A HIERARCHICAL HEURISTIC ENGINEERING DESIGN METHOD FOR NUCLEAR SYSTEMS BASED ON THE PROGRESSIVE DEFINITION OF CONSTRAINTS

A Dissertation

by

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ABSTRACT

A novel hierarchical heuristic engineering design method is developed that is based on levels of abstraction. This was created in response to the dearth of engineering design methods geared for nuclear engineering and the inability of existing engineering design methodologies to address the complexity of the nuclear system. Constraint selection is regarded as a more critical decision in nuclear system design than optimization of the design problem. The nuclear system is defined as a nested hierarchy of systems that is decomposed from the most abstract to the most concrete. Objectives and constraints for each level are created that bring further specificity to the objectives and constraints of the preceding (more abstract) level. The design or solution of the more abstract level is the basis for the more concrete level of abstraction. A sequence of levels of abstraction is presented that should be sufficient for the majority of nuclear systems but this sequence is not mandated. The engineer is encouraged to adapt his levels of abstraction to his problem's design space. A comprehensive set of heuristics is provided for the method. The developed hierarchical design method is agnostic to the optimization method within the level of abstraction although heuristics are provided to select the one best suited to the problem. The method is illustrated in the design of a materials testing fast spectrum reactor. It is used to optimize both global reactor systems and smaller problems dealing with a specific aspect of the system. The method was compared against Axiomatic Design using the materials testing reactor and nuclear system example problems. The method possessed greater flexibility than Axiomatic Design. Traditional engineering design

methodologies and their variants do not provide enough emphasis on constraint development and were not examined as in-depth.

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1. INTRODUCTION

This dissertation develops a novel method, called the progressive definition of constraints, which will guide the design of nuclear reactors. No engineering design method tuned for nuclear systems exists nor is there one that could be readily adapted for nuclear systems. This dissertation is meant to provide such a method. The method emphasizes the correct choice of constraints and objectives over the correct choice of design parameters. Selecting appropriate constraints is a more critical decision in nuclear systems than choosing the optimal design. Even though it is more properly an engineering design method rather than an optimization algorithm, much of the terminology used in this dissertation comes from the field of optimization algorithms. The constraints and objectives are gradually developed through a hierarchical nesting of design features into levels of abstraction. The levels of abstraction are evolved as the design progresses, although many nuclear system will follow specific paths outlined later in the document. The method can be used at any level of the design process and its basic form is presented later in this document. Each level's constraint and objective functions are determined by reference to heuristic processes which will be defined as part of this dissertation. The method can be used in two ways; first as a consistent method used from the outset of the design process; or second, as a corrective method after some work on a project has been completed and difficulties have emerged. The method explored in this dissertation can be used to guide the engineer or manager from a general set of constraints and objective functions to a more specific set of constraints and objective functions that can be used to provide a more detailed design. Abstract constraints and objectives are decomposed into concrete constraints and objectives as the design progresses, necessitating that the design parameters become progressively more concrete. These more concrete constraints and objective functions are based on the optimal design parameters of the previous level of abstraction and are additive to the preexisting constraints. This dissertation will contain heuristics to guide the choice of objective function and constraints. Essential to the heuristics is understanding the design process of a series of smaller problems at different levels of conceptualization. Decisions about the concepts and how to solve them play a large role in the heuristics.

The method consists of two parts, a simple two-step process and a set of heuristics which guide the process. The first step is to define the function of the system; the second step is to define the components necessary to fulfill the stated function. This process is repeated until the entire system is defined. The rationale behind this process is provided in Chapter 2, along with optional methodologies for defining the systems. The method can only be understood through the decomposition of the system into levels of abstraction. Engineering design is not performed with respect to the physical system, but with respect to a concept about how the system should perform. Example heuristics include: each level of abstraction should be as narrow as possible; constraints must address the system inputs, outputs, environment, and operation; constraints are often related to cost, safety, or materials availability; and safety constraints are often related to temperature. The translation of a physical system to a conceptual one introduces a great deal of systemic error. Naturally this error is reduced when the engineer fully understands the system, but two techniques are available. The first is to revert back to a previous level of abstraction when a difficulty is encountered. A more detailed analysis could expose certain flaws not immediately noted in the preceding levels. Analysis can be reused when the design is reverted. The second is to fork the design whenever a future difficulties are foreseen. Difficulties need not be technical, but could be derived from changing political and economic concerns. Literature reviews are encouraged for every aspect of design, but especially in foreseeing future technical issues. An example literature review is provided in the appendix.

This dissertation is composed of six sections. The first section, Introduction, introduces previous engineering design methods, PIRT, evolutionary algorithms, and nuclear system design space. The most promising design method, Axiomatic Design, is discussed in greater detail in the appendix. Robust Design is more closely examined in the appendix where it is contrasted with High Reliability Organizations. The second section, Method, states the design method, provides some background theory, and uses simple examples to illustrate the concepts. The third section, Heuristics, lists the heuristics necessary to implement the method. The heuristics are quite numerous and vary in importance; some are merely advice while others are essential. The fourth section, Demonstration of the Method, uses the method to design a SFR for materials irradiation. The reactor is not described. The fifth section, Advantages of the Method, details the benefits of the method over the other engineering design methods outlined in the dissertation. The sixth section, Conclusion, summarizes the method and outlines main finding of the dissertation. The seventh section, Appendix, provides some of the background literature for the heuristics, analyzes an engineering design method for nuclear systems, and provides a historical example of nuclear system design.

1.1 Description of PIRT

Essentially, PIRT is a process that organizes expert opinion. The traditional use of PIRT is to define relevant thermal hydraulic and system phenomena involved in nuclear accidents but it can be expanded beyond its traditional use. PIRT begins by querying a group of experts for a problem statement and a list of PIRT objectives. It then asks them to classify accident scenarios, parameters, partitions, and system designs. Based on their answers to the previous steps, the group of experts then identifies the phenomena of interest. After identifying the phenomena, the experts then classify the knowledge base of the phenomena and classify the degree to the phenomena affect the problem. The PIRT can be expanded to include more quantitative information like scaling information based on Π-groups. Π-groups can be used to estimate the effect of thermal hydraulic phenomenon on a given situation by comparing the characteristic phenomena time to the residence time in the system [Luo, 2012]. A scheme applicable to this original purpose is outlined in [Wilson, 1998] while noting that various researchers have expanded the topic to include other situations. The scheme is powerful enough to be used as a design algorithm, but would be cumbersome. Certain steps could be discarded as they are not general enough, but the algorithm still requires too many steps. It would be more powerful if the scope of the design could be more focused on constraints. Constraints of the design are hidden within each step and are not explicitly stated. PIRT processes assume a design and discuss phenomena. Reactor design algorithms must assume phenomena and discuss designs. For this reason, the PIRT algorithm can be rewritten. PIRT, as originally conceived, relies only on expert opinion. The incorporation of Π-groups into the PIRT process can be likened to the performance of the scoping studies in a design process. The proposed method does not strictly follow this PIRT process but uses it as inspiration. The PIRT process as originally described can be used to find constraints or conduct scoping studies within a broader method, a use described later in this document. Heuristics 4.3 and 6.3 as well as section 6.6 discuss the use of PIRT.

1.2 Evolutionary Algorithms

Evolutionary algorithms are a heuristic optimization technique that is based off the biological process of evolution [Lee and El-Sharkawi, 2008]. Heuristic is an ancient Greek word but in the modern context means rule or guideline. Broadly speaking, all optimization techniques are heuristic in that they make use of some rules. Heuristic optimization methods make use of heuristics to eliminate the need for the computation of derivatives of the objective function. As such, heuristic algorithms do not place any limits of the properties of the objective function or constraints. Heuristics are used to guide the choice of the next generation while only taking into account the function evaluations of the current and previous generations. Evolutionary algorithms use a very small sample of the design space and replace underperforming solutions with better performing solutions as the algorithm progresses. The process of replacing the poorer performing solutions with better performing solutions happens in generations. The algorithm begins by randomly sampling the solution space which creates 20 potential solutions. The number of solutions analyzed per generation has been shown through scoping studies to be approximately 20 but can vary depending on the problem. These 20 solutions are evaluated and ranked. The ranked solutions are then paired off and the sample's characteristics are shared between the paired solutions to create a new set of samples. This represents the division of chromosomes and mating in biology. Usually the characteristics of the solutions are then randomly mutated by some operator with variable strength and frequency. This represents mutation within biology. The mutation may happen before the recombination operator. Mutation may be thought of as an operator which enables more of the search space to be explored while the recombination operator may be thought of as an operator which keeps certain desirable characteristics within the generations. They are many variations on this theme. The most basic concerns the number of solutions in each generation which can also be thought of as the size of the sample as the algorithm samples the design space. The best sample may also be mated with the best sample of the current generation. The sample can either directly represent the function or be encoded. Suppose that enrichment was a characteristic and that 5%, 7.5%, 10%, 15%, and 19.75% enrichments were possible. The enrichment options could be directly represented or encoded in binary. Binary is not necessary but is a common choice for encoding the characteristics. 5% could become 001, 7.5% 010, etc. The encoding process is based on the use of genes to describe biological creatures.

1.3 Previous use of EA in Nuclear Applications

Pereira and Lapa demonstrated the effectiveness of genetic algorithms to solve core design problems [Pereira and Lapa, 2003]. They solved a three region LWR system with variable clad type, fuel type, fuel dimensions, and fuel enrichment to flatten the power profile while ensuring an under moderated system and a minimum criticality. In the course of Pereira and Lapa's investigation of the issues described in their journal article, the authors proposed an island genetic algorithm, wherein the solution population is divided into separate populations. These smaller populations are ranked, paired, and mutated separately with limited sharing of the design information between the groups of populations. This enables the exploration of multiple local optima at once, helping to prevent premature convergence to a suboptimal solution. As each population had limited contact with each other and function evaluations are time consuming, communication between nodes is unimportant and a LAN network is sufficient even though communication between nodes is slow. The island approach maintains genetic diversity while still allowing a gradual convergence to an optimal. Pereira and Sacco propose a modification to the island genetic algorithm to enhance niching [Pereira and Sacco, 2008].

Sacco et al propose a particle collision algorithm and compare it to a great deluge algorithm [Sacco et al, 2006]. Both algorithms are variations of simulated annealing, where the algorithm is allowed to seek out less favorable populations than the current with a certain probability. This probability decreases with the number of generations, forcing the algorithm to converge after many iterations but allowing it to investigate the search space early in the study. Sacco et al present a modification of the particle collision algorithm in which high fitness regions are explored with the Nelder-Mead Simplex algorithm and low fitness regions are not [Sacco et al, 2008]. The algorithm is found to perform better that the original particle collision algorithm, a genetic algorithm, a particle swarm algorithm, and the great deluge algorithm. de Lima et al propose a modification to the ant colony algorithm for use in a PWR core reload [de Lima et al, 2008]. A $1/8^{th}$ core symmetry of Angra-1 is used to demonstrate the benefits of an island model, wherein multiple populations with limited communication between the populations. Jiang, S et al consider an estimation of distribution algorithm enhanced with a set of heuristics to solve a fuel loading problem [Jian et al, 2006]. The problem is question is substantially easier than the others, only requiring that k_{eff} be maximized with 5 different types of assemblies in 24 locations. There are constraints about the number of placement of the assemblies. A neural network is generated for function evaluations to speed up computational time. The estimation of distribution algorithm is an evolutionary algorithm that builds a probability distribution model from which samples are pulled. This model may be likened to a simple model of the design space as is updated with more information at each generation. The variables are assumed to be independent of each other, implying that the assemblies do not communicate. The authors recognize the non-physicality of this assumption, but incorporating dependencies increases the number of simulations required to build the model. As such, they develop a set of heuristics based on the worth of each assembly type in each location surrounded by a single assembly with medium reactivity worth to account for neutron communication. The problem being solved is quite simple and characterized by multiple local optima and a single large global optimal. Essentially, the optimal design is to place the highest worth assemblies in the center of the core, followed by the next lowest worth, etc. It would be interesting to revisit this idea with a more complex problem. Kumar and Tsvetkov propose a multivariate regression analysis and modules to accelerate fitness evaluations [Kumar and Tsvetkov, 2015]. The design space is broken up into various modules characterized by a few inputs and outputs. Higher fidelity tools are used to generate regression models for each module, with the genetic algorithm being applied to the regression models. Four modules are used: fuel pin cell, whole core, hot channel, and Brayton cycle. This scheme can be used with other types of evolutionary algorithm. It should also be noted that Kumar and Tsvetkov mention the use of genetic algorithms for plant availability and maintenance scheduling, problems noted in [Lee and El-Sharkawi, 2008] as well. Mishra et al uses an estimation of distribution algorithm to load an Indian pressurized heavy water reactor [Mishra et al, 2011]. This reactor uses natural uranium and is a variant of the CANDU using heavy water as both moderator and coolant. The fresh core exhibits power peaking in the center of the core due to the use of NU and the lack of burnable absorbers, so certain positions are to be loaded with depleted uranium to reduce power peaking and enable higher total core powers. An estimation of distribution algorithm uses the fitness's of the ranked population within the generation of an evolutionary algorithm to estimate the probability a given location is occupied by a depleted uranium assembly. The next generation uses the probability density function estimated by the previous generation to generate a new population. This proceeds until converged. Montes-Tadeo et al used a Heuristic-Knowledge Method to devise enrichment and gadolinia distributions in a BWR fuel assembly and compares the technique to five standard evolutionary algorithm [Montes-Tadeo et al, 2015]. The five standard evolutionary algorithms are Ant Colony System, Artificial Neural Networks, Genetic Algorithms, Greedy Search, and a hybrid of Path Relinking and Scatter Search. The Heuristic-Knowledge Method uses design specific heuristics to accelerate the convergence towards the optimal solution. Amongst others, the method doesn't consider all of the design space simultaneously, instead finding the optimum of different designs spaces within each generation. Uranium enrichment is optimized first and this information is used to optimize the gadolinia distribution in each generation. Individual pins also have unique heuristics. Tavron and Shwageraus uses a particle swarm algorithm to optimize the fuel cycle of a pebble bed reactor [Tavron and Shwageraus, 2015]. The algorithm improved systems performance by 4.5%, which the author's call a modest improvement. Wang et al use an improved genetic-simplex algorithm to optimize a PWR steam cycle [Wang et al, 2017]. The authors note that genetic algorithms have excellent global search behaviors but poor local optimization behaviors, while the simplex algorithm has poor global search behaviors but excellent local optimization behaviors. Their method generates some population using a standard genetic algorithm then divides this population and performs the simplex method on the local search space. This method increased electrical output by 23.8 MW or increased thermal efficiency from 33.87% to 34.69%. The simplex method requires that the constraints and objective function be formulated as a polynomial, which means it cannot be used for core design calculations. Heuristics 4 provides guidance on the correct use of evolutionary algorithms in the nuclear system design. The heuristic was developed with reference to the research presented in this section.

1.4 Overview of Advancements in Engineering Design

Many different authors have contributed to the field of engineering design and this section will provide a thorough review of pertinent design schemes. The first schemes are various interpretations of the traditional top down engineering approach that can be summarized as: problem definition, brainstorming, down-selection, evaluation, and implementation. Each author expands on some aspects of the basic method. Other top down models such as the Vee model, Spiral model, and Waterfall model are also presented. These models are expansions of the traditional top down method which incorporate some aspects of bottom up design methods. All of the books listed in this section stressed the importance of material properties in engineering design, a fact which will be incorporated into the proposed method. An engineering design method is presented in Chapter 2 of [Gregory, 1966]. The algorithm begins by recognizing the engineering need including the economic, social, and geopolitical environements of the system. The state of the art which includes both research and general scientific concepts, is combined with the recognition of need to develop design concepts. These design concepts are studied and failures result in a reevaluation of the state of the art. Implementation of the design, including production and marketing, results in this design becoming state of the art. While very useful, this algorithm lacks specifics tuned for nuclear reactors. It also assumes that the recognition of need and definition of state of the art happen without much difficulty.

While possible in other engineering fields, these stages are of paramount importance in nuclear engineering. The definition of state of the art and recognition of need are difficult in the design of nuclear systems and require special heuristics for implementation. Consider a safety criterion for nuclear reactors, the large release frequency of less than 1E-6 per year. The exact translation of this to steady state operating conditions depends on safety studies which simulate accident scenarios like the loss of flow accident and loss of offsite power accident. The characteristics of these studies are set by the NRC. These studies normally require extensive knowledge of the system, meaning that the system must be very well defined early in the design process. This is not feasible, so simple models are used in lieu of complex models at this stage. The process of simplifying the safety analyses so that they are suitable for early design work is not trivial and involves expert opinion to set reasonable constraints. Once the system is developed, the actual system behavior can be verified.

A sequential process with multiple iterations at each stage is advocated in [Simon, 1975]. The recommended sequence is: definition of the problem statment and a formulation of needs; information collection; modeling; definition of the value statement or judgement criterion; synthesis of alternatives; analysis and testing; evaluation; decision making; optimization; iteration; communication to others. The author also advocated changing the order of the steps depending on the scenario. The placement of the modeling step before the definition of the value statement and synthesis of alternatives is somewhat peculiar. The modeling step generates mathematical, graphical, iconic, or analog models of the system behavior based on information collected about the system environment. The modeling step provides a way to meausre the performance of the design concepts and also guides the engineer in the synthesis of alternatives. The modeling step could prematurely bias the engineer towards certain design concepts and should be relocated to after the development of design concepts. System models are often concept dependent and their full definition can be time consuming especially if that design concept proves infeasible. Decision matrices, trees, and networks are all recommended for solving optimization problems. Such methods were commonly recommended by other authors. A general design scheme is given in [Dieter, 1983] and [Dieter, 1991] as: recognition of need; definition of problem; gathering of information; conceptualization; evaluation; communication of the design. This six step process is recast into a seven phase detailed morphology of design: feasibility study, preliminary design, detailed design, planning for manufacture, planning for distribution, planning for use, and planning for retirement. A generic model of the problem-solving process is also given in [Dieter, 1983] and [Dieter, 1991]. It is composed of five steps, with the result of each step feeding into the next step. The first step is entitled decision maker in which the external stimulus, organizational pressures, and personal characteristics of the enginneer are taken into account. The second step is problem definition and it requires data and information of the problem at hand. Logic, mathematics, and scientific principles play a role in the the third step, analysis. The fourth step is entitled decision where conflicts are resolved and risks are evaluated. The fifth and last step is called consequences in which the effects of the decision are studied. Feedback, new data, economic effects, and behavioral effects all serve to modify the decision. This problem solving scheme is more useful than the other design schemes because of its greater focus on problem definition. Conceptual design considerations and decision theory, elaborated on in [Dieter, 1991], are useful as part of the design framework, and should be integrated into a complete algorthm. Psychological aspects of problem solving are emphasized, including the distinction between the conscious, preconscious, and unconscious mind. The preconscious mind is said to be the avenue by which invention occurs, making relations between the information our mind stores before analysis by the conscious mind. Maslow's hierarchy of needs is used as a model for the recongition of needs and definition of problem stages of engineering design. Poor coping strategies when faced with stress are also listed as well as the correct responses to stress. These pyschological aspects of problem solving were unique amongst the various books examined.

