Risk-Informed Design Guidance for a Generation-IV Gas-Cooled Fast Reactor Emergency Core Cooling System

by

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B.S., Engineering Physics (2002)
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Submitted to the Department of Nuclear Engineering
in Partial Fulfillment of the Requirements for the Degree of

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ABSTRACT

Fundamental objectives of sustainability, economics, safety and reliability, and proliferation resistance, physical protection and stakeholder relations must be considered during the design of an advanced reactor. However, an advanced reactor's core damage frequency dominates all other considerations at the preliminary stage of reactor design. An iterative four-step methodology to guide the MIT gas-cooled fast reactor emergency core cooling system design through PRA insights was utilized based upon the preliminary stage of the reactor design and activities currently ongoing in the nuclear industry, regulator, and universities regarding advanced reactors. Advanced reactor designs also face an uncertain regulatory environment. It was concluded from the move towards risk-informed regulations of current reactors, that there will be some level of probabilistic insights in the regulations and supporting regulatory documents for advanced, "Generation-IV" nuclear reactors. The four-step methodology is moreover used to help designers analyze designs under potential risk-informed regulations and predict design justifications the regulator will require during the licensing process. The iterative design guidance methodology led to a reduction of the CDF contribution due to a LOCA of over three orders of magnitude from the baseline ECCS design (from 1.19×10^{-5} to 6.48×10^{-8} for the 3x100% loop configuration) and potential ECCS licensing issues were identified. This illustrates the value of formal design guidance based upon PRA.

Thesis Supervisor: George E. Apostolakis
Title: Professor of Nuclear Engineering

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Nomenclature

10CFR50	Code of Federal Regulations, Title 10, Part 50
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CECFP	Conditional Early Containment Failure Probability
DOE	United States Department of Energy
ECCS	Emergency Core Cooling System
GDC	General Design Criteria
GFR	Gas-Cooled Fast Reactor
GIF	Generation-IV International Forum
LERF	Large Early Release Frequency
LOOP	Loss of Offsite Power
LOSP	Loss of Station Power
LWR	Light Water Reactor
NEI	Nuclear Energy Institute
PRA	Probabilistic Risk Assessment
RAW	Risk Achievement Worth
RSS	Reactor Shutdown System
SFC	Single Failure Criterion
SRP	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants
USNRC	United States Nuclear Regulatory Commission

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I. INTRODUCTION

In anticipation of growing energy demands on a world-wide level, the United States Department of Energy (DOE) and the Generation-IV International Forum (GIF) are undertaking efforts to inspire international participation in research to develop innovative nuclear energy systems known as Generation-IV. However, new nuclear reactor concepts face many design and licensing challenges.

The Generation-IV Technology Roadmap [1] established that new nuclear energy systems should be economically competitive in future energy markets while increasing sustainability, safety and reliability, proliferation resistance and physical protection as compared to current nuclear reactors. However, at this preliminary stage of advanced reactor design – where major safety systems and fundamental aspects of reactor designs are still being formulated – the Core Damage Frequency (CDF) was found to dominate all other considerations. CDF is used as a surrogate to measure the risk of a reactor design in Probabilistic Risk Assessment (PRA).

Advanced reactor designs also face an uncertain regulatory environment. Current regulations are deterministic and focused on Light Water Reactors (LWRs). It was surmised from the move towards risk-informed regulations of current reactors, that there will be some level of probabilistic insights in the regulations and supporting regulatory documents for Generation-IV reactors. Regulations will not be promulgated for specific advanced reactor types until a design is submitted to the United States Nuclear Regulatory Commission (USNRC) for licensing.

In light of nuclear industry, regulatory, and university activities, a four-step methodology adapted from [2] was utilized for design guidance of advanced reactors. Step 1 of the methodology is to formulate a bare-bones design (of either the entire reactor or the individual safety system being analyzed). In the second step, unacceptable designs are screened through deterministic and probabilistic screening criteria. Designs that do not pass the screening are iteratively modified until they are deemed acceptable. Step 3 is to quantitatively analyze acceptable designs and iteratively modify to guide the designs based upon PRA insights. In the final step, the designs ranked via PRA are deliberated upon by the decision makers.

The use of the four-step methodology allows for transparent and readily defensible decisions. At this preliminary stage of advanced reactor design, such as the Gas-Cooled Fast Reactor (GFR) of the case study, PRA is used in the methodology as the primary tool for design guidance. Reference [3] illustrates an example of PRA being used to guide the design of a nuclear reactor.

Other important advanced reactor objectives, such as economics, sustainability, stakeholders, reliability, physical protection and proliferation resistance, are considered during the deliberation. Potential advanced reactor regulations are considered during the methodology. This allows for possible points of conflict with current LWR regulations and the prediction of justifications the USNRC may require during the licensing process to be identified and considered. Utilization of the four-step methodology leads to better advanced reactor design decisions in an uncertain regulatory environment.

II. CURRENT ACTIVITIES

Current activities related to the regulation and screening of advanced reactor designs are outlined in this section. These activities were expressed in documents and presentations by the nuclear power industry, regulator, and academia. From the groundwork laid by these current activities, it has been inferred that the four-step methodology, developed in [2] and summarized in Section III, can be used to guide the design of an advanced reactor.

II.A. USNRC Activities

This section provides an overview of both current Emergency Core Cooling System (ECCS, the safety system examined in the case study) regulations and the USNRC's approach to risk-informing Title 10, Part 50 of the Code of Federal Regulations (10 CFR 50). Current ECCS regulations can be found in 10 CFR 50.46, Appendix K of 10 CFR 50, and General Design Criterion (GDC) 35 in Appendix A of 10 CFR 50. For the purposes of the case study, only GDC 35 will be considered. 10 CFR 50.46 and Appendix K are wholly focused on light water reactor specifications and an attempt to adapt these regulations to the Gas-Cooled Fast Reactor would give rise to issues that are outside of the scope and the intent of this research. Reference [4] contains a feasibility study of risk-informing 10 CFR 50.46, Appendix K, and GDC 35. It provides a discussion of the issues related to risk-informing ECCS regulations for LWRs and some insights into the issues related to applying these regulations to advanced reactors can be inferred from reference [4].

GDC 35 can be considered in the following framework. Current regulations for the licensing of nuclear power plants in the United States demand that applicants demonstrate that their design presents no 'undue risk to the health and safety of the public.' 10 CFR 50 establishes the minimum design requirements for water-cooled reactors in Appendix A, "General Design Criteria for Nuclear Power Plants" [5]. These General Design Criteria (GDC) require reactors to be designed with sufficient margin to assure safety against postulated accident sequences [5]. "Postulated accidents" are also known as Design Basis Accidents (DBAs). Uncertainties regarding the safety of a new reactor design are addressed by protecting against DBAs and by meeting or exceeding the GDC.

10 CFR 50.34 requires applications for construction permits to include the principal design criteria. Appendix A of 10CFR50 states that "the principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." [5] Appendix A states that the "General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants" [5].

The USNRC defines a design basis accident as "A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to assure public health and safety" [6]. DBAs are intended to 'bound' sets of accident initiators.

Our case study involves three ECCS designs for a GFR currently under development at MIT. GDC 35 was used as a deterministic screening criterion for these design options as it is the General Design Criteria that addresses the ECCS system.

Criterion 35 of 10CFR50, Appendix A reads as follows:

> "Criterion 35-Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure." [5].

As written and implemented, GDC 35 treats the double-ended break of the largest pipe in the reactor coolant system in addition to offsite power being unavailable and a single failure in the most critical place as the DBA for the ECCS [7].

The approach towards risk-informing 10 CFR 50 has been dubbed "Option 3" as it is consistent with the third option towards the risk-informing of 10 CFR 50 outlined in reference [8]. The regulations in 10 CFR 50 are specific to light water reactors. However, this approach is not reactor-specific and has been proposed in [9] as a methodology for creating risk-informed advanced reactor regulations, such as those that will need to be created for Gas-Cooled Fast Reactors. The USNRC has proposed to integrate three concepts in developing the framework for risk-informing the technical requirements of advanced reactor regulations. The combination of these concepts by the Nuclear Regulatory Commission is shown in Figure 2-1.

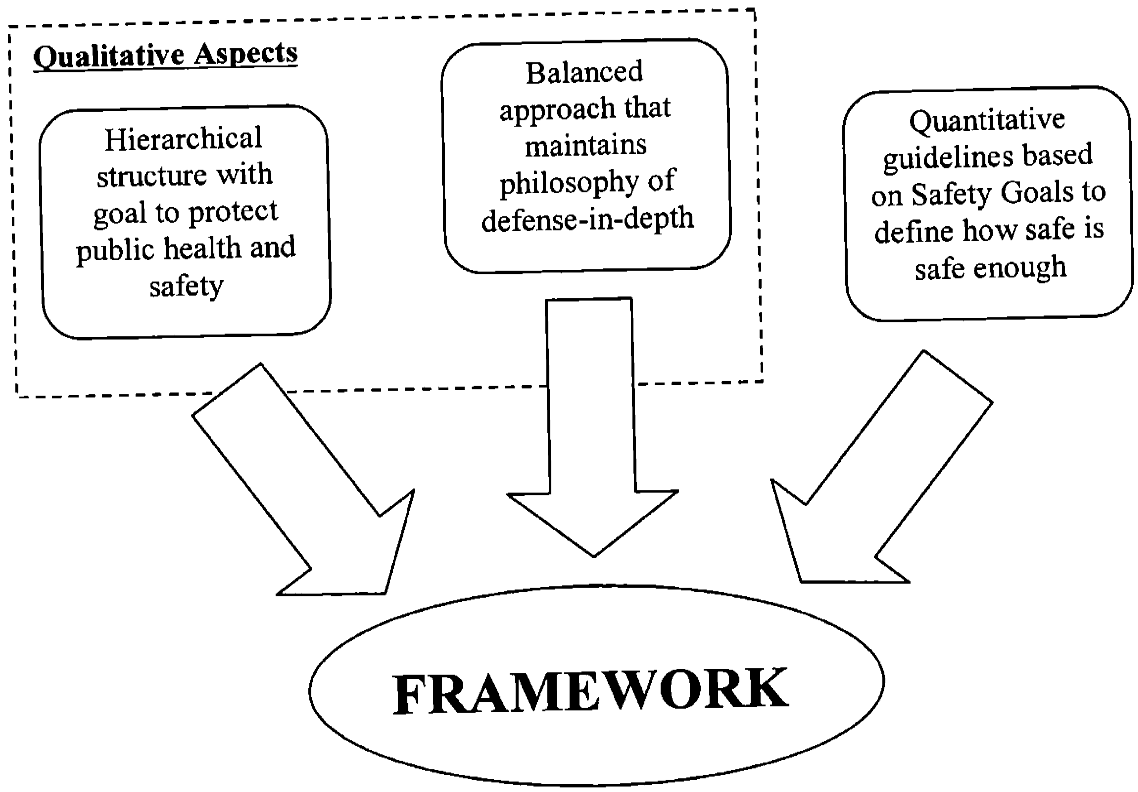

Figure 2-1. USNRC Risk-Informed Regulations Framework Concepts [4]

The three concepts the USNRC utilized in creating a framework for risk-informed regulations were – a hierarchical framework structure with the goal of protecting the public health and safety, a balanced regulatory approach that maintains the philosophy of defense-in-depth, and quantitative guidelines based on Safety Goals to define how safe is safe enough for advanced nuclear power plants. The USNRC's hierarchical framework structure is illustrated in Figure 2-2. The risk-informed regulatory framework is divided up into four hierarchical levels. Each lower level in the Framework Hierarchy describes a method to meet the next level up in the framework. The top level in the Nuclear

Regulatory Commission's framework, or Level 1, is the goal of protecting the public health or safety. Level 2 are the USNRC Reactor Inspection and Oversight Program cornerstones for safe nuclear reactor operation that are needed to meet the top level goal. Level 3 are the strategies for realizing the cornerstones and Level 4 are the tactics used in creating and implementing the regulations.

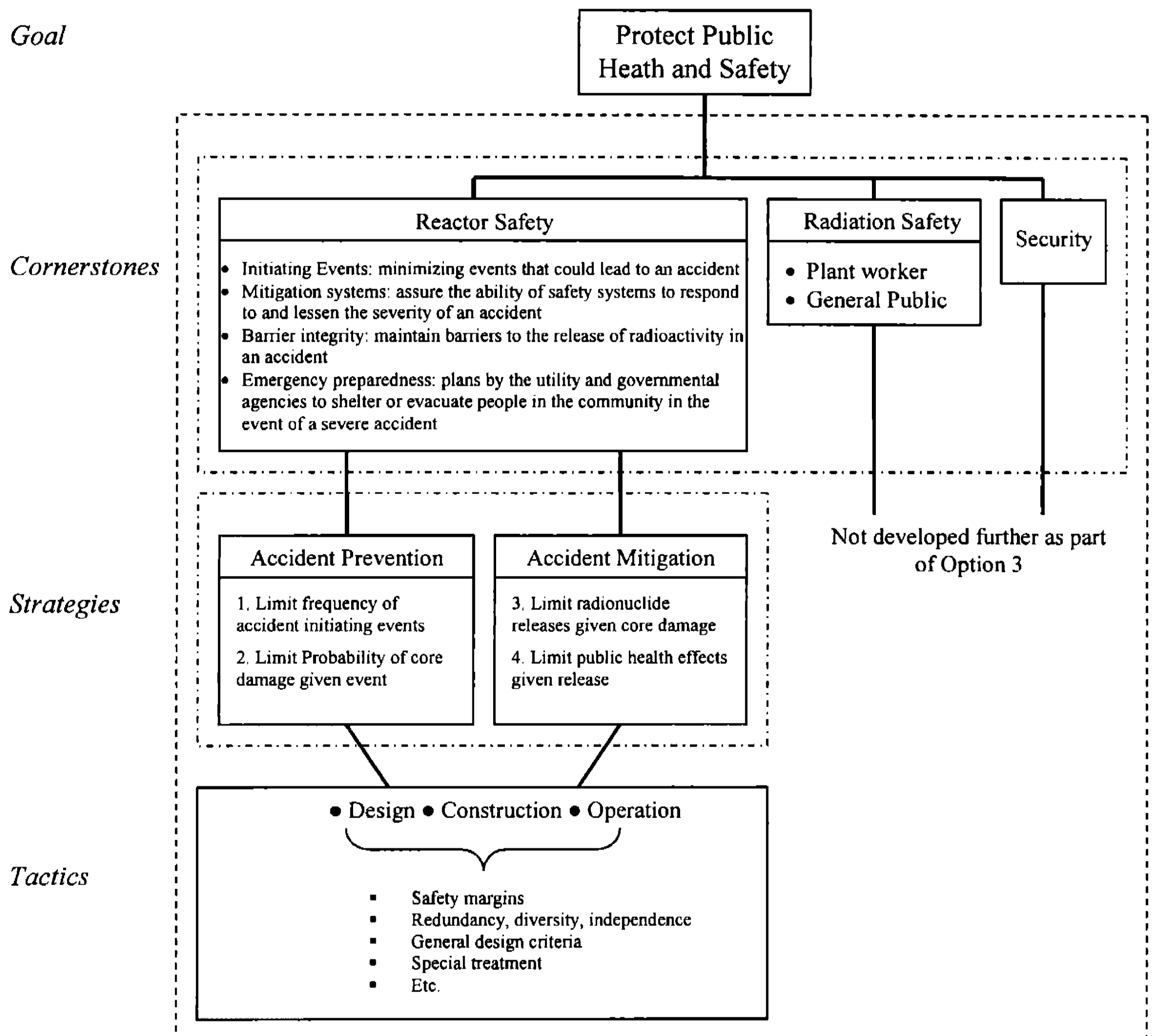

Figure 2-2. USNRC Risk-Informed Regulations Framework Hierarchy [4]

Figure 2-2 illustrates the USNRC's commitment to the defense-in-depth philosophy. This point is important to keep in mind when guiding the design of advanced reactors. If a potential advanced reactor design is scrutinized while keeping the viewpoint of defense-in-depth in mind, it is possible to predict questions that may arise and design justifications that may by required during a review of the design by the regulator.

As can be seen from Figure 2-2, the USNRC applies a defense-in-depth strategy when formulating and implementing regulations at multiple levels in the hierarchy. At Level 3 of the hierarchy, accident prevention and mitigation strategies are applied. When formulating a regulation, the USNRC looks to first limit the frequency of accident initiating events. Second, the probability of core damage given an initiating event is limited. The third strategy is to limit radionuclide releases during core damage. Finally, the USNRC considers how to limit the public health effects due to a core damage accident when making and employing a regulation under the risk-informed framework.

At the tactics level of the framework hierarchy, risk-informed regulations – the most likely form of advanced reactor regulations – are to be created and implemented so as to meet several defense in depth principles. These include (but are not limited to): assuring that a reasonable balance is provided among the high level strategies, avoiding over-reliance on programmatic activities to compensate for weakness in plant design and not degrading the independence of barriers. The framework hierarchy for current reactors in reference [4] also looks to maintain the defense-in-depth objectives of the GDC in Appendix A of 10 CFR 50. The GDC were created for light-water reactors, however the introduction states that the GDC are "considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units."

As can be seen from Figure 2-2, the USNRC did not develop further the Radiation Safety and Security cornerstones as a part of the risk-informed framework. During the formal design guidance analysis, these cornerstones of safe reactor operation were not taken into account. Any such considerations of these cornerstones would be taken into account during the deliberation phase of the analysis, as is explained in Section III.

To incorporate risk-insights into the framework illustrated in Figure 2-2, the USNRC included surrogate risk guidelines associated with the Level 3 in Figure 2-2, strategies for risk-informed technical requirements. As noted in reference [4], "The quantitative guidelines are not proposed regulatory requirements and will not appear in risk-informed regulations; however, they may appear in implementing documents such as regulatory guides when probabilistic analyses are deemed appropriate." While the NRC has stated that surrogate risk guidelines will not be in the advanced reactor regulations, they will be used in this research as probabilistic screening criteria for GFR ECCS design options since they will likely appear in regulatory support documents, i.e. Regulatory Guides.

The surrogate risk guidelines proposed by the USNRC for light water reactor regulations are illustrated in Figure 2-3 [10]. As can be seen, the structure of these guidelines is consistent with the defense-in-depth approach towards accident prevention and mitigation. A Level 1 PRA of the advanced reactor design must result in mean CDF and LERF values equal to or lower than the surrogate risk guidelines of 10^{-4} and 10^{-5} respectively. Individual sequences cannot contribute more than 10% to the total CDF.

Further, initiators are broken down into three categories: anticipated initiators, infrequent initiators, and rare initiators. Their frequencies and corresponding conditional core damage probabilities are shown in Fig. 2-3. As compared to GDC 35, the surrogate risk guidelines are more complete as uncertainties are accounted through the use of mean CDF and LERF values and common-cause failures are incorporated into the probabilistic accident sequence analysis. A discussion of the framework, intentions, and scope of these quantitative guidelines can be found in [10].

	(1) Prevention-Mitigation Assessment: Consider the Strategies in Pairs			
	Prevent		**Mitigate**	
	Core Damage Frequency $\leq 10^{-4}$/year		Conditional Probability of Early Containment Failure** $\leq 10^{-1}$	
	(2) Initiator-Defense Assessment: Consider the Strategies Individually (Preferred)			
	Limit the Frequency of Accident Initiating Events (Initiators)	Limit the Probability of Core Damage Given Accident Initiation	Limit Radionuclide Release During Core Damage Accidents	Limit Public Health Effects Due to Core Damage Accidents
	Initiator Frequency	Conditional Core Damage Probability	Conditional Early Containment Failure Probability**	Conditional Individual Fatality Probability
Anticipated Initiators	≤ 1/year	$\leq 10^{-4}$	$\leq 10^{-1}$	*
Infrequent Initiators	$\leq 10^{-2}$/year	$\leq 10^{-2}$	$\leq 10^{-1}$	*
Rare Initiators	$\leq 10^{-6}$/year	≤ 1	≤ 1	*

Notes:
The product across each row gives LERF < 10^{-5}/year. Responding systems and procedures are not designed for rare events. When applying the quantitative guidelines in this figure, in general, no individual sequence should contribute more than 10% of the value listed.
* No quantitative guideline propose, using LERF as a surrogate.
** This strategy does not imply that risks associated with late containment failure can or will be ignored. Potential causes of late containment failure and associated mechanisms for radionuclide removal prior to containment failure will be considered. A quantitative guideline of ≤ 0.1 is proposed for the probability of a late large release given a core damage accident

Figure 2-3. Surrogate Risk Guidelines [10]

In Reference [9] there was a suggestion that the quantitative guidelines may be made more stringent for advanced reactor regulations. The surrogate risk guidelines of 10^{-5} for CDF and 10^{-6} for LERF were suggested for advanced reactors. However, in light of commentary from the nuclear power industry and further communication from USNRC staff [11], the surrogate risk guidelines illustrated in Figure 2-3 appear to be the most likely guidelines for advanced reactor risk-informed technical requirements.

II.B. Nuclear Industry Activities

In 2002 the Nuclear Energy Institute (NEI) published a white paper [12] describing a new and optional risk-informed, performance-based regulatory framework for commercial nuclear power reactors that they developed. This regulatory framework focused mainly on technical and operational requirements. The white paper included a complete set of regulations for a new Part to 10 CFR, Part 53. The proposed Part 53 was intended to be an optional alternative to 10 CFR 50. As stated by the NEI, "the intent (of 10 CFR 53) is to provide the same standards of protection for the public and environment as current regulations, while providing for a more cost effective, efficient and safety focused means of licensing and regulating commercial nuclear power reactors."

In the white paper, a defense-in-depth design process under their risk-informed, performance-based framework is outlined. This process is illustrated in Figure 2-4. Starting from an initial reactor design, an iterative process is employed to address key uncertainties in the design. First, a PRA of the design is performed. Upon completion of the initial design PRA, the design is adapted, as needed, to meet risk acceptance criteria defined in the Part 53 regulations. Next, the defense-in-depth prospects are considered to balance any unacceptable risk uncertainties identified in the design.

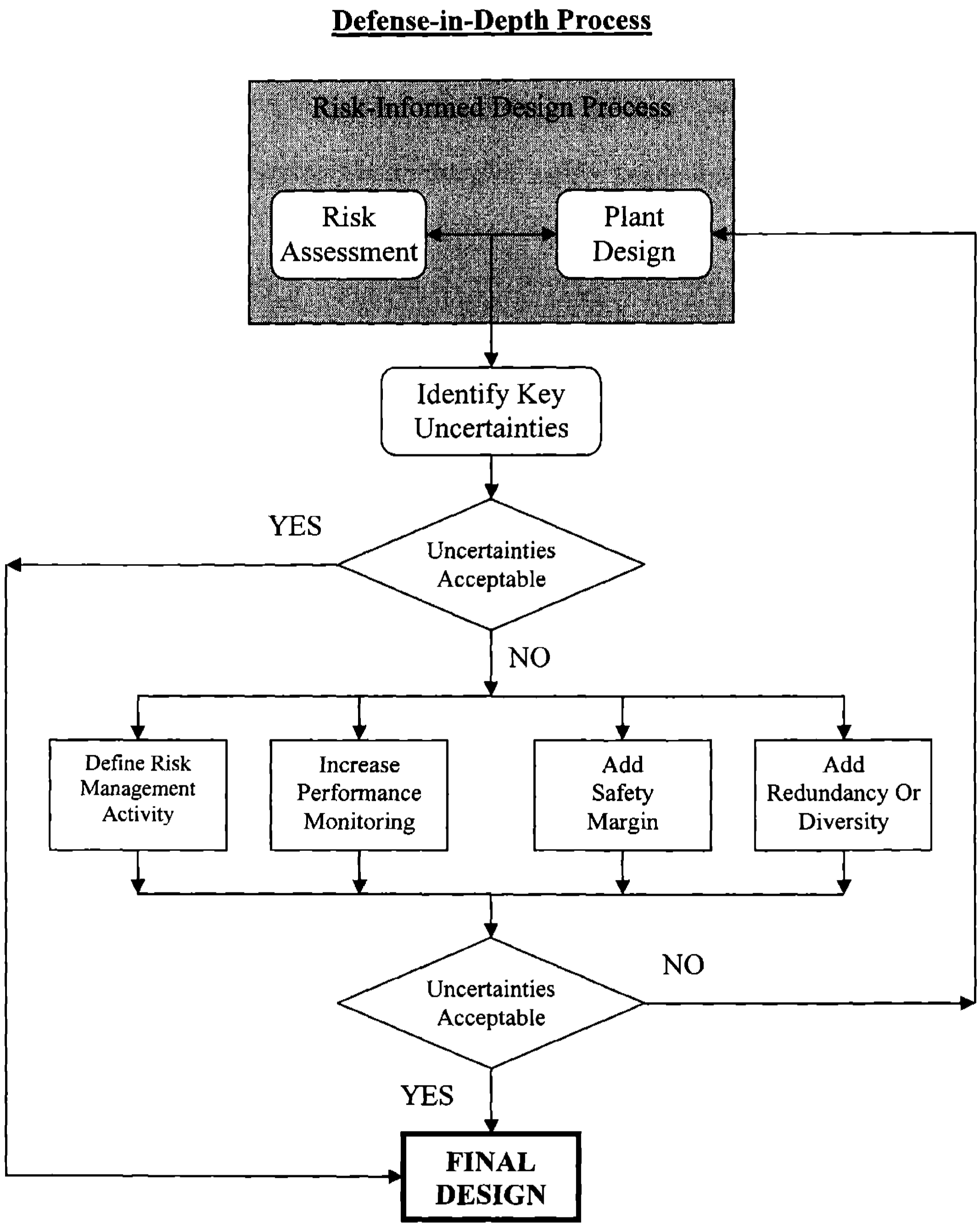

Figure 2-4. NEI Defense-in-Depth Design Application for Part 53 [12]

Under the NEI's framework, there are four defense-in-depth options – to define risk management activity, to increase performance monitoring, to add safety margin, and to add redundancy or diversity. The PRA is modified to reflect all alterations in the plant operations and design resulting from defense-in-depth addition. The process is iterated until the uncertainties are considered to be acceptable for the commercial nuclear power plant design.

II.C University Activities

Research is in progress and has been presented regarding risk-informed and risk-based regulations and reactor design at MIT. A framework for risk-based regulation and design is illustrated in Figure 2-5 [13]. Similar to the USNRC framework hierarchy, the MIT framework for risk-informed regulations has a top-level goal of "Public Health and Safety as a Result of Civilian Reactor Operation." In the MIT framework hierarchy, the top level goal is supported and formulated in increasing fine detail descending through the approach, PRA strategies, tactics, and finally implementation for regulation and design.

The crux of the MIT framework is the use of PRA as the principal decision support tool by both the designer and the regulator. The PRA would be used as a method for expressing the viewpoints of the reactor designer and licenser. Under the MIT framework, the PRA is a Bayesian decision tool that explicitly incorporates uncertainties into the reactor licensing process.

For the MIT framework to be effective, two key elements are required. First, the scope of the PRA must include all operating phenomena of the reactor. This allows for the discussion of all reactor design issues to take place in the context of the PRA. This is important because where the models and data in the PRA cannot be supported by deterministic analytical results; the subjective judgments regarding the PRA of the design are explicit to the regulator and designer. Second, the regulations must be promulgated in terms of acceptable levels of unavailability of safety functions. This allows the PRA to be utilized as a Bayesian decision tool.

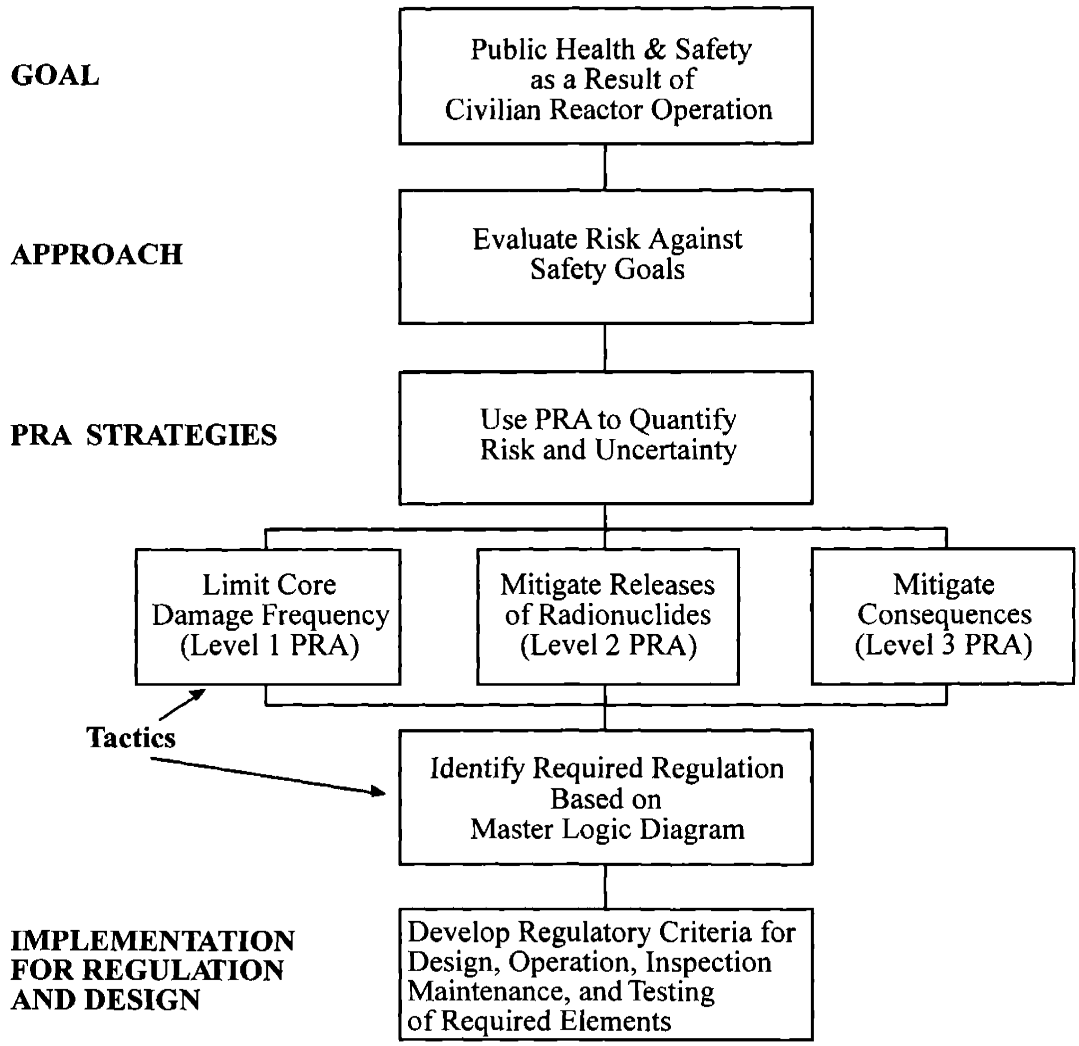

Figure 2-5. MIT Framework for Risk-Informed Regulation and Design [13]

Under the MIT framework, a risk-driven generic design methodology was outlined in reference [13]. This process is illustrated in Figure 2-6. The process begins with a bare-bones plant design. Next, deterministic analyses of the plant are carried out to identify possible failure modes. A PRA of the bare-bones plant design is then performed to identify the dominant failure modes. The results of the PRA are compared against surrogate risk guidelines, i.e. the USNRC's guidelines outlined in Section II.A. Safety features are then added to the bare-bones nuclear power plant design. A PRA of the plant design with the added safety features for mitigation or prevention of the dominant failure modes is then performed. The results of this PRA are then evaluated against the surrogate risk guidelines. This process is iterated until all of the acceptability criteria are met, resulting in a generic risk-driven plant design.

