studied here, however, is a long-term SBO in which the AC Power is not recovered for a lengthy period of time.

From SBO conditions, the scenario is expected to progress in one of two ways, depending on whether battery power is depleted before the PSP heat limits are reached or not. If the heat limits of the PSP are reached before battery power depletes, the ADS will be activated, and blowdown will occur. The RSVs will open, venting an enormous amount of steam to the wetwell to be condensed by the cold inventory of the wetwell, depressurizing the reactor. Because of the low pressure in the RPV, HPCI and RCIC become inoperable. Under normal conditions, LPCI would become operable with the lowered RPV pressure, but with the loss of AC power, LPCI is inoperable. With no LPCI to reflood the core with cold water, the diminished core inventory will soon boil away, uncovering the core and leading to core damage.

If battery power depletes before the heat limits are reached, then HPCI, RCIC, and the ADS all stop functioning. Without battery power, the valves that open and close to regulate and provide steam to the secondary turbine that powers HPCI and RCIC are rendered inoperable, rendering HPCI and RCIC non-functional. Similarly, the valves that comprise the ADS cannot be opened without battery power, rendering the ADS inoperable. Without high pressure core makeup or the ability to depressurize, the core contents will boil away, the fuel will uncover, and core damage will ensue.

If sufficient molten corium is present and no way to cool it arises, the lower head of the RPV will eventually be breached and molten corium will begin to attack the containment base-mat. If the molten corium can be cooled before the RPV is breached, lower head integrity can be maintained, preventing corium-concrete interaction and the hydrogen generation that entails. It is assumed that any ventilation system will have its own independent supply of power, such as a DC battery power supply separate from the batteries that provide DC power for the ADS and HPCI/RCIC and will retain functionality even without AC power and after the main battery power supply fails. Similarly, in cases where PARs are being evaluated, they will function regardless of AC power or any lack thereof. If steam pressure buildup and hydrogen accumulation in the containment can be controlled, radionuclide release can likely be prevented. If radionuclide release occurs, the magnitude of the release is the metric by which offsite consequences are evaluated.

### 4.5. *Stochastic Parameter Analysis*

To evaluate the impact of an NPP safety upgrade on accident scenario progression and outcomes, stochastic parameter analysis is used. By randomly sampling a number of stochastic parameters important to the outcome of a severe accident scenario and running a variety of accident scenario models with these randomly sampled stochastic parameters, the probability and consequences of a particular variation of a severe accident scenario, with and without upgrades, can be analyzed.

The stochastic parameters sampled are the EDG failure time, the Battery Power lifetime, the wetwell initial temperature, the wetwell initial level, and the containment failure pressure. The EDG failure time strongly influences the decay heat of the scenario; the reactor is assumed to SCRAM at the onset of LOOP conditions and that plant is assumed to remain stable until the onset of SBO conditions. Because of this, any time between the onset of LOOP conditions and the onset of SBO conditions will reduce the amount of decay heat produced at the onset of SBO conditions. The wetwell initial temperature and level both can significantly influence the timing of when the wetwell heat absorption temperature and pressure limits are reached, which can have major repercussions for the overall progression of the accident scenario. Finally, the containment failure pressure can dictate whether the containment actually fails or not in near-miss scenarios, potentially entirely preventing the release of radionuclides even without ventilation systems. The stochastic parameters examined are shown below in Table 2, alongside the range of values considered for each stochastic parameter.

Table 2. Stochastic Parameters analyzed

| **Stochastic Parameter** | **Parameter Range Considered** |
|---|---|
| Diesel Generator Failure Time | 0 hours – 24 hours |
| Battery Power Lifetime | 4 hours – 6 hours |

| | |
|---|---|
| Wetwell Initial Level | 7 m – 15 m |
| Wetwell Initial Temperature | 285 K - 315 K |
| Containment Failure Pressure | 341.6 kPa – 512.4 kPa |

As well as the stochastic parameters examined, the impact of a Reactor Power increase was assessed. The scenario was examined at 100%, 110%, and 120% of nominal reactor power.

To perform all of these analyses, RAVEN was used to generate and execute thousands of randomly sampled MELCOR runs to examine the impact of the stochastic parameters in great detail. Within each run, the stochastic parameters are sampled with uniform probability, to ensure that not only the likely scenarios are examined, but also the unlikely scenarios that would not necessarily be sampled if the sampling were done according to the probability by which the scenarios are eventually weighted. Once a particular set of parameters has been sampled, they are plugged into a MELCOR input deck, the model is executed, and the desired points of output data are retrieved from the MELCOR output file and written down, alongside the full set of stochastic parameters that correspond to that output, in a CSV file for easy retrieval and analysis.

### 4.6. *Stochastic Parameter PDF Variation Sensitivity*

A major input to RAVEN is the PDF of each stochastic parameter being sampled. As well as the PDFs used in the initial primary analysis, alternative PDFs were examined to study the impact of change in the probabilistic weighting of data points. A strength of the RAVEN code is that it is able to randomly sample an appropriate number of sets of input to satisfy requirements, run the requested MELCOR models, and then reuse the output data acquired. Feeding the condensed output data back into RAVEN allows for the data set to be reexamined with different stochastic parameter PDFs without requiring the computational expenditure of rerunning all of the MELCOR models. The PDFs used in the main analysis are below in Table 3.

Table 3. Stochastic Parameter Primary Probability Distribution Functions

| **Stochastic Parameter** | **Distribution** |
|---|---|
| Diesel Generator Failure Time (h) | Exponential, $\lambda = 0.1919\ h^{-1}$ |
| Battery Power Lifetime (h) | Triangular, min = 4, peak = 5, max = 6 |
| Wetwell Initial Level (m) | Triangular, min = 7, peak = 11, max = 15 |
| Wetwell Initial Temperature (K) | Triangular, min = 285, peak = 300, max = 315 |
| Containment Failure Pressure (kPa) | Triangular, min = 341.6, peak = 427, max = 512.4 |

The Diesel Generator Failure Time distribution was fabricated with the goal of examining an Exponential function wherein there was a 99% chance of the EDGs failing within 24 hours of the onset of LOOP conditions. The Battery Power Lifetime distribution was obtained from prior work with Drs. Diego Mandelli and Curtis Smith at INL (Mandelli et al. 2013). The Containment Failure Pressure distribution was created using the design pressure of a BWR Mk1 Containment (427 kPa) and examining a triangular distribution for the failure pressure from 80% to 120% of the nominal containment design pressure. During the literature survey, little data was found for the specific details of the Wetwell Initial Level and Wetwell Initial Temperature, and the research team exercised engineering judgement, based on prior work experience regarding Station Blackouts in BWRs with Mk1 Containments (Mandelli et al. 2015), to create a reasonable and realistic PDF for these stochastic parameters.

The nominal Containment Failure Pressure of a BWR Mk1 Containment was pulled from a May 22, 2012 USNRC briefing to the Advisory Committee on Reactor Safeguards on Filtered Containment Venting Systems (USNRC 2012) and was found to be 427.0 kPa. The Containment Failure Pressure was assigned a triangular PDF that ranged from 80% of the nominal failure pressure, 341.6 kPa, to 120% of the nominal failure pressure, 512.4 kPa, with the peak of the PDF at the nominal failure pressure.

Many of the stochastic parameter distributions here are simple triangular distributions about a nominal value, with the distributions assumed to be as such for the purposes of demonstration. A study attempting to draw practical conclusions about a specific power plant would need to draw upon operational experience and reliability data, such as the wealth of information in NUREG/CR-6890 (USNRC 2005a). NUREG/CR-6890 has hundreds of pages of data on LOOP occurrences alone, before even beginning to plumb the depths of the intricacies of a full blackout. Nominal battery lives and emergency power configurations for various US power plants are also listed and analyzing a particular plant would require consulting this document, among others, to get a detailed look at the intricacies of the plant's response to LOOP and SBO conditions. For operator performance, particularly with regards to the plant operational procedures that would affect the PSP Initial Level and Temperature, the DC Lifetime, and the Containment Failure Pressure, human reliability analyses, like the SPAR-H method, would need to be consulted. Detailed documentation of the SPAR-H human reliability analysis method can be found in NUREG/CR-6883 (USNRC 2005b). The SPAR-H method includes models for plant maintenance schedules, expected operator responses during an accident scenario, and the potential for repair and recovery of failed equipment during an accident scenario. Drawing upon decades of operational experience and extensive component testing allows analysts to create reasonable and defensible failure rates and PDFs for real-world practical application and would be necessary to use this methodology to analyze a particular power plant.

To examine the impact of variation in the PDFs, alternate sets of PDFs were used in further examinations of the accident scenario. Two other sets of PDFs were used. The first set of PDFs was a set of triangular distributions with the apex of the distribution for each stochastic parameter at the expected worst possible value, based on the results of the initial examination of each stochastic parameter individually. The seconds set of PDFs was the inverse, a set of triangular distributions with the apex of the distribution for each stochastic parameter at the expected best possible value.

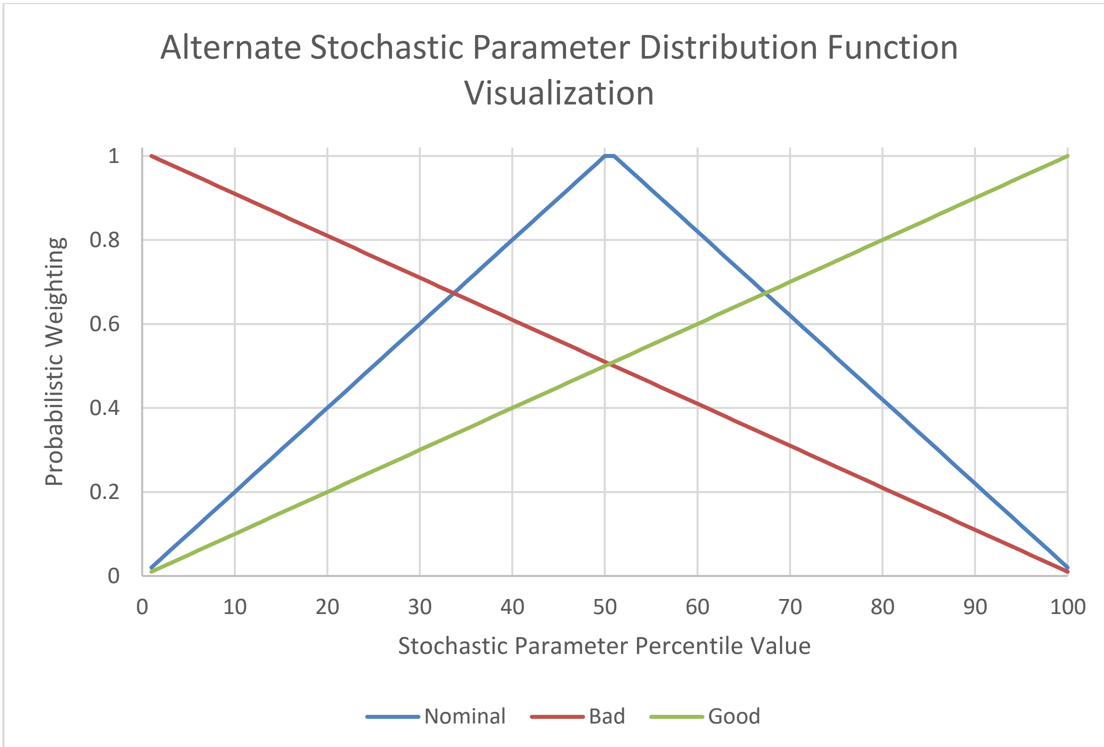


Figure 7 - Alternate Stochastic Parameter Distribution Function Visualization

The expected best-case and worst-case values for each stochastic parameter were determined in the initial single stochastic parameter sensitivity studies.  In these studies, each stochastic parameter was scrutinized by pinning each other stochastic parameter to a single value, and the parameter being scrutinized was varied and its impact examined. The values for which the Release Activity trended downwards were considered to be the good values, and the values for which the Release Activity trended upwards were considered to be the bad values. For example, for EDG Failure Time, a later EDG Failure Time was associated with a lower Release Activity, so the good values were decided to be the longest EDG Failure Times possible. Figure 7 shows, for the stochastic parameters where the nominal PDF was a triangular distribution and the low value was the worst-case scenario, a visual comparison between the nominal PDF, the “Bad” PDF, and the “Good” PDF. Though it cannot be represented visually, extrapolating this general idea across five dimensions of stochastic parameter PDFs is the method by which the alternate stochastic parameter PDFs were evaluated. By altering the stochastic parameter PDFs drastically, we can examine some of the larger potential changes in the Release Activity expected value and CDFs that can happen without resorting to PDFs with singularities or extremely sharply peaked curves. Varying the stochastic parameter PDFs significantly and examining the resulting impacts on the release activity expected values and CDFs allows us to examine how sensitive the methodology is to broad, untargeted changes in stochastic parameter PDFs.

### 4.7. *Probabilistic Consequence Evaluation*

RAVEN was used both to generate stochastically sampled parameters and to perform data analysis of the results from the MELCOR models run. Multiplying the probability of any given particular scenario occurring with the offsite consequences of that particular scenario allows us to evaluate the risk posed by that specific variety of an accident scenario. Then, by comparing the risk posed by an accident scenario with various NPP upgrades (or lack thereof) the probabilistic efficacy of the upgrade can be established. To illustrate the principles behind this analysis more clearly, what follows is a discussion of a constructed example problem and analysis.

The illustrative problem uses two stochastic parameters and 1,000 value bins for these parameters, ranging from 1 to 1000. As part of the analysis, we assign each stochastic parameter value a probability of occurrence. The probability of any given stochastic parameter combination is simply the probability of each selection multiplied together, or P(X) * P(Y). The demonstration problem uses a simple bell curve truncated at 3.33σ for both parameter's probability distribution functions. It is worth noting that for the purposes of analyses, it is advantageous to sample points using a flat distribution, to better capture outlier scenarios that might otherwise be overlooked. The probability of the scenario occurring can be factored in as part of the "weight" of the point.

In this demonstration problem, the consequences of each sampled scenario are evaluated as (2X+Y)^2, where X and Y are the sampled parameters. This method of deciding the consequences of a chosen scenario has no real-world implications and was chosen to be both non-linear and also very simple to solve. With the consequences and probability in hand, evaluating the risk of any particular scenario is easily done. Given the simplicity of the "model" used, it would be easy to simply use a spreadsheet to calculate the risk of every scenario, sum them appropriately, and apply the effects of upgrades as desired. However, the actual research project described in this document uses Monte-Carlo sampling to select the scenarios to be evaluated. RAVEN would be overkill for this simple application, so a simple random sampling function was used to generate 1000 scenarios to use as the evaluation basis. Figure 8 is a simple scatter graph of the sampled scenarios, to visually show the arrangement of all the scenarios. While there is some variation in the density of sampled points due to the nature of random sampling, a sample size of 1000 samples appears to be enough to generate a statistically significant sample size that will appropriately reflect the nature of the underlying problem, while only requiring 1/1000$^{th}$ of all possible cases to be evaluated.

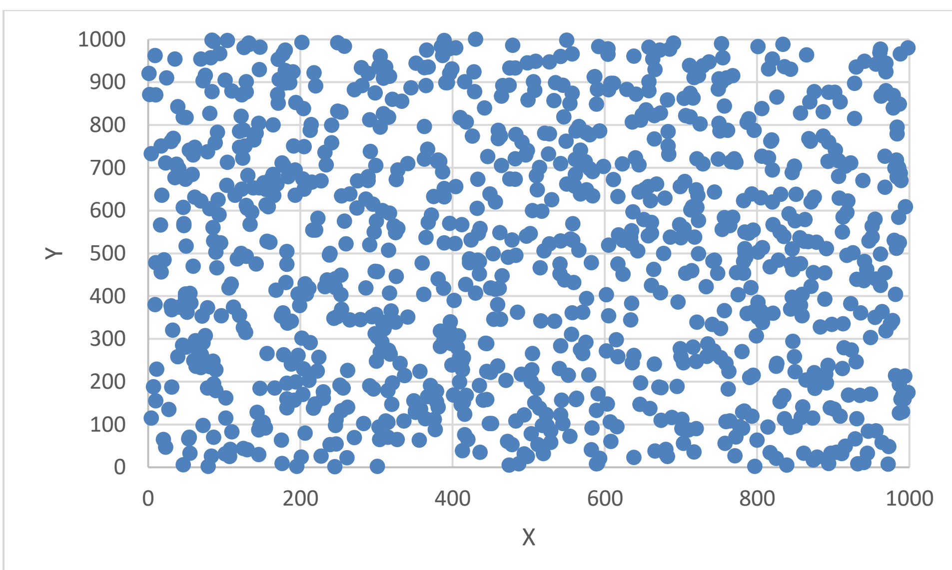


Figure 8. Example problem sampled parameters, graphed

From here, we calculate the risk by multiplying the probability and consequences of each scenario. The total magnitude of this number is somewhat meaningless, as it has no units or scale, but is of worth for comparative analysis – we can evaluate what ‘areas’ of the problem dominate the risk presented.

Table 4. Demonstration problem initial results lumped by area of the sample space

| | | | | |
|---|---|---|---|---|
| Y | 701-1000 | 41.08 | 240.88 | 46.26 |
| | 401-700 | 201.18 | 1090.47 | 298.69 |
| | 1-400 | 52.16 | 224.52 | 71.84 |
| | | 1-400 | 401-700 | 701-1000 |
| | | | X | |

Because of the very high probability of sampling between 0.4 and 0.7 for either parameter, these samples dominate the risk posed across all scenarios. Additionally, large values for X and/or Y generally posed greater risk than small values for X and/or Y. It is also worth noting that the small values are binned together in greater numbers than the medium or large bins – the small bin for each parameter contains 400 possible values where the medium and large value bins only contain 300 possible values. To address the risk of these scenarios, six upgrades are proposed. Upgrade A reduces the consequences of all scenarios by 20%, Upgrade B reduces the consequences of scenarios for which X is between 1 and 400 by 100%, Upgrade C reduces the consequences of scenarios for which Y is between 1 and 400 by 100%, Upgrade D reduces the consequences of scenarios for which both X and Y are between 1 and 400 by 35%, Upgrade E reduces the consequences of scenarios for which X is between 701 and 1000 by 50%, and finally, Upgrade F reduces the consequences of scenarios for which Y is between 701 and 1000 by 50%. The targeted impact of the upgrades is tabulated below in Table 5, along with the benefit from risk reduction of each upgrade.

Table 5. Demonstration problem proposed upgrades

| | Parameters Affected | Consequence Reduction | Risk Reduction |
|---|---|---|---|
| Upgrade A | All | 20% | 0.2 |
| Upgrade B | X = 1 - 400 | 100% | 0.130 |

| | | | |
|---|---|---|---|
| Upgrade C | Y = 1 - 400 | 100% | 0.154 |
| Upgrade D | X&Y = 401 - 700 | 35% | 0.168 |
| Upgrade E | X = 701 - 1000 | 50% | 0.092 |
| Upgrade F | Y = 701 - 1000 | 50% | 0.072 |

Upgrade A was the most effective upgrade, even though it provided the smallest reduction in consequences, because of the sheer number of upgrades it targeted. The second most effective upgrade was Upgrade D. It targeted far fewer scenarios, but these scenarios were much more likely to occur – the chance of any particular scenario sampled using the bell-curve probability distribution function (PDF) meeting the criterion for Upgrade D was 43.1%, even though these scenarios made up only 9% of the total sample space. Additionally, between Upgrades B and C, Upgrade C was slightly more effective as targeting low values of Y allowed X to be potentially large, increasing the consequences mitigated as the formula for consequences was (2X+Y)^2. Similarly comparing between Upgrades E and F, Upgrade E was more effective specifically because it targeted large values of X.

As well as evaluating the upgrades individually, the methodology used in this project can be used to evaluate combinations of upgrades. With that in mind, the upgrades in this demonstration problem were also evaluated in several configurations of multiple upgrades. For this evaluation, any time multiple upgrades affected the same scenario, only the effects of the most potent upgrade were applied. The results of this evaluation are shown below in Table 6.

Table 6. Upgrade combination evaluation

| Upgrades | Risk Reduction | Summed Risk Reduction of Original Upgrades | Combination Efficiency |
|---|---|---|---|
| A and D | 0.27 | 0.37 | 0.74 |
| B and C | 0.26 | 0.28 | 0.92 |
| B and E | 0.22 | 0.22 | 1 |
| B and F | 0.19 | 0.20 | 0.96 |
| C and E | 0.23 | 0.25 | 0.94 |
| C and F | 0.23 | 0.23 | 1 |
| E and F | 0.15 | 0.16 | 0.94 |
| B, C, E, and F | 0.39 | 0.45 | 0.87 |

| | | | |
|---|---|---|---|
| B, C, D, E, and F | 0.56 | 0.62 | 0.91 |
| All | 0.56 | 0.82 | 0.68 |

The most impactful configuration was a tie between combining all the upgrades and all of the upgrades barring Upgrade A. Upgrades B through F, when combined, cover all possible scenarios, making the addition of Upgrade A strictly wasteful unless we attempt to account for the upgrades having a chance of failing. The most efficient upgrade combinations were B/E and C/F, as these upgrades have no overlapping scenarios. Adding upgrade D to either of these pairings would continue to improve the risk reduction without compromising efficiency, as Upgrade D has no overlap with either of these upgrade pairings.

Lastly, for the sake of a cost-comparison, we assume the upgrades all have an equal cost. For any single upgrade, Upgrade A is the most cost-efficient upgrade to implement. If more risk reduction is desired than Upgrade A can provide alone, Upgrades A and D together represent the most potent choice evaluated. However, if three or more upgrades are desired to be implemented, Upgrades B, C, and D together appear to be an ideal choice – Upgrade A combines poorly with other upgrades compared to Upgrade D, and Upgrades B and C pair well without any overlap with Upgrade D.

### 4.8. *Severe Accident Consequence Evaluation*

An important and necessary piece of the demonstration problem detailed in Section 4.6 is the evaluation of consequences so that risk, and thus risk reduction, can be calculated. In this project, we assume the plant cannot be recovered after a severe accident, as plant operation has never been recovered at a commercial nuclear power plant after a severe accident. With this in mind, it is not necessary to calculate the cost of damages to the plant incurred by the severe accident, as the cost will remain constant with or without NPP upgrades to mitigate the offsite consequences of a severe accident. As the impact of the upgrades is entirely aimed at reducing the offsite consequences of a severe accident and the onsite consequences are expected to remain constant regardless of the upgrades, only the offsite consequences will be evaluated for analysis of the risk reduction benefit presented by implementing the proposed NPP safety upgrades.

To analyze the offsite consequences, a simple metric of the radionuclide release source term is used. There is support in the literature that this is a reasonable metric by which to evaluate the offsite consequences of a radionuclide release from a severe accident, as the economic consequences of a severe accident were found to scale very linearly with the radionuclide release source term of an accident (Silva et al. 2014). A full Level 3 PRA using a code like OSCAAR or MACCS is beyond the scope of this project. Using the radionuclide release source term in lieu of such an analysis greatly reduced the amount of work necessary for this project without significantly degrading the results obtained.

MELCOR is capable of tracking the movement of radioactive materials by the chemical class of the material. By default, MELCOR tracks classes as Barium, Iodine, Tellurium, Ruthenium, Cesium, Lanthanum, Uranium Oxide, Cadmium, Silver, Cesium Iodide, and Cesium Molybdate. To evaluate the activity of the release, control functions were implemented within RAVEN to track the total mass of radioactive material released to the environment. Using values from the MelMACCS software package from SNL for the conversion of MELCOR chemical groups to MACCS input activities for a typical BWR we were able to convert these mass values to appropriate measurements of the activity of each element, and to sum these figures to obtain a calculation of the overall release activity. The values, listed in Curies per gram of release, are listed below in Table 7.

Table 7. Release Mass-Activity Conversion Factors

| Chemical Group | Curies per gram of release |
|---|---|
| Xenon | 4.604429 |
| Cesium | 2.060442 |

| | |
|---|---|
| Barium | 17.53105 |
| Iodine | 222.0967 |
| Tellurium | 32.10857 |
| Ruthenium | 9.499643 |
| Molybdenum | 11.92256 |
| Cerium | 10.85798 |
| Lanthanum | 9.772958 |

4.9. *RAVEN Data Analysis*

RAVEN assigns all data points a probabilistic weighting for use in many statistical calculations. The weighting of a particular point of data is calculated as follows

$$w = \prod_{i=1}^{m} pdf_i(x)$$

Where m is the number of stochastic parameters being sampled, n is the number of sample points, and $pdf_i(x)$ is the value of the PDF at the value sampled for that stochastic parameter. By multiplying, in our case, the probability for all 6 stochastic parameters together we can find, without normalization, the individual weight that a particular sampling should be assigned. By summing the non-normalized weights of all points together and dividing each non-normalized weight by this summed value, we can normalize the weight of each point such that the weights will sum to 1.

To calculate the Expected Value of a desired output variable, RAVEN simply does a weighted summation of the value of that variable for every run, as follows

$$\bar{x} = \frac{1}{V_1} \sum_{j=1}^{n} w_j x_j$$

Where

$$V_1 = \sum_{j=1}^{n} w_j$$

The Variance, $\sigma^2$, and Standard Deviation, $\sigma$, of a variable are measures of how spread out the data is. RAVEN calculates the Variance and Standard Deviation of a sampling as follows

$$\sigma^2 = \frac{1}{n}\sum_{j=1}^{n} w_j\left(x_j - \bar{x}\right)^2$$

to create a measure of variance that is unbiased with respect to sample size, RAVEN does the following operation

$$\sigma_{corrected} = \frac{V_1^2}{V_1^2 - V_2}\sigma_{uncorrected}$$

Where

$$V_2 = \sum_{j=1}^{n} w_j^2$$

The Skewness, $\gamma$, is a measurement of the asymmetry of the distribution of a variable about its mean, or expected value. For a positive skewness, the distribution "leans" left of the mean value towards smaller values, and for a negative skewness, the distribution "leans" to the right of its mean, towards greater values. RAVEN calculates the Skewness as follows

$$\gamma_1 = \frac{\frac{1}{V_1}\sum_{j=1}^{n} w_j\left(x_j - \bar{x}\right)^3}{\frac{1}{V_1}\sum_{j=1}^{n} w_j(x_i - \bar{x})^{3/2}}$$

RAVEN performs a correction to get a Skewness calculation that is unbiased with regards to sample size

$$\gamma_{1,corrected} = \frac{V_1^3}{V_1^3 - 3V_1V_2 + 2V_3}\gamma_{1,uncorrected}x \frac{1}{\left(\frac{V_1^2}{V_1^2 - V_2}\right)^{3/2}}$$

Where $V_1$ and $V_2$ are as previously described and $V_3$ is calculated as

$$V_3 = \sum_{j=1}^{n} w_j^3$$

The Kurtosis is a measure of how peaked the distribution is. It is defined in a way such that a Normal distribution will give a Kurtosis of 0. A positive Kurtosis indicates that the distribution is sharply peaked, a negative Kurtosis indicates a flatter top to the distribution. RAVEN calculates the Kurtosis as

$$\gamma_2 = \frac{\frac{1}{V_1}\sum_{j=1}^{n} w_i(x_i - \bar{x})^4 - 3(\frac{1}{V_1}\sum_{j=1}^{n} w_i(x_i - x)^2)^2}{(\frac{1}{V_1}\sum_{j=1}^{n} w_i(x_i - \bar{x})^2)^2}$$

As with other calculations, to unbias the Kurtosis calculation with regards to sample size, RAVEN does the following

$$\gamma_2, corrected = \frac{V_1^2(V_1^4 - 4V_1V_3 + 3V_2^2)}{(V_1^2 - V_2)(V_1^4 - 6V_1^2V_2 + 8V_1V_3 + 3V_2^2 - 6V_4)}\sum_{j=1}^{n} w_i(x_i - \bar{x})^4 -$$

$$\frac{3V_1^2(V_1^4 - 2V_1^2V_2 + 4V_1V_3 - 3V_2^2)}{(V_1^2 - V_2)(V_1^4 - 6V_1^2V_2 + 8V_1V_3 + 3V_2^2 - 6V_4)}\sum_{j=1}^{n} w_i(x_i - \bar{x})^2$$

Where $V_1$, $V_2$, and $V_3$ are as before, and $V_4$ is calculated as

$$V_3 = \sum_{j=1}^{n} w_j^4$$

The Median of a distribution is the number separating the greater half from the lower half of all possible values. Formally, RAVEN calculates the median of a distribution as the number that satisfies

$$P(X \leq m) = P(X \geq m) = \frac{1}{2}$$

RAVEN's percentile calculations are similar, in some ways, to the Median calculation – the Median is the 50th percentile. A percentile is the number below which a requested percentage of the observed data group falls. Formally, RAVEN calculates a requested percentile value Z as the number that satisfies

$$P(X \leq m) = \frac{Z}{100}$$

Informally, the Covariance of two variables is how much the two variables tend to correlate in value. A positive Covariance indicates that as one variable increases, so will the other. A negative Covariance indicates that as one variable increases, the other will decrease. Formally, RAVEN calculates the Covariance between two variables, X and Y, as

$$\Sigma(X, Y) = \frac{1}{V_1} \sum_{j=1}^{n} w_j (X_j - \bar{X})(Y_j - \bar{Y})$$

The Pearson product-moment correlation coefficient, or just the correlation, of two variables is related to the Covariance as follows

$$\Gamma(X, Y) = \frac{\Sigma(X, Y)}{\sigma_X \sigma_Y}$$

For a simple test and demonstration of RAVEN's functionality, a classic Monte-Carlo problem, the estimation of pi, was performed. By sampling two variables between -1.0 and 1.0, inclusive, then taking the square root of the sum of their squares, we can estimate the distance the point lies from the origin. Geometrically, if a circle is inscribed inside of a square and the diameter of the circle is the same length as the sides of the square, the probability that a point sampled to be somewhere inside the square will be inside the circle is equal to pi/4. By sampling sufficient points, an estimate of pi can be obtained. To test RAVEN's Monte-Carlo capabilities, 10,000 points of data comprised of two numbers sampled between -1.0 and 1.0, inclusive, were generated. The square root of the sum of the squares of these two numbers was obtained to calculate the distance from the origin. Dividing the number of points calculated to be inside the circle by the total number of points sampled, then multiplying this by 4, our estimate of pi is calculated.

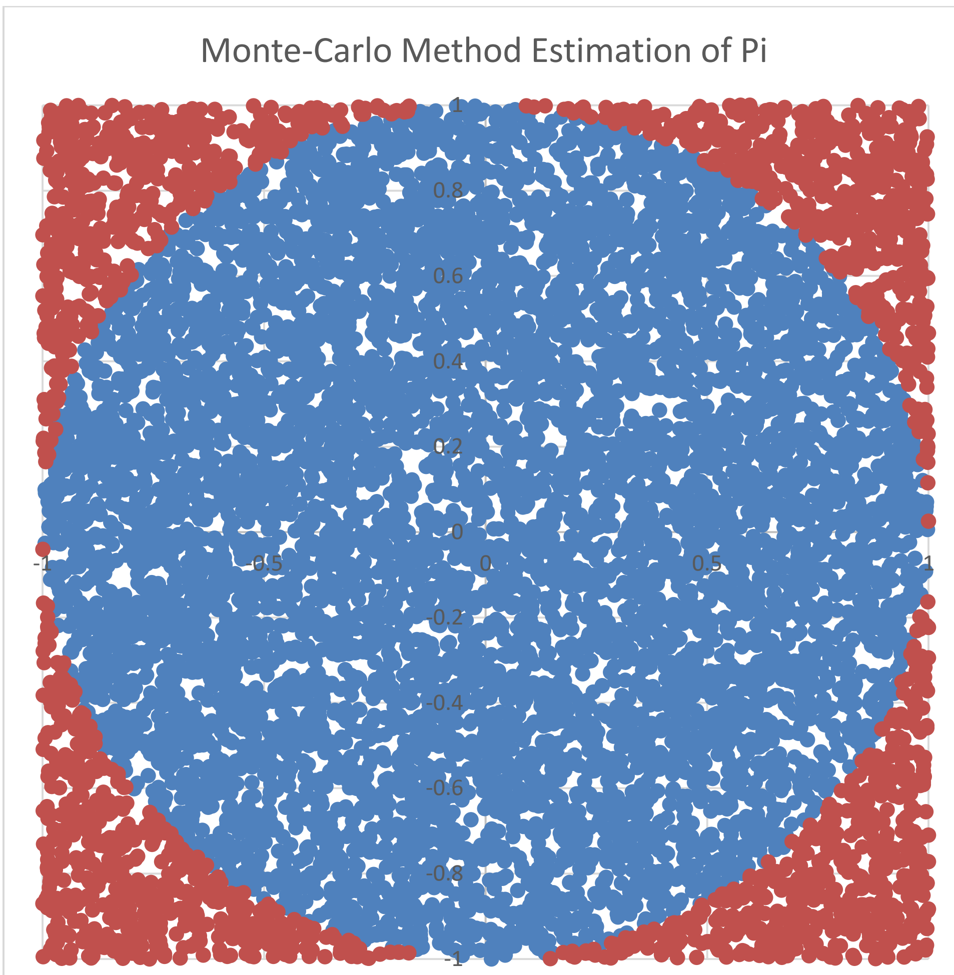


Figure 9. RAVEN Monte-Carlo Demonstration Example

Figure *9* shows a visualization of the results of the RAVEN Monte-Carlo estimation of pi. Of the 10,000 points of data sampled for the estimation of pi, 7,833 of them were inside the circle, giving us a result of 3.1332 as the estimate for pi. The true value of pi, truncated to 4 decimal places, is 3.1416, giving a percent error of 0.267% - a close estimate of pi!

### 4.10. *Uncertainty Quantification*

To ensure that a sufficient number of data points were generated and analyzed, the Dvoretzky-Kiefer-Wolfowitz (DKW) inequality was used to quantify the uncertainty in the cumulative density function (CDF) of the accident scenario release source term activity for the single stochastic parameter exploratory tests performed in the initial analyses. For empirically determined CDFs obtained using independent and identically distributed stochastic parameters, the DKW inequality states that the probability that the 'true' CDF value is less than ε from the empirically determined CDF is given by

$$\text{Probability}\ (Abs(CDF_{True} - CDF_{Determined}) > \varepsilon) \leq 2e^{-2n\varepsilon^2}$$

Rearranging this equation, for a desired confidence level β and *n* samples, our confidence interval is

$$\varepsilon = \sqrt{\frac{1}{2n}\ln(\frac{2}{1-\beta})}$$

Rearranging again, for a desired confidence level β and a desired uncertainty ε, we need

$$n = \frac{\ln(\frac{2}{1-\beta})}{2\varepsilon^2}$$

samples to achieve the desired confidence level and uncertainty. Using a 95% confidence interval, our uncertainty for a given number of samples can be calculated to be

$$\varepsilon = \sqrt{\frac{1.8444397}{n}}$$

Using this simplified equation, we can calculate that 2,000 points of data would give a 3.0% uncertainty, 1,000 points of data would give a 4.3% uncertainty, 500 points of data would give a 6.1% uncertainty, 300 points of data would give a 7.8% uncertainty, 200 points of data would give a 9.6% uncertainty, and 100 points of data would give a 13.6% uncertainty. It was later discovered that the method below – the Welch Unpooled Variance T-Test Confidence Interval Means Difference Analysis – allowed for the use of ~150 sample points per configuration without the loss of statistical significance.

For calculating the Confidence Interval for the difference between mean values for each upgrade configuration test and the base case test, a Welch Unpooled Variance T-Test based Confidence Interval was used (Welch 1947). This test is broadly applicable, as the only assumptions it demands are that the two sets of sampled data being compared are normally distributed and independent. It assumes nothing about the population variance of either distribution involved and does not require the population variance (or "true" variance) to be known in advance. Welch's T-Test has been found to be robust against Type 1 error (false rejection of the null hypothesis) in prior examinations (Derrick, Toher, and White 2016) and prior research has found it to be superior to the widely used Student's T-Test, as the Student's T-Test demands equal variances and can be unreliable when equal variances are incorrectly assumed (Ruxton 2006).

The data produced in the multi-parameter analyses is not normal, so some legwork is necessary to show that Welch's T-Test is still appropriate. The law of large numbers states that when a sequence of independent and identically distributed random variables $X_n$ is drawn from a distributed of expected value μ and finite variance $\sigma^2$, the sample means converge towards the expected value μ as we take progressively larger sample sizes. In other words

$$\lim_{n\to\infty} \overline{X_n} = \mu$$

where n is the number of samples and $\overline{X_n}$ is the mean of the sampled values.

The Central Limit Theorem extends this result and states that as more samples are taken, the probability distribution of the difference between the sample mean and the true mean, μ, approximates a normal distribution with mean 0 and variance $\sigma^2$. Formally, it states that

$$\lim_{n\to\infty} \sqrt{n}\,(X_n - \mu) = N(0, \sigma^2)$$

where $N(0, \sigma^2)$ is a normal distribution with mean 0 and variance $\sigma^2$. The practical use of this here is that even though the data produced in the analyses is not normal, the probability distribution of the difference between the sampled mean release activity for each upgrade configuration and the "true" mean release activity for each upgrade is normal, and the difference in mean release activity for each upgrade configuration can be examined with Welch's T-Test for statistical significance and bounded with confidence intervals.

In using the Welch treatment, initially the Mean Difference Standard Error is calculated as

$$\mathrm{SE}(\overline{X_1} - \overline{X_2}) = \sqrt{\frac{s_1^2}{n_1} + \frac{s_2^2}{n_2}}$$

where $\overline{X_1}$ and $\overline{X_2}$ are the calculated Mean values for the two upgrade configuration tests, $s_1$ and $s_2$ are the calculated sigma values for each test, and $n_1$ and $n_2$ are the number of samples in each test. To find the T-value used for each test, we use the Welch-Satterthwaite approximation to the degrees of freedom,

$$df = \frac{(\frac{s_1^2}{n_1} + \frac{s_2^2}{n_2})^2}{\frac{1}{n_1 - 1}(\frac{s_1^2}{n_1})^2 + \frac{1}{n_2 - 1}(\frac{s_2^2}{n_2})^2}$$

then using the calculated degrees of freedom, a T-table can be consulted for the appropriate value to match the number of degrees of freedom and the desired level of confidence in the resulting confidence interval. Finally, the confidence interval for the difference in means can be calculated as

$$\overline{X_1} - \overline{X_2} \pm t_{\frac{\alpha}{2}} * SE(\overline{X_1} - \overline{X_2})$$

where α is one minus the desired confidence – in this case, for a 95% confidence, α is equal to 0.05.

### 4.11. *Upgrade Cost Evaluation*

Much of the data used to evaluate the cost of implementing upgrades came from nuclear power plant license renewal applications. As part of power plant license renewals, the NRC requires a Generic Environmental Impact Statement (GEIS) to be submitted for each plant applying for a license renewal. As part of the GEIS, an analysis of Severe Accident Mitigation Alternatives (SAMA) must be performed and submitted (USNRC NUREG-1437, multiple years). Including in the SAMA analyses are cost estimates of proposed power plant alterations. The SAMA data from GEIS reports, as it relates to the upgrades we intend to analyze, has been examined and is presented in this section.

The cost of Hardened Containment Ventilation Systems (HCVS) were analyzed in the GEIS reports for the Cooper Nuclear Station, the Hope Creek Nuclear Generating Station, the LaSalle County Nuclear Generating Station, and the Seabrook Station Nuclear Power Plant. Additionally, a 2012 USNRC report on HCVS and Filtered Containment Ventilation Systems (FCVS) discussed potential costs. These costs are tabulated below in Table 8.

Table 8. HCVS Cost Estimates

| HCVS Cost Reported | Cost Estimate Source |
|---|---|
| $1,000,000 | Cooper GEIS Report |
| $25,000,000 | Hope Creek GEIS Report |

| | |
|---|---|
| $13,000,000 | LaSalle GEIS Report |
| $3,000,000 | Seabrook GEIS Report |
| $2,027,000 | USNRC HCVS/FCVS Report |

The cost of the FCVS upgrade was analyzed in the GEIS reports for the Braidwood Nuclear Generating Station, the Byron Nuclear Generating Station, the James A. FitzPatrick Nuclear Power Plant, the Pilgrim Power Plant, and the Seabrook Station Nuclear Power Plant. Additionally, a 2012 NRC presentation on the FCVS upgrade discusses potential costs, as well as a 2012 USNRC report on HCVS and FCVS systems. These costs are tabulated below in Table 9.

Table 9. FCVS Cost Estimates

| FCVS Cost Reported | Cost Estimate Source |
|---|---|
| $5,700,000 | Braidwood GEIS Report |
| $5,700,000 | Byron GEIS Report |
| $1,500,000 | FitzPatrick GEIS Report |
| $3,000,000 | Pilgrim GEIS Report |
| $20,000,000 | Seabrook GEIS Report |
| $16,127,000 | USNRC HCVS/FCVS Report |

The cost of the Hydrogen Igniters upgrade was analyzed in the GEIS reports for the LaSalle County Nuclear Generating Station, the McGuire Nuclear Station, and the Seabrook Station Nuclear Power Plant. These costs are tabulated below in Table 10.

Table 10. Hydrogen Igniter Cost Estimates

| Hydrogen Igniter Cost Reported | Cost Estimate Source |
|---|---|
| $205,000 | LaSalle GEIS Report |
| $205,000 | McGuire GEIS Report |
| $100,000 | Seabrook GEIS Report |

The cost of the Hydrogen Igniter Backup Power System upgrade was analyzed in the GEIS reports for the Arkansas Nuclear One nuclear power plant, the Donald C. Cook Nuclear Plant. The McGuire Nuclear Station, and the Seabrook Station Nuclear Power Plant. The costs are tabulated below in Table 11.

Table 11. Hydrogen Igniter Backup Power System Cost Estimates

| Hydrogen Igniter Backup Power System Cost Reported | Cost Estimate Source |
|---|---|
| $1,000,000 | Arkansas Nuclear One GEIS Report |
| $147,000 | Cook GEIS Report |
| $540,000 | McGuire GEIS Report |
| $200,000 | Point Beach GEIS Report |

The cost of the Passive Autocatalytic Recombiners (PARs) upgrade was analyzed in the GEIS reports for the Braidwood Nuclear Generating Station, the Byron Nuclear Generating Station, the Enrico Fermi Nuclear Generating Station, the McGuire Nuclear Station, the Monticello Nuclear Generating Plant, the Seabrook Station Nuclear Power Plant, and the Turkey Point Nuclear Generating Station. It is worth noting that Turkey Point contains a $45,000 cost estimate per PAR unit, not for the total cost of adding 34 PARs to the reactor unit. These costs are tabulated below in Table 12.

Table 12. PAR Cost Estimates

| PAR Cost Reported | Cost Estimate Source |
|---|---|
| $760,000 | Braidwood GEIS Report |
| $760,000 | Byron GEIS Report |
| $760.000 | Fermi GEIS Report |
| $750,000 | McGuire GEIS Report |
| $760,000 | Monticello GEIS Report |
| $1,530,000 | Turkey Point GEIS Report |

The cost of an in-containment core inventory makeup tank upgrade was not directly examined in any publicly available sources that were found. However, the cost of expanding a power plant's containment building, a necessary step in adding the room necessary to implement the upgrade at a power plant, was analyzed in the GEIS reports for the Pilgrim Nuclear Generating Station and the James A. FitzPatrick Nuclear Power Plant. In both reports, it was estimated to cost $8,000,000. The cost of the tank itself was interpolated using data for stainless steel water tanks found in a Michigan state government analysis of the cost of various water tanks (State of Michigan 2003). To match the capacity of the AP1000 IRWST, the cost of a 600,000-gallon elevated steel water tank was examined and found to be $640,000. Together, the expansion of containment and installation of an elevated steel water tank that could provide gravity driven low pressure coolant injection to the core after depressurization would cost, at a minimum estimate, $8,640,000. For a high-end estimate of the cost of this estimate, $46,200,000 was used – five times the low-end cost estimate was well in line with the ratio between the high-end and low-end cost estimates found for other upgrades.

To assign a figure to the cost of Accident Tolerant Fuels (ATFs), we are using potential ATF costs of 75%, 100%, 105%, 110%, and 125% of the current cost of fuel to examine the impact of a variety of potential costs, per year, of switching to ATFs. Using a 2017 Nuclear Energy Institute report on the costs of nuclear power, we found an average estimate of $7/MWh for fuel costs for a nuclear power plant to be a reasonable estimate of the costs of nuclear power, with reported costs ranging from $6.15/MWh to $8.67/MWh (Nuclear Energy Institute 2017). For an 1,100 MWe power plant, this comes out to an estimated fuel cost of approximately $67.5 million per year. From this, our upgrade cost estimate for ATF ranges from a $16.875M savings every year to a $16.875M per year cost.

Lastly, to estimate the cost of the procedural change to manually activate the ADS several hours into an SBO scenario, we examined the cost of similar procedure-based upgrades within available GEIS reports and found minor procedural changes to be estimated to cost anywhere from $26,000 to $50,000 to $146,000 in the Pilgrim and Sequoyah GEIS reports.

### 4.12. *Comparative Safety Upgrade Economic Analysis*

The final step of the analysis is to take the compiled risk reduction data for all upgrades and potential upgrade combinations evaluated and compare them against the cost of installing, operating, and maintaining those upgrades. This is a fairly simple, if laborious affair, as data for the costs of upgrades is often sparse and difficult to obtain. However, once data is obtained one can compare the benefits gained in the form of risk reduction from an upgrade to the costs of implementing that upgrade, in the manner presented in Section 4.6, to find the most cost-efficient use of resources for improving safety.

Major uncertainties to account for during this analysis come in the form of uncertainties in the Expected Value and CDF of the release activity for each upgrade configuration. It is expected that uncertainties in upgrade costs will dominate over uncertainties in the average release activities, as uncertainties in the cost of the upgrade were found to regularly be massive – up to an order of magnitude difference between the least and greatest cost estimates found in literature for a particular upgrade.

### 4.13. *Assumptions and Justification*

As with any research effort, assumptions must be made to reduce the scope of the work to something that can be reasonably accomplished – humorous, given that the nature of the project is to explore assumptions that are normally made in PRA analysis. The various modeling assumptions made in this research are discussed and justified in this section.

An extremely simple, fast running model of a BWR was used. It was assumed that this model would provide a reasonable approximation of the true behavior of a BWR in an SBO scenario. This assumption is justified both by benchmarking the model against the study performed at the Technical Research Centre of Finland by Tuomo Sevon (Sevon 2015) and by the source of the underlying model modified as part of this research effort. The base model that was modified in this project was originally developed at Sandia National Lab as a simple example problem of a BWR to include in the distribution package for the MELCOR modeling code. This model and its development are discussed in more detail in Section 4.2. Its use is justified both by the documentation and understanding of its functionality shown in Section 4.2 as well as its origin, having been developed as a fast running, reasonably accurate model of a BWR by the expert MELCOR development team at SNL.

For the purposes of the LT-SBO progression, it was assumed that there would be no firewater injection, no AC power recovery, and that the time between the onset of LOOP conditions and the beginning of the LT-SBO would be no greater than 24 hours. The lack of firewater injection is a modeling simplification, representative of the type of disastrous scenario that would cause such an extensive LT-SBO and likely cause other extensive infrastructural damage. These simplifications were made to allow for a simple, fast running model to be used. Additionally, and for similar reasons, it was assumed that the reactor would not be in the middle of an overpower transient as the LOOP or SBO conditions occurred.

As well as the above assumptions with regards to what would go wrong, assumptions were made with regards to what would go right. It was assumed that there were no concurrent failures alongside the LT-SBO, that there would be no occurrences of ATWS conditions, that the analyzed upgrades would work as intended. This was done as a reasonable modeling simplification, as the systems involved would be designed, as all nuclear safety systems are, to be incredibly reliable, and the odds of any of these systems failing alongside the multiple other simultaneous failures involved in an LT-SBO decreases the likelihood of the scenario from an already low frequency to essentially negligible.

As a modeling simplification, it was assumed that the power plant concrete base mat was thick enough to prevent melt-through via corium-concrete interactions. This was assumed as a modelling simplification, and it is believed that the slow progression of the accident scenario inherent to an LT-SBO will not allow for the level of base mat melt through necessary to rupture the base mat. Examination of the accident progression of the disaster at Fukushima Daiichi supports this, the base-mat did not fail at any of the three units that suffered core meltdowns (IAEA 2015).

For the purposes of offsite consequence economic evaluation, it was assumed that the release timing would not affect the impact of the radioactive release significantly compared to alterations in the magnitude of the release. Silva et al. found an $R^2 = 0.982$ correlation between the cost per severe accident and the magnitude of the release activity source term (Silva et al. 2014). They found that very early releases and very long release durations both could increase the cost of the accident, even for the same source term release activity, but that the release activity magnitude was by far the more important factor in the offsite consequence of a radioactive release.

Additionally, for the purposes of comparative severe accident consequence evaluation, it was assumed that the power plant is lost and must be decommissioned and replaced, and that the cost of decommissioning does not vary significantly with release activity, compared to the overall cost of the accident scenario. This is in line with Silva et al.'s findings, as decommissioning was never found to be more than a few percent of the overall cost of the accident, and the cost of replacing the plant does not scale with release activity at all.

For the purposes of the stochastic parameter analysis, it was assumed that relatively simple and unrealistic stochastic parameter PDFs would be sufficient to demonstrate the success of the methodology. A more real-world applicable analysis would demand thorough research into component failure rates, human performance analyses, and initiating event frequencies to provide data that can defensibly be used to make decisions with regards to nuclear power plant safety. This is justified by the fact that the PDFs used for each stochastic parameter can readily be changed without altering the outcome of a particular modeling point of data, just how heavily it is probabilistically weighted.

It was assumed that stochastic parameter values outside of the chosen bounds either would not occur or would not be significantly influential to the overall outcome of the accident scenario. In the cases of the Containment Failure Pressure, DC Lifetime, Wetwell Initial Temperature, and Wetwell Initial Level, it simply very unlikely for values outside of the sampling bounds to occur – the bounds of the sampling space were picked out of practical consideration of the realities of the power plant. Extremely weak or extremely strong containments, very hot or very cold wetwells, very over or under filled wetwells, or extremely long or short dc lifetimes are unlikely to deviate outside of accepted operating conditions. In the case of the EDG Failure Time, it was assumed that the EDGs would fail within one day of the onset of LOOP conditions, as testing in the stochastic parameter sensitivity study showed that the difference between EDG failure times of ~24h and ~48h were not significant compared to the differences made by power plant upgrade configurations.

The second assumption made for the stochastic parameter analysis was that any sharp, highly localized discontinuities that would cause major, unseen deviation between the sampled data and the true population mean would not bias the model output data significantly in favor of any particular upgrade configurations. This is a necessary assumption for the use of Monte-Carlo sampling techniques, and the appropriateness of this assumption is checked by examining the CDFs of the various upgrade configuration analysis data sets – sharp, localized non-linearities in an upgrade configuration's data set result in enormous jumps in the release activity CDF in the output. While some jumps were observed in the release activity CDFs, none of them were found to bias the methodology final results significantly. The nature of these discontinuities is explored further in Section 5.7, and were universally observed to be attributable to relatively small variations in the accumulation and deflagration of hydrogen leading to containment failure in some cases, and greatly increasing the release activity in these cases.

# 5. Results

## 5.1. *Stochastic Parameter Sensitivity Studies*

### 5.1.1. Initial Wetwell Level

In general, a higher Initial Wetwell Level corresponded to a reduction in release magnitude. The greater volume of water allowed more heat to be transferred to the Wetwell before the actuation of the ADS was demanded, and the greater bulk of water into which radioactively contaminated steam was vented allowed for more material to scrub aerosols from the steam.

Table 13. Initial Wetwell Level Results Summary

| Configuration | Mean Release Activity | Median | Max | Sigma | 45th Percentile | 55th Percentile |
|---|---|---|---|---|---|---|
| No Upgrades | 2779.94 | 2638.69 | 6922.18 | 1129.27 | 2320.66 | 2739.5 |
| IRWST | 2603.47 | 2835.33 | 3890.74 | 620.06 | 2641.73 | 2890.03 |
| ADS, IRWST | 2243.51 | 2118.87 | 3890.74 | 519.16 | 2028.88 | 2186.00 |

As seen in Table 13, the results were sufficiently noisy that it was difficult to draw any good conclusions regarding the efficacy of any of the upgrades. This was acceptable, given that the purpose of this examination was predominantly to develop a better understanding of the effects of the Initial Wetwell Level on the progression of the accident, as well as how this impact changed when the accident progression was potentially altered by upgrades believed to be relevant.

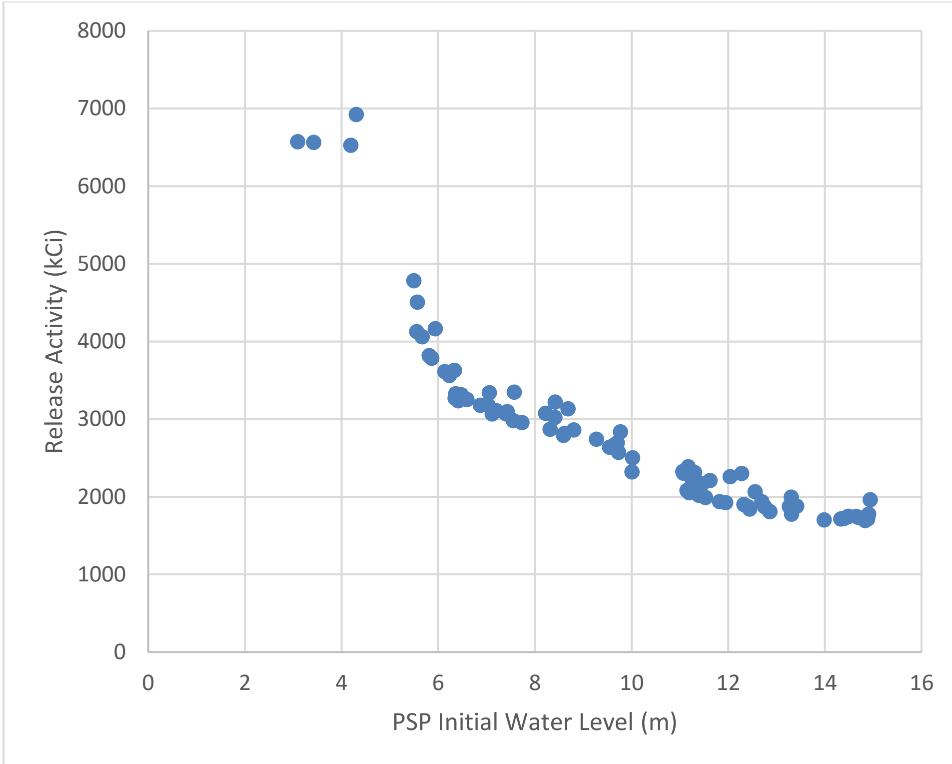


Figure 10. Initial Wetwell Level vs Release Activity, No Upgrades

Shown in Figure 10, with no upgrade, the release activity showed a relatively large sensitivity to the initial wetwell level, with an underfilled wetwell causing a greater release activity. With an extremely, dangerously underfilled wetwell, the release activity increased massively because the badly insufficient amount of water in the wetwell prevented the effective use of the wetwell as a heatsink to delay the onset of RPV pressurization and the onset of core damage, or for there to be sufficient water present to effectively scrub radioactive materials as they were released from the RPV.

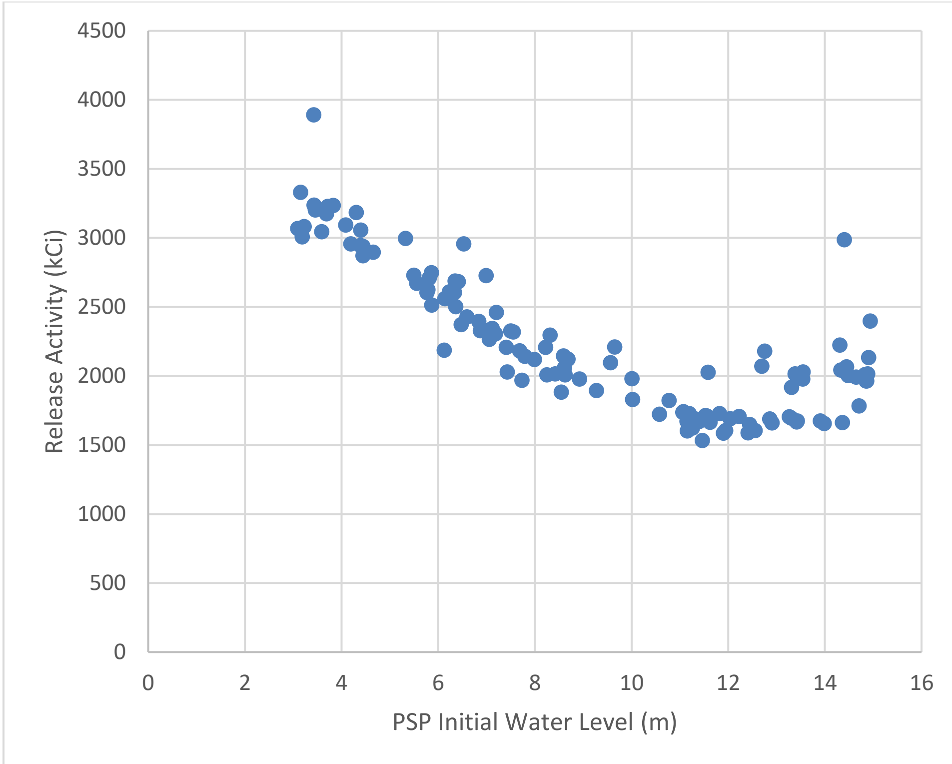


Figure 11. Initial Wetwell Level vs Release Activity, with Manual ADS and IRWST Upgrades

The results of the Initial Wetwell examination with the Manual ADS and IRWST Upgrades are shown in Figure 11. With the upgrades, even a badly underfilled wetwell was greatly mitigated. A more filled wetwell generally correlated to a lower release activity, though a very high wetwell initial level lead to overly early actuation of the ADS and a greater Release Activity due to the faster accident scenario progression leading to a hotter, more energetic containment environment and release.

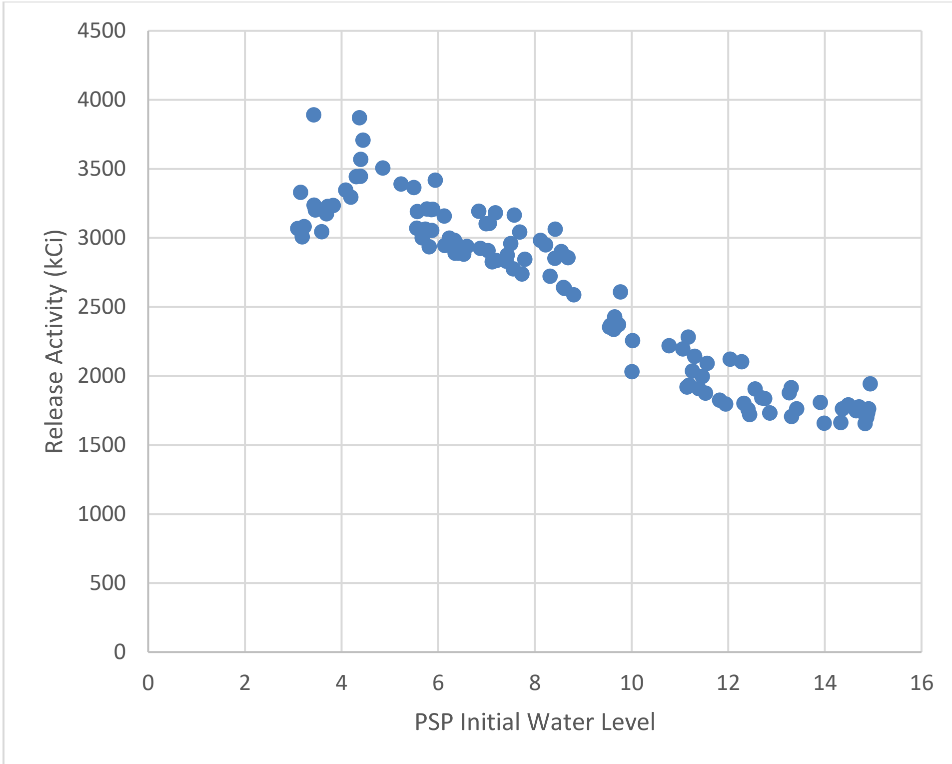


Figure 12. Initial Wetwell Level vs Release Activity, with IRWST Upgrade

The IRWST upgrade configuration Initial Wetwell Level sensitivity results are shown in Figure 12. The IRWST upgrade alone was not as effective as the Manual ADS and IRWST upgrades together. As with the other two cases, a greater PSP Initial Water Level generally correlated relatively strongly with a lower Release Activity, and the inclusion of the IRWST negated the problems associated with the overfilled Wetwell leading to overly early actuation of the ADS.

With a nominal Initial Wetwell Level of 11m, a range of Initial Wetwell Levels from 7m to 15m was used. While some of the testing results indicate that an Initial Wetwell Level below 6m would significantly impact the Release Activity, these were judged to be unrealistic values for the Initial Wetwell Level as they are at or below roughly half of the nominal level and the likelihood that such a badly underfilled Wetwell would go unnoticed for long was deemed minimal. At the upper end of the spectrum, overfilling the Wetwell beyond 15m quickly approaches inventory levels that would trigger a reactor SCRAM and ADS actuation on the possibility of a LOCA causing the high inventory level, and a yet further overfilled Wetwell was judged extremely unlikely.

### 5.1.2. Initial Wetwell Temperature

The Release Activity showed modest sensitivity to the Initial Wetwell Temperature, with higher temperatures corresponding to a faster accident progression and a larger release due to the reduced heat absorption capacity of the Wetwell resulting from the higher initial temperature.