A 10-step algorithm is given in [Middendorf, 1969]. The sequence begins with determining specifications to satisfy a given need. The need is provided by the supervisor; this may not be applicable to nuclear reactor design as the need should be determined by a feasibility study. The next step is to perform a feasibility study based on the specifications and needs. Third step is a search for patents. After searching for patents alternative design concepts are developed. Criteria are developed and the most promising design concept is selected. It is somewhat unusual that design concepts are developed before the judging criteria. Step six includes the development of a mathematical model that pertains to the selected design model. This model is used to determine basic dimensions and materials of the product and specifications in step seven. The model is then used to optimize the design selected in step five. Step nine involves evaluating the optimized design with higher order models. The last step involves communicating the design to administration and manufacturing personnel. This scheme involves two layers of modeling which is often necessary in reactor design. Step seven, determining basic relationships between dimensions and materials, is a useful step and can be subsumed in a general feasibility study. Reactor design typically makes use of several levels of engineering models which may or may not be performed in parallel. Step three, a patent search, can be broadened into a general search in academic papers, national lab reports, and industry reports. Step five could be placed between steps three and four. This algorithm is focused on brainstorming design concepts and studying them. The criteria defined in step five are of great importance, and different models may require different criteria. The definition of criteria and models is also not straight forward in reactor design. Feasibility studies and PIRT will be important in the definition of criteria. These criteria may be regarded as constraints.

The engineering process is divided into four main phases in [Pahl and Beitz, 1984]: clarification of the task; conceptual design; embodiment design; and detail design. Each phase has a chapter devoted to it which provides heuristics for their correct implementation. The entire process is summed up by the concepts of clarity, simplicity, and safety. Each step involves a separate iteration and is explained in detail throughout the book. The first step, clarification of the task, considers the following: possible company shortcomings; the state of technology; standards and guidelines; and future developments. Conceptual design is meant to identify the essential problems; establish function solutions; search for solution principles; combine and define concept variants;

evaluate against technical and economic criteria. The embodiment phase, or step three, determines the layout and forms of the product or system to meet the criteria. A separate algorithm is recommended for the embodiment phase. Detail design involves the optimization of the principle, layout, and forms. The algorithm emphasizes the establishment of principles and criteria for the design along with a thorough investigation of the design space. Such a scheme is well suited for reactor design, and concepts from this scheme will be applied for the algorithm described later. The method used by the Department of Defense to perform Research and Development is recommended in [Gibson, 1968]. Step one is research into the natural physical phenomenon. Step two is exploratory development for specific military problems. Research is confined to feasibility studies of proposed solutions. Step three is advanced research into all feasible solutions. Typical large projects are the VTOL aircraft or the X-16 aircraft. Step four is engineering development of ideas which have been extensively studied but not yet put into production. Management and support of the proposed ideas concerns the step five and includes the operation of the facilities and products. The last step is operational system development and the management of the approved systems and products. Step two must be modified to include problem definition as properly defining the problem is a major concern in preliminary design [Gibson, 1968]. Steps three through six concern design iterations. He also emphasizes that engineering design is an iterative process, and gives the steps of the iterative process as recognize, compare, and evaluate. Multiple design iterations are essential, but more emphasis must be placed on the definition of constraints and the objective function. Interestingly, the author lists six tools of engineering design rather than any method or scheme: economics, energy, thermodynamics, information concepts, human factors, and optimization theory.

Two general algorithms for the design process are provided in [Ray, 1985]. The first is called the Morphology of Design and is an analogy to the life cycle of a product. Step one is to identify the problem and then perform a feasibility study. Steps three and four are preliminary design and detailed design. Steps five and six are production and usage. Step seven is obsolescence. Obsolescence considers both premature obsolescence and what to do after the product's expected lifetime. In nuclear reactor design, this may include decommissioning the plant and any additional costs. Reactors that would require reprocessing would need additional consideration. These considerations are very common in the field of nuclear engineering and are normally considered in the feasibility stage. The second scheme is called Anatomy of the Design Process. It comprises four stages: identifying the problem and evaluating the need; information retrieval and assessment; evaluating the alternatives; communication and implementation.

Software design methods have developed in parallel with mechanical engineering design methods sometimes borrowing concepts. The Waterfall method was originally developed in the 1970's as an extension of traditional top down engineering design [Lui, 2016]. It has seven steps: Requirements and specification, Preliminary conceptual design, Logical design, Detailed design and testing, Operational implementation, Evaluation and modification, and Operational deployment. The design analysis and implementation steps in traditional engineering design are expanded into four individual steps. The method is notable because it incorporates feedback between each step and the preceding step. The engineer is meant use information in the following level to reevaluate the current level before proceeding. Each level is visited at least twice. This simple model was developed into the spiral model. The spiral model was developed by B. A. Boehm in 1986 [Lui, 2016]. The spiral model repeats traditional top down engineering design until the correct design is achieved. It is based on rapid prototyping where designs can be quickly implemented and studied. Making mistakes in the prototype produces no significant negative consequence as the prototype costs little to make. The spiral model is very common for systems whose complex nature limits what engineers can know about the behavior of the design before it is implemented. The engineers discover the requirements and the system and the behavior of the system by developing and studying these prototypes. While very useful for such design situations, nuclear system design is not conducive to rapid prototypes. The system has to perform as expected without full size testing although very limited prototyping is often feasible. The sequence, repeated as needed, is: Development test and verification, Evaluation and decision making, Define objectives, and Planning for the next phase.

Lastly, the Vee model is a compromise between the Waterfall method and the Spiral method. The left side of the “V” is similar to top down engineering design whereby broad concepts are translated into specific designs and components. These lower level systems and components are integrated together to form a system. The Vee model is quite popular and forms the basis for [Buede, 2009]. While used in other fields, the author does not feel that the second half of the “V” is necessary for engineering design.

1.5 Axiomatic Design

Axiomatic Design (AD) is an engineering design process which makes use of two axioms or heuristics to ensure successful design [Lee and Suh, 2006]. The AD has many features which will be adopted into the method developed in this dissertation. AD divides the design process into four domains; the customer domain, functional domain, physical domain, and the process domain. The customer specifies desired attributes and the engineer creates functional requirements to meet them. Design parameters are created to satisfy the function requirements and process variables are created to satisfy the design parameters. The functional requirements, design parameters, and process variables are vectors. The mapping from one vector to another can be represented by matrices. The first axiom of AD is to always maintain the independence of the functional requirements. This is done in two ways. Firstly, the functional requirements must not reference each other. Secondly, the design parameters must not satisfy multiple functional requirements simultaneously. Ideally, the covariant matrix would be diagonal, where each functional requirement is satisfied by each design parameter. However, this may not be possible. In that case the matrix should be lower or upper triangular where a single functional requirement would be satisfied by a single design parameter. The next functional requirement is then satisfied by the previously satisfied design parameter and one more design parameter. As the first design parameter was satisfied by the first functional requirement, the second design parameter must be satisfied by the first and second functional requirements and the first design parameter. This proceeds until all the functional requirements are satisfied. While a full matrix would produce a unique solution, the design parameters would be much more dependent on the functional requirements. Any change in a functional requirement would change every design parameter. The system is also much more expensive to solve. In practice, the matrix involves nonlinear relationships between the functional requirements. Many of the functional requirement and design parameters cannot be mathematically expressed; the vector and matrix analogy is used for clarity. Special corollaries are used whenever number of functional requirements is not equal to the number of design parameters. The method is also adaptable for the gradual definition of the solution space. The functional requirements and design parameters can be iterated on to progress from more general definition to more specific definitions of concepts. In this manner, AD can encompass a method based on the definition of constraints. However, the proposed method will still be an improvement over AD. The method will focus on the correct definition of functional requirements and developing ways satisfy them. Axioms and heuristics can be developed that are specific to nuclear engineering.

The mapping from the design parameters to the process variables makes use of the matrix analogy. The overall form of the matrix relating the functional requirements to the design parameters multiplied by the matrix relating the design parameters to the process variables must be either diagonal or triangular. The process variables must be considered when formulating the design of a product. Designs are compared on the basis of information. Information is defined as the logarithm of the inverse of the probability of the design satisfying the functional requirements. The design with the lowest information, and therefore the largest probability of success, is chosen. In general, a coupled design characterized by a full matrix would have much greater information as the likelihood of satisfying the functional requirements is lower considering uncertainties in actual operation and the exact nature of the relationships between the functional requirements, design parameters, and process variables. Information is a way of taking into account uncertainties in the actual use, performance, or development of the design. The first axiom could be generalized to say that a given property of the design should satisfy a single constraint of the system. The second axiom could be generalized to say that a design should be chosen which minimizes the uncertainty of the design's performance while maximizing the objective function.

1.6 The Design Space of Nuclear Systems

The field of nuclear systems is approximately 70 years old, beginning with Chicago Pile I in 1942. Since then most conceivable fission reactors have been examined in various levels of detail. Designs relating to large power reactors and graphite plutonium producing reactors were extensively studied for civil and defense purposes, respectively. On the other hand, only one molten salt reactor was ever built in the US, although the Chinese are currently building a very similar system as a test reactor. The design space for nuclear reactors is therefore highly known in some branches but unknown in others. System behavior for the less known systems must be gleaned from reference to systems in other industries with similar components.

Any design method or algorithm must be developed with as great an allowance for innovation as possible. Reference must be made to previous designs to encourage the development of new ideas or to reexamine an old discarded idea. The previous designs vary in detail from broad concepts to specific layouts of plants and reactor cores. Much of the practical detail of nuclear systems relates to joints, fastenings, valves, pumps, flanges, and pipes which are often taken from related industry or innovated on the fly. Given the variable knowledge of nuclear systems and the desire to allow innovation, the method must not be unnecessarily restrictive as to the design choices. As a general principle, the method should favor creativity while adding heuristics only when necessary so as to accommodate innovation.

Nuclear systems are constrained by materials as in other fields; however nuclear engineering imposes additional constraints not present in other fields. Neutrons are damaging to all materials, fast neutrons especially. The microstructure of materials is degraded by neutrons striking the lattice with enough energy to displace atoms, forming defects. Nuclear fuel must also fission, which radically alters the mechanical properties of the materials. As such, nuclear fuel is normally contained in a cladding, which provides structural strength. Reactor vessels also face neutron damage, which is complicated by their very long lifetimes and very stringent safety requirements. The standard material choices in civil and mechanical engineering such as steel, concrete, air, plastic, and water are also used in nuclear engineering. Nuclear fuels are diverse both in composition and property and structural components are often exposed to stresses unique to more standard applications. Specialized absorber materials and moderating materials add their unique properties. Consider the design of a bridge. The most likely choices are either steel or concrete, the decision of which will based on the expected loads and stresses. This process is also well codified. Consider a chemical plant. Based on the nature of the chemicals to be used the vessel, pipes, and fastener materials are chosen for chemical compatibility. Consider a reactor; should uranium dioxide or uranium zirconium metal be used? What about uranium carbide or uranium nitride? The choice of plutonium versus uranium is must include an analysis of fuel reprocessing which adds complications. In the design of an entire country's fuel cycle, plutonium versus uranium will have to be examined in greater detail. The choice of material is also not thoroughly codified, although certain requirements can be stated with certainty. For these reasons, the method will focus on material choice.

It is desired to have an actual method to ensure the optimally of a design. Many statements of the design process are general so as to be applicable to a variety of systems. As the method is restricted to nuclear systems, it must be specific enough to give solid advice while preserving innovation. The translation of utility requirements to lower level objectives and constraints is obvious from the outset. Therefore, the method should allow for the gradual definition of the search space as it evolves. A generational form of the method would enable the design to progress from less specific to more specific.

2. HIERARCHICAL HEURISTIC ENGINEERING DESIGN METHOD

The nuclear system comprises thousands of components in hundreds of systems. Fully specifying such a system would involve defining thousands of design variables and the functional relationships between them in the constraints and objectives. The ideal engineering design method would do exactly that, considering every variable, relationship, objective, and constraint simultaneously assuring that the optimal system is derived. Such a method is impractical. For this reason, the method must partition the system into smaller solvable problems. Boundaries of the smaller solvable problems are between different systems which have minimal effect on each other and between systems at different levels of conceptualization. These smaller, nested problems are called levels of abstraction, a term which will be defined later. These smaller problems are arranged in a hierarchy, and are solved in a specific order. Each problem is analyzed in the same manner. Each problem has some objective functions, constraints, and design variables. Design variables are found which satisfy the constraints and yield the best objectives (maximize or minimize). This framework is borrowed from mathematical optimization. Such a framework places great emphasis on the proper definition of constraints and objectives. The framework is considered to be the simplest possible framework necessary for problem solving. Proper creation of the levels of abstraction is essential and is discussed at length within this section and in Chapter 3. The proper creation of constraints is also important and discussed at length within this section and in Chapter 3.

Conceptualization, or the translations of physical systems into concepts, is at the heart of the method. By conceptualizing the nuclear systems they can be analyzed according to their features and not their design variables. Conceptualization is defined in the Merriam-Webster dictionary as the process of forming a concept, which is an entirely unhelpful definition. Conceptualization in this dissertation means to conceive of the design in decreasingly abstract terms as the design progresses from the most abstract to the most specific. Conceptualization allows design variables to be grouped together by function. Decomposition of the nuclear system results in levels of abstraction, a term which will be used throughout this dissertation. Each level of abstraction describes the same system but in increasing detail. While used in the design method as a top down process, the levels of abstraction are best understood from the bottom up. Consider a single component, like a primary pump. This component increases the pressure and/or velocity of a liquid and operates in the containment. In more general terms, the pump performs some function (pressure increase) on an input (coolant) to generate some output (coolant at higher pressure) and operates within an environment (containment). This pump is a part of the primary loop system, which has its own function, input/output, and environment. Generally speaking, systems/components are defined by functionality and distinguished by their feeding into or extracting from a larger system. The pump can be grouped with the related components/systems (primary loop) to form a system based on their shared functions and input/output. This system can be grouped with related systems into still broader systems until the entire nuclear system is conceived of as a whole. The whole nuclear system is now defined as a collection of nested subsystems described in varying levels of abstraction. All of the components of the nuclear system function as a single part, even if certain elements like the radiation alarm detectors and secondary loop valves are rather uncoupled. Similar components exist in the entire reactor but should not be grouped together if intermediate systems/components separate them. For example, pressure gauges in a secondary feed-water line and the primary loop are not to be grouped together, as these two pressure gauges are separated by the steam generator. Higher and lower levels of systems are levels of abstraction within the entire nuclear system. Each system/component has a function(s); the function can be described as more or less broad in parallel to the systems itself. Each system/component has an input and output; it performs some function on the input to generate some output which is processed as an input by some other system/component. Each system/component operates in an environment which implicitly affects the system/component. The Fukushima accident can be understood as a change in environment certain systems were not designed to withstand.

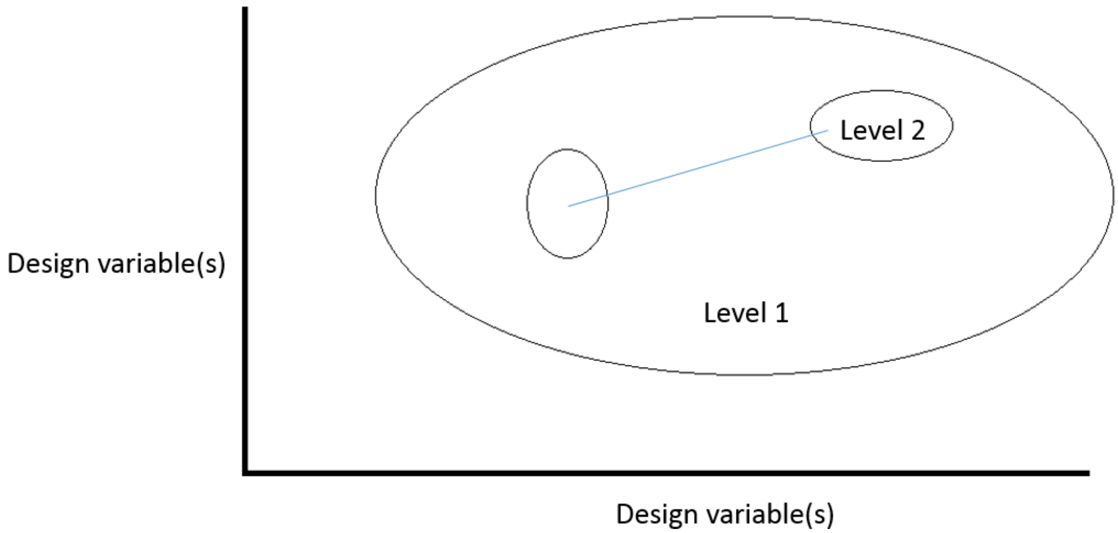

Figure 2.1 Visual depiction of two levels of abstraction. The 2D surface represents the entire design space of nuclear systems. Each level operates in the solution space of the preceding level.

A visual representation of two levels of abstraction is shown in Figure 2.1. The x-y plane represents the entire design space of nuclear systems which consists of thousands of design variables. Levels are bound by their constraints and the functionality of the levels is described by the objective function. The following level is defined in the solution space of the preceding level. Levels can be discontinuous as shown in Figure 2.1. Higher levels do not deal with all of the design variables themselves, but with groups of design variables. Therefore, the 2D surface shown in Figure 2.1 could represent two independent groups of design variables. Level 1 operates within this paradigm, which is translated to different sets of design variables in Level 2. For this reason, Level 2 would look somewhat different as the design space is redrawn including more information. The x axis in Figure 2.1 could be various primary loop layouts and the y axis could be various fuel vectors. Level 2 is then two different primary loop layouts with very similar fuel vectors. The loop configurations in Level 1 could consider the number of components and their inputs/outputs. Level 2 could consider the dimensions, mass flow rates, temperatures, etc. of the two select layouts. The fuel vector may or may not be further defined; it is often best to specify exact enrichments when performing the core design. The independent variables are well and truly independent; fuel vector and primary loop layout due not affect each other baring certain safety related constraints. Figure 2.2 is a different representation of the levels of abstraction highlighting the relationship between the levels, abstraction, and the size of the design space. Each level is narrower and less abstract than the preceding level.

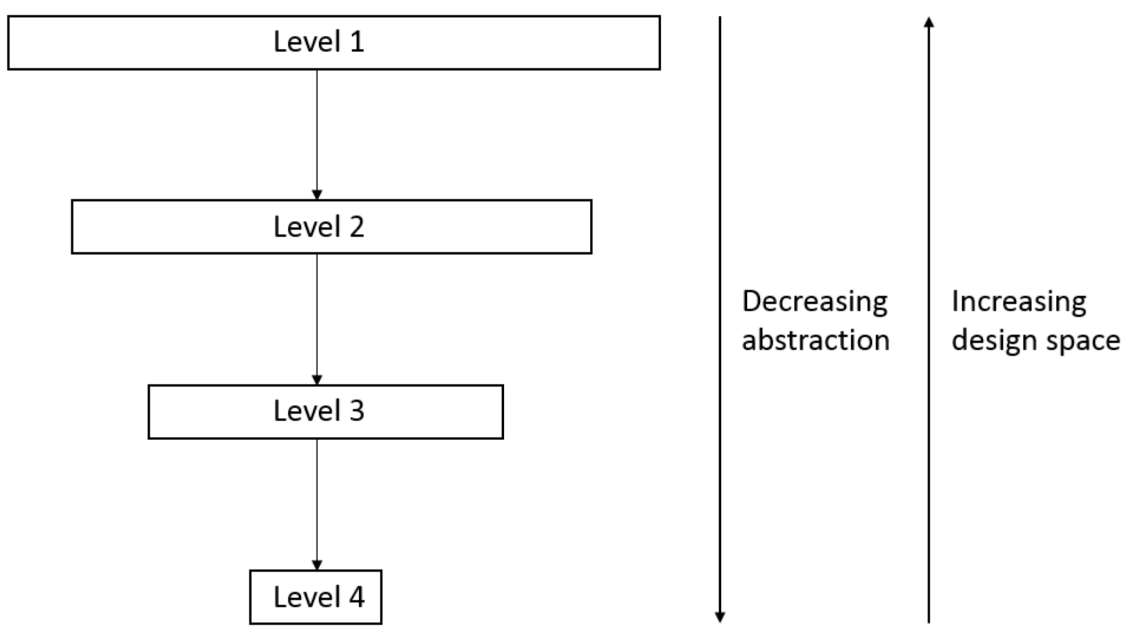

Figure 2.2 Another visual depiction of levels of abstraction highlighting the relationships of abstraction and size of the design space in the various levels.