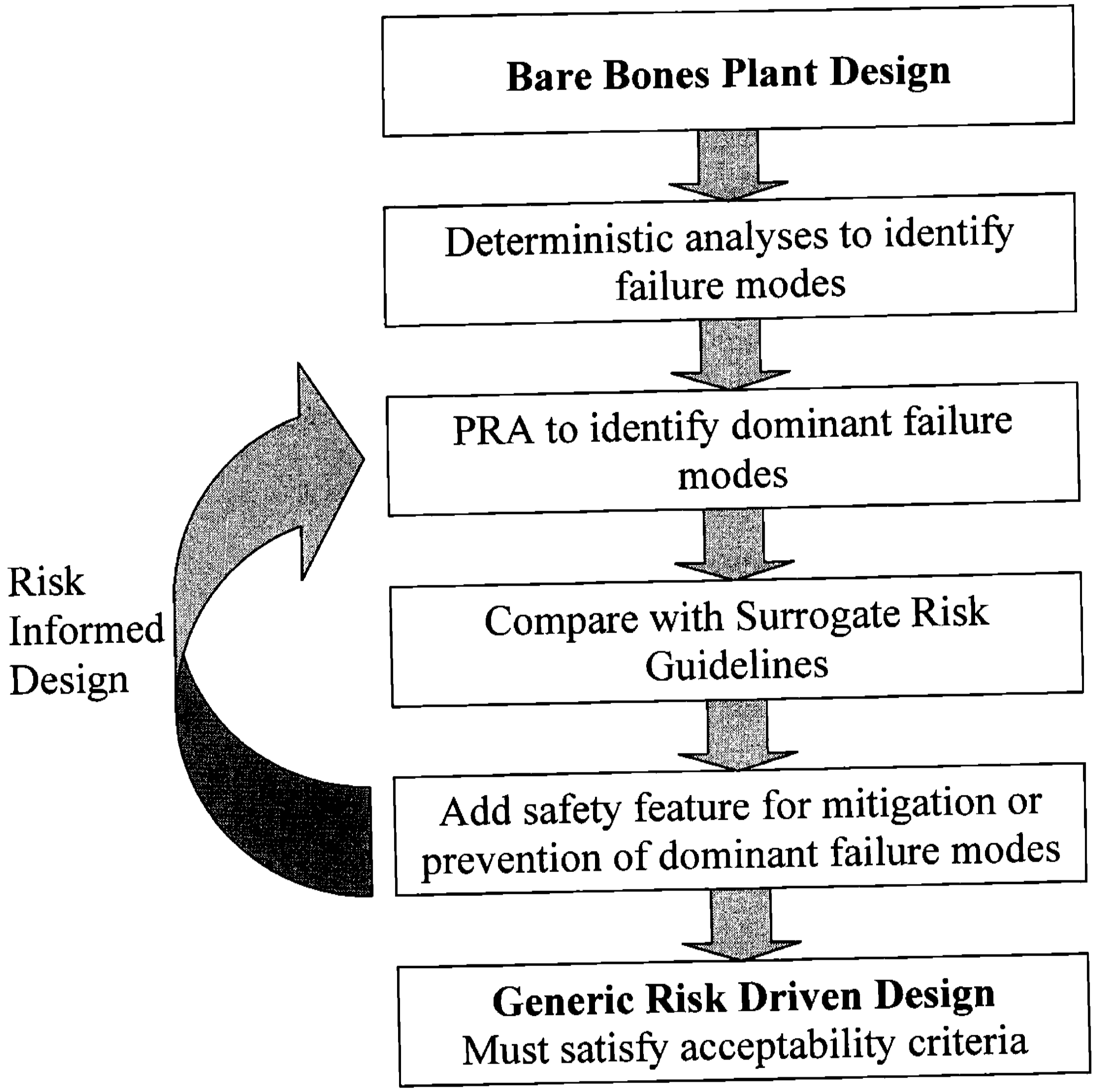

Figure 2-6. Bare-Bones Plant Design Methodology [13]

III. METHODOLOGY OVERVIEW

Based upon the current activities related to the design and regulation of advanced reactors, it can be inferred that the methodology presented in reference [2] can be used to guide the design of an advanced non-light water nuclear reactor. Guiding the design of the Generation-IV GFR is made more difficult because regulations for future advanced reactors have not been developed. To better understand how to make better and readily defensible plant design decisions in such an uncertain regulatory construct, the four-step decision making methodology presented in [2] was tailored for use by advanced reactor designers. It is based upon the assumption that advanced reactor regulations will be risk-informed, as the current activities in Section II indicate. This methodology will be used in the case study to for selecting plant design options.

The decision-making methodology presented in [2] is illustrated in Figure 3-1. This methodology consists of four steps. First, the decision options are formulated for the decision problem. This step should be as inclusive as possible. Second, decision options that are not deemed acceptable are removed. Decision options are removed to reduce the burden of analyzing design options that are clearly unacceptable. Third, the remaining decision options are ranked using multi-attribute utility theory (MAUT). MAUT helps decision makers to illustrate uncertainties which otherwise might not be quantified. It also allows the decision maker to quantify the relative desirability of the various outcomes. By doing so, the decision maker can better understand the degree to which one decision option is preferred over another. MAUT establishes formal rules for ranking the decision options. This helps to ensure that the decision-making process stays consistent when faced with a multitude of decisions. A more thorough discussion of MAUT can be found in reference [14]. Fourth, the decision options ranked via MAUT are deliberated upon by the decision makers. The deliberation is necessary because the decision analysis for the design options might not capture everything that the decision makers deem important and may not be comprehensive.

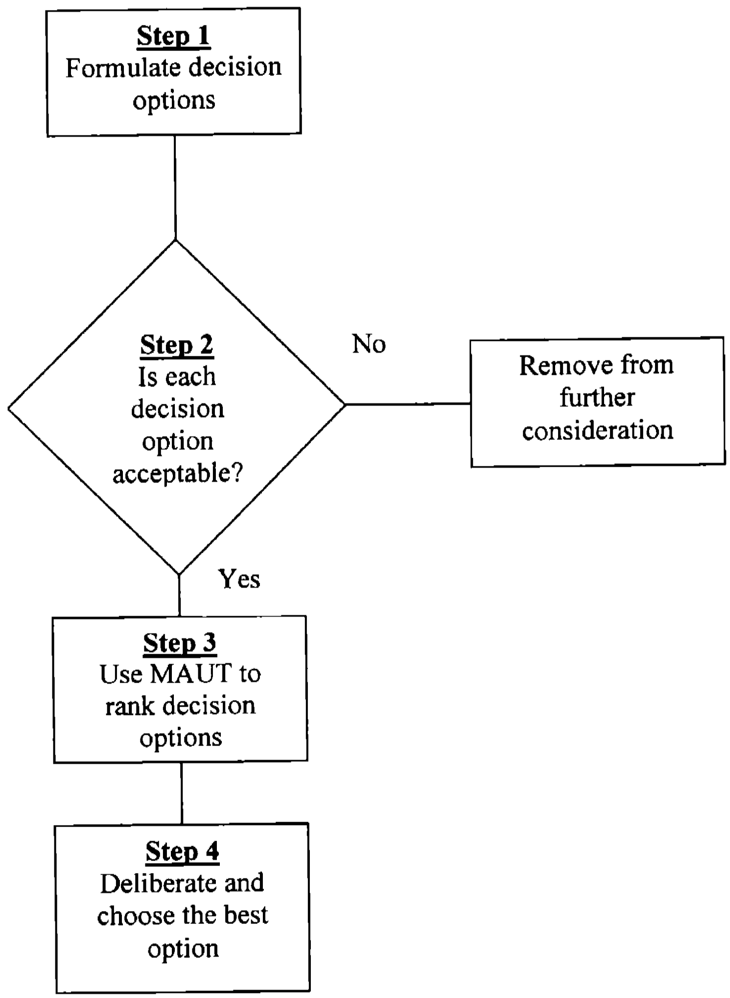

Figure 3-1. Decision Making Methodology Presented in Reference [2]

The designer decision-making methodology for selecting plant design options under risk-informed regulations is illustrated in Figure 3-2. As can be seen in a comparison of Figure 3-1 and 3-2, this methodology is a modified version of the methodology presented in [2]. The modifications were necessary, as is explained below, to account for certain characteristics of advanced reactor design and the preliminary stage of design of the case study reactor. This methodology can be applied to many different aspects of the plant design. For instance, the methodology could be applied to select the best major safety system design, or whether the plant should have a direct or indirect power cycle, or the overall plant design as a whole. The case study presented in this paper will examine Emergency Core Cooling System (ECCS) designs. The goal of the methodology is to help decision makers choose better advanced reactor design options.

Step 1 is to formulate an initial design. This is typically accomplished by the plant designers using engineering judgment and intuition. For the case study, a Bare-bones ECCS design was formulated in step 1 by the GFR designers to be analyzed in the methodology.

Step 2 is to screen out unacceptable designs through deterministic and probabilistic criteria. For nuclear power plants, the major sources for screening unacceptable designs are regulations and supporting regulatory documents. Current nuclear power plant safety regulations are solely comprised of deterministic screening criteria. It is assumed that a light-water reactor that does not violate any of the screening criteria is acceptably safe. However, current regulations never quantify "how safe is safe enough."

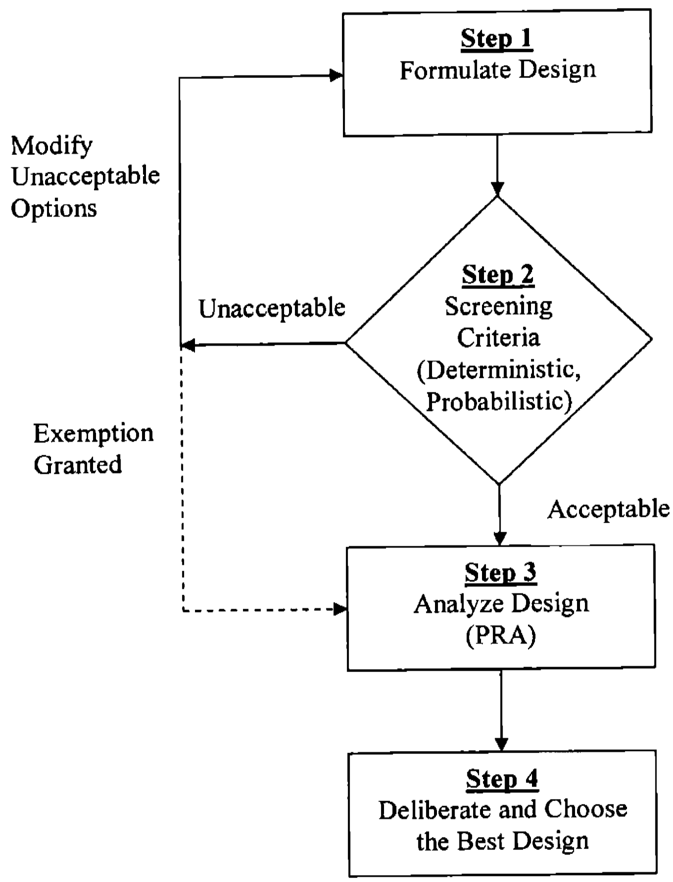

Figure 3-2. Iterative Design Guidance Methodology

Designs that do not pass the screening criteria are deemed unacceptable. Under current regulations, proposed advanced reactor designs that do not meet the GDC or defend against DBAs would be screened out from further consideration unless the designer applied for an exemption. An example of a US Nuclear Regulatory Commission (USNRC) staff assessment of an advanced reactor applicant's licensing approach and application for exemptions based upon current regulations can be found in [15].

An example of criteria used to determine an unacceptable design under current regulations would be the single failure criterion (SFC) established in Appendix A of 10CFR50 [5]. This criterion states that a single failure of active components, including valves and pumps, should not lead to the failure of a safety system [16,17]. However, the single failure criterion does not apply to passive components [18]. So, for instance, a single loop "passive" ECCS that includes one check valve, would violate the single failure criterion. However, it is possible to apply for an exemption to have the check valve deemed passive. Therefore, such a decision option would have to be modified to not violate the single failure criterion or an exemption would need to be applied for before being analyzed in step three.

While the method for screening designs in step 2 is exactly the same under current or risk-informed regulations, the criteria that are used to screen design options are greatly reduced in risk-informed regulations. Many deterministic screening criteria such as the single failure criterion of GDC 17, 21, 24, 34, 35, 38, 41, and 44 are likely to be replaced in risk-informed regulations [7]. Also, design basis accidents that are shown to contribute little to a plant's total core damage frequency are candidates for replacement with a reliability goal in risk-informed regulations.

One area that will not be changed in risk-informed regulations is the requirement for a number of diverse initiator prevention and mitigation systems such as the ECCS, the shutdown cooling system (SCS), on/off-site power requirements, and the reactor shutdown system (RSS). Plant design options that, for example, do not include an ECCS in the overall plant design would be judged unacceptable in step 2 under risk-informed regulations.

The other part of Step 2 is to screen out unacceptable designs through probabilistic screening criteria. The surrogate risk guidelines outlined in Section II.A will be used in the case study as probabilistic screening criteria.

The probabilistic screening in Step 2 requires a PRA to be performed. Quantitative reliability numbers will be found for the plant design options in Step 2. Any option whose risk exceeds the risk guidelines of the risk-informed regulations will be deemed unacceptable. These plant design options can either be removed from further consideration or modified in an attempt to meet the reliability guidelines of the regulations. Only the surrogate risk guideline of Core Damage Frequency (CDF) will be used in the case study. It is currently impossible to calculate the conditional containment failure probability, as the containment has yet to be designed for the GFR of the case study.

Starting from a Bare-bones design, in Step 1, the design is screened in Step 2 and modified iteratively until it is deemed acceptable. It may be desirable to modify a design deemed acceptable in Step 2 due to some characteristic of the design that the analyst may see. Iterating new designs off of a design that has been deemed acceptable in Step 2 creates design options to be analyzed in Step 3 and Deliberated upon in Step 4.

Step 3 is to quantitatively analyze designs. MAUT is used in Step 3 of reference [2] to quantitatively analyze designs. However, the reactor used in the case study is at such an early stage of development that many of the objectives that would be analyzed via MAUT are extremely difficult if not impossible to quantify. In the course of this research, a MAUT analysis was attempted for the ECCS designs of the case study. It was confirmed that many of the objectives in the ECCS value tree could not be quantified. Further, it was found that the ECCS design CDF contribution to the total CDF dominated all other considerations in a MAUT at this preliminary stage of GFR ECCS design.

PRA is therefore used as the design analysis tool in the modified methodology. The PRA [19-21] is a primary decision support tool due to its ability to integrate all of the elements of system performance and to represent the uncertainties in the results and its transparency for the safety regulators [22]. Any considerations beyond those captured in the PRA of the ECCS and supporting systems were considered during the final step of the methodology.

In the fourth and final step, the designs ranked via PRA are deliberated upon by the decision makers. The deliberation is necessary because the PRA for the design options might not capture everything that the decision makers deem important and may not be comprehensive. In the event that the decision makers are not thoroughly satisfied with any of the plant options or in the event that the analysis suggested possible improvements to the design options the methodology can be iterated until the decision makers are satisfied.

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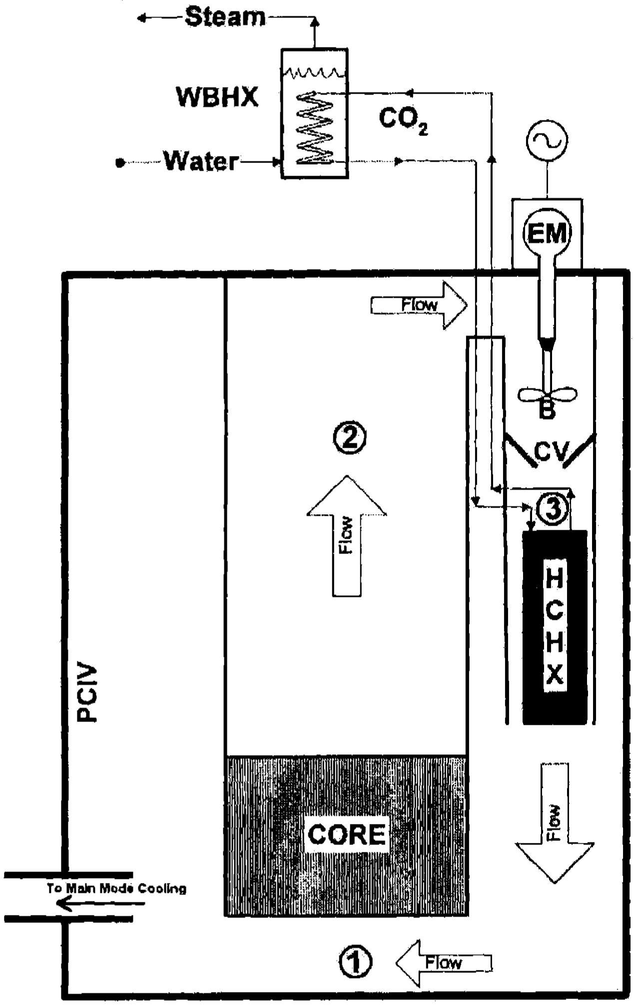
IV. CASE STUDY

The case study involves the design of a Generation-IV Gas-Cooled Fast Reactor (GFR) [2] currently under development at MIT. A bare-bones emergency core cooling system design is analyzed and modified iteratively using Steps 1-3 of the four-step methodology outlined in Section II.

IV.A. Step 1: Formulation of the Emergency Core Cooling System Design

Step 1 of the four-step methodology is to formulate a bare-bones ECCS design. For the case study, the initial design was produced by the GFR design team. This is a group of engineers whose expertise lies in such fields as reactor physics and thermohydraulics. These designs were developed based upon engineering judgment and intuition and then given to the GFR PRA group to analyze.

Figure 4-1 illustrates the bare-bones emergency core cooling system design. The ECCS is intended to prevent a Loss of Coolant Accident (LOCA) initiating event from leading to core damage. For the initial design, the most likely method of losing the reactor coolant (CO_2) would be a pipe break in the main-mode cooling. The reactor coolant was designed to be pressurized in the MIT GFR design to approximately 200 atmospheres (atm) during normal operation. A pipe break in the main mode cooling would allow the coolant to escape through the main mode cooling loop pipe (pictured in Fig. 4-1). This would cause the reactor to depressurize and the main mode cooling system to fail. At this point, the emergency core cooling system would be required to operate to prevent core damage.



Legend
EM = Electric Motor
CV = Check Valve
B = Blower
HCHX = Heatric Heat Exchanger
WBHX = Water-Boiler Heat Exchanger
PCIV = Prestressed Cast Iron Vessel

Not to Scale

Figure 4-1. Bare-Bones ECCS Design

Core cooling is accomplished by the blower (labeled B) moving the primary coolant (CO_2) through the core (point 1 to point 2), past a check valve that prevents backflow during non-emergency operation (point 2 to point 3), and then through the Heatric Heat Exchanger [23] (point 3 to point 1). Heat is transferred from the primary CO_2 to a secondary CO_2 loop in the Heatric heat exchanger. The secondary CO_2 flows via natural convection to the water-boiler heat exchanger (labeled WBHX). Heat is then transferred from the secondary CO_2 to water, which boils to form steam and is rejected to the ultimate heat sink.

Beyond physical failures of any of the components described so far, critical concerns are supplying AC power to the blower and DC power to the instrumentation and control systems. The bare-bones ECCS design provided only offsite power to the blower. Instrumentation and control is powered by a single DC battery in the bare-bones design. The design of the ECCS (including the power supply systems) will be modified from the bare-bones design based upon the screening criteria in Step 2 and the PRA results in Step 3 of the four-step methodology.

IV.B Step 2: Screening Criteria for the ECCS

Step 2 is to screen out unacceptable plant design options through deterministic and probabilistic criteria. For nuclear power plants, the major sources for screening unacceptable options are regulations and supporting regulatory documents. As was outlined in Section II.A, the deterministic screening criteria used in our case study will be General Design Criterion 35 and the ECCS DBA of the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP) [23] of current light-water reactor regulations. The CDF surrogate risk guidelines, also outlined in Section II.A, will be used as the probabilistic screening criteria in Step 2.

The applicable ECCS requirements of GDC 35 relating to the MIT GFR are illustrated in Figure 4-2. Abundant Emergency Core Cooling, provided by the ECCS, is required by GDC 35 to be available in the event of a single failure of an ECCS component and either the loss of onsite or offsite power.

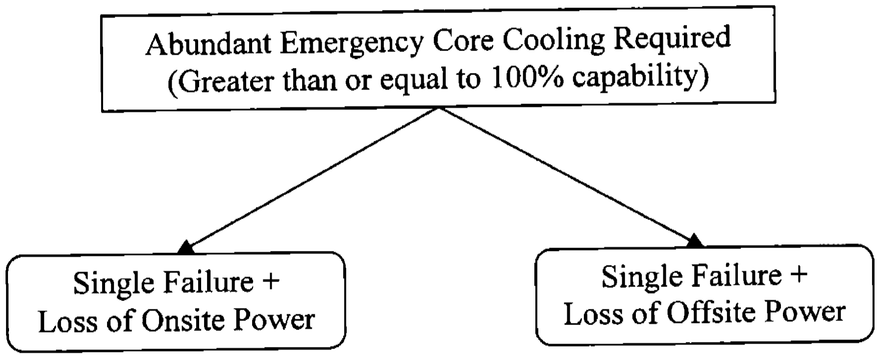


Figure 4-2. GDC 35 Requirements

The deterministic design basis accident assumptions for the ECCS are illustrated in Figure 4-3. According to the SRP, an Emergency Core Cooling System must be designed to withstand the following postulated LOCA – a double-ended break of the largest reactor coolant line, the concurrent loss of offsite power, and a single failure of an active ECCS component in the worst possible place.

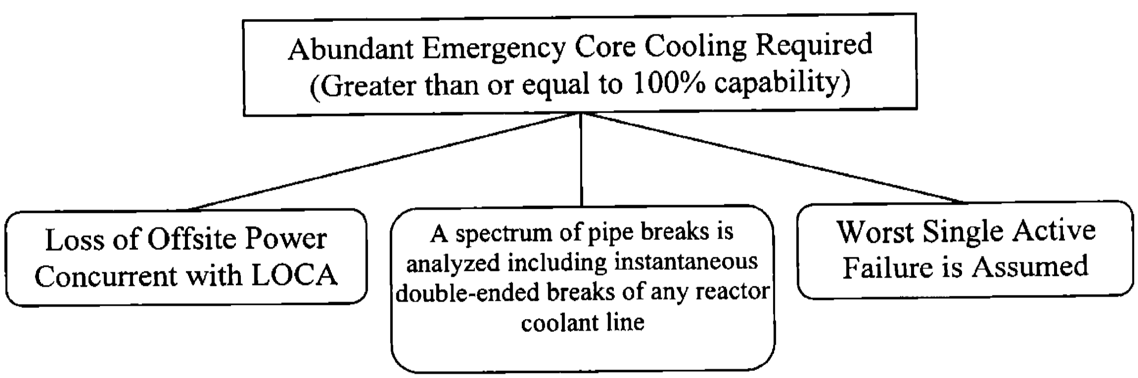


Figure 4-3. Standard Review Plan ECCS Design Basis Accident Assumptions

The probabilistic screening criteria are based upon the surrogate risk guidelines illustrated in Figure 2-3 (Section II.A). To develop the screening criteria, information on the LOCAs (the initiating events emergency core cooling systems are designed to mitigate) was needed. In light-water reactors, LOCAs are divided into at least three categories: Large, Medium, and Small break LOCAs. These categories were created to reflect the consequences and frequencies of the range of break sizes. The large break LOCA has the most adverse consequences, but the lowest frequency of occurrence while the small break LOCA has the lowest consequences and the highest frequency of occurrence. However, the main mode-cooling system for the MIT GFR has not yet been designed; therefore the range of possible pipe break sizes is not known. This implies that the consequences and frequencies of a pipe break had to be hypothesized. Because of the gas coolant, the frequencies of large, medium and small pipe break LOCAs were summed and the consequences were conservatively assumed to be the most severe since any size pipe break will lead to a relatively rapid depressurization as compared to light-water reactors. Also, it is possible for a LOCA to occur due to reactor vessel rupture. This accident sequence was not considered in the PRA due to its low likelihood of occurrence relative to a main-mode cooling pipe break and since the reactor vessel design or material has not been decided upon.

The LOCA frequencies were taken from the AP-1000 PRA. The frequencies of small, medium, and large, LOCAs were $5x10^{-4}$, $4x10^{-5}$, $5x10^{-6}$ (per reactor year) respectively. The summation of the range of pipe breaks leads to the use of $5.45x10^{-4}$ (per reactor year) as the LOCA frequency for the analysis of ECCS designs. It is recognized that the AP-1000 pipe failure data may not be the optimal data to use for a gas-cooled reactor. However, pipe failure data for the AP-1000 are much more current than any previous Gas-Cooled Fast Reactor design and reflect the state-of-the-art in pipe materials and manufacturing. Also, the MIT GFR is at such an early stage of design that the use of generic failure data is warranted until more details about the plant are developed.

A LOCA frequency of 5.45×10^{-4} falls under the "Infrequent Initiator" (initiator frequency per reactor year is less than 10^{-2} and greater than 10^{-5}) category of the surrogate risk guidelines (Figure 2-3). A Conditional Core Damage Probability (CCDP) of $\leq 10^{-2}$ is required for infrequent initiators. Therefore, the probability the ECCS fails to provide adequate core cooling, leading to core damage, must be less than 10^{-2} given a LOCA. Another screening criterion resulting from the surrogate risk guidelines outlined in Section II.A is that no individual sequence should contribute more than 10% of total CDF. Assuming a baseline CDF of 10^{-4} per reactor year (the largest acceptable CDF under the surrogate risk guidelines), implies that an individual sequence contribution to CDF can be no more than 10^{-5} per reactor year. In the case of the MIT GFR LOCA accident sequence, this criterion is automatically met if the CCDP guideline is met. This is because a LOCA frequency of 5.45×10^{-4} and the maximum allowed CCDP of 10^{-2} leads to a CDF of 5.45×10^{-6} per reactor year.

A final probabilistic screening criterion resultant from the surrogate risk guidelines outlined in Section II.A is the Conditional Early Containment Failure Probability (CECFP). The surrogate risk guidelines give a CECFP requirement of $\leq 10^{-1}$. Multiplying the CECFP to the maximum allowed CDF of 5.45×10^{-6} per reactor year leads to a Large Early Release Frequency (LERF) requirement of less than 5.45×10^{-7} per reactor year for the LOCA accident sequence of the MIT GFR. LERF will not be used as a probabilistic screening criterion in our case study, as the containment has not yet been designed. Since the CECFP was impossible to calculate at this stage of the MIT GFR design, the sole probabilistic screening criterion used in our case study was therefore the maximum allowed CCDP of 10^{-2}.

Failure data used in the ECCS case study was gathered from multiple sources [4, 21, 23-26]. Section A.1 in the Appendix lists the failure data used in the case study. Gas reactor data was difficult to obtain as the current U.S. commercial reactor fleet is comprised entirely of light-water reactors. It is recognized that the LWR component failure data used for some of the components in the case study is not optimal. However, ECCS components have not been designed in detail, therefore generic LWR failure data was viewed as an acceptable approximation for the design guidance of the ECCS.

Figure 4-4 illustrates the event tree used in the PRA of the Bare-Bones ECCS. As can be seen from the event tree, failures of ECCS components are not the only consideration in the analysis. In the event of a LOCA, the Reactor Shutdown System (RSS) – the system that trips the reactor - is required to function. As is the convention, "up" in the bare-bones event tree correlates to system success while "down" illustrates system failure. It was conservatively assumed for the case study that the failure of the reactor to trip led directly to core damage. Sequence 8 in Figure 4-4 illustrates the failure of the RSS leading directly to core damage. Since the RSS has not yet been designed, the failure probability of the reactor shutdown system was estimated based upon system failure probabilities in [19, 27] and can be found in Section A.1 of the Appendix. Moving from left to right along the event tree, the supply of offsite power to the ECCS is considered next. Assuming the reactor successfully trips, power is required to spin the blower (via the electric motor) in the bare-bones design illustrated in Figure 4-1. The probability of the loss of offsite power was taken from [4]. If offsite power is unavailable, power can still be supplied to the electric motor (EM in Fig. 4-1) by onsite diesel generators. In the bare-bones design, failure of offsite and onsite AC power results in an ECCS that can not perform its safety function (which leads to core damage).