Table 14. Initial Wetwell Temperature Results Summary

| Configuration | Mean Release Activity | Median | Max | Sigma | 45th Percentile | 55th Percentile |
|---|---|---|---|---|---|---|
| No Upgrades | 2851.687 | 2909.48 | 3335.81 | 243.9108 | 2863.82 | 2918.34 |
| ADS | 2813.789 | 2820.05 | 3395.77 | 286.3523 | 2811.05 | 2861.3 |
| IRWST | 2664.355 | 2706.43 | 3180.22 | 232.8598 | 2684.29 | 2720.37 |
| ADS, IRWST | 2025.064 | 2037.11 | 2580.61 | 222.6017 | 2003.15 | 2083.05 |

The results of the Initial Wetwell Temperature sensitivity study are shown in Table 14. The Manual ADS, IRWST, and Manual ADS and IRWST upgrade configurations did not significantly alter the correlation between Release Activity and Initial Wetwell Temperature, though the Manual ADS and IRWST upgrade configuration did show a significant reduction in Release Activity across all sampled scenarios.

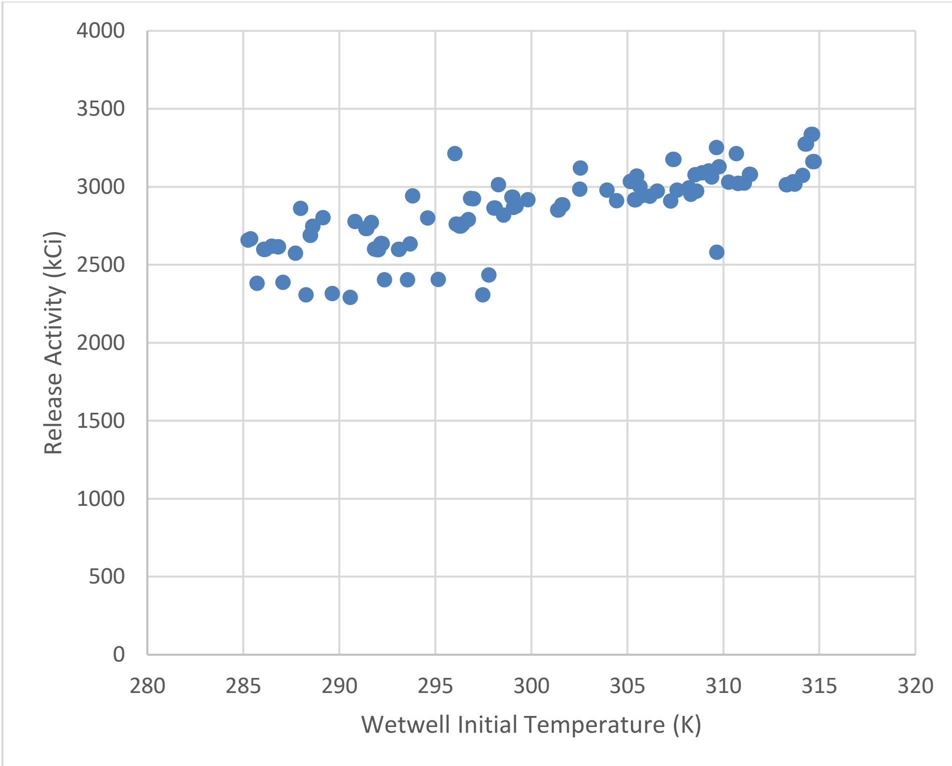


Figure 13. Initial Wetwell Temperature vs Release Activity, No Upgrades

Figure 13 shows that the Release Activity increased modestly with increasing Wetwell Initial Temperature. With an elevated Wetwell Initial Temperature, the wetwell is less able to act as a heatsink to condense steam vented from the RPV through the SRVs, accelerating the pace of the accident scenario, creating a hotter and more energetic release, increasing the amount of material released, increasing the Release Activity.

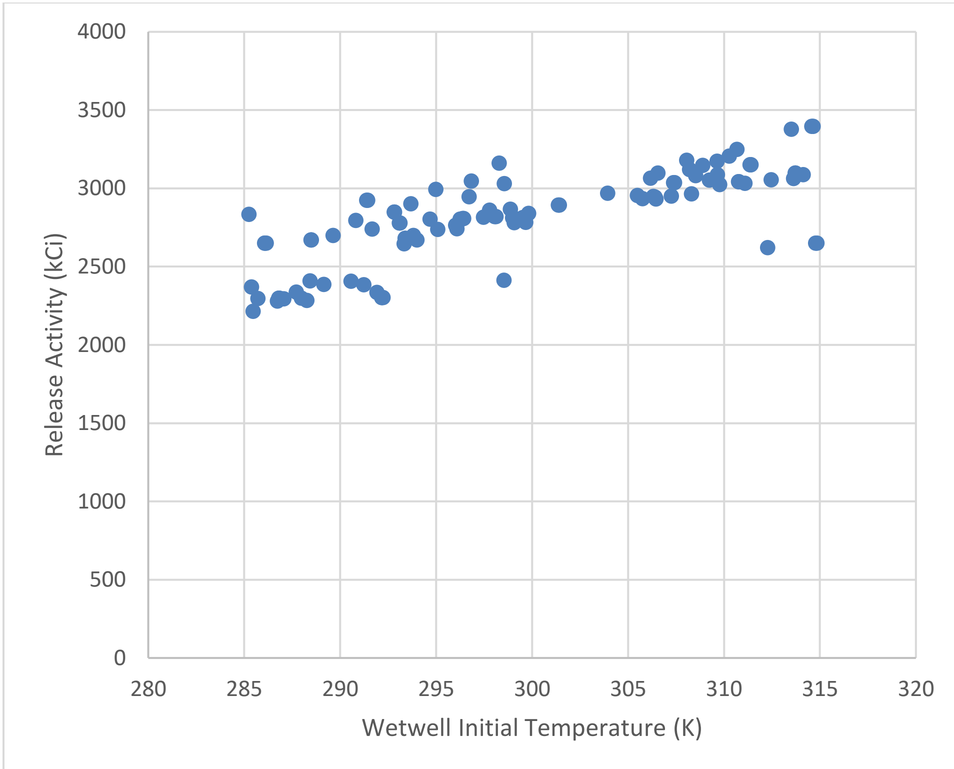


Figure 14. Initial Wetwell Temperature vs Release Activity, Manual ADS Upgrade

As seen in Figure 14, the Manual ADS upgrade alone had a minimal impact on the Release Activity, compared to no upgrades. The Manual ADS upgrade is intended to be paired with the IRWST upgrade to ensure that it is able to function properly in all scenarios, and by itself has little impact. The main purpose of the Manual ADS upgrade is to ensure the IRWST upgrade is always able to function, rather than run the risk of a combination of a short Battery Lifetime and a warm, underfilled Wetwell causing RPV blowdown to fail, preventing the use of the IRWST until the RPV depressurizes via lower head failure.

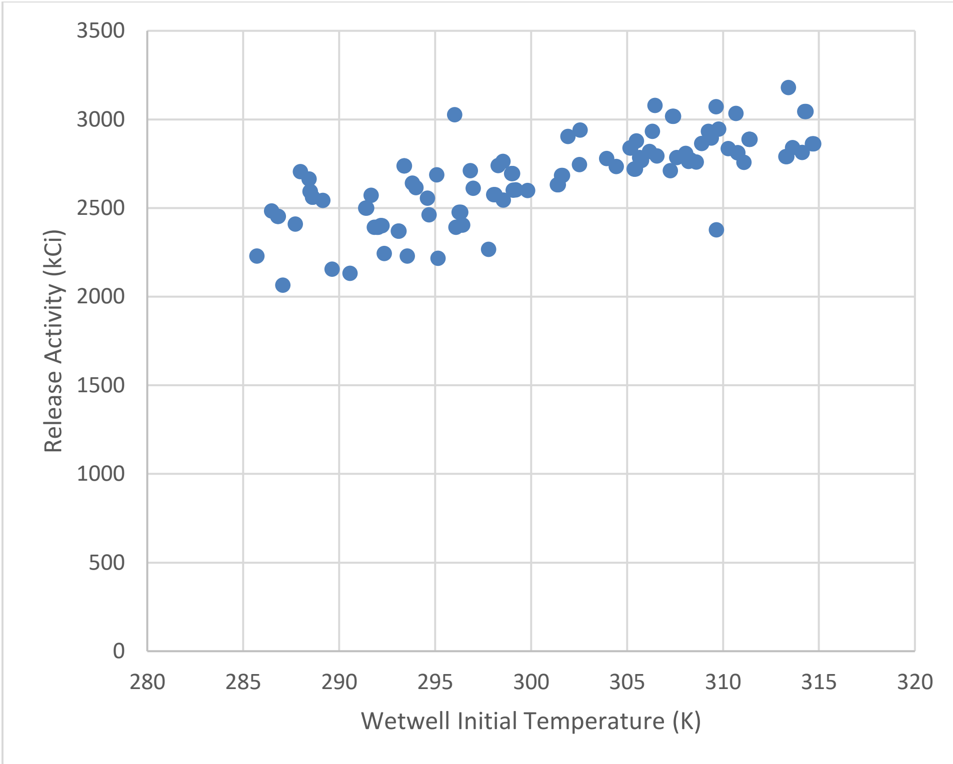


Figure 15. Initial Wetwell Temperature vs Release Activity, IRWST Upgrade

Figure 15 shows that the IRWST upgrade showed marginal improvements across the board as the Wetwell Initial Temperature varied, compared to the no upgrades configuration. Both with a colder than expected and a warmer than expected Wetwell Initial Temperature, the IRWST upgrade alone reduced the Release Activity by roughly 10%.

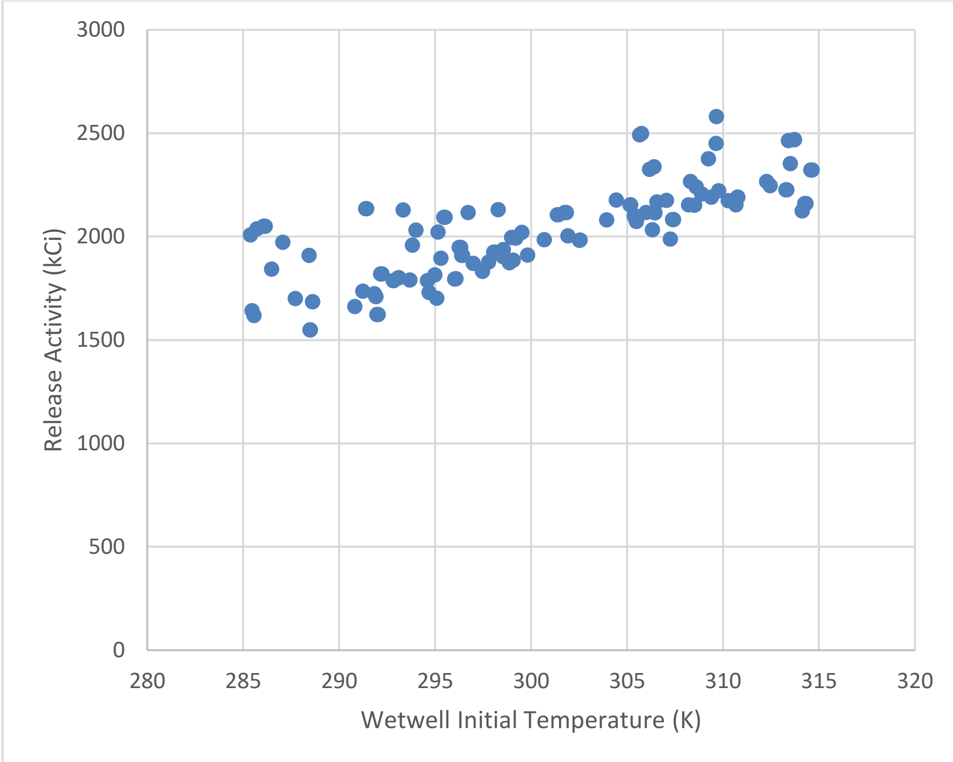


Figure 16. Initial Wetwell Temperature vs Release Activity, Manual ADS + IRWST Upgrades

Figure 16 shows the results of the Initial Wetwell Temperature sensitivity study using an upgrade configuration that implemented the Manual ADS and IRWST upgrades. Together, the Manual ADS and IRWST upgrades provided significant gains, where either upgrade alone did not have a major impact. Across essentially all sampled points, the IRWST upgrade, enabled by the support of the Manual ADS upgrade to work properly in all cases, provided nearly a 30% reduction in Release Activity.

While it seems plausible that the Release Activity trends with the Wetwell Initial Temperature would continue in both directions, a Wetwell Initial Temperature of under 285 K (or roughly 50º F) or above 315 K (roughly 110º F) were deemed unlikely and not considered in later analyses. At the cold end of things, temperatures below 285 K for the Wetwell are simply very unlikely because the inside of reactor buildings are frequently very warm (uncomfortably so, even) and even temperatures of 285 K would require active refrigeration of the Wetwell. Looking to the hotter temperatures, 315 K is approaching the point at which the Wetwell temperature will trigger the ADS system, and temperatures that, if they match the general temperature inside the reactor building, would threaten worker safety through hyperthermia ad would not be tolerable working conditions.

### 5.1.3. EDG Failure Time

The Release Activity showed a significant sensitivity to the EDG Failure Time, with longer EDG Failure Times corresponding to a significant decrease in Release Activity. Problems were encountered for very long EDG Failure Times, but the issues present were not addressed as it was anticipated that they would not impact later analyses.

Table 15. EDG Failure Time Results Summary

| Configuration | Mean Release Activity | Median | Max | Sigma | 45th Percentile | 55th Percentile |
|---|---|---|---|---|---|---|
| No Upgrades | 3092.66 | 3131.55 | 3354.14 | 84.58 | 3131.55 | 3131.55 |
| IRWST, ADS | 2328.11 | 2347.35 | 2347.35 | 90.78 | 2347.35 | 2347.35 |
| PARs, Vents | 2385.82 | 2420.79 | 2569.1 | 138.41 | 2420.79 | 2420.79 |

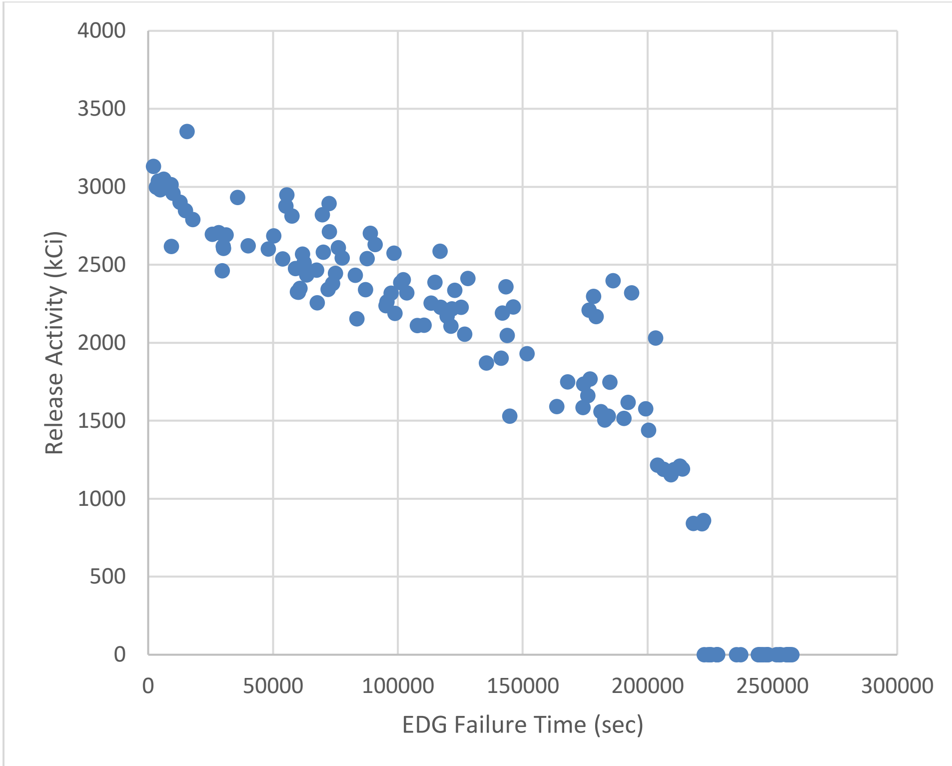


Figure 17. EDG Failure Time vs Release Activity, No Upgrades

Figure 17 shows that with no upgrades, the Release Activity showed a significant decline over time with increasing EDG Failure Time until dropping off to zero Release Activity for extremely long EDG Failure Times. For very long EDG Failure Times (generally, 200,000 seconds and greater), zero Release Activity is observed. Due to the method by which the EDG Failure Time was modeled, the simulation was ending before the onset of radionuclide release, as the model simulated up to 72 hours after LOOP conditions and not from the time of the SBO itself. This was not anticipated to be an influencing factor in later analyses, as values for the EDG Failure Time of greater than 24 hours were not considered in later analyses.

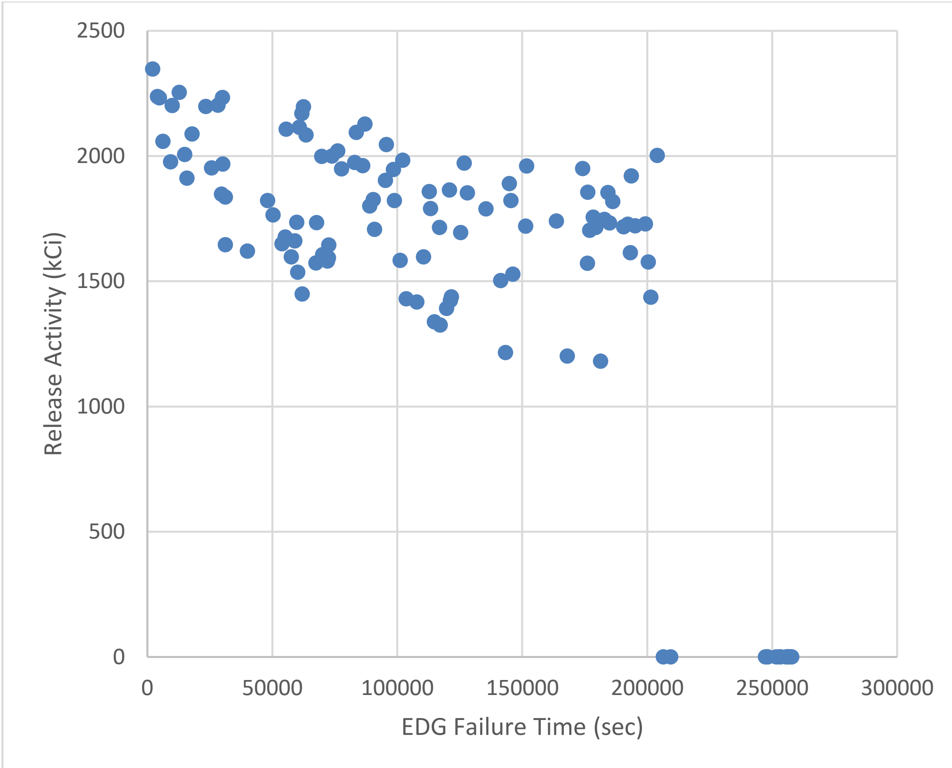


Figure 18. EDG Failure Time vs Release Activity, Manual ADS and IRWST Upgrades

It can be seen in Figure 18 that the Manual ADS and IRWST upgrades together, as well as significantly reducing the Release Activity source term in all sampled scenarios, significantly reduced the sensitivity of the Release Activity to the EDG Failure Time. Little reduction can be seen in the Release Activity as the EDG Failure Time increases in the Manual ADS and IRWST upgrade configuration, where significant reduction in Release Activity can be seen in the No Upgrades configuration.

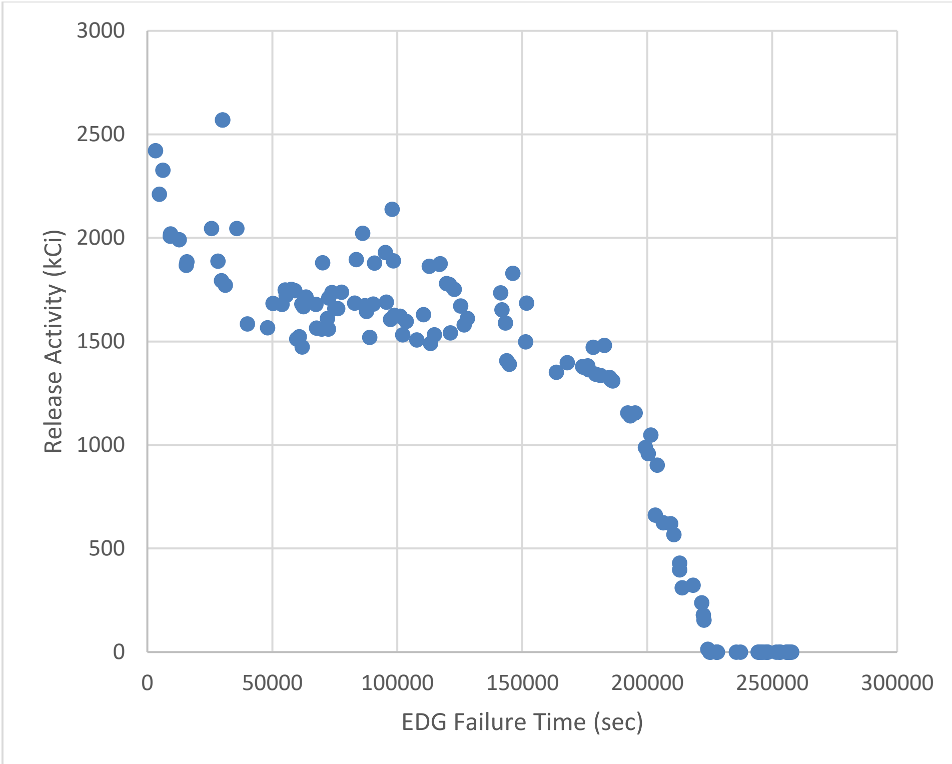


Figure 19. EDG Failure Time vs Release Activity, PARs and Hardened Vents Upgrades

As seen in Figure 19, the PARs and Hardened Containment Vents together provided a significant reduction in Release Activity, compared to No Upgrades, and provided an average reduction in Release Activity comparable to the average Release Activity for the Manual ADS and IRWST Upgrade configuration.

While longer EDG Failure Times corresponded to further reductions in Release Activity, for the sake of analysis it was assumed that the EDG would fail within the first 24 hours of the simulation,

### 5.1.4. Containment Failure Pressure

The Release Activity was found to have a binary relationship with the Containment Failure Pressure – either the Containment Failure Pressure was sampled to be low enough to cause the containment to fail when it would not with a stronger containment, increasing the release activity.

Table 16. Containment Failure Pressure Results Summary

| Configuration | Mean Release Activity | Median | Max | Sigma | 45th Percentile | 55th Percentile |
|---|---|---|---|---|---|---|
| No Upgrades | 2607.4 | 2588.53 | 3126.3 | 97.40059 | 2588.53 | 2588.55 |
| PARs | 2615.598 | 2593.56 | 3149.45 | 107.2556 | 2593.56 | 2593.6 |
| Vents | 2407.04 | 2407.04 | 2407.04 | 4.57E-13 | 2407.04 | 2407.04 |

Table 16 shows the tabulated statistics of the various upgrade configurations implemented in this sensitivity study. The Containment Failure Pressure was not, in these scenarios, observed to make any significant difference in Release Activity at all other than an increase in Release Activity for very weak containments. Essentially, the accident scenarios were too severe, without mitigating upgrades, for a mildly increased Containment Failure Pressure to make a difference, and not challenging enough to threaten Containment Integrity when appropriate mitigating upgrades – specifically, Hardened Ventilation – were implemented.

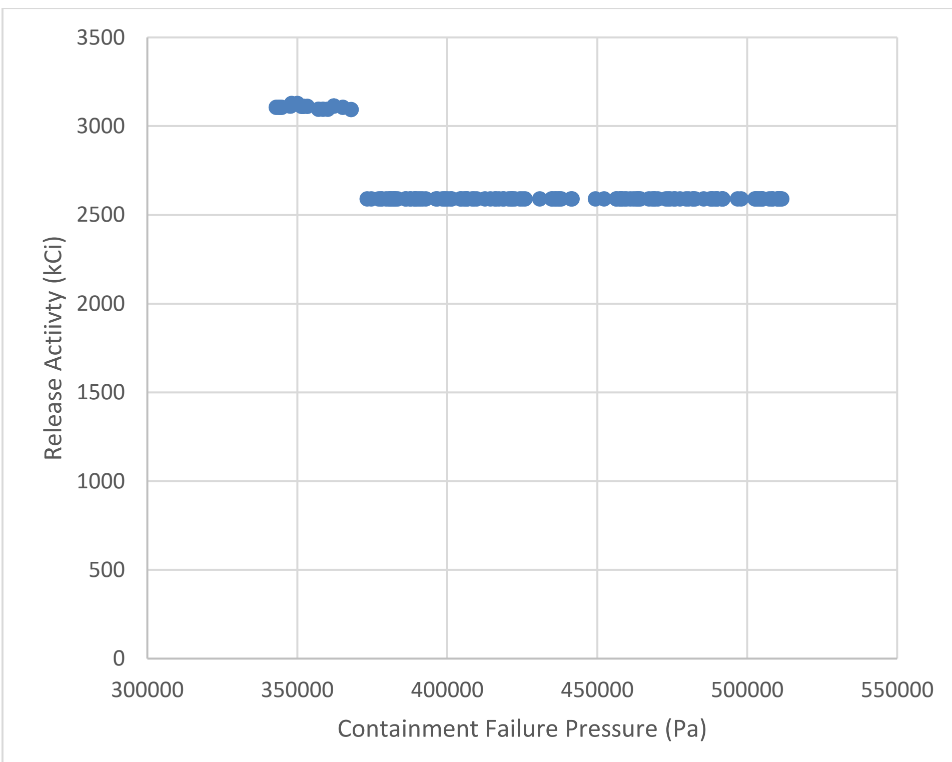


Figure 20. Containment Failure Pressure vs Release Activity, No Upgrades

Figure 20 shows the results of varying the Containment Failure Pressure in an unupgraded plant model. As discussed previously, varying the Containment Failure Pressure alone did not alter the Release Activity at all until it was sampled as being approximately 80% or less of the nominal value, at which point the containment began to fail on an early, small deflagration of hydrogen in the Wetwell that, while causing a pressure spike, was not a major deflagration that would rupture a properly constructed and maintained containment capable of withstanding its design pressure, rather than failing early. This early failure lead to an increase in Release Activity.

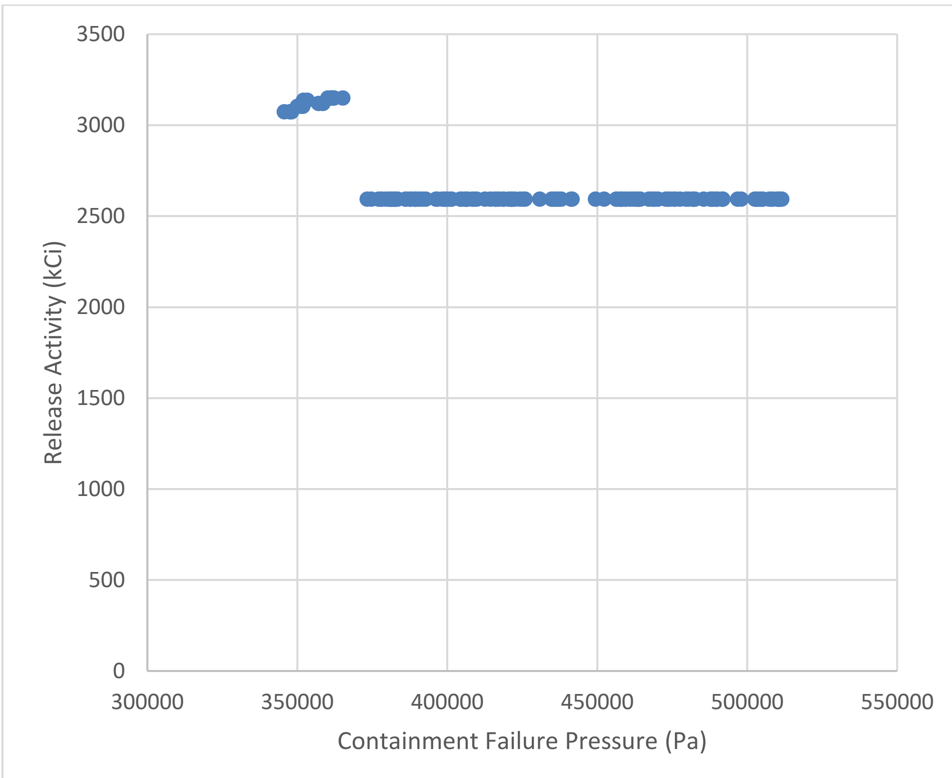


Figure 21. Containment Failure Pressure vs Release Activity, PAR Upgrade

Figure 21 shows the results of varying the Containment Failure Pressure in a plant model with only the PAR upgrade implemented. The PAR upgrade did not make a significant difference in this examination, as when the Containment Failure Pressure was sampled as being very low, or an abnormally weak containment, an early, relatively weak deflagration in the Wetwell was able to create a large enough spike in containment pressure to fail the weaker containment. Because the

PARs were implemented in the containment, they were unable to mitigate the accumulation of hydrogen in the Wetwell and did not significantly change the Release Activity in these scenarios. Without ventilation, though, even with PARs implemented the containment would inevitably fail to steam accumulation and slow overpressurization of the containment.

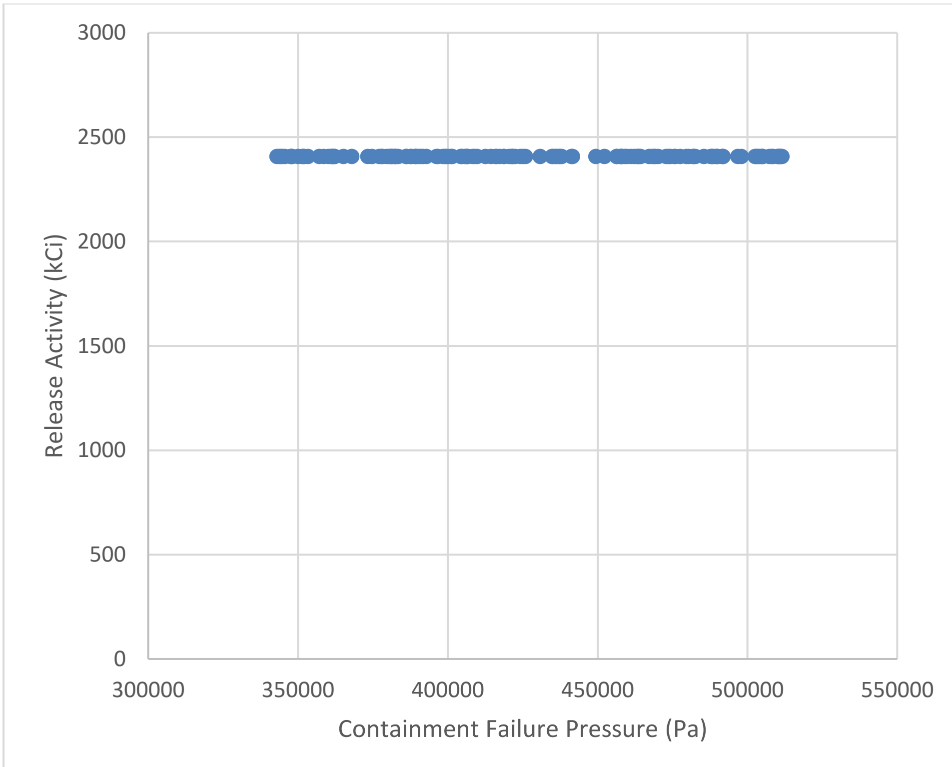


Figure 22. Containment Failure Pressure vs Release Activity, Vent Upgrade

As seen in Figure 22, with the Hardened Vent upgrade implemented, even the weakest Containment Failure Pressures sampled did not lead to the containment rupturing, and the release activity was significantly reduced in cases with extremely weak containments but had no impact other than that.

### 5.1.5.DC Lifetime

Table 17 presents the summary results of the DC Lifetime sensitivity study with various upgrade configurations. The Manual ADS had little impact on the Release Activity, but the IRWST upgrade provided a reasonable reduction in Release Activity and the Manual ADS and IRWST upgrades together had provided a major reduction in Release Activity.

Table 17. DC Lifetime Results Summary

| Configuration | Mean Release Activity | Median | Max | Sigma | 45th Percentile | 55th Percentile |
|---|---|---|---|---|---|---|
| Manual ADS | 3628.72 | 3564.82 | 4836.98 | 415.7644 | 3522.74 | 3589 |
| No Upgrades | 3560.447 | 3473.06 | 4508.27 | 358.7578 | 3408.58 | 3533.93 |
| IRWST | 2990.061 | 2949.14 | 3417.93 | 147.7714 | 2942.57 | 2970.97 |
| Manual ADS, IRWST | 2444.034 | 2411.54 | 2777.48 | 126.1509 | 2403.84 | 2419.56 |

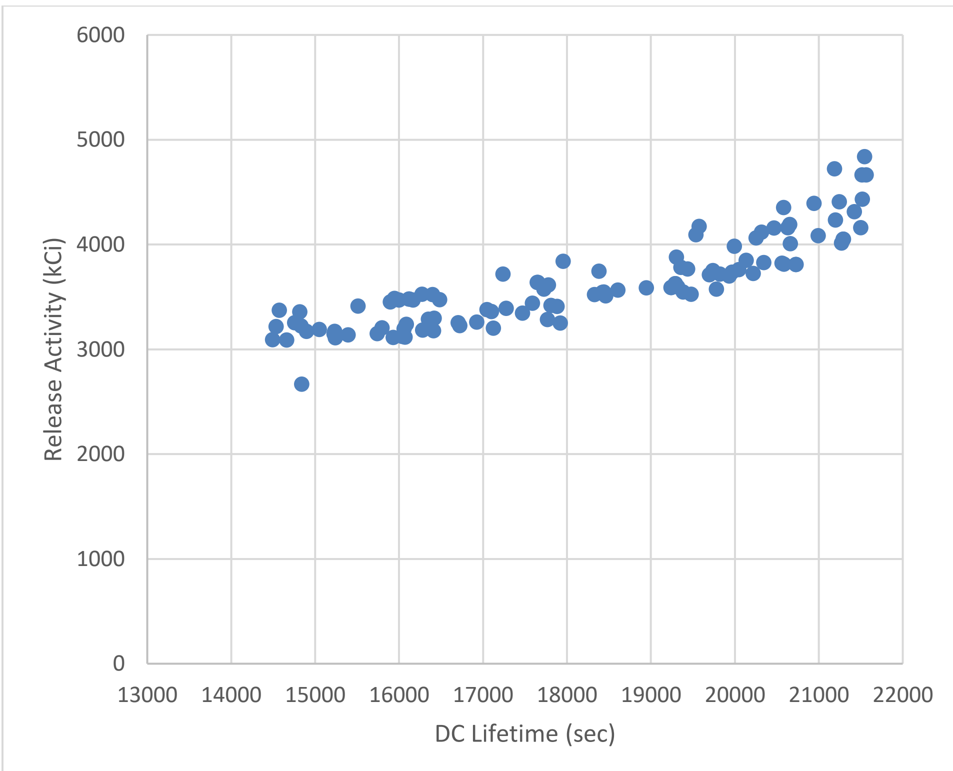


Figure 23. DC Lifetime vs Release Activity, Manual ADS Upgrade

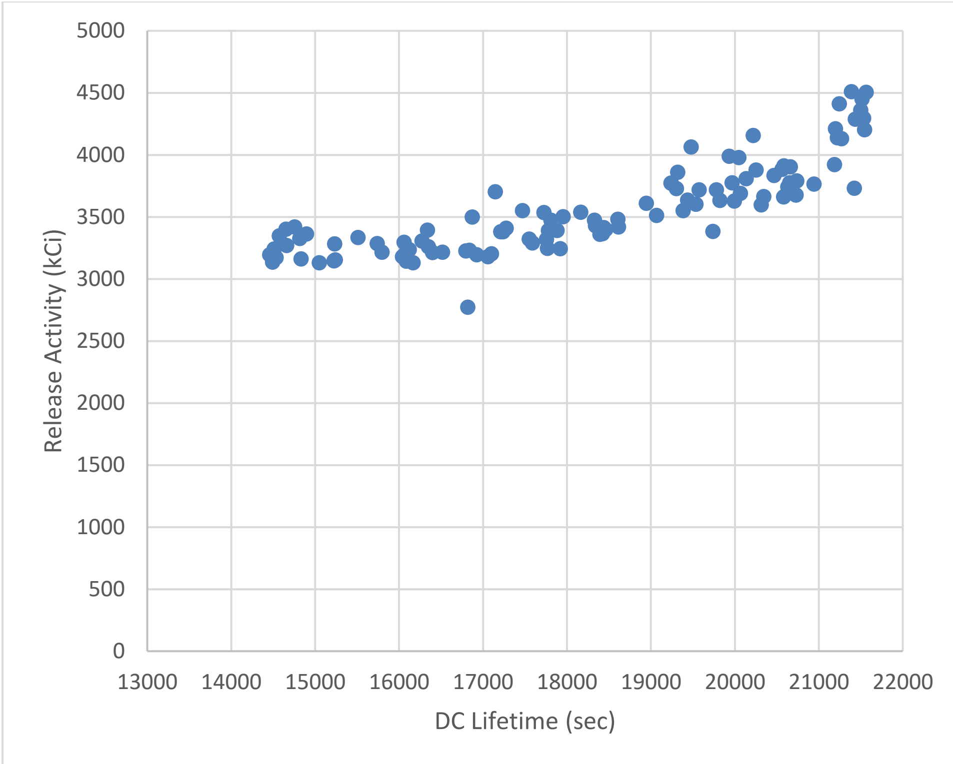


Figure 24. DC Lifetime vs Release Activity, No Upgrades

Interestingly, with No Upgrades and with only the Manual ADS upgrade, the Release Activity trended upward with increasing DC Lifetime, as seen in Figure 23 and Figure 24. The physical mechanism behind this trend is that the longer DC Lifetime increases the pressure and amount of steam present in the Wetwell, slowing the hydrogen burn and reducing the pressure spike created within the wetwell. This in turn reduces the amount of water forced backwards through the downcomers from the wetwell into the drywell. This, in turn, reduces the amount of water present in the drywell when radioactive aerosols begin to relocate through the drywell, eventually passing to the environment through the ruptured containment. With less water present in the drywell, fewer aerosols are scrubbed, greatly increasing the Release Activity of, in particular, the Molybdenum released. Through similar mechanisms, the Manual ADS upgrade, without the accompanying IRWST upgrade, actually increased the Release Activity in many cases sampled to have long DC Lifetimes.

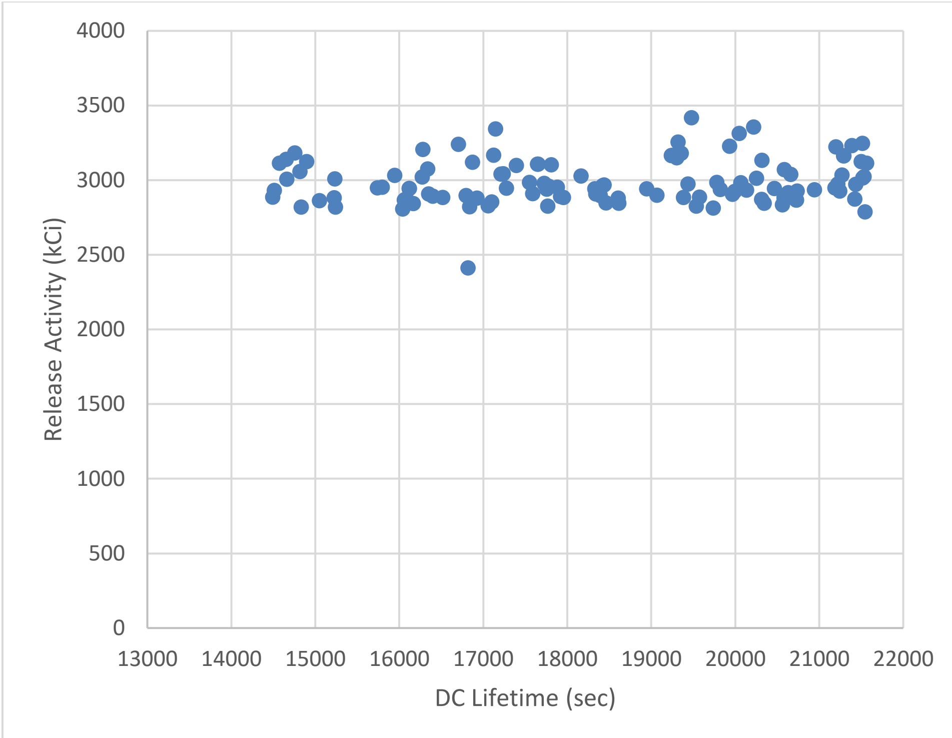


Figure 25. DC Lifetime vs Release Activity, IRWST Upgrade

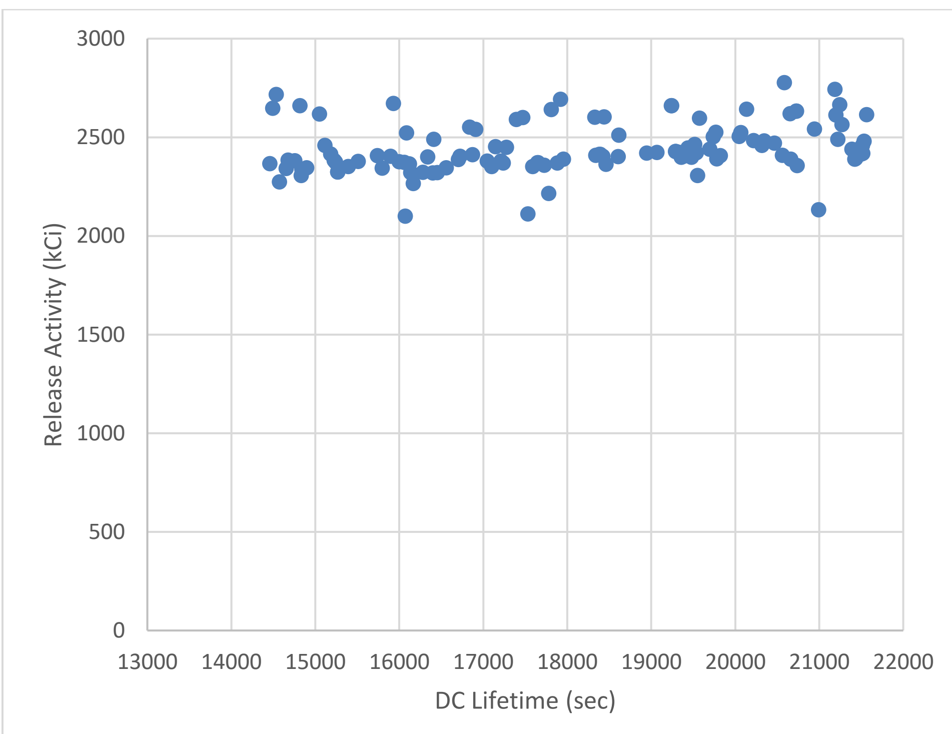


Figure 26. DC Lifetime vs Release Activity, Manual ADS and IRWST Upgrades

As seen in Figure 25 and Figure 26 the IRWST upgrade configuration and the Manual ADS and IRWST upgrade configuration made the Release Activity essentially completely insensitive to the DC Lifetime. Even when the DC power failed before the reactor was depressurized to allow the IRWST to be used before the onset of core damage, the eventual failure of the lower head of the RPV depressurizes the RPV and allows the IRWST to quench the molten corium, reducing the release activity. Adding the Manual ADS upgrade to allow the IRWST to add water to the core earlier reduced the Release Activity significantly by delaying the onset of core degradation and preventing some core relocation and radionuclide release.

## 5.2. *Individual Upgrade Impact EDG Sensitivity Study*

To guide later analysis, an initial examination of the impact of each upgrade was performed. For the purposes of these upgrades, conservative values were used for all stochastic parameters but the Emergency Diesel Generator Failure Time, which was varied across a significantly wider range than was used in later analyses, for the sake of a thorough examination.

### 5.2.1. Individual Upgrade Impact EDG Sensitivity Study Results Summary

Table 18 presents a summary of the results of the Individual Upgrade Impact Analysis. The upgrades are shown, ranked from least Mean Release Activity to greatest Mean Release Activity, as well as an identical analysis with no upgrades for comparison. The release activity is in units of kilocuries (kCi). 125 cases of MELCOR, each with the EDG Failure Time randomly sampled, were run for each upgrade configuration.

The most effective upgrade examined was easily the Aerosol Filtered Containment Venting System, with an expected release reduction of nearly 45%. The expected release activity for the Aerosol Filter configuration was 1713.1 +/- 370.0 kCi, while the expected release activity for the configuration with no upgrades was 3092.7 +/- 253.7 kCi, showing a marked and drastic reduction in the release activity. More detailed examinations of each upgrade analysis follow in the next sections

Table 18. Summarized Results of Individual Upgrade Impact EDG Sensitivity Study

| Configuration | Mean Release Activity | Median | Max | Sigma | 5th Percentile | 95th Percentile | Skewness | Kurtosis | Release Activity, EDG Failure Time Covariance | Pearson |
|---|---|---|---|---|---|---|---|---|---|---|
| Aerosol Filter | 1713.119 | 1743.21 | 2624.34 | 123.3283 | 1553.55 | 1743.21 | -12.6193 | 38.3036 | -138982 | -0.01436 |
| Vents | 2360.362 | 2391.22 | 2640.25 | 125.8475 | 2200.3 | 2391.22 | -13.0623 | 123.5748 | -146534 | -0.01484 |
| ATF | 2695.318 | 2678.33 | 2929.98 | 101.3707 | 2678.33 | 2807.91 | 44.0914 | 2642.456 | 156558.9 | 0.019912 |
| IRWST | 2886.764 | 2909.14 | 3130.99 | 52.45376 | 2843.82 | 2909.14 | -5.18958 | 49.90606 | -60284.2 | -0.01412 |
| ADS | 3002.992 | 2978.64 | 3149.02 | 40.68331 | 2978.64 | 3045.7 | 1.616496 | -3.02357 | 23470.39 | 0.007096 |
| PAR | 3052.411 | 2975.08 | 3346.34 | 167.4533 | 2966.19 | 3254.45 | 2.754481 | -2.51529 | 34997.89 | 0.00258 |
| No Upgrades | 3092.657 | 3131.55 | 3354.14 | 84.58144 | 2997.38 | 3131.55 | -2.91139 | -3.66035 | -88965.7 | -0.01301 |
| Igniter | 3092.657 | 3131.55 | 3354.14 | 84.58144 | 2997.38 | 3131.55 | -2.91139 | -3.66035 | -88965.7 | -0.01301 |

### 5.2.2. No Upgrades

For the sake of comparison to the single upgrade configurations, a model configuration with no upgrades implemented was analyzed. Figure 27 shows a plot of the Emergency Diesel Generator Time vs the Release Activity (in kCi) for every scenario examined in the No Upgrades configuration examination.

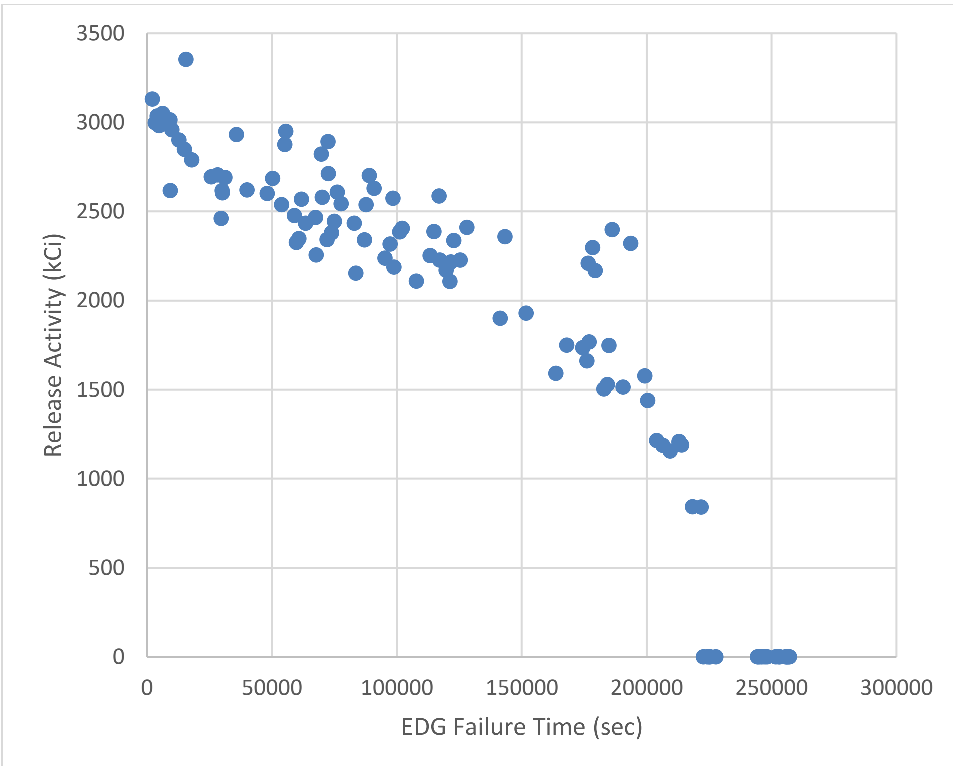


Figure 27. No Upgrades Analysis for comparison with Individual Upgrade Impact Analyses

As seen in Figure 27, with no upgrades, the Release Activity shows a fair amount of sensitivity to the EDG Failure Time, decreasing steadily with increased EDG Failure Time.

### 5.2.3. Accident Tolerant Fuels

Figure 28 shows a plot of the Emergency Diesel Generator Time vs the Release Activity (in kCi) for every scenario examined in the Accident Tolerant Fuels configuration examination.

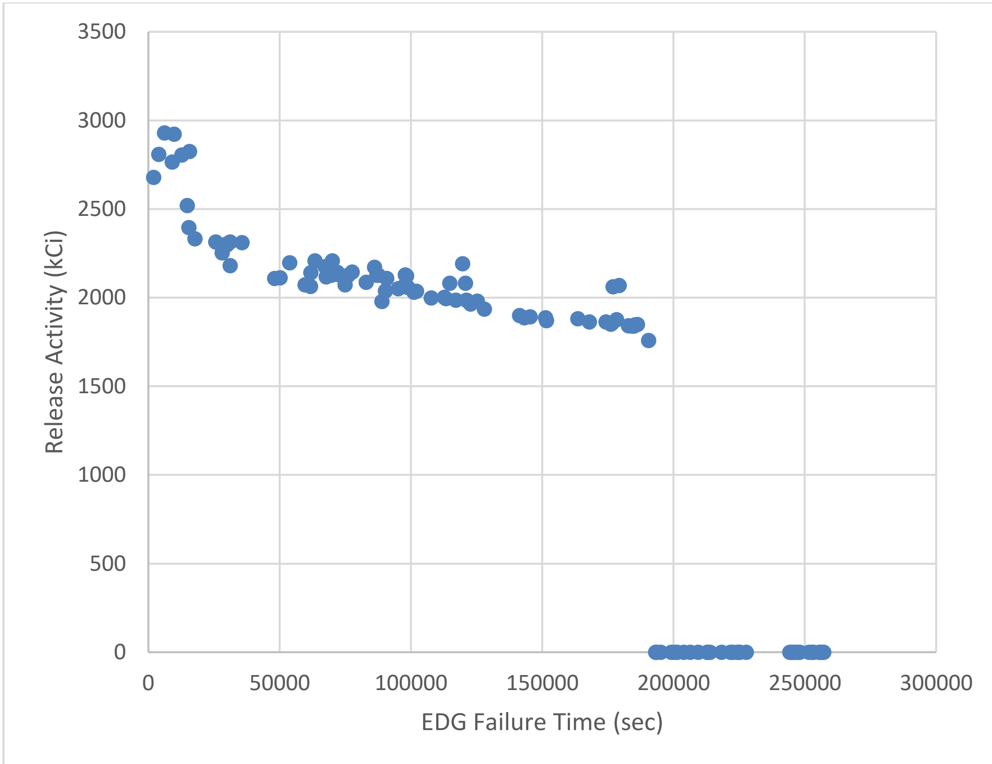


Figure 28. Accident Tolerant Fuel Individual Upgrade Impact Analysis

As seen in Figure 28, the release activity trended downward with increasing EDG Failure Time. At 190,000 seconds sampled for the EDG Failure Time, the release activity dropped to 0 – no release – because the accident scenario no longer caused high enough temperatures to fail the cladding before the end of the simulation. As per the No Upgrades scenario, this is because the simulation is ending before the onset of radionuclide release. It is of interest that the ATF upgrade delayed the onset of radionuclide release so greatly. Due to the limitations of the modeling for Accident Tolerant Fuels this was not anticipated, as the ATF model used only examined the effects of the ATF upgrade on hydrogen generation, as discussed in Section 4.3, by altering the minimum cladding oxidation temperature and did not account for the potential impact of Accident Tolerant Fuels on the physical failure of cladding. It is likely that the internal models of MELCOR are impacted in other ways by this change in minimum oxidation temperature and the lack of oxidation at fuel cladding temperatures that would otherwise lead to significant oxidation.

The presence of the ATF upgrade made the Release Activity less sensitive to the EDG Failure Time. This conclusion is not borne out in the statistics presented in Section 5.2.1, however as noise present within the data was enormously magnified by the probabilistic weighting, which very heavily weights the data in favor of short (under six hours) EDG Failure Times. A flaw in the analysis here was discovered as a result. Using uniform stochastic parameter sampling, while excellent for ensuring that less likely scenarios are reasonably well sampled without taking potentially millions of sample points (as discussed previously, the anticipated probability of the EDG Failure Time being greater than six hours is on the order of 1E-10), can lead to heavily weighted areas being potentially under sampled. Later analyses corrected this by using both more sample points and restricting the sample range to not include scenarios with essentially zero probability of occurring, in an effort to prevent the waste of computing resources on essentially impossible scenarios.

### 5.2.4. Hardened Containment Ventilation

Figure 29 shows a plot of the Emergency Diesel Generator Time vs the Release Activity (in kCi) for every scenario examined in the Hardened Containment Ventilation configuration examination.

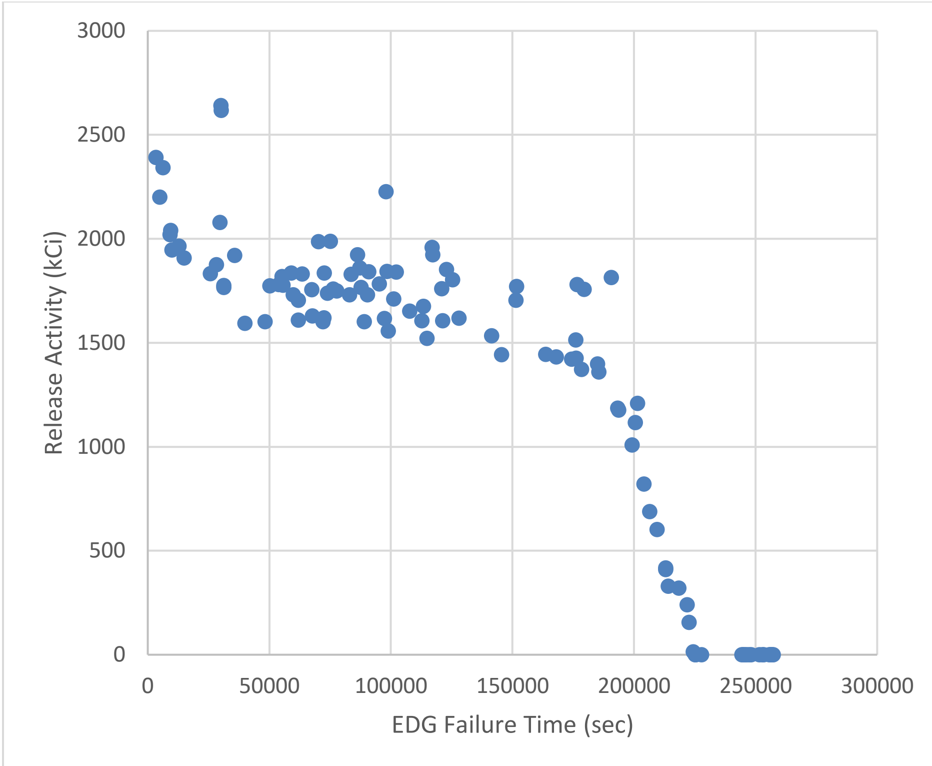


Figure 29. Hardened Containment Ventilation Individual Upgrade Impact Analysis

As in Figure 29, the data appears to suggest a potential downward trend in Release Activity with increasing EDG Failure Time, but the results are noisy enough that a trend is unclear. As with the ATF results, around the 190,000 mark for the EDG Failure Time, the release activity begins to drop sharply. Further investigation bore out that this was indeed simply because the simulation was being cut short before the bulk of the eventual radionuclide release could occur. However, comparing the results between these two scenarios, despite the similar appearance of the graphs, closer examination of Figure 28 and Figure 29 shows that the magnitude of release, across almost all EDG Failure Times, was significantly diminished in the HCVS configuration compared to the No Upgrades configuration.

#### 5.2.5. Filtered Containment Ventilation Filter Comparison

To examine the efficacy of various kinds of potential filtration systems, Aerosol Filters, Vapor Filters, and both kinds of filters together were compared to an Unfiltered Hardened

Containment Venting system and a No Upgrades configuration. Unfiltered Hardened Containment Venting provided a significant reduction in the Release Activity expected value, with an expected Release Activity of 2360.4 +/- 377.5 kCi, compared to the No Upgrades configuration's expected Release Activity of 3092.7 +/- 253.7 kCi.  Compared to the basic unfiltered vents, adding a Vapor Filter was not significantly effective, providing only a reduction to 2208.6 +/- 140.1 kCi, which is within the 95% confidence interval for the expected value of the Release Activity for the unfiltered vents. The Aerosol Filter addition to the vents was much more effective and provided a clear and significant reduction in the expected Release Activity. As with the Vapor Filter compared to the unfiltered vents, using both filters together did not provide a significant reduction in the expected Release Activity compared to using the Aerosol Filter alone. These results are presented in Table 19.

Table 19. Filtered Containment Ventilation Filter Comparison

| Configuration | Mean Release Activity | Median | Max | Sigma | 5th Percentile | 95th Percentile | Skewness | Kurtosis | Release Activity, EDG Failure Time Covariance | Pearson |
|---|---|---|---|---|---|---|---|---|---|---|
| Both Filter | 1561.36 | 1571.6 | 2622.7 | 41.390 | 1523.62 | 1571.6 | -15.8644 | 716.3752 | -59669.5 | -0.01837 |
| Aerosol Filter | 1713.1 | 1743.2 | 2624.3 | 123.32 | 1553.55 | 1743.21 | -12.6193 | 38.3036 | -138982 | -0.01436 |
| Vapor Filter | 2208.6 | 2219.6 | 2638.6 | 46.732 | 2170.37 | 2219.61 | -21.0912 | 1866.097 | -67222.1 | -0.01833 |
| No Filter | 2360.3 | 2391.2 | 2640.2 | 125.85 | 2200.3 | 2391.22 | -13.0623 | 123.5748 | -146534 | -0.01484 |
| No Upgrades | 3092.7 | 3131.5 | 3354.1 | 84.580 | 2997.38 | 3131.55 | -2.911 | -3.6604 | -88965.7 | -0.013 |

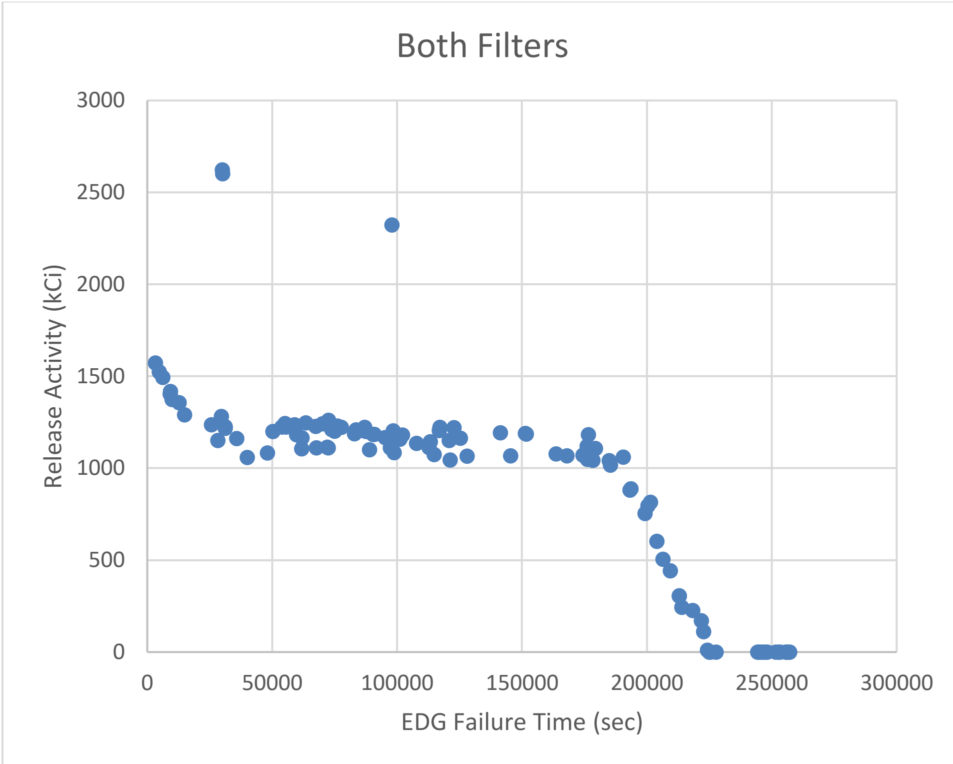


Figure 30. Aerosol and Vapor Filtered Ventilation Initial Upgrade Impact Assessment

As seen in Figure 30, both filters together nearly flattened the Release Activity for the majority of examined cases, with Xenon comprising essentially the entirety of the Release Activity. Xenon was set to pass through the Vapor Filter unimpeded to reflect the difficulties of filtering against noble gases and was found to be largely insensitive to the sampled EDG Failure Time.