While these levels of abstraction are best understood by considering a preexisting system from the bottom up the method must consider an undefined system. For this reason, the design method must be top down. The first level is always the overall purpose of the reactor. There are only three options; electricity production, materials testing, and process heat. Electricity is by far the most common application of nuclear energy, but materials testing is a necessary step in the development of new nuclear materials. Process heat is the least common application of nuclear power although desalination plants have been built in the past. New uses for process heat could include the production of hydrogen using temperatures above 700 °C. Electricity production is a subset of process heat as electricity is predominantly produced using a thermodynamic cycle. They are maintained as different applications for two reasons. Process heat applications will use the outlet temperature of the coolant as the prime objective of the system while outlet temperature is not a primary objective in electricity applications. Process heat and electricity applications have very different secondary coolant loops which can affect the design of the primary. The most common combination of overall purposes is electricity and process heat. Once determined, the second level of abstraction must be determined. There are no specific requirements for the second level of abstraction (and subsequent levels) like there are for the first. Despite this fact, heuristics which guide the definition of levels have been developed and are presented in Chapter 3.

Before these design algorithms are presented, some comments about the translation of the levels of abstraction into engineering design problems would be pertinent. The function of the system defined in the level of abstraction becomes the objective function of the design problem. The input/outputs become the design variables and constraints of the problem. The environment becomes a constraint. The methodology used to solve this problem is dependent on the nature of the constraints and objective function. In general, problems with numerical design variables, constraints, and objectives will be solved used a mathematical optimization method. Problems with qualitative or simple mathematical constraints and objectives can be solved through a literature review. PIRT is a useful technique when performing literature review as it encourages the specialists to think critically about the issue. The levels of abstraction concept is highly general and gives little advice into the actual design of a nuclear system. The heuristics presented in Chapter 3 are essential for the implementation of the method as they guide the choice of level. Some of the more important heuristics will be briefly summarized here. Implicit to the preceding discussion is the concept of extent, or what each level does and does not consider. The definition of extent is often concurrent with the definition of the objective and constraints. Put simply, the level's extent should be as small as possible. Once the extent of the level is known defining the constraints is straightforward. The objective for the current level contains the previous objective but seeks to refine it. The degree of refinement is determined by the extent of the level. The extent of the level can be increased if some future analysis will have an outstanding effect of the system. This can occur in pipe networks where thermal fatigue can cause unexpected joint failure where fluids of different temperatures come into contact. Such phenomena can be mitigated with an appropriate constraint or by extending the level to consider this phenomena. Explicitly considering this phenomena would greatly increase computational time as complex phenomena would have to be modeled for all proposed system configurations whereas the usage of an appropriate constraint would allow a small set of layouts to be defined and the complex calculations would be performed on them only. This consideration highlights the emphasis on the definition of constraints. It is also possible to go back to a previous level of abstraction and rework the design from an earlier point with new information. Perhaps the best solution to these issues is to anticipate the future developments and fork the design. Multiple designs could be created which anticipate the future information's effects on the design. Each design would be pursued until the required information is obtained. The extent of the levels is highly design dependent and a method which defines appropriate levels of abstraction is given in this chapter.

The preceding discussion can be condensed into two heuristics: first develop the levels of abstraction (conceptualization) and second translate the levels into design problems and consider them in descending level of abstraction. These two heuristics are given in Chapter 3 along with any others necessary to implement the design. Chapter 3 also expands on the heuristics presented in this chapter and gives additional advice on how to implement them. The design of nuclear systems can be condensed into two algorithms presented in this section. The first focuses on the correct definitions of constraints and objectives. The second algorithm specifies levels of abstraction which should be applicable to the majority of nuclear systems.

2.1 Required Input Data

There are two types of input data the method requires: user requirements and physical data. The chief concern for the user or operator is cost. Other concerns could be the neutron flux spectrum and magnitude in irradiation positions for a materials testing reactor or the outlet temperature for a desalination plant. Other user requirements could relate to the outer structure of the reactor. In the past reactor owners have perceived the tall dome containments as threatening to the layman. This may or may not be a realistic concern, but a desire to make the reactor look like a conventional industrial site has affected nuclear plant design. Physical data takes several forms but at its most basic form it consists of material properties. The most important material by mass is normally concrete followed by steel. Nickel alloys could be used instead of steel if corrosion is especially prominent. Water and air are commonly used as the final heat rejection open loop. Coolants occupy the next largest portion of the reactor's mass. The core composition is a small portion of the overall system mass while the numerous secondary metals, ceramics, and plastics in the systems occupy the smallest portion of the system mass. The method is unsuitable for the development of the components themselves, so performance data for basic components is considered an input. Depending on the desired level of abstraction, an entire core can be considered a component if the only concern is the configuration of the secondary loop. Pumps, feedwater heaters, pipes, turbines are usually assumed to be indivisible i.e. they cannot be broken down into component parts. Heat exchangers may or may not be subject to the restriction. Basic structural shapes like walls, domes, slabs, seismic dampers, shielding blocks, stairwells, etc may also be considered indivisible. Basic component data and material data is often freely available for scoping calculations. Safety related calculations must use actual proprietary data if this is not given freely.

2.2 Objective based Design Algorithm

The traditional top down engineering design method can be reoriented to an algorithm that explicitly concerns constraints and objectives. In this vein, a sequence has been developed to guide the correct definition of objectives and constraints. The steps are not levels of abstraction but have many similar characteristics. Lower level objectives and constraints function within the envelope of the higher level objectives and constraints. The first three steps concern general properties of the design before it is specified in detail, reflecting a general progression from more to less abstraction. After the completion of Step 4 in the sequence, the designer can consider further design choices as a refinement of the concept or he can revert to Step 1 for a broadening of the search space. The inability to find a solution may be caused by over constraining the search space. If this happens, the engineer can revert to the previous step and redefine the objective and constraints. An infeasible solution is in many ways just as useful as a feasible solution. The fundamental sequence is presented below. Two objective steps are needed because the primary objective is meant to be quite broad. For example, the primary objective for most power plants is to provide affordable electricity as safely as possible. This is insufficient for designing a reactor. Secondary objectives or constraints, such as the use of LEU or the requirement of a certain power production cycle enable design choices to be made. It is the author's belief that engineering design is often determined by factors not initially considered relevant. It is hoped that by requiring the engineer to specify two levels of objectives the engineer will reflect more deeply on the purpose of the system. Likewise the engineer should focus more closely on the correct definition of constraints. The heuristics presented in chapter 3 would be very useful in this algorithm.

1. Define the primary objectives
2. Define the secondary objectives and constraints
3. Define the materials and primary design choices
4. Define the secondary design choices
5. Define the tertiary design choices
6. Consider the further design choices as needed
7. Go back to Step 1 and repeat if desired but use the result of Step 6 as the primary objective

As an example of the method, suppose that an LWR is favored as a reactor type (Step 1). A consultation of the relevant operating costs and safety of BWRs versus PWRs could result in the choice of PWRs with fuel enriched to less than 5% (Step 2). The next step would assume PWRs with fuel enriched to less than 5% and compare the relevant safety and manufacturing considerations of UO_2 versus uranium metal versus UN to yield UO_2 as the favored fuel (Step 3). Step 4 might entail zoning the core by varying enrichment in several regions of the core to flatten the power profile while maintaining a certain k_{eff}. Step 1 used expert opinion and political considerations while Step 2 used expert opinion, political considerations, a PIRT, and economic analysis. Step 3 used economic analysis and expert opinion while Step 4 used a standard evolutionary algorithm or genetic algorithm to find the optimal solution. Each step uses the solution of the previous step and adds more objectives and constraints. The actual process of finding the optimal solution is straight forward and borrowed from other researchers. The Step 2 objective could be to maximize safety and minimize cost. Step 2 constraints could be the political cost associated with LEU enriched to greater than 5% and the technological maturity of the PWR versus BWR. Step 3 could seek to find the cheapest fuel type while maintaining a certain level of safety in response to a set of accident types. Step 4 would focus on minimizing the power per assembly (objective) while maintaining a certain k_{eff} and moderator coefficient of reactivity (constraints). The number of batches would be chosen for economic reasons, and the core shuffling pattern would also have to be considered in this step. Step 5, if performed, could entail the implementation of burnable absorbers to maximize burnup and cycle length. This is visualized in Figure 2.3.

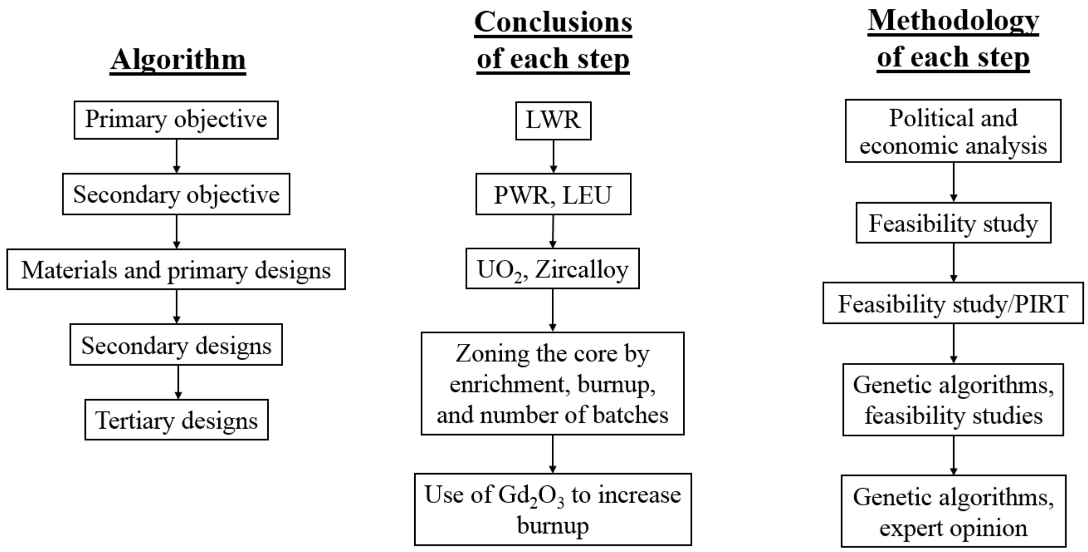

Figure 2.3 Representation of Example 1. The column on the left is the algorithm simply stated. The middle column shows the conclusion of each step, which is used as the input for the next step. The right column shows the decision-making algorithm for each step.

The previous example concludes with the implementation of Gd_2O_3 as a burnable absorber. A separate study into the use of burnable absorbers to increase burnup will illustrate two other features of the algorithm. The algorithm can be used to design local problems. It can also be used in a more iterative manner by going from Step 5 back to Step 1. As Steps 1 and 2 are both definitions of objectives, restarting the design process will enable a more applicable definition of the scope of the problem. For these reasons, Step 1 begins with the desire to increase the burnup of a preexisting core. Various methods to increase burnup are examined in a feasibility study; the economic prospects of each are judged. Each is judged to be possible; however, the branches are not given for the sake of brevity. Burnable absorbers are also the most difficult to optimize. The choice of burnable absorber material is subject to another feasibility study wherein the cost of the material is weighed against the cost saved by increased fuel utilization. The use of Gd_2O_3 in the assemblies and within the core is determined by a classical optimization method such as genetic algorithms. The layout is optimized for k_{eff} and cycle length. It is constrained by a relationship governing the cost of UO_2 versus the cost of Gd_2O_3. The cost of the fuel per core lifetime must be less than the cost without any Gd_2O_3.

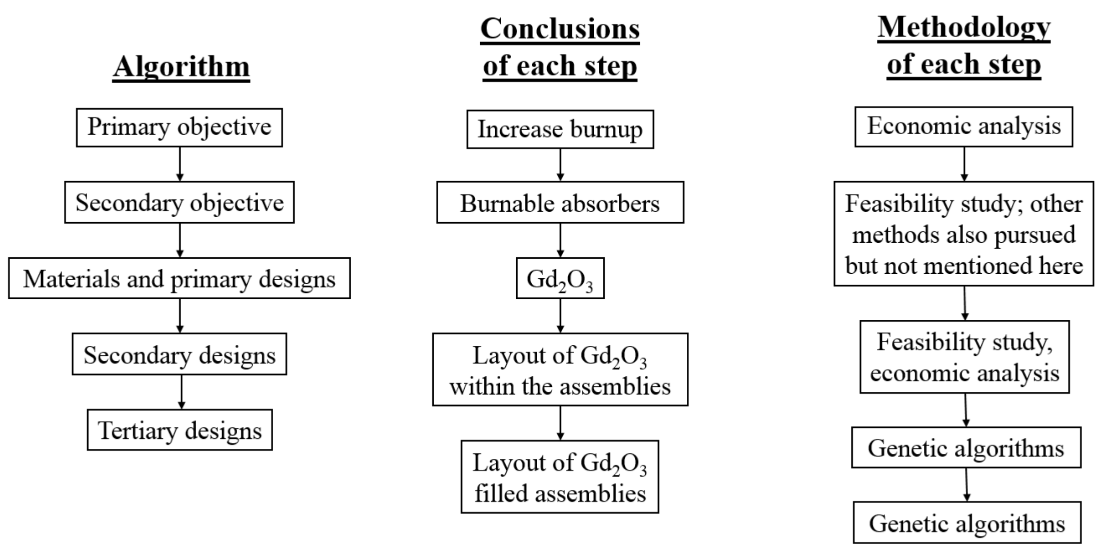

Figure 2.4 Representation of Example 2. An application of the algorithm to the use of burnable absorbers. It should be mentioned that burnable absorbers can be pursued in addition to other methods, like increasing the enrichment, number of batches, or reducing the coolant temperature and EOL. Additional methods result in branching.

2.3 Abstraction based Design Algorithm

The method described earlier down not specify the content of each level but allows to be discovered by the engineer. It so happens that the majority of nuclear systems will follow the same sequence of considerations when the levels of abstraction are defined. That sequence is shown in below. Each numbered step is the extent of the level of abstraction.

1. Purpose of the reactor
2. Heat removal scheme and fuel vector
3. Layout of the primary and secondary systems
4. Core design and primary system component design
5. Further core and heat exchanger design
6. Shielding and physical locations of the components
7. Finalization of the reactor layout

The scheme is best understood from the bottom up. The last level, finalization of the reactor layout, obviously requires the previous steps to implement. Shielding can only be designed once the neutron source and the systems between the source and the operators are well defined. In the majority of reactors shielding is not a determining factor in the design. Usually more concrete is added until the desired gamma flux is attained. Neutron flux is usually mitigated by in-core shields to protect the reactor vessel. The physical locations of the components can only be deduced once the number and size of the components is known. The physical locations could be determined in levels 4, 5, and 7 but are determined in level 6 because this aspect follows quite smoothly from the system layout concerns in earlier levels. The core and primary loop are intricate components of the nuclear system so their design requires two levels of abstraction. The number of levels of abstraction can be reduced to one or increased to three if the design is simple or complex, respectively. Level 3 determines the components necessary for the heat transport systems. Obviously, determining which components are necessary must occur before the components can be designed in great detail. This stage can require basic knowledge of the components themselves and for certain designs level’s 3 and 4 can be combined. Level 2 determines the type of the heat removal system and the type of nuclear fuel. Level 1 is the necessary first level of the method.

This algorithm was developed by considering the PWR, the most common type of commercial reactor. The BWR follows the same general scheme. The algorithm, developed with respect to the most common types of reactor, was compared with the CANDU, SFR, and VHTR. Figure 2.7 provides an outline of the steps needed to design a PWR. It is abstracted from manuals about the PWR [the Westinghouse, 1984].

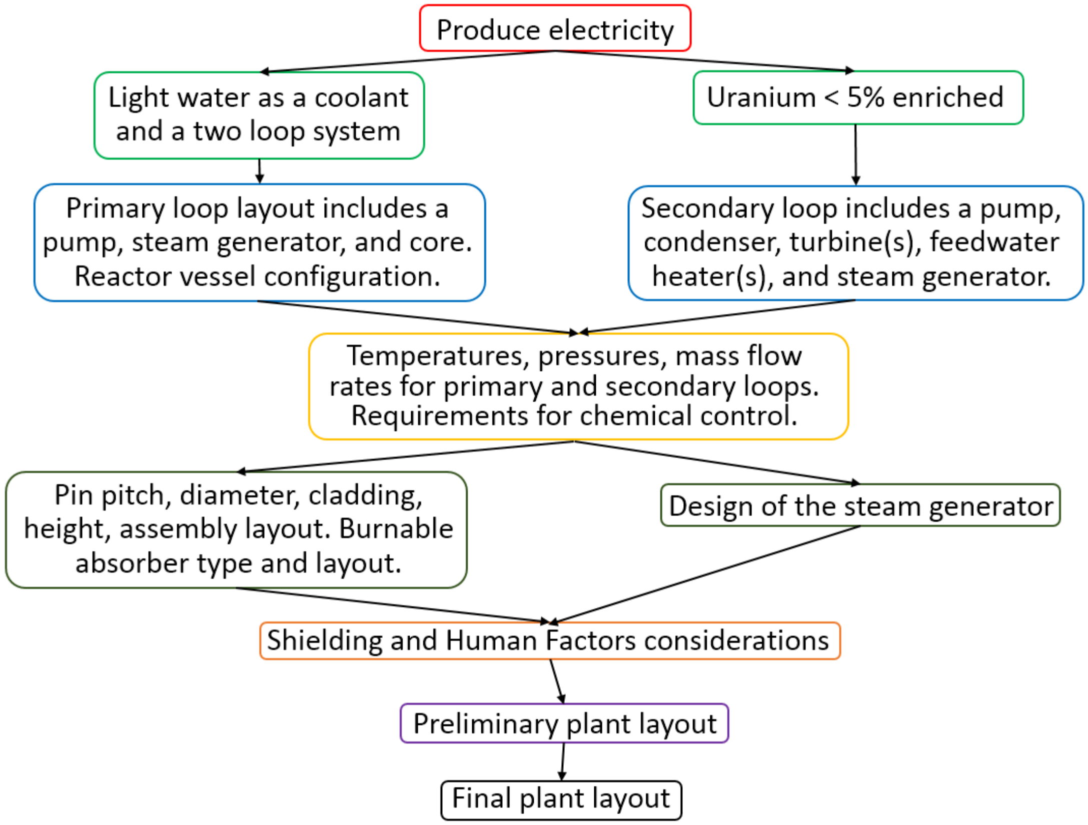

Figure 2.5 Recommended steps in the design of a PWR. This was used as a baseline for the determination of the abstraction based design algorithm. The steps are color coded.