Assuming that either onsite or offsite power is available, the availability of onsite DC power for instrumentation and control must next be considered in the bare-bones analysis. It was assumed for the case study that loss of DC power resulted in the unavailability of the ECCS since the system could not be controlled or monitored. Finally, assuming that DC power is available in conjunction with onsite or offsite power, the emergency core cooling system itself is considered in the analysis. As the ECCS is the only safety system in place to prevent core damage in the event of a LOCA, the failure of the ECCS leads directly to core damage. Fault trees for the onsite diesels, onsite DC power, and the emergency core cooling system can be found in Section A.2 of the Appendix. The fault trees illustrated in the Appendix have 2x100% capable loops for redundant systems. As will be seen in the analysis results, other redundant configurations were considered. Fault trees for these other configurations were created, but are not illustrated in the Appendix as their structure can be inferred from the fault trees illustrated in the Appendix.

The probabilistic analysis was carried out using the SAPHIRE computer code [28]. Common-cause failures were addressed for similar, redundant components via the Beta-factor model using the generic value of $\beta = 0.1$. As has been mentioned earlier, due to the current status of the GFR design, design details are sparse and therefore generic failure data are utilized. Uncertainty analysis was carried out using the Monte Carlo method with a sample size of 10,000 [29].

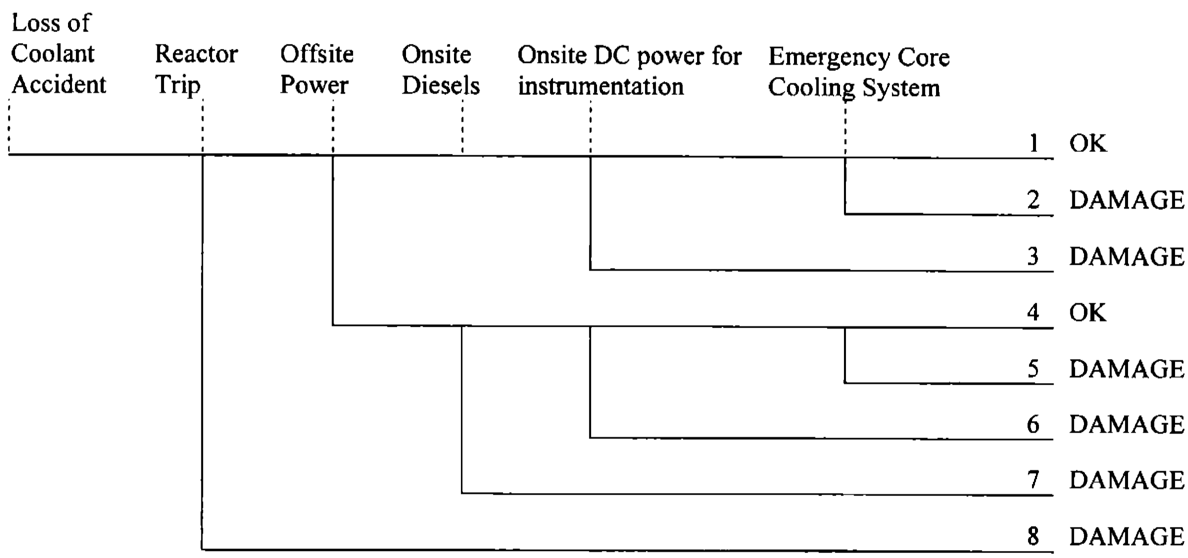


Figure 4-4. Bare-Bones ECCS Event Tree

The initial ECCS design screened under the deterministic and probabilistic screening criteria outlined above is illustrated in Fig. 4-1. The ECCS and supporting systems were configured initially with no onsite diesel generators and only one (100% capable) onsite DC battery. The results of the comparison of the bare-bones design versus the deterministic screening criteria are listed in Table 4-1. The results of the comparison of the bare-bones design versus the probabilistic screening criterion are listed in Table 4-2. As can be seen from Table 4-1, the initial design (Design 1) does not meet the deterministic screening criteria of GDC 35 for any number of redundant ECCS loops. The initial design violates GDC 35 by both not providing an onsite AC power supply and by a single failure of the one DC battery leading directly to failure of the safety system.

The initial bare-bones design also does not meet the CCDP probabilistic screening criterion for any loop configuration, as all numbers of ECCS loops result in a CCDP larger than 10^{-2}. As per the iterative four-step design guidance methodology, illustrated in Fig. 3-2, the initial bare-bones design was modified.

Table 4-1. Bare Bones Design: Meets Deterministic Screening Criteria?

		Number of ECCS Loops					
Design Number	Configuration	1x100%*	2x100%	3x50%	3x100%	4x50%	Comments
		Meet Deterministic Screening Criteria?					
1	No Diesels, 1x100% DC Battery	No	No	No	No	No	Violates GDC 35, no onsite AC power, Single Failure criterion (SFC)
2	1x100% Diesel, 1x100% DC Battery	No	No	No	No	No	Violates SFC
3	1x100% Diesel, 2x100% DC Battery	No	No	No	No	No	Violates SFC + Loss of Offsite Power
4	2x100% Diesel, 2x100% Battery	No	Yes	Yes	Yes	Yes	1x100% violates SFC

*Violates single failure criterion of GDC 35

Table 4-2. Bare Bones Design: Meets Probabilistic Screening Criterion?

Conditional Core Damage Probability given LOCA* ($\leq 10^{-2}$ Acceptable)						
		Number of ECCS Loops				
Design Number	Configuration	1x100%**	2x100%	3x50%	3x100%	4x50%
		Meet Probabilistic Screening Criterion? (Mean CCDP)				
1	No Diesels, 1x100% DC Battery	No (2.35E-2)	No (2.19E-2)	No (2.19E-2)	No (2.19E-2)	No (2.19E-2)
2	1x100% Diesel, 1x100% DC Battery	Yes (4.11E-3)	Yes (2.38E-3)	Yes (2.40E-3)	Yes (2.38E-3)	Yes (2.38E-3)
3	1x100% Diesel, 2x100% DC Battery	Yes (3.49E-3)	Yes (1.76E-3)	Yes (1.78E-3)	Yes (1.75E-3)	Yes (1.75E-3)
4	2x100% Diesel, 2x100% Battery	Yes (2.50E-3)	Yes (7.57E-4)	Yes (7.74E-4)	Yes (7.49E-4)	Yes (7.49E-4)

*LOCA Frequency = 5.45E-04
**Violates single failure criterion of GDC 35

Design 2 added a 1x100% capable onsite emergency diesel generator to provide onsite AC power in the event of a loss of offsite power. The comparisons of ECCS design 2 against the deterministic and probabilistic screening criteria are illustrated in Tables 5-1 and 5-2 respectively. Design number 2 violates the single failure criterion of GDC 35 as a failure of one DC battery leads directly to core damage. Somewhat surprisingly, design 2 meets the probabilistic screening criterion for all ECCS loop configurations. This suggests that the deterministic screening criteria taken from current regulations may be overly conservative as compared to the probabilistic screening criterion. However, one can not draw definitive conclusions regarding deterministic screening criteria, such as the requirement for onsite AC power or the single failure criterion, from one accident sequence. This is because these criteria may affect other accident sequences and safety systems. Therefore, disagreement on the acceptability of design options based upon deterministic and probabilistic screening criteria in this case study does not necessarily have any regulatory implications. Also, the non-regulatory consideration of online maintenance needs to be taken into account. Online maintenance could not be performed on 1x100% configurations of safety systems. Hence, the reactor would have to be shut down all maintenance of that system.

Since design 2 violated the SFC of GDC 35, the design was modified and analyzed as design 3. This design added a DC battery to increase the capability of the onsite DC power system to 2x100%. Design number 3 also does not meet the deterministic screening criteria. GDC 35 states that, "...for onsite electric power system operation (assuming offsite power is not available)...the system safety function can be accomplished, assuming a single failure." Design 3 violates this section of GDC 35 (as is shown in Table 4-1) since a single failure of the 1x100% diesel generator (assuming offsite power is not available) would prohibit the ECCS from accomplishing its safety function. The probabilistic screening criterion is met for design 3 (as is shown in Table 4-2), however the design needed to be modified according to the iterative four-step methodology.

Design 4 added a diesel generator to increase the capability of the onsite emergency diesel system to 2x100%. This design meets both the deterministic and probabilistic screening criteria of Step 2 of the four-step design guidance methodology. This ECCS design configuration therefore was able to move on to Step 3.

IV.C Step 3: Analysis of ECCS Designs

Providing a highly reliable means of cooling the core in the event of a LOCA is a critical concern of the GFR design team. In the case study, Step 3 of the four-step methodology is to analyze and modify the ECCS design based upon PRA insights. The first ECCS design modification that passed the deterministic and probabilistic screening criteria of Step 2, design 4, was used as the basis for the ECCS iterative design guidance of Step 3.

As can be seen from Table 4-3, for instance, there is an insignificant improvement in CCDP when adding redundant ECCS loops beyond 2x100% capability. This is due to the use of the Beta factor to model common-cause failures. For example, a 2 component parallel system (2x100% capable) requires failure of both components for the system to fail. Under the Beta factor model (using β=0.1), identical components can either fail randomly or all components can fail due to a common cause. Using a component failure probability for the two components, A and B, of, $u=1x10^{-3}$, for the 2x100% capable system, the probability of failure of the 2x100% capable system due to random causes is:

$$P_{2x100\%random} = P(A) * P(B) = u^2 = 1x10^{-6}$$

The CCF probability of the 2x100% capable system is:

$$P_{2x100\%CCF} = \beta * u = 1x10^{-4}$$

The total 2x100% capable system failure probability is:

$$P_{2x100\%fail} = P_{random} + P_{CCF} = 1.01x10^{-4}$$

Adding an identical redundant component, C, to bring the system capability to 3x100% does little to change the total failure probability in the Beta factor model. The total failure probability of the 3x100% capable system is:

$$P_{3x100\%fail} = P_{random} + P_{CCF} = P(A) * P(B) * P(C) + P_{CCF} = u^3 + \beta * u = 1x10^{-9} + 1x10^{-4} \cong 1x10^{-4}$$

It can be seen that adding identical, redundant components beyond 2x100% does little to decrease the system failure probability when using the beta factor common cause failure model. Other models exist that do not so conservatively describe common cause failures, such as the Multiple Greek Letter model and Alpha factor model. However, all models are approximations of the actual CCF rates and mechanisms. The preliminary stage of the design, use of generic failure data, and the desire for a relatively quick and straightforward analysis suggest that despite the beta factor model being the most conservative of the models, it is an acceptable model for the design guidance methodology.

In the CCF literature reviewed [30-34] and communications with CCF experts [35, 36], no instances of the quantitative modeling of design changes formulated to defend against CCFs were found. However, there is some guidance on methods to qualitatively reduce CCFs during the design stage. Reduction of CCFs is therefore left to Step 4, the deliberation phase of the design guidance methodology. A discussion of the coupling factors and possible methods to qualitatively reduce common cause failures in reactor design is given in Section IV.D.

The conditional core damage probability given a LOCA for each ECCS design and each configuration of ECCS loops analyzed (1x100%, 2x100%, 3x50%, 3x100%, and 4x50%) are given in Tables 5-4 through 5-6. Cut sets, Fussel-Vesely, and Risk-Achievement Worth (RAW) importance measures were used to determine risk-significant components and configurations. Because of common cause failures, the mean CCDP of deterministically acceptable ECCS loop configurations (all except 1x100%) are nearly identical. Therefore, as regards insights into risk-significant components and configurations, one ECCS loop configuration (for the case study, 3x100% was used) is representative of the other acceptable ECCS loop configurations. The Section A.3 of the Appendix contains tables listing the cut sets that contribute to 99% of the total risk along with component rankings sorted by Fussel-Vesely and RAW for each ECCS design analyzed in Step 3.

Design 4 consists of a 2x100% onsite diesel power system and a 2x100% onsite DC battery power system to supplement the bare-bones ECCS design illustrated in Figure 4-1. Using the 3x100% capable ECCS loop configuration to represent the deterministically acceptable configurations, the mean CCDP of Design 4 was 7.49×10^{-4}.

Three significant insights which into this ECCS design were found. First, 24.1% of the CCDP of design 4 results from the Loss of Offsite Power (LOOP) in combination with the CCF of the onsite emergency diesels. Second, 17.3% of the risk of ECCS failure given a LOCA for design 4 came from a LOOP plus the loss of onsite emergency diesels due to random failures. Finally, 16% of the CCDP was from the failure of the one onsite DC power transmission loop.

Table 4-3. Iterative ECCS Design Guidance

		Conditional Core Damage Probability given LOCA*					
		Number of ECCS Loops					
Design Number	Configuration	1x100%**	2x100%	3x50%	3x100%	4x50%	3x100% ECCS Insights
		Mean CCDP					
4	2x100% Diesel, 2x100% Battery	2.50E-3	7.57E-4	7.74E-4	7.49E-4	7.49E-4	• LOOP + CCF of diesels accounts for 24.1% of risk • LOOP + random failure of diesels accounts for 17.3% of risk • 1 DC Transmission loop accounts for 16% of risk
5	2x100% Diesel, 2x100% Battery, 2x100% Transmission	2.36E-3	6.69E-4	6.85E-4	6.62E-4	6.62E-4	• LOOP + CCF of diesels accounts for 28.1% of risk • LOOP + random failure of diesels accounts for 20.2% of risk
6	3x100% Diesel, 2x100% Battery, 2x100% Transmission	2.21E-3	5.18E-4	5.34E-4	5.11E-4	5.11E-4	• LOOP + CCF of diesels accounts for 34.8% of risk • LOOP + random failure of diesels accounts for 1.4% of risk • CCF of electric motor accounts for 15.7% of risk

*LOCA Frequency = 5.45E-04
**Violates single failure criterion of GDC 35

Design 5 addressed the third insight of design 4, the single DC transmission loop. In design 5, a redundant DC transmission loop was added to bring the DC transmission capability to 2x100%. This led to a mean CCDP for design 5 (3x100% ECCS loops) of 6.62×10^{-4}. For this configuration, LOOP plus the CCF of the onsite diesels accounted for

28.1% of the CCDP. LOOP in combination with random failures leading to the failure of the onsite AC power source accounted for 20.2% of the risk in design 5.

Design 6 addressed the random failures of the diesel generators in design 5 by adding a third diesel generator. Design 6 consists of 3x100% onsite diesels, 2x100% DC batteries, and 2x100% DC transmission lines to supplement the bare-bones ECCS design illustrated in Figure 4-1. The mean CCDP for design 5 (3x100% ECCS loops) was found to be 5.11×10^{-4}. LOOP plus the CCF of the onsite diesels accounted for 34.8% of the CCDP of design 6. LOOP in combination with random failures leading to the failure of the onsite AC power source only accounted for 1.4% of the risk in design 6. The CCF of the electric motors on the ECCS accounted for another 15.7% of the CCDP. In total, 51.9% of the CCDP resulted from failure of the ECCS electric motors or failure to supply power to the ECCS electric motors in design 6.

Three designs were proposed to reduce the CCDP contributions from the CCF of the ECCS electric motors and the Loss of Station Power (loss of both off- and on-site AC power). The first two designs – the secondary onsite turbine design and the secondary onsite microturbine design - added a secondary onsite AC power source providing power to a second electric motor (of a different design than the original electric motor) that could also spin the blower.

The secondary onsite turbine design is illustrated in Figure 4-5. The blower can be spun in this design option by the original electric motor (labeled EM1 in Fig. 4-5) that receives power from either offsite power or onsite emergency diesels. Failing that, the blower can be spun by a second, diverse electric motor (labeled EM2). In the event of a loss of both offsite power and the on-site emergency diesels the valve labeled VE opens.

Nitrogen then flows from the accumulator to the turbine outside of the PCIV (labeled T). The nitrogen spins the turbine, which in turn spins the electric generator (labeled G).

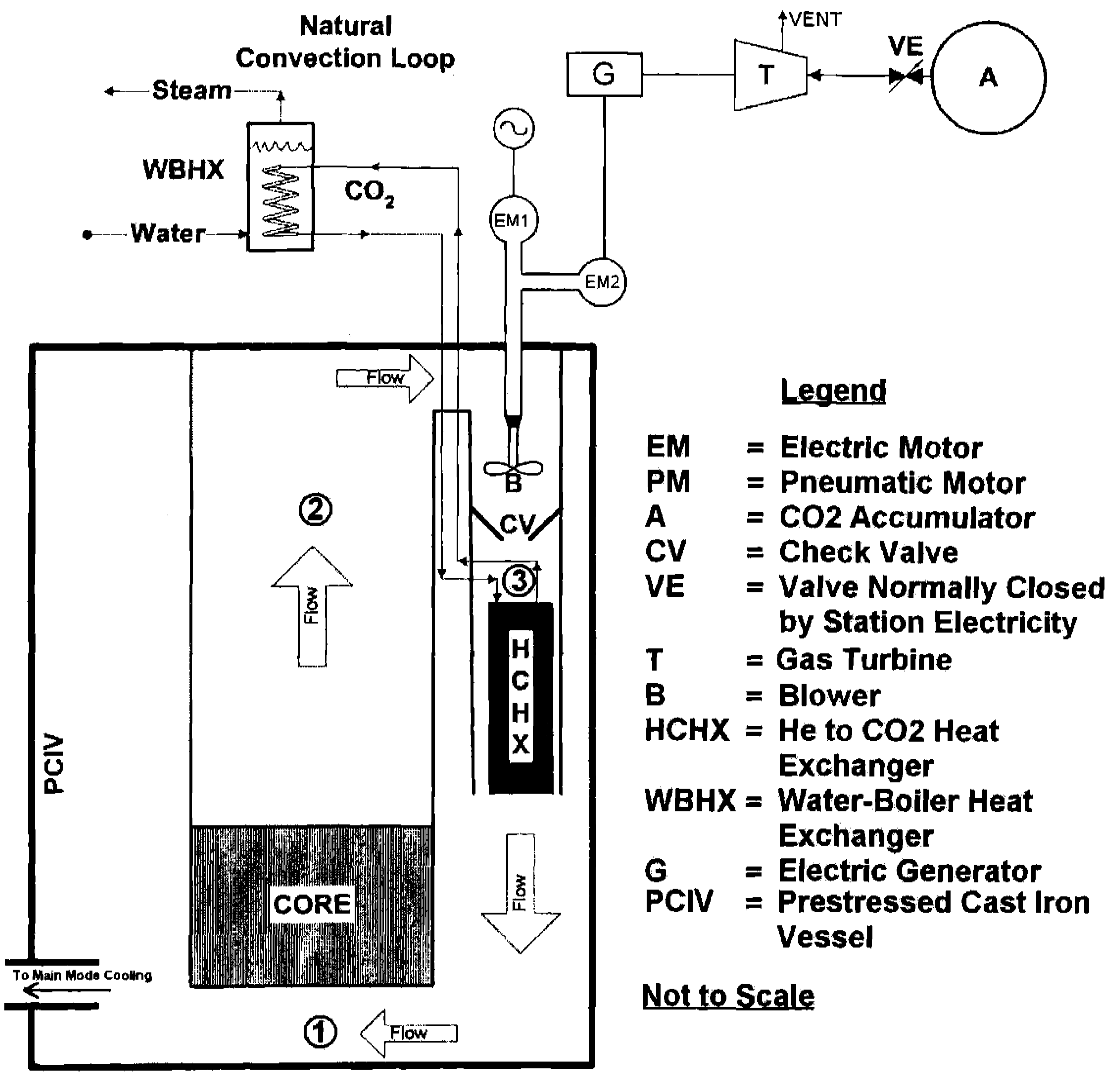


Figure 4-5. Secondary Onsite AC Power Design: Turbine

The generator then powers a second electric motor (labeled EM2) on the same drive shaft as electric motor 1. The blower is then spun by the second electric motor. It should be noted that similar to the onsite emergency diesel generators, the number of secondary onsite turbine loops can be independent of the number of ECCS loops. A 100 m^3

accumulator tank at 10 MPa would provide approximately one day of emergency power per loop.

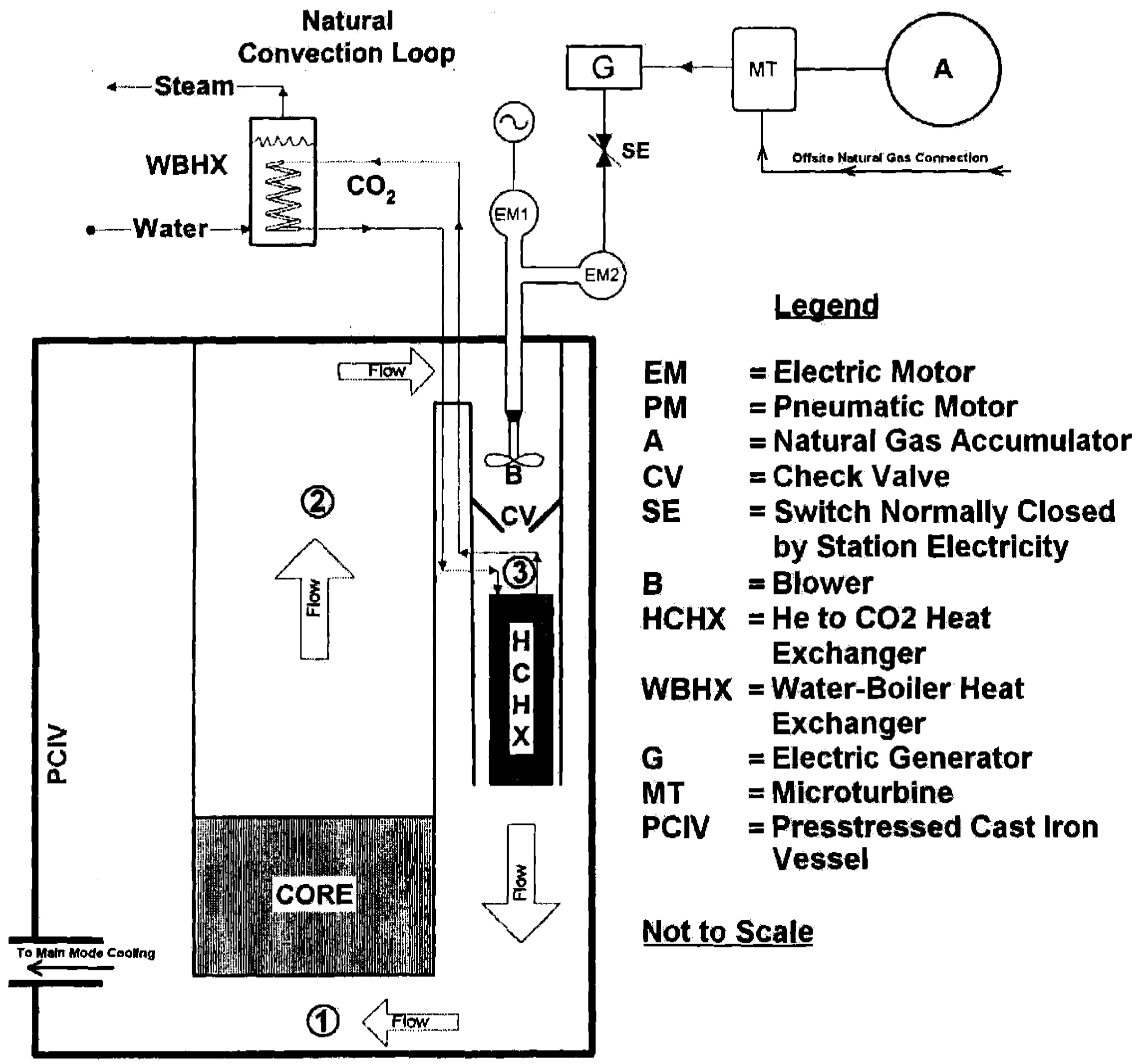


Figure 4-6. Secondary Onsite AC Power Design: Microturbine

Figure 4-6 illustrates the secondary onsite microturbine design. Power is supplied to the blower by two separate sources. In the case of offsite power or on-site emergency diesel availability, the blower is spun by electric motor 1 (labeled EM1). Failing that, the blower can be spun by a second, diverse electric motor (labeled EM2). In the event of a station blackout the electric switch labeled SE opens. Natural gas constantly flows from to the microturbine via an offsite natural gas connection. The accumulator tank is provided in case of the loss of offsite natural gas. A 100 m^3 accumulator tank at 10 MPa would provide approximately ten days of emergency power per loop. The microturbine is powered and spun via natural gas combustion, which in turn spins the electric generator. The generator then powers a second electric motor (labeled EM2) on the same drive shaft as electric motor 1. The blower is then spun by the second electric motor. It should also be noted that the number of secondary onsite turbine loops can be independent of the number of ECCS loops.

Figure 4-7 illustrates the event tree used to analyze the secondary onsite turbine and microturbine designs. As can be seen, adding a diverse secondary onsite AC power source creates an additional system success path as compared to the bare-bones ECCS event tree.

Table 4-4 shows the analysis results of the secondary onsite AC power designs. Design 7, 1x100% secondary onsite turbine loops, had a mean CCDP (3x100% ECCS loops) of 2.14×10^{-4}. Analysis of the cut sets of design 7 revealed that ~96% of the CCDP was due to common cause failures of ECCS or onsite DC power components. As such, little more can be done to quantitatively improve the ECCS and supporting systems design. Diverse ECCS and onsite DC power loops would need to be added to design 7 in order to significantly impact the CCDP.

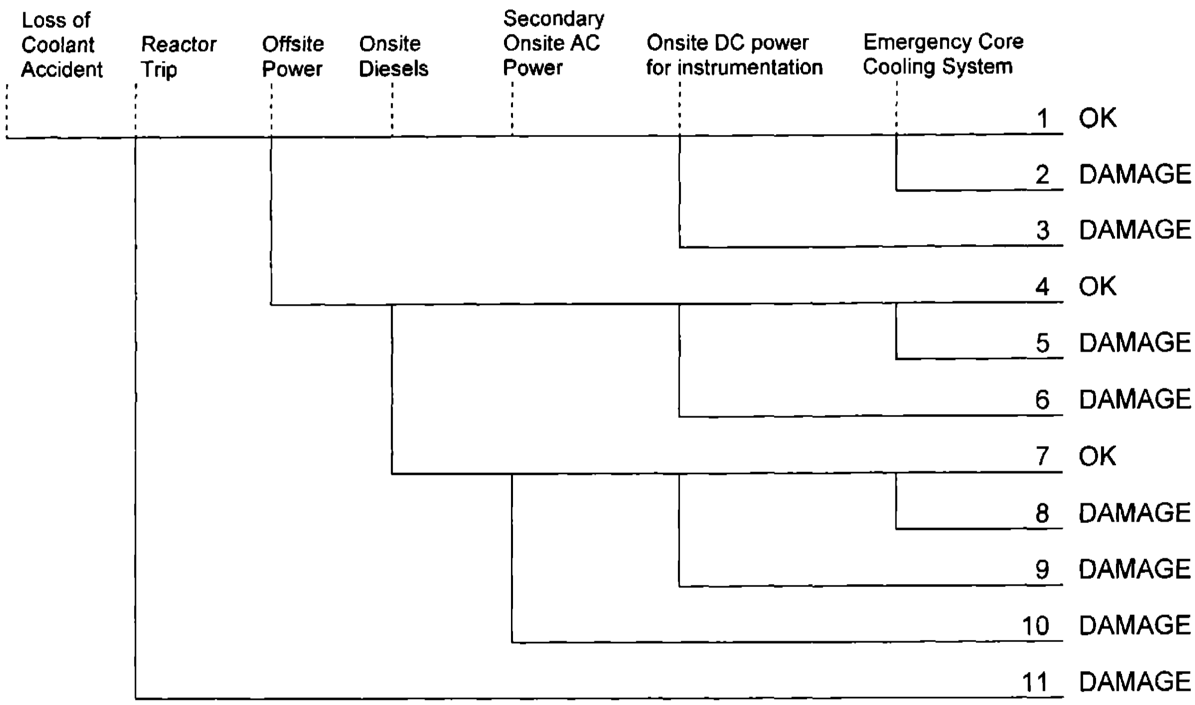


Figure 4-7. Secondary Onsite AC Power ECCS Event Tree

A mean CCDP of 2.04×10^{-4} was found for design 8, 1x100% secondary onsite microturbine loops. Cut set analysis showed that ~99% of the CCDP for design 8 was due to common cause failures of ECCS or onsite DC power components. Like design 7, little more can be done to quantitatively improve the ECCS and supporting systems design. In order to significantly impact the CCDP, diverse ECCS and onsite DC power loops would need to be added to design 8.

The third design proposed to reduce the CCDP contributions from the CCF of the ECCS electric motors and the loss of station power is illustrated in Figure 4-8. In this design, nitrogen accumulators provide a passive means of spinning the blower in the event of a LOCA. For the nitrogen accumulator design, power can be supplied to the blower by three diverse sources. In the case of offsite power or onsite emergency diesel availability, the blower is spun by an electric motor (labeled EM). The third possibility for moving coolant past the core involves the N_2 accumulator (labeled A). When primary pressure is lost due to the LOCA, the valve labeled VP opens. In the event of a station blackout the valve labeled VE opens. Nitrogen then flows from the accumulator to the turbine (labeled T). The nitrogen spins the turbine, which in turn spins the blower (labeled B). A 100 m^3 accumulator tank at 10 MPa would provide approximately one day of emergency power per loop.

Unlike the secondary onsite turbine and microturbine design options, the nitrogen accumulator system is part of an ECCS loop. This is reflected in the nitrogen accumulator ECCS design event tree illustrated in Figure 4-9. Also, because the nitrogen accumulator system is passive, onsite DC power for instrumentation and control is not required for system success. This leads to 6 success paths in the event of a LOCA as compared to the 2 success paths of the initial bare-bones ECCS design.