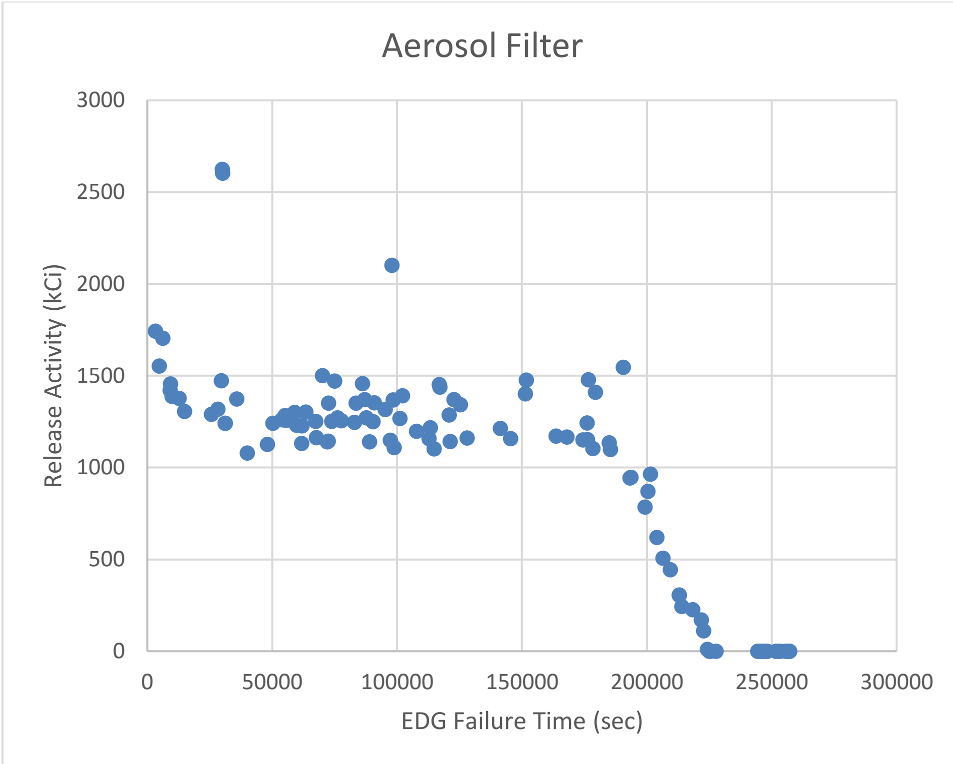


Figure 31. Aerosol Filtered Containment Ventilation Initial Upgrade Impact Assessment

It can be seen in Figure 31 that the Aerosol Filtered Containment Ventilation system both greatly reduced the magnitude of the Release Activity and also made the Release Activity fairly insensitive to the EDG Failure Time for many of the samples. It appears possible that very short EDG Failure Times, under two hours, correspond to an increase in the Release Activity, but the data is inconclusive. As with the previous tests, very long EDG Failure Times, around sixty hours and longer, lead to a drastic reduction in Release Activity. Further investigation showed that the insensitivity to the EDG Failure Time resulted from Xenon and other vaporous radionuclides dominating the Release Activity in many cases. Additionally, the Aerosol Filtered Containment Ventilation system provided yet more reduction in Release Activity than the Hardened Containment Ventilation system.

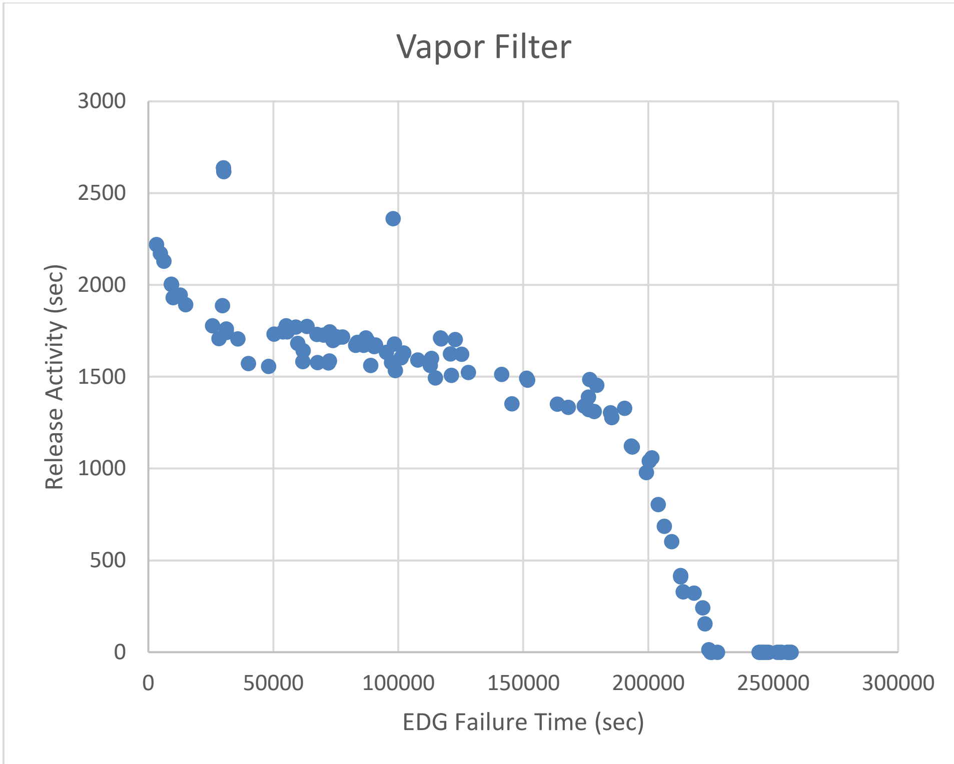


Figure 32. Vapor Filtered Containment Ventilation Initial Upgrade Impact Assessment

Comparing the Vapor Filtered Ventilation results in Figure 32 to the Unfiltered Ventilation below in Figure 33, most of the impact of the Vapor Filter was to ‘flatten’ out some of the noise in the data seen in Figure 33. Other that suppressing fluctuations in the Release Activity from variations the quantity of vapor radionuclides released, the Vapor Filter did not have a major impact on the Release Activity in most scenarios examined.

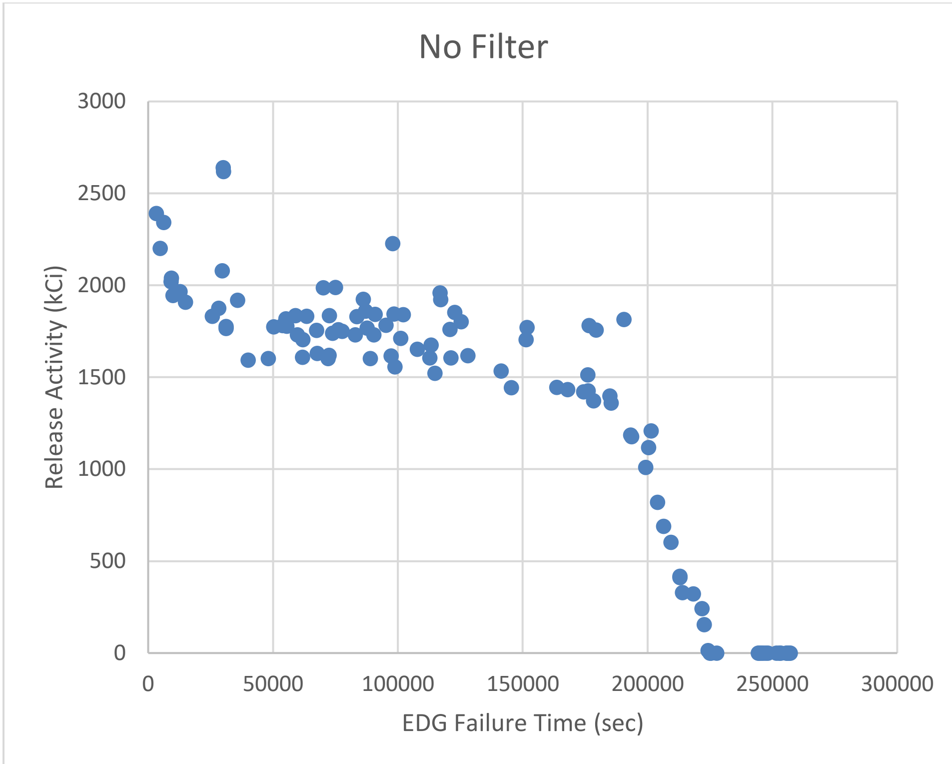


Figure 33. Unfiltered Containment Ventilation Initial Upgrade Impact Assessment

The Unfiltered Containment Ventilation is identical to the Hardened Containment Ventilation and is shown here for the sake of comparison to the filtered cases. The impact of this upgrade is discussed above, in Section 5.2.4.

### 5.2.6. Hydrogen Igniters

Figure 34 shows a plot of the Emergency Diesel Generator Time vs the Release Activity (in kCi) for every scenario examined in the Hydrogen Igniter configuration examination.

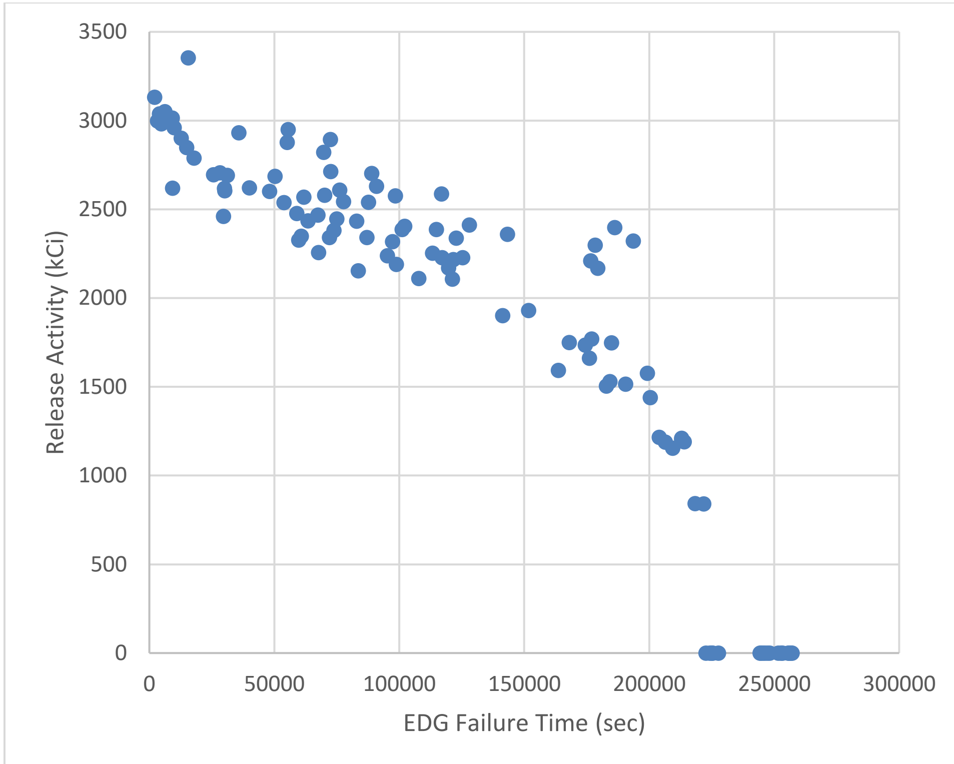


Figure 34. Hydrogen Igniter Individual Upgrade Impact Analysis

Comparing Figure 27, the No Upgrade configuration results, to Figure 34, the Hydrogen Igniter Individual Upgrade Impact Analysis results, the Hydrogen Igniters made no difference whatsoever in the outcomes of any scenarios. This conclusion is borne out by the results in Table 18, which shows identical values for the two configurations. It was verified that the input decks used for these configurations were appropriately created, and no error was made. Using conservative values for all stochastic parameters but the EDG Failure Time meant that power was never recovered, preventing the Hydrogen Igniter upgrade from ever functioning, resulting in a configuration that was functionally identical to the No Upgrades configuration. With this in mind, further analysis of the Hydrogen Igniters included an assumption that (at additional cost) a redundant power supply for the Hydrogen Igniters would be implemented to allow them to function even during an LT-SBO.

### 5.2.7. In-Containment Refueling Water Storage Tank

Figure 35 shows a plot of the Emergency Diesel Generator Time vs the Release Activity (in kCi) for every scenario examined in the In-Containment Refueling Water Storage Tank configuration examination.

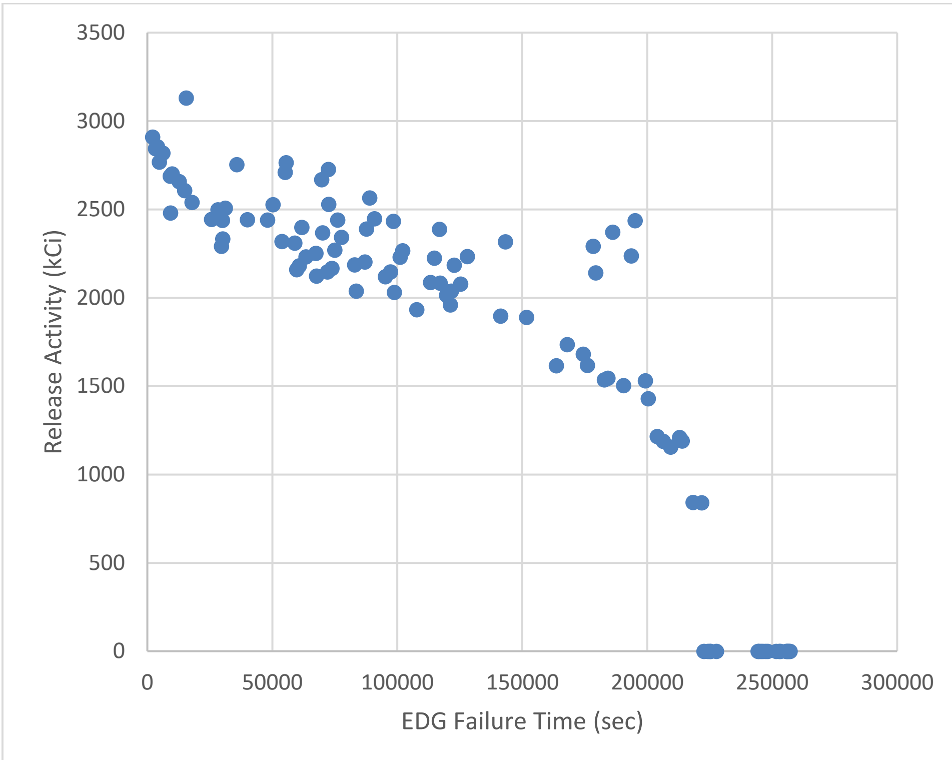


Figure 35. In-Containment Refueling Water Storage Tank Individual Upgrade Impact Analysis

Comparing Figure 27, the results for the No Upgrade configuration, and Figure 35, the results for the IRWST configuration, the IRWST upgrade may have provided a minor reduction in the Release Activity of the scenario. However, with an expected value of 2,886.8 +/- 157.4 kCi, compared to the No Upgrade configuration's 3092.7 +/- 253.7, no conclusive evidence of a reduction in Release Activity can be had.

### 5.2.8. Manual ADS Actuation

Figure 36 shows a plot of the Emergency Diesel Generator Time vs the Release Activity (in kCi) for every scenario examined in the Manual ADS Actuation configuration examination.

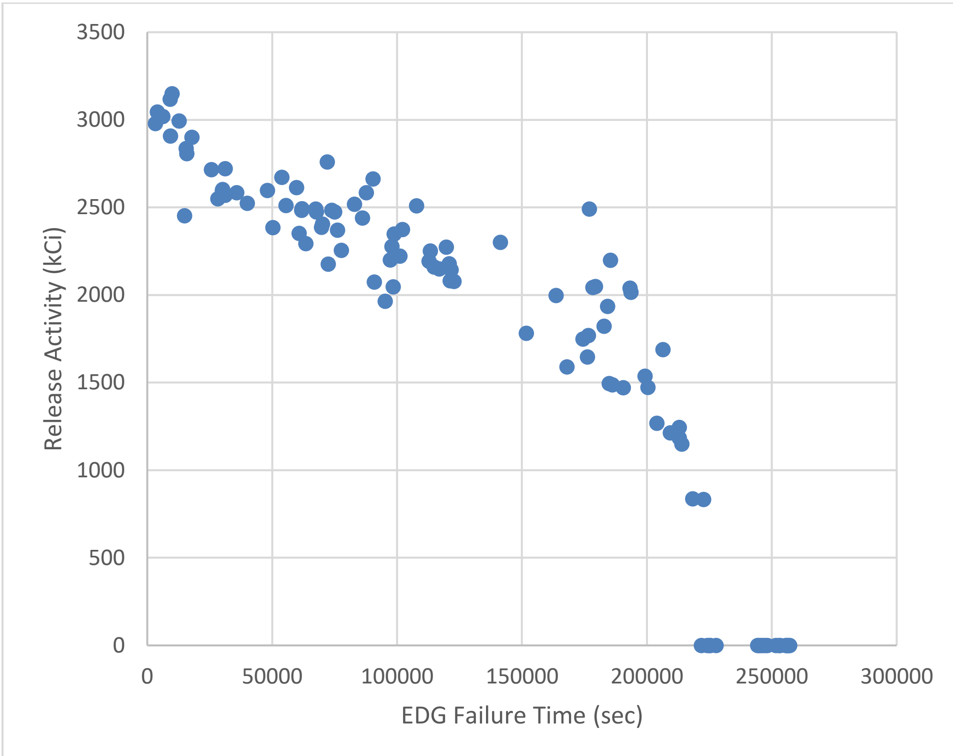


Figure 36. Manual ADS Actuation Individual Upgrade Impact Analysis

As seen in Table 18 and by comparing Figure 27 and Figure 36, the Manual ADS upgrade, by itself, appears to have had essentially no impact on the Release Activity in each scenario. This is in line with engineering judgement regarding the upgrade, as it is primarily intended as an accompanying piece to increase the efficacy of the IRWST upgrade. The results of an initial analysis of the impact of these two upgrades in tandem are presented in Section 5.3.4.

### 5.2.9. Passive Autocatalytic Recombiners

Figure 37 shows a plot of the Emergency Diesel Generator Time vs the Release Activity (in kCi) for every scenario examined in the Passive Autocatalytic Recombiners configuration examination

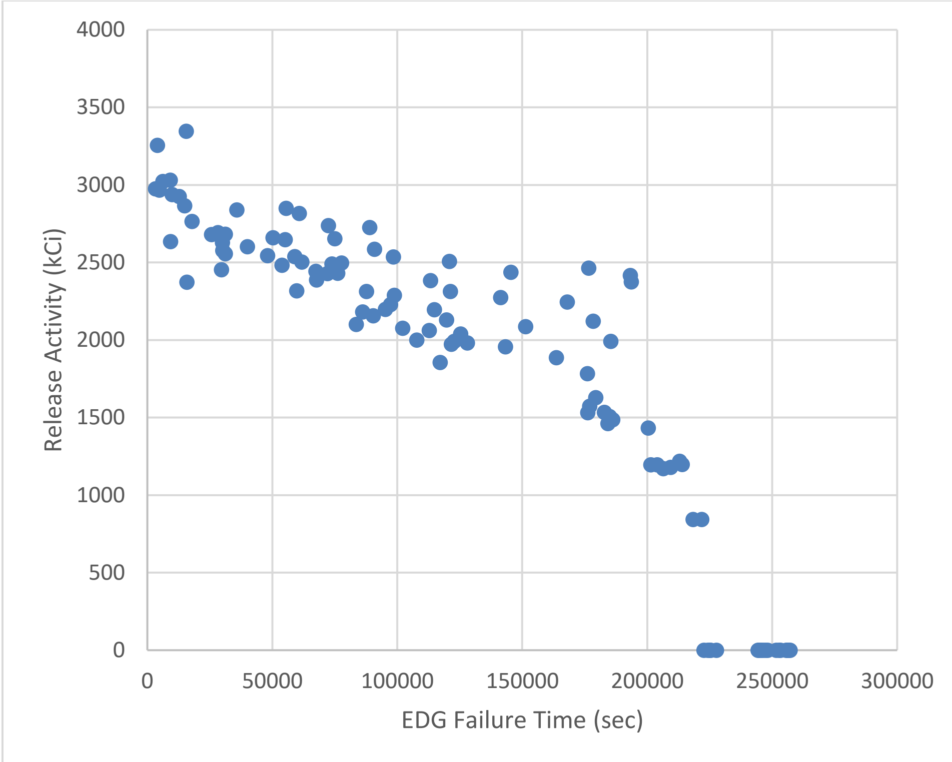


Figure 37. Passive Autocatalytic Recombiner Individual Upgrade Impact Analysis

It was initially expected that the PAR upgrade, even alone, would have a significantly more noteworthy impact than was found in this analysis. The PAR upgrade was not observed to have any serious, discernible impact on the expected Release Activity in the Station Blackout scenario examined. Its affects in tandem with other upgrades – notably a containment ventilation system, filtered or not – were more pronounced, as discussed in Section 5.3.6.

## 5.3. *Upgrade Configuration Impact EDG Sensitivity Study*

### 5.3.1. Accident Tolerant Fuels

Initial examination of the ATF upgrade in combination with other upgrades was largely inconclusive, and indicative of potential problems with the highly simplified implementation of the physical mechanisms of a real-world ATF upgrade. Summary results of this test are shown in Table 20.

Table 20. Accident Tolerant Fuel Upgrade Configuration Impact EDG Sensitivity Study

| Configuration | Mean Release Activity | Median | Max | Sigma | 5th Percentile | 95th Percentile | Skewness | Kurtosis | Release Activity, EDG Failure Time Covariance | Pearson |
|---|---|---|---|---|---|---|---|---|---|---|
| ATF, ADS, IRWST | 2326.0 | 2319.2 | 2432.8 | 23.00 | 2319.2 | 2342.7 | -915.36 | 4461124.9 | 5370.9 | 0.0034 |
| ATF | 2695.3 | 2678.3 | 2929.9 | 101.37 | 2678.3 | 2807.9 | 44.09 | 2642.5 | 156558.9 | 0.0199 |
| No Upgrades | 3092.7 | 3131.6 | 3354.1 | 84.58 | 2997.4 | 3131.6 | -2.911 | -3.6604 | -88965.7 | -0.0130 |
| ATF, ADS, IRWST, Vents | 3698.3 | 3698.6 | 3771.4 | 304.31 | 3698.6 | 3698.6 | -5462.0 | 342637.5 | -1180950.1 | -0.0538 |
| ATF, Vents | 6988.3 | 7022.5 | 7022.5 | 2124.7 | 7022.5 | 7022.5 | -2345.7 | -306511.0 | -6273148.0 | -0.0393 |

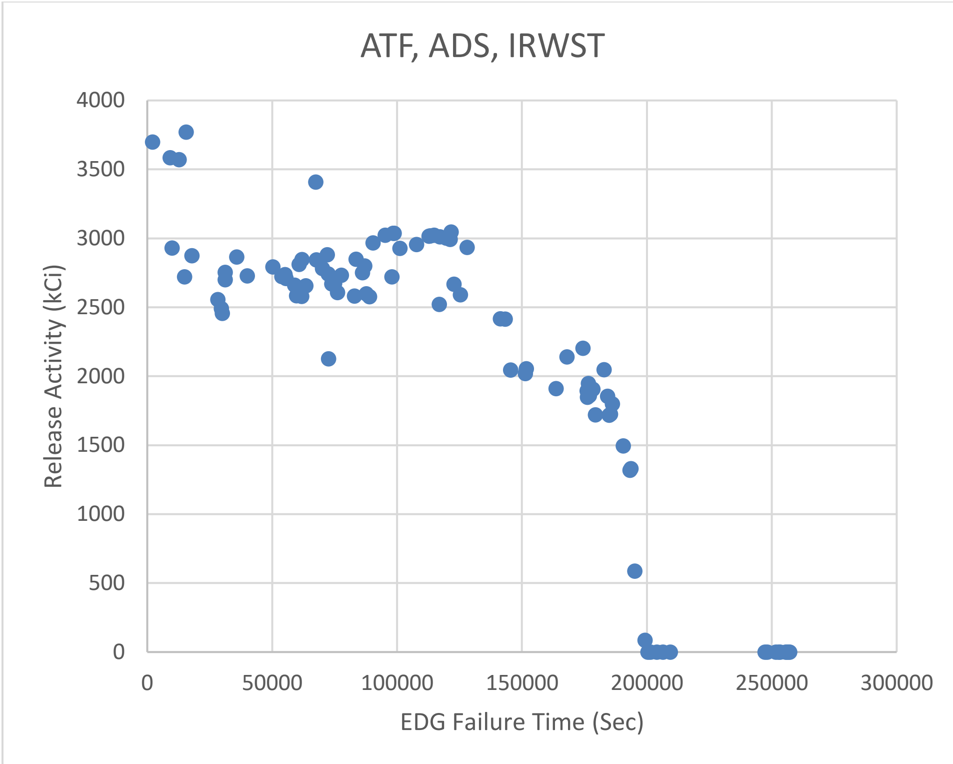


Figure 38. ATF, ADS, and IRWST Upgrade Combination Initial Impact Assessment Results

Shown in Figure 38, the combination of ATF, ADS, and IRWST showed significant reduction compared to No Upgrades, but was not a major improvement over the ADS and IRWST upgrades without the ATF upgrade (Section 5.3.4). The ATF upgrade itself does not appear, here, to have had a significant impact.

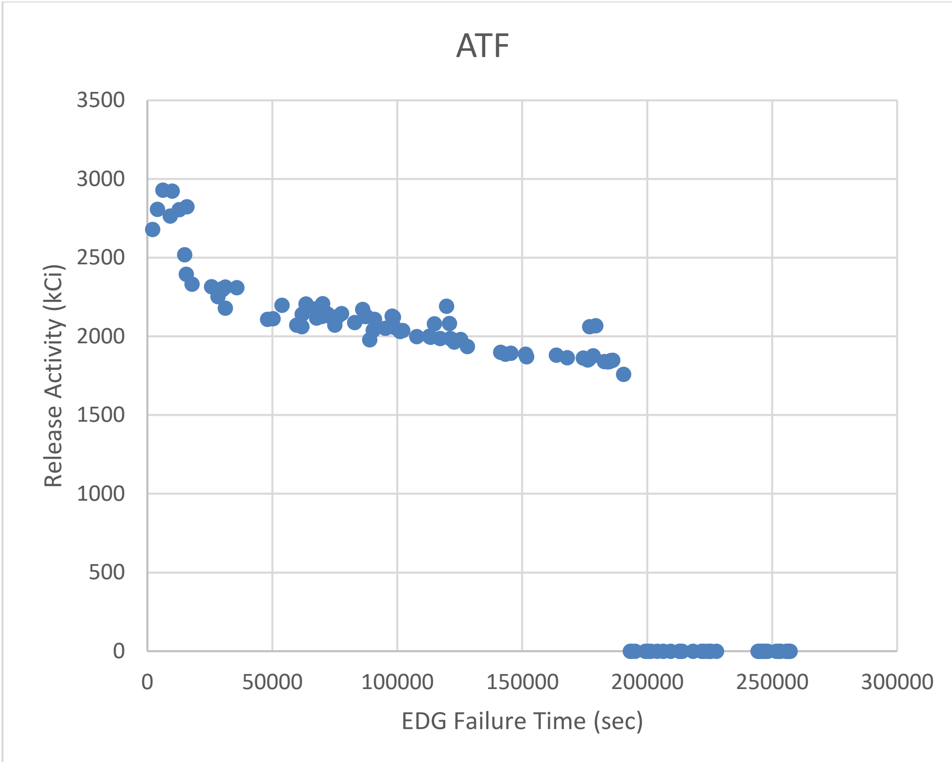


Figure 39. ATF Upgrade Initial Impact Assessment Results

The ATF upgrade alone provided results, shown in Figure 39 that suggest a reduction in Release Activity, but there is overlap between the 95% confidence uncertainty interval around the ATF upgrade results (2695.3 +/- 304.11 kCi) and the No Upgrades results (3092.7 +/- 253.74 kCi), so the results are somewhat inconclusive. Additionally, in problems with very short EDG Failure Times, the ATF upgrade did not have a major impact compared to No Upgrades, and most of the gains were in scenarios where the EDG sample time was sampled as 8 hours or longer.

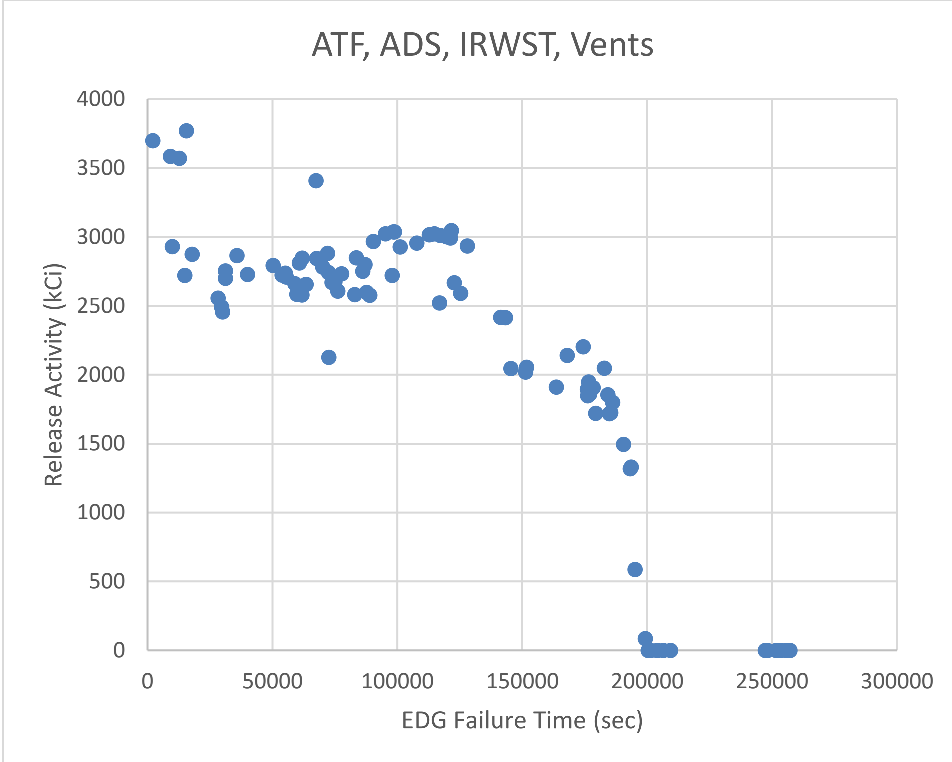


Figure 40. ATF, ADS, IRWST, and Hardened Vents Upgrade Combination Initial Impact Assessment Results

The ATF, ADS, IRWST, and Vents configuration seemed to potentially make the problem worse, bizarrely, as seen in Figure 40, but as discussed previously, the methodology used for these initial examinations was discovered to be overly vulnerable to statistical noise in the results because of the heavy weighting of a small portion of the sampling space, functionally reducing the number of sample points. It was later discovered that these points were not simple erroneous points, but a flaw in the highly simplistic ATF model. This is discussed in more detail in Section 5.4.1, in the discussion regarding Figure 62.

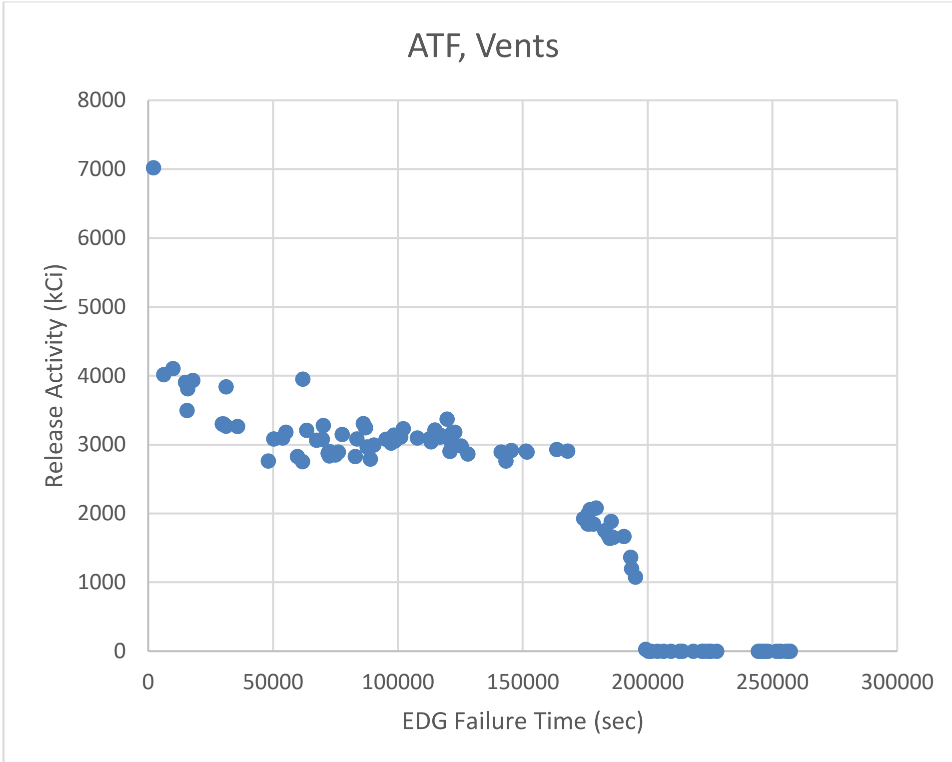


Figure 41. ATF and Vents Upgrade Combination Initial Impact Assessment Results

The problems in this test and the ATF, ADS, IRWST, and Vents test is reflected in the abnormally high Sigma value for both tests, with the 95% confidence uncertainty interval for the ATF, ADS, IRWST, and Vents expected value Release Activity being 3698.3 +/- 912.9 kCi and the ATF and Vents expected value Release Activity being 6988.3 +/- 6373.85 kCi, these latter results being essentially meaningless.

### 5.3.2. Hardened Containment Ventilation

Table 21 presents a summary of the results of the Hardened Containment Ventilation upgrade configuration EDG sensitivity study. The Hardened Containment Vents, by themselves, provided a significant reduction in Release Activity. Adding the IRWST, without the Manual ADS, on top of the Hardened Containment vents did little, but testing the IRWST, Manual ADS, and Hardened Containment Vents together provided a major reduction in Release Activity.

Table 21. Hardened Containment Ventilation Upgrade Configuration EDG Sensitivity Study

| Configuration | Mean Release Activity | Median | Max | Sigma | 5th Percentile | 95th Percentile | Skewness | Kurtosis | Release Activity, EDG Failure Time Covariance | Pearson |
|---|---|---|---|---|---|---|---|---|---|---|
| Vents, IRWST, ADS | 1517.9 | 1518.4 | 1865.2 | 25.135 | 1489.7 | 1537.6 | -13.148 | 162.07 | -12689.2 | -0.00678 |
| Vents | 2360.4 | 2391.2 | 2640.3 | 125.85 | 2200.3 | 2391.2 | -13.062 | 123.57 | -146534 | -0.01484 |
| Vent, IRWST | 2411.0 | 2298.4 | 2791.8 | 325.39 | 2093.7 | 2791.9 | 2.1189 | -3.0909 | 1486.5 | 5.83E-05 |
| No Upgrades | 3092.7 | 3131.6 | 3354.1 | 84.580 | 2997.4 | 3131.6 | -2.9110 | -3.6604 | -88965.7 | -0.013 |

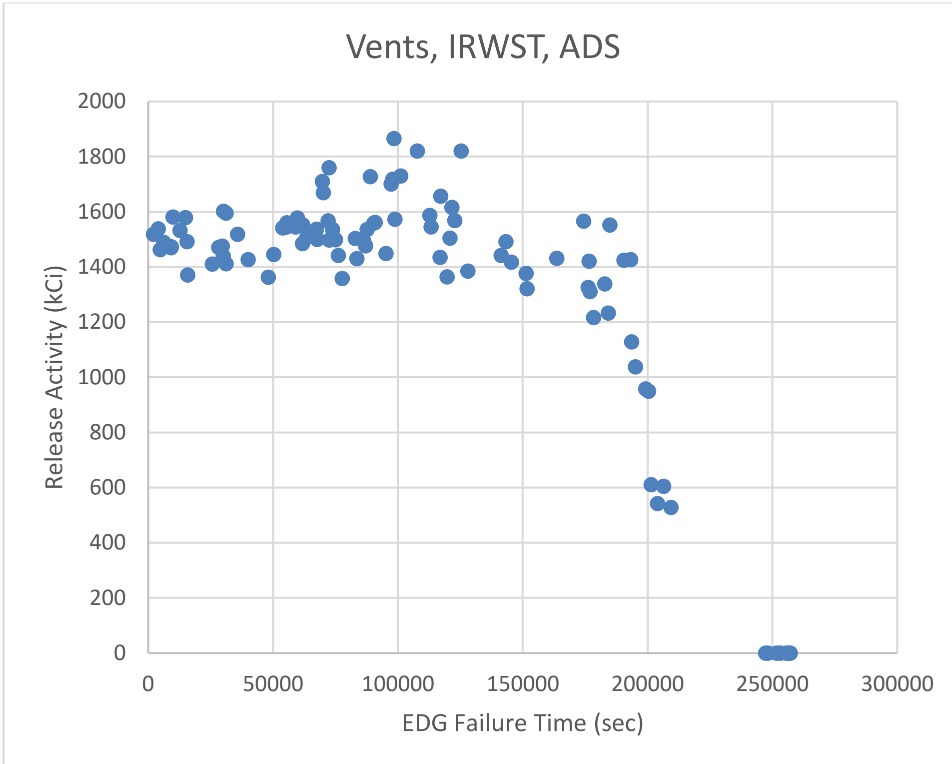


Figure 42. Vents, IRWST, and ADS Upgrade Combination Initial Impact Assessment Results

Shown in Figure 42, the combination of Hardened Vents, IRWST, and ADS provided a marked improvement over the Hardened Vents alone, or, as seen in Section 5.3.4, the ADS and IRWST without Hardened Vents. The expected value for the Vents, IRWST, and ADS configuration Release Activity (1517.9 +/- 75.4 kCi) was significantly lower than the expected value for Vents (2360.4 +/- 377.4 kCi) or the ADS and IRWST (2328.1 +/- 272.3 kCi).

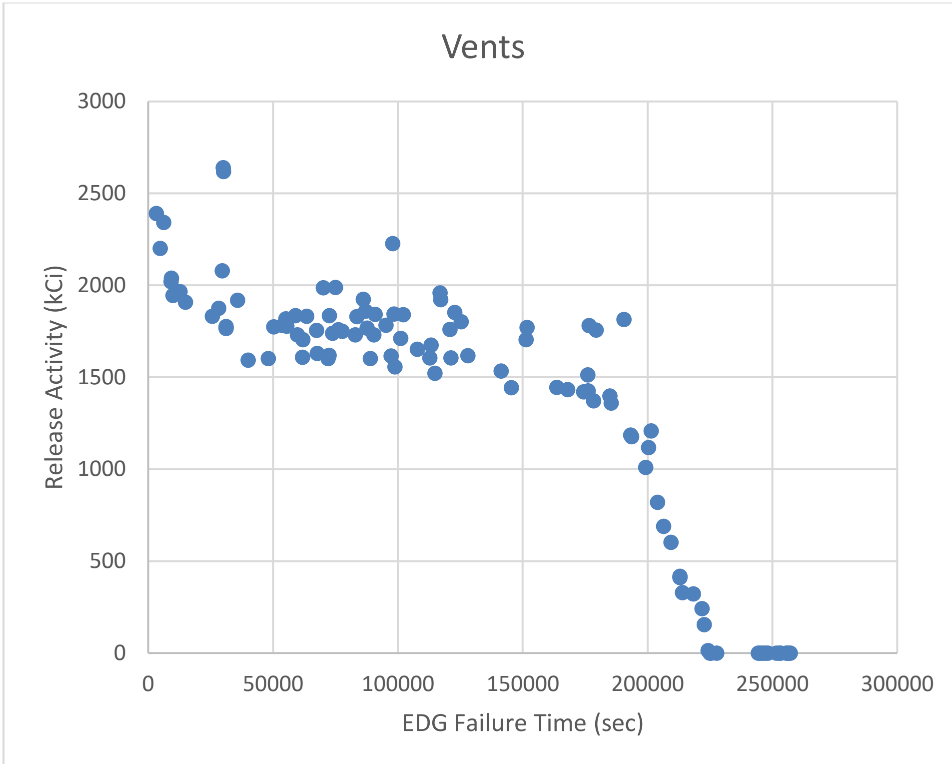


Figure 43. Vents Initial Upgrade Impact Assessment Results

Shown in Figure 43, the Hardened Vents alone also provided a significant reduction in the expected value Release Activity compared to No Upgrades. By keeping the containment depressurized and preventing a buildup of containment leading to a more violent depressurization that would have encouraged the transport of aerosolized particulate, the Release Activity was significantly reduced.

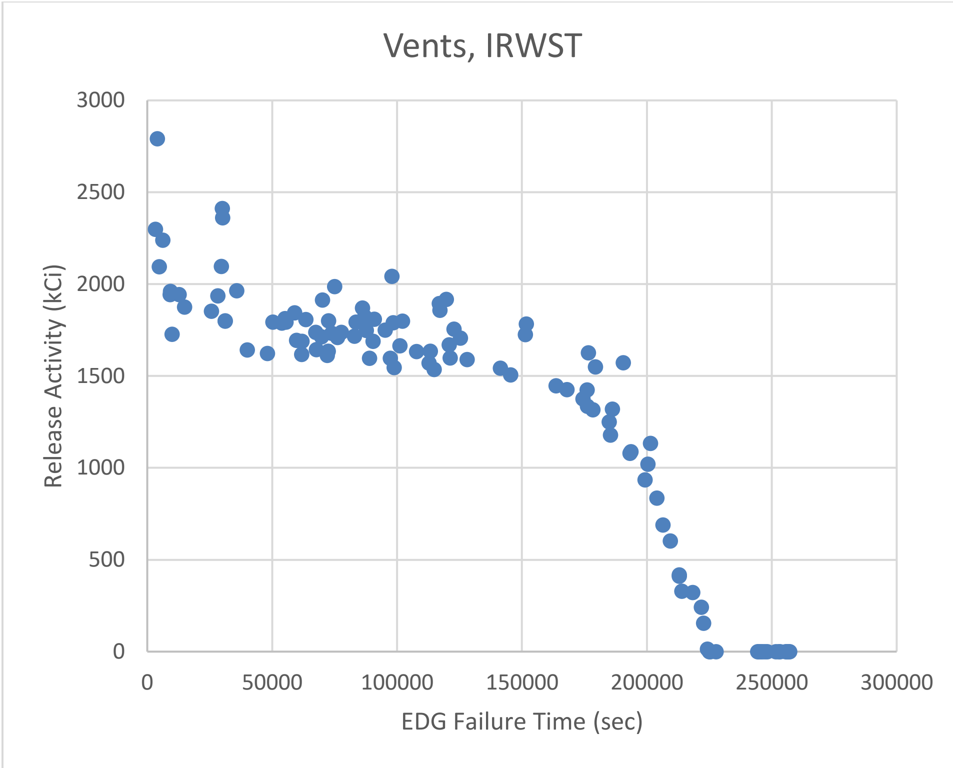


Figure 44. Vents and IRWST Upgrade Combination Initial Impact Assessment Results

The Hardened Vents and IRWST results, shown in Figure 44, indicated a reduction in Release Activity similar to the Vents alone across many cases, and via similar mechanisms. Combining the two upgrades did not provide a significant increase in reduction compared to the Hardened Vents alone and adding an aerosol filter to the ventilation system appears to be a significantly more effective way to boost the efficacy of the Hardened Vents system than adding the IRWST on top of the Hardened Vents system.

**5.3.3. Hydrogen Igniters**

Comparing upgrade combinations including Hydrogen Igniters to similar upgrade configurations only differing in through their lack of Hydrogen Igniters, the Hydrogen Igniters were found to have literally no impact on the outcome of any scenario examined. This is shown in the results presented in Table 22. The use of conservative values for stochastic parameters other than the EDG Failure Time meant that AC Power was never recovered in time to allow the

Hydrogen Igniters to operate in a timely fashion to mitigate the accumulation of hydrogen in the containment. Later analysis examined the possibility of Hydrogen Igniters with an independent power system, allowing them to function even during an SBO, at the tradeoff of a heightened cost.

Table 22. Hydrogen Igniter Upgrade Configuration EDG Sensitivity Study

| Configuration | Mean Release Activity | Median | Max | Sigma | 5th Percentile | 95th Percentile | Skewness | Kurtosis | Release Activity, EDG Failure Time Covariance | Pearson |
|---|---|---|---|---|---|---|---|---|---|---|
| Igniter, Aerosol | 1713.12 | 1743.2 | 2624.3 | 123.33 | 1553.6 | 1743.2 | -12.619 | 38.304 | -138982 | -0.01436 |
| Aerosol Filter | 1713.12 | 1743.2 | 2624.3 | 123.33 | 1553.6 | 1743.2 | -12.619 | 38.304 | -138982 | -0.01436 |
| Igniter, Vent | 2360.36 | 2391.2 | 2640.3 | 125.85 | 2200.3 | 2391.2 | -13.062 | 123.57 | -146534 | -0.01484 |
| Vents | 2360.36 | 2391.2 | 2640.3 | 125.85 | 2200.3 | 2391.2 | -13.062 | 123.57 | -146534 | -0.01484 |
| No Upgrades | 3092.66 | 3131.6 | 3354.1 | 84.581 | 2997.4 | 3131.6 | -2.9114 | -3.6604 | -88965.7 | -0.01301 |
| Igniter | 3092.66 | 3131.6 | 3354.1 | 84.581 | 2997.4 | 3131.6 | -2.9114 | -3.6604 | -88965.7 | -0.01301 |

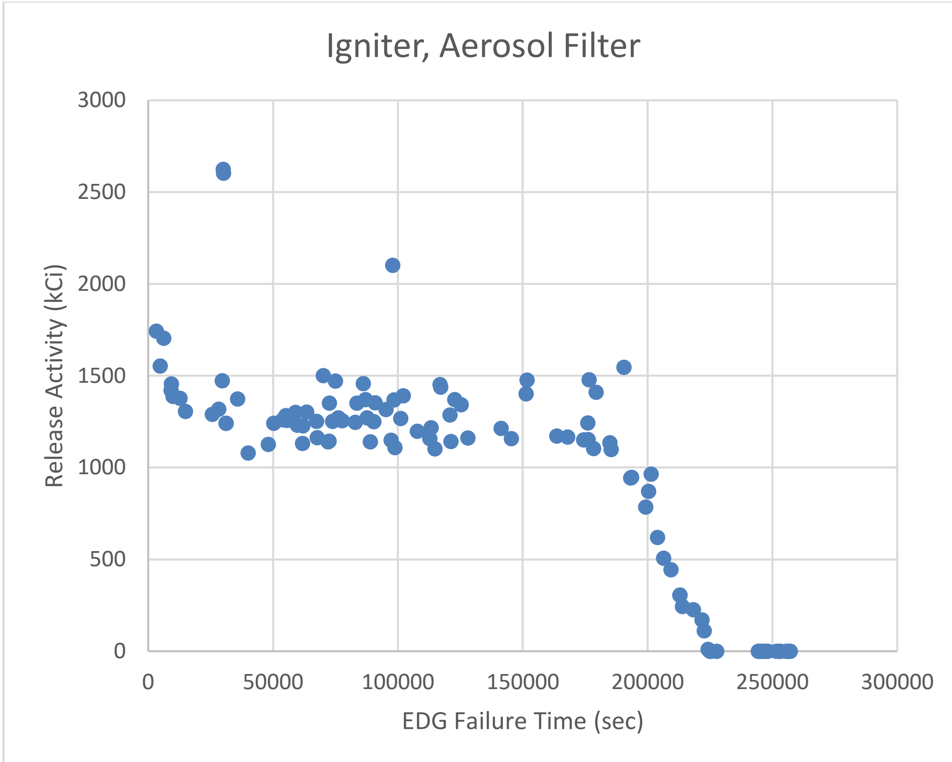


Figure 45. Hydrogen Igniter and Aerosol Filtered Containment Ventilation Initial Upgrade Combination Impact Assessment

Comparing Figure 45 and Figure 31, no difference between the two configurations can be observed in any of the sampled cases. Hydrogen Igniters were not found to have any impact on the LT-SBO with Aerosol Filtered Vents scenario compared to the same scenario without Hydrogen Igniters.

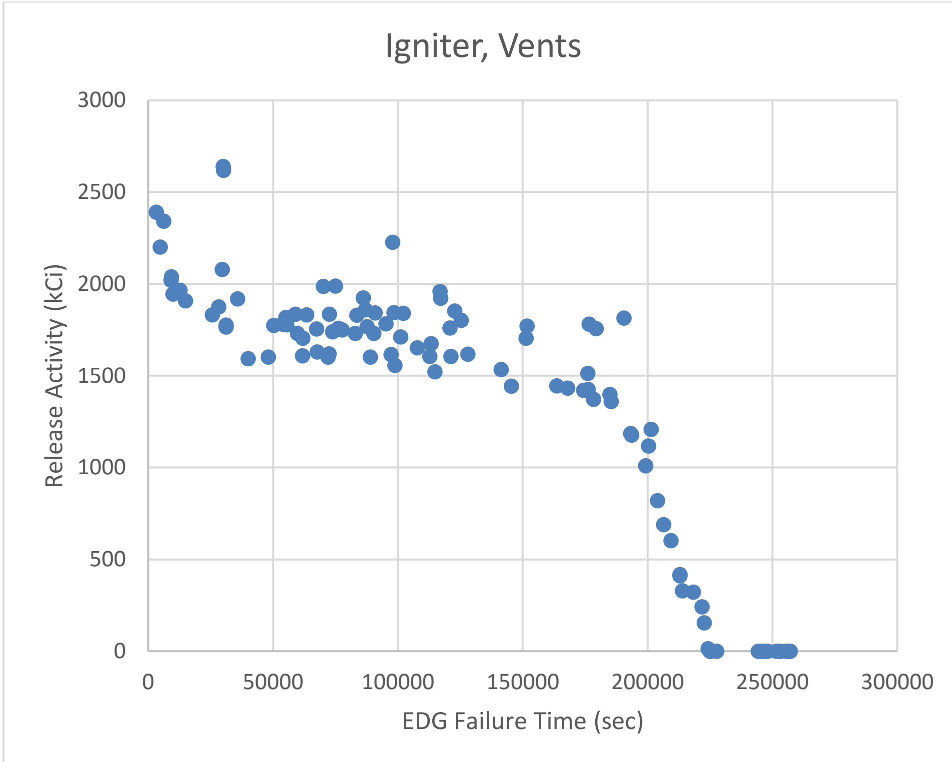


Figure 46. Hydrogen Igniter and Unfiltered Containment Ventilation Initial Upgrade Combination Impact Assessment

Comparing Figure 46 and Figure 33, no difference between the two configurations can be observed in any of the sampled cases. Hydrogen Igniters were not found to have any impact on the LT-SBO with Unfiltered Containment Vents scenario compared to the same scenario without Hydrogen Igniters.

### 5.3.4. In-Containment Refueling Water Storage Tank (IRWST)

The IRWST upgrade provided a moderate reduction in Release Activity but showed little synergistic effect with any of the upgrades it was tested alongside, other than the Manual ADS upgrade that is specifically intended to be paired with the IRWST upgrade. When tested alongside the Hardened Containment Ventilation upgrade, the IRWST and Hardened Containment Vents together provided a reduction in Release Activity similar to the Hardened Containment Vents alone. When the IRWST and PAR upgrades were tested together, they showed a reduction in Release

Activity similar to the IRWST upgrade alone, though this may simply be that the PARs had a small impact, as seen in Section 5.2.9. The summarized results of these tests are shown in Table 23.

Table 23. In-Containment Refueling Water Storage Tank Upgrade Configuration EDG Sensitivity Study

| Configuration | Mean Release Activity | Median | Max | Sigma | 5th Percentile | 95th Percentile | Skewness | Kurtosis | Release Activity, EDG Failure Time Covariance | Pearson |
|---|---|---|---|---|---|---|---|---|---|---|
| IRWST, ADS | 2328.1 | 2347.4 | 2347.4 | 90.783 | 2233.0 | 2347.4 | -13.829 | 153.28 | -146427 | -0.02166 |
| IRWST, Vent | 2411.0 | 2298.4 | 2791.9 | 325.39 | 2093.7 | 2791.9 | 2.1189 | -3.0909 | 1486.55 | 5.83E-05 |
| IRWST, PAR | 2849.3 | 2772.9 | 3077.9 | 172.24 | 2749.4 | 3057.3 | 2.7755 | -2.9034 | 28575.3 | 0.002058 |
| IRWST | 2886.8 | 2909.1 | 3131.0 | 52.454 | 2843.8 | 2909.1 | -5.1896 | 49.9061 | -60284.2 | -0.01412 |
| No Upgrades | 3092.7 | 3131.6 | 3354.1 | 84.581 | 2997.4 | 3131.6 | -2.9114 | -3.6604 | -88965.7 | -0.01301 |

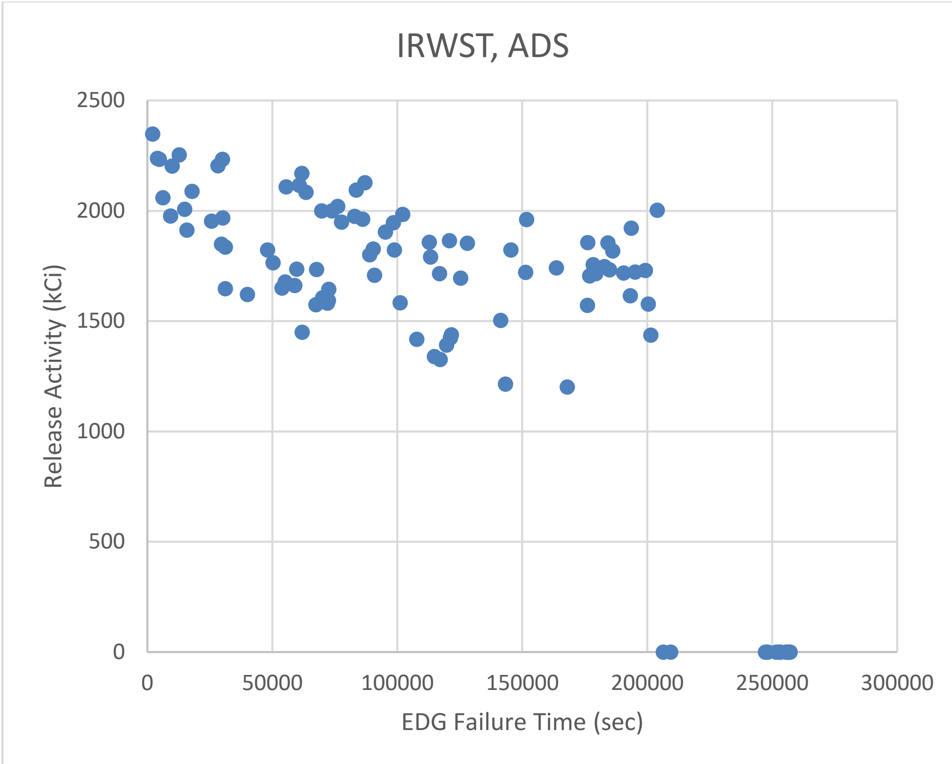


Figure 47. IRWST and Manual ADS Initial Upgrade Combination Impact Assessment

Shown in Figure 47, the IRWST and Manual ADS upgrades together were drastically more effective than the IRWST alone. The IRWST alone reduced the mean Release Activity by roughly 10%, where the IRWST and ADS together reduced the mean Release Activity by approximately 25%. Ensuring that the RPV was successfully depressurized, allowing the IRWST to always be used, meant that cold water could be added to the core well into the transient, delaying and mitigating the eventual core damage, relocation, and release of radionuclides.

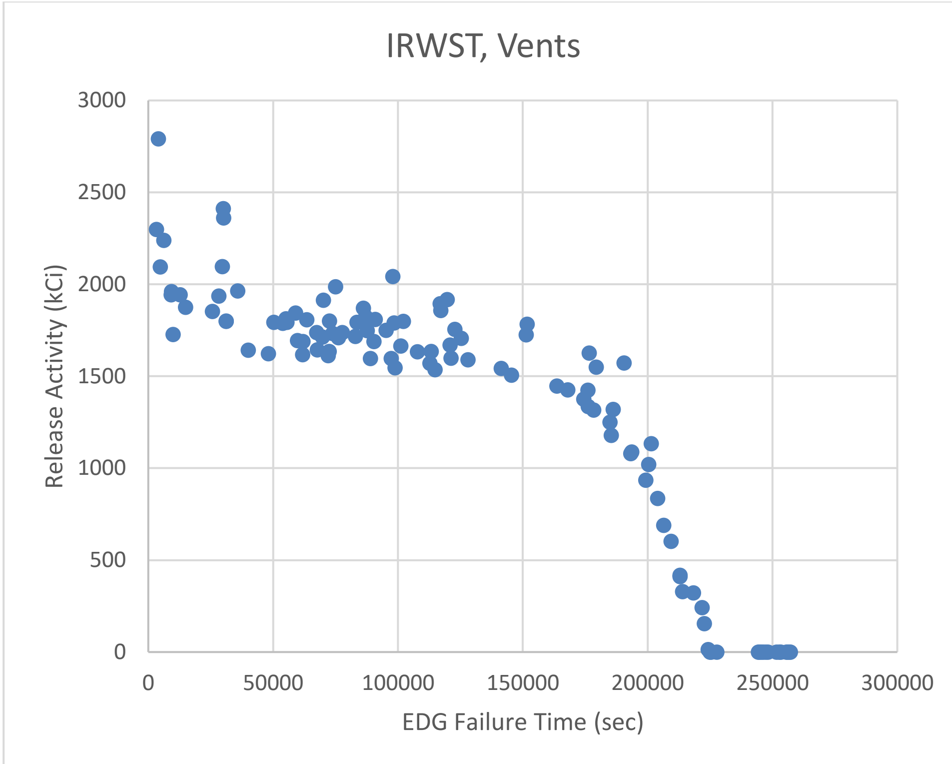


Figure 48. IRWST and Hardened Containment Ventilation Initial Upgrade Combination Impact Assessment

The results for IRWST and Hardened Containment Vents together are shown in Figure 48. The upgrades together were nearly as effective as the IRWST and Manual ADS, providing a similar reduction in Release Activity, though the reduction was similar to the Hardened Containment Vents alone. upon containment failure, further reducing the Release Activity.

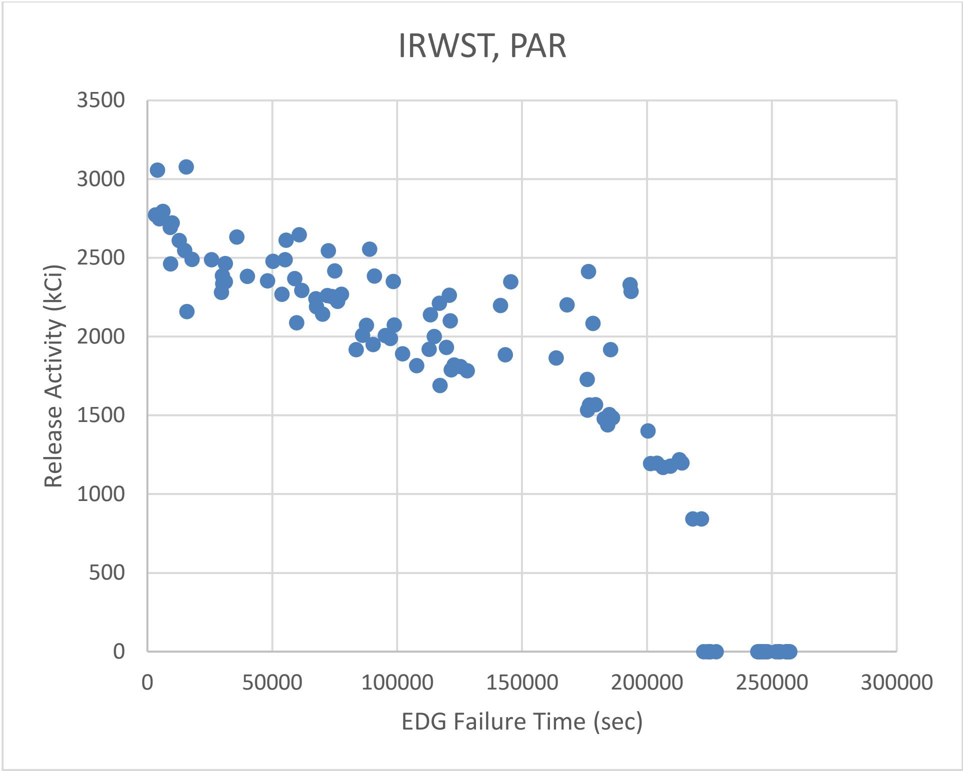


Figure 49. IRWST and Passive Autocatalytic Recombiner Initial Upgrade Combination Impact Assessment

As shown in Figure 49, adding the PAR on top of the IRWST did not show significant synergistic interaction between the two upgrades. The Release Activity expected value did not significantly decrease compared to the IRWST alone, and visual inspection of the data shows the two sets to be very similar. The mechanisms of mitigation are very similar, primarily through the IRWST quenching the hot core and relocated corium in cases where core damage and significant relocation has already occurred.