8 sequential steps are listed although 7 levels of abstraction are recommended. The outline presented in Figure 2.5 is specially tuned for the PWR and focuses on the greater complexity of the coolant systems as opposed to the core design. The layout of the burnable absorbers coupled with shuffling schemes is considered the most complex aspect of nuclear core design while the layout of the secondary loop is considered the most complex aspect of the overall nuclear system. Where possible the tasks within the steps are divided so that they can be considered separately without including unnecessary

information. This outline demonstrates the division of tasks within levels as previously mentioned. The second step divides the choice of light water as a coolant and a two loop system with the usage of uranium at an enrichment of less than 5%. Enriched uranium is the only commercially available nuclear fuel in the USA. In countries like France, a mixture of plutonium and uranium is available. With enriched uranium heavy water as a coolant is not necessary to achieve criticality. Light water is also chosen because of its widespread availability. LWRs have negative void coefficients in contrast to heavy water reactors, but this is a minor concern at the early stages of the design. 5% is the maximum enrichment for commercial fuel in the USA. Perhaps the strongest political reason for PWRs is the legacy of the submarine program which was the basis for the US commercial program. The choice of a two loop system is another legacy of the submarine program. The BWR, which does not have a second loop, has reduced capital costs with the same level of safety as a PWR (arguably). The relative performance of BWRs and PWRs is beyond the scope of this dissertation. The third step divides the primary and secondary loops which are joined by the steam generator. The third and fourth steps can be performed as a single large step. The secondary loops are quite complicated requiring dozens of components and lines far in addition to simplified models (1 pump, 1 SG, 1 condenser, 1 turbine). Balancing the temperatures and pressures of two phase water requires engineering software and a mathematical optimization method. Performing these calculations with varying system layouts would consume more computational time than choosing a few best layouts to analyze in detail. Extensive design work specifying the thermal hydraulic conditions of the inlets and outlets of the core is necessary before any core design is performed. This is a trait of the abstraction based design algorithm. With fully specified temperatures, pressures, and the overall power, there is very little needed information by the core from the secondary side. When the PWR was developed severe accident response was not really considered and the algorithm in Figure 2.5 reflects this. Severe accident analysis is not considered unless the system is very well specified. This could be done after step 5 is completed but before step 6. Steps 3, 4, and 5 can be repeated if severe accident analysis proves the infeasibility of the proposed system. Fewer iterations would be performed if severe accident performance could be gleaned in step 4. Step 3 could include the layouts of the redundant safety systems and step 4 could include their required performance.

2.4 Mathematical Description of the Method

It is a commonly held belief that mathematics can be used to explain engineering and scientific problems better than words. I disagree with this sentiment but I acquiesce for the sake of satisfying the demands of those who decide whether or not I can graduate. The underlying rationale behind their desire for mathematics seems to be that mathematics makes everything more scientific. I acknowledge that mathematics is used in the sciences as a notation, but it is important to remember that mathematics serves science not the other way around. The concepts that mathematical notation elucidates should be understood apart from their mathematical notation. Only then can true understanding be attained. Mathematical notation is often obscure which limits its widespread adoption amongst engineers. If the notation hinders engineer's understanding of the text, then it should be discarded in favor of words and figures.

This section will discuss mathematical notation which can be used to explain the method. The most basic notation is that of the optimization problem which is adopted from the notation used in the optimization courses I took. Design variables are represented by the variable x, the vector containing all design variables. The objective is represented by f(x), where the objective f is some determination scheme that uses the design variables x. Constraints are labeled as g(x) or h(x) depending on whether or not they are equal to 0. For this work all constraints will be labeled as g and no assumptions are made as to what they equal. The next two equations neatly summarize the notation:

$$minimize\ \forall f_i(x);$$

$$subject\ to\ \forall \mathrm{g}_i(x).$$

The subscript i refers to a particular f(x) or g(x) while the symbol ∀ means all. The equations given above translate as; minimize all objective functions subject to all constraints over all of the design variables. The "minimize" and "subject to" do not have an associated mathematical symbol so the English words are used.

The levels of abstraction are described with set theory. In set theory capital letters are used i.e. A, B etc. The letters will be used in an alphanumeric fashion, where A denotes the set of all feasible designs in the first level of abstraction, B denotes the set of all feasible designs in the second level of abstraction, C denotes the set of the third level of abstraction, and so on. Each set of design variables, denoted by x, is contained within the appropriate level of abstraction as denoted in the following equation:

$$\forall x \in .$$

The levels of abstraction can be reformulated meaning that the design variables may change between levels. Therefore, the following equation may or may not be true:

$$x \in A, B, C.$$

Each level of abstraction is solved in the same manner, denoted in the preceding equations. The relationship between the current and preceding constraints and objective functions is also undetermined because of the redefinition of the levels of abstraction. If the levels are not redefined, then the constraints and objectives are additive. For this reason, the set of all constraints and objective functions increases.

2.5 Ensuring Nuclear Safety within the Design Process

In standard practice the safety of nuclear power is ensured through extensive deterministic and probabilistic assessments which examine system response to a variety of scenarios. There are several different safety criteria used by regulatory agencies. The most basic criteria requires that fuel temperature does not exceed safety limits when subjected to the design basis accident with the failure of the single most important safety related component. The design basis accident is a low probability (~1E-4/reactor-year) accident which could occur in a single reactor in a fleet of 100 reactors operating for 100 years. Fuel temperature safety limits are determined by the cladding's ability to maintain its fission product barrier. Beyond basis accidents or severe accidents are any scenarios that are more dangerous and have lower probabilities of occurring than the design basis accidents. Design basis accidents can be earthquakes, fires, tsunamis, pipe breaks, equipment failures, etc and are dependent on the nature and location of the plant. These analyses can be called deterministic as there are specific accidents that must be overcome assuming the single fault criterion. Probabilistic risk assessment considers all conceivable accidents types and calculates the consequences considering the probabilities of component performance and the probability of the scenario occurring. Consequence analysis could limit itself to the number of fuels pins which exceed their safety limits or estimate the fission product inventory released by the fuel into the coolant, breaking the coolant barrier, breaching containment, and spreading over the environment causing cancer deaths. PRA and deterministic safety analysis have been used in the nuclear industry for several decades. The most advanced PRA calculations yield an overall estimate for the number of excess cancer deaths caused by the nuclear power plant per year. These calculations are risk, not actual deaths and there utility is debated within the nuclear community. There is an overwhelming abundance of information from international (IAEA), state (NRC, DOE etc), industry (EPRI), and academic (journals) about the correct application of safety analysis tools.

While safety analysis is a very well developed field in nuclear engineering, the analyses were developed to model existing systems. This is sensible, as these safety analyses tools were developed after the beginning of commercial nuclear power. These tools will not be useful at the earliest stages of nuclear system design where the majority of safety related decisions will be made. Only after the system is designed can its safety be measured, a wholly unsuitable situation. Therefore, some additional tools must be used to ensure nuclear safety at the higher levels of abstraction. The first tool is of course literature review. Any system that is being designed likely will be different than previous designs. However, the developing system will be similar to previous systems and their existing safety analyses could be appropriated considering the differences in design. PIRT is a useful technique in this process as it would allow researchers to focus on the safety performance of previous systems while judging the differences between the systems. Scoping calculations have a long tradition within engineering where lower levels are analyzed with simple mathematical models and higher levels are analyzed with complex mathematical models. Unfortunately, this technique is limited by the physics of two phase flows. This phenomena is difficult to predict in transient simulations and requires complex models like RELAP5-3D or MELCOR. Single phase transients like those in VHTRs or SFRs can be approximated with lower order models. In small sodium cooled reactors the sodium in each component within the system will not have drastically different temperatures. This means that the entire sodium inventory could be regarded as a single control volume and could be analyzed with only Conservation of Energy. Core wide reactivity coefficients can be estimated once the fuel, cladding, core height, and layout are selected. VHTRs and GFRs can also be analyzed in a similar manner as their gaseous coolant will not undergo a phase change. The earliest these analyses can be performed is Level 4, once the basics of the core are known.

As outlined in the Appendix, three different approaches to reliability in engineering have been developed. The first approaches the issue from a sociological perspective and identifies characteristics of organizations which are highly reliable. From their research it can be shown that the engineers designing nuclear system must have nuclear safety as a fundamental concern. The second approach analyzes organizational structure with control theory, an implication whose utility is undetermined. Control theory may be a useful analytical tool when incorporated into some other design tool but it does not appear to offer any significant benefit to encourage its use. The third approach is called Robust Design. In Robust Design, the best system is not the one which outperforms the others under operating conditions but the one which performs decently well under a wide variety of conditions which may be encountered in its lifetime. This principle is highly relevant to nuclear systems as a way to implicitly guarantee safety. The accident at Fukushima occurred because the system was subjected to a change in environment that was not anticipated. This accident could be analyzed under the framework of Level I and Level III Robust Design analysis. Sensitivity studies of system performance to small changes in design variables, system environment, inputs, etc. would be a useful at each level of abstraction. Designs with less sensitivity are to be favored over designs with greater sensitivity. A common theme of PRA is the degradation of the system's components performance in adverse conditions. Multiple component failures across different safety systems result in catastrophic system failure. It is believed that the precepts of Robust Design could be used to ensure nuclear safety from the highest levels of design. Axiomatic Design is discussed in the introductory literature review. Its first precept is that the ideal system will have each functional requirement satisfied by a single design variable. The second precept states that the ideal system has the highest probability of satisfying the functional requirements in a manner similar to PRA and Robust Design. A design which does not satisfy these precepts does not mean that the design will not function; rather that the design will fail to perform as well as a design which satisfies the two axioms. The first axiom is more powerful than the second, which can be subsumed into probabilistic analyses.

From the discussion in this section, that in the introduction, and that in the appendix, some conclusions about ensuring nuclear safety can be made. Deterministic and PRA should be used to verify system performance at lower levels of abstraction. Scoping calculations should be used at higher levels of abstraction while literature reviews should be used at all levels of abstraction. PIRT is especially useful as it organizes expert opinion in a systematic way. The safest design will have each functional requirement satisfied by a single design variable and will be the least sensitive to changes in environment, input, performance, etc. These precepts can be implemented at the earliest stages of the design process.

3. HEURISTICS

In this section the heuristics are provided as a series of bullet points. Heuristics are stated simply and organized by importance. The first two bulleted heuristics are the most important and are stated in the previous section. The heuristics related to constraint selection are also stated in the previous section and are of secondary importance. The next constraints relate to the choice of optimization method to solve the design problems. All other heuristics are of quaternary importance. The literature review which defends the heuristics specific to nuclear systems is presented in the appendix.

3.1 Conceptualization of the Levels

The first and most important part of the engineering design process, conceptualization defines the extent and abstraction of the levels themselves. Even before any attempt is made to solve the level, the degree of conceptualization dictates how qualitative and quantitative the level will be. Definition of the levels is highly flexible with several good paths but a great many more poor paths.

Table 3.1 Conceptualization heuristics

Number	Heuristic
1.0	Conceptualize the design process
1.1	Each system has a function, input, output, and associated environment

1.2	Systems can be identified by groups of lower level components/systems which have a similar function and whose inputs/outputs feed into each other
1.3	Systems can also be grouped together if they do not directly interact but perform a similar general function (like coolant meters) and have the same input/output sources.
1.4	Systems are nested together until the entire nuclear systems is defined
1.5	Information on the systems contained within the broader/larger ones can be reconstructed by the behavior of the broader/larger system with the addition of select constraints only applicable at lower levels.
1.6	The levels of abstraction within the system should be chosen to be as small as possible.
1.7	The first level of abstraction is always chosen to be the most abstract possible within the given problem framework.
1.7.1	The first level of abstraction must always answer the following question; what is the overall purpose of this reactor/system? Three answers are possible: electricity generation, process heat/desalination, or materials irradiation. There is a fourth option, which is the demonstrator reactor. The demonstrator reactor is a small-scale version of some other reactor and its purpose is to mimic the operational behavior of the larger reactor to work out any difficulties. The design process is very simple; copy the larger system and simplify until the desired size is reached. This is really an additional

	objective for reactors designed with one of the three purposes. The most common purpose of the nuclear system is to generate electricity, although materials irradiation is the most interesting and will receive attention. Any combination of the three answers is possible although electricity generation paired with desalination or process heat is the most frequent combination.
1.7.2	It is also possible to use the method with some of the nuclear system already designed. In this case, the first level of abstraction is defined with respect to the given higher level in the same way the second level of abstraction is defined in relation to the first. However, the reactor as a whole can only have one of three overall purposes.

3.2 Solving the Levels

Solving the levels of abstraction is straightforward compared to their definition. The form of the objective and constraints dictates the manner in which the level is solved.

Table 3.2 Level solution heuristics

Number	Heuristic
2.0	Solve the engineering design problems in order of decreasing abstraction

2.1	The system definition is directly related to the level of abstraction; by solving the design problem in decreasing level of abstraction the engineer solves the systems from the most abstract/broad to the most general/specific.
2.2	This necessarily implies that the engineer only considers information relevant to the current level of abstraction, saving constraints specific to lower levels until it is time to examine them.
2.3	It is desirable to solve the entire level of abstraction as a single engineering design problem. However, this is often impractical due to very different and highly independent system behavior. In these cases, the engineering design problems can be broken into multiple fields so long as the interfaces between the design problems are maintained. For example, the steam generator layout and reactor core can be defined in the same engineering design problem. However, they have very different physics, objectives, and constraints and should be considered separately ensuring the same mass flow rates, pressures, and temperatures.
2.4	By defining the nuclear system in levels of abstraction and considering the system in more abstract terms a great deal of systemic error has been introduced between actual system behavior and the presumptions of its behavior. This error is of course minimized by the correct definition of constraints but the engineer should leave room for it in his deliberations. Results of detailed analyses not performed until a later stage of the design process frequently change the nature of the design problem. The engineer has

	two solutions to this predicament. Firstly, go back to the previous level of abstraction and rework the design given the new information. Secondly, account for the future knowledge from the starting level by either defining the level in such a way that it accounts for the detailed phenomena or by defining multiple solutions at the starting level which accommodate the outcomes of the more detailed analyses. Any option is correct, although the latter options require a great deal of physical insight into the design.

3.3 Objectives and Constraints

Definition of the constraints happens after the conceptualization of the level. The order in which the heuristics are presented here is indicative of the relative importance of the heuristics. Definition of objectives is much easier than constraints, and there are correspondingly fewer heuristics pertaining to this.

Table 3.3 Heuristics for the definition of objectives and constraints

Number	Heuristic
3.0	Objective and constraints are only defined after the engineering problem is fully specified. The objective is related to the purpose or function of the system and is normally either maximized or minimized. This should be stated first. Constraints are normally some limiting conditions. Constraints must address the inputs, outputs, and environment assuming the function of the

	system becomes the objective. Ideally, one constraint each is defined to account for the inputs, outputs, and environment. However, they can be combined or expanded based on the needs of the design. Constraints frequently incorporate multiple design variables due to the dependencies.
3.1	Constraints should be as simple as possible, comprising one design variable. This is often not possible given the complex nature of the systems in question.
3.2	There is no restriction on the number of design variables, only that their number be minimized in any design problem.
3.3	Safety considerations are often constraints. Safety concerns must play a role in in every level of engineering design. The translation of safety considerations into constraints and objectives has its own set of heuristics presented in section 3.6.
3.4	Engineering feasibility is an important consideration within engineering design and can be implemented in the constraints or objectives.
3.4.1	Feasibility can be judged through consideration of the knowledge base of the design. Knowledge base is a term frequently used but uncritically defined. It can be qualitatively defined as experience with the manufacture and operation of a technology. This can be quantified as the number of times the technology was implemented, simulations of behavior, and breadth of experiments demonstrating technological response to off-normal scenarios.

3.4.2	Feasibility can be judged through the ease of attaining commercially available technologies. A technology with a single manufacturer is less feasible than one with several.
3.4.3	If the technology is not commercially available, feasibility can also be quantified in terms of years or cost to qualify a technology.
3.4.4	Feasibility can also be judged with standards used by government agencies like NASA.
3.4.5	Feasibility can be judged through the number and content of governmental reports on the technology.
3.4.6	Feasibility can be judged be examining the number and content of academic papers. Concepts that are the topic of academic papers with little to no commercial sponsors are less feasible than those with some funding be it commercial or governmental.
3.5	There are no restrictions on the way constraints are defined. The manner in which the constraints and objectives are defined determines the methodology used to solve them.
3.6	Constraints can be determined through the user's consideration of limitations stemming from the inputs, outputs, and environment. Environmental limitations are usually formulated as constraints while outputs, like power peaking factor, can be formulated as constraints or within the objective functions. Constraints for inputs often come into play when searches for

	optimal inputs are performed. The possible input space can be highly nonlinear and discontinuous. If this is the case uniform sampling of the input space will yield infeasible design variables but can still be performed as long as some kind of rejection technique is implemented or if the objective function weights infeasible solutions quite poorly.
3.7	Ultimately, only the engineer can properly account for the constraints. The best way to discover the constraints is for the user to consider what limits the design at each level of abstraction. That being said, some topics are more likely to yield useful constraints.
3.7.1	Materials properties dictate the majority of safety related constraints. Common constraints include stress/strain of fuel claddings, reactor vessels, primary components, and containments. Temperature, because of its ability to reduce the strength of materials, is also considered as a constraint. Chemical reactions are also temperature dependent e.g. hydrogen production/cladding embrittlement in LWRs and eutectic formation in steel clad metallic fuels. Efficiency of thermodynamic cycles increases with temperature. This can be formulated in the constraints or in the objective function.
3.7.2	Constraints can also be based on what is available for the engineers to use. This can be materials or components and their associated performances.
3.7.3	Cost can be implemented in either the constraints or objective functions. Material and component availability can be accounted for separately or by

	directly considering their costs. The same can be said for certain system configurations.

3.4 Methodology

In principle any level can be solved with any methodology. While this will change the solution the method will perform regardless. These heuristics provide advice on how to achieve the best solution with the least effort.

Table 3.4 Methodology heuristics

Number	Heurisitc
4.0	The methodology used to solve engineering design problems is based on the manner in which the constraints and objective function are defined. Design variables with mathematical formulations should be solved using a mathematical optimization method; design variables with non-mathematical formulation should be solved with non-mathematical methods. The three most useful methods in nuclear system design are evolutionary algorithms, dynamic programming, and PIRT.
4.1	The mathematical optimization tool should be the most simple that is capable of solving the problem. If the constraints and objectives are linear, then a linear programming technique should be chosen. If the constraints and objectives are nonlinear but quadratic, then a nonlinear dynamic method can

	be used. Combinatorial problems are normally solved with combinatorial optimization methods based on graph theory. Constraints and objectives with complicated functional dependencies that mix continuous and discrete formulations must use some form of heuristic algorithm. An enormous body of literature is available about the utility of the various methods.
4.2	Evolutionary algorithms (of which genetic algorithms are a subset) have received much academic attention and their usefulness can be mapped out for many types of problems in nuclear engineering. Neural networks are the most powerful tool for solving engineering design problems but are difficult to implement and are time consuming to find a solution. Genetic algorithms should be sufficient for the design of nuclear cores while dynamic programming is sufficient for the design of the power producing coolant loop. The design of heat exchangers may require a genetic algorithm. Neural networks should only be used if evolutionary algorithms prove insufficient or if future developments make them less time consuming to perform. Genetic algorithms are recommended as the generic optimization method because they are exceptionally robust and excellent at searching the design space. Evolutionary algorithms excel in finding many solutions with a high fitness but not in finding the solution with the highest fitness. Considering the systemic uncertainty in engineering design, a collection of good designs is more likely to contain the true best than the smaller collection of high fitness designs.

4.3	There are many high level decisions which must be made in the design of a nuclear system most of which can be made by performing a thorough literature review and comparing the relevant characteristics. PIRT could be used to guide particularly difficult literature reviews. Figure 1.1 summarizes the PIRT process for its original intention, but it could be tuned for engineering design. The first step would be to define the judgement criteria in detail and devise a framework for translating subjective judgements about feasibility and utility into numerical measurements. The second step would be to generate design concepts. The third step would be to evaluate each concept for its feasibility and ability to satisfy the objective using the numerical framework. The number of engineers helps estimate the uncertainty of any one numerical evaluation. The results are presented in step 4. This four step process can be expanded based on the desires of the engineers. The original PIRT process separates the analysis of thermal hydraulic phenomenon by time and system. A similar distinction could be made in the field of nuclear design between near-term and long-term effects and physical phenomena.