Table 4-4. Iterative ECCS Design Guidance: Secondary Onsite AC Power Designs

Conditional Core Damage Probability given LOCA*							
		Number of ECCS Loops					
Design Number	Configuration	1x100%**	2x100%	3x50%	3x100%	4x50%	2x100% ECCS Comments
		Mean CCDP					
7	3x100% Diesel, 2x100% Battery, 2x100% Transmission , 1x100% Secondary onsite Turbine	1.33E-3	2.19E-4	2.31E-4	2.14E-4	2.14E-4	• ~96% of risk due to CCF of ECCS or DC components
8	3x100% Diesel, 2x100% Battery, 2x100% Transmission , 1x100% Secondary onsite Microturbine	1.32E-3	2.08E-4	2.17E-4	2.04E-4	2.04E-4	• ~99% of risk due to CCF of ECCS or DC components

*LOCA Frequency = 5.45E-04
**Violates single failure criterion of GDC 35

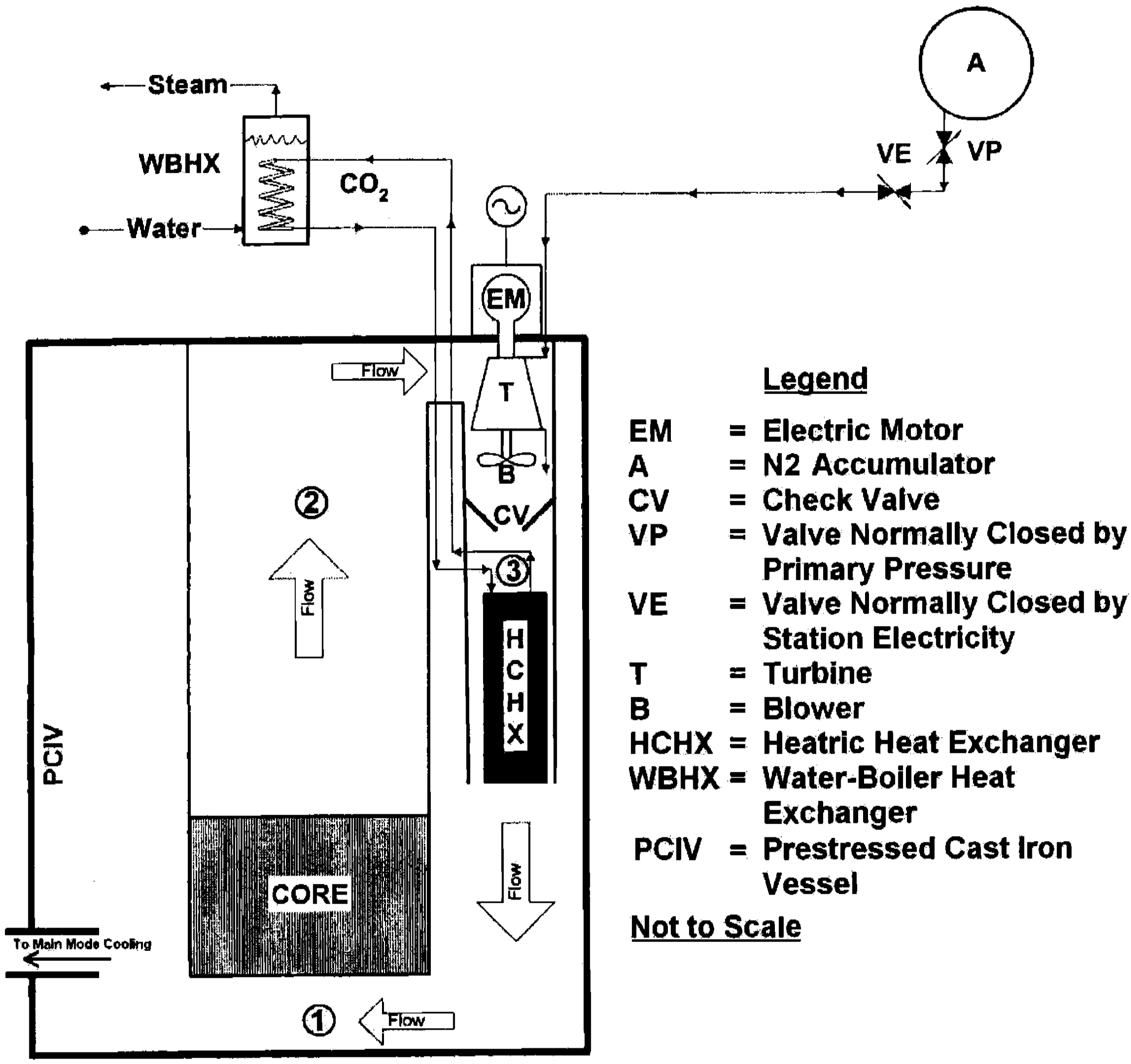


Figure 4-8. ECCS Design: Nitrogen Accumulator

Unfortunately, in addition to providing a passive means of performing emergency core cooling, the nitrogen accumulator design adds another path for the coolant to escape the reactor vessel. Piping is required to connect the nitrogen accumulators which were designed to be outside of the reactor vessel to each ECCS loop inside the reactor vessel. A break in this piping would lead to a LOCA. For the ECCS loop LOCA, the loop in which the LOCA occurred would be unable to perform its function of cooling the core. Figure 4-10 illustrates the ECCS loop LOCA event tree used in the analysis of the nitrogen accumulator ECCS design. The frequency of an ECCS Loop LOCA was taken from the AP-1000 PRA. The frequency of a small LOCA (5×10^{-4}) was used because it was the most likely LOCA. As for the LOCA initiator, it is recognized that the AP-1000 pipe failure data may not be the optimal data for the ECCS loop LOCA. But again, pipe failure data for the AP-1000 are much more current than any previous Gas-Cooled Fast Reactor design and reflect the state-of-the-art in pipe materials and manufacturing. Also, the MIT GFR is at such an early stage of design that the use of generic failure data is warranted until more details about the plant are developed.

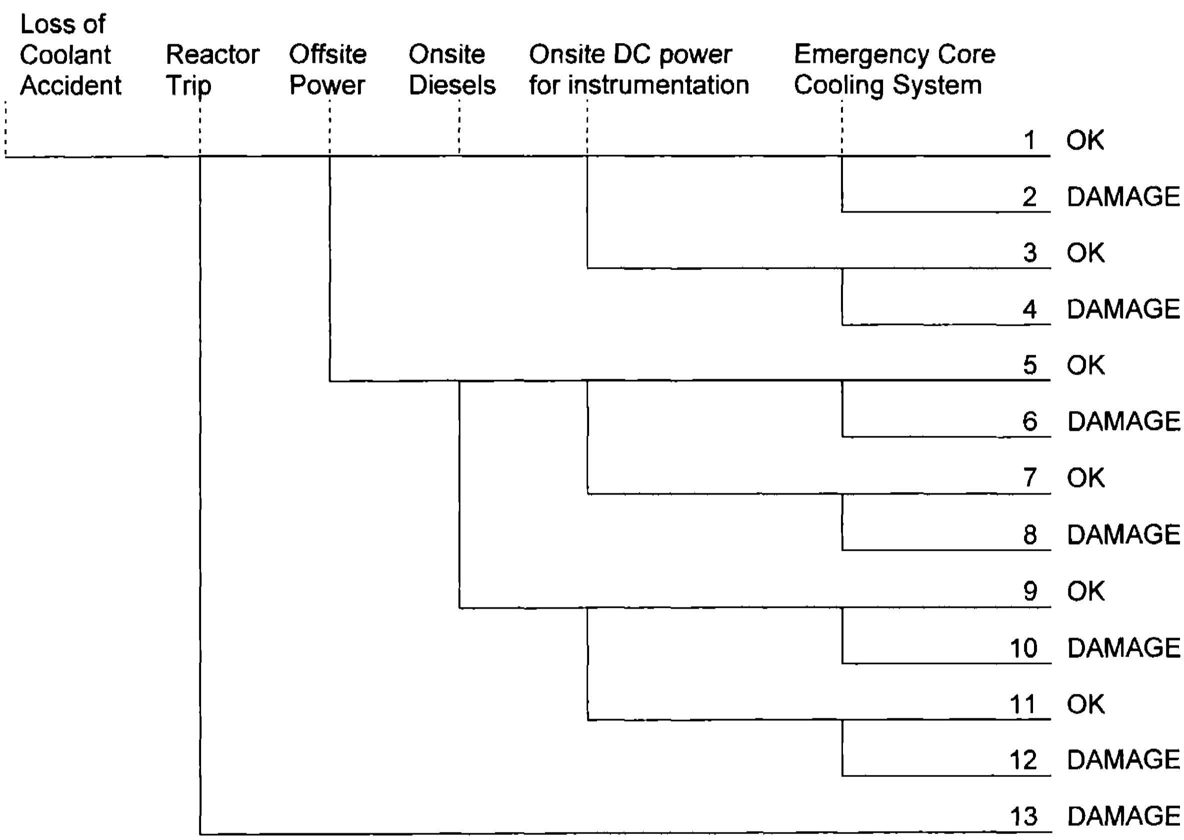


Figure 4-9. Nitrogen Accumulator ECCS Design Event Tree: LOCA

The nitrogen accumulator ECCS design addition provides an interesting insight into guiding reactor design through PRA. It was originally assumed that adding components onto a bare-bones plant design would be easily accomplished in the PRA. However, as the ECCS loop LOCA sequence illustrates, the addition of components onto a bare-bones design can vastly change the PRA.

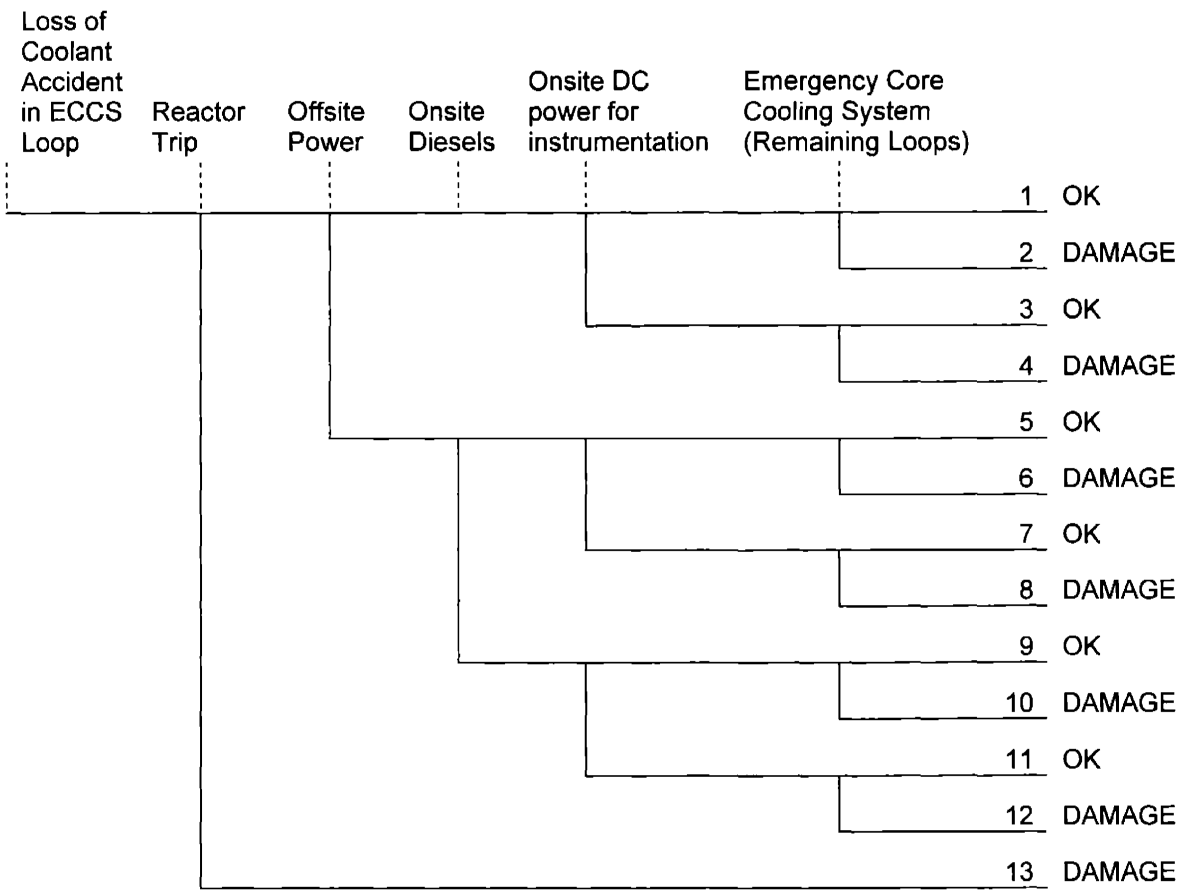


Figure 4-10. Nitrogen Accumulator ECCS Design Event Tree: ECCS Loop LOCA

Table 4-5 lists the analysis results of the nitrogen accumulator design addition (design 9). Unlike in previous designs, the 2x100% and 3x50% ECCS configurations are deterministically unacceptable in addition to the 1x100% ECCS configuration. This is due to the ECCS loop LOCA and the single failure criterion. For example, an ECCS loop

LOCA in the 3x50% ECCS configuration reduces 3x50% to 2x50%. According to the design basis accident, ECCS operation is required with offsite power unavailable and assuming a single failure. If the single failure is a physical failure of the blower (B in Fig. 4-8), for instance, in one of the remaining loops, then the emergency core cooling system would not supply adequate core cooling.

Table 4-5. Iterative ECCS Design Guidance: Nitrogen Accumulator (Design 9)

	Number of ECCS Loops					
Configuration	1x100%*	2x100%*	3x50%*	3x100%	4x50%	Comments
	CCDP given LOCA**					
3x100% Diesel, 2x100% Battery, 2x100% Transmission , Nitrogen Accumulator	6.85E-4	2.56E-7	2.58E-7	2.44E-7	2.44E-7	•
	CCDP given ECCS Loop LOCA***					
3x100% Diesel, 2x100% Battery, 2x100% Transmission , Nitrogen Accumulator	1.00E+0	1.23E-3	2.61E-3	1.27E-4	1.34E-4	• 1x100%, 2x100%, 3x50% violate SFC
	CDF of Nitrogen of Nitrogen Accumulator Design Due to LOCA and ECCS Loop LOCA					
3x100% Diesel, 2x100% Battery, 2x100% Transmission , Nitrogen Accumulator	5.00E-4	6.21E-7	1.40E-6	6.48E-8	6.94E-8	• CCF becomes a factor at 3x100%

*Violates single failure criterion of GDC 35
**LOCA Frequency = 5.45E-04
***ECCS Loop LOCA Frequency = 5.00E-04

As can be seen from Table 4-5, the 2x100% and 3x50% nitrogen accumulator ECCS configurations provide further examples of ECCS designs that pass the probabilistic screening criterion (CCDP $\leq 10^{-2}$) but violate the deterministic screening criteria outlined in Section IV.B. Because advanced reactor regulations have not yet been developed, it is unclear whether those nitrogen accumulator configurations would be acceptable in the eyes of the USNRC. In any event, an exemption [15] could be applied for, even if these configurations did not meet the regulations. Therefore, the 2x100% and 3x50% nitrogen accumulator ECCS design configurations can be considered during the deliberation.

Table 4-6. Results of the Iterative PRA ECCS Design Guidance

Design Number	Configuration	3x100% ECCS loops Mean CDF	Δ CDF over initial bare-bones design
1	No Diesels, 1x100% DC Battery	1.19E-05	0.00%
2	1x100% Diesel, 1x100% DC Battery	1.29E-06	-89.13%
3	1x100% Diesel, 2x100% DC Battery	9.49E-07	-92.01%
4	2x100% Diesel, 2x100% Battery	4.06E-07	-96.58%
5	2x100% Diesel, 2x100% Battery, 2x100% Transmission	3.59E-07	-96.98%
6	3x100% Diesel, 2x100% Battery, 2x100% Transmission	2.77E-07	-97.67%
7	3x100% Diesel, 2x100% Battery, 2x100% Transmission , 1x100% Secondary onsite Turbine	1.16E-07	-99.02%
8	3x100% Diesel, 2x100% Battery, 2x100% Transmission , 1x100% Secondary onsite Microturbine	1.11E-07	-99.07%
9	3x100% Diesel, 2x100% Battery, 2x100% Transmission, Nitrogen Accumulator	6.48E-08	-99.45%

Table 4-6 lists the mean core damage frequencies for designs considered during the four-step methodology and the percentage change in the mean CDF as compared to the initial bare-bones design. The CDFs listed are for the 3x100% ECCS configuration. It should be noted that for all designs, except for design 9 (the nitrogen accumulators design addition), that the 2x100%, 3x50%, and 4x50% ECCS loop configurations resulted in almost identical CDFs. Decision-makers should be aware of this when deliberating upon ECCS designs in Step 4 of the design guidance methodology.

IV.D Step 4: Deliberation

In the fourth and final step of the design guidance methodology, the designs are deliberated upon by the decision makers. Other considerations beyond the CDF of ECCS designs are reflected upon during the deliberation. Since a Generation-IV reactor was analyzed, the value tree presented in the Generation-IV Roadmap [1] by the Nuclear Energy Research Advisory Committee (NERAC) was looked at as a reference of objectives to be considered when designing an advanced nuclear reactor. This tree is shown in Figure 4-11.

In its value tree, NERAC presents four "Goal Areas," that will be termed fundamental objectives [14]. Fundamental objectives refer to attributes that reflect simply what is desired in a Generation-IV nuclear power plant. These fundamental objectives are sustainability, economics, safety and reliability, and proliferation resistance and physical protection.

A fundamental objective was added to the NERAC fundamental objectives to be considered during the deliberation – stakeholders. These are external parties whose views have an impact on the licensing of Generation-IV systems. Stakeholders include the regulator, the local government, and the public. Even though a decision option may have no significant impact on, for instance, the safety of a design, the potential impact on the stakeholders must be considered.

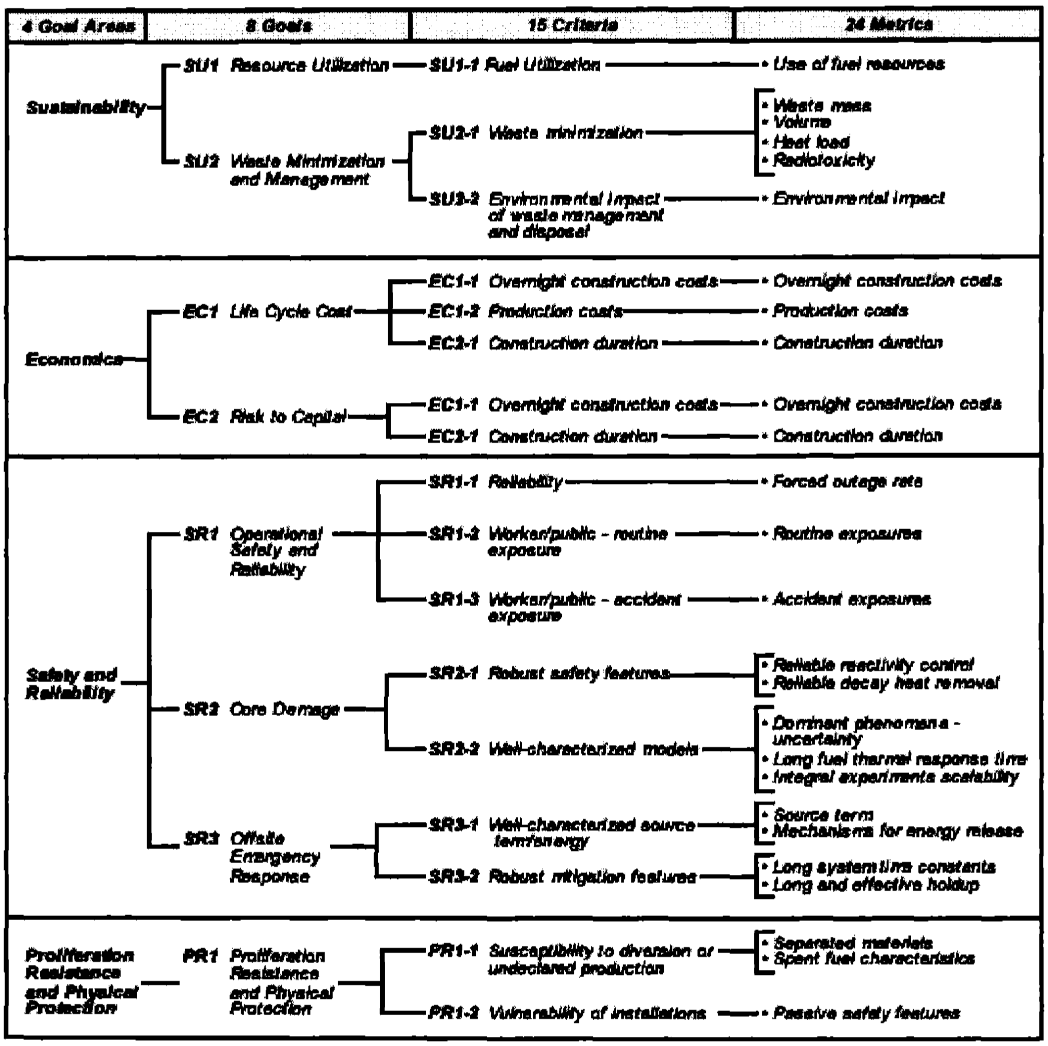


Figure 4-11. The NERAC-Generation-IV Value Tree [36]

During the deliberation, the decision makers consider the impact each ECCS design has on the fundamental objectives. The fundamental objectives considered during the MIT GFR ECCS deliberation are illustrated in Figure 4-12.

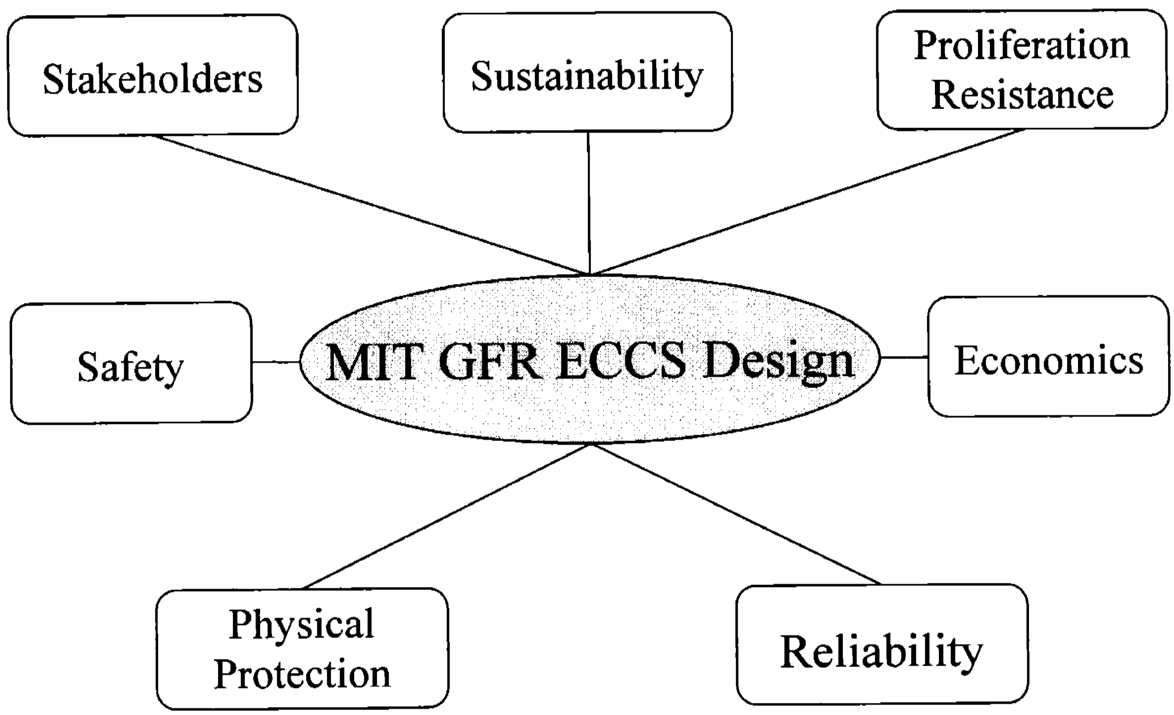


Figure 4-12. Fundamental Objectives for Deliberation

As was mentioned in Section IV.C, quantitative methods and modeling regarding the reduction of common cause failures are currently unavailable. Therefore, the reduction of CCFs is a process that must be undertaken during the deliberation. Figure 4-13 illustrates the share that each of the coupling factors have contributed to CCF in the current fleet of commercial LWRs [30]. A coupling factor is a characteristic of a group of components that identifies them as susceptible to the same cause of failure. Coupling factors identified in [30] were hardware, maintenance, operations, and environment. An example of a hardware coupling factor is the same defective design in multiple identical components. An example of an operational coupling factor would be an incorrect set point specified in the calibration procedure for multiple relief valves. An environmental coupling factor could be two check valves operating in a similar location and a property of that localized area (i.e., too hot) disables the check valves. An example of a maintenance coupling factor includes the same personnel incorrectly performing a maintenance procedure on all EDGs which disables all of them. It is apparent from these examples that only the hardware and environment coupling factors are readily affected by design changes. A sense of the approximate impact of the reduction or elimination of each coupling factor in the design of the MIT GFR can be gained from Figure 4-13.

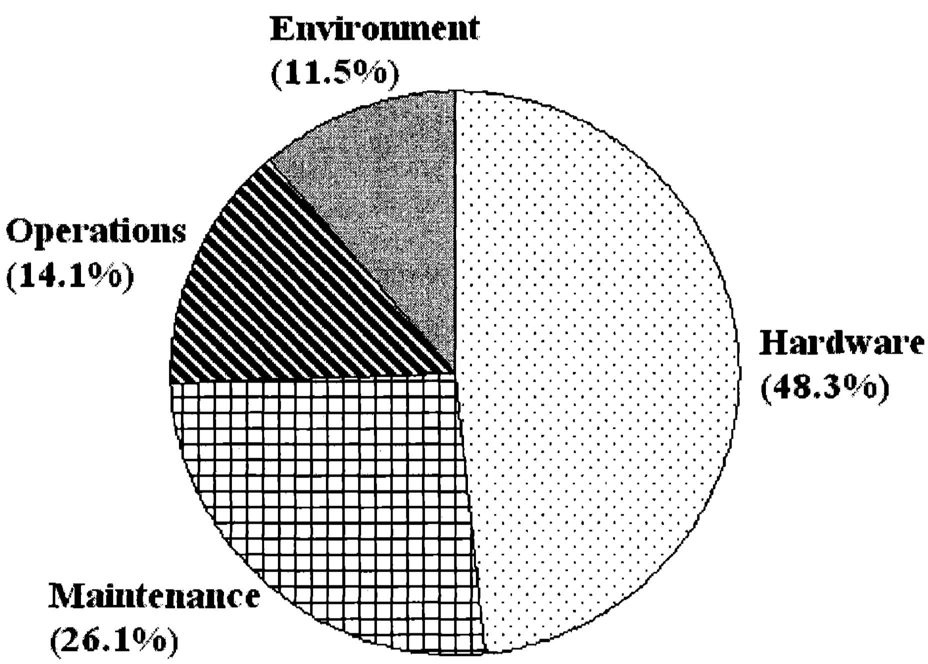


Figure 4-13. Distribution of complete CCF events by coupling factor (US LWR experience) [30]

Qualitative CCF insights into design configurations analyzed can be deduced from Fig. 4-13. For example, while the mean CCDP of design 8 was nearly identical for 2x100% capable and 3x50% capable ECCS loops, it is noted that the proximate CCF cause "location" would be reduced for the 3x50% capable ECCS loops. Since, ~99% of the CCDP for design option 8 was due to CCFs of ECCS or onsite DC components, the 3x50% capable configuration's reduction of the environmental coupling factor reduces the CCF rate which in turn would reduce the CCDP. Therefore, the 3x50% ECCS configuration may be more desirable than the 2x100% ECCS configuration for design 8.

Online maintenance was also considered during the deliberation. While it is possible that a 1x100% capable configuration may be allowed under probabilistic screening criterion, any maintenance on the loop could not take place while the reactor was online. The safety function of a 1x100% capable ECCS configuration could not be accomplished when the loop was down for testing or maintenance. Reference [37] gives acceptable increases in risk due to testing or maintenance.

In this case study, the GFR decision makers are still deliberating on the results of the ECCS design guidance analysis. In particular, the use of microturbine power packages is of interest because of their purported high reliability and the potential to run continuously – thereby providing assurance of readiness and elimination of the failure to start sequence. Microturbines are also a focus of further deliberation because similar components have never been use in previous reactors. In the event that the decision makers decide they are not thoroughly satisfied with any of the ECCS or if they see possible improvements of the ECCS design based upon the formal analysis, the design guidance methodology can be iterated until the decision makers are satisfied.

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V. CONCLUSIONS

The use of a four-step methodology to guide the MIT gas-cooled fast reactor emergency core cooling system design was devised from activities currently ongoing in the nuclear industry, regulator, and universities regarding advanced reactors and risk-informed regulations. This methodology was based upon [2]. The most substantial difference from the methodology presented in [2] was the use of PRA instead of MAUT in this case study as the main quantitative analysis tool. This modification was necessary to account for the preliminary stage of the MIT GFR. Modifications were also performed on Step 2 of the four-step methodology to reflect the uncertain regulatory environment that advanced reactors face and the possibility to apply for exemptions for designs that do not meet specified regulatory criteria. The iterative nature of the four-step methodology should also be highlighted – as it allows for design guidance based upon PRA and deterministic insights of previous designs.

Great care is necessary when modifying a design based upon insights discovered during the four-step methodology. Adding components or changing the configuration of components can vastly change the PRA model. It was originally assumed that adding components to a bare-bones advanced reactor design would simply translate to adding the component into the PRA model. However, as was the case when modeling design 9 (the nitrogen accumulator addition) new accident sequences can be introduced.

Cases have been found during the iterative four-step design guidance where ECCS loop configurations were acceptable according to a probabilistic screening criterion, but unacceptable under deterministic screening criteria. Further consideration is required to determine the significance of this incongruity. It should be noted that definitive conclusions regarding the acceptability of components and configurations cannot be drawn from one accident sequence. Possible dependencies concerning the same components in other accident sequences or other GFR systems make definitive conclusions regarding the necessity of certain deterministic screening criteria impossible at this preliminary stage of the MIT GFR design.