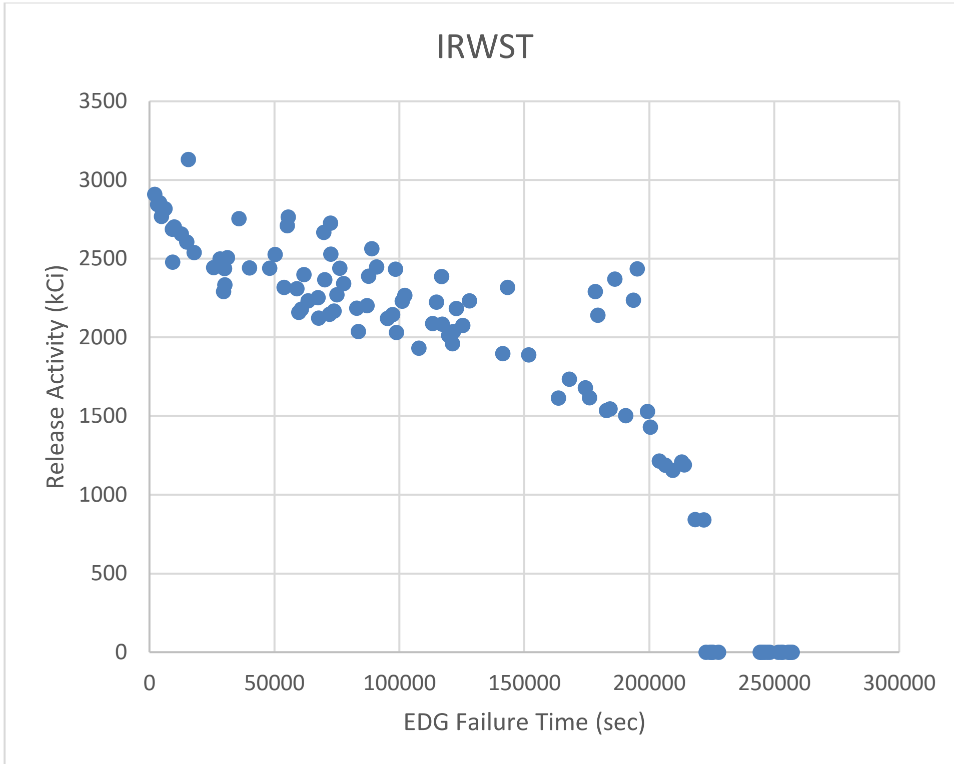


Figure 27. No Upgrades Analysis for comparison with Individual Upgrade Impact Analyses

Figure 27 is included here for the sake of comparison with the other configurations and is discussed in Section 5.2.7

### 5.3.5. Manual ADS Actuation

The Manual ADS upgrade was insignificant alone and did not show synergy with any of the upgrades it was tested alongside. This was anticipated, as the Manual ADS upgrade is intended to be paired with the IRWST upgrade and was confirmed by the results shown in Table 24.

Table 24. Manual ADS Upgrade Configuration EDG Sensitivity Study

| Configuration | Mean Release Activity | Median | Max | Sigma | 5th Percentile | 95th Percentile | Skewness | Kurtosis | Release Activity, EDG Failure Time Covariance | Pearson |
|---|---|---|---|---|---|---|---|---|---|---|
| ADS, PAR, Vent | 2435.9 | 1999.4 | 3000.6 | 632.80 | 1958.5 | 3000.6 | 0.5567 | -23.130 | -196957 | -0.00408 |
| ADS, PAR | 3000.7 | 2980.4 | 3197.0 | 49.188 | 2980.4 | 3021.5 | 10.912 | 302.94 | 36775 | 0.009286 |
| ADS | 3003.0 | 2978.6 | 3149.0 | 40.683 | 2978.6 | 3045.7 | 1.6165 | -3.0236 | 23470 | 0.007096 |
| No Upgrades | 3092.7 | 3131.6 | 3354.1 | 84.581 | 2997.4 | 3131.6 | -2.9114 | -3.6604 | -88966 | -0.01301 |

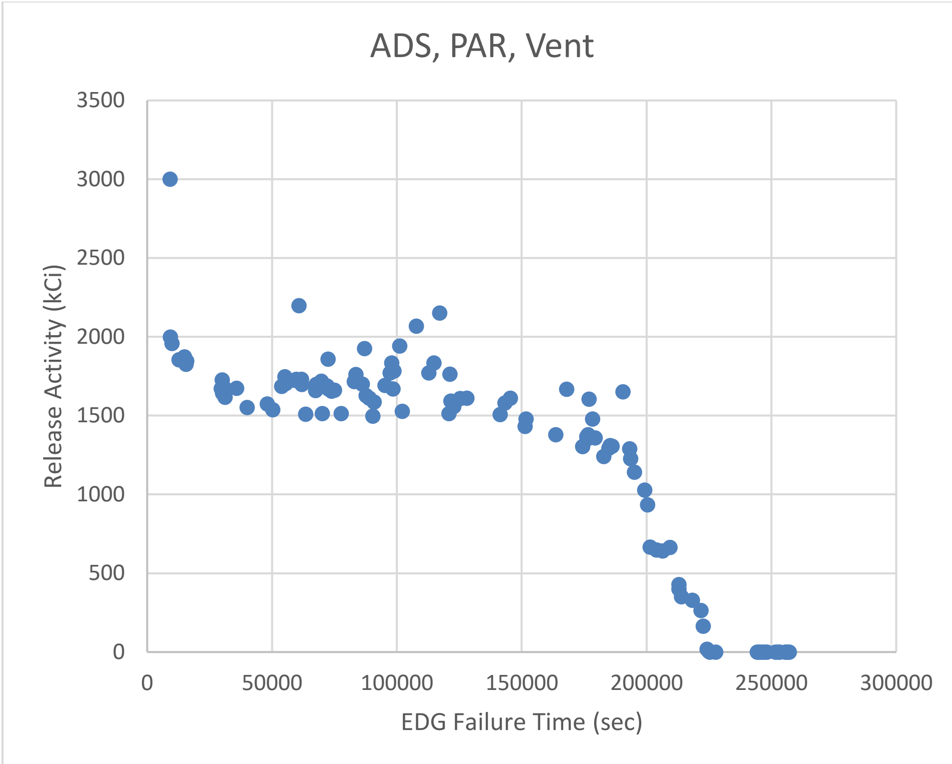


Figure 50. Manual ADS, PAR, and Hardened Ventilation Initial Upgrade Combination Impact Assessment

As seen in Figure 50, the Manual ADS, PAR, and unfiltered Ventilation combination was roughly as effective in the expected value for the Release Activity as the PAR and Vents were without the Manual ADS. The results were partially obscured by an aberrant, heavily weighted run with a much higher Release Activity than the rest of the runs, but visual inspection of the two data sets shows them to be very similar.

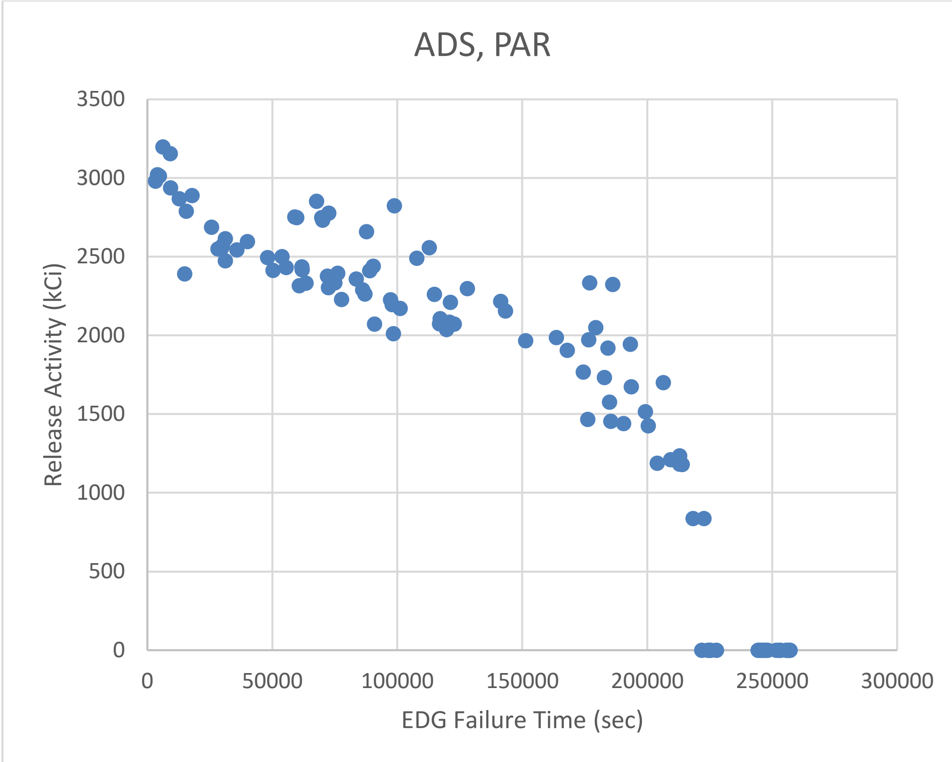


Figure 51. Manual ADS and PAR Initial Upgrade Combination Impact Assessment

As anticipated and shown in Figure 51, the Manual ADS upgrade had little to no discernible impact on the Release Activity without the upgrade it is intended to accompany, the IRWST upgrade. As with the Manual ADS, PAR, and Vents upgrade combination, the Manual ADS and PAR combination was very similar to the PAR upgrade alone. Without the IRWST upgrade it is intended to support, the Manual ADS upgrade has little impact. With this in mind, in the final analysis the IRWST and Manual ADS upgrades were packaged together, rather than examined individually, as the Manual ADS had essentially no impact on any case without the IRWST, and the cost of the IRWST utterly dwarfs the cost of the Manual ADS.

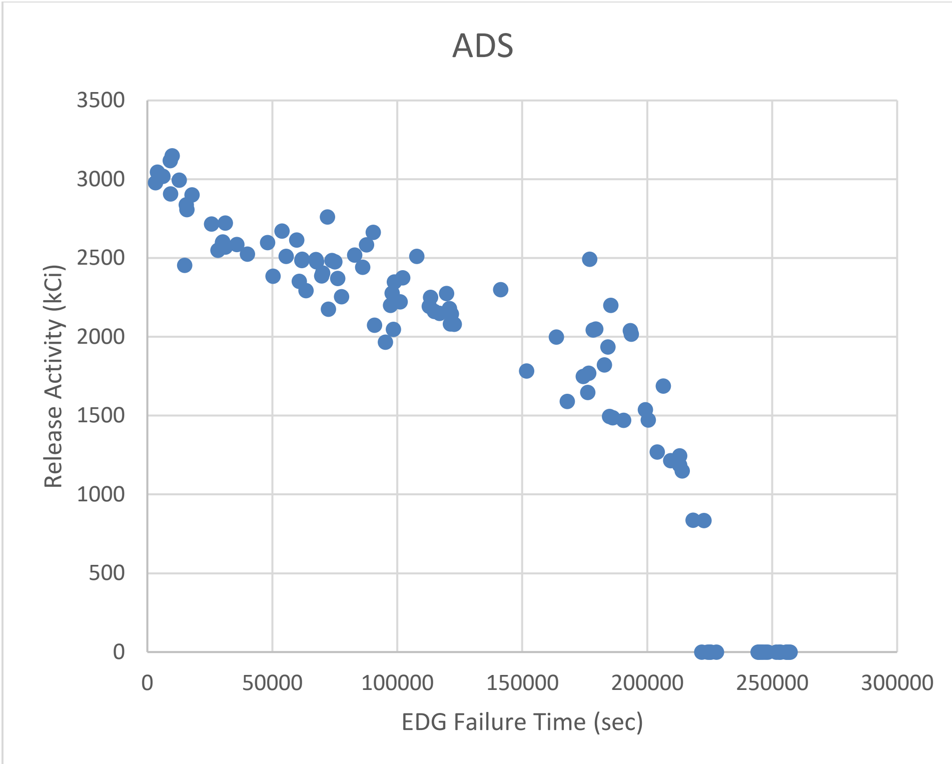


Figure 52. Manual Automatic Depressurization Actuation Initial Upgrade Impact Assessment

As shown in Figure 52, and comparing to Figure 27, the Manual ADS upgrade by itself had essentially no impact.

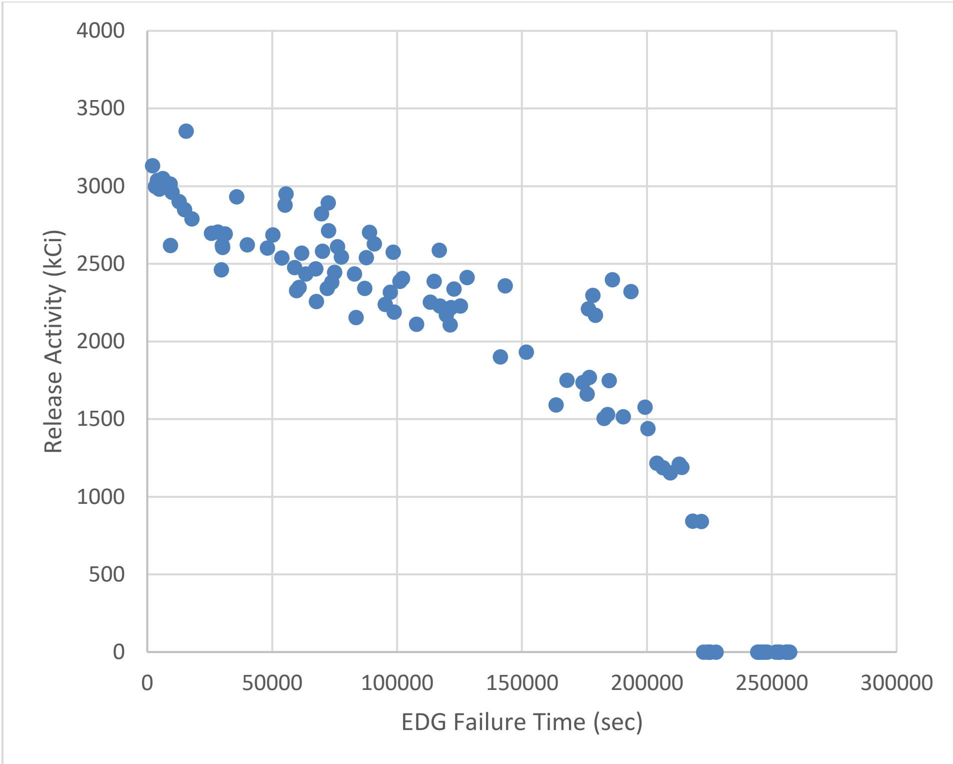


Figure 27. No Upgrades Analysis for comparison with Individual Upgrade Impact Analyses

Figure 27 is included here for the sake of comparison with the other configurations and is discussed in Section 5.2.1.

**5.3.6. Passive Autocatalytic Recombiners**

The results of the Passive Autocatalytic Recombiners examination are shown below in Table 25. The PARs were examined alongside various Hardened Containment Vents and various forms of Filtered Containment Vents to see how the upgrades interacted, as it was anticipated that containment ventilation upgrades would have the most interaction with PARs.

Table 25. Passive Autocatalytic Recombiner Upgrade Configuration EDG Sensitivity Study

| Configuration | Mean Release Activity | Median | Max | Sigma | 5th Percentile | 95th Percentile | Skewness | Kurtosis | Release Activity, EDG Failure Time Covariance | Pearson |
|---|---|---|---|---|---|---|---|---|---|---|
| PAR, Both Filters | 1524.7 | 1527.2 | 2609.0 | 16.900 | 1513.1 | 1527.2 | -15.543 | 1777.7 | -6917.2 | -0.0052 |
| PAR, Aerosol | 1569.6 | 1566.2 | 2610.3 | 99.388 | 1529.2 | 1566.2 | 24.854 | 1116.4 | 48103.4 | 0.00615 |
| PAR, Vents | 2218.6 | 2231.2 | 2626.6 | 124.11 | 2118.3 | 2231.2 | 11.998 | 474.16 | 10076.7 | 0.00103 |
| PAR | 3052.4 | 2975.1 | 3346.3 | 167.45 | 2966.2 | 3254.5 | 2.7545 | -2.5153 | 34997.9 | 0.00258 |
| No Upgrades | 3092.7 | 3131.6 | 3354.1 | 84.581 | 2997.4 | 3131.6 | -2.9114 | -3.6604 | -88965.7 | -0.01301 |

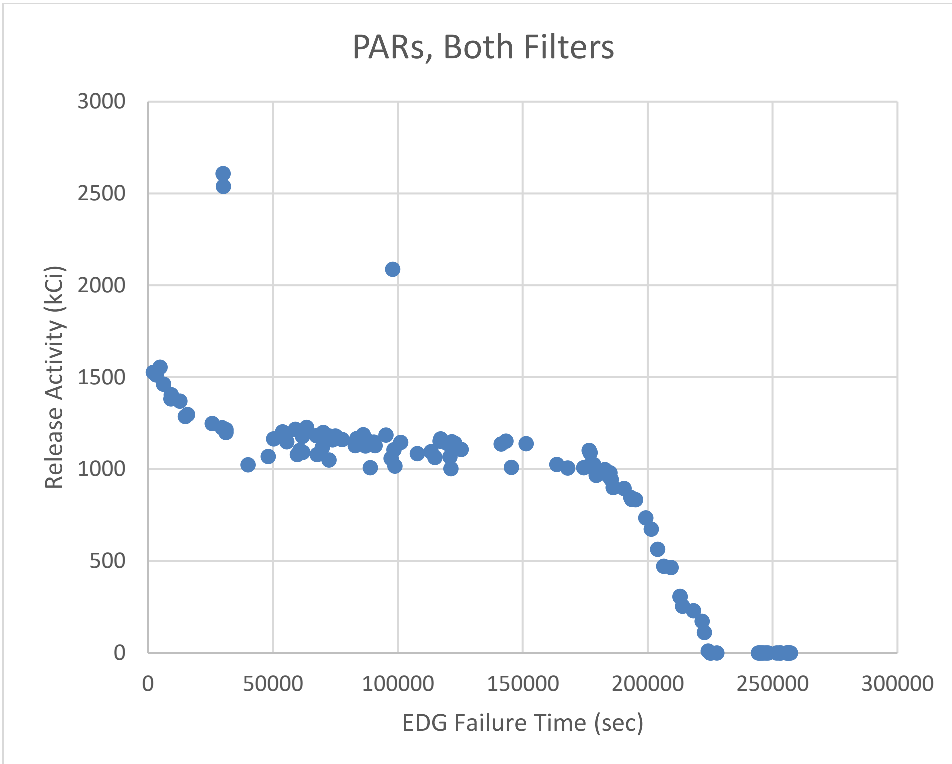


Figure 53. Passive Autocatalytic Recombiner and Vapor and Aerosol Filtered Ventilation Initial Upgrade Combination Impact Assessment

Shown in Figure 53, combining PARs with Aerosol and Vapor Filtered Vents minorly decreased the Release Activity compared to only having the doubly Filtered Vents. While the vents were able to largely prevent hydrogen deflagration from failing the containment, adding the PARs to prevent deflagration entirely meant that the pressure wave of a deflagration could not eject radioactive material from the containment. The anomalous points of high Release Activity were scenarios where circumstances came together to create a combustible mixture in the Wetwell, before the gas could pass to the Drywell and be oxidized, in such a configuration that the ensuing spike in pressure from the deflagration was large enough to rupture the containment.

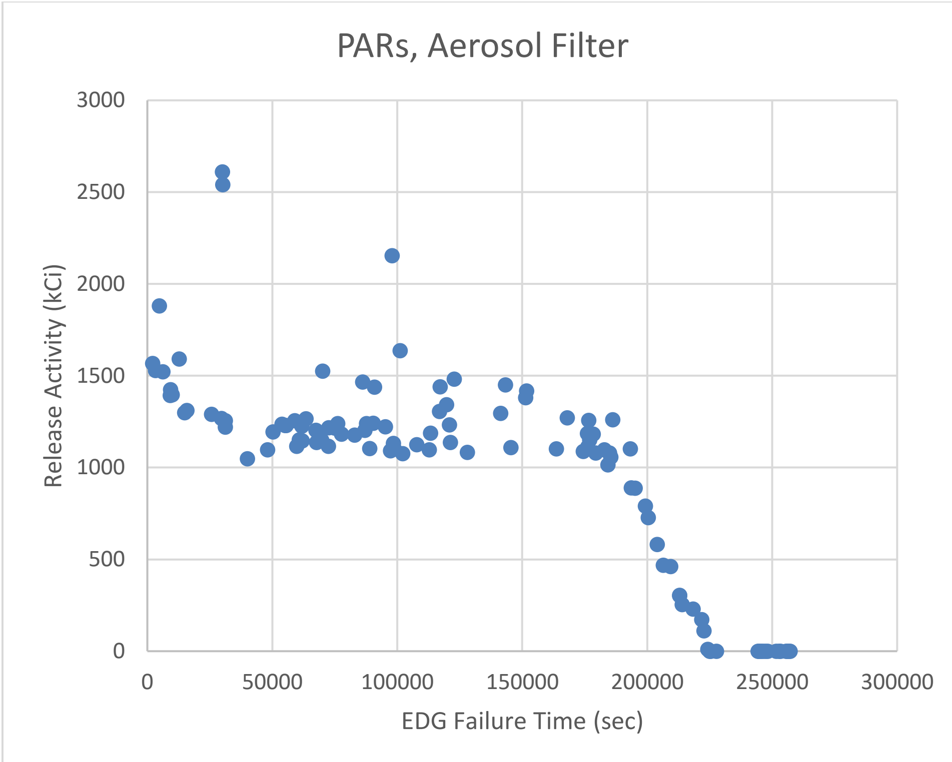


Figure 54. Passive Autocatalytic Recombiner and Aerosol Filtered Ventilation Initial Upgrade Combination Impact Assessment

Shown in Figure 54, the results from combining PARs with an Aerosol Filtered Vent was very similar to PARs and an Aerosol and Vapor Filtered Vent, albeit with slightly higher Release Activity from the lack of vapor filtration.

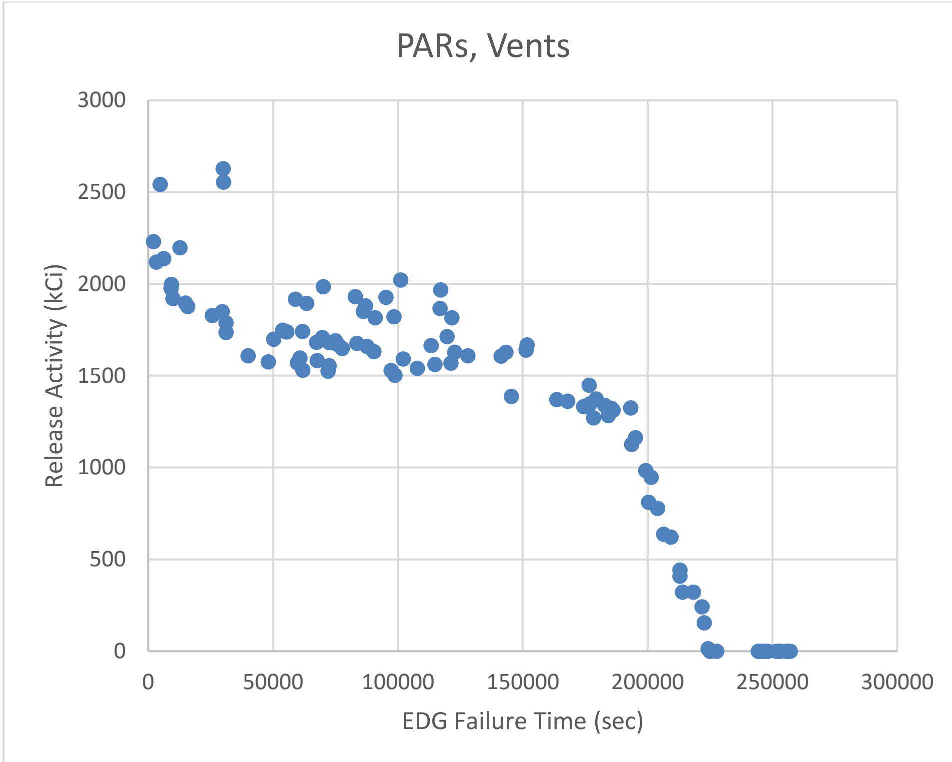


Figure 55. Passive Autocatalytic Recombiner and Hardened Containment Ventilation Upgrade Combination Impact Assessment

The results of examining PARs with Hardened Containment Vents are shown in Figure 55. As with the other combinations of Vents, PARs, and filtering systems, combining PARs and the Hardened Vents provided a modest improvement over the Hardened Vents alone, and via similar mechanism – the reduction in the potential for deflagration, in many cases, prevented the pressure spike from a deflagration from ejecting additional radioactive material into the environment.

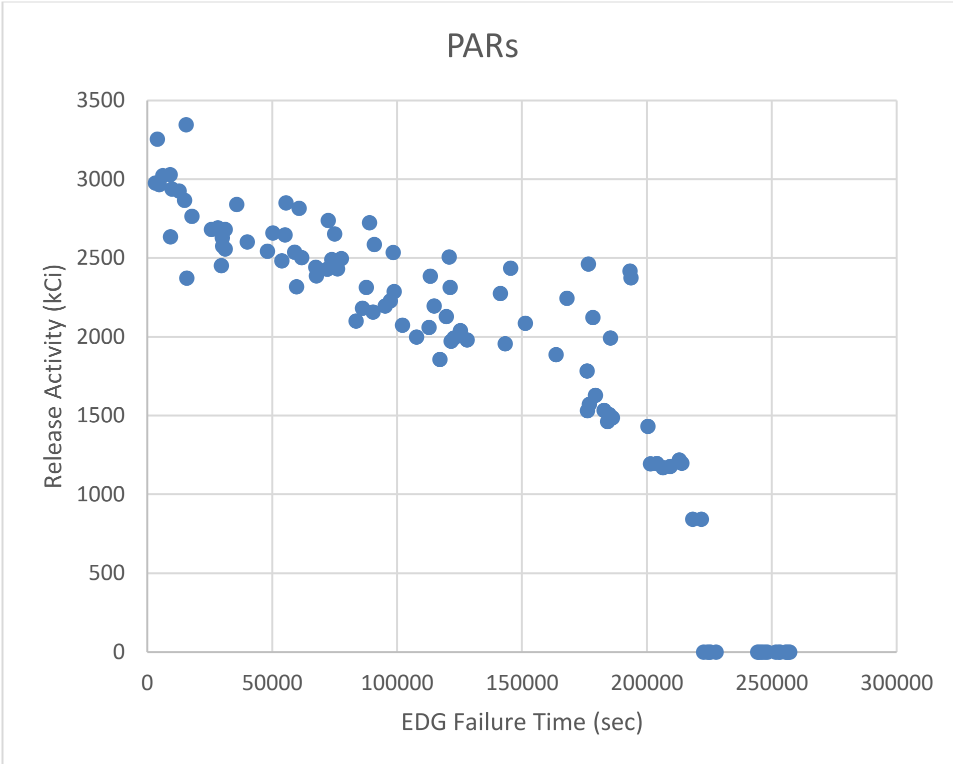


Figure 56. Passive Autocatalytic Recombiner Initial Upgrade Impact Assessment

The PAR upgrade was not found to have a major impact alone. The results in Figure 56 are shown here solely for comparison with the other cases in this section, and are discussed in more detail in Section 5.2.9.

## 5.4. *Initial Multi-Parameter Accident Scenario Upgrade Configuration Analyses*

### 5.4.1. Upgrade Configuration Release Activity CDF Comparison

Table 26 shows all upgrade configurations that were successfully analyzed in the initial Large-Scale Multi-Parameter analysis. Other tests were included, but unforeseen problems with the maximum job time allowed per test resulted in some tests failing. The source of this is unknown, as the maximum time allotted per MELCOR run, multiplied by the total number of runs, was well under the time allotted for the overall job.

Table 26 - Initial Multi-Parameter Accident Scenario Analysis Upgrade Configurations Tested

| Test Case | PARs | Manual ADS | IRWST | Vents | Filters | ATF |
|---|---|---|---|---|---|---|
| 1 | No | No | No | No | No | No |
| 2 | No | Yes | No | Yes | Yes | No |
| 3 | No | No | Yes | Yes | No | No |
| 4 | No | No | Yes | Yes | Yes | No |
| 5 | No | Yes | Yes | No | No | Yes |
| 6 | No | Yes | Yes | Yes | No | No |
| 7 | No | Yes | Yes | Yes | Yes | No |
| 8 | Yes | No | No | No | No | No |
| 9 | Yes | No | No | Yes | No | No |
| 10 | Yes | No | No | Yes | Yes | No |
| 11 | Yes | No | Yes | No | No | No |
| 12 | Yes | Yes | No | Yes | No | No |
| 13 | Yes | Yes | No | Yes | Yes | No |
| 14 | Yes | Yes | Yes | Yes | Yes | No |
| 15 | Yes | No | Yes | Yes | Yes | No |
| 16 | Yes | No | Yes | Yes | No | No |
| 17 | Yes | Yes | No | No | No | No |

After running these jobs, a serious error in the model arose that had not been encountered during earlier testing – vestigial functions to model early recovery of AC Power, believed to have been disabled entirely, were erroneously enabled during this testing. The primary fault with this model was in the LPCI, as it was incorrectly modeled and, once switched on, would generally flood the RPV with an unending torrent of water, eventually flooding both the RPV and the containment, displacing the gaseous contents of the containment and churning aerosol materials, greatly increasing the Release Activity. Attempts were made to salvage the data by removing all sample points for which the AC Power was recovered earlier than 68 hours into the Station Blackout, and the results are presented here for edification, though they are not used in the final cost-benefit analysis, for obvious reasons. Regardless of these problems, running hundreds of thousands of iterations of the various MELCOR models utilized provided enough data points that the majority of the data could be discarded while still leaving a statistically significant sample size.

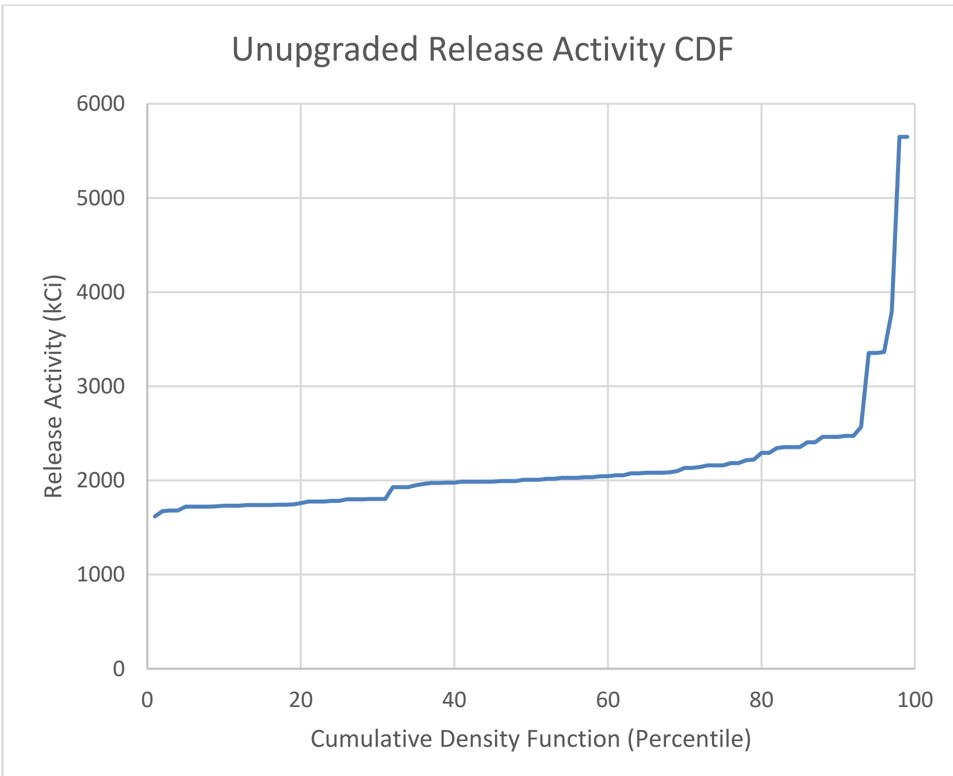


Figure 57 – Test-1, No Upgrades, Release Activity Cumulative Density Function

Figure 57 shows the CDF of the Release Activity for the non-upgraded plant response to a LT-SBO. The 5% CDF Xenon release activity was 1497 kCi, and the 95% CDF Xenon release activity was 2132 kCi – only a difference of 634 kCi, making the Xenon release relatively insensitive to effects that greatly altered the release activity from all other sources. The best-case scenarios were those in which the EDGs failed late and the Wetwell was very cold and neither overfilled nor underfilled. The worst-case scenarios were those in which circumstances aligned such that the containment was able to pressurize significantly before failing to a hydrogen deflagration, causing the most violent depressurization of containment possible and releasing the most radioactive material possible.

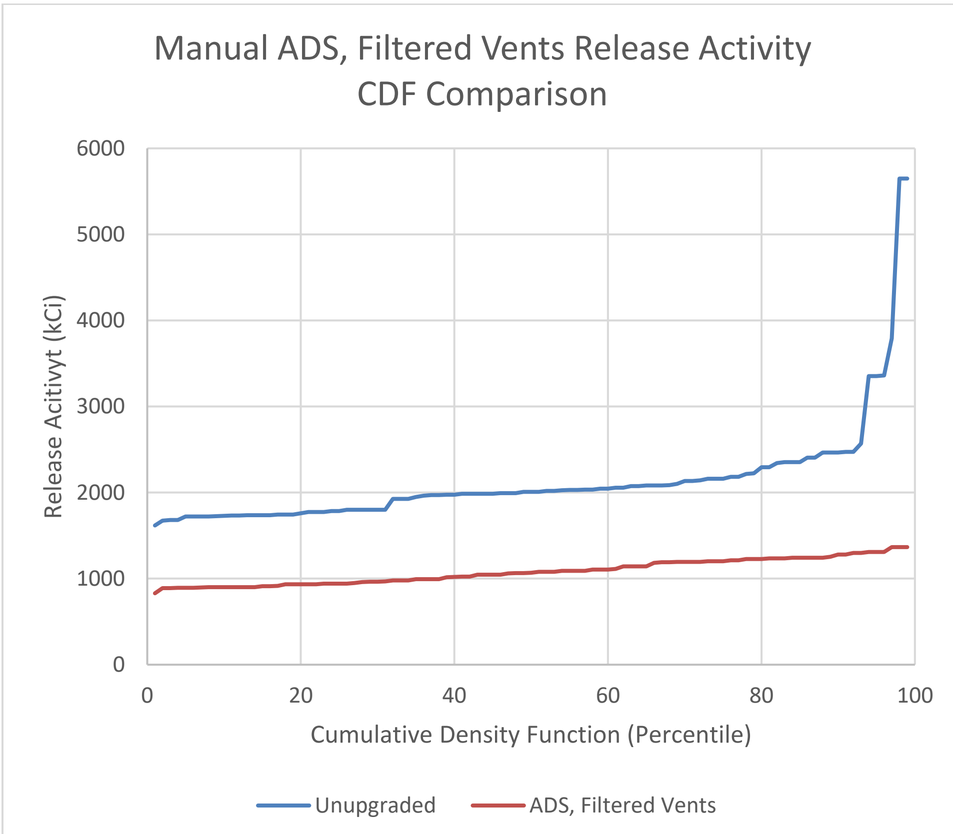


Figure 58 – Comparison between Test-2 and Unupgraded Test

Figure 58 shows a comparison of the Release Activity CDFs between this upgraded test and the unupgraded test. The Manual ADS had minimal impact or cost compared to the Filtered Vents, so this test was, in many ways, equivalent to testing the Filtered Vents alone. The filtration of the vents significantly reduced the activity released during favorable scenarios simply by removing much of the aerosols as they were vented, and the pressure relieving capabilities of the ventilation, combined with the filtration, drastically reduced the release activity in cases that would have otherwise led to major release via containment pressurization, failure, and violent depressurization.

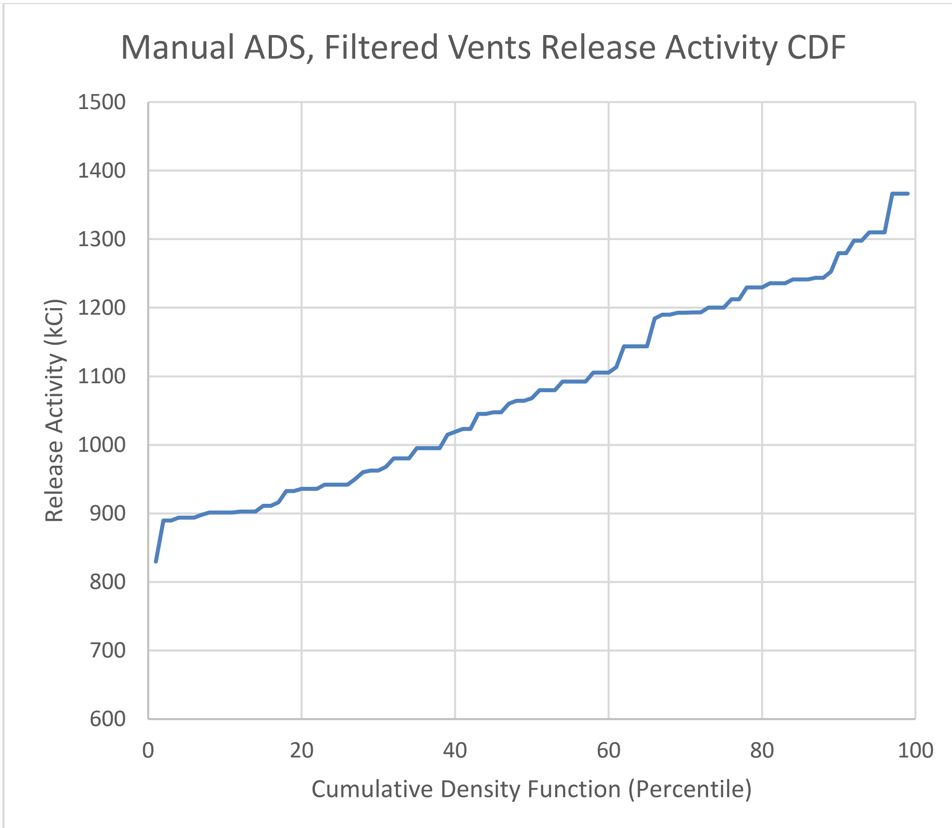


Figure 59 – Test-2, Manual ADS and Filtered Vents Release Activity Cumulative Density Function

Figure 59 shows the Release Activity CDF for the Manual ADS and Filtered Vents Release Activity with no comparison, to more effectively show details that were otherwise lost in Figure 58 due to the greater scale of the graph. The Release Activity in this test was found to be significantly negatively dependent on the EDG Failure Time – in other words, a longer EDG Failure Time lead to a significant reduction in Release Activity. The Release Activity was found to be positively dependent on Wetwell Initial Level and Wetwell Initial Temperature – a hot, overly full Wetwell corresponded generally with an increase in Release Activity. The Release Activity was very negatively dependent on the Containment Failure Pressure, as exceptionally weak containments were vulnerable to rupturing despite the inclusion of the vents, which significantly increased the Release Activity in these cases.

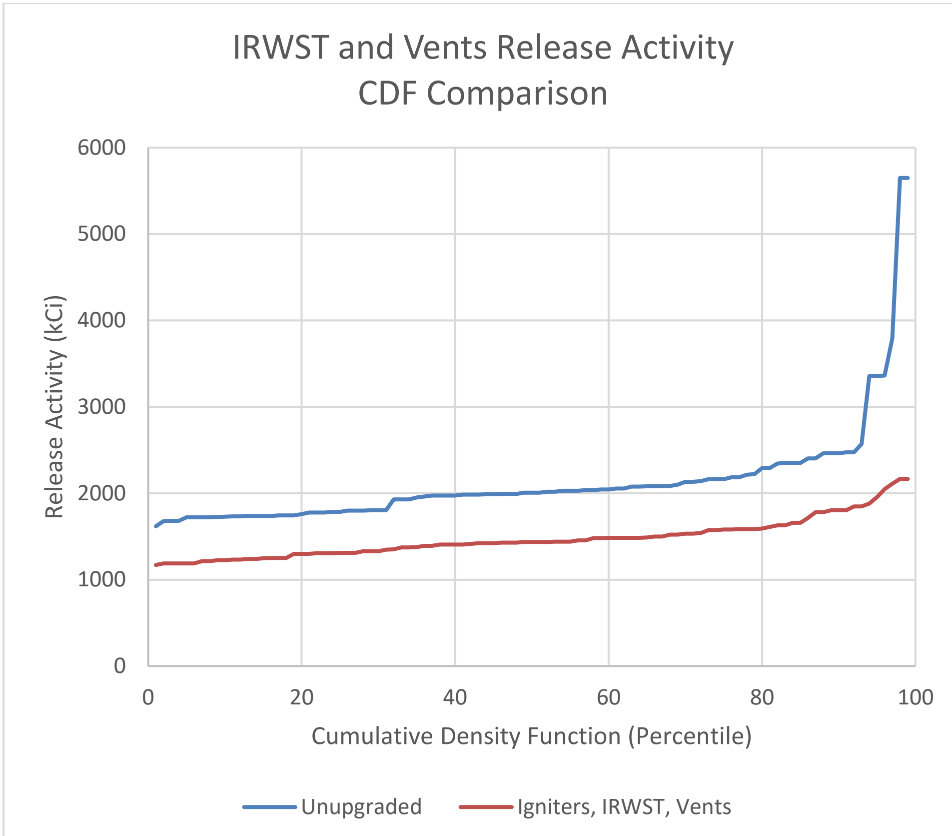


Figure 60 – Comparison between Test-3 and Unupgraded Test

Figure 60 shows a comparison of the Release Activity CDFs between this upgraded test and the unupgraded test. The combination of the IRWST and Hardened Vents together greatly reduced the release activity compared to the unupgraded case. A combination of pressure relief and quenching the core after core damage combined to greatly reduce the release activity, particularly in more worst-case scenario samplings.

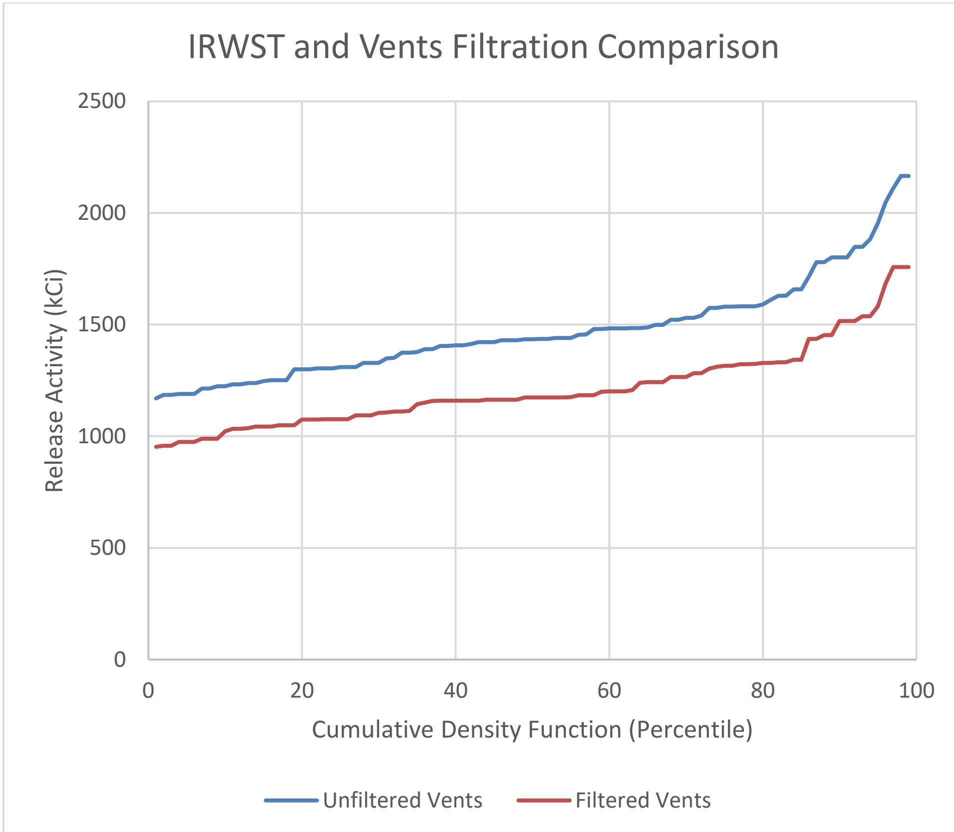


Figure 61 – IRWST, and Vents Filtration Release Activity CDF Comparison

Figure 61 shows a comparison of the Release Activity CDF for the IRWST and Hardened Vents upgrade configuration and the IRWST and Filtered Vents upgrade configuration. Worst-case scenarios in this test included a short EDG Failure Time and a hot, underfilled PSP. In general, a neither over nor under filled PSP led to lower Release Activity, DC Lifetime did not have a major impact on Release Activity, and cold PSPs lead to lower Release Activities than hotter PSPs. The ventilation filtration unit did not have an impact on the physics of the processes occurring inside the containment, and as such simply provided a reduction in the Release Activity across all cases. The exact fractional Release Activity reduction varied but was between 14% and 20% across the entire CDF – the filtration was more effective in cases where aerosols made up a greater portion of the Release Activity, and less effective in cases where the Release Activity of Xenon was proportionately greater. Across all cases, however, Xenon and other vapor Radionuclides release posed a minimum of 75% of the Release Activity, or the fractional Release Activity reduction from the aerosol filtration would have been significantly greater.

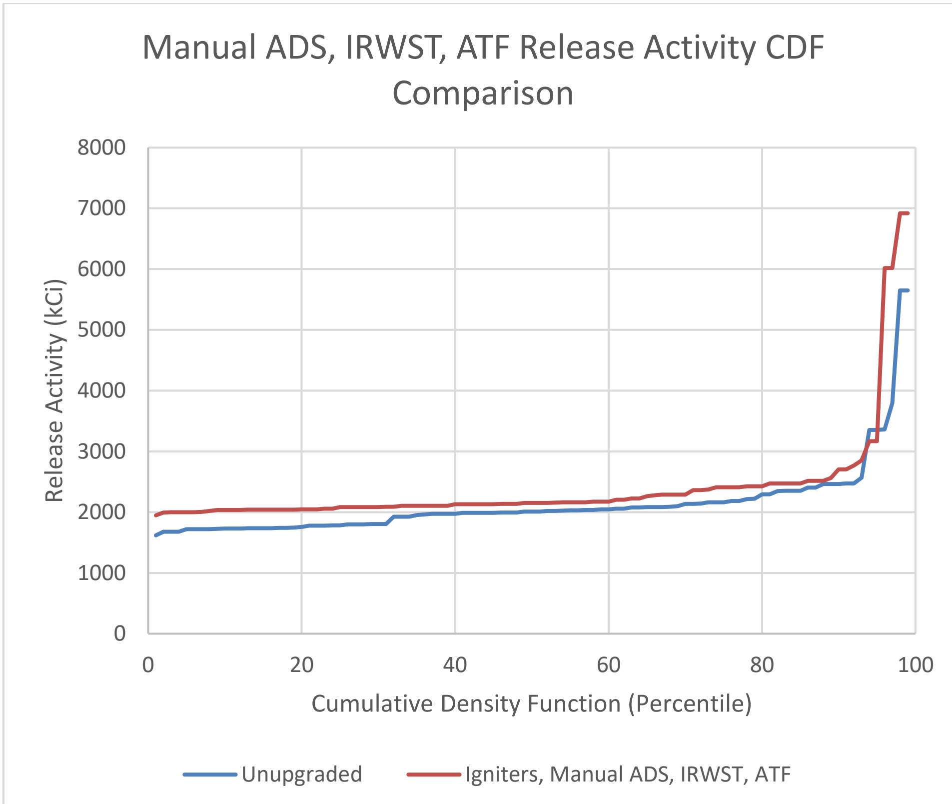


Figure 62 – Comparison between Test-5 and Unupgraded Test

Figure 62 shows a comparison in the Release Activity CDF for the Manual ADS, IRWST, and ATF configuration and the unupgraded configuration. In this test, it was discovered that the simplistic model for the effects of an ATF upgrade was fatally deficient. The flaw in it was that by increasing the minimum oxidation temperature in the Zircaloy oxidation model, the model essentially “saved” all of the zircaloy oxidation for the point at which the peak clad temperature climbed high enough for the zircaloy oxidation model to switch back on and immediately, at a very fast rate, begin oxidizing all of the hot zircaloy, rapidly generating large quantities of hydrogen and creating dangerously combustible and explosive gas mixtures. Because of this error, the ATF model is considered unfit for use and test results involving its use were discarded as unreliable. It was, however, considered a success that the methodology revealed the flaw in the upgrade model, as the ability to present surprising and/or concerning data when a serious flaw is present in the input to the methodology is as important as the ability to output sensible data when presented with sensible inputs – it was expected that the Manual ADS, IRWST, and ATF upgrades together would significantly reduce the Release Activity, not increase it, and the data shown in Figure 62 lead immediately to more investigation and the discovery of the flaw.

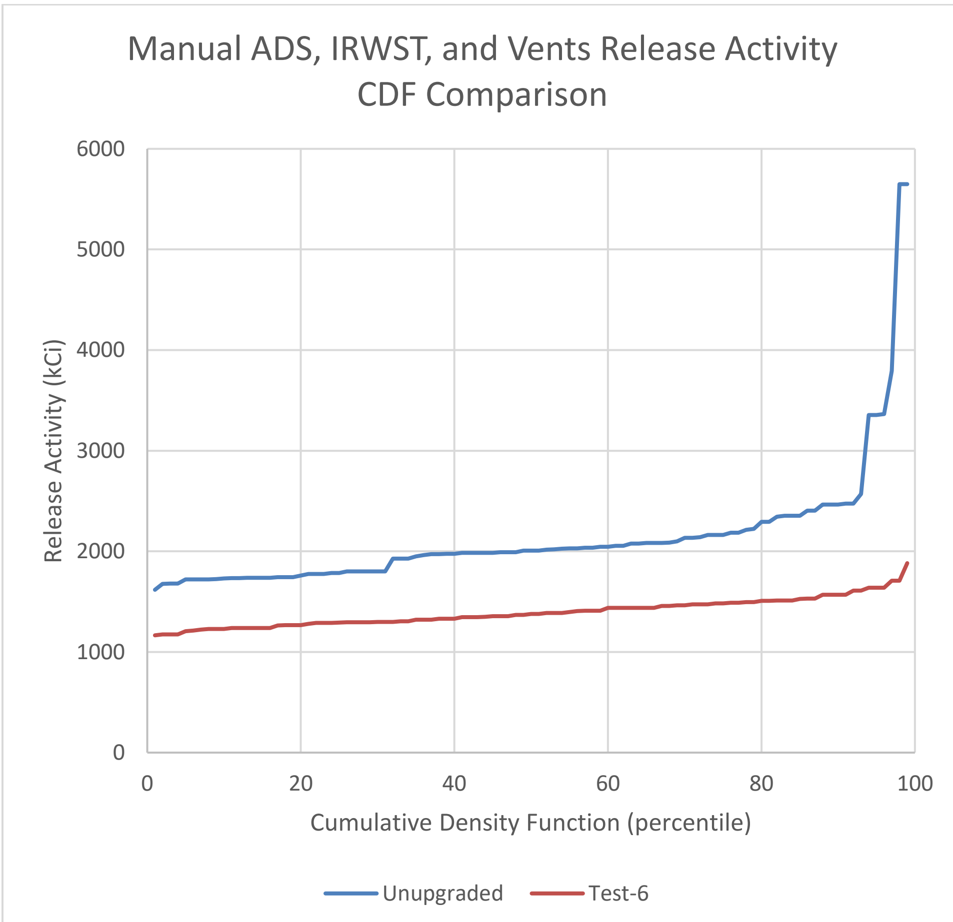


Figure 63 – Comparison between Test-6 and Unupgraded Test

Adding the Manual ADS, as expected, to Test-6 made it compare favorably to Test-3, which was an identical configuration, barring the exclusion of the Manual ADS. With a 99$^{th}$ percentile Release Activity of 1882 kCi, this upgrade configuration provided a significant reduction in release activity, particularly in worst case scenarios where the unupgraded configuration Release Activity climbed dramatically.

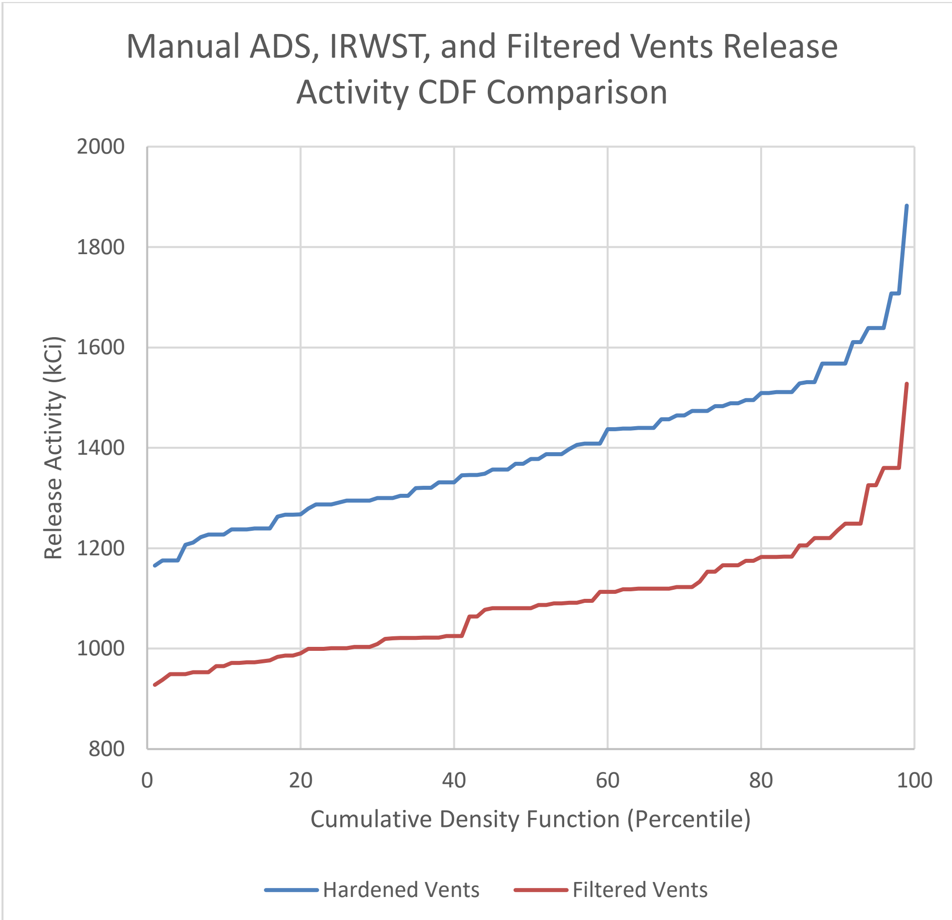


Figure 64 – Hardened Vents vs Filtered Vents, alongside Manual ADS and IRWST

Test-7 was identical to Test-6 other than adding filtration to the containment ventilation and provided a reduction in Release Activity of 15-20%, as shown in Figure 64.

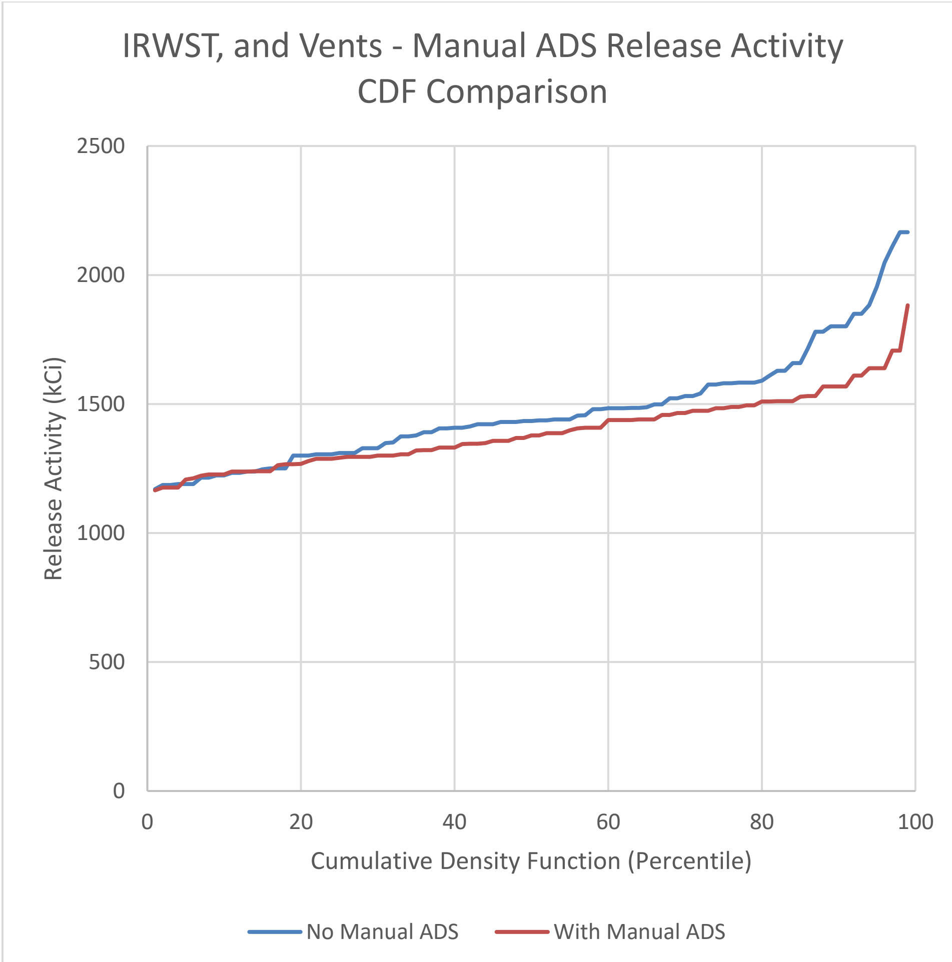


Figure 65 – Release Activity CDF Comparison for Test-3 and Test-6

As initial examination predicted and as shown in Figure 65, adding the Manual ADS Upgrade to an upgrade configuration that included the IRWST significantly improved the results in the new configuration. While the Manual ADS upgrade did not have a major impact in the smaller Release Activity cases examined, it significantly reduced the maximum Release Activity. Test-3, without the Manual ADS upgrade, had a 99$^{th}$ Percentile Release Activity of 2165 kCi, while Test-6, with the Manual ADS upgrade, had a 99$^{th}$ Percentile Release Activity of 1883 kCi, a reduction of 13%. The Manual ADS upgrade made the configuration Release Activity significantly less vulnerable to unfavorable Wetwell Initial Level and Wetwell Initial Temperature conditions, as well as reducing the Release Activity covariance with the EDG Failure Time, in that short EDG Failure Times were less harmful with the Manual ADS Upgrade than without.

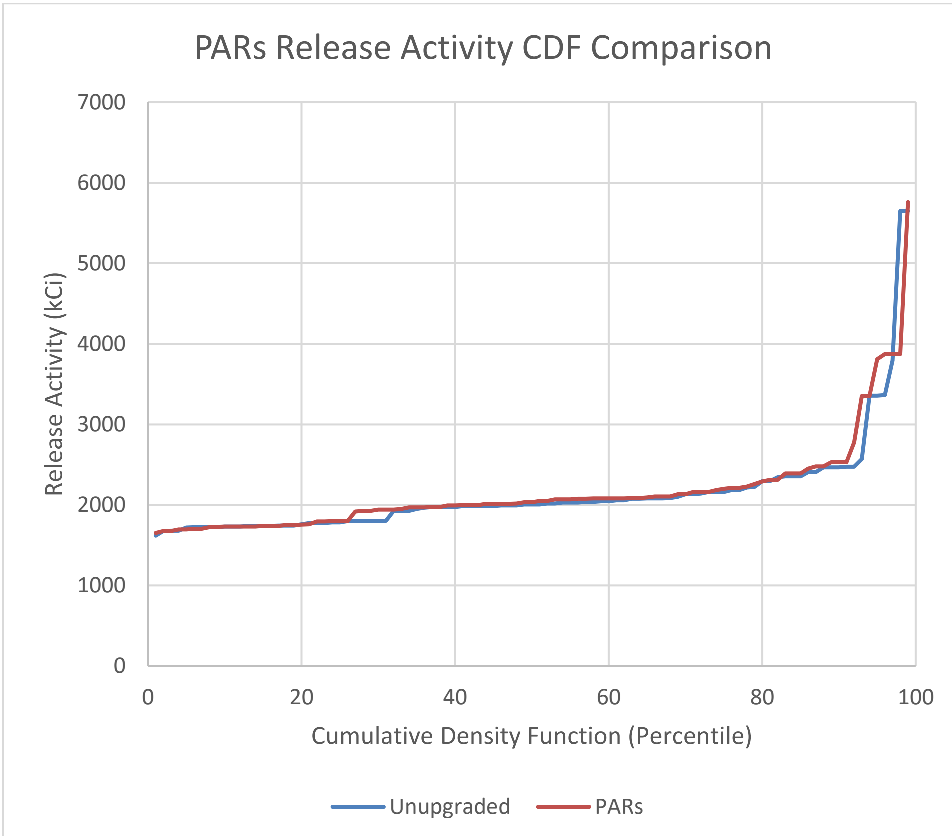


Figure 66 – Comparison between Test-8 and Unupgraded Test

Figure 66 shows the Release Activity CDF for the Unupgraded Test and for Test-8, which only implemented the Passive Autocatalytic Recombiner upgrade. While it was initially expected that the PARs would play a greater role in the accident scenario and provide a fair amount of reduction in Release Activity, this was not borne out in the data. The impact the PARs had on their own was largely negligible, and radionuclide release in these cases was driven by a mixture of deflagrations in the Wetwell, where hydrogen could potentially accumulate before transferring to the Drywell, where the PARs were implemented, to be mitigated, or by the slow accumulation of steam pressure to rupture the containment.