3.5 General Nuclear Design Process

These heuristics are not mandatory, but derived from experience dealing with nuclear design. They should be considered advice applicable in the majority of situations.

Table 3.5 General nuclear system related heuristics

Number	Heuristic
5.0	Constraints and functions can only be known through experience with the system. Similarly, where to divide the nuclear system into levels of abstraction can only be known through experience. Therefore, the remainder of the heuristics are meant for the nuclear systems only. These heuristics should be understood as nuclear engineering design best practices; some are merely advice whereas others are necessary.
5.1	The choice of heat transfer mechanism/coolant is normally the second level of abstraction.
5.2	LEU is to be used in all reactors aside from military reactors.
5.2.1	Exceptions to this can be made for materials irradiation facilities.
5.2.2	The international soft limit for enrichment in power reactors is 5%. However, high enrichments can be used for non-power reactors.
5.3	Fuel plates are favored in high density cores but generally cost more than pins. Fuel pins are easier to manufacture but have higher fuel centerline temperatures. This can pose a safety risk in certain accident scenarios and results in a higher Doppler penalty.
5.4	Steel, especially stainless steel, is the favored material in reactor design. Stainless steel is strong, chemically compatible with most coolants and fuel

	types, is reasonably cheap, has significant resonance integrals and thermal cross sections, has excellent irradiation behavior (select compositions), is easily machinable, and does not undergo exothermic reactions with water like zirconium. Nickel alloys or ceramics are to be used with fluoride or chloride salts, while zirconium alloys have to be used with most LWRs and CANDUs.
5.5	It is perfectly acceptable to bury the reactor in concrete. It makes an excellent shielding.
5.6	Core damage frequencies and severe accident initiating events have been historically underestimated. This must be taken into account when performing probabilistic risk assessments. Beyond Design Basis Accidents have occurred much more frequently than predicted, and should be analyzed.
5.7	Several collections of design criteria have been created by various regulatory agencies. Design criteria for light and heavy water reactors are more common due to their greater numbers. Design criteria for Generation IV designs are being developed by the IAEA and the NRC. Using such criteria accelerates the regulatory time but is not strictly necessary in the USA.
5.8	Extensive seismic isolation has not been approved by the NRC. Current designs mitigate earthquake damage by overdesigning the system to account for the full ground acceleration. Capital costs could be reduced by building the entire reactor on dampening pads not just specific portions of the building.
5.9	The system should be built with non-nuclear safety grade components to the greatest extent possible.

3.6 Nuclear Safety

The rationale behind the heuristics in this section can be found in section 6.2, located in the Appendix.

Table 3.6 Nuclear safety heuristics

Number	Heuristic
6.0	Nuclear safety is perhaps the greatest issue affecting nuclear power after cost. Thus, ensuring nuclear safety during the design process is essential.
6.1	Traditional safety assessments like PRA or deterministic analyses can be used at the lowest levels of abstraction once the system is well defined. These assessments cannot be used until after level 5 in the Abstraction based Design Algorithm. A number of documents are available for the correct implementation of these assessments so it is not summarized here.
6.2	Scoping calculations simplify system behavior and physics allowing for their use of higher levels of abstraction. Scoping calculations allow for prospective designs to be quickly analyzed.
6.2.1	Scoping calculations use simplified versions of the governing equations. For example, the homogeneous equilibrium model might be used instead of a two phase model. Similarly, simple hand calculations can be used to design

	reactors if appropriate cross sections are known. The same can be said of deterministic transport models.
6.2.2	Scoping calculations are normally steady state; transient analyses are either not performed or assured through significant safety factors on the steady state cases. Safety factors can be derived from literature reviews.
6.2.3	Scoping calculations typically make use of simplified system models. For example, in small SFRs the sodium temperature does not change around the loop, meaning that the entire sodium inventory can be considered a single control volume. In many scenarios only the average and hottest fuel channel need be analyzed.
6.2.4	Scoping calculations are easier to implement for thermal hydraulic models than neutronic models. Computational Fluid Dynamic models are the most advanced and are not usable for design calculations. 1D models with a few control volumes are suitable for design calculations. Complex turbulent behavior is modeled with experimental data in these 1D models. Performing these calculations is commonly taught in undergraduate mechanical or nuclear engineering courses and no other advice is needed.
6.2.5	Neutronics scoping calculations are often performed differently than thermal hydraulic scoping calculations. Neutronic scoping calculations typically use fully developed codes with simplified inputs (like MCNP). This is because of the difficulty in cross section generation when a new system is under development. Core materials are often determined before the core layout is

	chosen. Potential core layouts are generated and analyzed using the proposed materials at approximate temperatures. This is done to map out the design space identifying the best and worst design choices. Burnup calculations are often performed without the core in a critical state. If the core is not critical then the flux profile (in energy and space) will not be realistic; without a realistic flux profile, the reaction rates throughout the core will not be estimated correctly. However, for a variety of designs simulating the core non-critical will not result in significant errors. Full core burnup simulations are often performed with control rods in a set position. k_{eff} is often correctly predicted even if the flux profiles can be incorrect.
6.3	Literature review can be used in place of scoping calculations. Between reactors which were built at the very beginning of the nuclear era to the innumerable paper reactors available in academic journals, a large bank of reactor concepts exists. Performance data from these reactors can be extrapolated to the proposed system. PIRT can be used to identify the most important characteristics of the proposed system and judge how well a preexisting design matches the proposed system. It may be necessary to accumulate estimated system performance from several different preexisting designs. As mentioned earlier a literature review can be used to derive appropriate safety factors for steady state calculations. NUREG documents are especially useful but their scope is limited to the more common reactor types. VHTR and SFR concepts was licensed under the AEC and these

	documents can be used to estimate the safety performance of the proposed system. GFR limits can be appropriated from the SFR and VHTR while MSR concepts could use the MSRE documentation.
6.4	The precepts of Axiomatic Design and Robust Design could be used to ensure safety. The safest design will have each functional requirement satisfied by a single design variable and will be the least sensitive to changes in environment, input, performance, etc.
6.4.1	Axiomatic Design is discussed in the introductory literature review. Advice for the creation of functional requirements can be found in conjunction with the advice for definitions of the levels of abstraction and the translation of the levels to engineering design problems.
6.4.2	Sensitivity studies about the system performance in non-ideal environments, inputs, or component performance should be performed at each level of abstraction. Sensitivity should play a role in solving the related engineering design problem. Sensitivity at the higher levels of abstraction could be difficult to quantify but must be considered. This sensitivity will become more defined as the system becomes more concrete. Changes in environment are the most basic parameter to analyze. Changes to input analyze how the non-ideal output from a component could affect another component. Changes in the functionality of the system itself are perhaps the hardest parameters to analyze. These changes in functionality could be due to system degradation from a changing environment, degradation due to age, degradation due to a

	misunderstanding of the phenomena, degradation due to unforeseen complex interactions between the various system components.

4. FAST SPECTRUM MATERIALS TESTING REACTOR

The design of a materials testing reactor poses unique design challenges for the method and designer, allowing for a thorough exploration of the method. Materials testing reactors have to achieve set fluxes and/or power densities in certain locations and materials within the core. The irradiation positions must be reconfigurable. While the core design is highly constrained, the secondary systems are less constrained. Secondary systems often do not generate electricity unless a demonstrator plant is called for. Additional constraints for the core design and fewer constraints for the plant design simplify the design process of the higher levels of abstraction while making the lower levels more complex in contrast to the design of power reactors. Examples in Chapter 2 and the heuristics themselves focus on power reactors; this example is provided to demonstrate the versatility of the method. The initial objectives for the reactor came from a government document; additional constraints and objectives were provided by examining the behavior of other test reactors mentioned in the document. This research resulted in a number of constraints but no major objective. The objective came from focusing on a secondary concern wherein a demonstrator reactor is desired for a given Generation IV technology. The overall objective could then be stated as the smallest possible demonstrator reactor capable of fulfilling all the constraints listed in the document. Scoping calculations were used throughout this project. The SFR was chosen as the demonstrator reactor and a representative fuel geometry was adopted from a historical system with excellent performance. The overall height and diameter of the core was then chosen to achieve a suitable degree of excess reactivity. Core power was based on the power density in the historical system and the required flux levels in the irradiation positions. A thorough global and local characterization of the system was performed. Simple two parameter searches were then performed to allow a limited degree of optimization for the local behavior. The local behavior required additional objectives and constraints. Genetic algorithms were not used in this investigation, just a modified form of Robust Design focused on defining appropriate design factors to accommodate diverse neutronics requirements.

An outline of the reactor design process will be presented first. Levels of abstraction are numbered in the list. The objective, constraints, and methodology of each level of abstraction are provided while the manner in which the constraints were defined and the results of each engineering design problem are not defined. All constraints and objectives are additive from level to level. This study was performed to propose a new MTR; the neutronics of the core are paramount so non-nuclear aspects of the design were ignored. The design is meant to be a demonstrator reactor, so the non-nuclear aspects of the design would be scaled down from a hypothetical large scale SFR. The document which provided the purpose and needs of the reactor did not offer specifics.

1. Whether or not to build a new reactor or modify an existing reactor
    - Objective: satisfy all constraints while clarifying the original requirements. A durable, feasible design that is within or just beyond the realm of well-established technology is favored.
    - Constraints:

i. Coolant loops with multiple possible coolants; boiling water, helium, molten salt, and sodium were specifically mentioned

ii. Demonstrate an advanced reactor concept

iii. Large volume irradiation positions

iv. High flux thermal and fast spectrum positions for accelerated testing of long life materials

v. Support LWR and advanced reactor programs

- Methodology: extensive literature search, focusing on defining the size and fluxes of the irradiation positions.

2. The coolant and fuel vector of the reactor; set relevant thermal limits

- Objective: Satisfy all materials irradiation constraints while selecting the most feasible and promising reactor type
- Constraints:

i. Cannot use LEU

ii. Reasonable safety performance

- Methodology: extensive literature search, focusing on the available body of knowledge for the various reactor types. A PIRT study would have been applicable but was not performed because there were not enough parameters to warrant it.

3. Assembly and fuel dimensions

- Objective: highest flux
- Constraints:

i. Sufficiently negative power coefficients so that LOFA accidents will not result in fuel damage

ii. Durable, feasible assembly design

- Methodology: choose from a preexisting assembly design from proven performance. A PIRT study would have been useful in judging the merits of the possible design concepts.

4. The reflector material and core layout

- Objective: maximize fast fluxes and maximize the core lifetime; minimize the pressure drop (weak objective)

- Constraints:

i. Average coolant velocity less than 10 m/s [Fast, 2006]

ii. Core pressure drop less than 0.5 MPa [Fast, 2006]

iii. H/D=1

iv. Average linear power and maximum linear power must not exceed EBR-II values [Fast, 2006]

v. Fast flux should be at least 5E15 n/cm^2-s

- Methodology: scoping studies

5. Assembly specific design to attain specified fluxes and power densities

- Objective: maximize flux (strong) and minimize reactivity penalty (weak) of the irradiation positions

- Constraints:

i. Maintain prototypical power densities

ii. All assemblies must be independent

iii. Do not modify the barrier assemblies

iv. Water should be used sparingly

- Methodology: scoping studies

The initial requirements for this reactor were not specific enough for all levels of abstraction, and the definition of constraints played a large role in this design. Lower level constraints were chosen to elaborate on higher level constraints. It is possible for the design to be completely reconceptualized at different levels of abstraction; this was not the case in this design. Lower level design variables, objectives, and constraints are elaborations of those at higher levels. Figure 4.1 shows the relationships between the constraints at different levels of abstraction. There were two constraints in the lower levels of abstraction which do not have an origin in the preceding levels. Safety performance in Level 2 is a necessary consideration included in all nuclear design. H/D=1 was included in Level 4 to simplify the design scheme. Flattening the core (H/D<1) would reduce the pressure drop and increase neutron outleakage. A more thorough examination would not use this constraint. The design variables became more elaborate as the design progressed. A qualitative judgment of the safety performance of the reactor was translated into maximum velocities, maximum pressure drops, maximum pin powers, and sufficiently negative reactivity coefficients. This translation was possible because of the extensive knowledge base of SFRs that provided a baseline for a sufficiently safe SFR. Safety was assured by ensuring that this reactor did not exceed prototypic SFR steady state parameters.

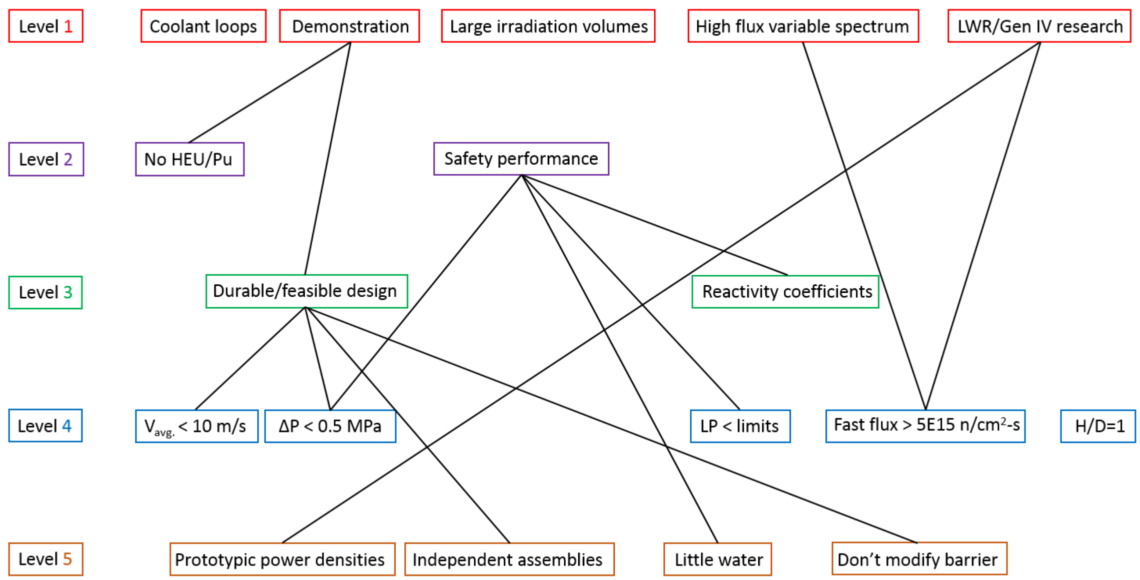

Figure 4.1 Relationships between the constraints showing how lower level constraints were derived from higher level constraints. All preceding constraints apply at each level of abstraction.

4.1 First Level of Abstraction

Materials testing reactors (MTR) exist to supply a high flux of neutrons for materials irradiation and general purpose experiments. The Advanced Test Reactor, High Flux Isotope Reactor, and university TRIGA reactors are all examples of MTRs where the primary purpose of the reactor is to supply neutrons for in core irradiation. HFIR is also notable for its beam ports, often used as a supply of cold neutrons. The overall design of a new MTR is presented in [Scherr, 2016] and [Scherr, 2018] but the design process is outlined here. The starting point for the reactor design was easy to decide, as it begins with a government document outlining the projected research needs. The Nuclear Reactor

Technology Subcommittee, a subcommittee of the Nuclear Energy Advisory Council, released a report on November 18, 2014 that requested designs for a MTR in support of the Generation IV and LWRS programs [Report, 2014]. The document requested a dual-purpose reactor, one that would enable testing of multiple reactor types and serve as a demonstration reactor for a Generation IV reactor type. Demonstration reactors, or demonstrator reactors, are proof of concept designs that demonstrate the performance of a large-scale design. The demonstrator reactor must replicate the systems and behaviors of the larger system. Equipment, fueling handling, reactor physics, maintenance schedules, fluid behavior, etc. are all meant to be similar to a larger system. The reactor is also meant to have diverse neutron spectrums which could be attained by reconfiguring the core. Coolant loops are commonly used in materials testing reactors to simulate appropriate thermal hydraulic behavior around irradiated material. Such systems were regarded as essential in the proposed reactor, and must be capable of accommodating a wide variety of coolants and fuel types. The coolant loops must be capable of both steady state and transient behavior. It is mentioned within [Report, 2014] that the irradiation position in the proposed reactor must be large. Large is not defined, but can be gleaned from studying potential objects to be irradiated and the current volumes of the irradiation positions in current testing reactors.

Since the document which defined the reactor requirements referenced the Generation IV and LWRS programs, studying those programs would yield additional constraints and objectives. The Light Water Reactor Sustainability program is a DOE funded program dedicated to increasing the useful lifetime of Generation II reactors

[Light, 2014]. Most of those reactors have attained 20-year license extensions and there is a focus on the next 20-year life extensions. This would bring the final reactor lifetime to 80 years. There are several aspects of this program, but the relevant parts are those which involve materials irradiation. The vessel and concrete are exposed to neutron fluences over the entire lifespan of the reactor, a process which could replicated in a test reactor. Large volumes of material would be very useful to irradiate so as to study irradiation effects. Replicating 80 years of irradiation in a reasonable amount of time means large volumes at a high flux. The claddings of LWRs are based on zirconium, which undergoes exothermic chemical reactions with light water at high temperature. This produces hydrogen gas, which can explode, and leaves the zirconium in a weaker ceramic form. The Fukushima accident is an excellent example of the pitfalls of zirconium alloy claddings. Steel and SiC are proposed as potential LWR claddings, so the reactor must be capable of replicating the operating conditions of those claddings [Light, 2014] [Bragg-Sitton, 2013]. The Generation IV forum was established in 2002 by the OECD Nuclear Energy Agency to promote research and development of Generation IV reactor designs in member countries. Six concepts were proposed: SFR, GFR, LFR, VHTR, MSR, and SCWR. The SFR and VHTR have received the most research and were projected to complete their viability studies earlier than the other four [Technology, 2014]. The modified update changes the projected finish times, but in the judgment of the author the SFR and VHTR are still the most likely to become actual systems [Technology, 2014]. The most common reactor type in the world, LWR, and the most likely Generation IV reactors, SFR and VHTR, should be the reactor types whose behavior the proposed reactor must be able to replicate. The

SCWR shares many characteristics with the LWR, the GFR and LFR share characteristics with the SFR, and the MSR could be replicated in most scenarios as its design is flexible. The three reactor types chosen well represent all possible reactor types and reduce the volume of analysis. The Fast Reactor Database 2006 Update is a collection of design information from historical fast reactors and proved very useful in this study [Fast, 2006]. The highest peak fast flux listed for any reactor type is 5E15 n/cm^{2}s for neutrons with energies greater than 0.1 MeV. A MTR with fluxes lower than the actual reactor would require more time to simulate the lifetime of nuclear materials. This could necessitate the building of an actual system without complete verification of the behavior of core materials at the end of life, an unfavorable scenario. For this reason, the peak fast flux in the reactor must be greater than 5E15 n/cm^{2}s. This value for fast flux was corroborated in personal correspondence with an INL researcher concerning a fast flux that would be required for materials irradiation research [Hayes, 2015].