Using the formal four-step methodology during the design stage of an advanced reactor can be useful in predicting possible questions or design justifications that may arise during the licensing process. First, current light water reactor regulations and possible risk-informed regulations were utilized during the screening of designs as an indicator of possible advanced reactor regulations. Second, PRA identifies risk-significant components that the USNRC may focus upon during the licensing process.

Other considerations beyond those encompassed in the PRA and in the formal analysis need to be taken into account during the deliberation. The impact of a design on the fundamental objectives of sustainability, economics, reliability, proliferation resistance, physical protection and stakeholder relations should be considered during the deliberation. Also, matters such as the possibility of online maintenance in addition to the contribution to the CDF of a design need to be addressed during Step 4. Qualitative methods for reducing the CDF due to common cause failures also are considered. No quantitative method for modeling reductions in CCF have been proposed, therefore considerations of CCF rates between designs and the impact of steps taken to reduce CCFs are considered qualitatively during the deliberation. Methods to more thoroughly model design changes that affect common cause failures, either qualitatively or preferably quantitatively, are an area for further study.

The iterative design guidance methodology led to a reduction of the CDF contribution due to a LOCA of over three orders of magnitude from the baseline ECCS design to Design 9 (from 1.19×10^{-5} to 6.48×10^{-8} for the 3x100% loop configuration) and potential ECCS licensing issues were identified. As such, this design guidance methodology was used to make better MIT GFR ECCS decisions and predict possible justification required by the regulator. This methodology is applicable to other GFR design options as well as other advanced reactors.

Currently, ranking GFR ECCS designs strictly by the CDF contribution due to a LOCA leads to the selection of Design 9 (the nitrogen accumulator design) as the best option. However, LWR initiating event data was used for the LOCA frequencies in this case study. Design 9 is particularly sensitive to changes in LOCA frequencies, as two LOCA sequences can challenge that ECCS design. Therefore, of the designs analyzed, the design that presently appears best is Design 8 (the secondary onsite AC power microturbine design) with a CDF contribution due to a LOCA of 1.11×10^{-7} and their elimination of the failure-to-start failure mode for an onsite AC power supply.

Many directions for future work are available to improve the design guidance of the MIT GFR emergency core cooling system and to guide the design of other MIT GFR systems. For instance, the collection of gas reactor component failure data would lead to less uncertainty in the results of the design guidance. Also, more information concerning the reliability of microturbines needs to be gathered. Mircoturbines are a newly developed technology that has never been used in a nuclear power plant. As such, they would be thoroughly scrutinized during the licensing process. Therefore, a concerted effort should be made during the design process to obtain accurate reliability and safety pertinent information regarding microturbines.

An overall goal for the core damage frequency of the MIT GFR needs to be developed. From this goal – target CDF contributions from individual accident sequences to the overall CDF can be determined. This will allow for the question of "How safe is safe enough?" to be answered for the safety systems designed to defend against accident sequences. The target CDF contributions from individual accident sequences also allows for an endpoint to be set for the PRA analysis in Step 3 of the iterative four-step design guidance methodology.

It is possible that the best ECCS design may not lead to the best MIT GFR safety system design when other accident sequences are considered. ECCS components can be used as part of other safety systems when faced with initiators other than LOCAs. Other accident sequences, resulting from initiating events such as the loss of offsite power or an inadvertent control rod withdrawal, need to be analyzed as the design of the GFR is further developed to ensure a safe and well balanced nuclear reactor.

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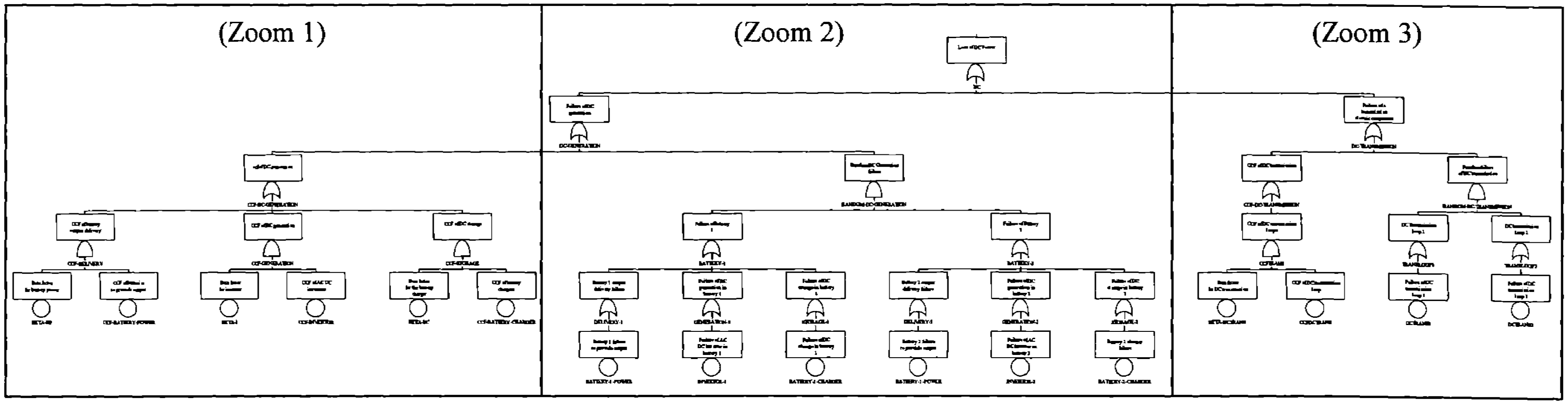
APPENDIX

A.1 Case Study Component Failure Data

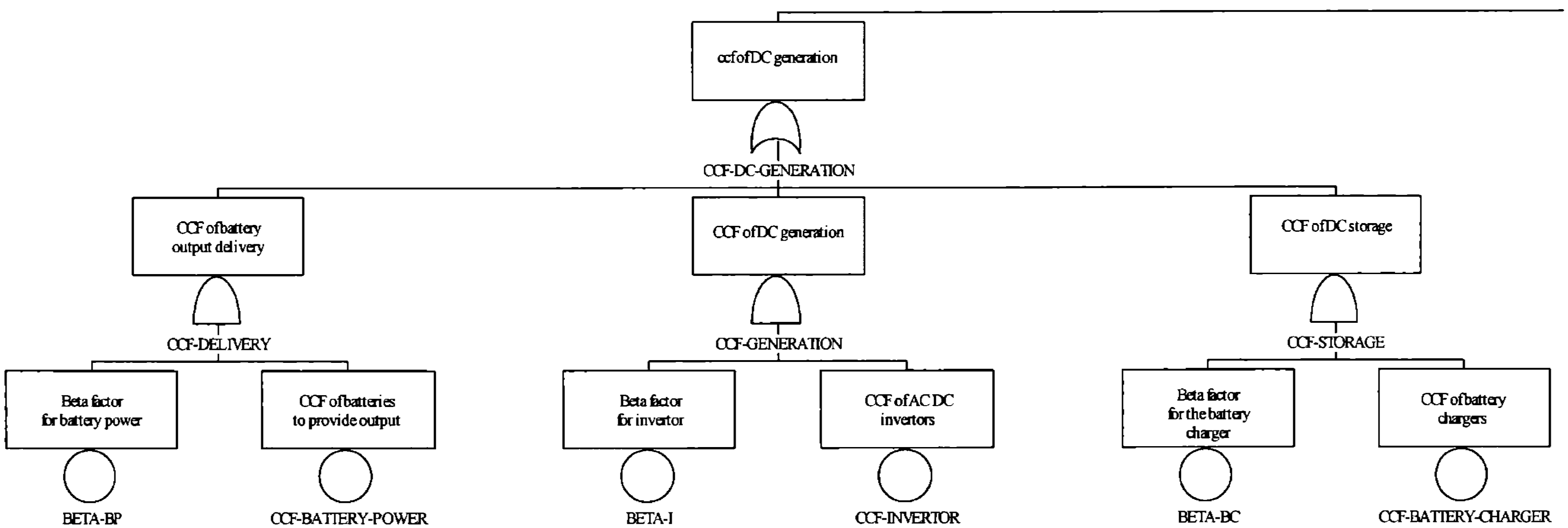
Device	Failure mode	Mean Failure Probability*	Error Factor	Source
Accumulator	All failure mode	2.40E-06	30	[23]
Check Valve	Failure to open	1.00E-04	3	[23]
Diesel	Failure to Start	1.40E-02	3	[23]
Diesel	Failure to run	5.76E-02	10	[23]
Electric motor	Failure to Start	3.75E-04	3	[21]
Electric motor	Failure to run	3.00E-04	3	[21]
Electrical Buswork	Failure during operation	4.80E-06	5	[23]
Heatric Heat Exchangers	Failure while operating	2.40E-05	10	[23]
Microturbine	Failure to run	6.00E-04	5	[24]
Offsite Power	Loss of Offsite Power	2.10E-02	3	[4]
Turbine	Failure to Start	2.00E-02	10	[23]
Turbine	Failure while running	1.44E-02	10	[23]
Blower	Physical failure while running	1.37E-06	5	[25, 26]
Electric Valve	To denergized position	1.00E-03	3	[23]
Pressure Valve	To denergized position	1.00E-03	3	[23]
Electric Switch	Failure on demand	1.00E-03	3	[23]
Generator	Failure during operation	4.80E-06	5	[23]
Reactor Trip	Failure on demand	1.00E-07	5	Estimate
DC Transmission	Failure during operation	2.40E-03	10	Estimate
Battery Power System	Failure during operation	4.80E-05	3	Estimate
Inverter	Failure during operation	4.80E-04	3	Estimate
Battery Charger	Failure during operation	1.68E-04	3	Estimate
CO2 Loop	Failure during operation	5.45E-04	10	[23]
Steam loop	Failure during operation	5.45E-04	10	[23]
WBHX	Failure while operating	2.40E-05	10	[23]
Automatic Activation	Failure on demand	1.00E-04	10	Estimate
Indication	Failure on demand	1.00E-06	10	Estimate
Manual Hardware Activation	Failure on demand	1.00E-04	10	Estimate
Operator Failure to Act	Failure on demand	1.00E-03	10	Estimate

*A run time of 24 hours per demand was used to obtain failure probabilities per demand for data given in failure rate per hour

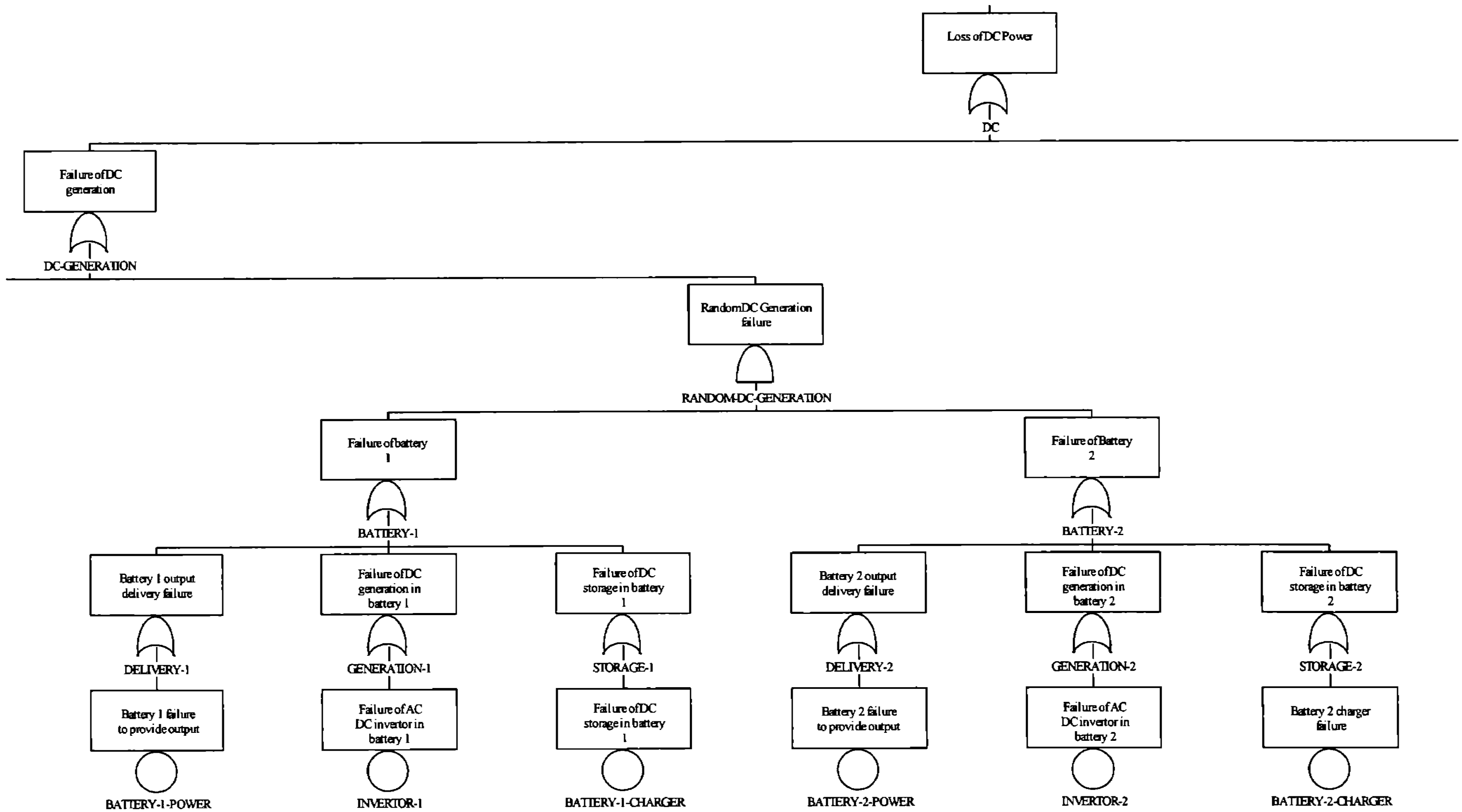
A.2 ECCS and Support System Fault Trees (2x100% capability)



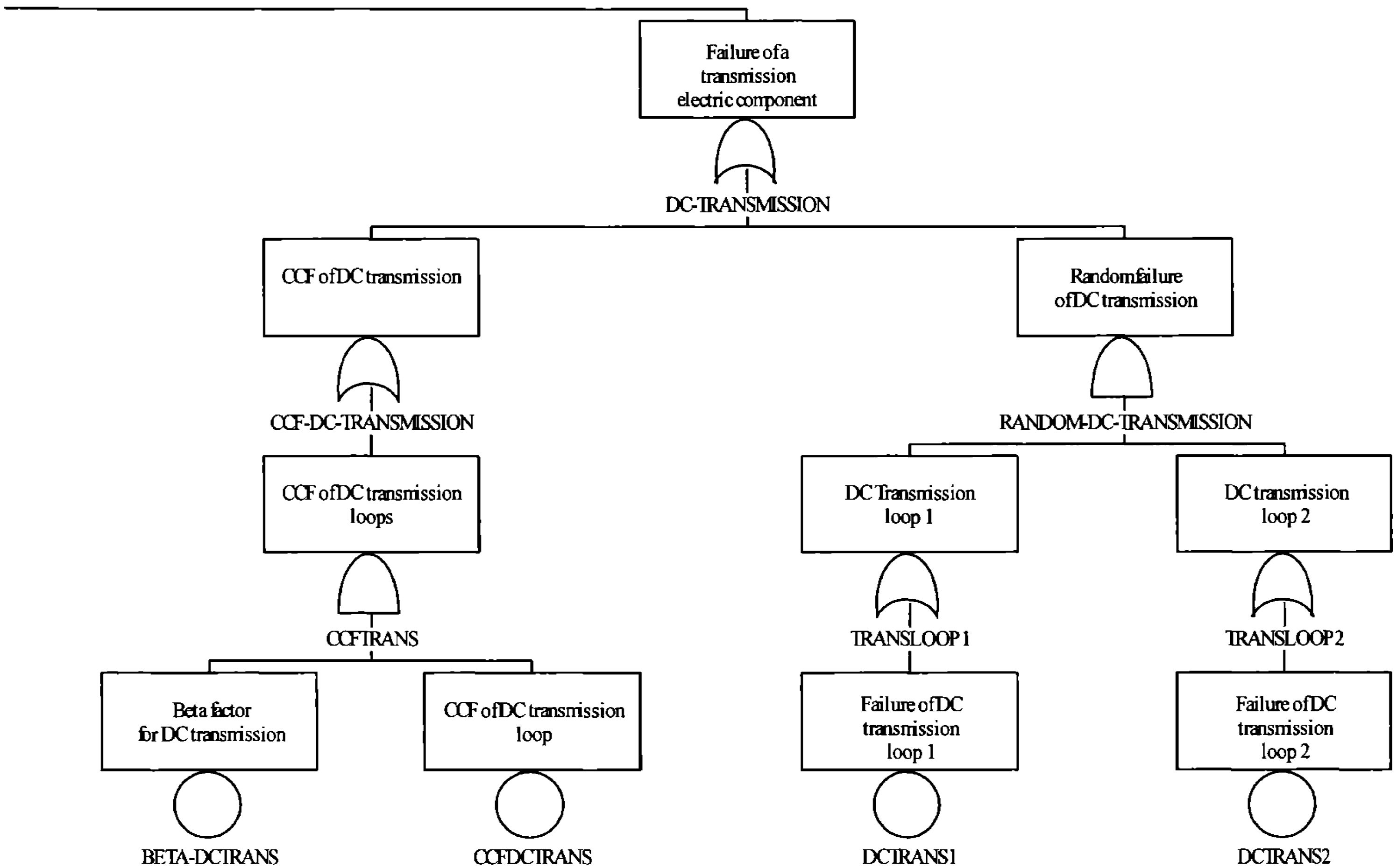
A.2.1. Onsite DC Power Fault Tree (2x100%): See Following "Zoomed In" Figures for enlarged, readable sections of the Fault Tree



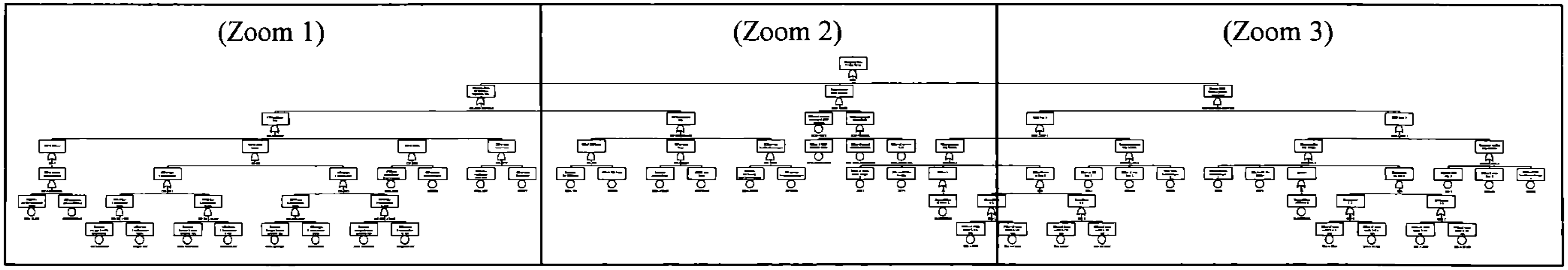
A.2.2. Onsite DC Power Fault Tree (2x100%) – Zoom 1



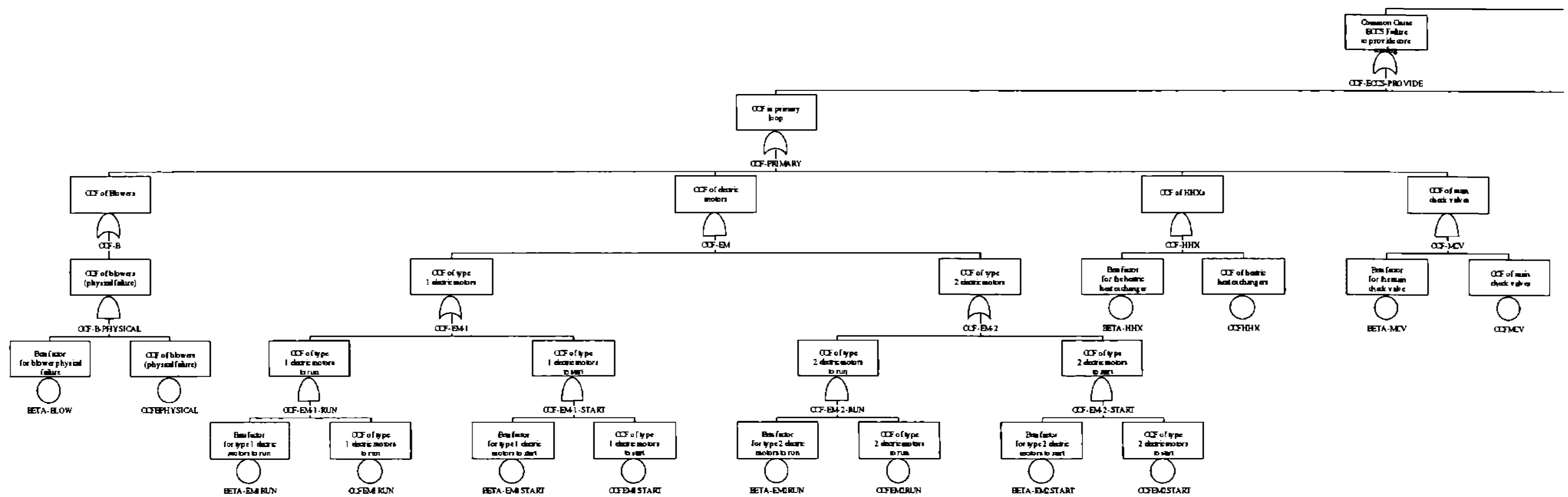
A.2.3. Onsite DC Power Fault Tree (2x100%) – Zoom 2



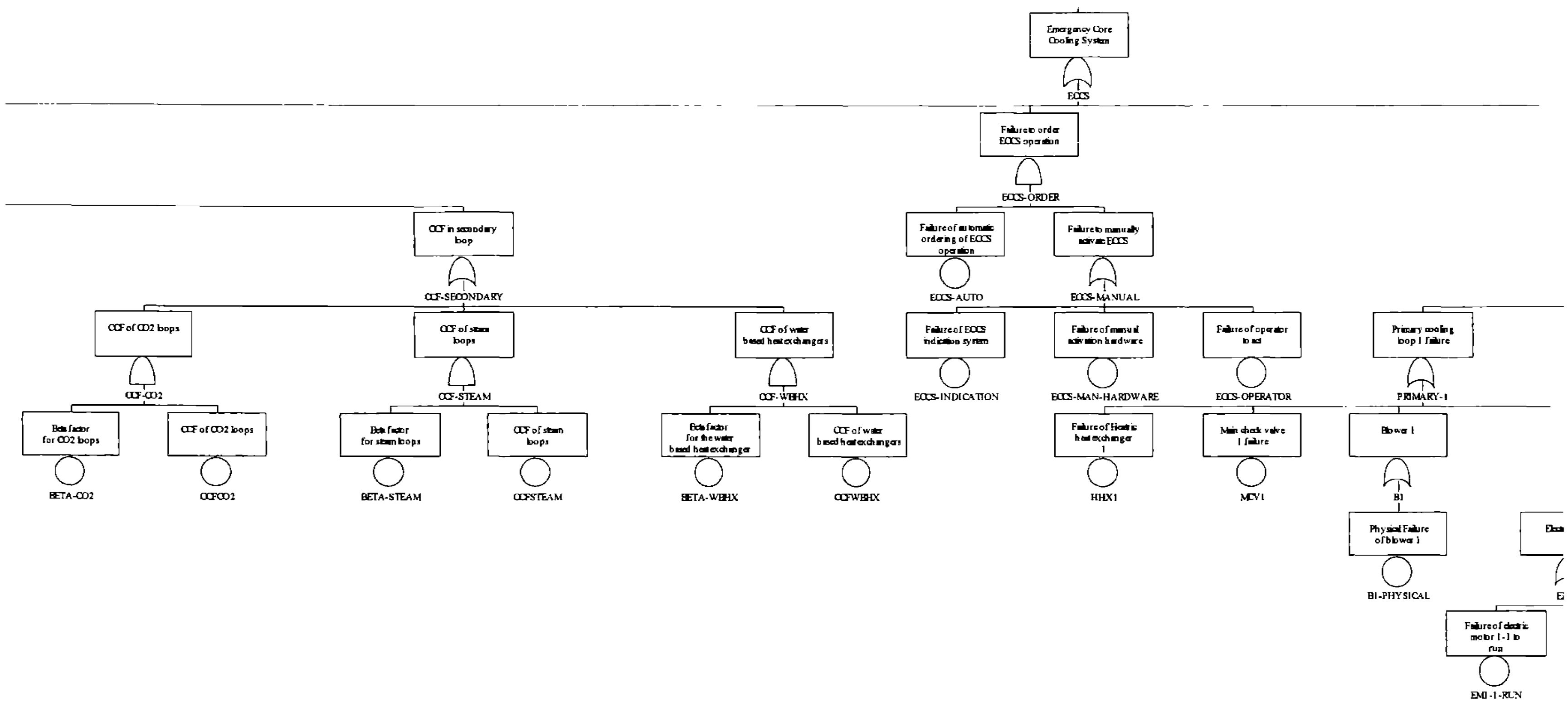
A.2.4. Onsite DC Power Fault Tree (2x100%) – Zoom 3



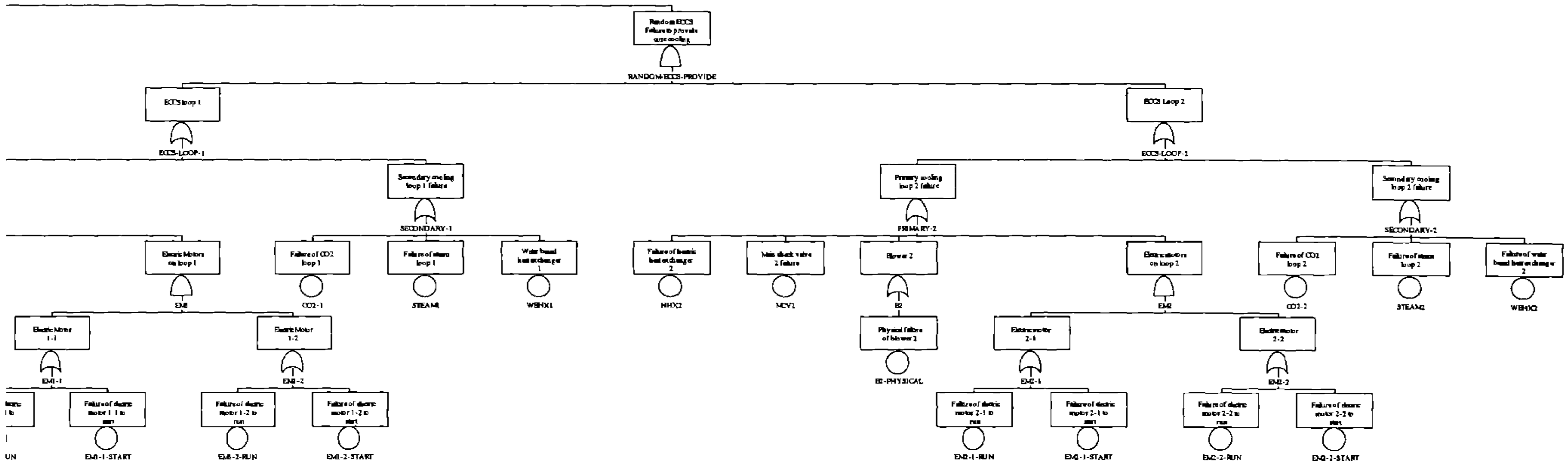
A.2.5. ECCS – Except Design 9 (2x100%): See Following "Zoomed In" Figures for enlarged, readable sections of the Fault Tree



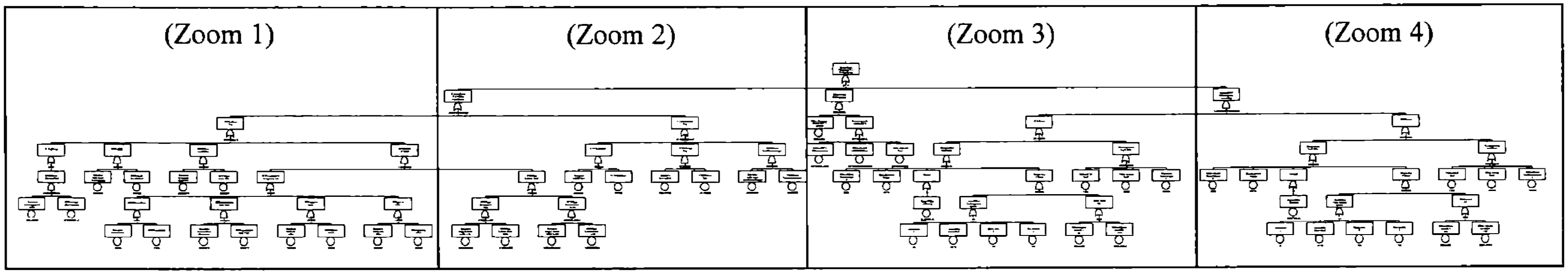
A.2.6. ECCS – Except Design 9 (2x100%) – Zoom 1