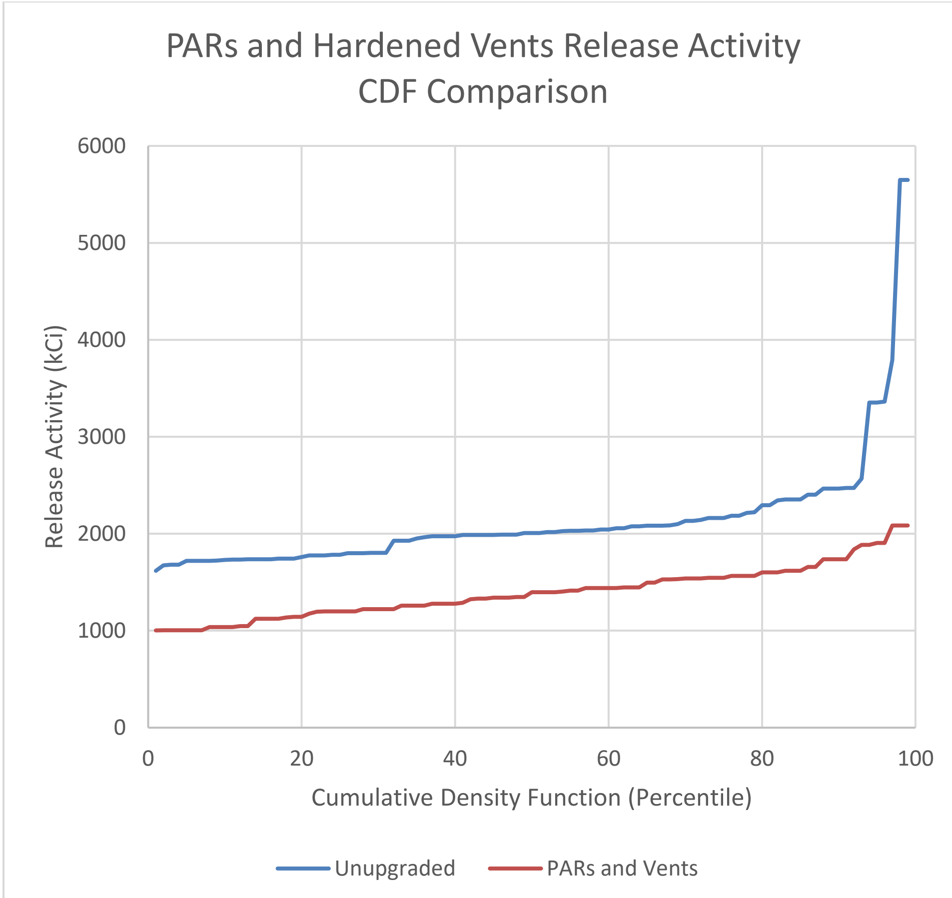


Figure 67 – Comparison between Test-9 and Unupgraded Test

Figure 67 shows a comparison of the Release Activity CDF between the PARs and Hardened Vents upgrade configuration and the unupgraded configuration. The upgrades had a significant impact on the low end of the Release Activity CDF, reducing the 1st Percentile Release Activity from 1617 kCi to 1003 kCi, and a major impact at the high end of the Release Activity CDF, reducing the 99th Percentile Release Activity from 5649 kCi to 2085 kCi.

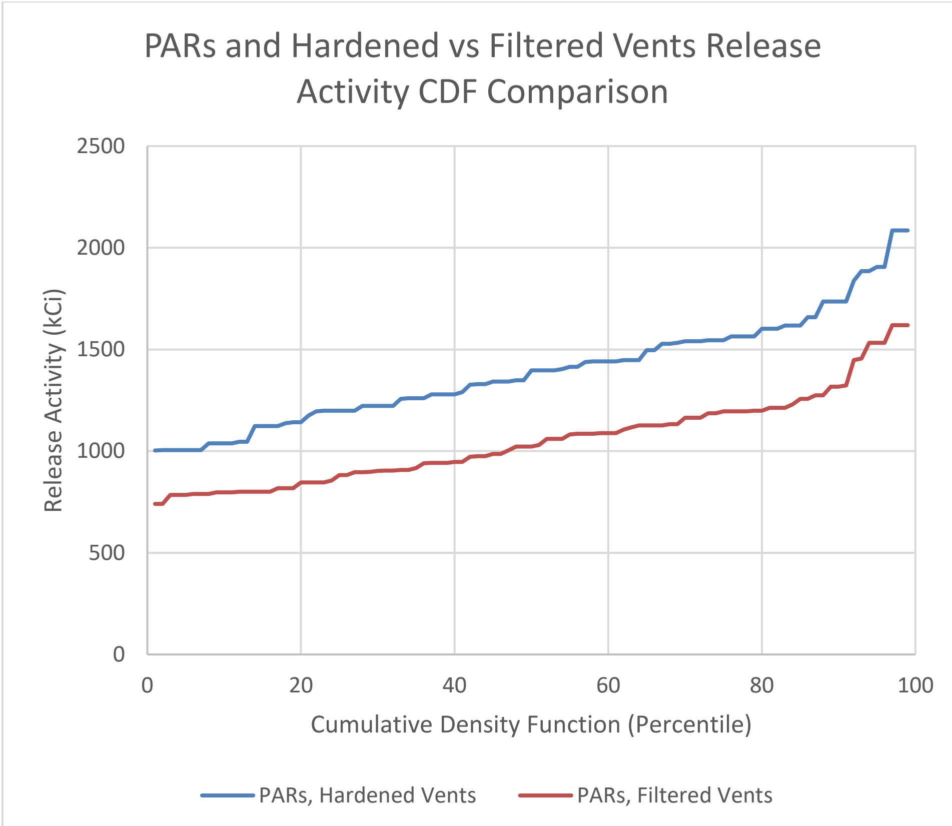


Figure 68 – PARs and Hardened vs Filtered Vent Comparison

Figure 68 shows a comparison between Test-9, which used an upgrade configuration of PARs and Hardened Vents, and Test-10, which used an upgrade configuration of PARs and Filtered Vents. The addition of filtration to the Hardened Vents reduced the Release Activity by roughly 25% across the entire Release Activity CDF, compared to the unfiltered vents.

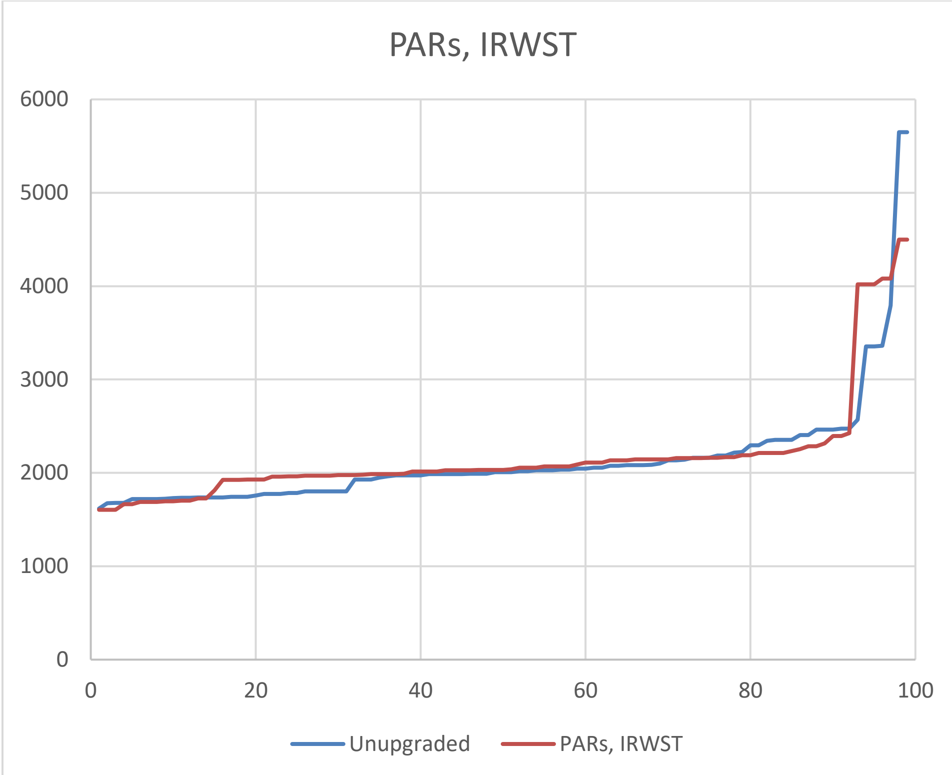


Figure 69 – Comparison between Test-11 and Unupgraded Test

Figure 69 shows a comparison between the results for the PAR and IRWST test and the unupgraded test. Interestingly, this upgrade configuration appears to have had anywhere between a minimal effect, where it was initially expected to be relatively impactful. Covariances between the Release Activity and stochastic parameters were generally found to be similar to the unupgraded test, with longer EDG Failure Times leading to lower Release Activities, DC Battery Life having little effect on the Release Activity, overfilled Wetwells leading to higher Release Activities, and hotter Wetwells leading to higher Release Activities. Lastly, the Release Activity was not found to be highly sensitive to the Containment Failure Pressure. Ultimately, this test was not re-examined because the IRWST without the Manual ADS upgrade was not expected to have a significant positive impact on worst-case scenarios, which include scenarios where DC Power fails before the ADS is actuated, preventing timely and effective use of the IRWST. Given that the Manual ADS is an extremely inexpensive upgrade compared to the IRWST and has had an overwhelming synergistic effect with the IRWST in all testing, there is little reason to believe that using the IRWST by itself will ever be more cost-effective than using the Manual ADS and IRWST upgrades together, and the matter was not pursued further.

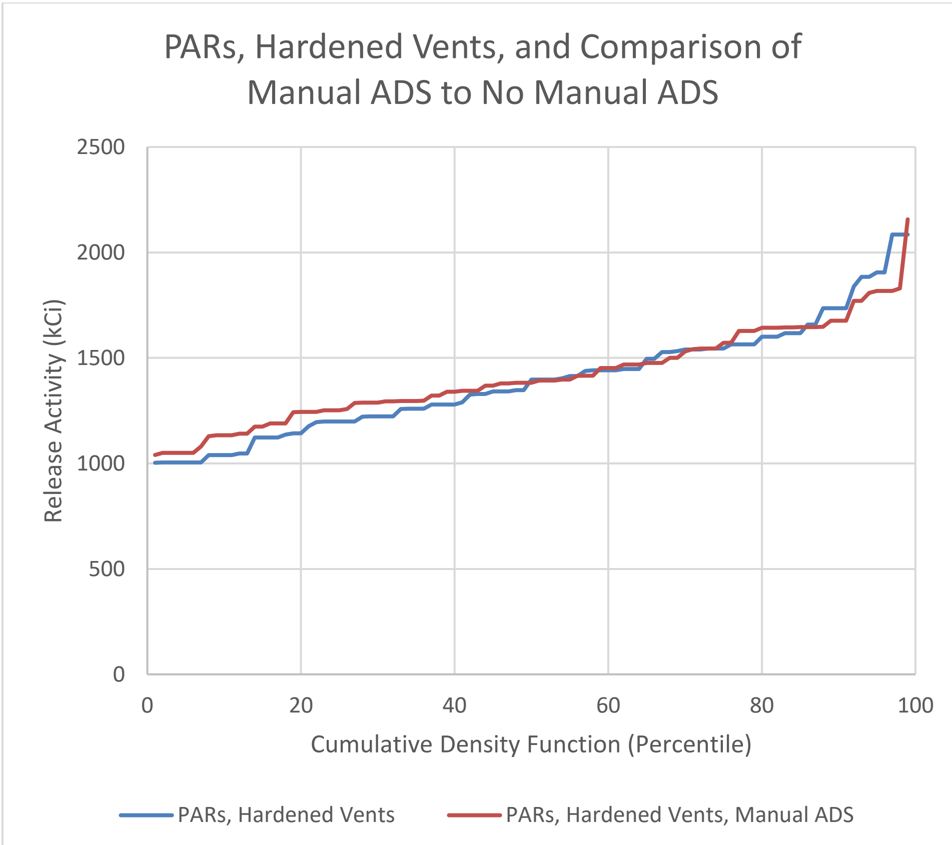


Figure 70 – Comparison of Manual ADS vs No Manual ADS, alongside PARs and Hardened Vents

Figure 70 shows an examination of the impact of the Manual ADS upgrade when it is implemented alongside PARs and Hardened Vents. Test-9, with only the PARs and Hardened Vents and no Manual ADS, and Test-12, which added the Manual ADS, were very similar, both visually and statistically. The Manual ADS upgrade had little impact, as to be expected without the upgrade it is intended to be implemented alongside, the IRWST upgrade.

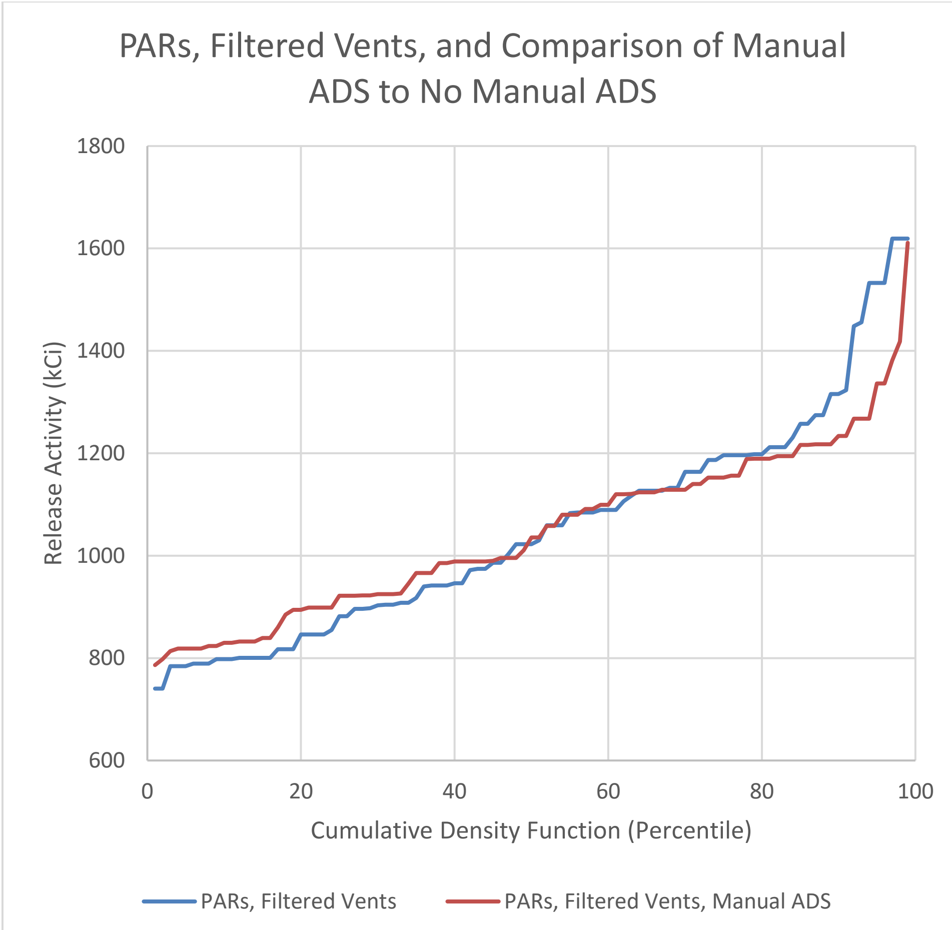


Figure 71 - Comparison of Manual ADS vs No Manual ADS, alongside PARs and Filtered Vents

As with Test-9 and Test-12 using PARs Hardened Vents, little impact was to be had from the Manual ADS when it was accompanied by PARs and Filtered Vents in comparison between Test-10 and Test-13. This is to be expected, as the implementation for Filtered Vents does not change the progression of the accident scenario, in comparison to the Hardened Vents, other than to remove outgoing Aerosols passing through the vent flowpath within MELCOR.

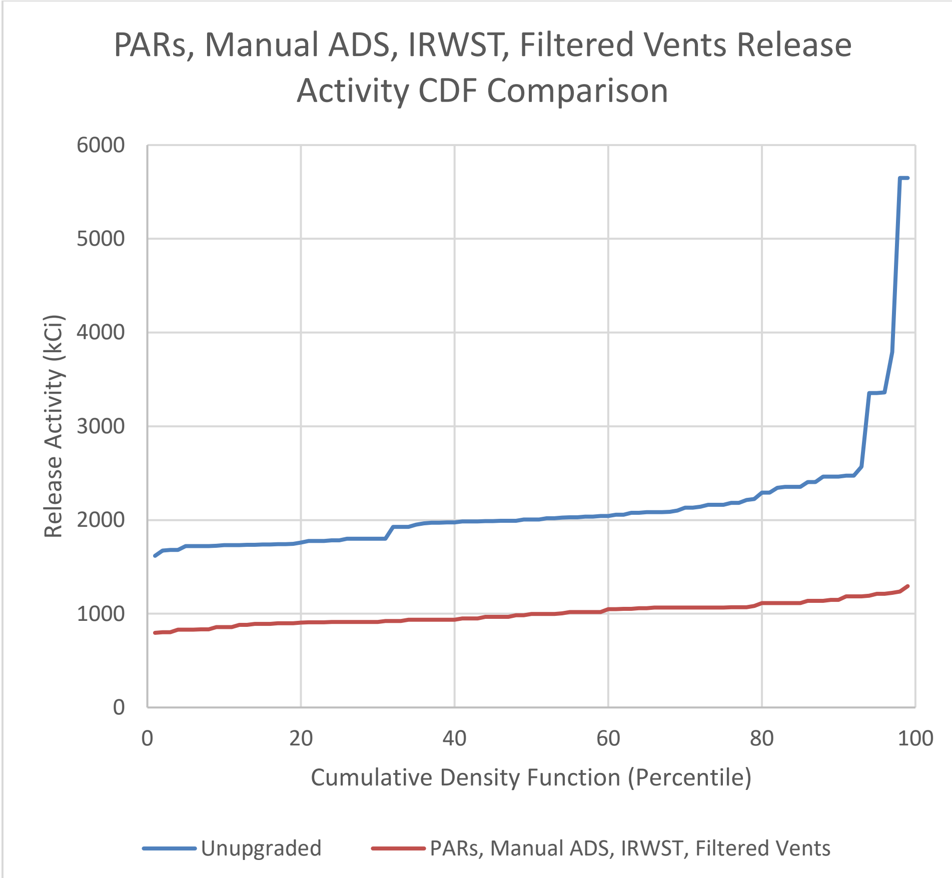


Figure 72 – Comparison between Test-14 and Unupgraded Test

Figure 72 shows the results of Test-14 compared to the unupgraded results. As was expected, combining many synergistic upgrades had an overwhelming reduction in Release Activity. At the low end of the spectrum, the Release Activity dropped from 1617 kCi to 796 kCi for the 1$^{st}$ Percentile Release Activity, a more than 50% reduction in Release Activity. At the high end of the spectrum, the 99th Percentile Release Activity dropped from 5649 kCi to 1294 kCi, nearly an 80% reduction. The 99$^{th}$ Percentile Release Activity for the heavily upgraded case was smaller than the 1$^{st}$ Percentile Release Activity for the unupgraded case. That said, these reductions in Release Activity come with a burdensome price tag, as discussed later.

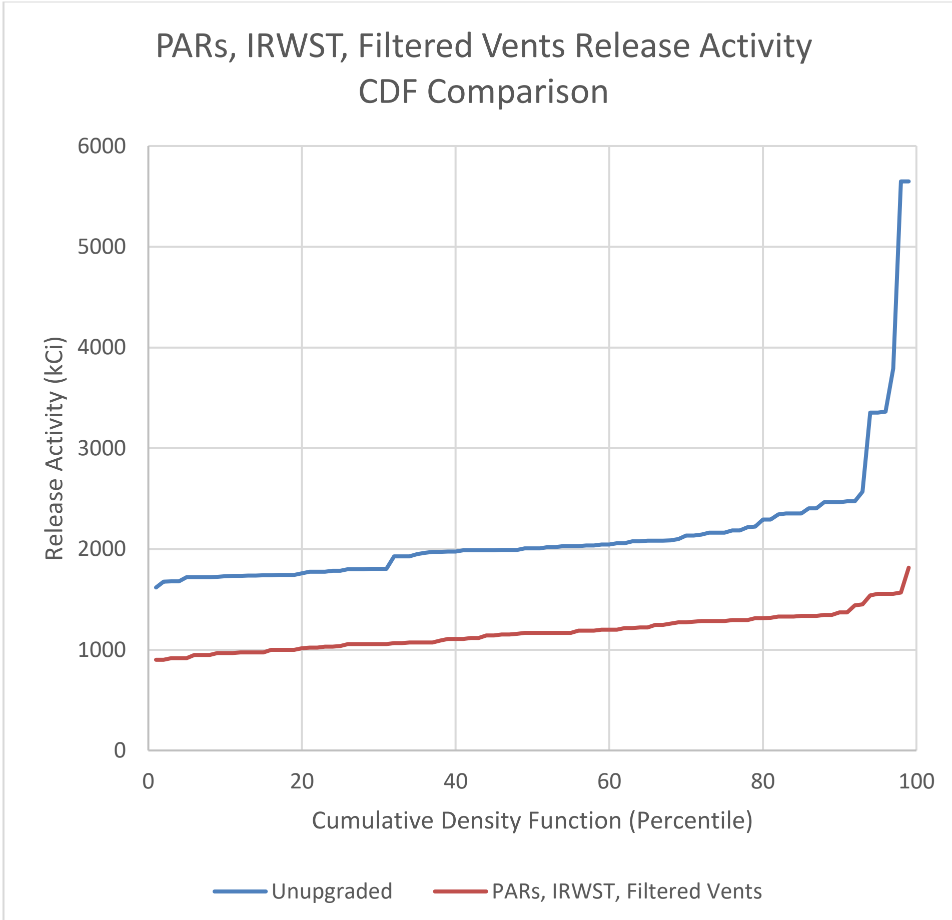


Figure 73 – Comparison between Test-15 and Unupgraded Test

Figure 73 shows a comparison between the results of Test-15 and the unupgraded test. Comparing Figure 72 and Figure 73 to examine the difference between Test-14 and Test-15, it can be seen that removing the Manual ADS upgrade significantly impacted the upgrade configuration's efficacy, though the upgrade configuration remains a very effective one. This is examined in greater detail below in Figure 74.

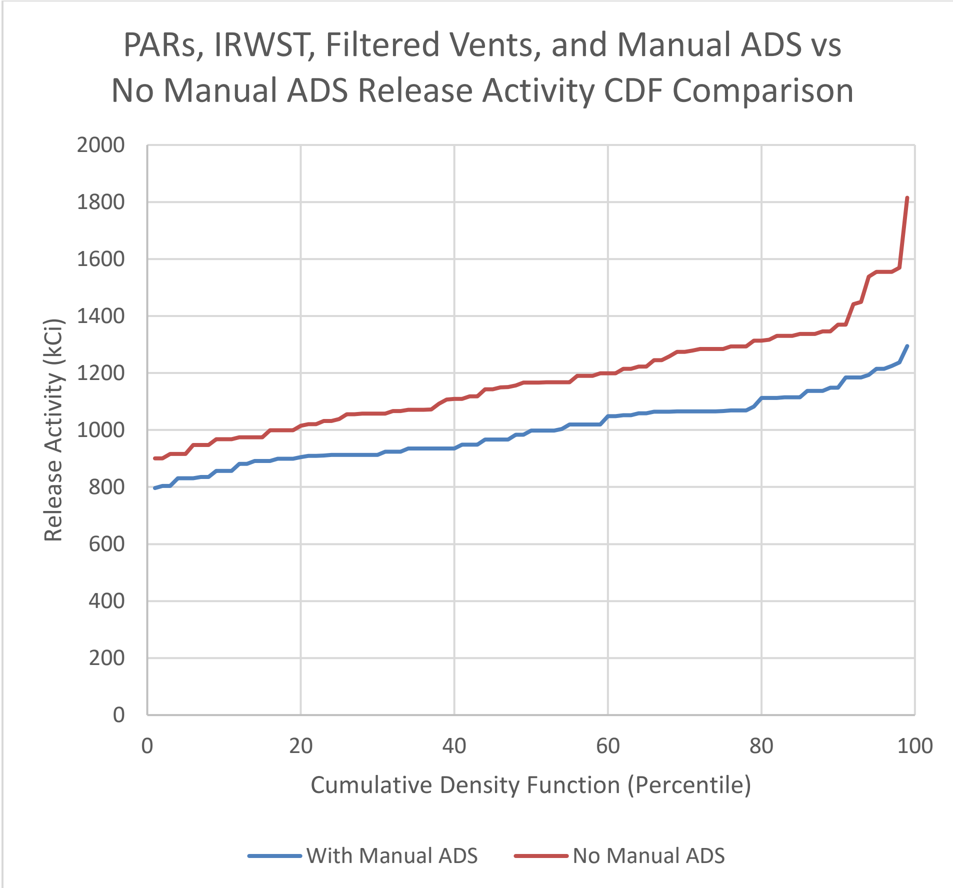


Figure 74 – Comparison of Manual ADS vs No Manual ADS, with PARs, IRWST, and Filtered Vents

Figure 74 shows a comparison of Test-14 and Test-15 in greater detail. Both tests' upgrade configurations included the PARs, IRWST, and Filtered Vents upgrades. Test-14 included the Manual ADS upgrade, while Test-15 did not. The inclusion of the Manual ADS made a major impact on the Release Activity CDF, greatly lowering it across the entire spectrum of possible Release Activities. The 1$^{st}$ Percentile Release Activity for Test-15 was 901 kCi and was reduced to 796 kCi in Test-14. The 99$^{th}$ Percentile Release Activity for Test-15 was 1814 kCi and was reduced to 1295 kCi in Test-14. With or without the Manual ADS, both upgrade configurations were marked improvements over the Unupgraded test configuration.

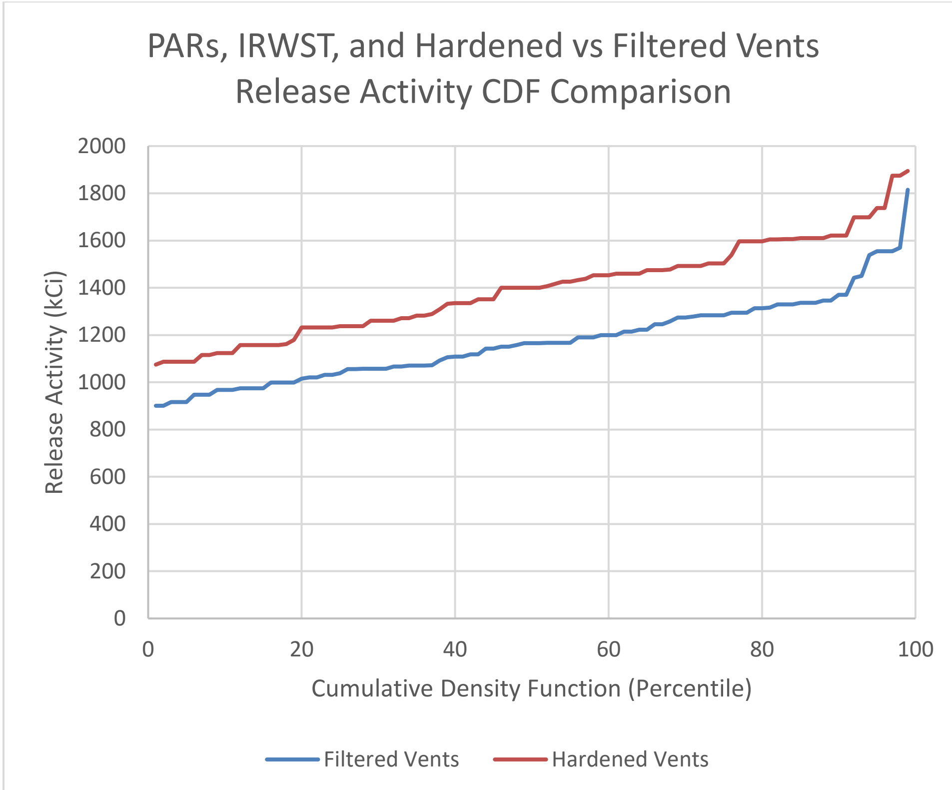


Figure 75 – Hardened vs Filtered Vents Comparison, alongside PARs and IRWST

Figure 75 shows a comparison between Test-15 and Test-16. Both tests' upgrade configurations include PARs, the IRWST upgrade, and Hardened Vents. Test-15 has an aerosol filtration system on the vents, where Test-16 does not. As with previous examinations of the aerosol filtration system, it provided between a 15% and 20% reduction in Release Activity, with the exception of a 4% reduction in Release Activity for the 99$^{th}$ Percentile Release Activity.

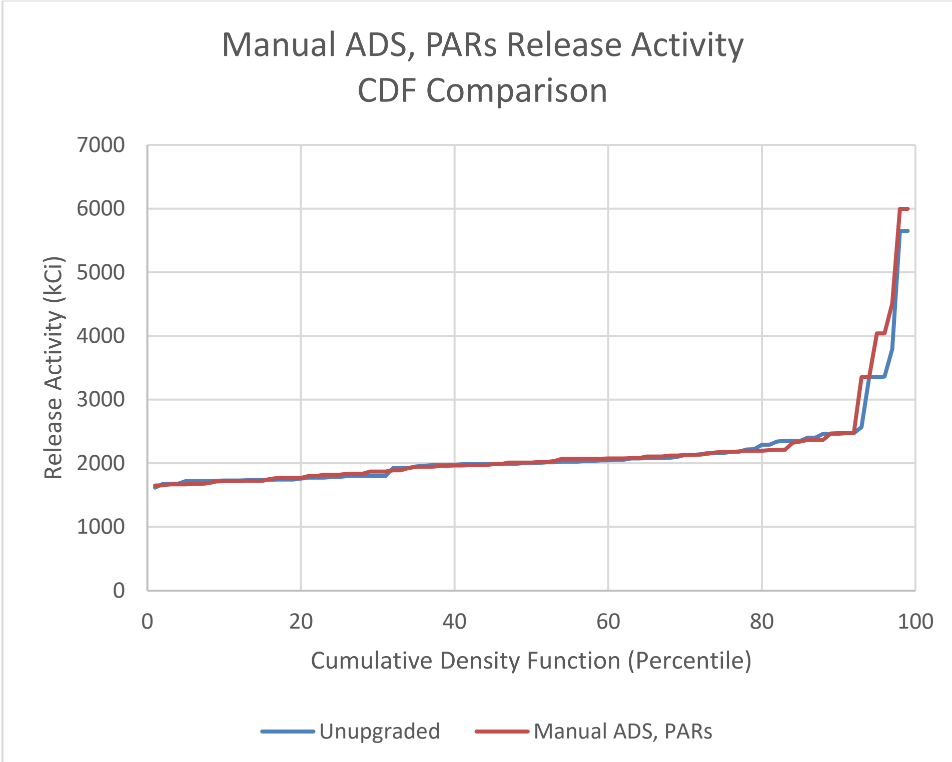


Figure 76 – Comparison between Test-17 and Unupgraded test

Figure 76 shows a comparison between Test-17 and the unupgraded test. The Manual ADS and PAR upgrades together had little to no impact on the Release Activity CDF when compared to the unupgraded Release Activity CDF. Based on prior modeling of the PAR upgrade and modeling of the Manual ADS upgrade without the IRWST, this was the expected result, but was examined anyway to confirm prior expectations.

Table 27. Initial Multi-Parameter Analysis Change in Release Activity Statistics

| Test Case | Number of Cases | Mean Release Activity | Release Activity Sigma | Mean Difference Standard Error | Degrees of Freedom | T-value | Mean Difference | 95% Confidence Interval Radius |
|---|---|---|---|---|---|---|---|---|
| 1 | 62 | 2138.87 | 640.06 | | | | | |
| 2 | 54 | 1081.34 | 148.34 | 83.76 | 68.46 | 1.97 | 1057.53 | 165.00 |
| 3 | 62 | 1476.11 | 230.37 | 86.39 | 76.54 | 1.97 | 662.77 | 170.19 |
| 4 | 60 | 1214.67 | 188.62 | 84.86 | 71.84 | 1.97 | 924.21 | 167.17 |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 5 | 63 | 2411.69 | 928.45 | 142.45 | 110.22 | 1.97 | -272.82 | 280.62 |
| 6 | 58 | 1393.56 | 140.86 | 83.37 | 67.29 | 1.97 | 745.32 | 164.23 |
| 7 | 56 | 1092.30 | 116.31 | 82.76 | 65.45 | 1.97 | 1046.57 | 163.04 |
| 8 | 65 | 2186.78 | 687.07 | 117.77 | 124.93 | 1.97 | -47.90 | 232.01 |
| 9 | 48 | 1395.16 | 271.87 | 90.26 | 86.64 | 1.97 | 743.71 | 177.82 |
| 10 | 55 | 1050.46 | 221.38 | 86.60 | 76.98 | 1.97 | 1088.42 | 170.59 |
| 11 | 47 | 2178.51 | 605.49 | 120.03 | 101.84 | 1.97 | -39.64 | 236.47 |
| 12 | 53 | 1413.01 | 225.40 | 86.98 | 78.06 | 1.97 | 725.86 | 171.36 |
| 13 | 53 | 1044.57 | 168.87 | 84.53 | 70.79 | 1.97 | 1094.30 | 166.53 |
| 14 | 52 | 1000.06 | 115.18 | 82.84 | 65.68 | 1.97 | 1138.81 | 163.20 |
| 15 | 47 | 1174.94 | 180.59 | 85.45 | 73.41 | 1.97 | 963.94 | 168.33 |
| 16 | 55 | 1395.52 | 204.23 | 85.83 | 74.69 | 1.97 | 743.35 | 169.08 |
| 17 | 61 | 2174.75 | 744.95 | 125.32 | 117.72 | 1.97 | -35.87 | 246.88 |

Table 27 presents an analysis of the difference in mean value between each test and Test-1, the unupgraded test. The results were calculated using a Welch T-test, as discussed in Section 4.10. The majority of the tests provided a major, statistically significant improvement over the base case. The T-value for each case was obtained with an online T-test table, and the mean difference for each case was calculated such that positive values for Mean Difference indicate a reduction in Release Activity.

Test-5, the failed ATF test, Test-8, the PARs test, Test-11, the PARs and IRWST, and Test-17, the PARs and Manual ADS test were the only tests that did not confer a statistically significant reduction in Release Activity when compared to the unupgraded base case. Visual examination of Figure 62, Figure 66, Figure 69, and Figure 76 confirms the conclusions garnered from statistical analysis, none of these upgrade configurations caused any significant reduction in Release Activity. Discarding these tests, heavily condensing the results, and sorting all tests but the base, unupgraded test by Mean Difference, and re-adding details of each upgrade configuration, we get

Table *28*.

Table 28. Initial Multi-Parameter Accident Scenario Upgrade Configuration Analysis Results, Ranked

| Test Case | Number of Cases | Mean Release Activity Difference (kCi) | 95% Confidence Interval Radius (kCi) | PARs | Manual ADS | IRWST | Vents | Filters | ATF |
|---|---|---|---|---|---|---|---|---|---|
| 1 | 62 | | | No | No | No | No | No | No |
| 14 | 52 | 1138.8 | 163.20 | Yes | Yes | Yes | Yes | Yes | No |
| 13 | 53 | 1094.3 | 166.53 | Yes | Yes | No | Yes | Yes | No |
| 10 | 55 | 1088.4 | 170.59 | Yes | No | No | Yes | Yes | No |
| 2 | 54 | 1057.5 | 165.00 | No | Yes | No | Yes | Yes | No |
| 7 | 56 | 1046.6 | 163.04 | No | Yes | Yes | Yes | Yes | No |
| 15 | 47 | 963.94 | 168.33 | Yes | No | Yes | Yes | Yes | No |
| 4 | 60 | 924.21 | 167.17 | No | No | Yes | Yes | Yes | No |
| 6 | 58 | 745.32 | 164.23 | No | Yes | Yes | Yes | No | No |
| 9 | 48 | 743.71 | 177.82 | Yes | No | No | Yes | No | No |
| 16 | 55 | 743.35 | 169.08 | Yes | No | Yes | Yes | No | No |
| 12 | 53 | 725.86 | 171.36 | Yes | Yes | No | Yes | No | No |
| 3 | 62 | 662.77 | 170.19 | No | No | Yes | Yes | No | No |

Unsurprisingly, Test-14, which included all of the successful upgrades, provided the greatest reduction. Combining the IRWST, Manual ADS, PARs, and Filtered Vents systems gave a massive reduction in Release Activity. Test-13 was a close second, though, and removes the extremely costly IRWST upgrade for a modest reduction in overall upgrade configuration efficacy.

### 5.5. *Refined Multi-Parameter Accident Scenario Upgrade Configuration Analysis*

After correcting the errors found within the MELCOR model used during the Initial Multi-Parameter Accident Scenario Upgrade Configuration Analysis, a second, refined analysis was performed. Fixing the errors both greatly improved the numerical stability of the MELCOR model and altered many of the results obtained. Additionally, the greater numerical stability meant that many more of the results were successful, allowing for far examination of individual upgrades in much greater detail. The results of this refined analysis are discussed in this section. A test matrix of all the upgrade combinations examined, and the numbering of the tests performed, is shown in Table 29.

Table 29. Refined Multi-Parameter Accident Scenario Upgrade Configuration Test Matrix

| Test Case | Filters | Vents | PARs | Igniters | IRWST |
|---|---|---|---|---|---|
| 1 | No | No | No | No | No |
| 2 | No | No | No | No | Yes |
| 3 | No | No | No | Yes | No |
| 4 | No | No | No | Yes | Yes |
| 5 | No | No | Yes | No | No |
| 6 | No | No | Yes | No | Yes |
| 7 | No | No | Yes | Yes | No |
| 8 | No | No | Yes | Yes | Yes |
| 9 | No | Yes | No | No | No |
| 10 | No | Yes | No | No | Yes |
| 11 | No | Yes | No | Yes | No |
| 12 | No | Yes | No | Yes | Yes |
| 13 | No | Yes | Yes | No | No |
| 14 | No | Yes | Yes | No | Yes |
| 15 | No | Yes | Yes | Yes | No |
| 16 | No | Yes | Yes | Yes | Yes |
| 17 | Yes | Yes | No | No | No |
| 18 | Yes | Yes | No | No | Yes |
| 19 | Yes | Yes | No | Yes | No |
| 20 | Yes | Yes | No | Yes | Yes |
| 21 | Yes | Yes | Yes | No | No |
| 22 | Yes | Yes | Yes | No | Yes |
| 23 | Yes | Yes | Yes | Yes | No |
| 24 | Yes | Yes | Yes | Yes | Yes |

### 5.5.1. General Upgrade Configuration Impact Examination, 100% Power

Table 30. 100% Power Upgrade Configuration Release Activity Results

| Test Case | Mean Release Activity | Median Release Activity | Maximum Release Activity | Sigma Release Activity |
|---|---|---|---|---|
| No Upgrades | 1768.2 | 1856.6 | 2400.6 | 474.4 |
| IRWST | 1729.7 | 1840.3 | 2540.3 | 482.7 |
| Igniters | 1796.8 | 1905.0 | 2472.2 | 494.4 |
| Igniters, IRWST | 1735.9 | 1840.2 | 2540.3 | 485.6 |

| | | | | |
|---|---|---|---|---|
| PARs | 1767.9 | 1888.6 | 2472.2 | 527.4 |
| PARs, IRWST | 1720.5 | 1818.9 | 2540.3 | 474.2 |
| PARs, Igniters | 1808.5 | 1914.4 | 2472.2 | 476.8 |
| PARs, Igniters, IRWST | 1739.6 | 1833.5 | 2540.3 | 481.0 |
| Vents | 1366.1 | 1344.3 | 2013.2 | 194.9 |
| Vents, IRWST | 1371.2 | 1366.9 | 1765.7 | 130.5 |
| Vents, Igniters | 1358.2 | 1329.4 | 2066.7 | 212.3 |
| Vents, Igniters, IRWST | 1347.0 | 1359.0 | 1694.1 | 126.8 |
| Vents, PARs | 1256.7 | 1236.1 | 2082.7 | 189.8 |
| Vents, PARs, IRWST | 1300.1 | 1287.3 | 1881.9 | 150.9 |
| Vents, PARs, Igniters | 1277.5 | 1264.6 | 2082.7 | 196.9 |
| Vents, PARs, Igniters, IRWST | 1302.9 | 1301.0 | 2098.1 | 144.7 |
| Vents, Filters | 1064.6 | 1050.3 | 1678.0 | 150.0 |
| Vents, Filters, IRWST | 1117.6 | 1111.8 | 1549.2 | 114.6 |
| Vents, Filters, Igniters | 1057.8 | 1059.4 | 1748.0 | 165.6 |
| Vents, Filters, Igniters, IRWST | 1093.0 | 1099.6 | 1334.4 | 101.2 |
| Vents, Filters, PARs | 970.1 | 971.2 | 1558.2 | 146.4 |
| Vents, Filters, PARs, IRWST | 1050.5 | 1036.7 | 1593.0 | 133.6 |
| Vents, Filters, PARs, Igniters | 986.5 | 982.3 | 1558.2 | 154.5 |
| Vents, Filters, PARs, Igniters, IRWST | 1054.1 | 1047.6 | 1769.9 | 128.4 |

Table 30 shows basic numerical statistics about all of the upgrade configurations tested at 100% Reactor Power. It is provided here as a reference for the reader. The cases are clustered by whether the containment was non-ventilated, ventilated without filtration (the Hardened Containment Ventilation upgrade, or "Vents" for short), or ventilated with filtration ("Vents, Filters") as these upgrades had the greatest impact on the Release Activity in this test.

Table 31. Comparison between all Upgrade Configurations and Unupgraded case, 100% Power

| Test Case | # of Points | Mean Release Activity | Sigma Release Activity | Mean Release Activity Difference | 95% Confidence Interval Radius | High Mean Difference | Low Mean Difference |
|---|---|---|---|---|---|---|---|
| No Upgrades | 162.00 | 1768.23 | 474.40 | | | | |
| IRWST | 164.00 | 1729.70 | 482.66 | 38.53 | 104.42 | 142.96 | -65.89 |
| Igniters | 162.00 | 1796.80 | 494.37 | -28.57 | 106.05 | 77.48 | -134.62 |
| Igniters, IRWST | 164.00 | 1735.91 | 485.60 | 32.31 | 104.75 | 137.06 | -72.43 |
| PARs | 164.00 | 1767.85 | 527.37 | 0.38 | 109.42 | 109.80 | -109.04 |
| PARs, IRWST | 157.00 | 1720.45 | 474.23 | 47.77 | 104.65 | 152.42 | -56.87 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| PARs, Igniters | 167.00 | 1808.49 | 476.76 | -40.27 | 103.31 | 63.05 | -143.58 |
| PARs, Igniters, IRWST | 157.00 | 1739.59 | 480.96 | 28.64 | 105.40 | 134.04 | -76.76 |
| Vents | 162.00 | 1366.14 | 194.94 | 402.09 | 79.38 | 481.47 | 322.71 |
| Vents, IRWST | 167.00 | 1371.25 | 130.54 | 396.98 | 76.08 | 473.06 | 320.91 |
| Vents, Igniters | 163.00 | 1358.22 | 212.28 | 410.01 | 80.40 | 490.41 | 329.61 |
| Vents, Igniters, IRWST | 166.00 | 1347.04 | 126.84 | 421.19 | 75.94 | 497.13 | 345.25 |
| Vents, PARs | 166.00 | 1256.71 | 189.75 | 511.52 | 78.95 | 590.47 | 432.57 |
| Vents, PARs, IRWST | 167.00 | 1300.11 | 150.95 | 468.12 | 76.95 | 545.07 | 391.17 |
| Vents, PARs, Igniters | 166.00 | 1277.53 | 196.90 | 490.70 | 79.36 | 570.06 | 411.34 |
| Vents, PARs, Igniters, IRWST | 163.00 | 1302.92 | 144.69 | 465.31 | 76.75 | 542.05 | 388.56 |
| Vents, Filters | 162.00 | 1064.57 | 149.95 | 703.66 | 77.01 | 780.67 | 626.65 |
| Vents, Filters, IRWST | 167.00 | 1117.55 | 114.65 | 650.67 | 75.48 | 726.15 | 575.20 |
| Vents, Filters, Igniters | 163.00 | 1057.84 | 165.64 | 710.38 | 77.75 | 788.13 | 632.64 |
| Vents, Filters, Igniters, IRWST | 166.00 | 1092.99 | 101.25 | 675.23 | 75.04 | 750.27 | 600.19 |
| Vents, Filters, PARs | 166.00 | 970.14 | 146.38 | 798.09 | 76.76 | 874.85 | 721.32 |
| Vents, Filters, PARs, IRWST | 167.00 | 1050.50 | 133.59 | 717.73 | 76.20 | 793.93 | 641.53 |
| Vents, Filters, PARs, Igniters | 166.00 | 986.47 | 154.47 | 781.76 | 77.13 | 858.89 | 704.63 |
| Vents, Filters, PARs, Igniters, IRWST | 163.00 | 1054.09 | 128.39 | 714.14 | 76.05 | 790.19 | 638.09 |

Table 31 shows a broad comparison between all of the tested upgrade configurations and the base case, for the 100% Reactor Power case. Using the Welch Treatment for generating a 95% Confidence Interval, none of the upgrades but the Hardened Vents and Filtered Vents created a statistically significant reduction in Release Activity. However, the possibility remains for statistically significant reductions in Release Activity when these upgrades are added on top of the Hardened Vents and/or Filtered Vents upgrades.  This possibility is examined in the following sections.

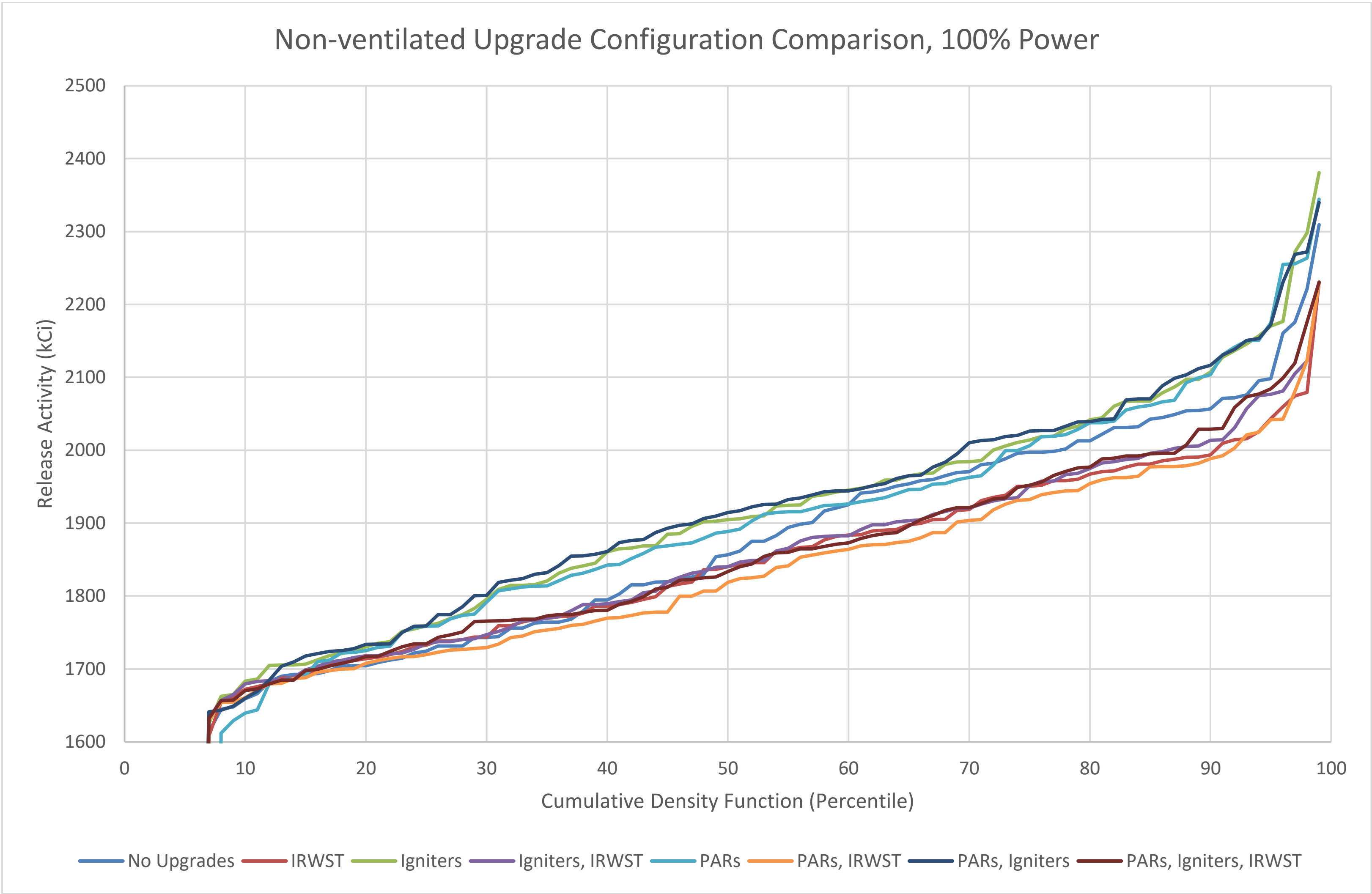


Figure 77 - Non-ventilated Containment Upgrade Configuration Results, 100% Power

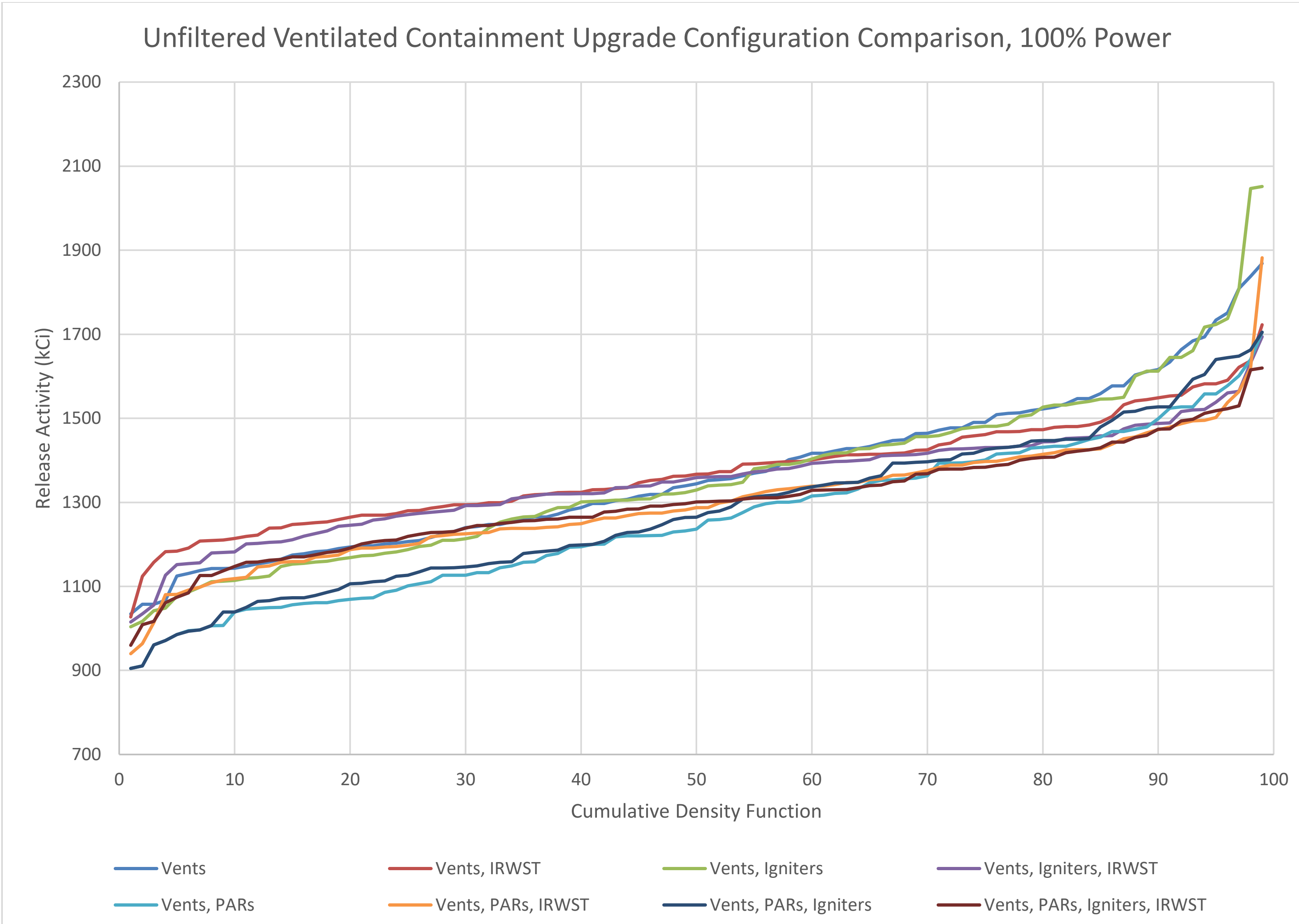


Figure 78 - Unfiltered Ventilated Containment Upgrade Configuration Comparison, 100% Power

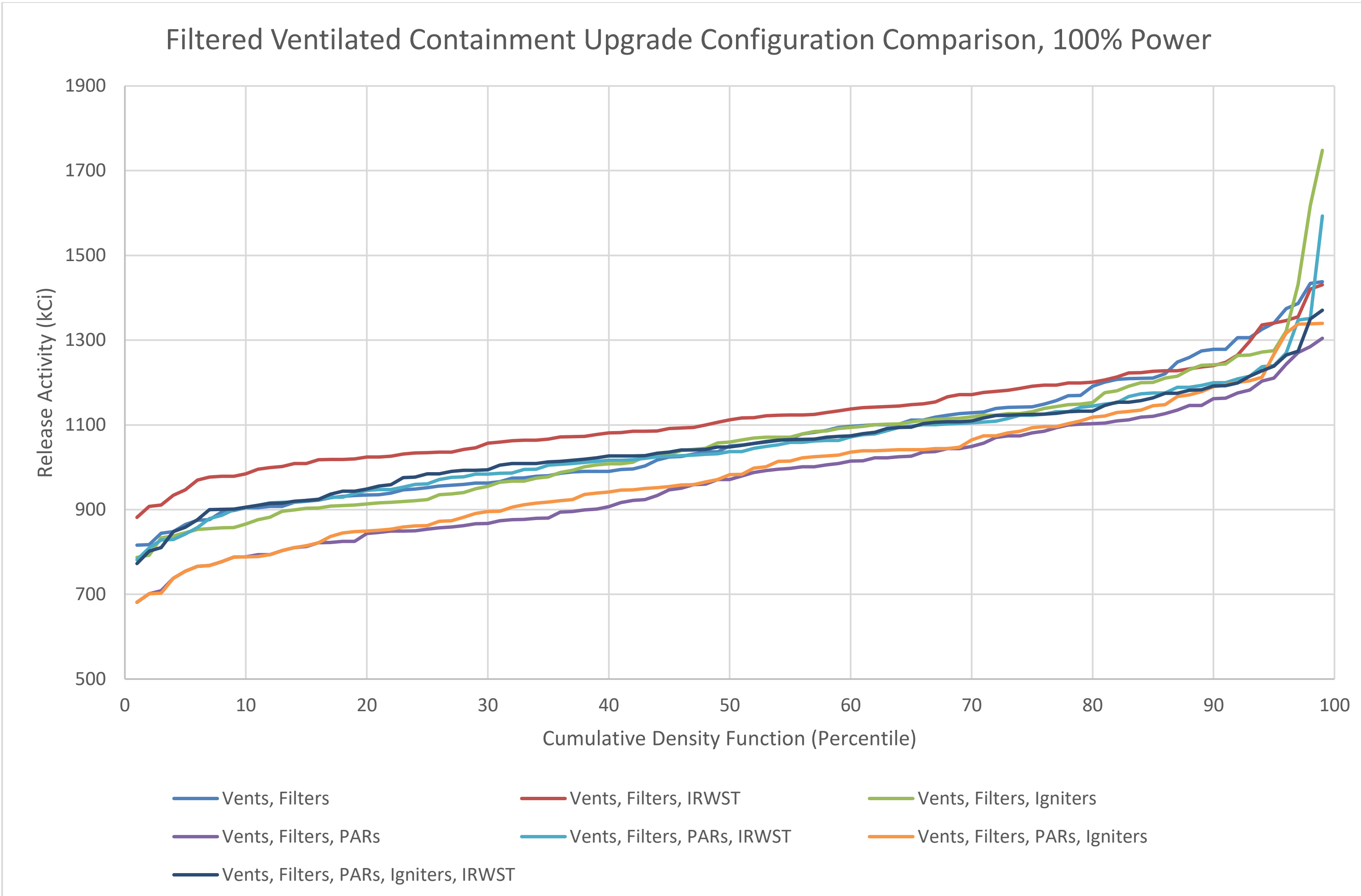


Figure 79 – Filtered Ventilated Containment Upgrade Configuration Comparison, 100% Power

Figure 77, Figure 78, and Figure 79 present a general visual overview of trends within the Release Activity CDFs for all upgrade configurations tested, at 100% power. Figure 77 cuts off the results at roughly the 8% mark intentionally, as the CDF value drops to 0 kCi Release Activity for non-ventilated configurations at this point. This represents collections of stochastic parameters sampled such that despite the LT-SBO, the containment did not rupture, preventing radionuclide release entirely. An interesting side effect of the implementation of Hardened Vents was that their presence made this small subset of sampled cases worse. There were cases sampled with a very long EDG Failure Time, a very cold and reasonably full, but not overfilled, Wetwell, a very long DC Power Lifetime, and a high Containment Failure Pressure in which, even without proper ventilation, the Containment did not fail, leading to no release, and zero Release Activity. This is shown in greater detail in Section 5.2.4 and Section 5.2.5, in which the Hardened Containment Ventilation upgrade and Filtered Containment Ventilation upgrade, respectively, are examined.

### 5.5.2. Hardened Containment Ventilation Examination

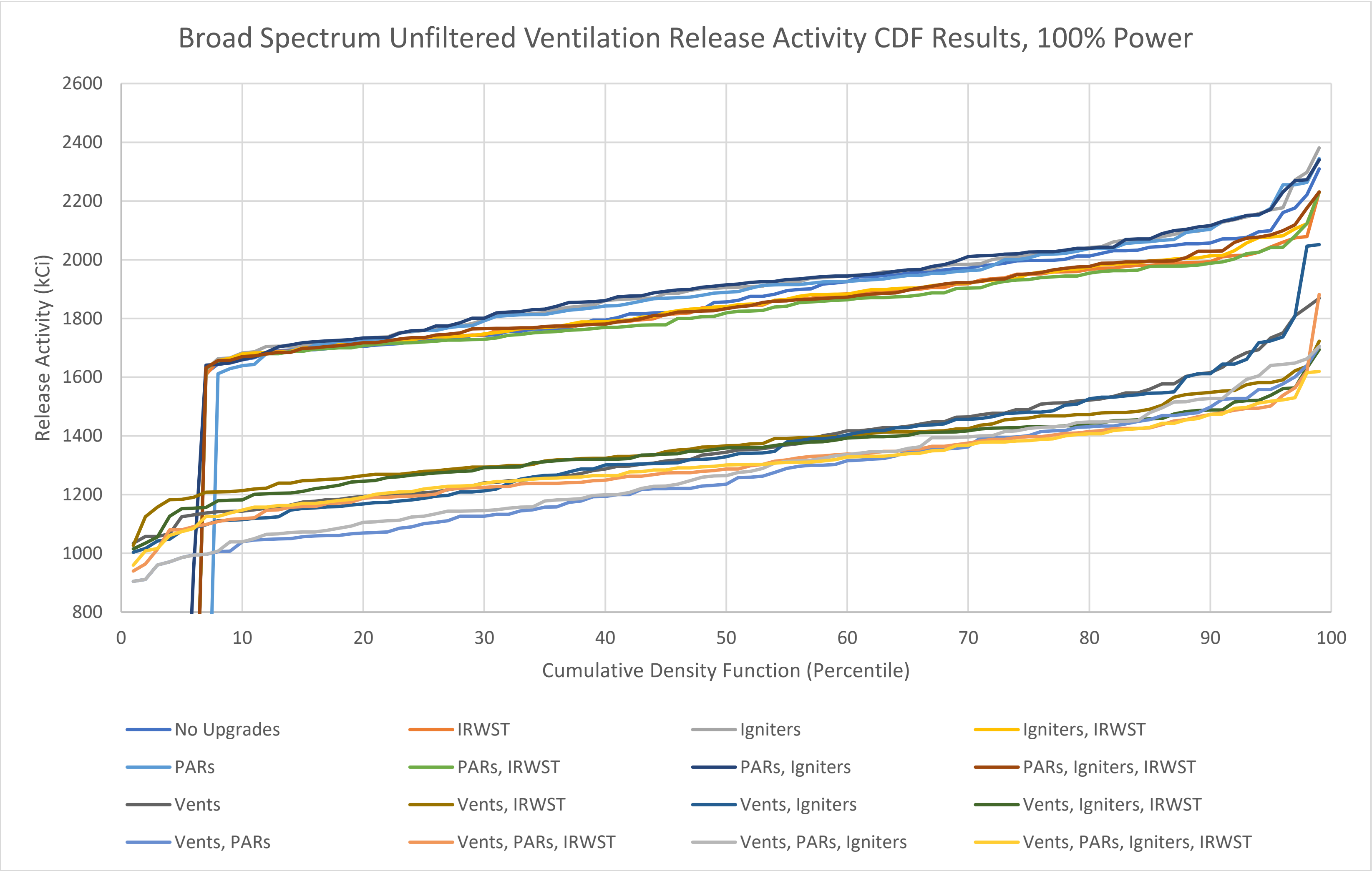


Figure 80 - Unfiltered Ventilation Broad Spectrum Comparison, 100% Power

Figure 80 presents a general visual overview of the impact of the Hardened Vents system. Visual examination of the comparison between the ventilated and unventilated containment cases show that, for whatever impact other upgrades have, they are dwarfed by the impact of the Hardened Vents system here – in both figures, the results are strongly clustered by whether each configuration included the Hardened Vents or not. As discussed in Section 5.5.1, for particularly favorable sets of stochastic parameters, it is possible for the Hardened Containment Ventilation upgrade to increase the Release Activity of the scenario by venting steam to the environment to relieve pressure within the containment when containment failure would not occur even without the Hardened Containment Ventilation upgrade, but this is a small minority of the CDF, and of the cases sampled. In general, the Hardened Containment Ventilation upgrade greatly reduces the Release Activity of each scenario.