Table 4.1 HFIR [Xoubi, 2005]

Nominal power	85 MWth
Maximum flux	2.6E10^{15} n/cm^{2}s thermal flux
Center flux trap size	12.7 cm diameter
Active fuel height	50.8 cm
Other positions in Be reflector	N/A and far less
Total in core irradiation volume	6500 cm^3

Table 4.2 ATR, fast flux E > 1 MeV [FY, 2009]

Maximum power	250 MWth
Normal operating power	110 MWth
Active core height	122 cm
Power tilt ratio	3:1
Largest corner traps (5.25 in diameter)	$4.4E10^{14}$/$2.2E10^{14}$ n/cm^{2}s thermal/fast
Other corner traps (3.0 in diameter)	$4.4E10^{14}$/$9.7E10^{13}$ n/cm^{2}s thermal/fast
Largest A traps (1.59 in diameter)	$1.9E10^{14}$/$1.7E10^{14}$ n/cm^{2}s thermal/fast
Other A traps (0.66 in and 0.5 in diameter)	$2.0E10^{14}$/$2.3E10^{14}$ n/cm^{2}s thermal/fast
Other positions are smaller	Other positions have lower fluxes
Total in core irradiation volume	98000 cm^3

Table 4.3 JHR, fast flux E > 0.9 MeV [Boyard 2005]

Nominal power	100 MWth
Active core height	60 cm
3 in core irradiation assemblies (94.5 mm)	$4.2E10^{14}$ /$2.9E10^{14}$ n/cm^{2}s fast/thermal
7 in the center of an assembly (32 mm)	$5.5E10^{14}$/$2.2E10^{14}$ n/cm^{2}s fast/thermal
12 positions between assemblies	N/A
6 PWR testing loops	$4.3E10^{14}$ n/cm^{2}s thermal 1% enriched pin
Total in core irradiation volume	16000 cm^3

Table 4.4 MBIR, fast flux E > 0.1 MeV [Tretiyakov, 2014]

Nominal power	150 MWth
Active fuel height	55 cm
Fast neutron fraction	~0.7
Maximum/average core flux	5.5E15/3.5E15 n/cm^{2}s in the core
14 materials testing assemblies, 7.22 cm flat to flat. Maximum/average flux	4.9E15/3.6E15 n/cm^{2}s
3 external loops, assemblies 5 cm by 144 cm	5.0E15, 2.0E15, 1.3E15 n/cm^{2}s
3 instrumented in core loops for alternate coolant types, assemblies 4.5 cm wide.	3.2-4.0E15 n/cm^{2}s
6 horizontal beam ports	N/A
Total in core irradiation volume	45000 cm^3

Table 4.5 MYRRHA, fast flux E > 0.75 MeV [Van Tichelen] [Abderrahim, 2012]

Subcritical/critical power	65 MWth/100 MWth
Subcritical fast/total flux in central channel	1.01E15/3.75E15 n/cm^{2}s
Critical fast/total flux in central channel	4.05E14/2.61E15 n/cm^{2}s
Subcritical fast/total flux in off central channel	4.2E14/2.6E15 n/cm^{2}s
Critical fast/total flux in off central channel	2.56E14/1.75E15 n/cm^{2}s
Total in core irradiation volume	39000 cm^3

The ATR has coolant loops which can accommodate different coolant types and can operate in transient mode [FY, 2009] [Gerstner, 2009]. The MBIR also possesses coolant loops but the coolant within them and whether or not alternative coolants are allowed are not stated. The JHR has PWR loops and it is not stated if other coolants can be used in them. The Fast Flux Test Facility is a 400 MWth SFR located in Hanford, Washington. Shutdown in 1996, it has been defueled and the sodium has been drained while it waits in limbo. This reactor could be restarted and it might be possible for it to be reconfigured as the US's new MTR. This possibility was not examined because of FFTF's maximum fast flux was less than 5E15 n/cm^2-s, concerns about how configurable FFTF could be made, the general age of the facility (perhaps the largest concern), the interest in a demonstrator reactor and the report's general emphasis on building a new reactor. Each current MTR was deemed to be insufficient for the same concerns reconfiguring FFTF was not pursued. Table 4.6 summarizes those concerns. The ability of each reactor to meet a certain characteristic was analyzed as high, medium, or low. While the report did emphasize the construction of a new reactor, this concern could be overridden if some existing reactor was shown to meet the capabilities sought. Flux shaping within the proposed fast reactors could not be found in literature and their fast fluxes, while high, did not exceed 5E15 n/cm^2-s. The judgement criteria in Table 4.6 are a combination of constraints and objectives. At this stage in the design, constraints and objectives can be reformulated in a number of ways. The last requirement judges the US's ability to conduct R&D. It was not formally stated in the NEAC report, but is an unstated assumption based on the other requirements listed.

Table 4.6 Summary of reactor characteristics evaluated with respect to the requirements derived from the NEAC report. L, M, and H mean Low, Medium, and High respectively.

Characteristic	HFIR	ATR	JHR	MBIR	MYRRHA	FFTF
High fast flux	L	L	L	M	M	M
High thermal flux	H	H	H	L	L	L
Variable spectrum	L	L	L	M	M	M
Multiple coolant loops	L	M	M	M	L	L
Large volumes	L	H	M	M	M	M
Long lifetime	L	L	L	H	H	M
Demonstrator	L	L	L	M	M	M
Ease of use for US R&D	H	H	M	L	L	H

4.2 Second Level of Abstraction

The choice of fuel vector was simple. The use of highly enriched uranium is regarded as a proliferation risk within the US. The Reduced Enrichment for Research and Test Reactor Program (RERTR) has initiated in 1978 and since then has assisted in the refueling of over 40 reactors from HEU to LEU [RERTR, 2017]. This program and others within the DOE and NNSA discourage the usage of HEU in current and future test reactors. While not strictly prohibited, using LEU would ameliorate any such concerns. Similarly, there is no commercial source for plutonium in the US and defense related plutonium has never been made available for research reactor and likely never will. For these reasons, the fuel vector of this reactor should be LEU, with HEU as a very distant possibility. The JHR will use 27% enriched fuel until its new high density LEU fuel is qualified while the MBIR uses 38.5% plutonium MOX. The ATR and HFIR use 93% enriched fuel, while EBR-II used HEU and FFTF used MOX. (EBR-II and FFTF were used to study a variety of different fuel forms and vectors but HEU-Zr was the most common). The lack of high fissile content fuels would dictate the size of the core and ultimately its power. The SFR has received a great deal of R&D all over the world. Large units like MONJU, Superphenix, BN-800, and BN-1200 demonstrate the concepts viability. The TWR, proposed by Terrpower LLC, is a SFR [Gilleland, 2016]. No other Generation IV reactor type has been demonstrated to the same degree as the SFR. The VHTR is arguably the second closest with the THTR-300, Fort St Vrain, and HTTR. A small concept is being constructed in China, but the VHTR demonstrators are all smaller than the SFR concepts. No similarly sized MSR, LFR, or GFR has been constructed. While the VHTR is a viable technology, attaining a high fast flux would be impossible. Molten lead is corrosive to most kinds of steel and its high melting point can pose operational difficulties. For these reasons, the SFR was chosen as the baseline reactor. It will serve as a demonstrator reactor for future SFR projects and should have enough flexibility to serve as a research reactor for a variety of reactor types.

4.3 Third Level of Abstraction

Assembly design is a complex topic within SFR design. Coming up with a new assembly design is not difficult as long as it does not diverge from established concepts. Ensuring suitable performance of the new concept would be a time consuming process, especially when the benefits of a new assembly are hard to quantify and often cannot be determined except through actual usage of the assembly and assembly restraint systems. For these reasons, it was decided to use a preexisting assembly design and core restraint system. This posed the problem of choosing the best design from the inordinate number of concepts, both paper and physical. A durable, feasible near term design with excellent safety performance was desired; this meant choosing from actual reactor designs that had excellent operational characteristics and verified safety performance. Fortunately, several designs presented themselves. Fuel within FFTF and EBR-II achieved burnups equal to or exceeding 20% without incident [Fast, 2006]. Fuel handling accidents were uncommon in all reactors examined. Based on a literature review of historical SFR experience, durability appears to be easily satisfied as long as assembly design does not depart from the historical design envelope. Safety, measured as several accident performance, does differ between the historical SFR concepts. Severe accidents were tested in Rapsodie, EBR-II, and FFTF. Rapsodie successfully underwent a planned unprotected loss of flow accident at 50% power [Liquid, 2007]. Rapsodie, due to its high fissile content in the core, did not have an appreciable Doppler coefficient of reactivity. In unprotected accidents, a high Doppler coefficient is actually detrimental to the safety of the reactor. As core power declines and flow rate slows, the rods will decline in temperature. As the rods lose heat, positive reactivity will be inserted from the Doppler effect, maintaining core power and core temperature. However, mass flow rate is determined by natural circulation, and the steel cladding can easily fail from the excessively high power and low mass flow rate. High Doppler coefficients are a feature of oxide fueled reactors because of the softer spectrum induced through scattering off oxygen. The resonances play a greater role than in a metallic fueled reactor which has a harder spectrum. In fuels with more fertile than fissile material, the absorption resonances will dominate in this region and the Doppler coefficient will be negative. In fuels with more fissile than fertile material, the fission resonances will dominate the absorption resonances and the Doppler coefficient can be negligible or positive. Given the use of 30% plutonium MOX, it is likely that Rapsodie would not have performed so well in its test if it had been using LEU as required for the proposed design.

FFTF was also subjected to an unprotected loss of flow accident at 50% power [Liquid, 2007]. Gas expansion models, which increase neutron outleakage as coolant velocity decreases, had to be inserted into the core for these tests to be performed safely. Before the test the pumps were operated at a high mass flow rate to reduce the core temperatures so that the rise in temperature induced by the test would not breach safety limits. FFTF could not have withstood the same accident at full power. This stands in contrast to EBR-II, which underwent an planned unprotected loss of flow test at full power on April 3, 1986. No safety limits were breached. Only passive safety systems were used, meaning that this test simulated a complete loss of offsite power with partial loss of onsite power type accident. It is possible that Rapsodie or FFTF would have performed similarly to EBR-II with metallic fuel. The fuel, coolant, and steel volume fractions of the EBR-II are similar to those of other designs, so a change in assembly design may not significantly change the neutronics, as long as the power density stays the same. EBR-II fuel pins are much smaller than the FFTF, allowing for lower pin linear powers and temperatures for the same power density. For this reason, a change in assembly design may necessitate a decrease in power density to maintain safe fuel temperatures. EBR-II had excellent operational performance and safety performance; for these reasons its design will be used as the basis for the reactor.

Metallic fuels have varying material compositions, but the most promising fuel form is a uranium-zirconium alloy, with 10% zirconium by mass [Chang, 2007]. Metallic fuel has a high thermal conductivity compared to oxide fuels. Although the melting point of metallic fuels is lower than that of uranium dioxide, the higher thermal conductivity compensates in severe accidents [Chang, 2007]. Metallic fuels have higher thermal expansion coefficients than uranium dioxide, meaning that the fuel assemblies expand both vertically and axially with increasing fuel temperature more than uranium dioxide fueled assemblies. While uranium dioxide has a low thermal expansion coefficient, the steel pin will expand with temperature and drag the fuel with it, causing the fuel to expand with the steel. The magnitude of this safety critical phenomenon depends on the fuel pin design. Radial and axial expansion increases neutron leakage and increases the amount of sodium in the core [Chang, 2007]. More sodium in the core inserts negative reactivity and reduces the coolant temperature, which lowers pin temperatures. Zirconium present in the fuel improves the irradiation swelling behavior of the fuel. Metallic fuel assemblies have been successfully irradiated to burnups of 19.9 atom percent [Chang, 2007]. Although metallic fuel does expand with burnup, an initial smear density of ~75% leaves enough room within the fuel pin for the fuel to expand [Chang, 2007]. Metallic fuels have a higher density and uranium weight percent than oxide fuels [Chang 2007]. A higher fissile density dramatically increases k_{eff}. These favorable characteristics underlie Terrapower's choice to use metallic fuels in the Traveling Wave Reactor [Gilleland, 2016].

The harder neutron spectrum of a metallic fueled SFR relative to an oxide fueled SFR has some other neutronics effects besides a reduced Doppler coefficient. Neutron importance, or adjoint flux, is a measure of the effect a neutron at a given energy and position will have on k_{eff} [Waltar, 2011]. The neutron importance within fast reactors increases for neutrons with energies above ~100 keV, depending on the reactor. At higher energies, the probability of fission increases while capture decreases. The number of neutrons released per fission increases with incident neutron energy, an effect especially pronounced in plutonium [Waltar, 2011]. Scenarios which reduce neutron energy, like added oxygen to the fuel or increasing the amount of sodium in the core, will cause neutrons to drop in importance reducing k_{eff}. The effect is especially pronounced if neutrons are shifted downwards in energy enough for them to occupy the resonance region, dominated by the ^{238}U capture resonances. This is one of the reasons UO_2 fueled assemblies have a lower k_{inf} than metallic fueled assemblies (the other being the reduction in fuel density). Reductions in the density of sodium tend to increase k_{eff} due to the spectral hardening [Waltar, 2011]. As the density of sodium decreases with increasing temperature, and sodium temperature increases with core power, an increase in core power will cause an increase of reactivity from the change in sodium density. This positive power coefficient is ameliorated by the radial and axial fuel coefficients, but can become an issue if the core sodium boils. Large scale boiling of sodium could counteract the expansion. Such a scenario is highly unlikely in an SFR because of the large margin to sodium boiling incorporated into the design from the outset. Oxide fueled SFRs have softer neutron spectrum and lower neutron importance than metallic fueled SFRs; the addition of sodium will have less of an effect on neutron importance so the increase in reactivity induced by an increase in power will be lessened for oxide fueled SFRs [Judd, 2014]. Sodium has a small but appreciable capture cross section at higher energies; this has a negative effect on k_{eff} but is small compared to the neutron softening induced by scattering. The high scattering cross sections are beneficial on the periphery of the reactor, where sodium acts to reflect neutrons back into the core. For these reasons, the sodium worth (reactivity induced by a decrease in density) is likely to be positive in the center of the core and negative on the periphery of the core in medium and large SFRs. Leakage will be more significant in small SFRs and can cause the core average sodium worth to be negligible or slightly negative.

4.4 Fourth Level of Abstraction

The scope of this level is the core design, quite a large topic. A full list of the relevant design variables is provided in Table 4.7 while the objectives and constraints are provided in Table 4.8. The design variables, constraints, and objective functions are all quantitative, suggesting a mathematical optimization method. This project was meant as a proof of concept for a new materials testing reactor; it does not have to be optimal as long as the concept satisfies all constraints and is reasonably successful.

Table 4.7 Design variables of relevance in the fourth level of abstraction, divided between predetermined and undetermined variables. Predetermined variables were defined in a previous level of abstraction but are important in the current level. Undetermined variables are to be determined in the current level.

Predetermined variables	**Undetermined variables**	**Undetermined variables**
Pin diameter	Core power	Mass flow rate
Pin pitch	Cladding/assembly material	Core temperatures
Assembly pitch	Active core height	Location of control rods
Assembly wall thick.	Axial reflector height	Control rod design
Fuel vector	Reflector pin diameter	Moderator material
	Reflector pin pitch	Moderator positions
	Reflector material	Irradiation positions
	Assembly number/layout	Enrichment zoning

For this reason, a deterministic solution methodology was devised. This methodology had to be revised because it initially ignored the core pressure drop. Fortunately, the necessary design changes could be easily implemented because the EBR-

II had a gap between the reflector and fissile zones to accommodate the different pin pitches.

Table 4.8 Objectives and constraints applicable in the fourth level of abstraction

Obj./Con.	**Characteristic**	**Max./Min.**	**Importance**
Objective	Fast flux	Maximize	Strong
	Core lifetime	Maximize	Strong
	Core power	Minimize	Weak
	Core pressure drop	Minimize	Weak
Constraint	Coolant velocity less than 10 m/s		
	Core pressure drop less than 0.5 MPa		
	H/D=1		
	Maximum and average linear powers less than EBR-II		
	Fast flux greater than 5E15 n/cm^2-s		
	Independence of assemblies		

The first step was to perform a literature review of possible reflectors for fast reactors. Breeder reactors typically use ^{238}U, but this would yield a weapons grade plutonium vector in the reflector and this could be construed as a proliferation risk. Steel has appreciable absorption cross sections in the resonance region. Scattering is necessary to reflect neutrons back into the driver but does soften the spectrum. MgO was recommended in previous literature so it was used for the initial investigations as the radial and axial reflector [Macdonald, 2010]. The second step was to copy the EBR-II fuel assembly and build a full core with 19.75% enriched U-10Zr. As EBR-II used HEU, the critical mass of the same design with LEU would have to be much larger. The number of fissile assemblies and their height must increase because the assembly cannot significantly change. The axial reflector in EBR-II was 36.1 cm tall and the gap between the reflector and fissile assembly was 10.0 cm. The axial reflector for this reactor was 35 cm and the gap between the reflector and fissile regions was 5 cm to make the dimensions similar to those of EBR-II. EBR-II was designed for breeding; the thickness of the reflector was presumably designed to capture as many neutrons as possible suggesting that any further height would be unnecessary. In order to maintain H/D=1, 7 rings of radial reflector were used. Starting with an active height of 50 cm, the fissile zone was made taller and the number of driver assemblies was increased until a core with a k_{eff} of ~1.08 was attained. This gives over 10 dollars of reactivity, which was felt to be a suitable margin for excess reactivity this early in the design phase. Two years after the completion of this project, the research was reexamined for journal papers. The initial design of the axial reflectors used the EBR-II lattice which had unacceptably high pressure drops. This was not an issue in EBR-II because EBR-II was much shorter than this core. For this reason the axial reflectors were redesigned to have the same lattice and pin diameter as the driver while being 50 cm tall.

The third step was the calculation of average and peak linear power, which yielded a total core power of 600 MWth. This step should have involved the calculation of the core pressure drop, but this was ignored at the time. The peak fast flux E>0.1 MeV was greater than 5E15 n/cm^2-s, satisfying a design requirement. The fourth step was a series of scoping studies where the MgO was replaced by various materials. It was found that moderators with beryllium or carbon thermalized the spectrum, causing power peaking in the periphery of the core. Additionally, softening the spectrum reduces k_{eff} because of the neutron importance as previously mentioned. High atomic mass reflectors like PbO did not perform as well as others which matched the research presented in [Macdonald, 2010]. The fifth step was the design of the moderating region. Graphite was chosen as the moderator material because of its high tolerance to radiation, ease of manufacture, and non-toxicity. The graphite was located in steel assemblies so the moderating region could be easily reconfigured. The high thermal flux from the moderator caused extensive localized power peaking in the driver assemblies nearest the core, so specialized barrier assemblies where developed which had lower enrichments near the graphite. The enrichment within the barrier assemblies gradually increased, maintaining a reasonably flat power profile within the assembly while allowing as much fissile material to be placed in the assembly as possible. The sixth core design step was the locations of the irradiation positions. It was discovered that assemblies with sodium in them reduced the fast flux by 20%. Test assemblies would have to be designed with minimal sodium, probably filled with an inert gas. This fact was not investigated further. In core irradiation positions were filled with void inside the assembly, but the assembly steel wall and inter-assembly sodium were modeled. 21 assemblies were removed and replaced with irradiation test assemblies. The highest fast flux is in the center of the core. The moderator region only occupies one portion of the outer reflector and the core possesses bilateral symmetry about the x-axis. However, the moderator region and barrier assemblies are designed to have as little effect on the driver as possible. Thus, the central driver possesses six fold symmetry, as it would if the moderator were replaced by MgO reflector assemblies. This symmetry was taken into account when locating the test assemblies. To satisfy the requirements for large diameter high flux positions, seven assemblies were grouped together to form a large position. Two of these positions were created and were located in the outer driver. The inner driver contained seven assemblies; one in the center and six surrounding it. Scoping simulations were performed to find those positions which yielded the greatest fluxes. The removal of those 21 assemblies significantly reduced k_{eff}; fuel diameter was increased by 4.9% to increase the amount of fuel in the core. The irradiation test assemblies did not contain axial reflectors, increasing outleakage especially considering that the test assemblies are located in high importance regions of the core. More fuel had to be added (10% increase in fuel mass per assembly) than was removed to compensate for the increased outleakage.