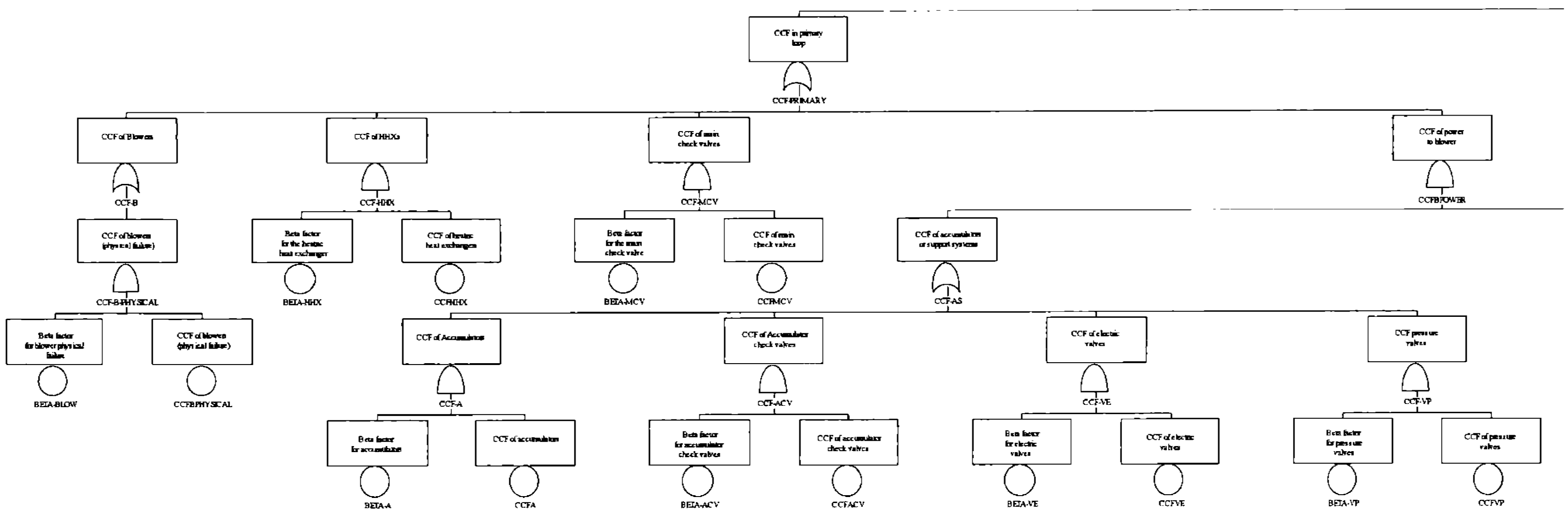
A.2.7. ECCS – Except Design 9 (2x100%) – Zoom 2



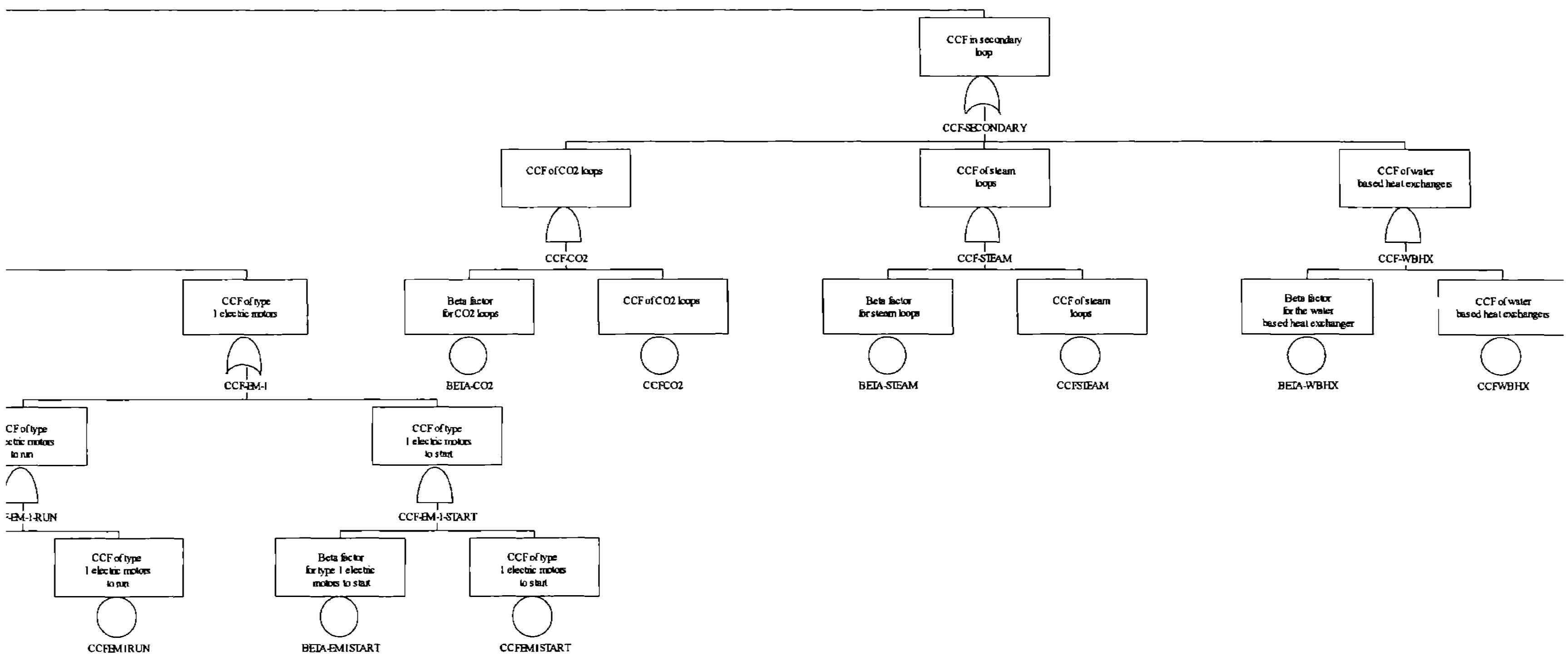
A.2.8. ECCS – Except Design 9 (2x100%) – Zoom 3



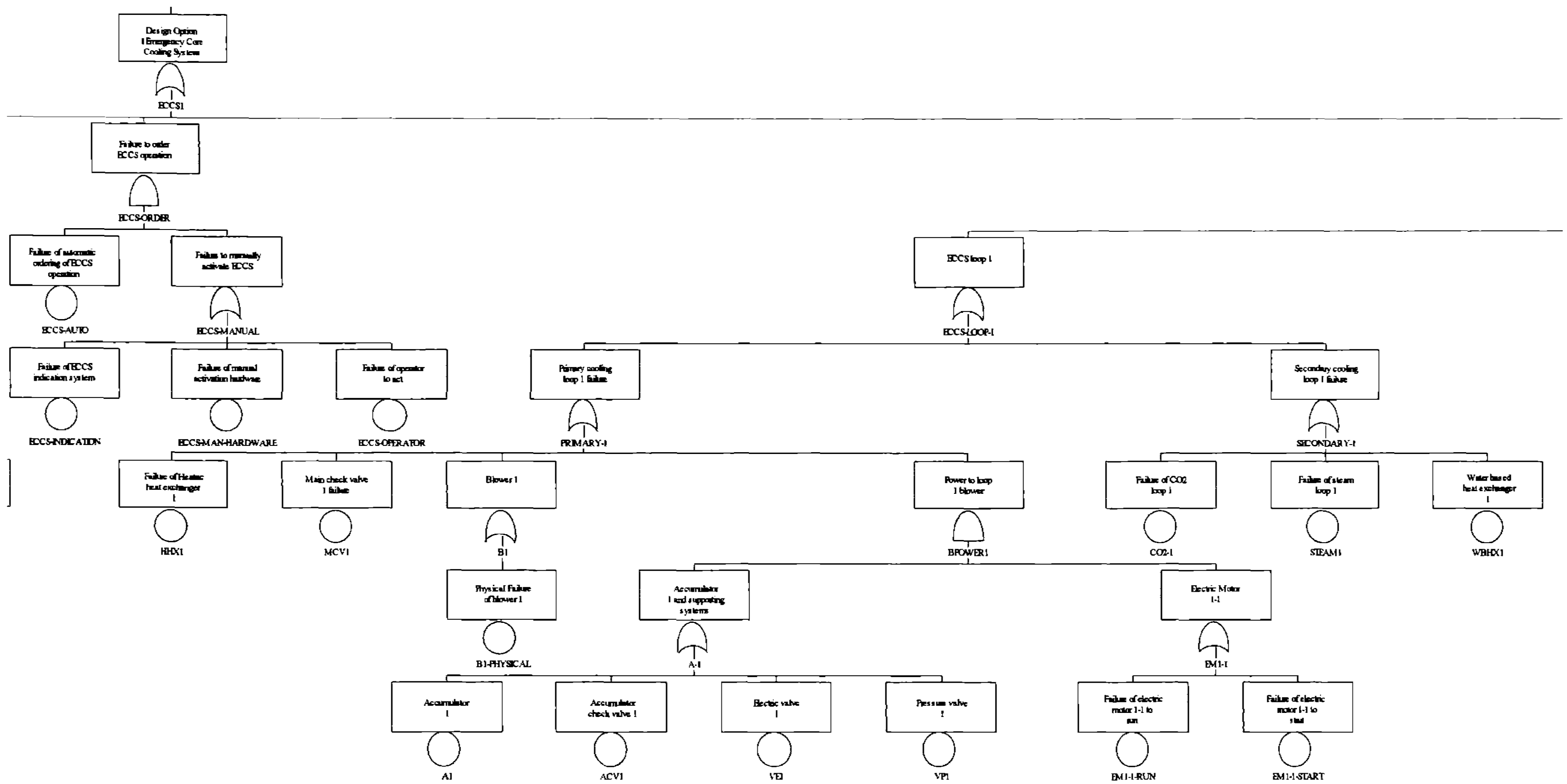
A.2.9. ECCS – Nitrogen Accumulator (Design 9) (2x100%): See Following "Zoomed In" Figures for enlarged, readable sections of the Fault Tree



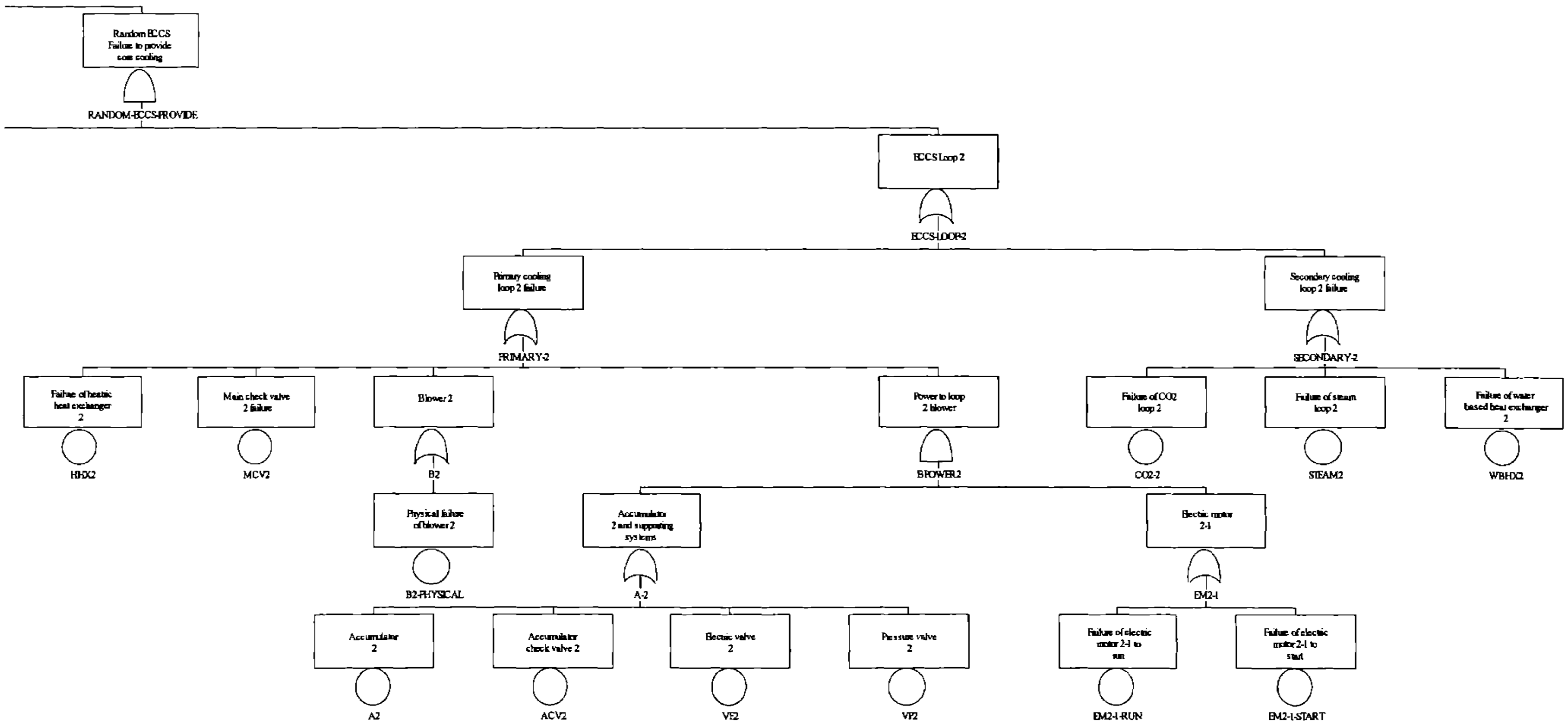
A.2.10. ECCS – Nitrogen Accumulator (Design 9) (2x100%) – Zoom 1



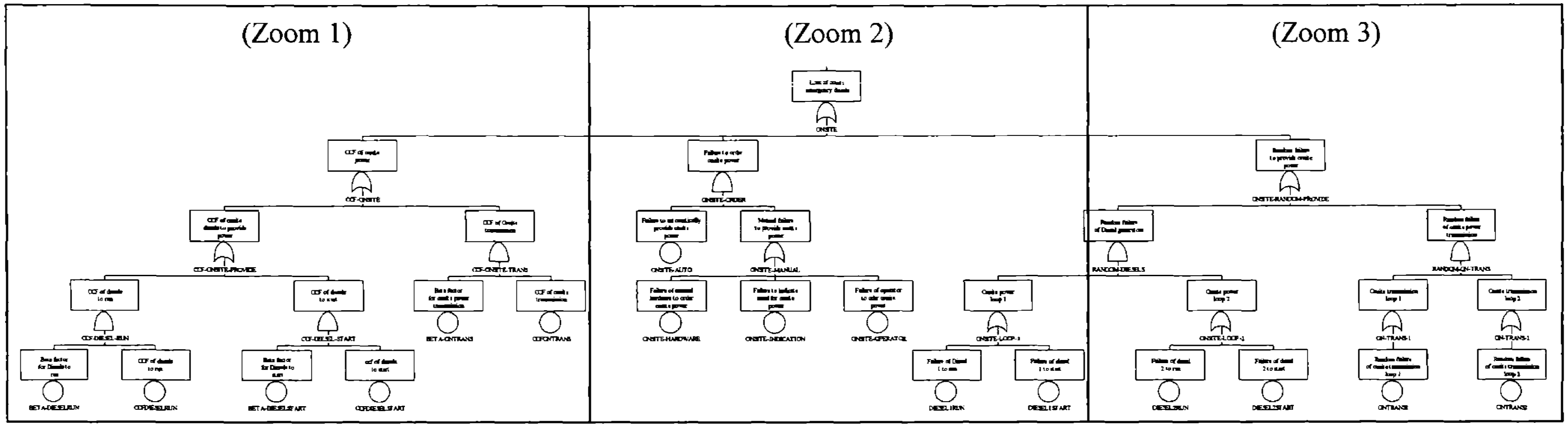
A.2.11. ECCS – Nitrogen Accumulator (Design 9) (2x100%) – Zoom 2



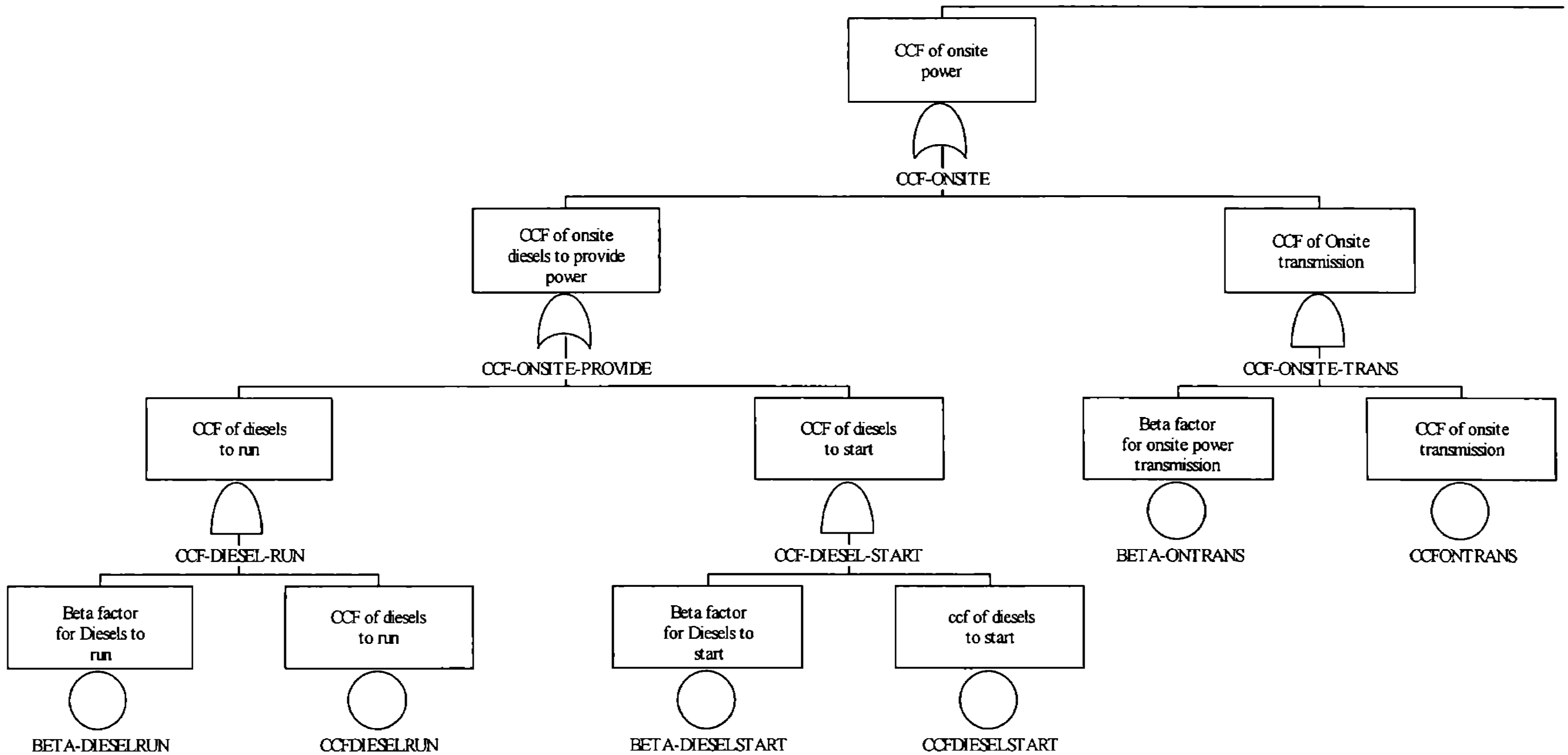
A.2.12. ECCS – Nitrogen Accumulator (Design 9) (2x100%) – Zoom 3



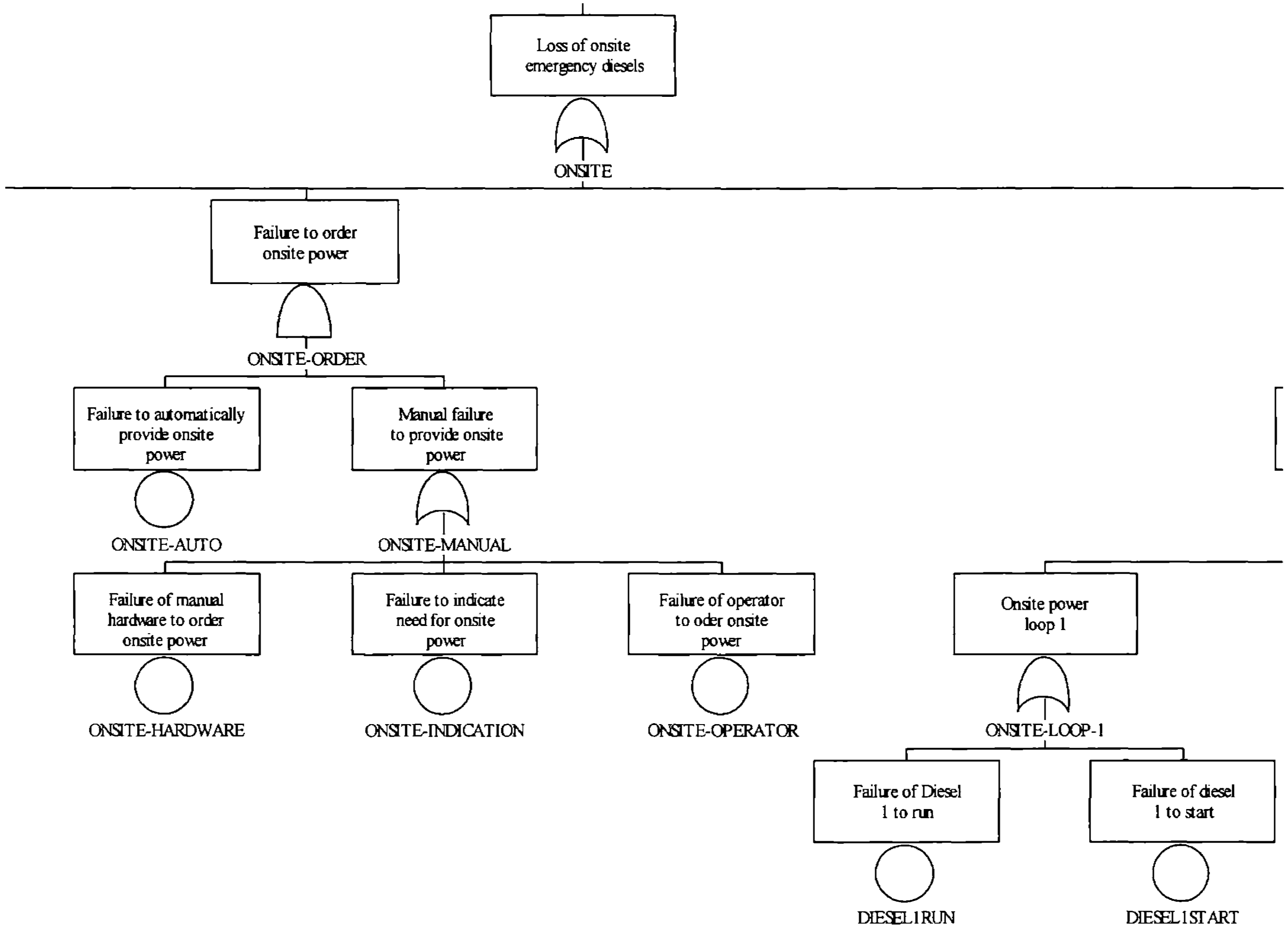
A.2.13. ECCS – Nitrogen Accumulator (Design 9) (2x100%) – Zoom 4



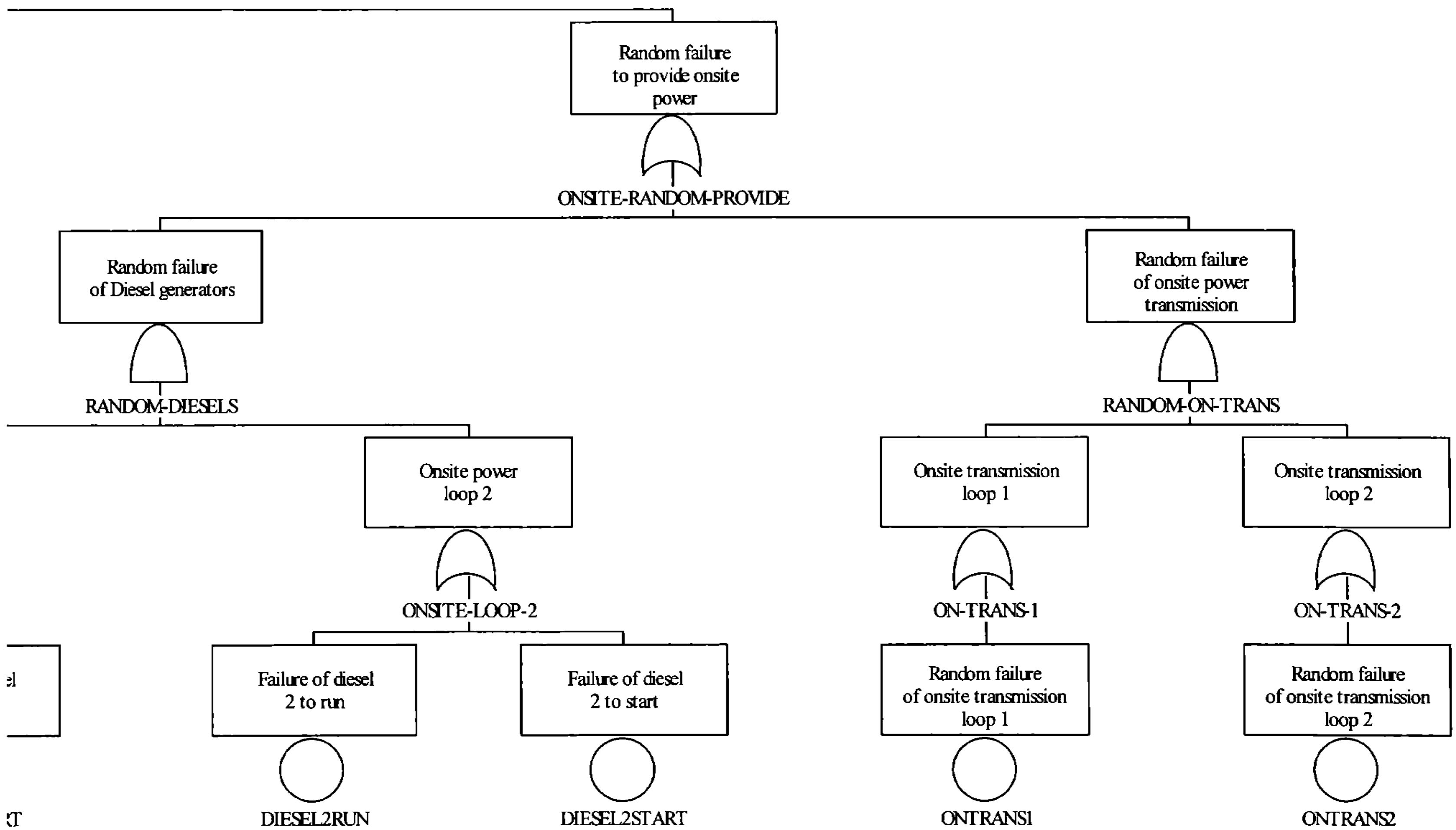
A.2.14. Onsite AC Power (2x100%): See Following "Zoomed In" Figures for enlarged, readable sections of the Fault Tree



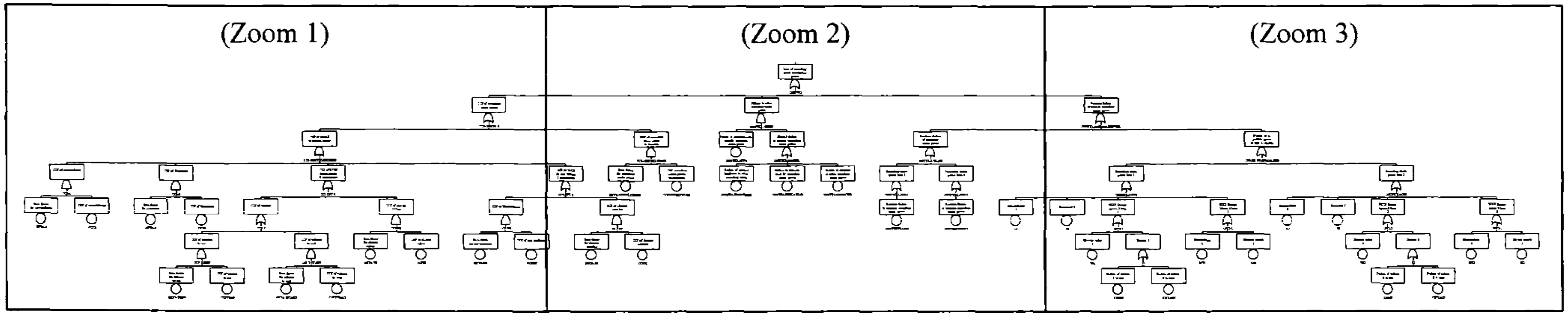
A.2.15. Onsite AC Power (2x100%) – Zoom 1