Table 32. Hardened Containment Ventilation Means Difference Analysis Results

| Test Case | Number of Points | Mean Release Activity | Sigma Release Activity | Mean Release Difference | 95% Confidence Interval Radius | High Mean Difference | Low Mean Difference |
|---|---|---|---|---|---|---|---|
| Vents | 162.0 | 1366.1 | 194.9 | 402.1 | 79.4 | 481.5 | 322.7 |
| Vents, IRWST | 167.0 | 1371.2 | 130.5 | 358.4 | 115.9 | 474.3 | 242.6 |
| Vents, Igniters | 163.0 | 1358.2 | 212.3 | 438.6 | 167.7 | 606.3 | 270.8 |
| Vents, Igniters, IRWST | 166.0 | 1347.0 | 126.8 | 388.9 | 194.7 | 583.6 | 194.2 |
| Vents, PARs | 166.0 | 1256.7 | 189.8 | 511.1 | 261.1 | 772.2 | 250.0 |
| Vents, PARs, IRWST | 167.0 | 1300.1 | 150.9 | 420.3 | 276.1 | 696.4 | 144.3 |
| Vents, PARs, Igniters | 166.0 | 1277.5 | 196.9 | 531.0 | 318.3 | 849.2 | 212.7 |
| Vents, PARs, Igniters, IRWST | 163.0 | 1302.9 | 144.7 | 436.7 | 359.0 | 795.7 | 77.7 |

Table 32 shows the results of comparing the Release Activity for each upgrade configuration containing the Hardened Containment Ventilation upgrade to the upgrade configuration that is identical, other than lacking the Hardened Containment Ventilation upgrade.

The Mean Release Activity is calculated using the methodology outlined for Mean, or Expected Value, in Section 4.9. Sigma Release Activity is, similarly, calculated using the methodology outlined in Section 4.9. The Mean Difference is obtained by subtracting, for example, the Mean Release Activity for the “Vents” from the Mean Release Activity for the “No Upgrades” case, less processed data for both of which can be found in Table 30. It is worth noting that a positive value here indicates a decrease in Release Activity. The 95% Confidence Interval Radius is calculated using the Welch T-Test Confidence Interval method discussed in Section Uncertainty Quantification, and the High Mean Difference and Low Mean Difference are respectively obtained by adding or subtracting the 95% Confidence Interval Radius from the Mean Release Difference to obtain a pair of values in which there is 95% confidence that the “true” Mean Release Difference lies between. If the Low Mean Difference value is positive, a statistically significant reduction in Release Activity was discovered. If the High Mean Difference value is negative, a statistically significant *increase* in Release Activity was discovered, boding ill for the potential of the upgrade. Lastly, if the Low Mean Difference value is negative and the High Mean Difference value is positive, the Confidence Interval spans across zero change in Release Activity, and no statistically significant change in Release Activity was found.

In every case, the Hardened Containment Ventilation upgrade produced a statistically significant reduction in Release Activity. The Hardened Containment Ventilation produced the greatest minimum reduction in Release Activity when it was the only upgrade implemented but provided the greatest average reduction when accompanied by both the Hydrogen Igniters with backup power and PAR upgrades. It is not a statistically significant finding, but results indicate that it is very much possible that the Hardened Containment Ventilation upgrade provides the greatest reduction in Release Activity when accompanied by effective hydrogen mitigation.

### 5.5.3. Filtered Containment Ventilation Examination

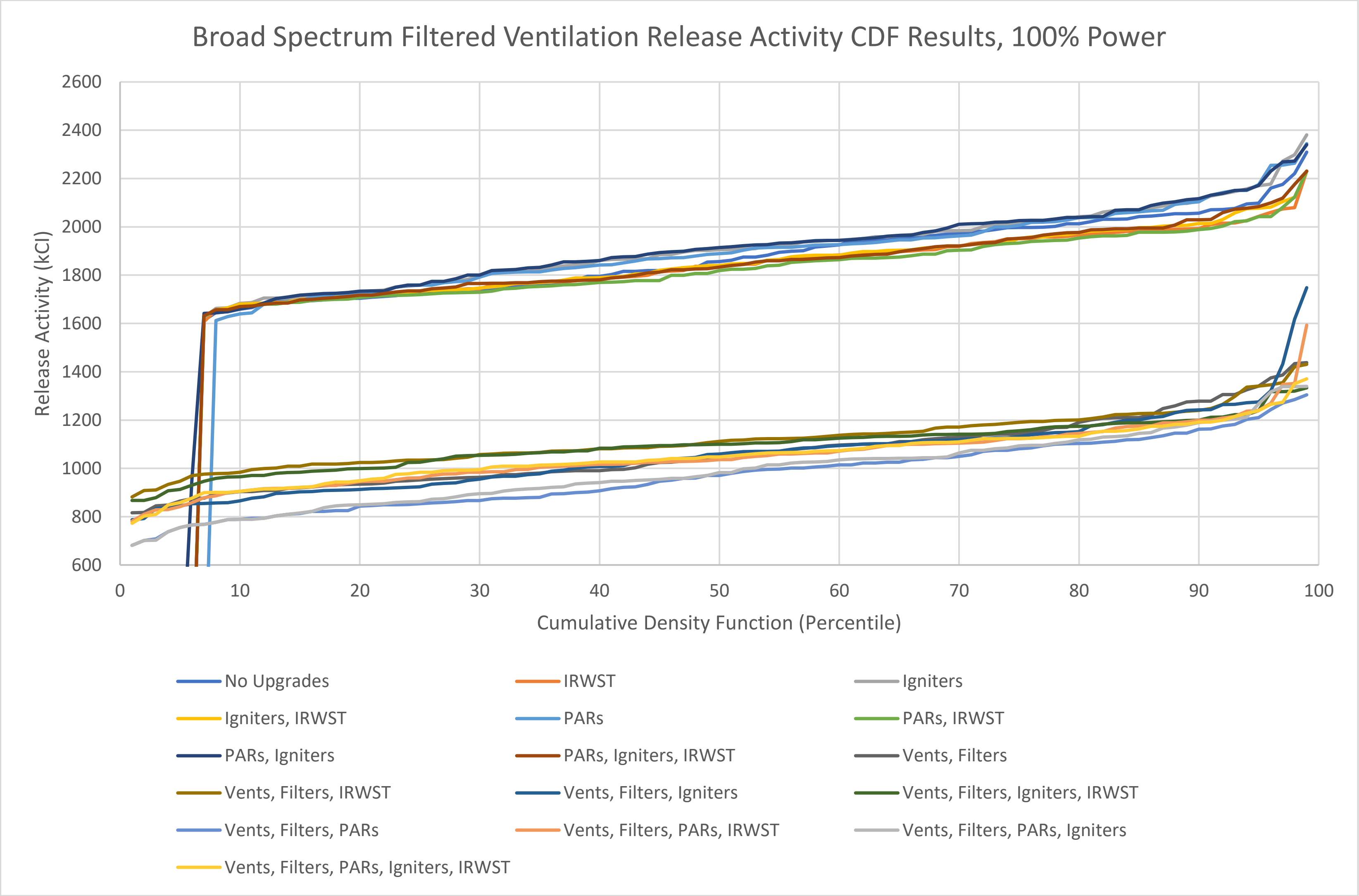


Figure 81 – Filtered Containment Ventilation Broad Spectrum Comparison, 100% Power

Figure 81 presents a general visual display of the impact of the Filtered Containment Ventilation upgrade on the accident scenario Release Activity CDF. Similar to the Hardened Containment Ventilation results, the impact of the Filtered Containment Ventilation upgrade was vastly greater than the impact of any other upgrades implemented alongside it, though the impact of the other upgrades is examined in greater detail in the coming sections.

Table 33. Filtered Containment Ventilation Means Difference Analysis Results, 100% Power

| Test Case | # of Points | Mean Release Activity (kCi) | Sigma Release Activity (kCi) | Mean Release Difference (kCi) | 95% Confidence Interval Radius | Low Mean Difference | High Mean Difference |
|---|---|---|---|---|---|---|---|
| Vents, Filters | 162 | 1064.6 | 149.95 | 301.57 | 38.066 | 339.64 | 263.50 |
| Vents, Filters, IRWST | 167 | 1117.6 | 114.65 | 253.69 | 39.930 | 293.62 | 213.76 |
| Vents, Filters, Igniters | 163 | 1057.8 | 165.64 | 300.38 | 83.726 | 384.10 | 216.65 |
| Vents, Filters, Igniters, IRWST | 166 | 1093.0 | 101.25 | 254.04 | 62.604 | 316.65 | 191.44 |
| Vents, Filters, PARs | 166 | 970.14 | 146.38 | 286.56 | 111.04 | 397.61 | 175.52 |
| Vents, Filters, PARs, IRWST | 167 | 1050.5 | 133.59 | 249.61 | 108.72 | 358.32 | 140.89 |
| Vents, Filters, PARs, Igniters | 166 | 986.47 | 154.47 | 291.06 | 154.81 | 445.87 | 136.25 |
| Vents, Filters, PARs, Igniters, IRWST | 163 | 1054.1 | 128.39 | 248.84 | 135.91 | 384.74 | 112.93 |

Table 33 shows the results of performing a Welch Treatment T-Test Confidence Interval Means Difference Analysis between each upgrade configuration that contained the Filtered Containment Ventilation upgrade and the corresponding upgrade configuration that had the Hardened Containment Ventilation upgrade, to examine the impact of the addition of ventilation filtration to each upgrade configuration. In every upgrade configuration, the addition of ventilation filtration provided a large, statistically significant reduction in Release Activity. The upgrade had the greatest average impact when upgrading from Hardened Containment Vents alone but had the greatest minimum Release Activity reduction when implemented alongside PARs and Igniters.

### 5.5.4. IRWST Examination

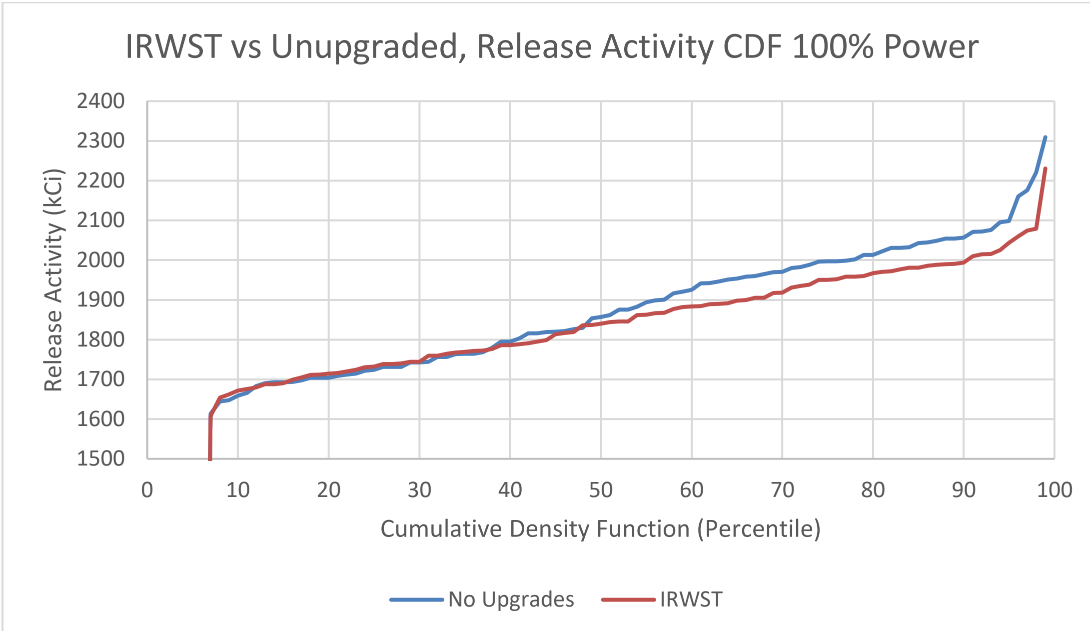


Figure 82 – Revised Multi-Parameter Analysis, IRWST vs Unupgraded Configuration, 100% Power

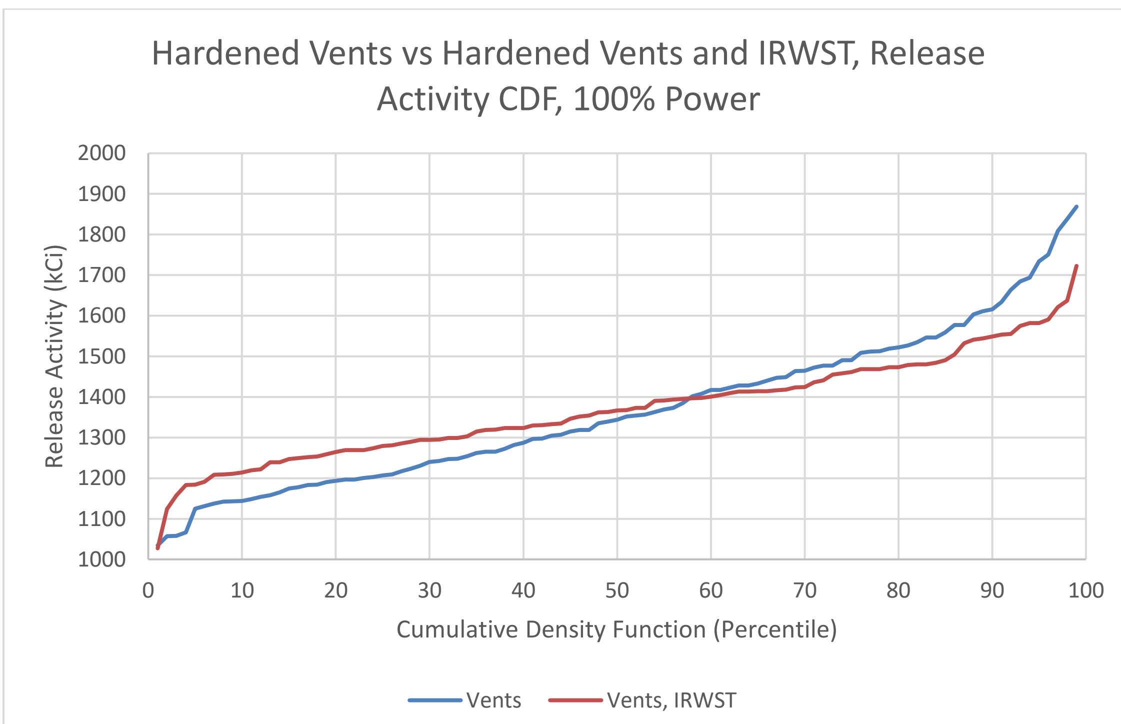


Figure 83 – Revised Multi-Parameter Analysis, Hardened Vents vs Hardened Vents and IRWST Configuration, 100% Power

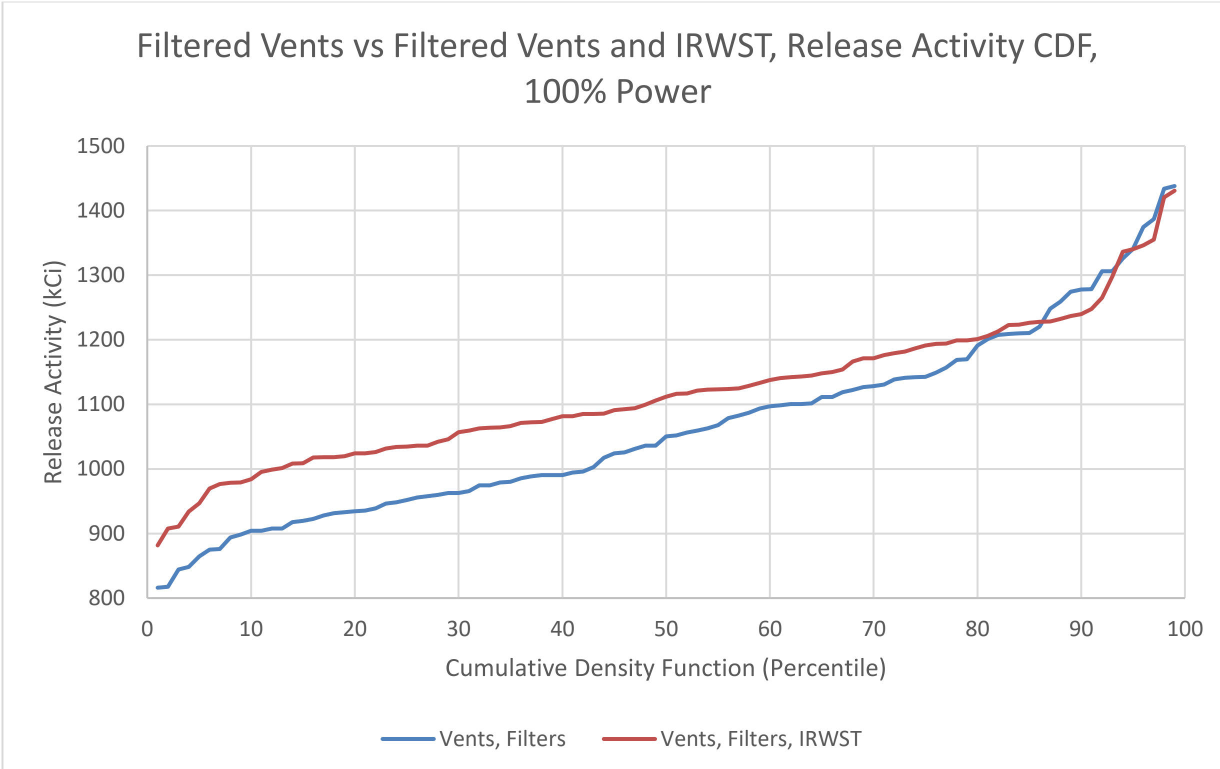


Figure 84 – Revised Multi-Parameter Analysis, Filtered Vents vs Filtered Vents and IRWST Configuration, 100% Power

Figure 82, Figure 83, and Figure 84 present a visual comparison of select IRWST results, comparing the unupgraded configuration, the Hardened Vents configuration, and the Filtered Vents configuration against the corresponding configurations that also include the IRWST. Rather than showing lumped results as with the Hardened Containment Ventilation and Filtered Containment Ventilation, the graphs shown here less aggressively display the data, as the change in Release Activity CDF from the IRWST upgrade was much less drastic than either ventilation upgrade and would get lost in the mass of data. For the purposes of this analysis, the Manual ADS and IRWST upgrades were lumped together, as prior modeling strongly suggested that there was little insight to be gained from pursuing them separately and combining them allowed for a significant reduction in the computational demands of this analysis.

Table 34. IRWST Means Difference Analysis Results

| Test Case | # of Points | Mean Release Activity | Sigma Release Activity | Mean Release Difference | 95% Confidence Interval Radius | Low Mean Difference | High Mean Difference |
|---|---|---|---|---|---|---|---|
| IRWST | 164.0 | 1729.7 | 482.7 | 38.5 | 104.4 | -65.9 | 143.0 |
| Igniters, IRWST | 164.0 | 1735.9 | 485.6 | 60.9 | 106.9 | -46.0 | 167.8 |
| PARs, IRWST | 157.0 | 1720.5 | 474.2 | 47.4 | 110.2 | -62.8 | 157.6 |
| PARs, Igniters, IRWST | 157.0 | 1739.6 | 481.0 | 68.9 | 104.9 | -36.0 | 173.8 |
| Vents, IRWST | 167.0 | 1371.2 | 130.5 | -5.1 | 36.1 | -41.3 | 31.0 |
| Vents, Igniters, IRWST | 166.0 | 1347.0 | 126.8 | 11.2 | 38.1 | -26.9 | 49.2 |
| Vents, PARs, IRWST | 167.0 | 1300.1 | 150.9 | -43.4 | 37.0 | -80.4 | -6.4 |
| Vents, PARs, Igniters, IRWST | 163.0 | 1302.9 | 144.7 | -25.4 | 37.5 | -62.9 | 12.1 |
| Vents, Filters, IRWST | 167.0 | 1117.6 | 114.6 | -53.0 | 29.1 | -82.0 | -23.9 |
| Vents, Filters, Igniters, IRWST | 166.0 | 1093.0 | 101.2 | -35.2 | 29.9 | -65.0 | -5.3 |
| Vents, Filters, PARs, IRWST | 167.0 | 1050.5 | 133.6 | -80.4 | 30.3 | -110.6 | -50.1 |
| Vents, Filters, PARs, Igniters, IRWST | 163.0 | 1054.1 | 128.4 | -67.6 | 30.8 | -98.4 | -36.8 |

Table 34 shows the results of performing a Welch Treatment T-Test Confidence Interval Means Difference Analysis between each upgrade configuration that included the IRWST upgrade and the corresponding upgrade configuration that did not contain the IRWST upgrade. While generally, on average it was more likely than not that the IRWST upgrade would confer at least a marginal reduction in Release Activity, no statistically significant reduction in Release Activity was found in any case, and in some cases a statistically significant increase in Release Activity was found.

### 5.5.5. PAR Examination

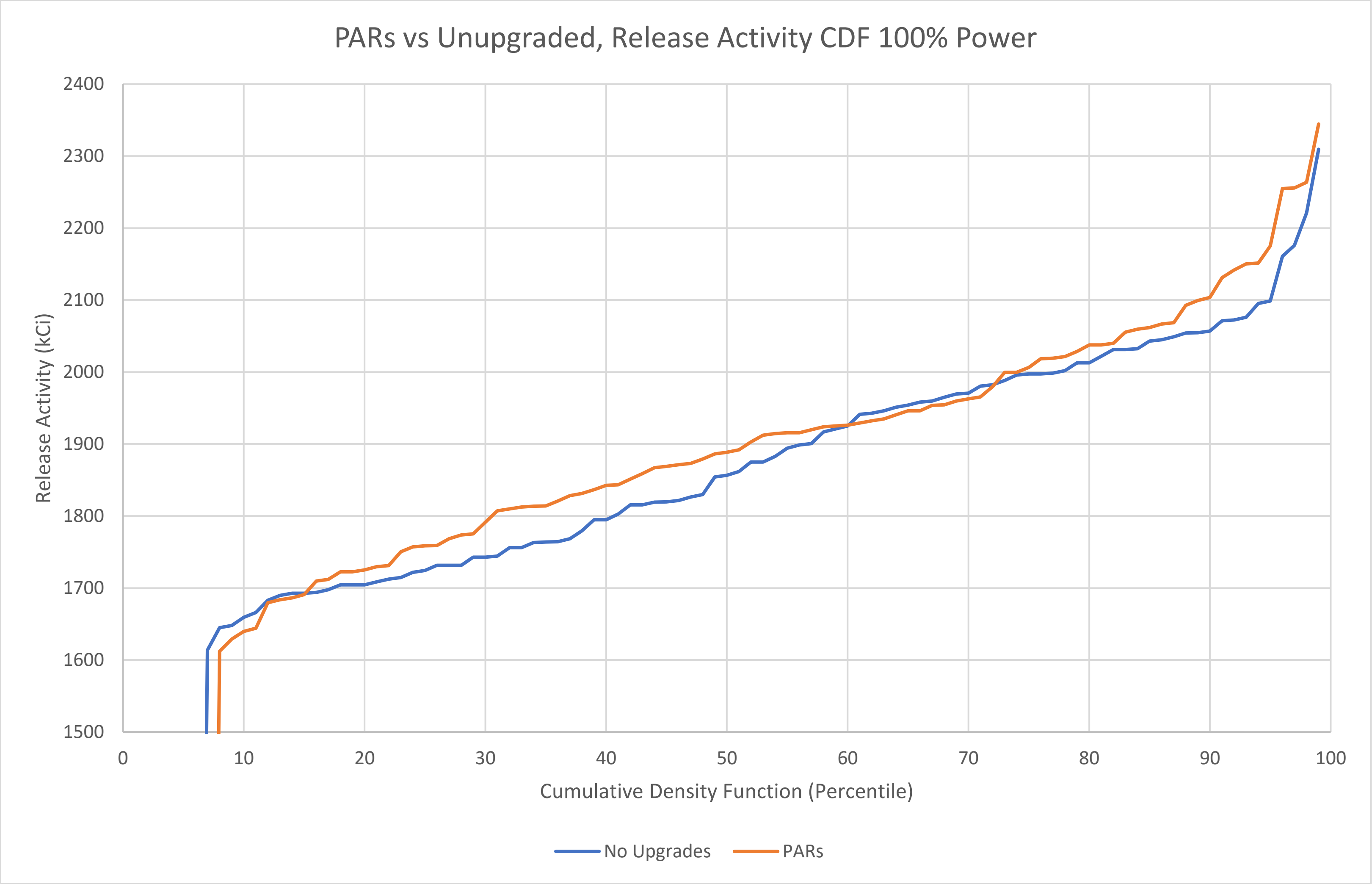


Figure 85 – Revised Multi-Parameter Analysis, PARs vs Unupgraded Configuration, 100% Power

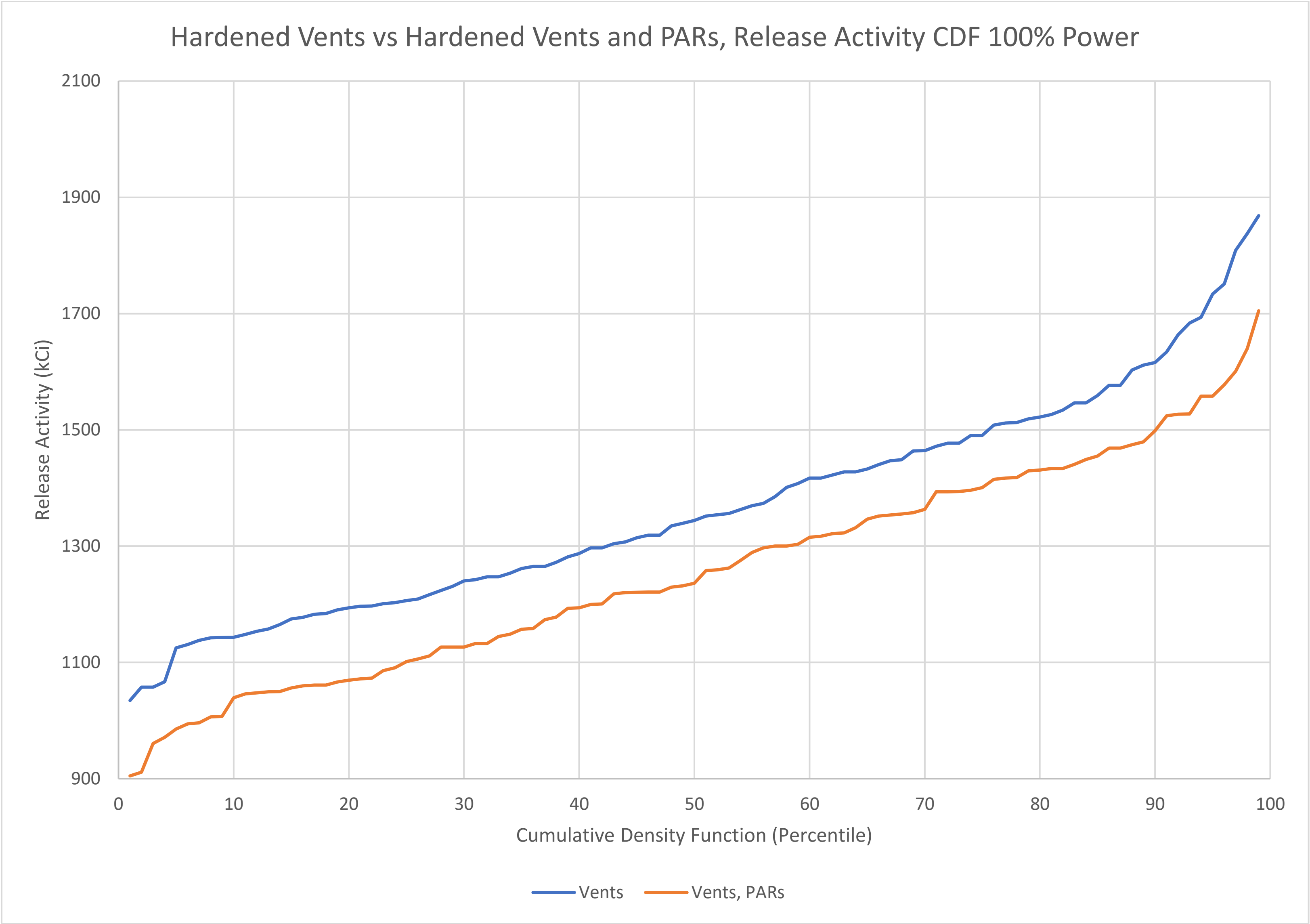


Figure 86 – Revised Multi-Parameter Analysis, Hardened Vents vs Hardened Vents and PAR Configuration, 100% Power

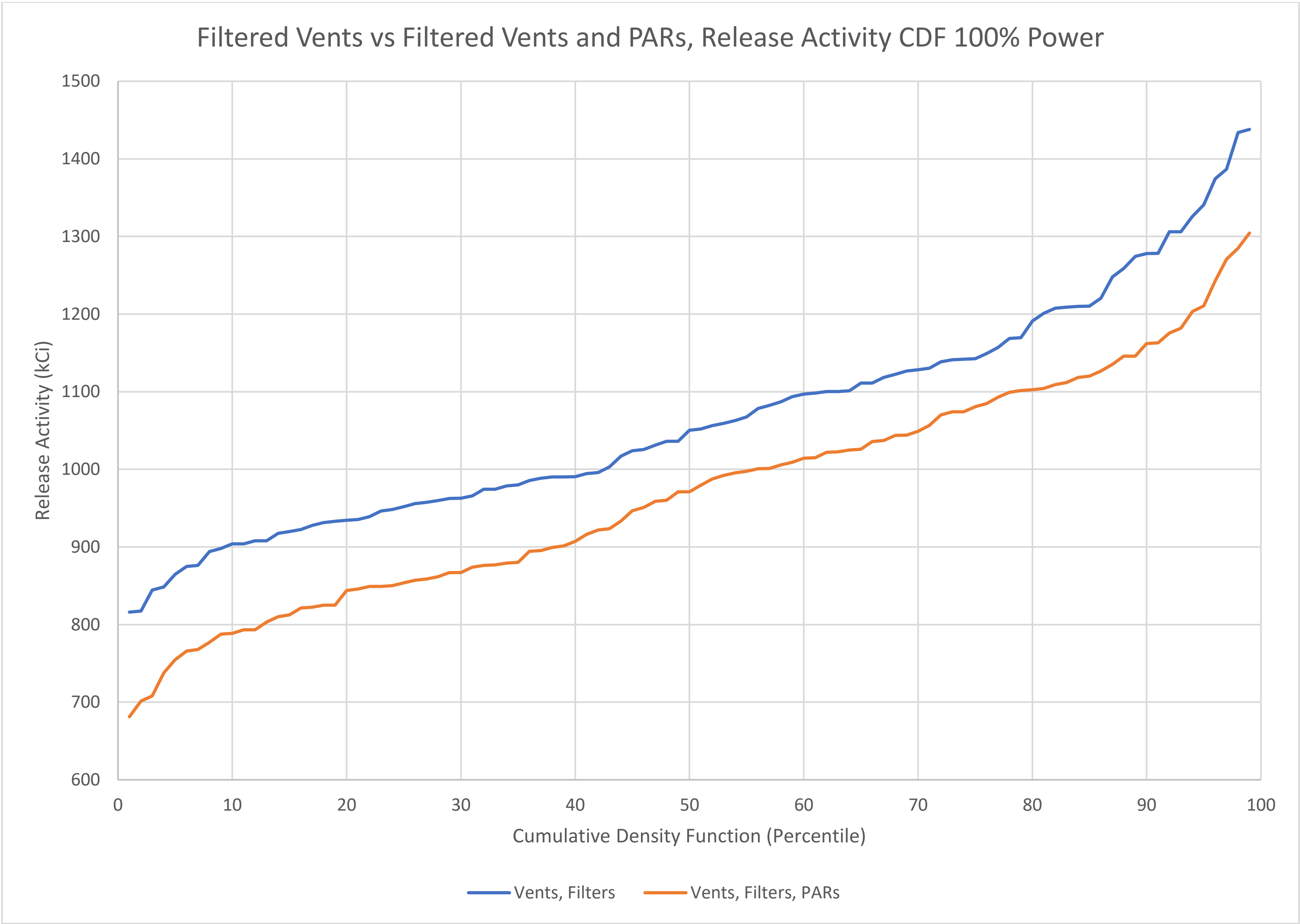


Figure 87 - Revised Multi-Parameter Analysis, Filtered Vents vs Filtered Vents and PAR Configuration, 100% Power

Figure 85, Figure 86, and Figure 87 present a visual comparison of select PAR results, comparing the unupgraded configuration, the Hardened Vents configuration, and the Filtered Vents configuration against the corresponding configurations that also include the PAR upgrade. Figure 85 cuts off the results at the 8$^{th}$ Percentile as the Release Activity for these results plummets to 0 kCi, as they represent cases in the non-ventilated containment configurations where the stochastic parameters sampled are such that the containment does not fail, causing no radioactive material at all to be released, and is of little visual interest. Visual examination of the results indicates that the impact of the PAR upgrade was, in general, minimal for the case in which PARs were implemented without the presence of a ventilation upgrade and provided a modest but significant reduction in Release Activity when implemented alongside the Hardened or Filtered Containment Ventilation upgrades.

Table 35. PAR Means Difference Analysis Results

| Test Case | # of Points | Mean Release Activity | Sigma Release Activity | Mean Difference | 95% Confidence Interval Radius | High Mean Difference | Low Mean Difference |
|---|---|---|---|---|---|---|---|
| PARs | 164 | 1767.85 | 527.37 | 0.3766 | 109.42 | 109.80 | -109.04 |
| PARs, IRWST | 157 | 1720.45 | 474.23 | 9.2418 | 105.22 | 114.47 | -95.982 |
| PARs, Igniters | 167 | 1808.50 | 476.76 | -11.694 | 105.53 | 93.839 | -117.23 |
| PARs, Igniters, IRWST | 157 | 1739.59 | 480.96 | -3.6755 | 106.29 | 102.62 | -109.97 |
| Vents, PARs | 166 | 1256.71 | 189.75 | 109.43 | 41.859 | 151.29 | 67.575 |
| Vents, PARs, IRWST | 167 | 1300.11 | 150.95 | 71.139 | 30.423 | 101.562 | 40.716 |
| Vents, PARs, Igniters | 166 | 1277.53 | 196.90 | 80.688 | 44.489 | 125.18 | 36.199 |
| Vents, PARs, Igniters, IRWST | 163 | 1302.92 | 144.69 | 44.115 | 29.572 | 73.687 | 14.543 |
| Vents, Filters, PARs | 166 | 970.143 | 146.38 | 94.425 | 32.243 | 126.67 | 62.182 |
| Vents, Filters, PARs, IRWST | 167 | 1050.50 | 133.59 | 67.052 | 26.836 | 93.888 | 40.217 |
| Vents, Filters, PARs, Igniters | 166 | 986.47 | 154.47 | 71.375 | 34.800 | 106.18 | 36.575 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| Vents, Filters, PARs, Igniters, IRWST | 163 | 1054.09 | 128.38 | 38.907 | 25.142 | 64.049 | 13.765 |

Table 35 shows the results of performing a Welch Treatment T-Interval Means Difference Analysis between each upgrade configuration that included the PAR upgrade and the corresponding upgrade configuration that did not contain the PAR upgrade. The reduction in Release Activity was, in general, relatively small. Without an accompanying ventilation system, filtered or hardened, the PAR upgrade provided no statistically significant reduction in release activity. Additionally, the PAR provided the greatest reduction in Release Activity when implemented alongside a ventilation system, and only a ventilation system, with no IRWST or Igniters present.

### 5.5.6. Hydrogen Igniter with Backup Power Examination

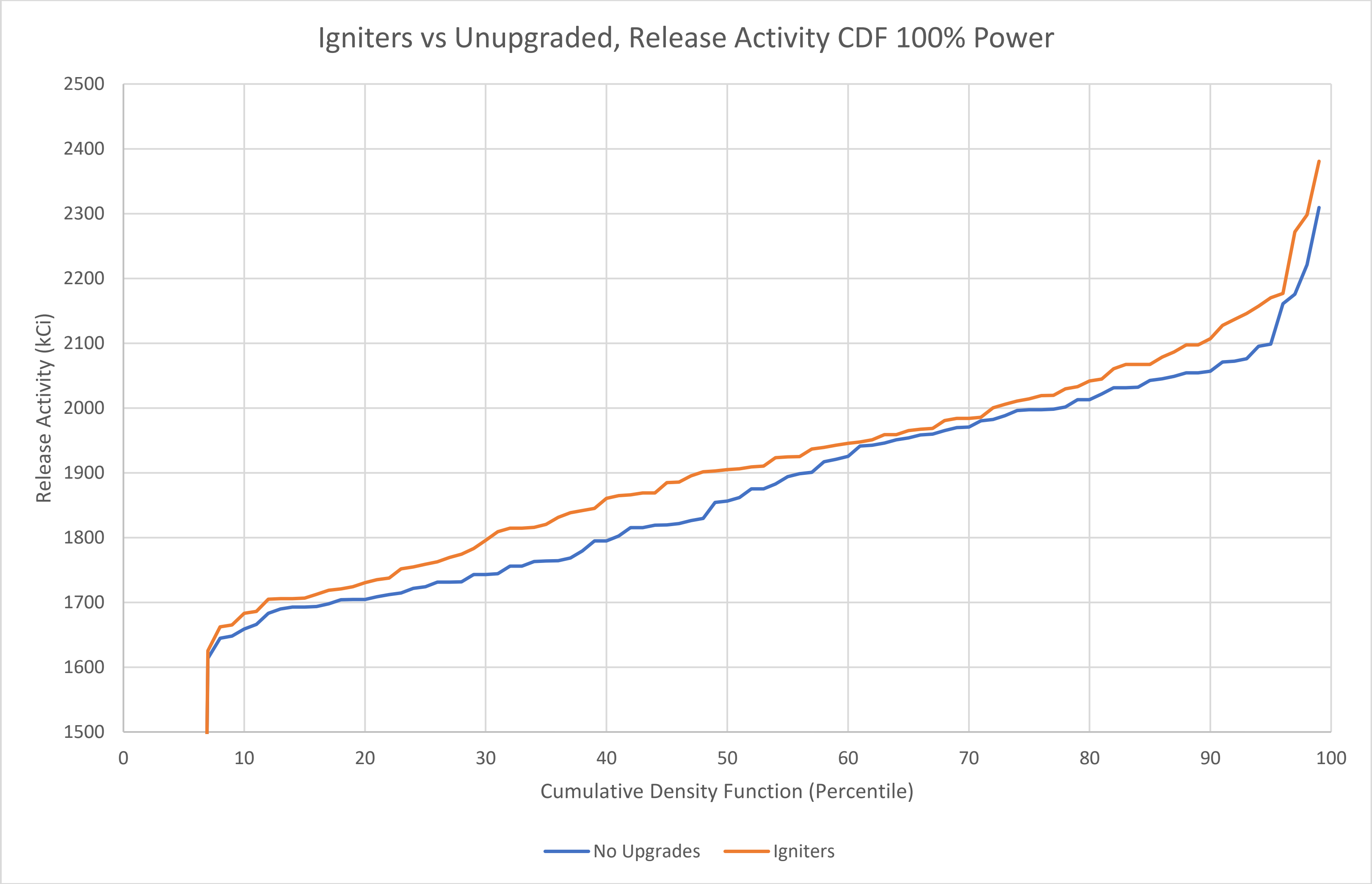


Figure 88 – Revised Multi-Parameter Analysis, Igniters vs Unupgraded Configuration, 100% Power

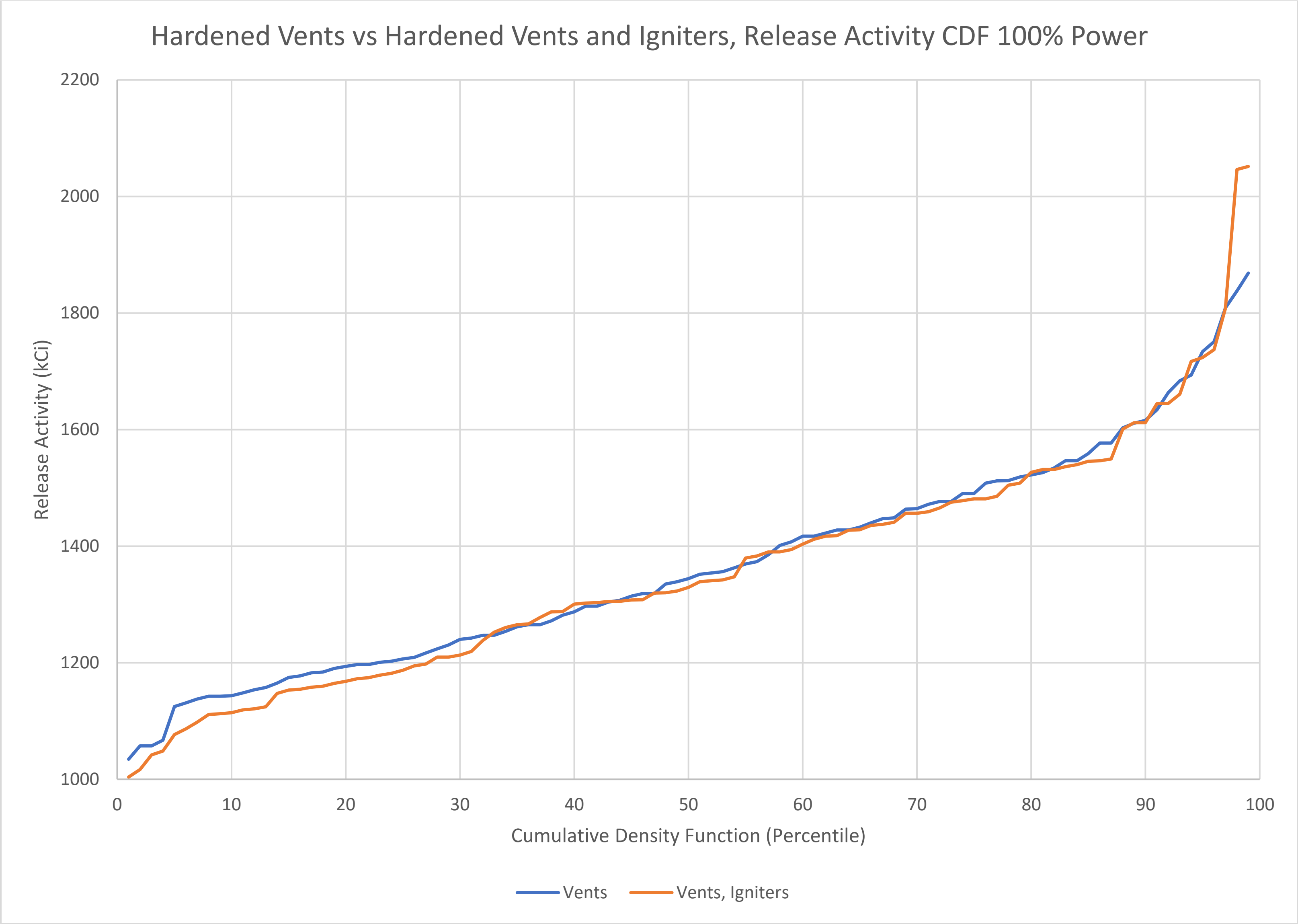


Figure 89 – Revised Multi-Parameter Analysis, Hardened Vents vs Hardened Vents and Igniters Configuration, 100% Power

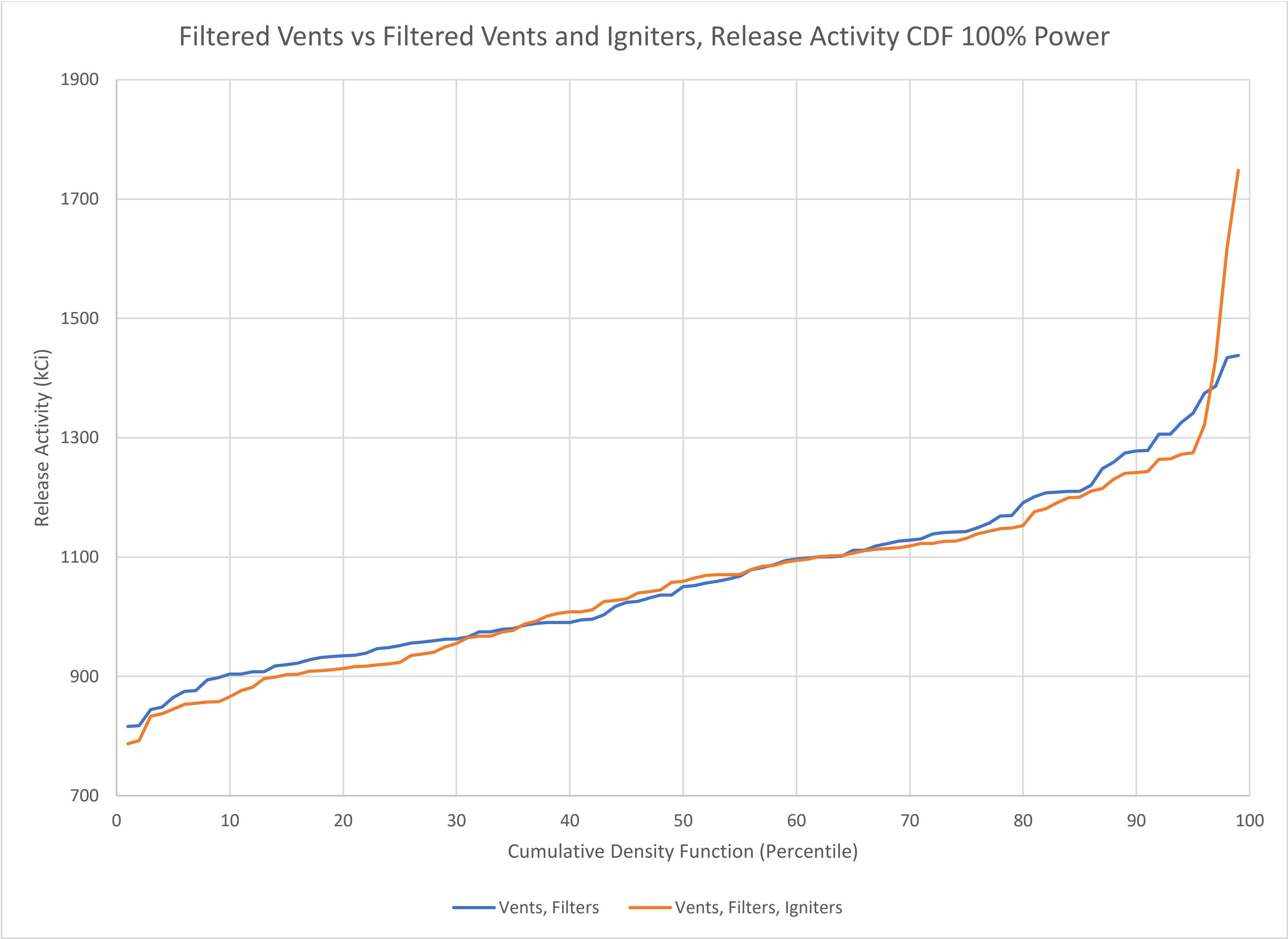


Figure 90 – Revised Multi-Parameter Analysis, Filtered Vents vs Filtered Vents and Igniters Configuration, 100% Power

Figure 88, Figure 89, and Figure 90 present a visual comparison of select Hydrogen Igniters with Backup Power results, comparing the unupgraded configuration, the Hardened Vents configuration, and the Filtered Vents configuration against the corresponding configurations that also include the Hydrogen Igniters with Backup Power upgrade. Figure 88 cuts off the results at the 8$^{th}$ Percentile as the Release Activity for these results plummets to 0 kCi, as they represent cases in the non-ventilated containment configurations where the stochastic parameters sampled are such that the containment does not fail, causing no radioactive material at all to be released, and is of little visual interest. Visual examination of the results indicates that the impact of the Hydrogen Igniter upgrade was, in general, minimal except for extreme cases in the 95$^{th}$ and higher percentile, where it increased the Release Activity. The mechanism for this increase in Release Activity is that the Hydrogen Igniters would ignite hydrogen in the Drywell, well after hydrogen had already accumulated in the Wetwell, and the burn would spread to the Wetwell and cause a deflagration in cases where otherwise the stoichiometric mix in either volume would not reach the appropriate concentration thresholds for spontaneous ignition. Given the DKW 95% confidence interval radius for the CDF percentile for these tests, roughly 12%, though, it is possible that these cases do not actually comprise ~5% of the possible outcomes and may range from essentially impossible scenarios to encompassing upwards of 17% of the possible results. Determining a more precise percentage of cases in which this scenario would occur would, however, demand tens of thousands of sample points of this particular upgrade configuration – having a 95% confidence DKW interval radius below 1% would require nearly 20,000 cases, and the number of cases required for further reduction in uncertainty grows inverse-quadratically with the desired uncertainty – 0.1% uncertainty would require nearly two million sample points, and is wholly impractical, even with the might of a supercomputer available.

### 5.5.7. Alternate Reactor Power Analysis

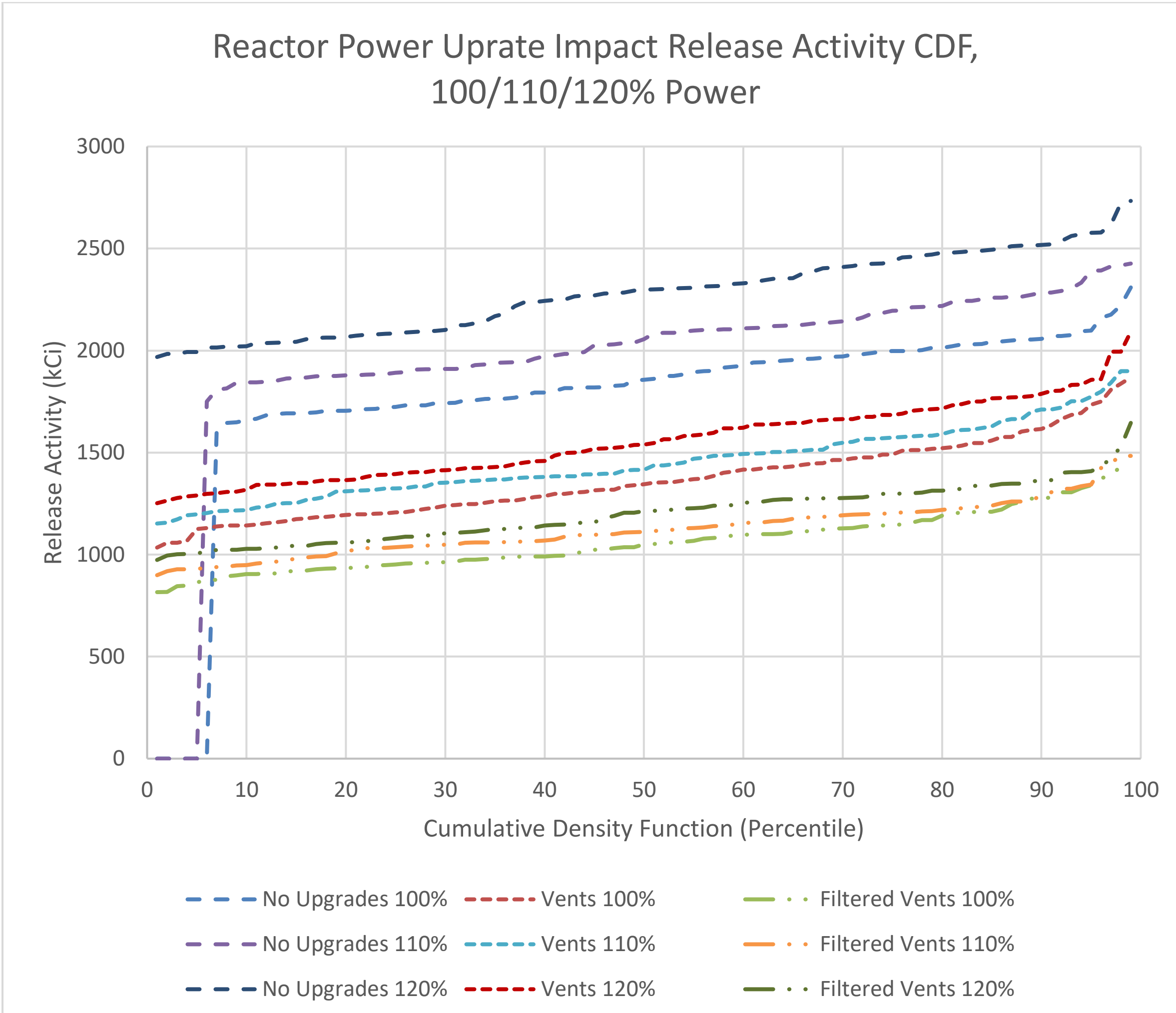


Figure 91 – Reactor Power Uprate Impact Release Activity CDF, 100/110/120% Power

Figure 91 shows a comparison of the Release Activity CDF for the unupgraded configuration, Hardened Containment Ventilation alone, and Filtered Containment Ventilation upgrades for 100%, 110%, and 120% power. As well as varying the five stochastic parameters used in this analysis, the impact of Reactor Power uprates to 110% and 120% was examined. Broadly speaking, this led to an increase in Release Activity in every examined case, and an uptick in the mean Release Activity for every upgrade configuration successfully examined. Four of the 24 upgrade configurations tests did not complete successfully at 110% power, but all others did, and all 24 tests finished successfully for both 100% and 120% power.

Table 36. 110% Reactor Power Means Difference Analysis Results

| Test Case | Number of Cases | Mean Release Activity | Release Activity Sigma | Mean Difference | 95% Confidence Interval Radius | High Mean Difference | Low Mean Difference |
|---|---|---|---|---|---|---|---|
| No Upgrades | 148.0 | 1947.0 | 517.0 | -178.8 | 128.30 | -50.501 | -307.12 |
| IRWST | 154.0 | 1970.0 | 245.0 | -201.8 | 95.742 | -106.06 | -297.54 |
| Igniters | N/A | N/A | N/A | | | | |
| Igniters, IRWST | 154.0 | 1945.1 | 331.1 | -176.9 | 104.05 | -72.799 | -280.91 |
| PARs | 158.0 | 1984.0 | 462.3 | -215.7 | 118.87 | -96.878 | -334.62 |
| PARs, IRWST | 160.0 | 1937.3 | 331.3 | -169.0 | 103.40 | -65.625 | -272.43 |
| PARs, Igniters | 148.0 | 1985.4 | 500.7 | -217.2 | 126.04 | -91.126 | -343.21 |
| PARs, Igniters, IRWST | N/A | N/A | N/A | | | | |
| Vents | 155.0 | 1453.6 | 181.3 | 314.6 | 90.837 | 405.44 | 223.761 |
| Vents, IRWST | 147.0 | 1463.2 | 141.4 | 305.1 | 88.654 | 393.72 | 216.413 |
| Vents, Igniters | 154.0 | 1441.1 | 185.7 | 327.1 | 91.170 | 418.29 | 235.96 |
| Vents, Igniters, IRWST | 159.0 | 1487.8 | 146.5 | 280.4 | 88.621 | 369.07 | 191.83 |
| Vents, PARs | 157.0 | 1360.2 | 195.6 | 408.0 | 91.733 | 499.73 | 316.27 |
| Vents, PARs, IRWST | N/A | N/A | N/A | | | | |
| Vents, PARs, Igniters | 156.0 | 1379.7 | 204.6 | 388.5 | 92.421 | 480.93 | 296.09 |
| Vents, PARs, Igniters, IRWST | 144.0 | 1400.3 | 130.8 | 367.9 | 88.153 | 456.06 | 279.75 |
| Vents, Filters | 155.0 | 1124.2 | 131.6 | 644.0 | 87.945 | 731.98 | 556.09 |
| Vents, Filters, IRWST | 147.0 | 1183.8 | 118.9 | 584.5 | 87.486 | 671.94 | 496.97 |
| Vents, Filters, Igniters | 154.0 | 1111.0 | 131.5 | 657.2 | 87.959 | 745.17 | 569.25 |
| Vents, Filters, Igniters, IRWST | 159.0 | 1198.2 | 122.6 | 570.0 | 87.441 | 657.47 | 482.59 |
| Vents, Filters, PARs | 157.0 | 1045.7 | 148.0 | 722.5 | 88.755 | 811.26 | 633.75 |
| Vents, Filters, PARs, IRWST | N/A | N/A | N/A | | | | |
| Vents, Filters, PARs, Igniters | 156.0 | 1056.6 | 156.9 | 711.7 | 89.28277 | 800.94 | 622.37 |
| Vents, Filters, PARs, Igniters, IRWST | 144.0 | 1124.4 | 118.0 | 643.8 | 87.50519 | 731.31 | 556.30 |

Table 36 shows the results of a Welch Treatment T-Interval Means Difference Analysis comparing every upgrade configuration at 110% Reactor Power to the unupgraded configuration at 100% Reactor Power. Every successfully tested upgrade configuration that did not include some form of containment ventilation showed a statistically significant increase in Mean Release Activity. Adding Hardened Containment Ventilation more than offset the increase in Release Activity from the increase in Reactor Power and created an overall reduction in mean Release Activity. Adding filtration to the containment ventilation yet further increased the reduction in Release Activity.

Table 37. 120% Reactor Power Means Difference Analysis Results

| Test Case | Number of Cases | Mean Release Activity | Release Activity Sigma | Mean Difference | 95% Confidence Interval Radius | High Mean Difference | Low Mean Difference |
|---|---|---|---|---|---|---|---|
| No Upgrades | 149.0 | 2267.3 | 283.8 | -499.0 | 109.95 | -389.07 | -608.98 |
| IRWST | 146.0 | 2173.6 | 157.6 | -405.4 | 100.90 | -304.51 | -506.31 |
| Igniters | 149.0 | 2283.4 | 274.2 | -515.2 | 109.111 | -406.04 | -624.27 |
| Igniters, IRWST | 146.0 | 2180.2 | 157.5 | -412.0 | 100.89 | -311.07 | -512.86 |
| PARs | 148.0 | 2291.4 | 290.3 | -523.2 | 110.63 | -412.534 | -633.80 |
| PARs, IRWST | 148.0 | 2142.9 | 293.7 | -374.7 | 110.94 | -263.78 | -485.66 |
| PARs, Igniters | 147.0 | 2304.4 | 283.5 | -536.2 | 110.10 | -426.08 | -646.28 |
| PARs, Igniters, IRWST | 149.0 | 2152.1 | 293.0 | -383.9 | 110.79 | -273.11 | -494.68 |
| Vents | 144.0 | 1555.8 | 191.2 | 212.4 | 103.02 | 315.43 | 109.40 |
| Vents, IRWST | 144.0 | 1608.5 | 178.0 | 159.7 | 102.17 | 261.88 | 57.552 |
| Vents, Igniters | 145.0 | 1538.2 | 210.0 | 230.1 | 104.27 | 334.34 | 125.80 |
| Vents, Igniters, IRWST | 148.0 | 1573.2 | 172.3 | 195.1 | 101.68 | 296.74 | 93.392 |
| Vents, PARs | 142.0 | 1456.7 | 223.5 | 311.5 | 105.44 | 416.97 | 206.09 |
| Vents, PARs, IRWST | 136.0 | 1484.7 | 167.5 | 283.5 | 101.82 | 385.36 | 181.71 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| Vents, PARs, Igniters | 142.0 | 1462.0 | 222.4 | 306.2 | 105.36 | 411.56 | 200.85 |
| Vents, PARs, Igniters, IRWST | 138.0 | 1491.2 | 164.4 | 277.0 | 101.55 | 378.59 | 175.48 |
| Vents, Filters | 144.0 | 1201.5 | 140.6 | 566.7 | 100.06 | 666.78 | 466.67 |
| Vents, Filters, IRWST | 144.0 | 1300.4 | 147.7 | 467.8 | 100.42 | 568.23 | 367.38 |
| Vents, Filters, Igniters | 145.0 | 1186.1 | 158.5 | 582.2 | 100.9826 | 683.143 | 481.1778 |
| Vents, Filters, Igniters, IRWST | 148.0 | 1263.0 | 138.4 | 505.2 | 99.86014 | 605.0808 | 405.3605 |
| Vents, Filters, PARs | 142.0 | 1114.1 | 171.4 | 654.1 | 101.8376 | 755.9427 | 552.2674 |
| Vents, Filters, PARs, IRWST | 136.0 | 1192.0 | 139.5 | 576.3 | 100.209 | 676.4637 | 476.0457 |
| Vents, Filters, PARs, Igniters | 142.0 | 1118.3 | 170.8 | 650.0 | 101.7983 | 751.7754 | 548.1788 |
| Vents, Filters, PARs, Igniters, IRWST | 138.0 | 1196.3 | 137.1 | 572.0 | 100.0337 | 672.0056 | 471.9382 |

Table 37 shows the results of a Welch Treatment T-Interval Means Difference Analysis comparing every upgrade configuration at 120% Reactor Power to the unupgraded configuration at 100% Reactor Power. Comparing the 120% Reactor Power results to the 110% Reactor Power results, every non-ventilated upgrade configuration showed a worse increase in mean Release Activity under 120% Reactor Power than 110% Reactor Power. Additionally, while every configuration containing the Hardened Containment Ventilation upgrade still provided for an overall reduction in mean Release Activity compared to the unupgraded configuration at 100%

Reactor Power, the reduction was smaller and, in some cases, approaching no longer being statistically significant. Lastly, the mean Release Activity reduction provided by the Filtered Containment Ventilation upgrade configurations was again larger, across the board, than the mean Release Activity reduction provided by the Hardened Containment Ventilation upgrade configurations, the reduction in Mean Release Activity was smaller at 120% Reactor Power than at 110% Reactor Power.