The seventh and last step was the design of the control systems. Two systems were used. The shutdown rods replace some of the MgO assemblies and in conjunction with the shim rods bring the core to far subcriticality. The in-core shim rods replace some of the driver assemblies and use the same pin diameter and pitch as the fuel assemblies. The B_4C filled rods are followed by driver fuel. The rods are inserted from the top. The bottom 150 cm of the rods is B_4C, while the top 100 cm of the rods contain fuel. The driver is not axial reflected, being filled with sodium above and containing B_4C below. Natural boron was used. The worth of each assembly was less than one dollar, so a rod ejection accident would not result in supercriticality. Ten control rod assemblies were required. These changes to the core reduced the excess reactivity of a fresh core. The eighth step involved scoping calculations to increase the core lifetime. These ideas were not successful but relied on reducing the enrichment to enhance capture within the ^{238}U and breed plutonium. All ideas were unsuccessful. However, the burnup of the core and driver assemblies was highly linear, so linear reactivity theory could be used to enhance burnup through shuffling the core. With a three batch shuffling strategy a cycle length of ~100 day is possible and the driver would attain an average burnup of 45 MWd/kgU, a value typical of PWRs.

This methodology yielded a design which was not optimal but did satisfy the constraints. The calculation of core pressure drop only considered the pin bundle; the pressure drop would be greater with the inlets/outlets are included. The excellent safety performance behavior of EBR-II can be emulated if all facets of the design are copied. For this reason, the methodology sought to change the core design as little as possible. The Fast Reactor Database did not provide the core pressure drop which is necessary to ensure similar safety performance. In loss of flow accidents natural circulation provides the cooling mechanism of the core. The balance between pressure losses in the circuit and the hydrostatic pressure from differences in density drive the flow. The steady state pressure drop in the core provides some insight into flow resistance under natural circulation. EBR-II had a large core with far more inactive assemblies than active assemblies. For this reason the average coolant velocity was 0.5 m/s but the maximum coolant velocity was 8 m/s. Maximum coolant velocity would occur in the hottest assembly; therefore similar pressure drops would occur in that assembly if the assembly design were to be exactly copied. However, the assembly is taller than the prototypic and uses active pin lattice for the reflector rather than the breeder lattice used in EBR-II. The change in geometry nullify any attempts to appropriate EBR-II's safety performance to this reactor. For this reason, it was decided to limit the core pressure drop to a value typical for other reactors of ~600 MWth [Fast, 2006]. Reliability requires analyzing the designs sensitivity to variability in inputs, environment, and performance. This was not performed because it was felt to be redundant with respect to safety performance as safety performance cannot be assured. The methodology used in this level of abstraction was deterministic and cannot be used to derive alternative concepts. Even if sensitivity coefficients could be computed there would not be another concept to compare them with. Due to the method of evaluating the core pressure drop sensitivity coefficients for one design would not be significantly different for alternative designs that similar to the parent design. Maintaining an H/D=1 is not necessary to satisfy the other constraints but does make the design process easier. Core with H/D=1 will have lower critical masses than cores where H/D≠1 because H/D=1 minimizes neutron outleakage. Flattening the core so that H/D<1 would reduce the core pressure drop. To compensate for the reduction in core height more active assemblies would have to be added. This core would have a larger critical mass and active volume reducing the power density for a constant core power. Maximizing flux could be accomplished by reducing enrichment or increasing power density. Power density is maximized by increasing core power or decreasing volume. For these reasons, it is expected that core pressure drop will positively correlate with higher fast flux. This methodology yielded a design that could be an excellent starting point for an evolutionary algorithm, shown in Figure 4.2.

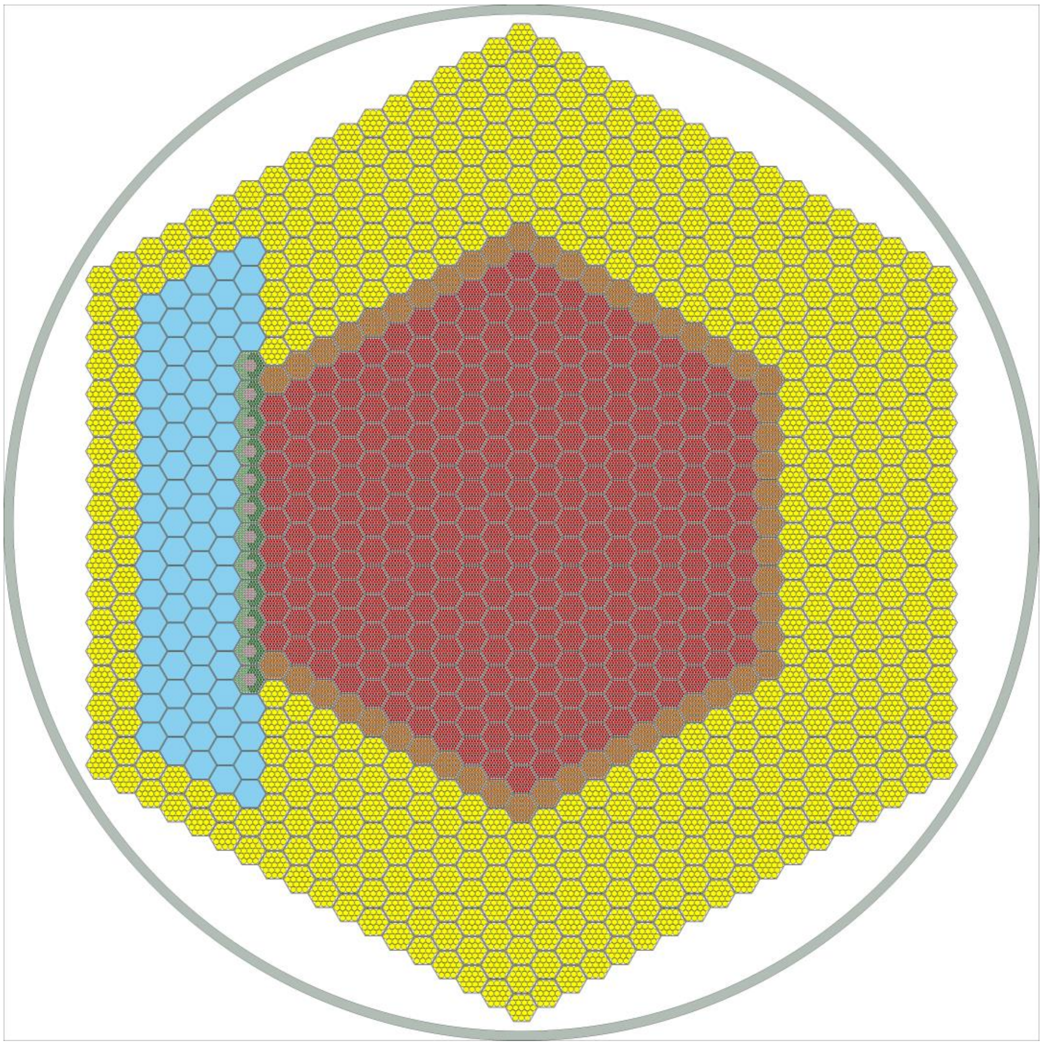

Figure 4.2 The core design at the end of the fourth level of abstraction. Red is 19.9% enriched U-10Zr, orange is 16% enriched U-10Zr, ligh blue is graphite, yellow is MgO, green is 15% enriched, pink is 12% enriched, light green is 6% enriched, white is sodium, and gray is steel.

4.5 Fifth Level of Abstraction

Flux within the outer reflectors is above 1E15 n/cm^2-s but is primarily in the epithermal range. Replicating the PWR and VHTR neutronic behaviors was achieved in the graphite moderator positions. There are two different kinds of test assemblies, active and passive. Active assemblies contain fissile material and are heated internally by fission. Passive assemblies do not contain fissile material and are heated externally through electric heaters or through gamma heating, neutron absorption, or neutron thermalization. A methodology similar to that used in the fourth level of abstraction was used in this section. The first step was to generate prototypic flux spectrum for the PWR and VHTR, deciding which locations within the PWR and VHTR are most in need of materials testing. One of the stated desires of the NEAC report was flow loop under prototypic conditions. For this reason the active assemblies were designed to mimic fuel assemblies within the core with flowing coolant. The coolant loops were not designed because their viability can be assumed through the extensive experience with flow loops in the ATR. The same flux spectrum (active core) was used in the passive assemblies. This may or may not be useful. The LWRS program is concerned with the performance of steel vessels after many years of stress corrosion cracking. The neutron spectrum at the core vessel is often quite different from the spectrum in the active core. For this reason the spectral adjusting done to exactly mimic the prototypic spectrum could be easily adjusted to any desired spectrum. Steel was used as a placeholder but a variety of materials, like potential claddings or graphite, could require a prototypic PWR or VHTR spectrum.

The second step was to develop prototypic configurations of the test assemblies, both similar to the prototype plants and similar to the standard test assembly configurations found in the ATR. Unfortunately, little information could be found online about exact dimensions of the transient test assemblies within the ATR. It is known that multiple layers of aluminum or steel are used to thermally isolate the test assembly from the core. The necessity of all the layers was not determined in this project, but it should be mentioned a water-sodium interaction would generate great quantities of sodium hydroxide and free hydrogen gas. This is not a favorable scenario. The PWR assemblies were double walled for this reason. The VHTR assemblies were not doubled walled because of the inert nature of helium gas and because a thick graphite block touches the steel wall, limiting sodium penetration in the event of a breach. Dimensions for the PWR assembly were derived from the AP1000 Design Control Document [AP1000, 2011]. A single infinitely reflected PWR pin was simulated in Serpent 2 and the flux spectrum for this simulation was used. Flux spectrum for prototypic PWR pins is different from an infinitely reflected 2.6% enriched pin, especially over the course of its lifetime. Reasons for this include: burnable absorbers hardening the spectrum of BOL while being ineffectual at EOL, control rods hardening the spectrum, burnup reducing enrichment and softening the spectrum, surrounding assemblies affecting spectrum, and the overestimation of fission neutrons implicit in all Monte Carlo codes where k_{eff} is significantly above 1. For all these reasons, the flux spectrum generated in this project would not match those in an actual reactor. However, it is not necessary that it exactly match. An enormous variability it spectrum shaping is possible; the spectrum presented here could be easily modified with the techniques used in the project demonstrating the viability of the overall concept. The spectrum, although inexact, is needed to provide something to benchmark against. The VHTR conditions were derived in a different fashion. The physical design of the assembly is derived from an MCNP deck provided by Dr. Tsvetkov for the HTTR, a VHTR test reactor located in Japan. The spectrum was derived from research presented in [Sterbentz, 2008]. The GT-MHR was used in [Sterbentz, 2008] and the four group flux spectrum from the central ring of fuel assemblies was used for benchmarking because that ring had the highest fast flux. Fast flux is the most damaging to materials. The prototypic HTTR assembly was simulated at the GT-MHR power density. Again, not realistic but the ease with which flux was shaped does not change the viability of the concept. The PWR was also simulated at the average power density of the core. The passive assemblies are modeled as steel cylinders 2 cm in diameter. The pins are cooled with a 0.5 cm thick sodium channel in direct contact with the steel. This is surrounded by a thin steel wall separating the sodium from the moderator. Only one pin is located in each test assembly to accommodate enough moderator.

Step three sought to match the prototype fluxes and power densities in the test assemblies. It was desired to maximize the flux while keeping the flux spectra as close to prototypic as possible. For the active assemblies this was accomplished by reducing the enrichment from prototypic. Gamma heating or neutron heating was ignored so power density was not considered in the passive assembly. Whenever possible, $ZrH_{1.6}$ was used in place of water for flux shaping. The most important constraint is keep the peak linear power in the barrier assemblies ~200 W/cm. The assemblies must be kept independent to preserve the configurability of the moderating region. $ZrH_{1.6}$ generates a high thermal flux. The shorter path-length of thermal neutrons compared to epithermal neutrons results in power peaking in the barrier assemblies with $ZrH_{1.6}$. For this reason absorber foils were used to block thermal neutrons. Blocking the thermal neutrons does prevent them from fissioning within the barrier assemblies and reduces k_{eff}. The number of possible configurations is too great for this project to examine. Therefore, as in level four, a deterministic methodology was developed which satisfies the constraints and maximizes the fluxes. Some consideration was given to the minimization of reactivity penalty. The reactivity penalty is greatest when a high thermal flux is present in the moderating region for three reasons. A softer spectrum has less neutron importance; low energy neutrons have shorter path-lengths and do not penetrate the core as readily as high energy neutrons. Neutrons on the periphery of the core have lower importance than neutrons in the center of the core. These two effects act to reduce k_{eff}; when the flux foils are used k_{eff} drops even more. Characterizations of the graphite region revealed that the peak fluxes were on the core centerline. For this reason the prototypic test assemblies were placed on either side of the core centerline, replacing graphite assemblies. The core has twofold symmetry and two test assemblies. This preserved symmetry and doubled the tallying positions, reducing the stochastic error by ~1.414. The moderating region is four assemblies across. The test assemblies were simulated at different distances from the core. The PWR had a higher fast flux fraction (flux in group over total flux) than the VHTR. For this reason the positions near the core were more suited to PWR testing while the positions far from the core were suited for VHTR testing. The VHTR assemblies had too great of an epithermal flux so

$ZrH_{1.6}$ assemblies were used to enhance the thermal flux. Conversely, the thermal flux in the PWR assemblies was too high so water was removed from the test assembly. Boron carbide was used in the VHTR and PWR test assemblies to exactly match the thermal and epithermal flux. At no point in this study were all four flux fractions matched to within 5%. Fine group flux calculations for the PWR showed that even when the four group flux fraction approximately matched, there were significant discrepancies. Test assemblies require fissile materials. The general scheme for this process was to match the flux fractions while holding the enrichment constant. Then once flux fractions were reasonable, enrichment was altered to match the desired power. Changing enrichment changed power, which changed the number of fissions in the test assembly, which changed the spectrum. However, since the majority of the neutrons within the test assembly came from without, this feedback mechanism was very low. The passive assemblies were modified in the same way. First the capsule (steel pin, sodium channel, steel wall) were moved along the core's x-axis. Then varying amounts of $ZrH_{1.6}$ and graphite were loaded into the assembly to achieve reasonable flux fractions. The capsules had to be located near the core to attain as high a fast flux as possible. It proved impossible to exactly match the fast flux fractions in the PWR or VHTR. The SFR contains massive nuclides which allow for scattering but minimize the energy loss in each collision. Thermal reactors contain low mass nuclides which maximize the energy loss per collision. For this reason fission neutrons in a PWR need only one or two scatters to fall out of the fast flux range ($E>0.1$ MeV). Dozens of scatters are needed in a SFR. The fast flux in the SFR is softer than the fast flux in a thermal reactor for this reason, even though the total flux in an SFR is harder than the total flux in a thermal reactor. This is shown in Figure 4.3.

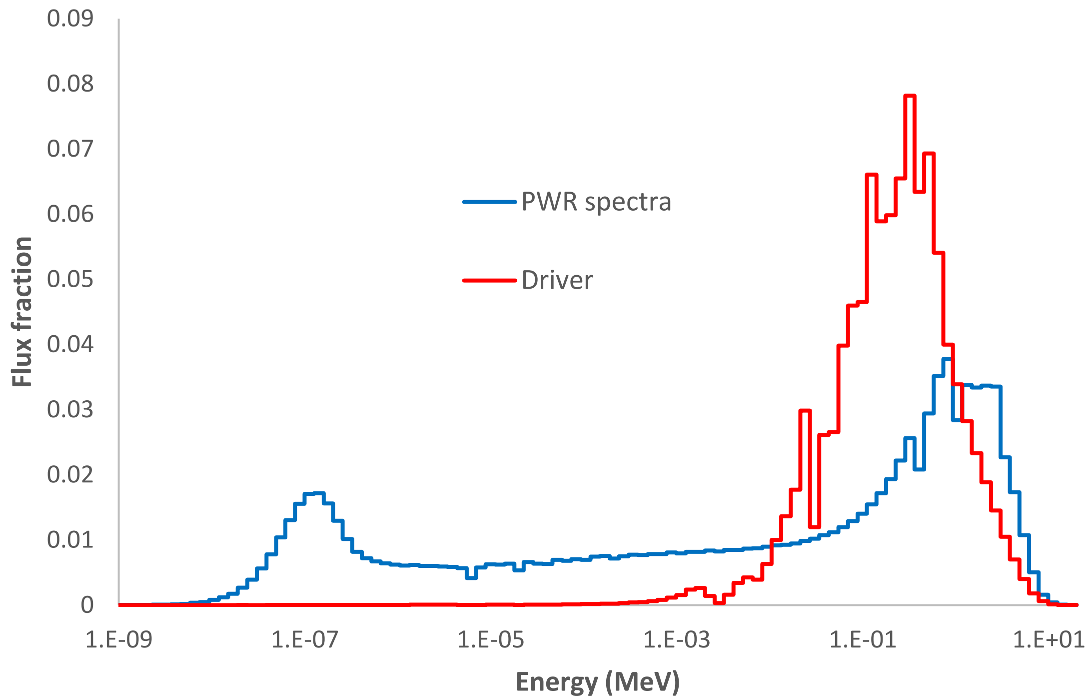

Figure 4.3 120 group flux in the prototypic PWR pin and in the central driver. Flux fraction is the flux in each group divided by total flux. The PWR spectra was tallied over the entire pin cell and the Driver was tallied over the entire active assemblies.

The design of a fast spectrum materials testing reactor presented in [Scherr, 2016] contained a unique concept that had no prior implementation. A method to change the fission rate within a test assembly was proposed. The test assembly is surrounded by control rods, the movement of which affects the fission rate in the test assembly. The control rods were tipped by graphite or MgO, and their movement affects the reflection of neutrons back into the core. Having to quickly change the shims to accommodate rapid changes in the test assembly fission rate could cause operational difficulties, so it was desired to minimize the reactivity worth of the transient test region. This was accomplished by cadmium doped MgO reflector assemblies. Cadmium has a very high thermal cross section and a low epithermal cross section. In small quantities fast and epithermal neutrons are either reflected or pass through the reflector assemblies. The control rods rest in light water and are well thermalized. Neutrons that enter the water filled control rod assemblies are absorbed in the cadmium if they escape the region. PWR, VHTR, and SFR test assemblies were simulated in this trap. The reactivity inserted by the movement of the rods was ~0.10$ and power in the test assembly could be reduced to 2% of nominal. Choosing the location of this flux trap was simple. It should be as far from the driver as possible so that as many neutrons are reflected back into the core as possible.

4.6 Summary of the FSMTR

The design process of a materials testing reactor to support LWRS and Advanced reactor programs was outlined, demonstrating the utility of the method outside of the field of power reactors. Design of a materials testing reactor places more emphasis on the in core behavior and less emphasis on the system behavior. Core design is highly neutronics focused and the problem definitions are well suited to evolutionary algorithms. Such algorithms were not utilized because the design was performed as a feasibility study. Deterministic algorithms were used to solve the levels of abstraction in place of mathematical optimization. Literature review was essential in the higher levels and was used in place of formal PIRT. The first level of abstraction was concerned with whether to build a new materials testing reactor or modify an existing reactor. Solved with an extensive literature review and a qualitative comparison of existing materials testing reactors, it was decided to design a new reactor despite the additional cost. The first level of abstraction was concerned with the information in the DOE report. The qualitative judgments outlined in that report where translated into decision variables and discretized into high, medium, or low for judging the suitability of current reactors. Thermal fluxes and in core volumes were judged on projected future needs and current abilities of the reactors. The second level of abstraction choose the coolant type and fuel vector. Literature review proved sufficient. There was no need for a comprehensive analysis like in the first level of abstraction. LEU is the required fuel vector, sodium has far more R&D, and metallic fuels have clear safety benefits.