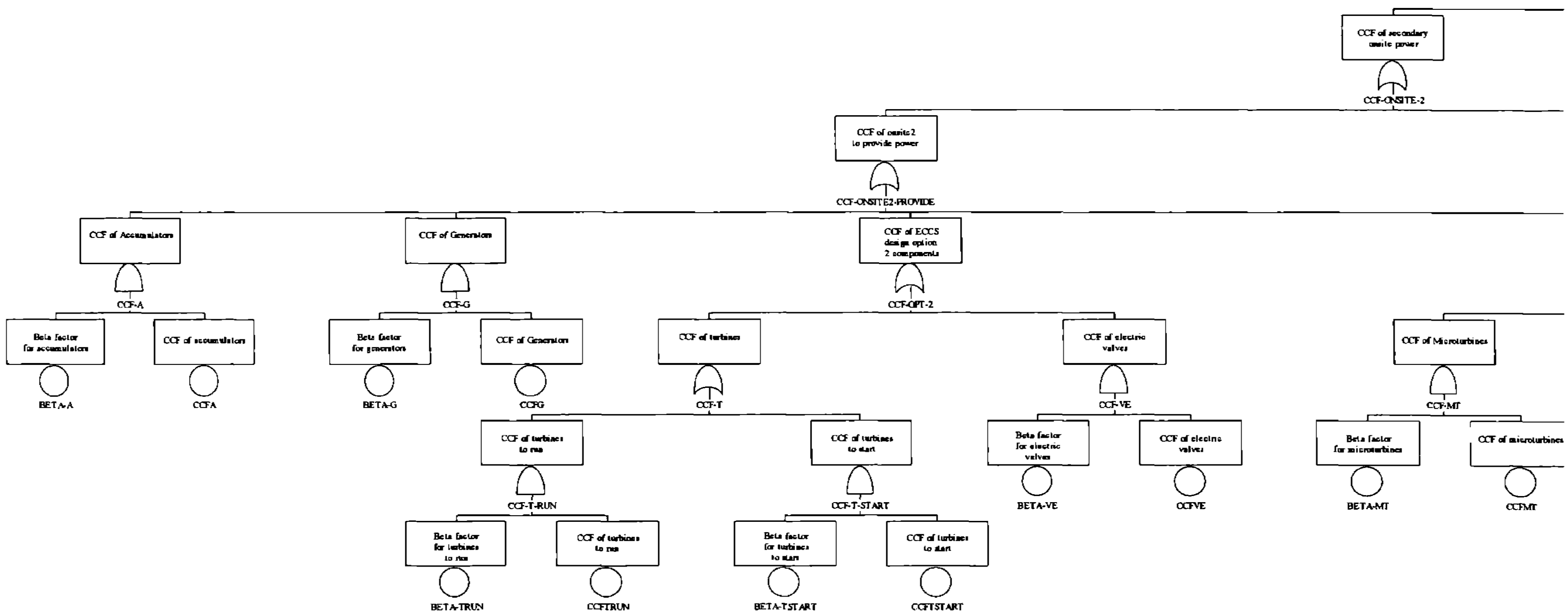
A.2.16. Onsite AC Power (2x100%) – Zoom 2



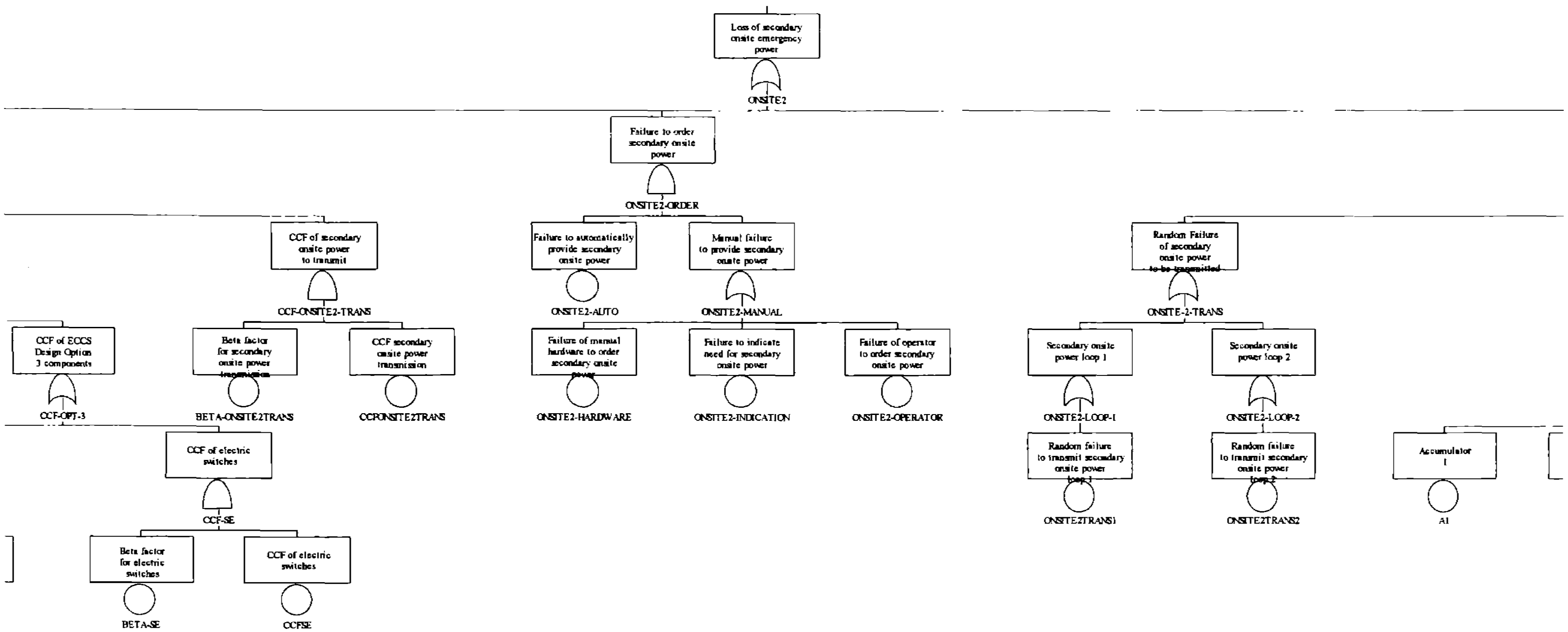
A.2.17. Onsite AC Power (2x100%) – Zoom 3



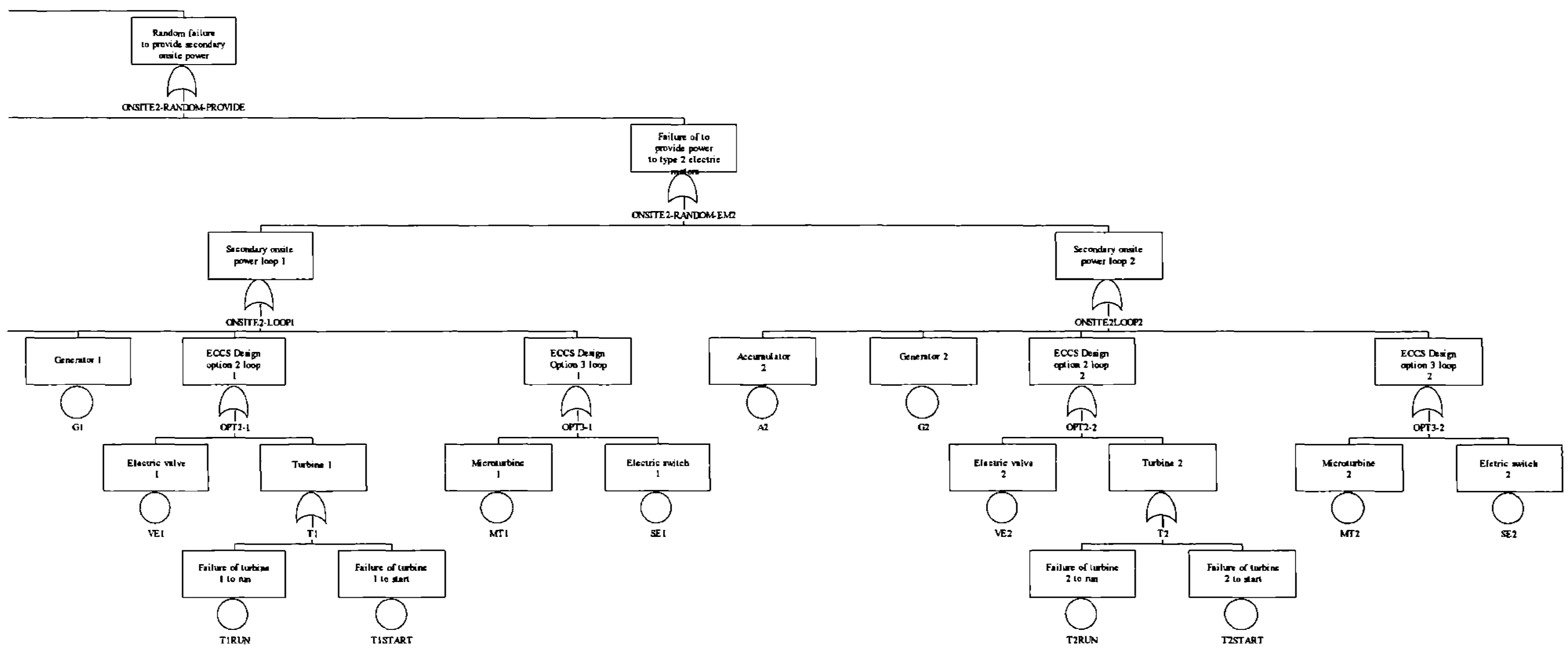
A.2.18. Secondary Onsite AC Power (2x100%): See Following "Zoomed In" Figures for enlarged, readable sections of the Fault Tree



A.2.19. Secondary Onsite AC Power (2x100%) – Zoom 1



A.2.20. Secondary Onsite AC Power (2x100%) – Zoom 2



A.2.21. Secondary Onsite AC Power (2x100%) – Zoom 3

A.3 Design Cut Sets, Importance Measures (ranked by Fussel-Vesely), and Uncertainty Data (3x100% ECCS capability)

Design 1 – Cut Sets Report (Top 99%)

Cut No.	% Total	% Cut	Frequency	Cut Sets
1	95.6	95.6	2.10E-02	IE->LOCA, LOOP, ONSITE-AUTO
				ONSITE-HARDWARE, ONSITE2-AUTO
				ONSITE2-HARDWARE, Seq->LOCA, 10
2	97.8	2.2	4.80E-04	IE->LOCA, INVERTOR-1, Seq->LOCA, 03
3	98.6	0.8	1.68E-04	IE->LOCA, BATTERY-1-CHARGER, Seq->LOCA, 03
4	99	0.5	1.00E-04	IE->LOCA, DCTRANS1, Seq->LOCA, 03

Design 1 – Importance Measures Report (Sorted by Fussel-Vesely Top 20)

Event Name	Num of Occ.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio
LOCA	501	1.00E+00	1.00E+00	-----------	1.00E+00
ONSITE-AUTO	1	1.00E+00	9.55E-01	2.23E+01	1.00E+00
ONSITE-HARDWARE	1	1.00E+00	9.55E-01	2.23E+01	1.00E+00
LOOP	1	2.10E-02	9.55E-01	2.23E+01	4.55E+01
ONSITE2-AUTO	1	1.00E+00	9.55E-01	2.23E+01	1.00E+00
ONSITE2-HARDWARE	1	1.00E+00	9.55E-01	2.23E+01	1.00E+00
INVERTOR-1	1	4.80E-04	2.14E-02	1.02E+00	4.55E+01
BATTERY-1-CHARGER	1	1.68E-04	7.48E-03	1.01E+00	4.55E+01
DCTRANS1	1	1.00E-04	4.45E-03	1.00E+00	4.55E+01
CCFSTEAM	1	5.45E-04	2.43E-03	1.00E+00	5.45E+00
CCFCO2	1	5.45E-04	2.43E-03	1.00E+00	5.45E+00
BETA-STEAM	1	1.00E-01	2.43E-03	1.00E+00	1.02E+00
BETA-CO2	1	1.00E-01	2.43E-03	1.00E+00	1.02E+00
BATTERY-1-POWER	1	4.80E-05	2.14E-03	1.00E+00	4.55E+01
BETA-EM1START	1	1.00E-01	1.67E-03	1.00E+00	1.02E+00
CCFEM1START	1	3.75E-04	1.67E-03	1.00E+00	5.45E+00
BETA-EM1RUN	1	1.00E-01	1.34E-03	1.00E+00	1.01E+00
CCFEM1RUN	1	3.00E-04	1.34E-03	1.00E+00	5.45E+00
BETA-MCV	1	1.00E-01	4.45E-04	1.00E+00	1.00E+00
CCFMCV	1	1.00E-04	4.45E-04	1.00E+00	5.45E+00

Design 1 Uncertainty Data

Distribution Quantile Level (in percent)	95% Confidence Interval on Quantile Level	Quantile Values	95% Confidence on Quantile Lower Bound	95% Confidence on Quantile Upper Bound
0.5	0.1	3.74E-03	3.50E-03	3.95E-03
1	0.2	4.33E-03	4.14E-03	4.52E-03
2.5	0.3	5.36E-03	5.23E-03	5.51E-03
5	0.4	6.44E-03	6.28E-03	6.57E-03
10	0.6	8.02E-03	7.84E-03	8.17E-03
20	0.8	1.05E-02	1.04E-02	1.07E-02
25	0.9	1.16E-02	1.14E-02	1.18E-02
30	0.9	1.27E-02	1.26E-02	1.29E-02
40	1	1.51E-02	1.48E-02	1.53E-02
50	1	1.77E-02	1.74E-02	1.80E-02
60	1	2.09E-02	2.05E-02	2.12E-02
70	0.9	2.49E-02	2.45E-02	2.53E-02
75	0.9	2.73E-02	2.69E-02	2.78E-02
80	0.8	3.03E-02	2.97E-02	3.08E-02
90	0.6	4.02E-02	3.93E-02	4.10E-02
95	0.4	5.15E-02	5.02E-02	5.30E-02
97.5	0.3	6.41E-02	6.20E-02	6.64E-02
99	0.2	8.13E-02	7.79E-02	8.62E-02
99.5	0.1	9.48E-02	8.88E-02	1.02E-01

Design 2 – Cut Sets Report (Top 99%)

Cut No.	% Total	% Cut	Frequency	Cut Sets
1	48.6	48.6	1.21E-03	IE->LOCA, DIESEL1RUN, LOOP, ONSITE2-AUTO
				ONSITE2-HARDWARE, Seq->LOCA, 10
2	67.8	19.3	4.80E-04	IE->LOCA, INVERTOR-1, Seq->LOCA, 03
3	79.6	11.8	2.94E-04	IE->LOCA, DIESEL1START, LOOP, ONSITE2-AUTO
				ONSITE2-HARDWARE, Seq->LOCA, 10
4	86.4	6.8	1.68E-04	IE->LOCA, BATTERY-1-CHARGER, Seq->LOCA, 03
5	90.4	4	1.00E-04	IE->LOCA, DCTRANS1, Seq->LOCA, 03
6	92.6	2.2	5.45E-05	IE->LOCA, BETA-CO2, CCFCO2, Seq->LOCA, 02
7	94.8	2.2	5.45E-05	IE->LOCA, BETA-STEAM, CCFSTEAM
				Seq->LOCA, 02
8	96.7	1.9	4.80E-05	IE->LOCA, BATTERY-1-POWER, Seq->LOCA, 03
9	98.2	1.5	3.75E-05	IE->LOCA, BETA-EM1START, CCFEM1START
10	99.5	1.2	3.00E-05	IE->LOCA, BETA-EM1RUN, CCFEM1RUN

Design 2 – Importance Measures Report (Sorted by Fussel-Vesely Top 20)

Event Name	Num of Occ.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio
LOCA	506	1.00E+00	1.00E+00	-----------	1.00E+00
LOOP	6	2.10E-02	6.04E-01	2.52E+00	2.88E+01
ONSITE2-HARDWARE	6	1.00E+00	6.04E-01	2.52E+00	1.00E+00
ONSITE2-AUTO	6	1.00E+00	6.04E-01	2.52E+00	1.00E+00
DIESEL1RUN	1	5.76E-02	4.85E-01	1.94E+00	8.93E+00
INVERTOR-1	1	4.80E-04	1.92E-01	1.24E+00	4.01E+02
DIESEL1START	1	1.40E-02	1.18E-01	1.13E+00	9.29E+00
BATTERY-1-CHARGER	1	1.68E-04	6.73E-02	1.07E+00	4.01E+02
DCTRANS1	1	1.00E-04	4.01E-02	1.04E+00	4.01E+02
CCFCO2	1	5.45E-04	2.18E-02	1.02E+00	4.10E+01
BETA-STEAM	1	1.00E-01	2.18E-02	1.02E+00	1.20E+00
BETA-CO2	1	1.00E-01	2.18E-02	1.02E+00	1.20E+00
CCFSTEAM	1	5.45E-04	2.18E-02	1.02E+00	4.10E+01
BATTERY-1-POWER	1	4.80E-05	1.92E-02	1.02E+00	4.01E+02
CCFEM1START	1	3.75E-04	1.50E-02	1.02E+00	4.10E+01
BETA-EM1START	1	1.00E-01	1.50E-02	1.02E+00	1.14E+00
BETA-EM1RUN	1	1.00E-01	1.20E-02	1.01E+00	1.11E+00
CCFEM1RUN	1	3.00E-04	1.20E-02	1.01E+00	4.10E+01
BETA-MCV	1	1.00E-01	4.00E-03	1.00E+00	1.04E+00
CCFMCV	1	1.00E-04	4.00E-03	1.00E+00	4.10E+01

Design 2 Uncertainty Data

Distribution Quantile Level (in percent)	95% Confidence Interval on Quantile Level	Quantile Values	95% Confidence on Quantile Lower Bound	95% Confidence on Quantile Upper Bound
0.5	0.1	5.46E-04	5.26E-04	5.69E-04
1	0.2	6.05E-04	5.83E-04	6.28E-04
2.5	0.3	7.02E-04	6.86E-04	7.16E-04
5	0.4	7.95E-04	7.79E-04	8.09E-04
10	0.6	9.23E-04	9.08E-04	9.36E-04
20	0.8	1.13E-03	1.11E-03	1.14E-03
25	0.9	1.22E-03	1.20E-03	1.23E-03
30	0.9	1.30E-03	1.29E-03	1.32E-03
40	1	1.49E-03	1.47E-03	1.51E-03
50	1	1.70E-03	1.68E-03	1.72E-03
60	1	1.95E-03	1.92E-03	1.98E-03
70	0.9	2.30E-03	2.27E-03	2.35E-03
75	0.9	2.58E-03	2.52E-03	2.63E-03
80	0.8	2.91E-03	2.86E-03	2.98E-03
90	0.6	4.14E-03	4.02E-03	4.29E-03
95	0.4	5.85E-03	5.65E-03	6.18E-03
97.5	0.3	6.41E-02	6.20E-02	6.64E-02
99	0.2	8.13E-02	7.79E-02	8.62E-02
99.5	0.1	9.48E-02	8.88E-02	1.02E-01

Design 3 – Cut Sets Report (Top 99%)

Cut No.	% Total	% Cut	Frequency	Cut Sets
1	64.8	64.8	1.21E-03	IE->LOCA, DIESEL1RUN, LOOP, ONSITE2-AU
				ONSITE2-HARDWARE, Seq->LOCA, 10
2	80.6	15.8	2.94E-04	IE->LOCA, DIESEL1START, LOOP, ONSITE2-
				ONSITE2-HARDWARE, Seq->LOCA, 10
3	85.9	5.4	1.00E-04	IE->LOCA, DCTRANS1, Seq->LOCA, 03
4	88.9	2.9	5.45E-05	IE->LOCA, BETA-CO2, CCFCO2, Seq->LOCA,
5	91.8	2.9	5.45E-05	IE->LOCA, BETA-STEAM, CCFSTEAM
				Seq->LOCA, 02
6	94.4	2.6	4.80E-05	IE->LOCA, BETA-I, CCF-INVERTOR
				Seq->LOCA, 03
7	96.4	2	3.75E-05	IE->LOCA, BETA-EM1START, CCFEM1START
				Seq->LOCA, 02
8	98	1.6	3.00E-05	IE->LOCA, BETA-EM1RUN, CCFEM1RUN
				Seq->LOCA, 02
9	98.9	0.9	1.68E-05	IE->LOCA, BETA-BC, CCF-BATTERY-CHARGER
				Seq->LOCA, 03
10	99.4	0.5	1.00E-05	IE->LOCA, BETA-MCV, CCFMCV, Seq->LOCA,

Design 3 – Importance Measures Report (Sorted by Fussel-Vesely Top 20)

Event Name	Num of Occ.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio
LOCA	515	1.00E+00	1.00E+00	-----------	1.00E+00
ONSITE2-HARDW	6	1.00E+00	8.06E-01	5.16E+00	1.00E+00
ONSITE2-AUTO	6	1.00E+00	8.06E-01	5.16E+00	1.00E+00
LOOP	6	2.10E-02	8.06E-01	5.16E+00	3.82E+01
DIESEL1RUN	1	5.76E-02	6.48E-01	2.84E+00	1.16E+01
DIESEL1START	1	1.40E-02	1.57E-01	1.19E+00	1.21E+01
DCTRANS1	1	1.00E-04	5.35E-02	1.06E+00	5.36E+02
BETA-CO2	1	1.00E-01	2.92E-02	1.03E+00	1.26E+00
CCFCO2	1	5.45E-04	2.92E-02	1.03E+00	5.45E+01
CCFSTEAM	1	5.45E-04	2.92E-02	1.03E+00	5.45E+01
BETA-STEAM	1	1.00E-01	2.92E-02	1.03E+00	1.26E+00
CCF-INVERTOR	1	4.80E-04	2.57E-02	1.03E+00	5.45E+01
BETA-I	1	1.00E-01	2.57E-02	1.03E+00	1.23E+00
CCFEM1START	1	3.75E-04	2.01E-02	1.02E+00	5.45E+01
BETA-EM1START	1	1.00E-01	2.01E-02	1.02E+00	1.18E+00
BETA-EM1RUN	1	1.00E-01	1.60E-02	1.02E+00	1.14E+00
CCFEM1RUN	1	3.00E-04	1.60E-02	1.02E+00	5.45E+01
CCF-BATTERY-C	1	1.68E-04	8.98E-03	1.01E+00	5.45E+01
BETA-BC	1	1.00E-01	8.98E-03	1.01E+00	1.08E+00
BETA-MCV	1	1.00E-01	5.35E-03	1.01E+00	1.05E+00

Design 3 Uncertainty Data

Distribution Quantile Level (in percent)	95% Confidence Interval on Quantile Level	Quantile Values	95% Confidence on Quantile Lower Bound	95% Confidence on Quantile Upper Bound
0.5	0.1	2.34E-04	2.17E-04	2.41E-04
1	0.2	2.57E-04	2.48E-04	2.67E-04
2.5	0.3	3.08E-04	3.00E-04	3.16E-04
5	0.4	3.63E-04	3.55E-04	3.71E-04
10	0.6	4.38E-04	4.30E-04	4.48E-04
20	0.8	5.70E-04	5.59E-04	5.82E-04
25	0.9	6.34E-04	6.23E-04	6.44E-04
30	0.9	6.96E-04	6.87E-04	7.09E-04
40	1	8.41E-04	8.24E-04	8.58E-04
50	1	1.01E-03	9.94E-04	1.03E-03
60	1	1.23E-03	1.20E-03	1.26E-03
70	0.9	1.56E-03	1.53E-03	1.60E-03
75	0.9	1.80E-03	1.76E-03	1.85E-03
80	0.8	2.12E-03	2.07E-03	2.19E-03
90	0.6	3.40E-03	3.29E-03	3.55E-03
95	0.4	5.29E-03	5.04E-03	5.52E-03
97.5	0.3	7.90E-03	7.27E-03	8.45E-03
99	0.2	1.28E-02	1.18E-02	1.45E-02
99.5	0.1	1.92E-02	1.65E-02	2.19E-02

Design 4 – Cut Sets Report (Top 99%)

Cut No.	% Total	% Cut	Frequency	Cut Sets
1	19.5	19.5	1.21E-04	IE->LOCA, BETA-DIESELRUN, CCFDIESELRUN
				LOOP, ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
2	35.6	16.1	1.00E-04	IE->LOCA, DCTRANS1, Seq->LOCA, 03
3	46.8	11.2	6.97E-05	IE->LOCA, DIESEL1RUN, DIESEL2RUN, LOOP
				ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
4	55.5	8.8	5.45E-05	IE->LOCA, BETA-CO2, CCFCO2, Seq->LOCA, 02
5	64.3	8.8	5.45E-05	IE->LOCA, BETA-STEAM, CCFSTEAM
				Seq->LOCA, 02
6	72	7.7	4.80E-05	IE->LOCA, BETA-I, CCF-INVERTOR
				Seq->LOCA, 03
7	78.1	6	3.75E-05	IE->LOCA, BETA-EM1START, CCFEM1START
				Seq->LOCA, 02
8	82.9	4.8	3.00E-05	IE->LOCA, BETA-EM1RUN, CCFEM1RUN
				Seq->LOCA, 02
9	87.6	4.7	2.94E-05	IE->LOCA, BETA-DIESELSTART, CCFDIESELSTART
				LOOP, ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
10	90.4	2.7	1.69E-05	IE->LOCA, DIESEL1START, DIESEL2RUN, LOOP
				ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
11	93.1	2.7	1.69E-05	IE->LOCA, DIESEL1RUN, DIESEL2START, LOOP
				ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
12	95.8	2.7	1.68E-05	IE->LOCA, BETA-BC, CCF-BATTERY-CHARGER
				Seq->LOCA, 03
13	97.4	1.6	1.00E-05	IE->LOCA, BETA-MCV, CCFMCV, Seq->LOCA, 02
14	98.2	0.8	4.80E-06	IE->LOCA, BETA-BP, CCF-BATTERY-POWER
				Seq->LOCA, 03
15	98.9	0.7	4.12E-06	IE->LOCA, DIESEL1START, DIESEL2START, LOOP
				ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
16	99.3	0.4	2.40E-06	IE->LOCA, BETA-HHX, CCFHHX, Seq->LOCA, 02

Design 4 – Importance Measures Report (Sorted by Fussel-Vesely Top 20)

Event Name	Num of Occ.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio
LOCA	519	1.00E+00	1.00E+00	-----------	1.00E+00
ONSITE2-AUTO	10	1.00E+00	4.18E-01	1.72E+00	1.00E+00
LOOP	10	2.10E-02	4.18E-01	1.72E+00	2.04E+01
ONSITE2-HARDWARE	10	1.00E+00	4.18E-01	1.72E+00	1.00E+00
CCFDIESELRUN	1	5.76E-02	1.95E-01	1.24E+00	4.18E+00
BETA-DIESELRUN	1	1.00E-01	1.95E-01	1.24E+00	2.75E+00
DCTRANS1	1	1.00E-04	1.61E-01	1.19E+00	1.61E+03
DIESEL1RUN	2	5.76E-02	1.39E-01	1.16E+00	3.28E+00
DIESEL2RUN	2	5.76E-02	1.39E-01	1.16E+00	3.28E+00
CCFCO2	1	5.45E-04	8.76E-02	1.10E+00	1.62E+02
BETA-CO2	1	1.00E-01	8.76E-02	1.10E+00	1.79E+00
BETA-STEAM	1	1.00E-01	8.76E-02	1.10E+00	1.79E+00
CCFSTEAM	1	5.45E-04	8.76E-02	1.10E+00	1.62E+02
BETA-I	1	1.00E-01	7.72E-02	1.08E+00	1.69E+00
CCF-INVERTOR	1	4.80E-04	7.72E-02	1.08E+00	1.62E+02
CCFEM1START	1	3.75E-04	6.03E-02	1.06E+00	1.62E+02
BETA-EM1START	1	1.00E-01	6.03E-02	1.06E+00	1.54E+00
BETA-EM1RUN	1	1.00E-01	4.82E-02	1.05E+00	1.43E+00
CCFEM1RUN	1	3.00E-04	4.82E-02	1.05E+00	1.62E+02
CCFDIESELSTART	1	1.40E-02	4.73E-02	1.05E+00	4.33E+00

Design 4 Uncertainty Data

Distribution Quantile Level (in percent)	95% Confidence Interval on Quantile Level	Quantile Values	95% Confidence on Quantile Lower Bound	95% Confidence on Quantile Upper Bound
0.5	0.1	2.34E-04	2.17E-04	2.41E-04
1	0.2	2.57E-04	2.48E-04	2.67E-04
2.5	0.3	3.08E-04	3.00E-04	3.16E-04
5	0.4	3.63E-04	3.55E-04	3.71E-04
10	0.6	4.38E-04	4.30E-04	4.48E-04
20	0.8	5.70E-04	5.59E-04	5.82E-04
25	0.9	6.34E-04	6.23E-04	6.44E-04
30	0.9	6.96E-04	6.87E-04	7.09E-04
40	1	8.41E-04	8.24E-04	8.58E-04
50	1	1.01E-03	9.94E-04	1.03E-03
60	1	1.23E-03	1.20E-03	1.26E-03
70	0.9	1.56E-03	1.53E-03	1.60E-03
75	0.9	1.80E-03	1.76E-03	1.85E-03
80	0.8	2.12E-03	2.07E-03	2.19E-03
90	0.6	3.40E-03	3.29E-03	3.55E-03
95	0.4	5.29E-03	5.04E-03	5.52E-03
97.5	0.3	7.90E-03	7.27E-03	8.45E-03
99	0.2	1.28E-02	1.18E-02	1.45E-02
99.5	0.1	1.92E-02	1.65E-02	2.19E-02

Design 5 – Cut Sets Report (Top 99%)

Cut No.	% Total	% Cut	Frequency	Cut Sets
1	22.8	22.8	1.21E-04	IE->LOCA, BETA-DIESELRUN, CCFDIESELRUN
				LOOP, ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
2	35.9	13.1	6.97E-05	IE->LOCA, DIESEL1RUN, DIESEL2RUN, LOOP
				ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
3	46.1	10.3	5.45E-05	IE->LOCA, BETA-CO2, CCFCO2, Seq->LOCA, 02
4	56.4	10.3	5.45E-05	IE->LOCA, BETA-STEAM, CCFSTEAM
				Seq->LOCA, 02
5	65.4	9	4.80E-05	IE->LOCA, BETA-I, CCF-INVERTOR
				Seq->LOCA, 03
6	72.5	7.1	3.75E-05	IE->LOCA, BETA-EM1START, CCFEM1START
				Seq->LOCA, 02
7	78.1	5.7	3.00E-05	IE->LOCA, BETA-EM1RUN, CCFEM1RUN
				Seq->LOCA, 02
8	83.6	5.5	2.94E-05	IE->LOCA, BETA-DIESELSTART, CCFDIESELSTART
				LOOP, ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
9	86.8	3.2	1.69E-05	IE->LOCA, DIESEL1START, DIESEL2RUN, LOOP
				ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
10	90	3.2	1.69E-05	IE->LOCA, DIESEL1RUN, DIESEL2START, LOOP
				ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
11	93.2	3.2	1.68E-05	IE->LOCA, BETA-BC, CCF-BATTERY-CHARGER
				Seq->LOCA, 03
12	95.1	1.9	1.00E-05	IE->LOCA, BETA-MCV, CCFMCV, Seq->LOCA, 02
13	97	1.9	1.00E-05	IE->LOCA, BETA-DCTRANS, CCFDCTRANS
				Seq->LOCA, 03
14	97.9	0.9	4.80E-06	IE->LOCA, BETA-BP, CCF-BATTERY-POWER
				Seq->LOCA, 03
15	98.6	0.8	4.12E-06	IE->LOCA, DIESEL1START, DIESEL2START, LOOP
				ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
16	99.1	0.5	2.40E-06	IE->LOCA, BETA-HHX, CCFHHX, Seq->LOCA, 02