### 5.5.8. Stochastic Parameter Alternate Probability Distribution Function Analysis

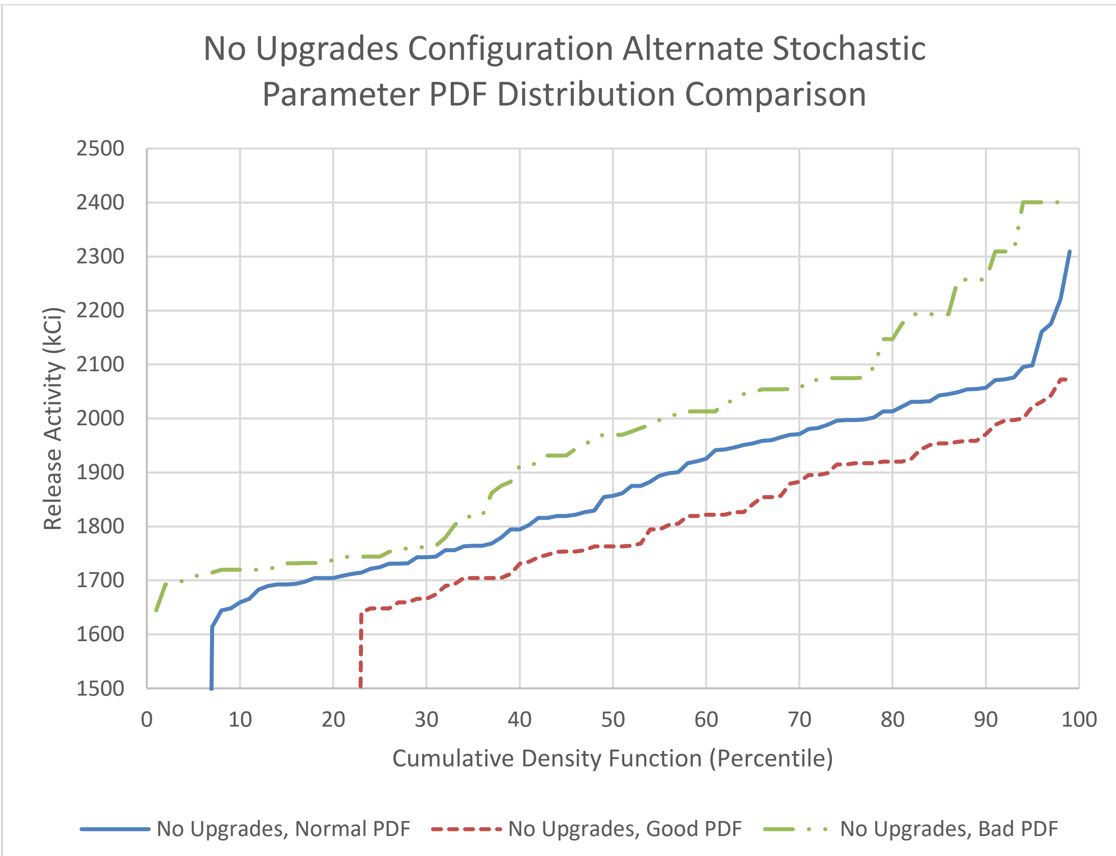


Figure 92 – Stochastic Parameter Alternate PDF Comparison for Unupgraded Configuration

Figure 92 shows a comparison between the unupgraded configuration Release Activity CDFs for the primary set of PDFs used in the cost-benefit analysis, a set of “Good” PDFs that are Triangular Distributions with the apex of the distribution at the expected best-case-scenario value for each stochastic parameter, and a set of "Bad" PDFs that are Triangular Distributions with the apex of the distribution at the expected worst-case-scenario value for each stochastic parameter. The Y-axis of the graph cuts off at 1500 kCi, as the Normal PDFs CDF and the Good PDFs CDF values drop from ~1650 kCi Release Activity to 0 kCi Release Activity as the CDF transitions from representing cases in which the containment ruptures to cases in which the containment does not rupture and for which there is no release.

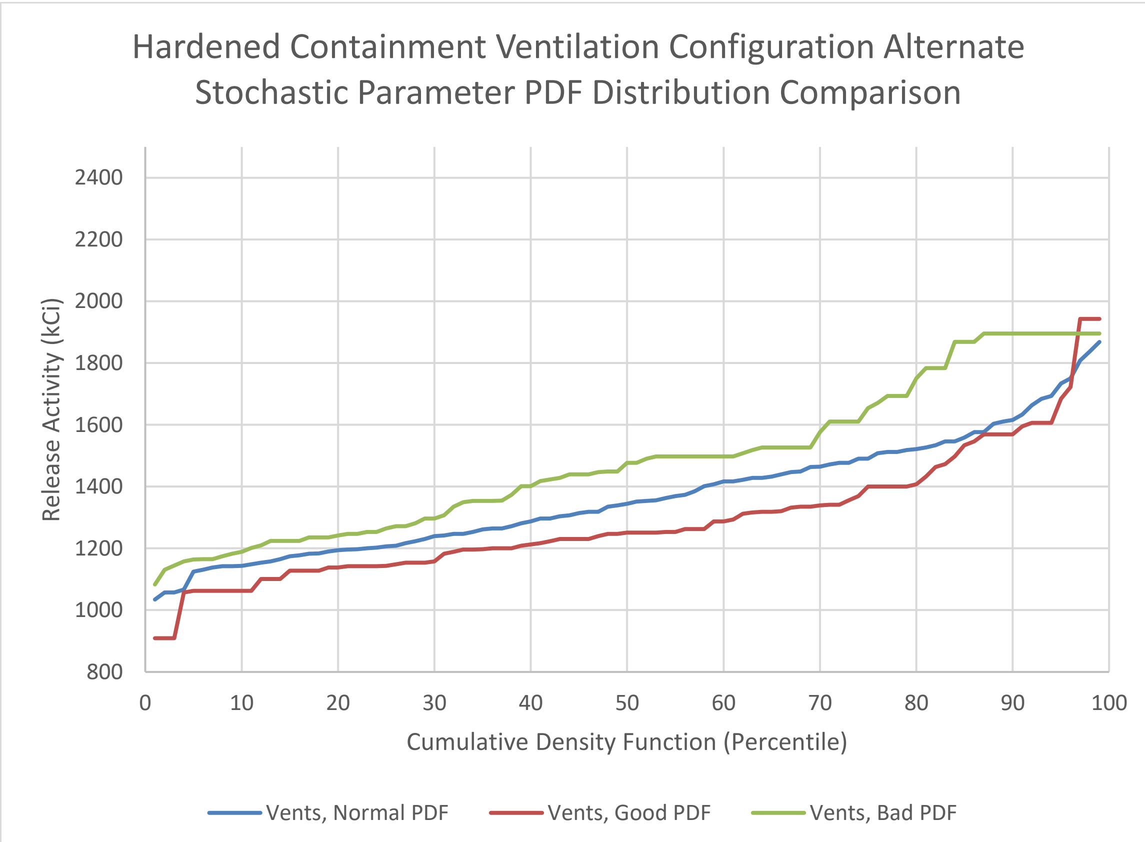


Figure 93 – Stochastic Parameter Alternate PDF Comparison for Hardened Vents Configuration

Figure 93 shows an alternate PDF comparison for the Hardened Containment Ventilation configuration. Interestingly, the Good PDF results showed the highest Release Activity for the 97$^{th}$ through 99$^{th}$ Percentile. A heavily weighted case in which the EDG Failure Time was long, the DC Lifetime was long, the Wetwell was very full and at a reasonable temperature, and the Containment Failure Pressure was very high. One flaw in the assignment of "Good" and "Bad" values to various stochastic parameters that was found here is the possibility, as happened here, of various stochastic parameters sampling to values that are, individually speaking, each expected to be favorable values, but which combine to create a set of circumstances that prove unfavorable. Additionally, due to the nature of the Wetwell heat capacity limit curves, either an overly full or an overly empty Wetwell can, depending on how the scenario unfolds, become unfavorable. Here, the stoichiometrics of hydrogen deflagrations meant that the stochastic parameters sampled created an environment in the Wetwell in which more hydrogen could accumulate before reaching a combustible mixture, generating a larger, more violent hydrogen explosion, propelling aerosolized radioactive materials through the containment vent and into the environment.

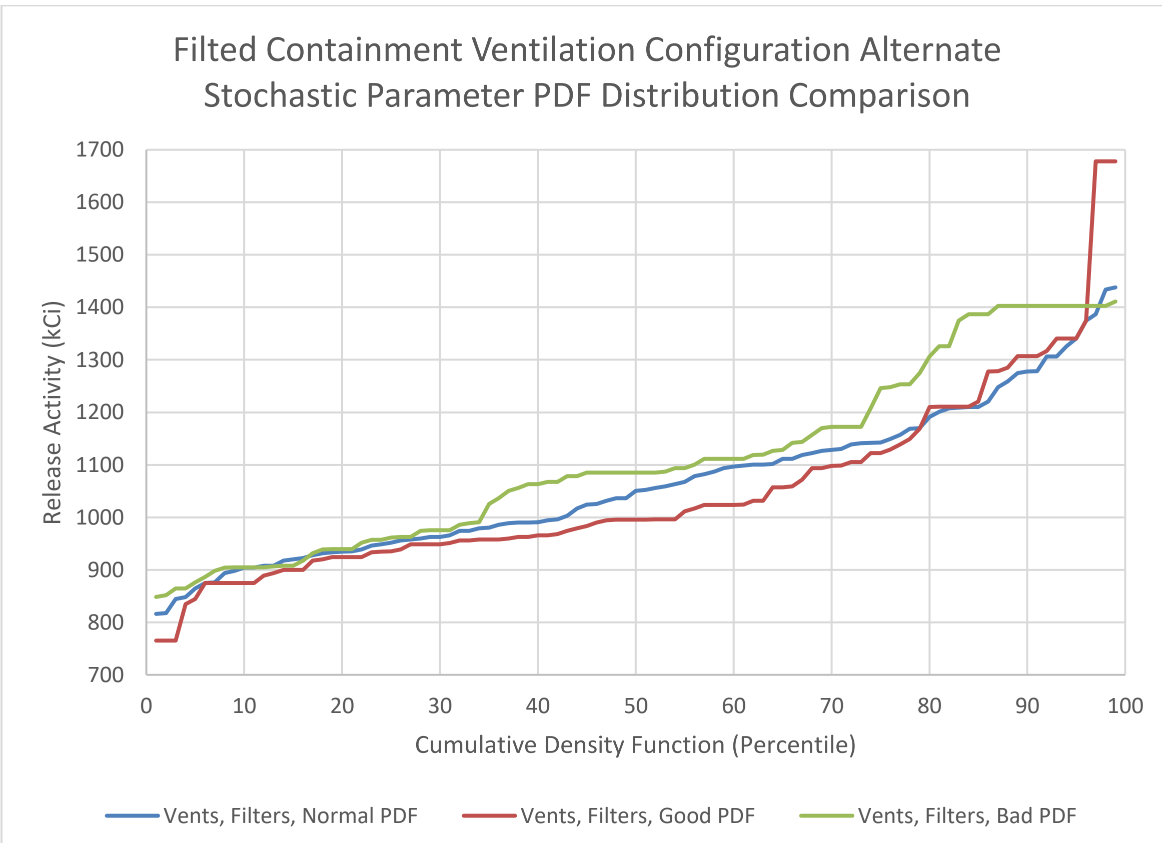


Figure 94 – Stochastic Parameter Alternate PDF Comparison for Hardened Vents Configuration

Figure 94 shows an alternate PDF comparison for the Filtered Containment Ventilation configuration. As with the Hardened Containment Ventilation alternate PDF comparison, a heavily weighted "spoiler" with high release activity causes the expected "Good" PDFs to have the highest Release Activity for the 97th through 99th Percentile. The addition of the Filtered Containment Vents made the Release Activities less sensitive to changes in the PDFs – in configurations with Filtered Containment Ventilation, the significant majority of the Release Activity was from released Xenon. The highest non-Xenon Release Activity sampled was 678 kCi and was the "spoiler" case that gives the "Good" PDFs data set the high 97th-99th Percentile Release Activities, and the lowest Xenon Release Activity sampled was 722 kCi.

Table 38. Mean Difference Analysis of Upgrade Configuration Release Reduction for alternate PDFs

| Test Case | Normal PDF Mean Difference | Normal PDF 95% Confidence Interval Radius | Good PDF Mean Difference | Good PDF 95% Confidence Interval Radius | Bad PDF Mean Difference | Bad PDF 95% Confidence Interval Radius |
|---|---|---|---|---|---|---|
| IRWST | 38.53249 | 104.4232 | 107.9086 | 178.537 | 83.09207 | 42.64947 |
| Igniters | -28.5718 | 106.0493 | 205.0016 | 184.4583 | -89.0814 | 55.87161 |
| Igniters, IRWST | 32.31442 | 104.7455 | 102.4311 | 179.0262 | 78.46058 | 42.82807 |
| PARs | 0.376563 | 109.4209 | 156.11 | 179.7184 | -99.5267 | 55.9089 |
| PARs, IRWST | 47.77434 | 104.6458 | 137.993 | 181.7285 | 109.4197 | 43.0278 |
| PARs, Igniters | -40.2656 | 103.3135 | 86.52522 | 175.8422 | -109.754 | 54.8255 |
| PARs, Igniters, IRWST | 28.63896 | 105.4015 | 122.1872 | 183.1346 | 100.1689 | 43.2442 |
| Vents | 402.0897 | 79.38415 | 138.0693 | 124.5048 | 473.0075 | 54.2936 |
| Vents, IRWST | 396.9816 | 76.07558 | 77.17701 | 122.2419 | 574.9565 | 42.25763 |
| Vents, Igniters | 410.0081 | 80.40128 | 169.8528 | 124.2894 | 479.05 | 54.87842 |
| Vents, Igniters, IRWST | 421.1903 | 75.94451 | 116.892 | 121.3392 | 564.1843 | 42.72507 |
| Vents, PARs | 511.5236 | 78.95097 | 263.9375 | 123.4028 | 538.4242 | 62.17052 |
| Vents, PARs, IRWST | 468.1205 | 76.94787 | 170.8637 | 122.1689 | 641.5487 | 42.43261 |
| Vents, PARs, Igniters | 490.6964 | 79.35905 | 235.5488 | 123.5669 | 528.9635 | 61.89609 |
| Vents, PARs, Igniters, IRWST | 465.3053 | 76.74562 | 147.1505 | 121.729 | 645.9956 | 43.23346 |
| Vents, Filters | 703.6609 | 77.00738 | 378.0967 | 123.5512 | 848.1233 | 47.23796 |
| Vents, Filters, IRWST | 650.6743 | 75.4779 | 294.796 | 122.0573 | 864.054 | 40.85712 |
| Vents, Filters, Igniters | 710.3839 | 77.74764 | 406.8326 | 123.317 | 852.1201 | 47.29121 |

| | | | | | | |
|---|---|---|---|---|---|---|
| Vents, Filters, Igniters, IRWST | 675.2338 | 75.0409 | 344.9657 | 120.967 | 854.1229 | 41.15096 |
| Vents, Filters, PARs | 798.0854 | 76.76182 | 496.0351 | 122.5797 | 900.7714 | 53.03301 |
| Vents, Filters, PARs, IRWST | 717.7267 | 76.19828 | 382.422 | 121.8184 | 934.7593 | 41.23279 |
| Vents, Filters, PARs, Igniters | 781.7588 | 77.1317 | 472.5074 | 122.7386 | 893.3658 | 52.92835 |
| Vents, Filters, PARs, Igniters, IRWST | 714.141 | 76.05214 | 369.5632 | 121.4671 | 922.6418 | 42.39263 |

Table 38 shows a comparison of the results of Welch Treatment T-Interval Mean Difference Analysis for each upgrade configuration when using each of the three sets of stochastic parameter PDFs compared. Interestingly, when the Normal PDFs were used, Igniters did not provide a statistically significant reduction or increase in Release Activity. When the Good PDFs were used, the Igniters provided a statistically significant reduction in Release Activity, and when the Bad PDFs were used, the Igniters caused a statistically significant increase in Release Activity. The mechanism by which the reduction in Release Activity for Igniters occurred with the Good PDFs was that in many cases, the inclusion of Igniters prevented the containment from failing in the non-ventilated Igniters configuration, in cases that were significantly more heavily weighted in the Good PDFs than the Normal PDFs. In similar vein, the PARs transitioned from providing no noticeable change in Release Activity with the Normal PDFs to nearing a statistically significant reduction in Release Activity for the Good PDFs and to causing a statistically significant increase in Release Activity for Bad PDFs. For both the Igniters and the PARs, the significant worsening of the upgrade from the use of the Bad PDFs was the result of more heavily weighting scenarios in which circumstances aligned such that hydrogen was mitigated enough to prevent early deflagrations that would rupture and depressurize containment while it was at a low pressure, which caused the containment pressure to build significantly higher before the containment ultimately failed, leading to a more violent depressurization with a greater release.

The Hardened and Filtered Containment Ventilation systems saw their efficacy cut drastically when the Good PDFs were used, as it increased the percent of cases in which the containment was not expected to fail from 6% with the Normal PDFs to 22% with the Good PDFs. For the Igniters Good PDF case, the containment was expected to withstand the accident 34% of the time. The IRWST had no statistically significant impact on the Release Activity for the Normal PDFs or Good PDFs but provided a small but statistically significant reduction in Release Activity for the Bad PDFs.

## 5.6. *Power Plant Upgrade Cost-Benefit Analysis*

### 5.6.1. Individual Upgrade Costs

Table 39. Individual Upgrade Costs

| Upgrade | Cost (Low) | Cost (High) | Cost (Mid) |
|---|---|---|---|
| PARs | 7.50E+05 | 1.50E+06 | 1.13E+06 |
| Manual ADS | 5.00E+04 | 1.46E+04 | 1.50E+05 |
| IRWST | 8.60E+06 | 4.30E+07 | 2.58E+07 |
| Hardened Vents | 1.00E+06 | 2.50E+06 | 1.75E+06 |
| Filtered Vents | 1.50E+06 | 2.00E+07 | 1.08E+07 |
| Igniters | 1.00E+05 | 2.05E+05 | 1.53E+05 |
| Igniter Backup Power | 1.47E+05 | 1.00E+06 | 5.74E+05 |

### 5.6.2. Cost-Benefit Upgrade Analysis for EDG Sensitivity Study Results

This section discusses preliminary cost-benefit analysis results from Sections 5.2 and 5.3, in which all stochastic parameters but the EDG Failure Time were assigned conservative values and only the EDG Failure Time was varied, for an initial examination of some of the impacts of a variety of upgrade configurations.

Table 40. Upgrade Configurations Costs Tabulated for EDG Sensitivity Study Results

| Configuration | Low Cost | High Cost |
|---|---|---|
| Vents, IRWST, ADS | 9.65E+06 | 3.37E+07 |
| PAR, Aerosol | 2.25E+06 | 2.15E+07 |
| Aerosol Filter | 1.50E+06 | 2.00E+07 |
| Igniter, Aerosol | 1.60E+06 | 2.02E+07 |

| | | |
|---|---|---|
| PAR, Vents | 1.75E+06 | 2.65E+07 |
| ATF, ADS, IRWST | -3.31E+08 | 3.50E+08 |
| IRWST, ADS | 8.65E+06 | 8.65E+06 |
| Hardened Vents | 1.00E+06 | 2.50E+07 |
| Igniter, Vent | 1.10E+06 | 2.52E+07 |
| Vent, IRWST | 9.60E+06 | 3.36E+07 |
| ADS, PAR, Vent | 1.80E+06 | 2.66E+07 |
| ATF | -3.40E+08 | 3.40E+08 |
| IRWST, PAR | 9.35E+06 | 1.01E+07 |
| IRWST | 8.60E+06 | 8.60E+06 |
| ADS, PAR | 8.00E+05 | 1.55E+06 |
| ADS | 5.00E+04 | 5.00E+04 |
| PAR | 7.50E+05 | 1.50E+06 |
| Igniter | 1.00E+05 | 2.05E+05 |
| ATF, ADS, IRWST, Vents | -3.30E+08 | 3.74E+08 |

Table 40 is a listing of the high end and low end cost estimates for each upgrade configuration examined in Sections 5.2 and 5.3. For the purposes of examining the Accident Tolerant Fuels upgrade, a use lifetime of 20 years was used to convert the potential costs from costs per year to a lifetime cost of the upgrade.

Table 41. Power Plant Upgrade Configuration EDG Sensitivity Study Benefits Tabulated

| Configuration | Mean Release Activity (kCi) | Sigma Release Activity | Change in Release Activity (kCi) | Release Activity Change Sigma | R.A.C. 5th Percentile | R.A.C. 95th Percentile |
|---|---|---|---|---|---|---|
| Vents, IRWST, ADS | 1517.87 | 25.13 | 1574.79 | 88.24 | 1310.08 | 1839.50 |
| PAR, Aerosol | 1569.56 | 99.39 | 1523.10 | 130.51 | 1131.57 | 1914.62 |
| Aerosol Filter | 1713.12 | 123.33 | 1379.54 | 149.55 | 930.90 | 1828.18 |
| Igniter, Aerosol | 1713.12 | 123.33 | 1379.54 | 149.55 | 930.90 | 1828.18 |
| PAR, Vents | 2218.57 | 124.11 | 874.10 | 150.19 | 423.53 | 1324.65 |
| ATF, ADS, IRWST | 2326.00 | 23.00 | 766.66 | 87.65 | 503.70 | 1029.62 |
| IRWST, ADS | 2328.11 | 90.78 | 764.54 | 124.08 | 392.31 | 1136.78 |
| Hardened Vents | 2360.36 | 125.85 | 732.30 | 151.63 | 277.41 | 1187.19 |
| Igniter, Vent | 2360.36 | 125.85 | 732.30 | 151.63 | 277.41 | 1187.19 |
| Vent, IRWST | 2411.04 | 325.39 | 681.62 | 336.21 | -327.00 | 1690.24 |
| ADS, PAR, Vent | 2435.86 | 632.80 | 656.80 | 638.43 | -1258.48 | 2572.08 |
| ATF | 2695.32 | 101.37 | 397.34 | 132.02 | 1.270 | 793.41 |

| | | | | | | |
|---|---|---|---|---|---|---|
| IRWST, PAR | 2849.33 | 172.24 | 243.33 | 191.89 | -332.34 | 818.99 |
| IRWST | 2886.76 | 52.45 | 205.89 | 99.53 | -92.68 | 504.47 |
| ADS, PAR | 3000.66 | 49.19 | 92.00 | 97.84 | -201.54 | 385.53 |
| ADS | 3002.99 | 40.68 | 89.67 | 93.86 | -191.91 | 371.24 |
| PAR | 3052.41 | 167.45 | 40.25 | 187.60 | -522.56 | 603.05 |
| Igniter | 3092.66 | 84.58 | 0.00 | 119.62 | -358.85 | 358.85 |
| ATF, ADS, IRWST, Vents | 3698.30 | 304.31 | -605.643 | 315.84 | -1553.18 | 341.89 |

Table 41 shows a tabulation of the Mean Release Activity and sigma for each upgrade configuration, as well as the change in Release Activity, compared to the No Upgrades configuration, for each upgrade configuration. The Release Activity Change Sigma was calculated by adding, in quadrature, the uncertainty in the Mean Release Activity for each upgrade configuration to the uncertainty in the Mean Release Activity for the No Upgrades configuration. Finally, the Release Activity Change (R.A.C.) 5$^{th}$ and 95$^{th}$ percentiles were calculated by subtracting or adding, respectively, three times sigma from or to the Change in Release Activity expected value.

Table 42. Power Plant Upgrade EDG Sensitivity Study Cost-Benefit Ratio, Low Cost

| Configuration | Low Cost | Avg kCi/$ | 5th percentile kCi/$ | 95th percentile kCi/$ |
|---|---|---|---|---|
| Aerosol Filter | 1.50E+06 | 9.20E-04 | 6.21E-04 | 1.22E-03 |
| Igniter, Aerosol | 1.60E+06 | 8.62E-04 | 5.82E-04 | 1.14E-03 |
| PAR, Aerosol | 2.25E+06 | 6.77E-04 | 5.03E-04 | 8.51E-04 |
| Hardened Vents | 1.00E+06 | 7.32E-04 | 2.77E-04 | 1.19E-03 |
| Igniter, Vent | 1.10E+06 | 6.66E-04 | 2.52E-04 | 1.08E-03 |
| PAR, Vents | 1.75E+06 | 4.99E-04 | 2.42E-04 | 7.57E-04 |
| Vents, IRWST, ADS | 9.65E+06 | 1.63E-04 | 1.36E-04 | 1.91E-04 |
| IRWST, ADS | 8.65E+06 | 8.84E-05 | 4.54E-05 | 1.31E-04 |
| ATF, ADS, IRWST, Vents | -3.30E+08 | 1.83E-06 | 4.70E-06 | -1.03E-06 |
| ATF | -3.40E+08 | -1.17E-06 | -3.74E-09 | -2.33E-06 |
| ATF, ADS, IRWST | -3.31E+08 | -2.32E-06 | -1.52E-06 | -3.11E-06 |
| IRWST | 8.60E+06 | 2.39E-05 | -1.08E-05 | 5.87E-05 |
| Vent, IRWST | 9.60E+06 | 7.10E-05 | -3.41E-05 | 1.76E-04 |
| IRWST, PAR | 9.35E+06 | 2.60E-05 | -3.55E-05 | 8.76E-05 |
| ADS, PAR | 8.00E+05 | 1.15E-04 | -2.52E-04 | 4.82E-04 |

| | | | | |
|---|---|---|---|---|
| PAR | 7.50E+05 | 5.37E-05 | -6.97E-04 | 8.04E-04 |
| ADS, PAR, Vent | 1.80E+06 | 3.65E-04 | -6.99E-04 | 1.43E-03 |
| Igniter | 1.00E+05 | 0.00E+00 | -3.59E-03 | 3.59E-03 |
| ADS | 5.00E+04 | 1.79E-03 | -3.84E-03 | 7.42E-03 |

Table 43. Power Plant Upgrade EDG Sensitivity Study Cost-Benefit Ratio, High Cost

| Configuration | High Cost | Avg kCi/$ | 5th percentile kCi/$ | 95th percentile kCi/$ |
|---|---|---|---|---|
| PAR, Aerosol | 2.15E+07 | 7.08E-05 | 5.26E-05 | 8.91E-05 |
| Aerosol Filter | 2.00E+07 | 6.90E-05 | 4.65E-05 | 9.14E-05 |
| Igniter, Aerosol | 2.02E+07 | 6.83E-05 | 4.61E-05 | 9.05E-05 |
| IRWST, ADS | 8.65E+06 | 8.84E-05 | 4.54E-05 | 1.31E-04 |
| Vents, IRWST, ADS | 3.37E+07 | 4.68E-05 | 3.89E-05 | 5.47E-05 |
| PAR, Vents | 2.65E+07 | 3.30E-05 | 1.60E-05 | 5.00E-05 |
| Hardened Vents | 2.50E+07 | 2.93E-05 | 1.11E-05 | 4.75E-05 |
| Igniter, Vent | 2.52E+07 | 2.91E-05 | 1.10E-05 | 4.71E-05 |
| ATF, ADS, IRWST | 3.50E+08 | 2.19E-06 | 1.44E-06 | 2.94E-06 |
| ATF | 3.40E+08 | 1.17E-06 | 3.74E-09 | 2.33E-06 |
| ATF, ADS, IRWST, Vents | 3.74E+08 | -1.62E-06 | -4.16E-06 | 9.15E-07 |
| Vent, IRWST | 3.36E+07 | 2.03E-05 | -9.73E-06 | 5.03E-05 |
| IRWST | 8.60E+06 | 2.39E-05 | -1.08E-05 | 5.87E-05 |
| IRWST, PAR | 1.01E+07 | 2.41E-05 | -3.29E-05 | 8.11E-05 |
| ADS, PAR, Vent | 2.66E+07 | 2.47E-05 | -4.74E-05 | 9.69E-05 |
| ADS, PAR | 1.55E+06 | 5.94E-05 | -1.30E-04 | 2.49E-04 |
| PAR | 1.50E+06 | 2.68E-05 | -3.48E-04 | 4.02E-04 |
| Igniter | 2.05E+05 | 0.00E+00 | -1.75E-03 | 1.75E-03 |
| ADS | 5.00E+04 | 1.79E-03 | -3.84E-03 | 7.42E-03 |

Table 42 and

Table *43* show the final results of the Upgrade EDG Sensitivity Cost-Benefit analysis. Both tables are sorted, from largest to smallest, by the 5th percentile estimate of kCi of release prevented per dollar spent on the upgrade. Sorting the table this way allows us to conservatively estimate the most economically effective upgrades, though the data can readily be sorted along any of the columns.

### 5.6.3. Refined Multi-Parameter Accident Scenario Upgrade Configuration Cost-Benefit Analysis

Table 44. Refined Multi-Parameter Analysis Upgrade Configuration Costs

| Test Case | Filters | Vents | PARs | Igniters | IRWST | Cost (High) | Cost (Mid) | Cost (Low) |
|---|---|---|---|---|---|---|---|---|
| 1 | No | No | No | No | No | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| 2 | No | No | No | No | Yes | 8.65E+07 | 4.74E+07 | 8.65E+06 |
| 3 | No | No | No | Yes | No | 1.21E+06 | 7.03E+05 | 2.00E+05 |
| 4 | No | No | No | Yes | Yes | 8.77E+07 | 4.81E+07 | 8.85E+06 |
| 5 | No | No | Yes | No | No | 1.50E+06 | 1.13E+06 | 7.50E+05 |
| 6 | No | No | Yes | No | Yes | 8.80E+07 | 4.86E+07 | 9.40E+06 |
| 7 | No | No | Yes | Yes | No | 2.71E+06 | 1.83E+06 | 9.50E+05 |
| 8 | No | No | Yes | Yes | Yes | 8.92E+07 | 4.93E+07 | 9.60E+06 |
| 9 | No | Yes | No | No | No | 2.50E+07 | 1.30E+07 | 1.00E+06 |
| 10 | No | Yes | No | No | Yes | 1.12E+08 | 6.04E+07 | 9.65E+06 |
| 11 | No | Yes | No | Yes | No | 2.62E+07 | 1.37E+07 | 1.20E+06 |
| 12 | No | Yes | No | Yes | Yes | 1.13E+08 | 6.11E+07 | 9.85E+06 |
| 13 | No | Yes | Yes | No | No | 2.65E+07 | 1.41E+07 | 1.75E+06 |
| 14 | No | Yes | Yes | No | Yes | 1.13E+08 | 6.16E+07 | 1.04E+07 |
| 15 | No | Yes | Yes | Yes | No | 2.77E+07 | 1.48E+07 | 1.95E+06 |
| 16 | No | Yes | Yes | Yes | Yes | 1.14E+08 | 6.23E+07 | 1.06E+07 |
| 17 | Yes | Yes | No | No | No | 2.00E+07 | 1.08E+07 | 1.50E+06 |
| 18 | Yes | Yes | No | No | Yes | 1.07E+08 | 5.82E+07 | 1.02E+07 |
| 19 | Yes | Yes | No | Yes | No | 2.12E+07 | 1.15E+07 | 1.70E+06 |
| 20 | Yes | Yes | No | Yes | Yes | 1.08E+08 | 5.89E+07 | 1.04E+07 |
| 21 | Yes | Yes | Yes | No | No | 2.15E+07 | 1.19E+07 | 2.25E+06 |
| 22 | Yes | Yes | Yes | No | Yes | 1.08E+08 | 5.93E+07 | 1.09E+07 |
| 23 | Yes | Yes | Yes | Yes | No | 2.27E+07 | 1.26E+07 | 2.45E+06 |
| 24 | Yes | Yes | Yes | Yes | Yes | 1.09E+08 | 6.00E+07 | 1.11E+07 |

Table 44 shows the tabulated costs of every upgrade configuration examined in the Refined Multi-Parameter Upgrade Configuration Accident Scenario Analysis discussed in Section 5.5. For determining upgrade configuration costs for this analysis, the cost of implementing the IRWST included the cost of adding the Manual ADS upgrade and adding the cost of the Hydrogen Igniters included the cost of adding the Hydrogen Igniters Backup Power upgrade, as these upgrades were paired within the analysis for the sake of computational expediency. A simple cost-benefit ratio, in units of \$/kCi, can be obtained using the results of Table 31, the Means Difference Analysis between each upgrade configuration and the base case, and the costs tabulated in Table 44.

Table 45 – Refined Multi-Parameter Analysis General 100% Power Cost-Benefit Results

| Test Case | Filters | Vents | PARs | Igniters | IRWST | Cost (High) | Cost (Mid) | Cost (Low) | Mean Release Activity Change | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 7 | No | No | Yes | Yes | No | 2.71E+06 | 1.83E+06 | 9.50E+05 | -40.2656 | -6.72E+04 | -4.54E+04 | -2.36E+04 |
| 3 | No | No | No | Yes | No | 1.21E+06 | 7.03E+05 | 2.00E+05 | -28.5718 | -4.22E+04 | -2.46E+04 | -7.00E+03 |
| 21 | Yes | Yes | Yes | No | No | 2.15E+07 | 1.19E+07 | 2.25E+06 | 798.0854 | 2.69E+04 | 1.49E+04 | 2.82E+03 |
| 17 | Yes | Yes | No | No | No | 2.00E+07 | 1.08E+07 | 1.50E+06 | 703.6609 | 2.84E+04 | 1.53E+04 | 2.13E+03 |
| 23 | Yes | Yes | Yes | Yes | No | 2.27E+07 | 1.26E+07 | 2.45E+06 | 781.7588 | 2.90E+04 | 1.61E+04 | 3.13E+03 |
| 19 | Yes | Yes | No | Yes | No | 2.12E+07 | 1.15E+07 | 1.70E+06 | 710.3839 | 2.99E+04 | 1.61E+04 | 2.39E+03 |
| 13 | No | Yes | Yes | No | No | 2.65E+07 | 1.41E+07 | 1.75E+06 | 511.5236 | 5.18E+04 | 2.76E+04 | 3.42E+03 |
| 15 | No | Yes | Yes | Yes | No | 2.77E+07 | 1.48E+07 | 1.95E+06 | 490.6964 | 5.65E+04 | 3.02E+04 | 3.97E+03 |
| 9 | No | Yes | No | No | No | 2.50E+07 | 1.30E+07 | 1.00E+06 | 402.0897 | 6.22E+04 | 3.23E+04 | 2.49E+03 |
| 11 | No | Yes | No | Yes | No | 2.62E+07 | 1.37E+07 | 1.20E+06 | 410.0081 | 6.39E+04 | 3.34E+04 | 2.93E+03 |
| 22 | Yes | Yes | Yes | No | Yes | 1.08E+08 | 5.93E+07 | 1.09E+07 | 717.7267 | 1.50E+05 | 8.27E+04 | 1.52E+04 |
| 24 | Yes | Yes | Yes | Yes | Yes | 1.09E+08 | 6.00E+07 | 1.11E+07 | 714.141 | 1.53E+05 | 8.40E+04 | 1.55E+04 |
| 20 | Yes | Yes | No | Yes | Yes | 1.08E+08 | 5.89E+07 | 1.04E+07 | 675.2338 | 1.60E+05 | 8.72E+04 | 1.53E+04 |
| 18 | Yes | Yes | No | No | Yes | 1.07E+08 | 5.82E+07 | 1.02E+07 | 650.6743 | 1.64E+05 | 8.94E+04 | 1.56E+04 |
| 14 | No | Yes | Yes | No | Yes | 1.13E+08 | 6.16E+07 | 1.04E+07 | 468.1205 | 2.41E+05 | 1.32E+05 | 2.22E+04 |
| 16 | No | Yes | Yes | Yes | Yes | 1.14E+08 | 6.23E+07 | 1.06E+07 | 465.3053 | 2.45E+05 | 1.34E+05 | 2.28E+04 |
| 12 | No | Yes | No | Yes | Yes | 1.13E+08 | 6.11E+07 | 9.85E+06 | 421.1903 | 2.68E+05 | 1.45E+05 | 2.34E+04 |
| 10 | No | Yes | No | No | Yes | 1.12E+08 | 6.04E+07 | 9.65E+06 | 396.9816 | 2.81E+05 | 1.52E+05 | 2.43E+04 |
| 6 | No | No | Yes | No | Yes | 8.80E+07 | 4.86E+07 | 9.40E+06 | 47.77434 | 1.84E+06 | 1.02E+06 | 1.97E+05 |
| 2 | No | No | No | No | Yes | 8.65E+07 | 4.74E+07 | 8.65E+06 | 38.53249 | 2.24E+06 | 1.23E+06 | 2.24E+05 |
| 4 | No | No | No | Yes | Yes | 8.77E+07 | 4.81E+07 | 8.85E+06 | 32.31442 | 2.71E+06 | 1.49E+06 | 2.74E+05 |
| 8 | No | No | Yes | Yes | Yes | 8.92E+07 | 4.93E+07 | 9.60E+06 | 28.63896 | 3.11E+06 | 1.72E+06 | 3.35E+05 |
| 5 | No | No | Yes | No | No | 1.50E+06 | 1.13E+06 | 7.50E+05 | 0.376563 | 3.98E+06 | 2.99E+06 | 1.99E+06 |

Table 45 presents a general overview analysis of the cost-benefit ratio of every upgrade configuration examined, sorted by Cost per Mean kCi prevented, Mid Cost, from least to greatest. Care must be taken when a negative cost-benefit ratio is produced, as it can mean either that the cost is negative, or the benefit is negative – in other words, a negative cost-benefit ratio is either indicative of a definite "should" or a definite "should not" with regards to implementation. In the case of Test-7 and Test-3 in the Refined Multi-Parameter analysis, the results are indicative of a definite "should not." Ignoring these results, Tests 21, 17, 23, and 19 are the clear winners, being by far the most cost-effective configurations tested. These configurations all include Filtered Containment Ventilation and differ in their inclusion or exclusion of PARs and Hydrogen Igniters with Backup Power.

As well as comparing the upgrade configurations as full units, the wealth of data available allows each configuration containing an upgrade to be compared to the corresponding upgrade that lacks the upgrade, to examine each upgrade in extreme detail with regards to potential synergies with other upgrades.

Table 46. IRWST Individual Upgrade Multi-Parameter Cost-Benefit Analysis, 100% Reactor Power

| Test Case | Filters | Vents | PARs | Igniters | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|
| 10 | No | Yes | No | No | -5.11 | -1.69E+07 | -9.23E+06 | -1.69E+06 |
| 16 | No | Yes | Yes | Yes | -25.39 | -3.41E+06 | -1.86E+06 | -3.41E+05 |
| 20 | Yes | Yes | No | Yes | -35.15 | -2.46E+06 | -1.34E+06 | -2.46E+05 |
| 14 | No | Yes | Yes | No | -43.40 | -1.99E+06 | -1.09E+06 | -1.99E+05 |
| 18 | Yes | Yes | No | No | -52.99 | -1.63E+06 | -8.90E+05 | -1.63E+05 |
| 24 | Yes | Yes | Yes | Yes | -67.62 | -1.28E+06 | -6.97E+05 | -1.28E+05 |
| 22 | Yes | Yes | Yes | No | -80.36 | -1.08E+06 | -5.87E+05 | -1.08E+05 |
| 8 | No | No | Yes | Yes | 68.90 | 1.26E+06 | 6.84E+05 | 1.26E+05 |
| 4 | No | No | No | Yes | 60.89 | 1.42E+06 | 7.74E+05 | 1.42E+05 |
| 6 | No | No | Yes | No | 47.40 | 1.82E+06 | 9.95E+05 | 1.82E+05 |
| 2 | No | No | No | No | 38.53 | 2.24E+06 | 1.22E+06 | 2.24E+05 |
| 12 | No | Yes | No | Yes | 11.18 | 7.74E+06 | 4.22E+06 | 7.74E+05 |

Table 46 shows a cost-benefit analysis of the Manual ADS and IRWST upgrade for the Refined Multi-Parameter Analysis at 100% Reactor Power. Each upgrade configuration containing the IRWST upgrade is compared to the corresponding upgrade configuration that does not include the IRWST upgrade. Even under the best of circumstances, when the IRWST was implemented alongside PARs and Hydrogen Igniters with Backup Power and without Hardened or Filtered Containment Vents, the IRWST was highly cost-ineffective, costing hundreds of times more on a $/kCi basis than more cost-effective upgrades.

Table 47. Hydrogen Igniter with Backup Power Individual Upgrade Multi-Parameter Cost-Benefit Analysis, 100% Reactor Power

| Test Case | Filters | Vents | PARs | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|
| 16 | No | Yes | Yes | Yes | -2.82 | -4.28E+05 | -2.50E+05 | -7.10E+04 |
| 24 | Yes | Yes | Yes | Yes | -3.59 | -3.36E+05 | -1.96E+05 | -5.58E+04 |
| 4 | No | No | No | Yes | -6.22 | -1.94E+05 | -1.13E+05 | -3.22E+04 |
| 23 | Yes | Yes | Yes | No | -16.33 | -7.38E+04 | -4.30E+04 | -1.22E+04 |
| 8 | No | No | Yes | Yes | -19.14 | -6.30E+04 | -3.67E+04 | -1.05E+04 |
| 15 | No | Yes | Yes | No | -20.83 | -5.79E+04 | -3.37E+04 | -9.60E+03 |
| 3 | No | No | No | No | -28.57 | -4.22E+04 | -2.46E+04 | -7.00E+03 |
| 7 | No | No | Yes | No | -40.64 | -2.96E+04 | -1.73E+04 | -4.92E+03 |
| 20 | Yes | Yes | No | Yes | 24.56 | 4.91E+04 | 2.86E+04 | 8.14E+03 |
| 12 | No | Yes | No | Yes | 24.21 | 4.98E+04 | 2.90E+04 | 8.26E+03 |
| 11 | No | Yes | No | No | 7.92 | 1.52E+05 | 8.87E+04 | 2.53E+04 |
| 19 | Yes | Yes | No | No | 6.72 | 1.79E+05 | 1.04E+05 | 2.97E+04 |

Table 47 shows a cost-benefit analysis of the Hydrogen Igniter with Backup Power upgrade for the Refined Multi-Parameter Analysis at 100% Reactor Power. The Igniters were, in general, a relatively ineffective upgrade – only in Test Case 20 was the reduction in Release Activity provided by the Igniters actually statistically significant. The relatively good cost-benefit ratio of the Igniters is entirely a function of their low cost.

Table 48. Hardened Containment Ventilation Individual Upgrade Multi-Parameter Cost-Benefit Analysis, 100% Reactor Power

| Test Case | PARs | Igniters | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|
| 15 | Yes | Yes | No | 530.96 | 4.71E+04 | 2.45E+04 | 1.88E+03 |
| 13 | Yes | No | No | 511.15 | 4.89E+04 | 2.54E+04 | 1.96E+03 |
| 11 | No | Yes | No | 438.58 | 5.70E+04 | 2.96E+04 | 2.28E+03 |
| 16 | Yes | Yes | Yes | 436.67 | 5.73E+04 | 2.98E+04 | 2.29E+03 |
| 14 | Yes | No | Yes | 420.35 | 5.95E+04 | 3.09E+04 | 2.38E+03 |
| 9 | No | No | No | 402.09 | 6.22E+04 | 3.23E+04 | 2.49E+03 |
| 12 | No | Yes | Yes | 388.88 | 6.43E+04 | 3.34E+04 | 2.57E+03 |
| 10 | No | No | Yes | 358.45 | 6.97E+04 | 3.63E+04 | 2.79E+03 |

Table 48 shows a cost-benefit analysis of the Hardened Containment Ventilation upgrade for the Refined Multi-Parameter Analysis at 100% Reactor Power. The cost efficacy of the upgrade varies dramatically between the high cost-estimate and the low cost-estimate due to the wide variance in the cost-estimates for such upgrades found in the available literature. That said, the Hardened Containment Ventilation upgrade was, in general, one of the most cost-effective upgrades and would make for a reasonable, lower cost alternate to the Filtered Containment Ventilation upgrade, despite its superior cost efficacy in $/kCi of release prevented.

Table 49. Containment Ventilation Filter Individual Upgrade Multi-Parameter Cost-Benefit Analysis, 100% Reactor Power

| Test Case | PARs | Igniters | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|
| 17 | No | No | No | 301.57 | -1.66E+04 | -7.46E+03 | 1.66E+03 |
| 19 | No | Yes | No | 300.38 | -1.66E+04 | -7.49E+03 | 1.66E+03 |
| 23 | Yes | Yes | No | 291.06 | -1.72E+04 | -7.73E+03 | 1.72E+03 |
| 21 | Yes | No | No | 286.56 | -1.74E+04 | -7.85E+03 | 1.74E+03 |
| 20 | No | Yes | Yes | 254.04 | -1.97E+04 | -8.86E+03 | 1.97E+03 |
| 18 | No | No | Yes | 253.69 | -1.97E+04 | -8.87E+03 | 1.97E+03 |
| 22 | Yes | No | Yes | 249.61 | -2.00E+04 | -9.01E+03 | 2.00E+03 |
| 24 | Yes | Yes | Yes | 248.84 | -2.01E+04 | -9.04E+03 | 2.01E+03 |

Table 49 shows a cost-benefit analysis of the Filtered Containment Ventilation upgrade for the Refined Multi-Parameter Analysis at 100% Reactor Power. Unlike the other cost-benefit analyses, here the cost was determined by subtracting the cost-estimate for the Hardened Containment Ventilation from the cost-estimate for the Filtered Containment Ventilation, as the Mean Release Difference calculated for each upgrade configuration was the difference in release between each configuration with the Filtered Containment Ventilation upgrade and the corresponding configuration with only the Hardened Containment Ventilation upgrade. Due to a quirk of the available cost-estimates, for high and mid cost-estimates, the Filtered Containment Ventilation upgrade was cheaper than the Hardened Containment Ventilation upgrade. Unlike previous negative cost-benefit ratios that indicated a definite "should not" implement for an upgrade, the negative cost-benefit ratio here is a definite "should" implement. That said, the idea that the Filtered Containment Ventilation upgrade is cheaper than the Hardened Containment Ventilation upgrade is, in a word, unlikely. With that in mind, the cost per kCi prevented using the low cost-estimate, for which the Hardened Containment Ventilation upgrade is cheaper than the Filtered Containment Ventilation upgrade, is the metric by which the table is sorted. It can be seen, comparing Table 49 to Table 45, that the Filtered Containment Ventilation upgrade is very cost-competitive and appears to be the most cost-competitive upgrade tested – using the low cost-estimates, the Filtered Containment Ventilation without PARs upgrade configuration wins out over the PARs and Filtered Containment Ventilation upgrade configuration for cost-efficacy.

Table 50. PAR Individual Upgrade Multi-Parameter Cost-Benefit Analysis, 100% Reactor Power

| Test Case | Filters | Vents | Igniters | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|
| 8 | No | No | Yes | Yes | -3.68 | -4.08E+05 | -3.06E+05 | -2.04E+05 |
| 7 | No | No | Yes | No | -11.69 | -1.28E+05 | -9.62E+04 | -6.41E+04 |
| 13 | No | Yes | No | No | 109.43 | 1.37E+04 | 1.03E+04 | 6.85E+03 |
| 21 | Yes | Yes | No | No | 94.42 | 1.59E+04 | 1.19E+04 | 7.94E+03 |
| 15 | No | Yes | Yes | No | 80.69 | 1.86E+04 | 1.39E+04 | 9.30E+03 |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 23 | Yes | Yes | Yes | No | 71.37 | 2.10E+04 | 1.58E+04 | 1.05E+04 |
| 14 | No | Yes | No | Yes | 71.14 | 2.11E+04 | 1.58E+04 | 1.05E+04 |
| 22 | Yes | Yes | No | Yes | 67.05 | 2.24E+04 | 1.68E+04 | 1.12E+04 |
| 16 | No | Yes | Yes | Yes | 44.12 | 3.40E+04 | 2.55E+04 | 1.70E+04 |
| 24 | Yes | Yes | Yes | Yes | 38.91 | 3.86E+04 | 2.89E+04 | 1.93E+04 |
| 6 | No | No | No | Yes | 9.24 | 1.62E+05 | 1.22E+05 | 8.12E+04 |
| 5 | No | No | No | No | 0.38 | 3.98E+06 | 2.99E+06 | 1.99E+06 |

Table 50 shows a cost-benefit analysis of the PAR upgrade for the Refined Multi-Parameter Analysis at 100% Reactor Power. While the upgrade is almost wholly ineffective by itself, to the point that it provided no statistically significant reduction in Release Activity, it becomes fairly cost-effective when implemented alongside the Hardened Containment Ventilation upgrade and becomes a competitive choice for a plant owner wishing, or required, to reduce the risk of radionuclide release from an LT-SBO. The PAR remains relatively cost-effective when implemented with the Filtered Containment Ventilation upgrade, but loses some of its competitiveness, as the Filtered Containment Vents fared better without the PARs than the Hardened Containment Vents.

### 5.6.4. Alternate Reactor Power Upgrade Configuration Cost-Benefit Analysis

Table 51. 120% Reactor Power Multi-Parameter General Upgrade Cost-Benefit Analysis Results

| Test Case | Filters | Vents | PARs | Igniters | IRWST | Cost (High) | Cost (Mid) | Cost (Low) | Mean Release Activity Change | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 7 | No | No | Yes | Yes | No | 2.71E+06 | 1.83E+06 | 9.50E+05 | -37.16 | -7.28E+04 | -4.92E+04 | -2.56E+04 |
| 5 | No | No | Yes | No | No | 1.50E+06 | 1.13E+06 | 7.50E+05 | -24.15 | -6.21E+04 | -4.66E+04 | -3.11E+04 |
| 3 | No | No | No | Yes | No | 1.21E+06 | 7.03E+05 | 2.00E+05 | -16.13 | -7.47E+04 | -4.35E+04 | -1.24E+04 |
| 17 | Yes | Yes | No | No | No | 2.00E+07 | 1.08E+07 | 1.50E+06 | 1065.74 | 1.88E+04 | 1.01E+04 | 1.41E+03 |
| 21 | Yes | Yes | Yes | No | No | 2.15E+07 | 1.19E+07 | 2.25E+06 | 1153.13 | 1.86E+04 | 1.03E+04 | 1.95E+03 |
| 19 | Yes | Yes | No | Yes | No | 2.12E+07 | 1.15E+07 | 1.70E+06 | 1081.18 | 1.96E+04 | 1.06E+04 | 1.57E+03 |
| 23 | Yes | Yes | Yes | Yes | No | 2.27E+07 | 1.26E+07 | 2.45E+06 | 1149.00 | 1.98E+04 | 1.09E+04 | 2.13E+03 |
| 13 | No | Yes | Yes | No | No | 2.65E+07 | 1.41E+07 | 1.75E+06 | 810.55 | 3.27E+04 | 1.74E+04 | 2.16E+03 |
| 9 | No | Yes | No | No | No | 2.50E+07 | 1.30E+07 | 1.00E+06 | 711.44 | 3.51E+04 | 1.83E+04 | 1.41E+03 |
| 15 | No | Yes | Yes | Yes | No | 2.77E+07 | 1.48E+07 | 1.95E+06 | 805.23 | 3.44E+04 | 1.84E+04 | 2.42E+03 |
| 11 | No | Yes | No | Yes | No | 2.62E+07 | 1.37E+07 | 1.20E+06 | 729.09 | 3.59E+04 | 1.88E+04 | 1.65E+03 |
| 22 | Yes | Yes | Yes | No | Yes | 1.08E+08 | 5.93E+07 | 1.09E+07 | 1075.28 | 1.00E+05 | 5.52E+04 | 1.01E+04 |
| 24 | Yes | Yes | Yes | Yes | Yes | 1.09E+08 | 6.00E+07 | 1.11E+07 | 1071.00 | 1.02E+05 | 5.60E+04 | 1.04E+04 |
| 20 | Yes | Yes | No | Yes | Yes | 1.08E+08 | 5.89E+07 | 1.04E+07 | 1004.24 | 1.07E+05 | 5.86E+04 | 1.03E+04 |
| 18 | Yes | Yes | No | No | Yes | 1.07E+08 | 5.82E+07 | 1.02E+07 | 966.83 | 1.10E+05 | 6.02E+04 | 1.05E+04 |
| 14 | No | Yes | Yes | No | Yes | 1.13E+08 | 6.16E+07 | 1.04E+07 | 782.56 | 1.44E+05 | 7.87E+04 | 1.33E+04 |
| 16 | No | Yes | Yes | Yes | Yes | 1.14E+08 | 6.23E+07 | 1.06E+07 | 776.06 | 1.47E+05 | 8.02E+04 | 1.37E+04 |
| 12 | No | Yes | No | Yes | Yes | 1.13E+08 | 6.11E+07 | 9.85E+06 | 694.09 | 1.62E+05 | 8.81E+04 | 1.42E+04 |
| 10 | No | Yes | No | No | Yes | 1.12E+08 | 6.04E+07 | 9.65E+06 | 658.74 | 1.69E+05 | 9.18E+04 | 1.46E+04 |
| 6 | No | No | Yes | No | Yes | 8.80E+07 | 4.86E+07 | 9.40E+06 | 124.31 | 7.08E+05 | 3.91E+05 | 7.56E+04 |
| 8 | No | No | Yes | Yes | Yes | 8.92E+07 | 4.93E+07 | 9.60E+06 | 115.13 | 7.75E+05 | 4.28E+05 | 8.34E+04 |
| 2 | No | No | No | No | Yes | 8.65E+07 | 4.74E+07 | 8.65E+06 | 93.61 | 9.24E+05 | 5.07E+05 | 9.24E+04 |
| 4 | No | No | No | Yes | Yes | 8.77E+07 | 4.81E+07 | 8.85E+06 | 87.05 | 1.01E+06 | 5.53E+05 | 1.02E+05 |

Table 51 shows the results of a general 120% Reactor Power upgrade configuration cost-benefit analysis. The most notable shift in cost-efficacy, when compared to the 100% Reactor Power results, was that not having PARs in configurations included the Filtered Containment Vents became nominally more cost-effective than also including the PARs. Implementing Igniters alongside Filtered Containment Vents, when considering the low cost-estimates, was, using the nominal mean difference in Release Activity, more cost-effective than implementing PARs with the Filtered Containment Vents. Additionally, not including any form of hydrogen mitigation alongside Hardened Containment Vents was more cost-effective than including both PARs and Igniters where the reverse was true for 100% Reactor Power.

Table 52. IRWST Individual Upgrade Multi-Parameter Cost-Benefit Analysis, 120% Reactor Power

| Test Case | Filters | Vents | PARs | Igniters | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|
| 14 | No | Yes | Yes | No | -27.99 | -3.09E+06 | -1.68E+06 | -3.09E+05 |
| 16 | No | Yes | Yes | Yes | -29.17 | -2.97E+06 | -1.62E+06 | -2.97E+05 |
| 12 | No | Yes | No | Yes | -35.00 | -2.47E+06 | -1.35E+06 | -2.47E+05 |
| 10 | No | Yes | No | No | -52.70 | -1.64E+06 | -8.95E+05 | -1.64E+05 |
| 20 | Yes | Yes | No | Yes | -76.94 | -1.12E+06 | -6.13E+05 | -1.12E+05 |
| 22 | Yes | Yes | Yes | No | -77.85 | -1.11E+06 | -6.06E+05 | -1.11E+05 |
| 24 | Yes | Yes | Yes | Yes | -78.01 | -1.11E+06 | -6.04E+05 | -1.11E+05 |
| 18 | Yes | Yes | No | No | -98.92 | -8.74E+05 | -4.77E+05 | -8.74E+04 |
| 8 | No | No | Yes | Yes | 152.29 | 5.68E+05 | 3.10E+05 | 5.68E+04 |
| 6 | No | No | Yes | No | 148.46 | 5.83E+05 | 3.18E+05 | 5.83E+04 |
| 4 | No | No | No | Yes | 103.19 | 8.38E+05 | 4.57E+05 | 8.38E+04 |
| 2 | No | No | No | No | 93.61 | 9.24E+05 | 5.04E+05 | 9.24E+04 |

Table 52 shows a cost-benefit analysis of the IRWST upgrade for the Refined Multi-Parameter Analysis at 120% Reactor Power. The IRWST remained ineffective at 120% Reactor Power in most configurations and, even assuming the calculated Mean Difference Value was exactly correct was still very cost-ineffective when it did provide a reduction in mean Release Activity, on top of said reductions not actually being statistically significant.

Table 53. Hydrogen Igniters Individual Upgrade Multi-Parameter Cost-Benefit Analysis, 120% Reactor Power

| Test Case | Filters | Vents | PARs | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|
| 23 | Yes | Yes | Yes | No | -4.13 | -2.92E+05 | -1.70E+05 | -4.85E+04 |
| 24 | Yes | Yes | Yes | Yes | -4.28 | -2.81E+05 | -1.64E+05 | -4.67E+04 |
| 15 | No | Yes | Yes | No | -5.32 | -2.26E+05 | -1.32E+05 | -3.76E+04 |
| 16 | No | Yes | Yes | Yes | -6.50 | -1.85E+05 | -1.08E+05 | -3.08E+04 |
| 4 | No | No | No | Yes | -6.56 | -1.84E+05 | -1.07E+05 | -3.05E+04 |
| 8 | No | No | Yes | Yes | -9.18 | -1.31E+05 | -7.65E+04 | -2.18E+04 |
| 7 | No | No | Yes | No | -13.01 | -9.26E+04 | -5.40E+04 | -1.54E+04 |
| 3 | No | No | No | No | -16.13 | -7.47E+04 | -4.35E+04 | -1.24E+04 |
| 20 | Yes | Yes | No | Yes | 37.42 | 3.22E+04 | 1.88E+04 | 5.34E+03 |
| 12 | No | Yes | No | Yes | 35.35 | 3.41E+04 | 1.99E+04 | 5.66E+03 |
| 11 | No | Yes | No | No | 17.65 | 6.83E+04 | 3.98E+04 | 1.13E+04 |
| 19 | Yes | Yes | No | No | 15.44 | 7.80E+04 | 4.55E+04 | 1.30E+04 |

Table 53 shows a cost-benefit analysis of the Hydrogen Igniters with Backup Power upgrade for the Refined Multi-Parameter Analysis at 120% Reactor Power. The Hydrogen Igniters with Backup Power were slightly more effective at 120% Reactor Power than 100% Reactor Power, but the gains remained statistically insignificant and the Hydrogen Igniters remained cost-ineffective by and large.

Table 54. Hardened Containment Ventilation Individual Upgrade Multi-Parameter Cost-Benefit Analysis, 120% Reactor Power

| Test Case | PARs | Igniters | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|
| 15 | Yes | Yes | No | 842.3848 | 2.97E+04 | 1.54E+04 | 1.19E+03 |
| 13 | Yes | No | No | 834.6974 | 3.00E+04 | 1.56E+04 | 1.20E+03 |
| 11 | No | Yes | No | 745.2215 | 3.35E+04 | 1.74E+04 | 1.34E+03 |
| 9 | No | No | No | 711.4386 | 3.51E+04 | 1.83E+04 | 1.41E+03 |
| 16 | Yes | Yes | Yes | 660.9248 | 3.78E+04 | 1.97E+04 | 1.51E+03 |
| 14 | Yes | No | Yes | 658.2486 | 3.80E+04 | 1.97E+04 | 1.52E+03 |
| 12 | No | Yes | Yes | 607.0362 | 4.12E+04 | 2.14E+04 | 1.65E+03 |
| 10 | No | No | Yes | 565.13 | 4.42E+04 | 2.30E+04 | 1.77E+03 |

Table 54 shows a cost-benefit analysis of the Hardened Containment Ventilation upgrade for the Refined Multi-Parameter Analysis at 120% Reactor Power. Compared to other possible options, implementing the Hardened Containment Ventilation upgrade alone was significantly more cost-effective at 120% Reactor Power than when compared at 100% Reactor Power.

Table 55. Filtered Containment Ventilation Individual Upgrade Multi-Parameter Cost-Benefit Analysis, 120% Reactor Power

| Test Case | PARs | Igniters | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|
| 17 | No | No | No | 354.3054 | -1.41E+04 | -6.35E+03 | 1.41E+03 |
| 19 | No | Yes | No | 352.094 | -1.42E+04 | -6.39E+03 | 1.42E+03 |
| 23 | Yes | Yes | No | 343.7729 | -1.45E+04 | -6.55E+03 | 1.45E+03 |
| 21 | Yes | No | No | 342.5781 | -1.46E+04 | -6.57E+03 | 1.46E+03 |
| 20 | No | Yes | Yes | 310.1533 | -1.61E+04 | -7.25E+03 | 1.61E+03 |
| 18 | No | No | Yes | 308.084 | -1.62E+04 | -7.30E+03 | 1.62E+03 |
| 24 | Yes | Yes | Yes | 294.9396 | -1.70E+04 | -7.63E+03 | 1.70E+03 |
| 22 | Yes | No | Yes | 292.7216 | -1.71E+04 | -7.69E+03 | 1.71E+03 |

Table 55 shows a cost-benefit analysis of the Filtered Containment Ventilation upgrade for the Refined Multi-Parameter Analysis at 120% Reactor Power. In terms of relative cost-efficacy, Filtered Containment Vents were most cost-effective when implemented alone, by a slim, statistically insignificant margin at both 100% Reactor Power and 120% Reactor Power. Generally speaking, the change in Reactor Power did not affect the relative difference between potential Filtered Containment Ventilation implementations in any significant way.