Design 5 – Importance Measures Report (Sorted by Fussel-Vesely Top 20)

Event Name	Num of Occ.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio
LOCA	520	1.00E+00	1.00E+00	-----------	1.00E+00
ONSITE2-HARDWARE	10	1.00E+00	4.89E-01	1.96E+00	1.00E+00
LOOP	10	2.10E-02	4.89E-01	1.96E+00	2.37E+01
ONSITE2-AUTO	10	1.00E+00	4.89E-01	1.96E+00	1.00E+00
CCFDIESELRUN	1	5.76E-02	2.27E-01	1.29E+00	4.72E+00
BETA-DIESELRUN	1	1.00E-01	2.27E-01	1.29E+00	3.05E+00
DIESEL1RUN	2	5.76E-02	1.63E-01	1.19E+00	3.66E+00
DIESEL2RUN	2	5.76E-02	1.63E-01	1.19E+00	3.66E+00
CCFCO2	1	5.45E-04	1.02E-01	1.11E+00	1.89E+02
BETA-CO2	1	1.00E-01	1.02E-01	1.11E+00	1.92E+00
CCFSTEAM	1	5.45E-04	1.02E-01	1.11E+00	1.89E+02
BETA-STEAM	1	1.00E-01	1.02E-01	1.11E+00	1.92E+00
BETA-I	1	1.00E-01	9.02E-02	1.10E+00	1.81E+00
CCF-INVERTOR	1	4.80E-04	9.02E-02	1.10E+00	1.89E+02
CCFEM1START	1	3.75E-04	7.05E-02	1.08E+00	1.89E+02
BETA-EM1START	1	1.00E-01	7.05E-02	1.08E+00	1.63E+00
BETA-EM1RUN	1	1.00E-01	5.64E-02	1.06E+00	1.51E+00
CCFEM1RUN	1	3.00E-04	5.64E-02	1.06E+00	1.89E+02
CCFDIESELSTART	1	1.40E-02	5.53E-02	1.06E+00	4.89E+00
BETA-DIESELSTART	1	1.00E-01	5.53E-02	1.06E+00	1.50E+00

Design 5 Uncertainty Data

Distribution Quantile Level (in percent)	95% Confidence Interval on Quantile Level	Quantile Values	95% Confidence on Quantile Lower Bound	95% Confidence on Quantile Upper Bound
0.5	0.1	1.06E-04	1.02E-04	1.11E-04
1	0.2	1.18E-04	1.14E-04	1.22E-04
2.5	0.3	1.36E-04	1.34E-04	1.39E-04
5	0.4	1.55E-04	1.52E-04	1.58E-04
10	0.6	1.83E-04	1.80E-04	1.86E-04
20	0.8	2.23E-04	2.20E-04	2.26E-04
25	0.9	2.45E-04	2.41E-04	2.48E-04
30	0.9	2.66E-04	2.62E-04	2.69E-04
40	1	3.07E-04	3.02E-04	3.11E-04
50	1	3.57E-04	3.52E-04	3.62E-04
60	1	4.21E-04	4.14E-04	4.28E-04
70	0.9	5.07E-04	4.97E-04	5.18E-04
75	0.9	5.69E-04	5.57E-04	5.80E-04
80	0.8	6.64E-04	6.45E-04	6.77E-04
90	0.6	1.07E-03	1.03E-03	1.12E-03
95	0.4	1.75E-03	1.65E-03	1.86E-03
97.5	0.3	3.10E-03	2.81E-03	3.39E-03
99	0.2	6.58E-03	5.82E-03	7.84E-03
99.5	0.1	1.03E-02	8.83E-03	1.23E-02

Design 6 – Cut Sets Report (Top 99%)

Cut No.	% Total	% Cut	Frequency	Cut Sets
1	28	28	1.21E-04	IE->LOCA, BETA-DIESELRUN, CCFDIESELRUN
				LOOP, ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
2	40.6	12.6	5.45E-05	IE->LOCA, BETA-CO2, CCFCO2, Seq->LOCA,
3	53.3	12.6	5.45E-05	IE->LOCA, BETA-STEAM, CCFSTEAM
				Seq->LOCA, 02
4	64.4	11.1	4.80E-05	IE->LOCA, BETA-I, CCF-INVERTOR
				Seq->LOCA, 03
5	73.1	8.7	3.75E-05	IE->LOCA, BETA-EM1START, CCFEM1START
				Seq->LOCA, 02
6	80	7	3.00E-05	IE->LOCA, BETA-EM1RUN, CCFEM1RUN
				Seq->LOCA, 02
7	86.8	6.8	2.94E-05	IE->LOCA, BETA-DIESELSTART, CCFDIESELS
				LOOP, ONSITE2-AUTO, ONSITE2-HARDWARE
				Seq->LOCA, 10
8	90.7	3.9	1.68E-05	IE->LOCA, BETA-BC, CCF-BATTERY-CHARGER
				Seq->LOCA, 03
9	93.1	2.3	1.00E-05	IE->LOCA, BETA-MCV, CCFMCV, Seq->LOCA,
10	95.4	2.3	1.00E-05	IE->LOCA, BETA-DCTRANS, CCFDCTRANS
				Seq->LOCA, 03
11	96.5	1.1	4.80E-06	IE->LOCA, BETA-BP, CCF-BATTERY-POWER
				Seq->LOCA, 03
12	97.4	0.9	4.01E-06	IE->LOCA, DIESEL1RUN, DIESEL2RUN
				DIESEL3RUN, LOOP, ONSITE2-AUTO
				ONSITE2-HARDWARE, Seq->LOCA, 10
13	98	0.6	2.40E-06	IE->LOCA, BETA-HHX, CCFHHX, Seq->LOCA,
14	98.6	0.6	2.40E-06	IE->LOCA, BETA-WBHX, CCFWBHX, Seq->LOC
15	99	0.5	2.10E-06	IE->LOCA, LOOP, ONSITE2-AUTO

Design 6 – Importance Measures Report (Sorted by Fussel-Vesely Top 20)

Event Name	Num of Occ.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio
LOCA	524	1.00E+00	1.00E+00	-----------	1.00E+00
LOOP	14	2.10E-02	3.71E-01	1.59E+00	1.83E+01
ONSITE2-AUTO	14	1.00E+00	3.71E-01	1.59E+00	1.00E+00
ONSITE2-HARDWARE	14	1.00E+00	3.71E-01	1.59E+00	1.00E+00
CCFDIESELRUN	1	5.76E-02	2.80E-01	1.39E+00	5.58E+00
BETA-DIESELRUN	1	1.00E-01	2.80E-01	1.39E+00	3.52E+00
CCFCO2	1	5.45E-04	1.26E-01	1.14E+00	2.32E+02
CCFSTEAM	1	5.45E-04	1.26E-01	1.14E+00	2.32E+02
BETA-CO2	1	1.00E-01	1.26E-01	1.14E+00	2.14E+00
BETA-STEAM	1	1.00E-01	1.26E-01	1.14E+00	2.14E+00
BETA-I	1	1.00E-01	1.11E-01	1.13E+00	2.00E+00
CCF-INVERTOR	1	4.80E-04	1.11E-01	1.13E+00	2.32E+02
CCFEM1START	1	3.75E-04	8.68E-02	1.10E+00	2.32E+02
BETA-EM1START	1	1.00E-01	8.68E-02	1.10E+00	1.78E+00
BETA-EM1RUN	1	1.00E-01	6.94E-02	1.08E+00	1.63E+00
CCFEM1RUN	1	3.00E-04	6.94E-02	1.08E+00	2.32E+02
CCFDIESELSTART	1	1.40E-02	6.81E-02	1.07E+00	5.79E+00
BETA-DIESELSTART	1	1.00E-01	6.81E-02	1.07E+00	1.61E+00
BETA-BC	1	1.00E-01	3.89E-02	1.04E+00	1.35E+00
CCF-BATTERY-CHARGER	1	1.68E-04	3.89E-02	1.04E+00	2.32E+02

Design 6 Uncertainty Data

Distribution Quantile Level (in percent)	95% Confidence Interval on Quantile Level	Quantile Values	95% Confidence on Quantile Lower Bound	95% Confidence on Quantile Upper Bound
0.5	0.1	1.01E-04	9.82E-05	1.05E-04
1	0.2	1.11E-04	1.09E-04	1.15E-04
2.5	0.3	1.30E-04	1.26E-04	1.32E-04
5	0.4	1.45E-04	1.43E-04	1.48E-04
10	0.6	1.70E-04	1.68E-04	1.73E-04
20	0.8	2.07E-04	2.05E-04	2.10E-04
25	0.9	2.25E-04	2.21E-04	2.27E-04
30	0.9	2.42E-04	2.38E-04	2.45E-04
40	1	2.79E-04	2.74E-04	2.82E-04
50	1	3.21E-04	3.16E-04	3.25E-04
60	1	3.69E-04	3.64E-04	3.74E-04
70	0.9	4.32E-04	4.26E-04	4.39E-04
75	0.9	4.76E-04	4.70E-04	4.85E-04
80	0.8	5.35E-04	5.24E-04	5.46E-04
90	0.6	7.71E-04	7.52E-04	8.04E-04
95	0.4	1.18E-03	1.12E-03	1.25E-03
97.5	0.3	1.89E-03	1.72E-03	2.05E-03
99	0.2	4.11E-03	3.52E-03	4.72E-03
99.5	0.1	6.86E-03	5.66E-03	8.07E-03

Design 7 – Cut Sets Report (Top 99%)

Cut No.	% Total	% Cut	Frequency	Cut Sets
1	26	26	5.45E-05	IE->LOCA, BETA-CO2, CCFCO2, Seq->LOCA,
2	51.9	26	5.45E-05	IE->LOCA, BETA-STEAM, CCFSTEAM
				Seq->LOCA, 02
3	74.8	22.9	4.80E-05	IE->LOCA, BETA-I, CCF-INVERTOR
				Seq->LOCA, 03
4	82.8	8	1.68E-05	IE->LOCA, BETA-BC, CCF-BATTERY-CHARGER
				Seq->LOCA, 03
5	87.6	4.8	1.00E-05	IE->LOCA, BETA-MCV, CCFMCV, Seq->LOCA,
6	92.4	4.8	1.00E-05	IE->LOCA, BETA-DCTRANS, CCFDCTRANS
				Seq->LOCA, 03
7	94.6	2.3	4.80E-06	IE->LOCA, BETA-BP, CCF-BATTERY-POWER
				Seq->LOCA, 03
8	95.8	1.2	2.42E-06	IE->LOCA, BETA-DIESELRUN, CCFDIESELRUN
				LOOP, T1START, Seq->LOCA, 10
9	97	1.2	2.40E-06	IE->LOCA, BETA-HHX, CCFHHX, Seq->LOCA,
10	98.1	1.2	2.40E-06	IE->LOCA, BETA-WBHX, CCFWBHX, Seq->LOC
11	98.9	0.8	1.74E-06	IE->LOCA, BETA-DIESELRUN, CCFDIESELRUN
				LOOP, T1RUN, Seq->LOCA, 10
12	99.2	0.3	5.88E-07	IE->LOCA, BETA-DIESELSTART, CCFDIESELS

Design 7 – Importance Measures Report (Sorted by Fussel-Vesely Top 20)

Event Name	Num of Occ.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio
LOCA	526	1.00E+00	1.00E+00	-----------	1.00E+00
CCFCO2	1	5.45E-04	2.60E-01	1.35E+00	4.77E+02
CCFSTEAM	1	5.45E-04	2.60E-01	1.35E+00	4.77E+02
BETA-STEAM	1	1.00E-01	2.60E-01	1.35E+00	3.34E+00
BETA-CO2	1	1.00E-01	2.60E-01	1.35E+00	3.34E+00
BETA-I	1	1.00E-01	2.29E-01	1.30E+00	3.06E+00
CCF-INVERTOR	1	4.80E-04	2.29E-01	1.30E+00	4.77E+02
BETA-BC	1	1.00E-01	8.00E-02	1.09E+00	1.72E+00
CCF-BATTERY-C	1	1.68E-04	8.00E-02	1.09E+00	4.77E+02
CCFDCTRANS	1	1.00E-04	4.76E-02	1.05E+00	4.77E+02
CCFMCV	1	1.00E-04	4.76E-02	1.05E+00	4.77E+02
BETA-DCTRANS	1	1.00E-01	4.76E-02	1.05E+00	1.43E+00
BETA-MCV	1	1.00E-01	4.76E-02	1.05E+00	1.43E+00
LOOP	103	2.10E-02	2.71E-02	1.03E+00	2.26E+00
BETA-BP	1	1.00E-01	2.29E-02	1.02E+00	1.21E+00
CCF-BATTERY-P	1	4.80E-05	2.29E-02	1.02E+00	4.77E+02
CCFDIESELRUN	9	5.76E-02	2.05E-02	1.02E+00	1.34E+00
BETA-DIESELRU	9	1.00E-01	2.05E-02	1.02E+00	1.18E+00
T1START	14	2.00E-02	1.53E-02	1.02E+00	1.75E+00
CCFWBHX	1	2.40E-05	1.14E-02	1.01E+00	4.77E+02

Design 7 Uncertainty Data

Distribution Quantile Level (in percent)	95% Confidence Interval on Quantile Level	Quantile Values	95% Confidence on Quantile Lower Bound	95% Confidence on Quantile Upper Bound
0.5	0.1	3.98E-05	3.71E-05	4.13E-05
1	0.2	4.52E-05	4.30E-05	4.63E-05
2.5	0.3	5.44E-05	5.22E-05	5.55E-05
5	0.4	6.25E-05	6.16E-05	6.36E-05
10	0.6	7.60E-05	7.49E-05	7.71E-05
20	0.8	9.57E-05	9.41E-05	9.70E-05
25	0.9	1.05E-04	1.03E-04	1.06E-04
30	0.9	1.13E-04	1.11E-04	1.14E-04
40	1	1.32E-04	1.30E-04	1.34E-04
50	1	1.54E-04	1.51E-04	1.56E-04
60	1	1.80E-04	1.77E-04	1.83E-04
70	0.9	2.15E-04	2.12E-04	2.19E-04
75	0.9	2.38E-04	2.34E-04	2.43E-04
80	0.8	2.68E-04	2.62E-04	2.74E-04
90	0.6	3.86E-04	3.74E-04	3.95E-04
95	0.4	5.29E-04	5.07E-04	5.53E-04
97.5	0.3	7.46E-04	7.10E-04	7.90E-04
99	0.2	1.14E-03	1.05E-03	1.23E-03
99.5	0.1	1.48E-03	1.35E-03	1.74E-03

Design 8 – Cut Sets Report (Top 99%)

Cut No.	% Total	% Cut	Frequency	Cut Sets
1	26.5	26.5	5.45E-05	IE->LOCA, BETA-CO2, CCFCO2, Seq->LOCA,
2	53	26.5	5.45E-05	IE->LOCA, BETA-STEAM, CCFSTEAM
				Seq->LOCA, 02
3	76.3	23.3	4.80E-05	IE->LOCA, BETA-I, CCF-INVERTOR
				Seq->LOCA, 03
4	84.5	8.2	1.68E-05	IE->LOCA, BETA-BC, CCF-BATTERY-CHARGER
				Seq->LOCA, 03
5	89.3	4.9	1.00E-05	IE->LOCA, BETA-MCV, CCFMCV, Seq->LOCA,
6	94.2	4.9	1.00E-05	IE->LOCA, BETA-DCTRANS, CCFDCTRANS
				Seq->LOCA, 03
7	96.5	2.3	4.80E-06	IE->LOCA, BETA-BP, CCF-BATTERY-POWER
				Seq->LOCA, 03
8	97.7	1.2	2.40E-06	IE->LOCA, BETA-HHX, CCFHHX, Seq->LOCA,
9	98.9	1.2	2.40E-06	IE->LOCA, BETA-WBHX, CCFWBHX, Seq->LOC
10	99	0.2	2.97E-07	IE->LOCA, CO2-2, STEAM1, Seq->LOCA, 02

Design 8 – Importance Measures Report (Sorted by Fussel-Vesely Top 20)

Event Name	Num of Occ.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio
LOCA	325	1.00E+00	1.00E+00	9.27E+11	1.00E+00
CCFCO2	1	5.45E-04	2.65E-01	1.36E+00	4.87E+02
BETA-STEAM	1	1.00E-01	2.65E-01	1.36E+00	3.38E+00
BETA-CO2	1	1.00E-01	2.65E-01	1.36E+00	3.38E+00
CCFSTEAM	1	5.45E-04	2.65E-01	1.36E+00	4.87E+02
CCF-INVERTOR	1	4.80E-04	2.33E-01	1.30E+00	4.87E+02
BETA-I	1	1.00E-01	2.33E-01	1.30E+00	3.10E+00
CCF-BATTERY-CHARGER	1	1.68E-04	8.16E-02	1.09E+00	4.87E+02
BETA-BC	1	1.00E-01	8.16E-02	1.09E+00	1.74E+00
CCFDCTRANS	1	1.00E-04	4.86E-02	1.05E+00	4.87E+02
BETA-DCTRANS	1	1.00E-01	4.86E-02	1.05E+00	1.44E+00
BETA-MCV	1	1.00E-01	4.86E-02	1.05E+00	1.44E+00
CCFMCV	1	1.00E-04	4.86E-02	1.05E+00	4.87E+02
BETA-BP	1	1.00E-01	2.33E-02	1.02E+00	1.21E+00
CCF-BATTERY-POWER	1	4.80E-05	2.33E-02	1.02E+00	4.87E+02
BETA-HHX	1	1.00E-01	1.17E-02	1.01E+00	1.11E+00
CCFWBHX	1	2.40E-05	1.17E-02	1.01E+00	4.87E+02
BETA-WBHX	1	1.00E-01	1.17E-02	1.01E+00	1.11E+00
CCFHHX	1	2.40E-05	1.17E-02	1.01E+00	4.87E+02
STEAM1	10	5.45E-04	3.28E-03	1.00E+00	7.02E+00

Design 8 Uncertainty Data

Distribution Quantile Level (in percent)	95% Confidence Interval on Quantile Level	Quantile Values	95% Confidence on Quantile Lower Bound	95% Confidence on Quantile Upper Bound
0.5	0.1	3.76E-05	3.64E-05	3.94E-05
1	0.2	4.29E-05	4.10E-05	4.40E-05
2.5	0.3	5.12E-05	5.03E-05	5.25E-05
5	0.4	5.97E-05	5.85E-05	6.12E-05
10	0.6	7.28E-05	7.16E-05	7.37E-05
20	0.8	9.07E-05	8.94E-05	9.23E-05
25	0.9	9.97E-05	9.82E-05	1.01E-04
30	0.9	1.09E-04	1.07E-04	1.11E-04
40	1	1.27E-04	1.25E-04	1.28E-04
50	1	1.47E-04	1.45E-04	1.50E-04
60	1	1.72E-04	1.69E-04	1.76E-04
70	0.9	2.08E-04	2.04E-04	2.12E-04
75	0.9	2.32E-04	2.27E-04	2.37E-04
80	0.8	2.64E-04	2.58E-04	2.70E-04
90	0.6	3.72E-04	3.63E-04	3.83E-04
95	0.4	5.24E-04	4.99E-04	5.46E-04
97.5	0.3	7.22E-04	6.91E-04	7.67E-04
99	0.2	1.09E-03	9.99E-04	1.20E-03
99.5	0.1	1.52E-03	1.31E-03	1.80E-03

Design 9 LOCA – Cut Sets Report (Top 99%)

Cut No.	% Total	% Cut	Frequency	Cut Sets
1	42.2	42.2	1.00E-07	IE->LOCA, R-TRIP, Seq->OPT1LOCA, 13
2	84.4	42.2	1.00E-07	IE->LOCA, ECCS-AUTO, ECCS-OPERATOR
				Seq->OPT1LOCA, 02
3	89.5	5.1	1.21E-08	IE->LOCA, BETA-DIESELRUN, BETA-VP
				CCFDIESELRUN, CCFVP, LOOP
				Seq->OPT1LOCA, 10
4	93.7	4.2	1.00E-08	IE->LOCA, ECCS-AUTO, ECCS-MAN-HARDWARE
				Seq->OPT1LOCA, 02
5	95.7	2	4.80E-09	IE->LOCA, BETA-I, BETA-VP, CCF-INVERTOR
				CCFVP, Seq->OPT1LOCA, 04
6	97	1.2	2.94E-09	IE->LOCA, BETA-DIESELSTART, BETA-VP
				CCFDIESELSTART, CCFVP, LOOP
				Seq->OPT1LOCA, 10
7	97.7	0.7	1.68E-09	IE->LOCA, BETA-BC, BETA-VP
				CCF-BATTERY-CHARGER, CCFVP
				Seq->OPT1LOCA, 04
8	98.2	0.5	1.21E-09	IE->LOCA, BETA-ACV, BETA-DIESELRUN, CCFACV
				CCFDIESELRUN, LOOP, Seq->OPT1LOCA, 10
9	98.6	0.4	1.00E-09	IE->LOCA, BETA-DCTRANS, BETA-VP, CCFDCTRANS
				CCFVP, Seq->OPT1LOCA, 04
10	98.8	0.2	4.80E-10	IE->LOCA, BETA-BP, BETA-VP
				CCF-BATTERY-POWER, CCFVP, Seq->OPT1LOCA, 04
11	99	0.2	4.80E-10	IE->LOCA, BETA-ACV, BETA-I, CCF-INVERTOR

Design 9 LOCA – Importance Measures Report (Sorted by Fussel-Vesely Top 20)

Event Name	Num of Occ.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio
LOCA	345	1.00E+00	1.00E+00	-----------	1.00E+00
ECCS-AUTO	3	1.00E-04	4.64E-01	1.87E+00	4.64E+03
R-TRIP	1	1.00E-07	4.22E-01	1.73E+00	4.22E+06
ECCS-OPERATOR	1	1.00E-03	4.22E-01	1.73E+00	4.22E+02
BETA-VP	27	1.00E-01	1.01E-01	1.11E+00	1.91E+00
CCFVP	27	1.00E-03	1.01E-01	1.11E+00	1.02E+02
LOOP	182	2.10E-02	7.60E-02	1.08E+00	4.55E+00
BETA-DIESELRUN	18	1.00E-01	5.74E-02	1.06E+00	1.52E+00
CCFDIESELRUN	18	5.76E-02	5.74E-02	1.06E+00	1.94E+00
ECCS-MAN-HARDWARE	1	1.00E-04	4.22E-02	1.04E+00	4.23E+02
BETA-I	18	1.00E-01	2.28E-02	1.02E+00	1.21E+00
CCF-INVERTOR	18	4.80E-04	2.28E-02	1.02E+00	4.85E+01
BETA-DIESELSTART	18	1.00E-01	1.40E-02	1.01E+00	1.13E+00
CCFDIESELSTART	18	1.40E-02	1.40E-02	1.01E+00	1.98E+00
CCFACV	27	1.00E-04	1.01E-02	1.01E+00	1.02E+02
BETA-ACV	27	1.00E-01	1.01E-02	1.01E+00	1.09E+00
BETA-BC	18	1.00E-01	7.98E-03	1.01E+00	1.07E+00
CCF-BATTERY-CHARGER	18	1.68E-04	7.98E-03	1.01E+00	4.85E+01
CCFDCTRANS	18	1.00E-04	4.75E-03	1.01E+00	4.85E+01
BETA-DCTRANS	18	1.00E-01	4.75E-03	1.01E+00	1.04E+00

Design 9 LOCA - Uncertainty Data

Distribution Quantile Level (in percent)	95% Confidence Interval on Quantile Level	Quantile Values	95% Confidence on Quantile Lower Bound	95% Confidence on Quantile Upper Bound
0.5	0.1	2.92E-08	2.76E-08	3.11E-08
1	0.2	3.49E-08	3.38E-08	3.60E-08
2.5	0.3	4.30E-08	4.16E-08	4.43E-08
5	0.4	5.25E-08	5.10E-08	5.41E-08
10	0.6	6.57E-08	6.43E-08	6.71E-08
20	0.8	8.58E-08	8.42E-08	8.74E-08
25	0.9	9.56E-08	9.39E-08	9.72E-08
30	0.9	1.05E-07	1.04E-07	1.07E-07
40	1	1.27E-07	1.25E-07	1.29E-07
50	1	1.51E-07	1.48E-07	1.54E-07
60	1	1.79E-07	1.77E-07	1.82E-07
70	0.9	2.17E-07	2.13E-07	2.22E-07
75	0.9	2.44E-07	2.39E-07	2.49E-07
80	0.8	2.80E-07	2.73E-07	2.86E-07
90	0.6	4.19E-07	4.06E-07	4.35E-07
95	0.4	6.54E-07	6.19E-07	6.91E-07
97.5	0.3	1.03E-06	9.56E-07	1.15E-06
99	0.2	2.06E-06	1.80E-06	2.44E-06
99.5	0.1	2.92E-06	2.75E-06	3.25E-06

Design 9 ECCS Loop LOCA – Cut Sets Report (Top 99%)

Cut No.	% Total	% Cut	Frequency	Cut Sets
1	43.4	43.4	2.73E-08	IE->ECCSLOOPLOCA, RBETA-STEAM, RCCFSTEAM
				Seq->ECCSLOOPLOCA, 02
2	86.7	43.4	2.73E-08	IE->ECCSLOOPLOCA, RBETA-CO2, RCCFCO2
				Seq->ECCSLOOPLOCA, 02
3	94.7	8	5.00E-09	IE->ECCSLOOPLOCA, RBETA-MCV, RCCFMCV
				Seq->ECCSLOOPLOCA, 02
4	96.6	1.9	1.20E-09	IE->ECCSLOOPLOCA, RBETA-WBHX, RCCFWBHX
				Seq->ECCSLOOPLOCA, 02
5	98.5	1.9	1.20E-09	IE->ECCSLOOPLOCA, RBETA-HHX, RCCFHHX
				Seq->ECCSLOOPLOCA, 02
6	98.7	0.2	1.49E-10	IE->ECCSLOOPLOCA, RCO2-2, RSTEAM1
				Seq->ECCSLOOPLOCA, 02
7	99	0.2	1.49E-10	IE->ECCSLOOPLOCA, RCO2-1, RSTEAM2

Design 9 ECCS Loop LOCA – Importance Measures Report

Event Name	Num of Occ.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio
ECCSLOOPLOCA	295	5.00E-04	1.00E+00	-----------	2.00E+03
RCCFCO2	1	5.45E-04	4.33E-01	1.77E+00	7.96E+02
RBETA-CO2	1	1.00E-01	4.33E-01	1.77E+00	4.90E+00
RBETA-STEAM	1	1.00E-01	4.33E-01	1.77E+00	4.90E+00
RCCFSTEAM	1	5.45E-04	4.33E-01	1.77E+00	7.96E+02
RCCFMCV	1	1.00E-04	7.95E-02	1.09E+00	7.96E+02
RBETA-MCV	1	1.00E-01	7.95E-02	1.09E+00	1.72E+00
RBETA-WBHX	1	1.00E-01	1.91E-02	1.02E+00	1.17E+00
RCCFHHX	1	2.40E-05	1.91E-02	1.02E+00	7.96E+02
RBETA-HHX	1	1.00E-01	1.91E-02	1.02E+00	1.17E+00
RCCFWBHX	1	2.40E-05	1.91E-02	1.02E+00	7.96E+02
RSTEAM2	28	5.45E-04	5.38E-03	1.01E+00	1.09E+01
RCO2-2	28	5.45E-04	5.38E-03	1.01E+00	1.09E+01
RSTEAM1	28	5.45E-04	5.38E-03	1.01E+00	1.09E+01
RCO2-1	28	5.45E-04	5.38E-03	1.01E+00	1.09E+01
RBETA-BLOW	1	1.00E-01	1.09E-03	1.00E+00	1.01E+00
RCCFBPHYSIC	1	1.37E-06	1.09E-03	1.00E+00	7.96E+02
RMCV1	18	1.00E-04	9.87E-04	1.00E+00	1.09E+01
RECCS-AUTO	3	1.00E-04	8.76E-04	1.00E+00	9.75E+00
RECCS-OPERA	1	1.00E-03	7.95E-04	1.00E+00	1.79E+00

Design 9 ECCS Loop LOCA - Uncertainty Data

Distribution Quantile Level (in percent)	95% Confidence Interval on Quantile Level	Quantile Values	95% Confidence on Quantile Lower Bound	95% Confidence on Quantile Upper Bound
0.5	0.1	1.89E-10	1.65E-10	2.10E-10
1	0.2	2.82E-10	2.47E-10	3.07E-10
2.5	0.3	4.90E-10	4.49E-10	5.23E-10
5	0.4	8.45E-10	7.88E-10	9.05E-10
10	0.6	1.55E-09	1.47E-09	1.64E-09
20	0.8	3.34E-09	3.20E-09	3.48E-09
25	0.9	4.39E-09	4.22E-09	4.60E-09
30	0.9	5.58E-09	5.36E-09	5.83E-09
40	1	8.84E-09	8.43E-09	9.26E-09
50	1	1.34E-08	1.28E-08	1.40E-08
60	1	2.03E-08	1.94E-08	2.12E-08
70	0.9	3.30E-08	3.13E-08	3.49E-08
75	0.9	4.33E-08	4.13E-08	4.54E-08
80	0.8	5.78E-08	5.53E-08	6.04E-08
90	0.6	1.27E-07	1.20E-07	1.35E-07
95	0.4	2.39E-07	2.23E-07	2.67E-07
97.5	0.3	4.45E-07	4.07E-07	4.82E-07
99	0.2	7.89E-07	6.96E-07	9.51E-07
99.5	0.1	1.32E-06	1.12E-06	1.56E-06