Table 56. PARs Individual Upgrade Multi-Parameter Cost-Benefit Analysis, 120% Reactor Power

| Test Case | Filters | Vents | Igniters | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|
| 7 | No | No | Yes | No | -21.03 | -7.13E+04 | -5.35E+04 | -3.57E+04 |
| 5 | No | No | No | No | -24.15 | -6.21E+04 | -4.66E+04 | -3.11E+04 |
| 14 | No | Yes | No | Yes | 123.82 | 1.21E+04 | 9.09E+03 | 6.06E+03 |
| 22 | Yes | Yes | No | Yes | 108.45 | 1.38E+04 | 1.04E+04 | 6.92E+03 |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 13 | No | Yes | No | No | 99.11 | 1.51E+04 | 1.14E+04 | 7.57E+03 |
| 21 | Yes | Yes | No | No | 87.38 | 1.72E+04 | 1.29E+04 | 8.58E+03 |
| 16 | No | Yes | Yes | Yes | 81.96 | 1.83E+04 | 1.37E+04 | 9.15E+03 |
| 15 | No | Yes | Yes | No | 76.14 | 1.97E+04 | 1.48E+04 | 9.85E+03 |
| 23 | Yes | Yes | Yes | No | 67.82 | 2.21E+04 | 1.66E+04 | 1.11E+04 |
| 24 | Yes | Yes | Yes | Yes | 66.75 | 2.25E+04 | 1.69E+04 | 1.12E+04 |
| 6 | No | No | No | Yes | 30.70 | 4.89E+04 | 3.66E+04 | 2.44E+04 |
| 8 | No | No | Yes | Yes | 28.08 | 5.34E+04 | 4.01E+04 | 2.67E+04 |

Table 56 shows a cost-benefit analysis of the PARs upgrade for the Refined Multi-Parameter Analysis at 120% Reactor Power. The PARs cost-efficacy was not significantly changed moving from 100% Reactor Power to 120% Reactor Power.

### 5.6.5. Alternate Stochastic Parameter Probability Distribution Upgrade Configuration Cost-Benefit Analysis

Table 57. Good PDF Multi-Parameter General Upgrade Cost-Benefit Analysis Results

| Test Case | Filters | Vents | PARs | Igniters | IRWST | Cost (High) | Cost (Mid) | Cost (Low) | Good PDF Mean Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 3 | No | No | No | Yes | No | 1.21E+06 | 7.03E+05 | 2.00E+05 | 205.0016 | 5.88E+03 | 3.43E+03 | 9.76E+02 |
| 17 | Yes | Yes | No | No | No | 2.00E+07 | 1.08E+07 | 1.50E+06 | 378.0967 | 5.29E+04 | 2.84E+04 | 3.97E+03 |
| 19 | Yes | Yes | No | Yes | No | 2.12E+07 | 1.15E+07 | 1.70E+06 | 406.8326 | 5.21E+04 | 2.82E+04 | 4.18E+03 |
| 21 | Yes | Yes | Yes | No | No | 2.15E+07 | 1.19E+07 | 2.25E+06 | 496.0351 | 4.33E+04 | 2.39E+04 | 4.54E+03 |
| 5 | No | No | Yes | No | No | 1.50E+06 | 1.13E+06 | 7.50E+05 | 156.11 | 9.61E+03 | 7.21E+03 | 4.80E+03 |
| 23 | Yes | Yes | Yes | Yes | No | 2.27E+07 | 1.26E+07 | 2.45E+06 | 472.5074 | 4.81E+04 | 2.66E+04 | 5.19E+03 |
| 13 | No | Yes | Yes | No | No | 2.65E+07 | 1.41E+07 | 1.75E+06 | 263.9375 | 1.00E+05 | 5.35E+04 | 6.63E+03 |
| 11 | No | Yes | No | Yes | No | 2.62E+07 | 1.37E+07 | 1.20E+06 | 169.8528 | 1.54E+05 | 8.07E+04 | 7.06E+03 |
| 9 | No | Yes | No | No | No | 2.50E+07 | 1.30E+07 | 1.00E+06 | 138.0693 | 1.81E+05 | 9.42E+04 | 7.24E+03 |
| 15 | No | Yes | Yes | Yes | No | 2.77E+07 | 1.48E+07 | 1.95E+06 | 235.5488 | 1.18E+05 | 6.29E+04 | 8.28E+03 |
| 7 | No | No | Yes | Yes | No | 2.71E+06 | 1.83E+06 | 9.50E+05 | 86.52522 | 3.13E+04 | 2.11E+04 | 1.10E+04 |
| 22 | Yes | Yes | Yes | No | Yes | 1.08E+08 | 5.93E+07 | 1.09E+07 | 382.422 | 2.82E+05 | 1.55E+05 | 2.85E+04 |
| 20 | Yes | Yes | No | Yes | Yes | 1.08E+08 | 5.89E+07 | 1.04E+07 | 344.9657 | 3.12E+05 | 1.71E+05 | 3.00E+04 |
| 24 | Yes | Yes | Yes | Yes | Yes | 1.09E+08 | 6.00E+07 | 1.11E+07 | 369.5632 | 2.95E+05 | 1.62E+05 | 3.00E+04 |
| 18 | Yes | Yes | No | No | Yes | 1.07E+08 | 5.82E+07 | 1.02E+07 | 294.796 | 3.61E+05 | 1.97E+05 | 3.44E+04 |
| 14 | No | Yes | Yes | No | Yes | 1.13E+08 | 6.16E+07 | 1.04E+07 | 170.8637 | 6.61E+05 | 3.60E+05 | 6.09E+04 |
| 6 | No | No | Yes | No | Yes | 8.80E+07 | 4.86E+07 | 9.40E+06 | 137.993 | 6.38E+05 | 3.52E+05 | 6.81E+04 |
| 16 | No | Yes | Yes | Yes | Yes | 1.14E+08 | 6.23E+07 | 1.06E+07 | 147.1505 | 7.76E+05 | 4.23E+05 | 7.20E+04 |
| 8 | No | No | Yes | Yes | Yes | 8.92E+07 | 4.93E+07 | 9.60E+06 | 122.1872 | 7.30E+05 | 4.03E+05 | 7.86E+04 |
| 2 | No | No | No | No | Yes | 8.65E+07 | 4.74E+07 | 8.65E+06 | 107.9086 | 8.02E+05 | 4.40E+05 | 8.02E+04 |
| 12 | No | Yes | No | Yes | Yes | 1.13E+08 | 6.11E+07 | 9.85E+06 | 116.892 | 9.64E+05 | 5.23E+05 | 8.43E+04 |
| 4 | No | No | No | Yes | Yes | 8.77E+07 | 4.81E+07 | 8.85E+06 | 102.4311 | 8.56E+05 | 4.70E+05 | 8.64E+04 |
| 10 | No | Yes | No | No | Yes | 1.12E+08 | 6.04E+07 | 9.65E+06 | 77.17701 | 1.44E+06 | 7.83E+05 | 1.25E+05 |

Table 57 shows the results of a general 100% Reactor Power upgrade configuration cost-benefit analysis using the "Good" alternate PDFs examined in Section 5.5.7. Very notably, Hydrogen Igniters with Backup Power are, by a fair margin, the most efficient upgrade available with the "Good" alternate PDFs and are vastly more effective with the "Good" alternate PDFs than with the primary PDFs used. PARs and Filtered Containment Vents remained cost-effective with these alternate PDFs, and the IRWST upgrade remained dead last for cost-efficiency.

Table 58. Good PDF IRWST Individual Upgrade Cost-Benefit Analysis, 100% Reactor Power

| Test Case | Filters | Vents | PARs | Igniters | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|
| 6 | No | No | Yes | No | -18.1169 | -4.77E+06 | -2.62E+06 | -4.77E+05 |
| 12 | No | Yes | No | Yes | -52.9608 | -1.63E+06 | -8.96E+05 | -1.63E+05 |
| 10 | No | Yes | No | No | -60.8923 | -1.42E+06 | -7.79E+05 | -1.42E+05 |
| 20 | Yes | Yes | No | Yes | -61.8669 | -1.40E+06 | -7.67E+05 | -1.40E+05 |
| 18 | Yes | Yes | No | No | -83.3008 | -1.04E+06 | -5.70E+05 | -1.04E+05 |
| 16 | No | Yes | Yes | Yes | -88.3983 | -9.79E+05 | -5.37E+05 | -9.79E+04 |
| 14 | No | Yes | Yes | No | -93.0738 | -9.29E+05 | -5.10E+05 | -9.29E+04 |
| 4 | No | No | No | Yes | -102.571 | -8.43E+05 | -4.63E+05 | -8.43E+04 |
| 24 | Yes | Yes | Yes | Yes | -102.944 | -8.40E+05 | -4.61E+05 | -8.40E+04 |
| 22 | Yes | Yes | Yes | No | -113.613 | -7.61E+05 | -4.18E+05 | -7.61E+04 |
| 2 | No | No | No | No | 107.9086 | 8.02E+05 | 4.40E+05 | 8.02E+04 |
| 8 | No | No | Yes | Yes | 35.66199 | 2.43E+06 | 1.33E+06 | 2.43E+05 |

Table 58 shows the results of the cost-benefit analysis of the IRWST upgrade at 100% Reactor Power using the "Good" alternate PDFs. The alternate set of PDFs did not significantly alter the effectiveness (or, rather, ineffectiveness) of the IRWST upgrade – even for Test 2, in which the Mean Release Difference was significantly larger with the "Good" alternate PDFs, the results were not statistically significant.

Table 59. Good PDF PARs Individual Upgrade Cost-Benefit Analysis, 100% Reactor Power

| Test Case | Filters | Vents | Igniters | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|
| 7 | No | No | Yes | No | -118.476 | -1.27E+04 | -9.496E+03 | -6.33E+03 |
| 5 | No | No | No | No | 156.11 | 9.61E+03 | 7.206E+03 | 4.80E+03 |
| 13 | No | Yes | No | No | 125.8681 | 1.19E+04 | 8.938E+03 | 5.96E+03 |
| 21 | Yes | Yes | No | No | 117.9384 | 1.27E+04 | 9.539E+03 | 6.36E+03 |
| 14 | No | Yes | No | Yes | 93.6867 | 1.60E+04 | 1.201E+04 | 8.01E+03 |
| 22 | Yes | Yes | No | Yes | 87.62603 | 1.71E+04 | 1.284E+04 | 8.56E+03 |
| 15 | No | Yes | Yes | No | 65.69603 | 2.28E+04 | 1.712E+04 | 1.14E+04 |
| 23 | Yes | Yes | Yes | No | 65.67486 | 2.28E+04 | 1.713E+04 | 1.14E+04 |
| 16 | No | Yes | Yes | Yes | 30.25856 | 4.96E+04 | 3.718E+04 | 2.48E+04 |
| 6 | No | No | No | Yes | 30.08444 | 4.99E+04 | 3.739E+04 | 2.49E+04 |
| 24 | Yes | Yes | Yes | Yes | 24.59744 | 6.10E+04 | 4.574E+04 | 3.05E+04 |
| 8 | No | No | Yes | Yes | 19.75615 | 7.59E+04 | 5.694E+04 | 3.80E+04 |

Table 59 shows the results of the cost-benefit analysis of the PAR upgrade at 100% Reactor Power using the “Good” alternate PDFs. The Release Activity increase in Test 7 is not statistically significant, nor are any of the other changes in Release Activity with the addition of PARs to non-ventilated configurations. It would take roughly 500 sample tests per configuration to generate statistically significant data for these configurations, as the difference in Mean Release Activity between the configurations is small, and the variance in Release Activity within each configuration’s sample pool is large, making it burdensome to generate sufficient data to establish statistical significance in some cases.

Table 60. Good PDF Hardened Containment Vents Individual Upgrade Cost-Benefit Analysis, 100% Reactor Power

| Test Case | PARs | Igniters | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|
| 10 | No | No | Yes | -30.7316 | -8.13E+05 | -4.23E+05 | -3.25E+04 |
| 11 | No | Yes | No | -35.1489 | -7.11E+05 | -3.70E+05 | -2.85E+04 |
| 15 | Yes | Yes | No | 149.0236 | 1.68E+05 | 8.72E+04 | 6.71E+03 |
| 9 | No | No | No | 138.0693 | 1.81E+05 | 9.42E+04 | 7.24E+03 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| 13 | Yes | No | No | 107.8275 | 2.32E+05 | 1.21E+05 | 9.27E+03 |
| 14 | Yes | No | Yes | 32.87066 | 7.61E+05 | 3.95E+05 | 3.04E+04 |
| 16 | Yes | Yes | Yes | 24.96331 | 1.00E+06 | 5.21E+05 | 4.01E+04 |
| 12 | No | Yes | Yes | 14.4609 | 1.73E+06 | 8.99E+05 | 6.92E+04 |

Table 60 shows the results of the cost-benefit analysis of the Hardened Containment Ventilation upgrade at 100% Reactor Power using the "Good" alternate PDFs. Using the "Good" alternate PDFs causes the Hardened Containment Vents impact the Igniters configuration to change from a large, statistically significant reduction to statistical insignificance. In fact, using the "Good" alternate PDFs made the contribution of the Hardened Containment Ventilation upgrade statistically insignificant for all tested upgrade configurations except for testing the Hardened Containment Vents alone, and testing them with PARs and Igniters at the same time. The use of the "Good" alternate PDFs severely diminished the Release Activity reduction.

Table 61. Good PDF Filtered Containment Vents Individual Upgrade Cost-Benefit Analysis, 100% Reactor Power

| Filters | PARs | Igniters | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|
| 17 | No | No | No | 240.0274 | -2.08E+04 | -9.37E+03 | 2.08E+03 |
| 19 | No | Yes | No | 236.9798 | -2.11E+04 | -9.49E+03 | 2.11E+03 |
| 23 | Yes | Yes | No | 236.9587 | -2.11E+04 | -9.50E+03 | 2.11E+03 |
| 21 | Yes | No | No | 232.0976 | -2.15E+04 | -9.69E+03 | 2.15E+03 |
| 20 | No | Yes | Yes | 228.0738 | -2.19E+04 | -9.87E+03 | 2.19E+03 |
| 24 | Yes | Yes | Yes | 222.4126 | -2.25E+04 | -1.01E+04 | 2.25E+03 |
| 18 | No | No | Yes | 217.619 | -2.30E+04 | -1.03E+04 | 2.30E+03 |
| 22 | Yes | No | Yes | 211.5583 | -2.36E+04 | -1.06E+04 | 2.36E+03 |

Table 61 shows the results of the cost-benefit analysis of the Filtered Containment Ventilation upgrade at 100% Reactor Power using the "Good" alternate PDFs. Unlike the Hardened Containment Vents, the contribution of the Filtered Containment Vents remained statistically significant even after being diminished with by the alternate point weighting of the "Good" alternate PDFs. The Filtered Containment Vents remained cost-effective with the new stochastic parameter PDFs, and other than reducing the magnitude of the reduction in Release Activity from implementing the Filtered Containment Vents, the alternate PDFs changed little about the results.

Table 62. Good PDF Hydrogen Igniters with Backup Power Individual Upgrade Cost-Benefit Analysis, 100% Reactor Power

| Test Case | Filters | Vents | PARs | IRWST | Mean Release Difference | Cost per Mean kCi prevented, High Cost | Cost per Mean kCi prevented, Mid Cost | Cost per Mean kCi prevented, Low Cost |
|---|---|---|---|---|---|---|---|---|
| 4 | No | No | No | Yes | -5.478 | -2.20E+05 | -1.28E+05 | -3.65E+04 |
| 24 | Yes | Yes | Yes | Yes | -12.86 | -9.37E+04 | -5.46E+04 | -1.56E+04 |
| 8 | No | No | Yes | Yes | -15.81 | -7.62E+04 | -4.44E+04 | -1.27E+04 |
| 23 | Yes | Yes | Yes | No | -23.53 | -5.12E+04 | -2.99E+04 | -8.50E+03 |
| 16 | No | Yes | Yes | Yes | -23.71 | -5.08E+04 | -2.96E+04 | -8.43E+03 |
| 15 | No | Yes | Yes | No | -28.39 | -4.24E+04 | -2.47E+04 | -7.05E+03 |
| 7 | No | No | Yes | No | -69.58 | -1.73E+04 | -1.01E+04 | -2.87E+03 |
| 3 | No | No | No | No | 205.00 | 5.88E+03 | 3.43E+03 | 9.76E+02 |
| 20 | Yes | Yes | No | Yes | 50.17 | 2.40E+04 | 1.40E+04 | 3.99E+03 |
| 12 | No | Yes | No | Yes | 39.71 | 3.03E+04 | 1.77E+04 | 5.04E+03 |
| 11 | No | Yes | No | No | 31.78 | 3.79E+04 | 2.21E+04 | 6.29E+03 |
| 19 | Yes | Yes | No | No | 28.74 | 4.19E+04 | 2.44E+04 | 6.96E+03 |

Table 62 shows the results of the cost-benefit analysis of the Hydrogen Igniters with Backup Power upgrade at 100% Reactor Power using the “Good” alternate PDFs. For many cases, the impact of the Hydrogen Igniters was too small to be statistically significant, but the Igniters did provide a sizable and statistically significant reduction in Release Activity when implemented alone. The Hydrogen Igniters, implemented alone, examined with the “Good” alternate PDFs, were the only upgrade examined in this research to achieve a cost per kilocurie of reduction below 1,000 \$/kCi.

## 5.7. *Scatter, Noise, and Outliers*

Initial testing in the stochastic parameter sensitivity studies, and especially the initial upgrade testing, showed a fair amount of noise in the data. Noise in the initial upgrade testing has been discussed previously – a uniform sampling distribution, with a heavily probabilistic weighting towards very short EDG failure times caused the functional number of data points to be roughly one tenth the number of data points sampled, making the results highly susceptible to noise.

The noise present in the data was found to come from one of two sources. The first was that the release activity of vapor radionuclides was frequently unpredictable and not well correlated with the release activity of aerosol radionuclides. This would not be a significant issue, but for the existence of the Xenon class. Xenon was both unfilterable and also responsible for almost all of the release activity of vaporous radionuclides.

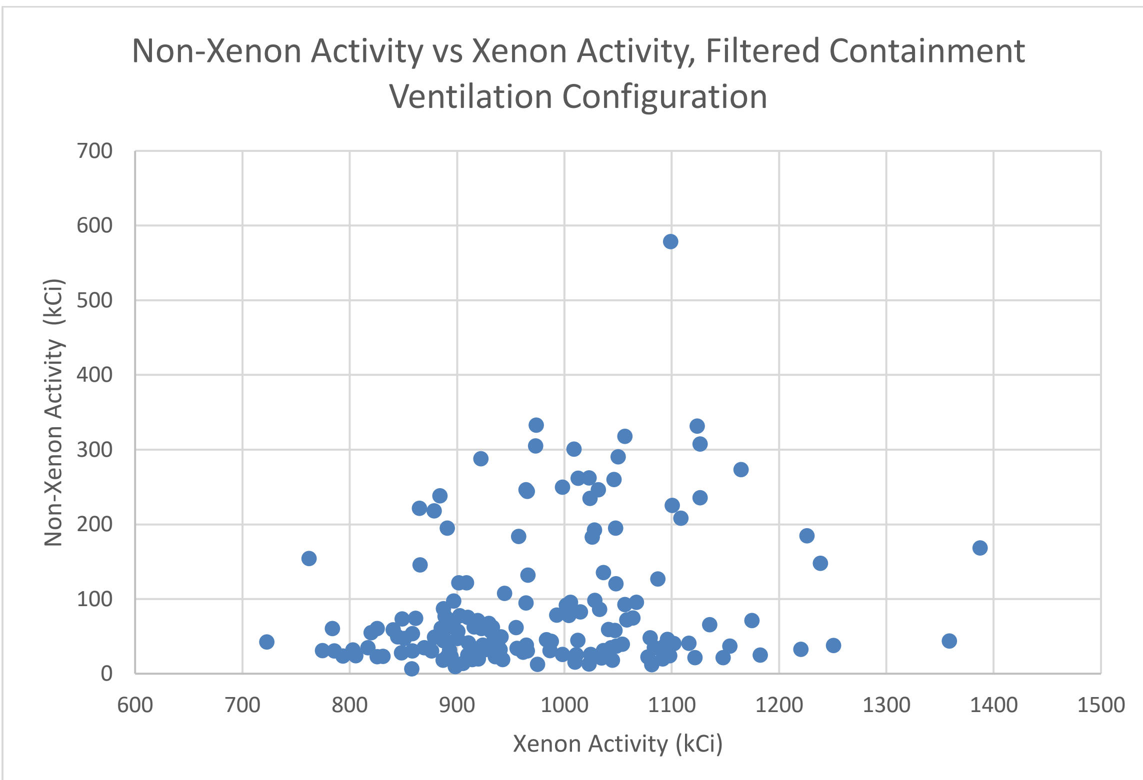


Figure 95 – Non-Xenon vs Xenon Release Activity for Filtered Containment Ventilation Configuration

Figure 95 shows a plotting of the Xenon Release Activity vs the non-Xenon Release Activity for all MELCOR models completed for the Filtered Containment Ventilation upgrade configuration multi-parameter analysis. The Xenon was found to be largely erratic with regards to how it behaved in relation to stochastic parameters, with its strongest stochastic correlations –

which were still fairly weak - lying with the delay or reduction of core damage, which kept more Xenon trapped in the fuel cladding, as once the Xenon was released from the fuel, it became essentially impossible to keep it from reaching the environment without keeping the containment entirely sealed.

The second source of significant noise was that hydrogen effects were sometimes subject to cliff edge effects – potentially favorable circumstances preventing an early, weak hydrogen burn would sometimes the lead to a greater buildup of hydrogen and a late, large burn that would increase the release activity. This was most readily apparent during the containment ventilation individual upgrade testing, the results of which are repeated below.

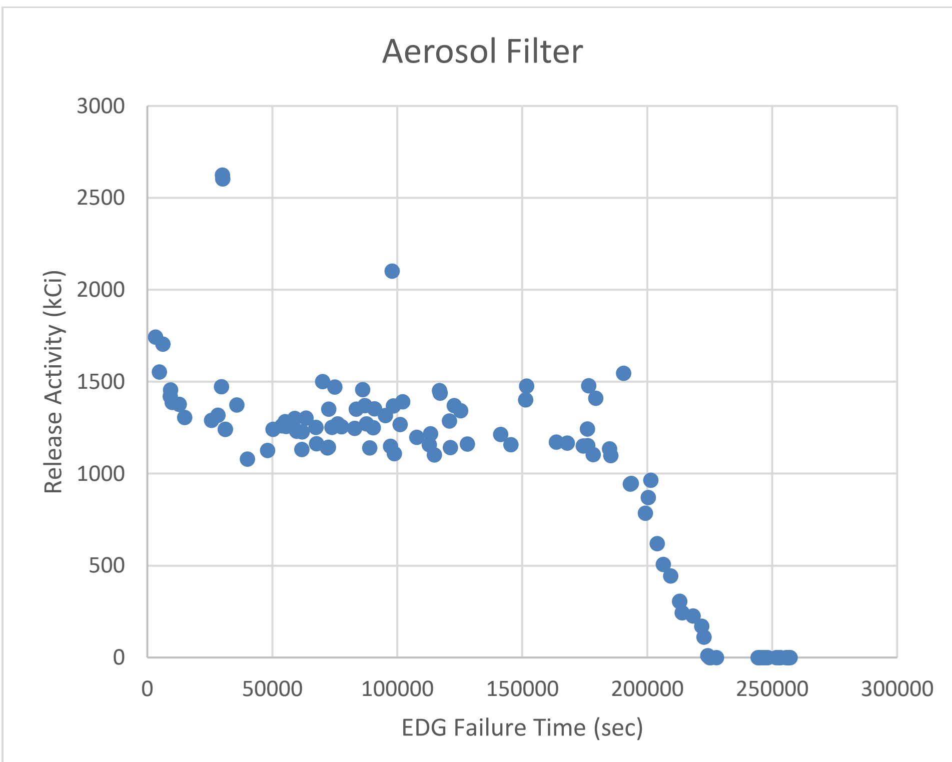


Figure 96 – Aerosol Filtered Containment Ventilation Initial Upgrade Impact Assessment

As seen in Figure 31, there are major, notable spikes in the release activity for the Aerosol Filtered Containment Ventilation test when the EDG Failure Time was sampled to be between 29,600 seconds and 31,200 seconds, as well as between 97,200 seconds and 98,400 seconds, major spikes in release activity were observed. Supplemental sample data in and around these discontinuities was generated to look at them in detail. This is examined in more detail below.

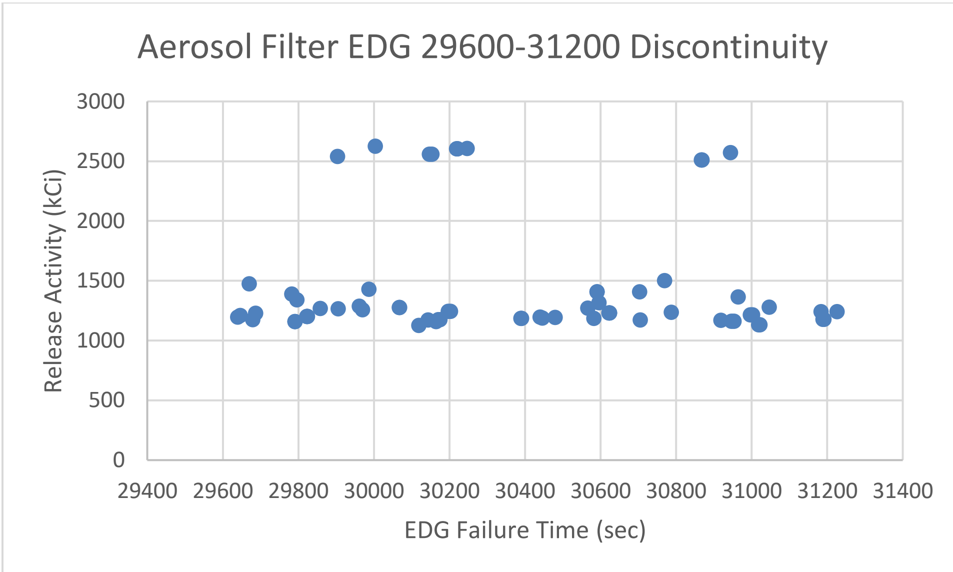


Figure 97 – Detailed view of discontinuous data for Aerosol Filter testing, low EDG Failure Time discontinuity

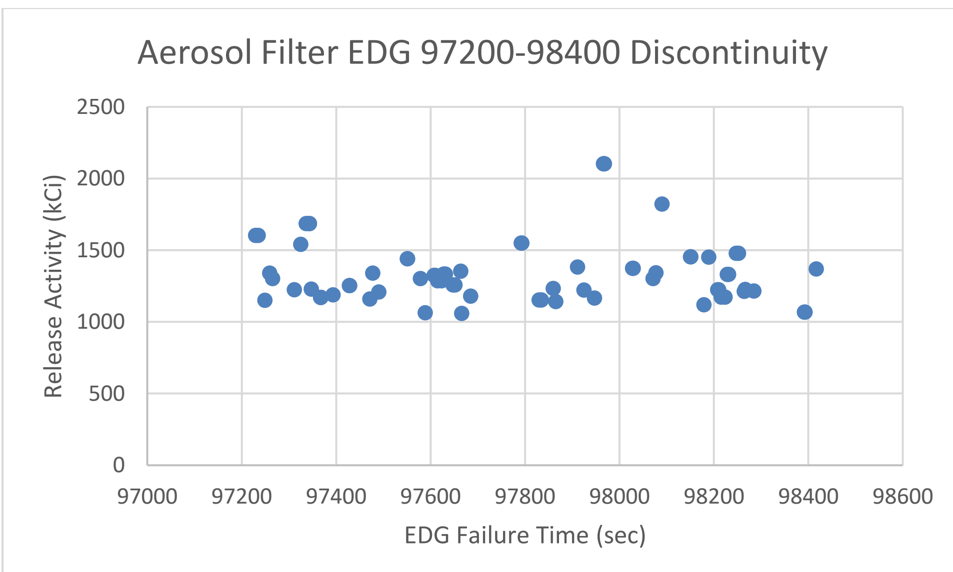


Figure 98 – Detailed view of discontinuous data for Aerosol Filter testing, high EDG Failure Time discontinuity

Figure 97 and Figure 98 present detailed looks at the supplemental data, taken for EDG Failure Times. The discontinuities remain unresolved and with no predictable rise and fall in the Release Activity, even at this higher sampling resolution. The phenomena behind this is that these cases are, truly, quite similar and there is a very severe cliff edge effect occurring here from relatively small variances in hydrogen generation and combustion creating pressure shockwaves just severe enough to burst the containment, which then enormously increases the release activity.

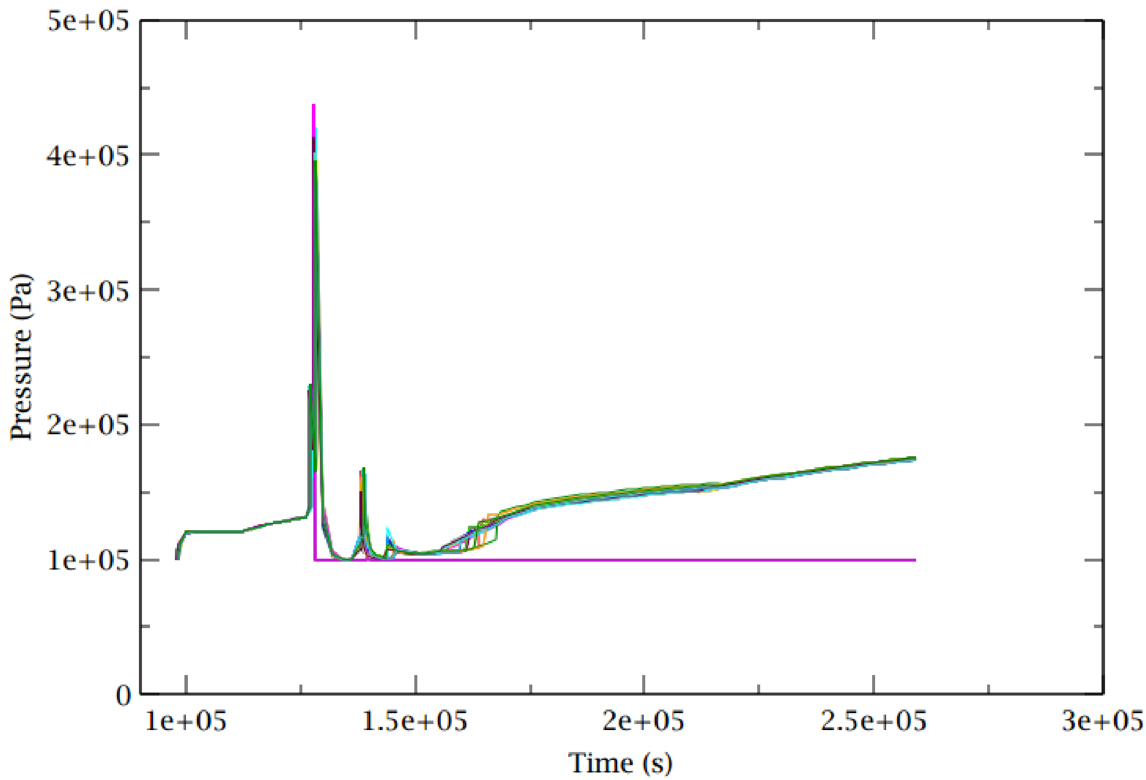


Figure 99 – General View of high EDG Failure Time data discontinuity containment pressure

Figure *99* shows a plotting of the containment pressure vs time for 30 different sampled runs with EDG Failure Time between 97,200 and 98,400 seconds. The runs are very close to identical until 127,800 seconds into the transient, at which point they branch into two groupings and are, within each grouping, nearly identical. What splits these groupings is relatively minor differences in the circumstances under which the first, and largest, hydrogen deflagration occurs that cause variations in the shockwave peak pressure. In most scenarios, the peak pressure during the hydrogen deflagration is close to the containment failure pressure but does not quite cause the containment to rupture. However, in the cases with the significant increase in release activity, the hydrogen deflagration is just violent enough to cause the containment to fail.

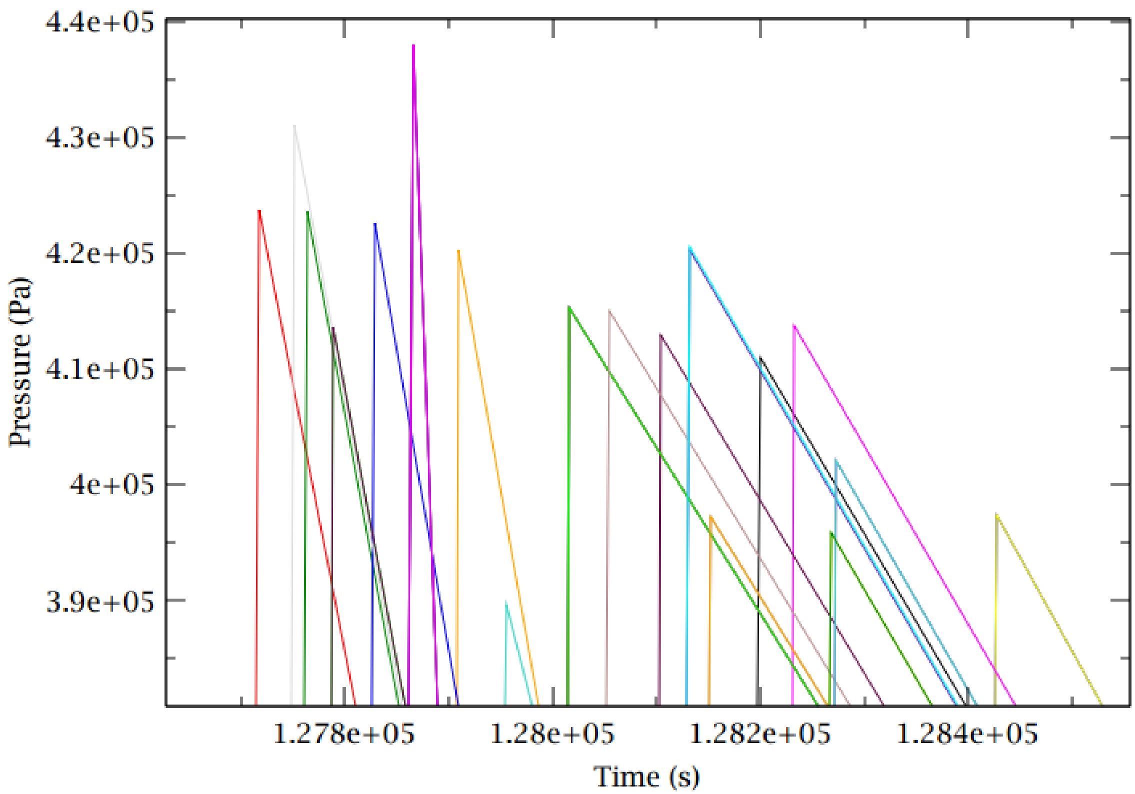


Figure 100 – Peak pressure view of high EDG Failure Time data discontinuity containment pressure

Figure 100 shows a zoomed in view of the containment pressure in all of the cases shown in Figure 99. While the time resolution of the plot file is not sufficiently fine to show the full rise and fall of the pressure because of the speed with which the pressure spike occurs and dissipates, Figure 100 shows how similar these cases are – the hydrogen deflagration pressure spikes all occur within roughly a minute of each other, and in every case modeled, the containment pressure rises dramatically before decreasing. The discontinuity lies in the few cases in which the containment pressure rises to exceed the containment failure pressure.

## 6. Conclusions

The goal of this project was to show prove the concept of RAVEN as a tool that can be used to rigorously explore the potential offsite consequences of a severe accident to provide and effective methodology for evaluating potential nuclear power plant upgrades to optimize changes to power plants to maximize gains in safety per dollar spent on upgrades.

The Filtered Containment Ventilation upgrade, using our model and stochastic parameter PDFs, was extremely effective. The Passive Autocatalytic Recombiners upgrade showed some promise, but only when implemented in tandem with some form of containment ventilation. The Igniter upgrades were found to be largely ineffective on their own, though it is certainly within the realm of possibility that other accident scenarios may pose a greater threat to containment integrity via hydrogen deflagration than steam accumulation, where the primary threat to containment integrity here was, generally speaking, through steam accumulation and failure via slow over-pressurization. The IRWST was, as expected, highly cost ineffective. It was simultaneously extremely expensive and largely ineffective, and as such it would almost entirely be a waste of money to try to retrofit it to an older power reactor. Depending on the stochastic parameters used, the Hydrogen Igniters with backup power showed promise, and the potential to entirely negate radionuclide release by maintaining containment integrity without using vents to relieve pressure is a promising one. Lastly, our findings agree with and support the NRC order to add the Hardened Containment Ventilation upgrade to Mk1 Containment BWRs. Their cost-effectiveness varied due to the widely differing cost-estimates found in the literature, but they were an extremely effective upgrade at mitigating the worst possible outcomes and were very cost-effective when considered using anything but the highest possible cost-estimates.

We have found that this methodology can very much be influenced by variations in cost estimates. The cost-efficacy of many upgrades, comparatively speaking, shifted when considering two upgrade configurations assuming high cost estimates vs assuming low cost estimates. This is potentially obvious, given that any cost-benefit analysis is vulnerable to flaws in the cost-estimation stage of the analysis, but it is worth discussing that this methodology for a cost-benefit analysis is no different.

Additionally, care must be taken when selecting sampling ranges and probabilistic weighting functions to ensure against under sampling heavily weighted areas of stochastic parameter probability distribution functions. The initial EDG Sensitivity study showed that the methodology can be very vulnerable to mismatches between the sampling space bounds and the probability distributions used to weight sample points that can lead to significant amplification of statistical uncertainty within the analysis by functionally removing the majority of data points from the analysis and producing a significantly reduced, potentially insufficient sample size.

It was observed that shifts in the probability distributions of upgrade configurations were capable of altering the results significantly – some upgrade configurations were found to be resilient in most, but not all, sampled cases, and heavily weighting the vulnerable points in the sampling space to target particular upgrade configurations drove down their cost-efficacy significantly, while leaving other upgrades comparatively unscathed. Similarly, even without particularly aiming to favor one upgrade over another, care must be taken to ensure that realistic stochastic parameter PDFs are used at the risk of creating a "garbage in, garbage out" analysis.

A flaw with this methodology is that, thus far, it does not significantly address non-linearities and 'cliff-edge effects,' as the use of blind, simple random sampling demands that the sample space be heavily saturated with sample points to the point that unseen cliff-edge effects can be reasonably ruled out, which demands heavily burdensome volumes of data. Potential solutions and refinements to this issue are discussed in the Future Work section.

While a 90% or better run success rate was achieved in the Refined Multi-Parameter Accident Scenario Upgrade Configuration Impact Analysis, any failures have the potential to call the results of the analysis into question due to the possibility of cliff-edge effects being hidden by unresolved singularities within the data leading to model crashes obscuring the impact of the cliff-edge effects. While no correlations between stochastic parameter inputs and crash rates were found within the Refined Multi-Parameter Accident Scenario Upgrade Configuration Impact Analysis, such effects were found within the Initial Multi-Parameter Accident Scenario Upgrade Configuration Impact Analysis with the vestigial AC Power Recovery options that were unintentionally included. Given the high dimensionality of the data involved, it can be difficult to sniff out more subtle flaws in the output data. Efforts to generate analyses applicable to real-world problems and specific power plants would do well to strive for as close to 100% success rate as can be achieved. Other means to address this issue are proposed in the Future Work section.

A surprising finding was that the number of sample points necessary to produce statistically significant results was often *vastly* smaller than initially expected. At the outset of the work that entails this research, it was anticipated that each individual upgrade configuration test would demand thousands, if not tens of thousands of data points to produce statistically conclusive and meaningful results. This was refined to hundreds of cases, to potentially a thousand, after examination of the DKW inequality discussed in Section 4.10. This was yet further refined to a few dozen to a hundred cases when the existence of the Welch Treatment T-Test Confidence Interval, also discussed in Section 4.10, was discovered and implemented. For upgrades with relatively small impacts, such as the PARs or the Hydrogen Igniters, more cases than this are necessary to achieve statistical significance, but if an upgrade is expected to make a moderate to major impact, comparatively few samples can be used.

The discovery of the relative ease of performing enough MELCOR analyses to provide a statistically significant sample size for many upgrades greatly improves the practical efficacy of this methodology, reducing the computational needs to compare a handful of upgrade configurations from demanding a super-computing cluster and thousands of CPUs to the possibility of performing such an analysis on a dedicated 8-core desktop computer, a vast improvement in usability. The efficacy of the use of RAVEN-MELCOR for the economic evaluation of power plant upgrades make it a valuable and potent tool for the cost-optimization of any potential safety improvements at a nuclear power plant, and a powerful addition to the RISMC methodology toolkit.

## 7. Future Work

Future work to extend this project would generally encompass developing more MELCOR models to evaluate other types of nuclear power plant and other accident scenarios. For a severe accident mitigation upgrade to be appropriately and completely examined for its overall usefulness, it must be examined through the lenses of many different potential severe accidents. It would also be valuable to perform an abbreviated Fault Tree/Event Tree analysis of the initiating events and event sequences leading to these severe accidents to gain a better understanding of the probabilities involved in the accident sequence even occurring, though it seems possible that such an undertaking would be a larger project than even a doctoral dissertation could encompass.

This methodology, as with any methodology based on a model of a physical system, is heavily dependent on both the accuracy and stability of the model. While the model used here was reasonably grounded in realism and was largely developed by world-class experts in the use of MELCOR for NPP modeling, it was a very coarse model that traded some degree of accuracy for run speed, in the name of expediency. Future efforts with this methodology would do well to ensure that ample resources and lead time are given to allow for a slower, more refined model to be used without compromising the project schedule.

A potential means to address model stability issues, and the possibility of unresolved singularities within the data obscuring deleterious outcomes within the scenario would be for

RAVEN to “quarantine” crashed runs of the MELCOR model used within a study. During early hand-testing of the MELCOR model used in this study, it was frequently found that altering various parameters of the model infinitesimally would correct the issue, as the crashes were numerical in nature and slight alterations to the scenario leading into the crash point would allow the exact combination of specifications that would end the run to be bypassed. With only one stochastic parameter, this could potentially be done by re-running the model using, say, 99.99%, 99.98%, 100.01%, and 100.02% of the stochastic parameter value that crashed the model. Assuming they succeed, this would wrap the crashed run in a set of successful runs to hedge against cliff edge effects hiding within unresolved singularities. If some percentage of the “bubble” also crashes, the point could be marked by RAVEN for further inquiry.

A flaw found with the methodology thus far was that the use of simple random sampling can force unfavorable tradeoffs between the confidence that the results have not overlooked any sharp cliff-edge effects and a reasonable runtime, particularly with a slower running model. While they were not used here due to the broad scope of the project and time limitations, RAVEN does offer the possibility of intelligent sampling strategies that make use of past data to inform future efforts. It is possible that by beginning with simple random sampling for an initial set of data points, then using these data points to explore further would allow for cliff-edge effects to be intelligently mapped out and explored. However, it is also possible that localized cliff-edge effects would be missed in the initial samplings and ignored entirely. In the end, intelligent use of resources and expert engineering insight must be utilized to ensure the appropriateness of the analysis.

The implementation of the ATF upgrade, and other potential upgrades, would also be a valuable extension of this work. The ATF upgrade is mentioned specifically as unsuccessful attempts at modeling the impacts of such an upgrade were made, and it is currently an extremely active area of interest within the field of Nuclear Engineering.

A flaw, with regards to usability, found with this methodology was that the amount of data generated was frequently highly cumbersome. Even with a reduced number of runs to be executed, and with much of the data from individual runs stripped down to facilitate handling the data, the amount of data handled quickly climbed into the hundreds of gigabytes of data and acquiring sufficient storage to house it for later examination, rather than pulling was what specifically requested and deleting everything else, was frequently a problem. Deleting all unrequested data made it difficult to look back in greater detail at runs of particular interest and frequently necessitated re-running particular data points by hand for further examination, a frequently time-consuming process that delayed project progress repeatedly. A means to address this issue would majorly improve the wieldiness of the methodology

Additionally, a full Level 3 PRA analysis, coupling RAVEN-MELCOR to MELMACCS and MACCS2, would be a valuable extension of this project to further prove the efficacy and legitimacy of using prevented Curies of release as a shorthand for the rapid evaluation of severe accident scenario offsite economic consequences. Coupling these codes together would allow for direct Monte-Carlo sampled stochastic parameter analysis of the offsite consequences of severe accidents, rather than stopping at the evaluation of the release source term activity and correlating that value to the general magnitude of the offsite consequences of the accident.

Applying this framework to the consequences of less than severe accidents within a power plant to evaluate strategies to mitigate on-site economic consequences to power plant owners and utilities would yet further enrich and add value to the RISMC methodology toolkit. In particular, the possibility of early AC Power recovery during an SBO may hold interest, especially with regards to examination of the Hydrogen Igniters without concurrent implementation of backup power systems.

Outside of analyzing accident scenarios at older plants, any of the industrial applications of Monte-Carlo methods discussed in the literature review would benefit from the level of detail and insight obtained from the RAVEN-driven Monte-Carlo integrated modeling approach demonstrated here. Lastly, this framework has applications for design alternative analysis for any nuclear power plant design that is still a work in progress, or as a replacement for best-estimate or bounding case analysis methodologies in essentially any application where assumptions regarding important and unpredictable scenario inputs must be made. If an assumption is made, it should be explored.

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**Appendix A: BWR safety system diagrams and further discussion**

The Standby Liquid Control System (SLCS), seen in Figure A-1, consists of a tank of heavily borated water, a set of two pumps, a set of two explosive valves, and the necessary pipes to connect the SLCS components and to connect the SLCS to the reactor vessel. The boron in the borated water acts as a neutron poison and will end the nuclear chain reaction in the core when injected. The SLCS never triggers automatically and must be started manually, and functions as a backup to the Reactor Protection System automatic SCRAM. When started by the operator, the SLCS explosive valves burst open and the SLCS pumps begin injecting the borated water into the core, killing the reaction.

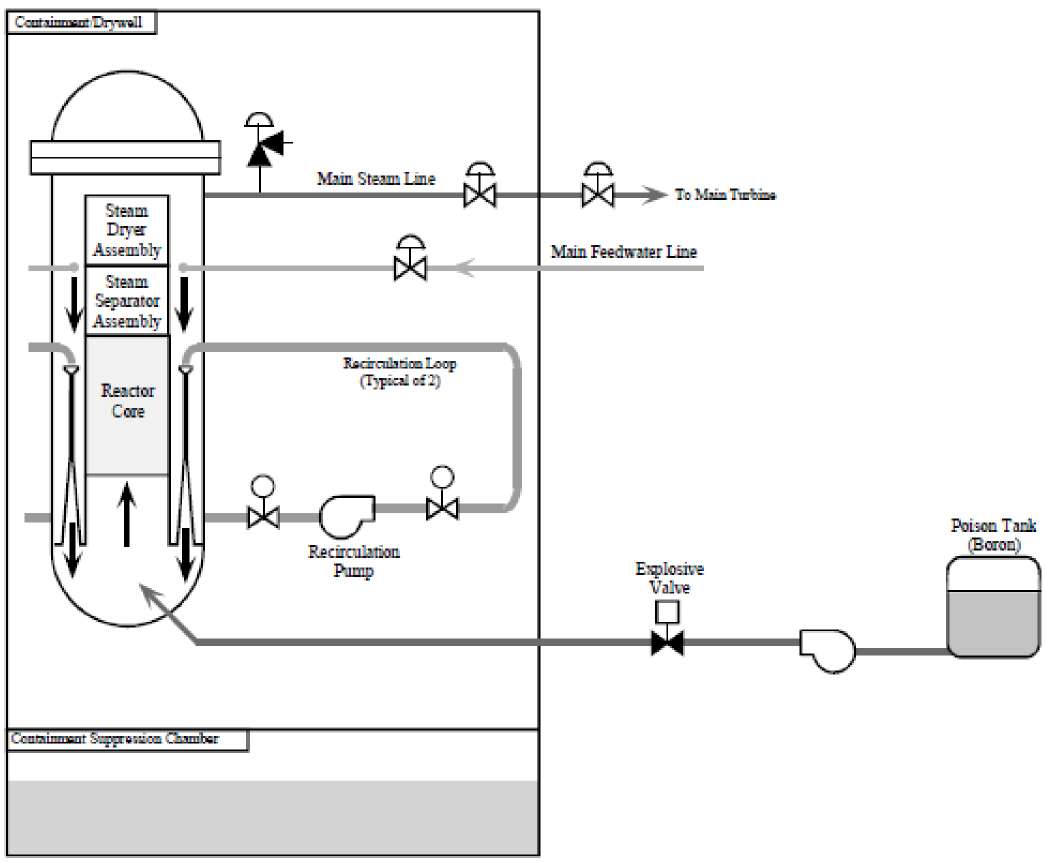


Figure A-1: Standby Liquid Control System (nrc.gov 2012)

The High Pressure Coolant Injection (HPCI) system and Automatical Depressurization System (ADS), seen in Figure A-2, serve to maintain core inventory and core pressure during accident scenarios. Additionally, the ADS can be used to depressurize the reactor in scenarios where the HPCI system can no longer be used to maintain core inventory or scenarios where the HPCI system is insufficient to maintain core inventory. The ADS is a system of hardwired logical trips and a series of safety relief valves used to vent steam from the core into the Containment Suppression Chamber (also called the Pressure Suppression Pool). The Containment Suppression Chamber is a large tank of subcooled water that is used to cool vented steam and provide a sink to which the ADS can safely vent steam.

Aside from a supply of water, the HPCI system requires no external support from other plant systems to operate and is an independent system capable of providing make up water to the core. The HPCI system can draw water either from the Condensate Storage Tank, a backup tank of water available for use with Emergency Core Cooling System subsystems, or from the Containment Suppression Chamber. It consists of control valves, a turbine that draws steam from the Main Steam Line for power and a pump, driven by the HPCI Turbine, that injects water from the Condensate Storage Tank or Containment Suppression Chamber into the reactor vessel.

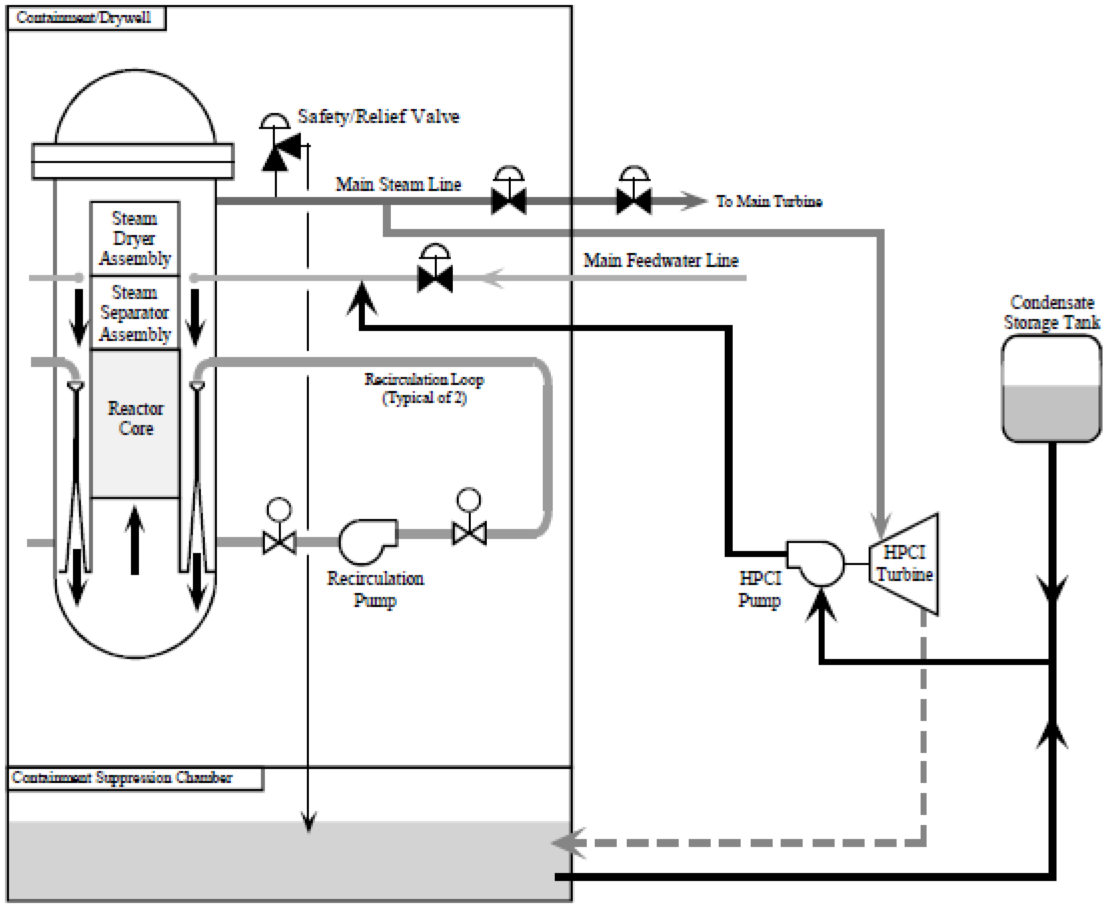


Figure A-2: High Pressure Coolant Injection and Automatic Depressurization Systems (nrc.gov 2012)

The Reactor Core Isolation Cooling (RCIC) system, seen in Figure A-3, fulfills a similar function as the HPCI system. Similar to the HPCI system, the RCIC system consists of control valves, a turbine that uses steam from the Main Steam Line for power, and a turbine driven pump that draws water from either the Condensate Storage Tank or the Containment Suppression Chamber to provide make up coolant and cooling for the core. As with the HPCI system, the RCIC system requires no external support from other plant systems aside from a supply of water. Most importantly, the HPCI and RCIC systems do not require external AC power, and can be operated for hours using DC battery power and steam from the Main Steam Line to provide core cooling during Station Blackout scenarios.

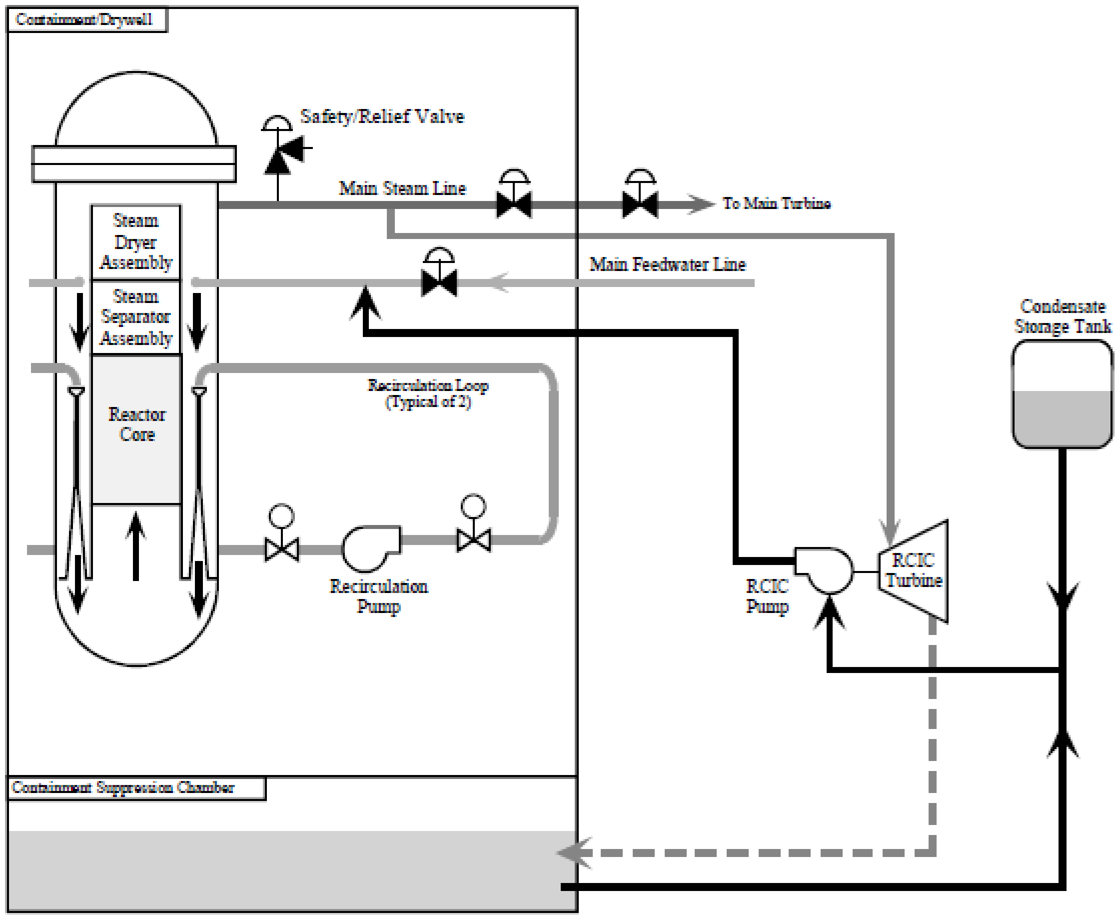


Figure A-3: Reactor Core Isolation Cooling system (nrc.gov 2012)

The low-pressure Emergency Core Cooling System subsystems, seen in figure A-4, consist of two independent systems, the Low Pressure Coolant Injection (LPCI) system and the Low Pressure Core Spray (LPCS) system. The LPCI system is, in reality, a second operational mode of the Residual Heat Removal (RHR) system. During normal shut down operations, the RHR system can be used to remove decay heat from the core indefinitely, keeping it in a cooled and stable configuration. During accident scenarios, typically large break loss of coolant accident (LOCA) conditions, the RHR system can be switched to its LPCI operational mode to inject large amounts of water from the Containment Suppression Chamber into the core to provide emergency makeup water to maintain core cooling. The RHR (and thus LPCI) system consists of control valves, piping, a heat exchanger used during normal shut down operations, and a pair of pumps that draw cold water from the Containment Suppression Chamber and inject it into the core.

The LPCS system is used to condense steam generated by a major accident scenario, keeping core pressure low so the LPCI and LPCS systems can operate, as high pressure in the core prevents these systems from functioning. It accomplishes this through a pair of pumps that draw water from the Pressure Suppression Chamber and injects it into the core from above, collapsing steam voids above the core to reduce core pressure. Additionally, the LPCS system can be used in accident scenarios where the LPCI system cannot adequately maintain the core water level to spray the fuel from above to maintain core cooling despite the fuel being uncovered.

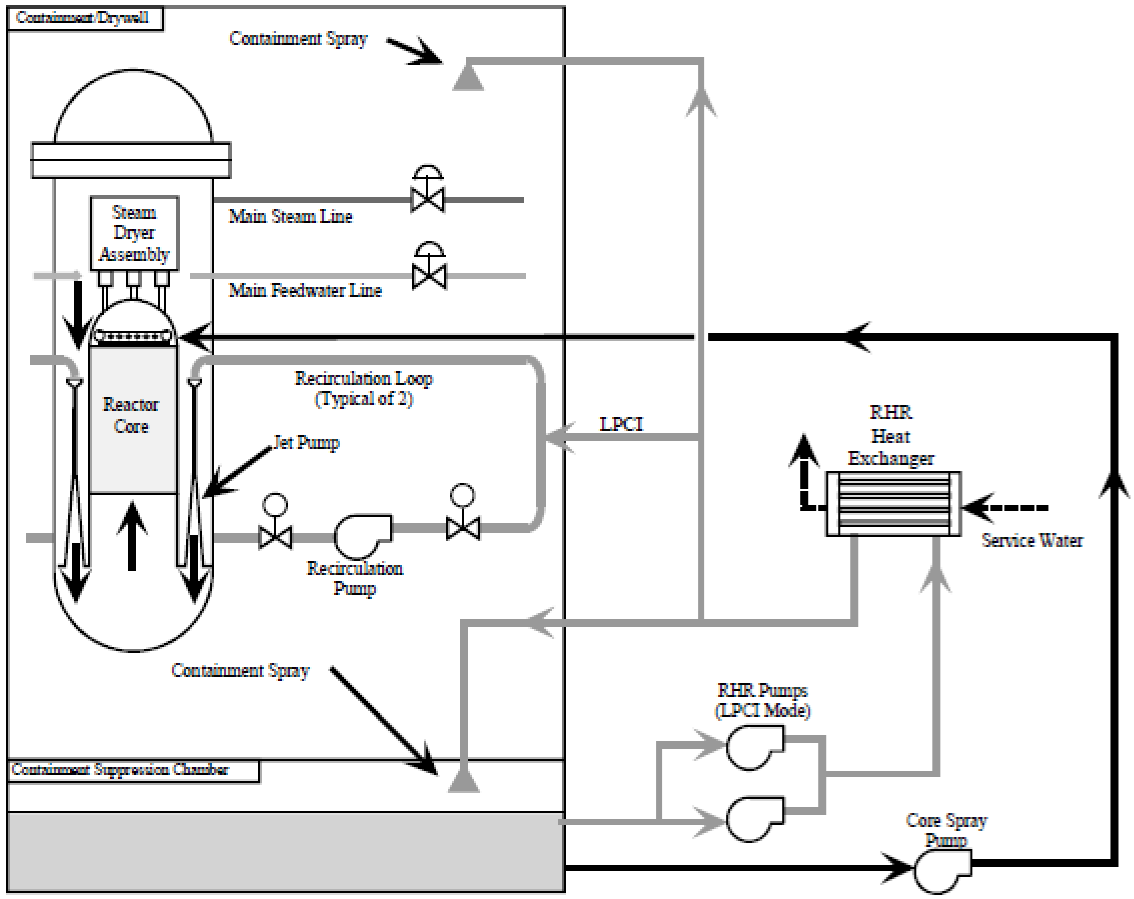


Figure A-4: Low Pressure Coolant Injection and Low Pressure

Core Spray systems (nrc.gov 2012)