The third level of abstraction defined the assembly. This level was the most difficult to perform. Assembly design is a very complex topic and must be considered with the choice of core restraint system, core temperatures, core shuffling systems, core layout, and fuel dimensions. Rather than optimizing all of these parameters, it was decided to simply adopt a system with proven performance. A literature review selected the EBR-II system as the ideal candidate, and the core design was based on that reactor. It was chosen primarily for its excellent >30 year operational record, where it achieved high annual capacity factors and had no major sodium leaks. The fourth level of abstraction defined the core layout. This level was perhaps the first quantitative level, where every design variable could be assigned some mathematical value. The fourth level was solved with a deterministic scheme that satisfied the design constraints but did not seek an optimal solution. The fifth level of abstraction concerned the irradiation positions. While quantitative, a deterministic solution methodology was also used. The utility of the method can be understood by considering that the fourth and fifth levels of abstraction define the core performance requested in the report. Five levels of abstraction were needed to completely satisfy all of the qualitative requirements of the report. These levels also ensure that alternatives to the final design are considered at least conceptually and not overlooked. Considering design variables best examined at different levels of abstraction (such as coolant type and core layout) would needlessly complicate the design process. Determining the design variables in a hierarchical sequence allows for greater focus on the system behavior while ensuring that all aspects of the design are considered. Design variables are combined at the higher levels of abstraction. A concept like safety performance is considered in level 2 then translated to the number and type of accidents as well as capacity factor and overall risk in level 3. Safety was further considered in level 4 through power density and pressure drop, becoming progressively quantitative through the levels. Ultimately, PRA and deterministic safety analysis would be needed before the reactor design could be truly complete. The translation of abstract design parameters like safety into concrete design parameters like maximum power density and pressure drop was accomplished through literature review. The process, and the detailed consideration of different design variables at different levels of abstraction, exemplify the benefits of the method.

5. ADVANTAGES OF THE METHOD

The proposed method is superior to the other methods for three reasons which will be elaborated on within this section. First, it prioritizes the generation of constraints/objectives over the design parameters. Heuristics are provided for the decomposition of constraints/objectives, a feature not found in any previous literature although it is of paramount importance. Second, the method includes fewer mandatory steps per generation. Thirdly, it uses multiple generations to evolve the constraints/objectives from the abstract to the concrete.

Every method will be proficient for solving some design problems but not proficient for others. The goal of this dissertation was to derive an engineering design methodology that is tuned for nuclear engineering. Once developed it was observed that this method may have benefits outside of nuclear engineering, although this possibility remains largely unexplored. Central to this method is the assertion that constraint selection is more important in the process than creation of design parameters or optimizing the design variables. This assertion stems from experience with nuclear system design and is exemplified in the fast materials testing reactor, where only broad objectives and constraints are provided. The best scheme to design the fast materials testing reactor is one which decomposes the high level constraints and objectives to low level constraints and objectives. This feature is utterly absent in the other engineering design methods. Traditional top down methods assume that the problem is fully defined. The Spiral model is well suited to prototyping, where the entire system is created then progressively altered.

The Waterfall model contains elements of decomposition but would have difficulty adapting to the needs of alternative nuclear designs. The Vee model decomposes the design requirements then constructs the objectives and constraints before solving the engineering design problem. While useful many iterations may be needed and the Vee model would prove cumbersome between generations. These models, especially the Vee model, are quite useful but can include many (potentially redundant) steps. A method was desired that consisted of as few steps as possible but that could be repeated until a suitable design was achieved. This scheme can be regarded as a generalization of the previously mentioned models. The proposed method can be reduced to a single generation and the steps from the Waterfall and Vee models incorporated into the generation. This is not recommended. Aspects of the steps within the Waterfall and Vee models have been incorporated as heuristics to guide the method. Axiomatic Design is analyzed in detail in section 7.5 and it did not demonstrate utility in nuclear system design. Robust Design is not an engineering design method per se but is a statistical methodology to demonstrate the resilience of a system. Certain aspects of Robust Design have been incorporate as heuristics.

6. CONCLUSIONS

This dissertation was created in response to three perceived concerns in the field of nuclear design. The first concern was the realization that nuclear systems are too complex for all design parameters to be considered simultaneously. The second concern is the lack of a nuclear system design methodology or a generic engineering design methodology that could be adapted for nuclear systems. The third is the deep seated intuition that the choice of constraints is essential in engineering design. Axiomatic Design was regarded as the most comprehensive engineering design method but its heuristics were found to be of little use to nuclear systems. However, Axiomatic Design shows potential for broad systems level analysis. It is impossible to restrict the number of design parameters which affect functional requirements, an important precept of Axiomatic Design (sections 1.5 and 7.5). Traditional top down engineering design assume constraints and guide the choice of design concepts. The same is true of the Waterfall and Vee models, which operate over hierarchical levels of abstraction. These methods are all insufficient. The traditional top down engineering design scheme and its derivatives are robust but provide limited assistance in choosing constraints, the most important aspect of design (section 1.4). Evolutionary algorithms are a class of mathematical optimization method and have proven themselves useful in solving engineering design problems (section 1.3). As currently conceived evolutionary algorithms are unsuitable as an engineering design methodology and it was decided to adapt evolutionary algorithms for this purpose. A general description of the characteristics of nuclear reactor design is provided followed by a description of the method with examples (section 2). The proposed methodology went through several phases before arriving at the form presented in section 2. The first phase used evolutionary algorithms but modified the constraints and objectives at each generation. This was called the progressive generation of constraints. Heuristics which guided the creation of new constraints where then developed. The development of these heuristics lead to the levels of abstraction concept where appropriate design parameters evolve out of the translation of abstract concepts to more concrete objectives and constraints. The meta-heuristics of Axiomatic Design are simple but powerful; equally powerful heuristics were developed for the levels of abstraction. Two heuristics are regarded as of the utmost importance. The first heuristic for the levels of abstraction is to conceptualize the design process, best understood through nested system hierarchies. The second heuristic is to solve the levels in order of decreasing abstraction. Throughout section 3 additional heuristics are provided on conceptualizing the systems, identifying constraints, picking the best solution method, ensuring nuclear safety, and general best nuclear design practices. This methodology, described in section 2 and further developed in section 3, was used to study a fast spectrum materials testing reactor. The method is agnostic as to the reactor and solution methodology, a characteristic aptly demonstrated by the complete omission of analysis of the fast spectrum reactor (section 4). This example was used to analyze Axiomatic Design. The appendix (section 7) contains important information that supports or clarifies content located in the main body. Historical precedent for the levels of abstraction is provided in section 7.1. The literature review and discussion of sodium fires demonstrates how to perform an engineering literature review and how to draw safety conscience conclusions for first-of-a-kind facilities. Robust Design (section 7.4) is incorporated into the safety heuristics in section 3.6. A brief primer on PIRT (section 7.6) provides principles incorporated into the heuristics.

The method begins at the highest possible level of abstraction, the overall purpose of the reactor. Some combination of three options are available: electricity, process heat, or neutrons for research. The engineer selects the quantity and quality of electricity, heat, or neutrons then moves to the second level of abstraction. The other levels are not specified and the engineer has enormous freedom in selecting the levels although sections 2.2 and 2.3 provide potential outlines which should be applicable for the majority of systems. The current level gives concreteness to the preceding levels objectives and constraints while referencing the chosen design parameters of the preceding level. New objective and constraints are added to more precisely define the preceding objectives and constraints. The new design parameters are also chosen to expand on the preceding design parameters. This process may be understood as the decomposition of nested system hierarchies (section 2). Constraints may be defined through consideration of system inputs, outputs, environments, and functions. Purpose is regarded as a combination of objectives and constraints. Constraints should be stated as simply as possible. While the extent level should be minimized, it is possible to create very large levels and then subdivide them if they prove too difficult to solve. The unstated goal of this dissertation was to find a general engineering design framework or philosophy taking a form similar to Axiomatic Design. This is likely impossible. The method outlined in this dissertation requires extensive understanding of the system behavior although it is hoped that sufficient heuristics have been provided to encourage the engineer to discover the constraints of his system. Advice for general nuclear system design is the most underdeveloped portion of this dissertation and could be the subject of future work.

REFERENCES

A comparison of US and Japanese regulatory requirements in effect at the time of the Fukushima accident. USNRC, 2013.

AP1000 Design Control Document: Revision 19. Westinghouse Electric Company LLC, 2011.

Abderrahim, H. A. et al. "MYRRHA – A multi-purpose fast spectrum research reactor." Energy Conversion and Management 63, 2012.

Accident Analysis for Nuclear Power Plants. IAEA, Safety Report Series No. 23, 2002.

"Advanced Reactor Design Criteria (ARDC) Development Process." NRC Public Meeting, 21 January, 2015. US DOE. Presentation.

Allen, Janet K. *Robust Design for Multiscale and Multidisciplinary Applications.* Journal of Mechanical Design. Volume 128, July 2006.

Boyard, M. et al. "The Jules Horowitz Reactor core and cooling system design." *JOINT MEETING of the National Organization of Test, Research, and Training Reactors and the International Group on Research Reactors September 12-16, 2005 at the Holiday Inn, Gaithersburg, MD.* National Institute of Standards and Technology, 2005.

Bragg-Sitton, S. "Status of SiC Research for Accident Tolerant Fuels." Idaho National Laboratory, US Department of Energy, 2013.

Bredimas, Alexandre and Nuttall and William J. *A Comparison of International Regulatory Organizations and Licensing Procedures for New Nuclear Power Plant*. Energy Policy Research Group, University of Cambridge, 2007.

Buede, Dennis. *The Engineering Design of Systems: Models and Methods*. Wiley Online Library, 2009.

Chang, Yoon Il. "Technical Rational for Metal Fuel In Fast Reactors." *Nuclear Engineering and Technology*. Volume 39 No. 3 (2007): 161-170.

Cochran, Thomas B. et al. *Fast Breeder Reactor Programs: History and Status.* International Panel on Fissile Materials, 2010.

Comparison of Canadian NPP Design Requirements with those of Foreign Regulators. CNSC, RSP-0273, 2011.

Comptroller and Auditor General, *Nuclear Power in the UK.* National Audit Office, The Department of Energy and Climate Change, 2016.

de Lima, Alan M.M. et al. "A nuclear reactor core fuel reload optimization using artificial ant colony connective networks." *Annals of Nuclear Energy,* Volume 35, 2008, Pages 1606-1612.

Design Features and Operating Experience of Experimental Fast Reactors. IAEA, No. NP-T-1.9, 2013.

Deterministic Safety Analysis for Nuclear Power Plant. IAEA, Specific Safety Guide No. SSG-2, 2009.

Dieter, George Ellwood. *Engineering Design.* McGraw-Hill Inc, 1983.

Dieter, George Ellwood. *Engineering Design A Materials and Processing Approach.* McGraw-Hill Inc, 1991.

"Fast Reactor Database 2006 Update." Vienna: International Atomic Energy Agency, 2006. IAEA-TECDOC-1531.

"FY 2009 Advanced Test Reactor National Scientific User Facility Users' Guide." Idaho Falls: Idaho National Laboratory. INL/EXT-08-14709. 2009.

General Design Criteria for Nuclear Power Plants. 10 CFR 50 Appendix A, 2015.

Gerstner, D. and Davis, C. "Thermal-Hydraulic Analysis Results of a Seismically-Induced Loss of Coolant Accident Involving Experiment Out-of-Pile Loop Piping at the Idaho National Laboratory Advanced Test Reactor." *EFCOG 2012 Safety Analysis Workshop May 2012 Santa Fe, NM.* Idaho National Laboratory, 2012. Presentation.

Gibson, John E. *Introduction to Engineering Design.* Holt, Rinehart, and Winston Inc, 1968.

Gilleland, J. et al. "The Traveling Wave Reactor: Design and Development." *Energy,* Volume 2, Issue 1, 2016, Pages 88-96.

Gregory, Sidney A editor. *The Design Method.* Butterworth Inc, 1966.

Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors. Idaho National Laboratories, US DOE, INL/EXT-14-31179 Revision 1. 2014.

Hayes, Steve. Personal conversation, 2015.

Hilton, B. A. et al. "U.S. Plans for the Next Fast Reactor Transmutation Fuels Irradiation Test." *Proceedings of GLOBAL 2007 conference on advanced nuclear fuel cycles and systems*, La Grange Park: American Nuclear Society, 2007.

Jiang, S. et al. "Estimation of distribution algorithms for nuclear reactor fuel management optimisation." *Annals of Nuclear Energy,* Volume 33, 2006, Pages 1039-1057.

Judd, A. M. *An Introduction to the Engineering of Fast Nuclear Reactors*. Cambridge University Press, 2014.

Kumar, Akansha and Tsvetkov, Pavel V. "A new approach to nuclear reactor design optimization using genetic algorithms and regression analysis." *Annals of Nuclear Energy*, Volume 85, 2015, Pages 27-35.

La Porte, Todd R. *High Reliability Organization: Unlikely, Demanding and At Risk.* Journal of Contingencies and Crisis Management. Volume 4, Number 2 June 1996.

Lee, Dai Gil and Suh, Nam Pyo. *Axiomatic Design and Fabrication of Composite Structures.* Oxford University Press, 2006.

Lee, Kwang Y. and El-Sharkawi, Mohamed A. editors. *Modern Heuristic Optimization Techniques Theory and Applications to Power Systems.* John Wiley & Sons, 2008.

Leveson, Nancy. *A New Accident Model for Engineering Safer Systems.* Safety Science. Volume 42 No 4, April 2004.

Leveson, Nancy et al. *Moving Beyond Normal Accidents and High Reliability Organizations: A Systems Approach to Safety in Complex Systems.* Organization Studies. Volume 30, Number 2 & 3, 2009.

“Light Water Reactor Sustainability Program Integrated Program Plan.” U.S. Department of Energy Office of Nuclear Energy, Revision 2, 2014.

Liquid Metal Cooled Reactors: Experience in Design and Operation. IAEA, IAEA-TECDOC-1569, 2007.

Lui, Dahai. *Systems Engineering Design Principles and Models.* Taylor and Francis Group LLC, 2016.

Luo, Hu. “Quantified PIRT and Uncertainty Quantification for Computer Code Validation.” Oregon State University, November 2012. Dissertation.

Macdonald, R. R. and Driscoll, M.J. “Magnesium Oxide: An Improved Reflector for Blanket-Free Fast Reactors.” *Transactions of the American Nuclear Society* Volume 102 (2010): 488-489.

Middendorf, William A. *Engineering Design.* Allyn and Bacon Inc., 1969.

Mishra, Surendra et al. “Optimization of depleted uranium loading in fresh core of large sized Indian PHWR by evolutionary algorithm.” *Annals of Nuclear Energy,* Volume 38, 2011, Pages 905-909.

Montes-Tadeo, Jose-Luis et al. “Searching for enrichment and gadolinia distributions in BWR fuel lattices through a Heuristic-Knowledge Method.” *Progress in Nuclear Energy*, Volume 85, 2015, Pages 213-227.

Nourbakhsh, H. P. *An Overview of Differences in Nuclear Safety Regulatory Approaches and Requirements Between United States and Other Countries*. Advisory Committee on Reactor Safeguards (ACRS), US NRC, 2004.

NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness. US NRC, ML16356A67, 2016.

NUCLEAR SAFETY: Countries' Regulatory Bodies Have Made Changes in Response to the Fukushima Daiichi Accident. USGAO, GAO-14-109, 2014.

Olivier, Tara J. et al. *Metal Fire Implications for Advanced Reactors, Part 1: Literature Review.* Sandia National Laboratories, SAND2007-6332, 2007.

Olivier, Tara J. et al. *Metal Fire Implications for Advanced Reactors, Part 1: Literature Review.* Sandia National Laboratories, SAND2008-6855, 2008.

Pahl, Gerhard and Beitz, Wolfgang. *Engineering Design.* The Design Council, 1984.

Pakhamov, I. "Approaches to Resolve Safety Issues Related to Sodium as a Fast Reactor Coolant." Second Joint GIF-IAEA/INPRO Workshop in Safety Aspects of Sodium-Cooled Fast Reactors, 30 November – 1 December 2011. Presentation.

Pereira, Claudio M.N.A and Lapa, Celso M.F. "Coarse-grained parallel genetic algorithm applied to a nuclear reactor core design optimization problem." *Annals of Nuclear Energy*, Volume 30, 2003, Pages 555-565.

Pereira, Claudio M.N.A and Sacco, Wagner F. "A parallel genetic algorithm with niching technique applied to a nuclear reactor core design optimization problem." *Progress in Nuclear Energy,* Volume 50, 2008, Pages 740-746.

Ray, Martin S. *Elements of Engineering Design.* Prentice Hall International, 1985.

"Report of the Nuclear Reactor Technology Subcommittee." Nuclear Energy Advisory Committee, US Department of Energy, 2014.

"RERTR Reduced Enrichment for Research and Test Reactors," *Nuclear Engineering Division at Argonne,* December 6 2017, http://www.rertr.anl.gov/index.html.

Sacco, Wagner F. et al. "Two stochastic optimization algorithms applied to nuclear reactor core design." *Progress in Nuclear Energy,* Volume 48, 2006, Pages 525-539.

Sacco, Wagner F. et al. "A Metropolis algorithm combined with Nelder-mead Simplex applied to nuclear reactor core design." *Annals of Nuclear Energy,* Volume 35, 2008, Pages 861-867.

Safety of Nuclear Power Plant: Design Specific Safety Requirements No. SSR-2/1 Rev. 1. IAEA, 2016.

Sastry, Rahul and Siegal, Bennett. *The French Connection: Comparing French and American Civilian Nuclear Energy Programs*. Stanford Journal of International Relations, Spring 2010.

Scherr, Jonathan B. "Fast Spectrum Materials Testing Reactor with variable energy spectra to support Advanced Reactors Program and Light Water Reactor Sustainability Program R&D." Texas A&M University, May 2016. Thesis.

Scherr, J. and Tsvetkov, P. V. "Reactor design strategy to support spectral variability within a sodium-cooled fast spectrum materials testing reactor." *Annals of Nuclear Energy*, Volume 113, March 2018.

Simon, Harold A. *A Student's Guide to Engineering Design.* Pergamon Press Ltd, 1975.

Sleeper, Andy. *Robust Design Simplified.* Successful Statistics LLC, 2007.

Sterbentz, J. W. *Calculated Neutron and Gamma-Ray Spectra Across the Prismatic Very High Temperature Reactor Core.* Idaho Falls: Idaho National Laboratory, 2008. INL/CON-08-13845 Preprint.

Subo, K. "Approaches to Resolve Safety Issues Related to Sodium as a Coolant." Second Joint GIF-IAEA/INPRO Workshop in Safety Aspects of Sodium-Cooled Fast Reactors, 30 November – 1 December 2011. Presentation.

Tavron, Barak and Shwageraus, Eugene. "Pebble bed reactor fuel cycle optimization using particle swarm algorithm." *Nuclear Engineering and Design*, Volume 307, 2016, Pages 96-105.

"Technology Roadmap Update for Generation IV Nuclear Energy Systems." OECD Nuclear Energy Agency for the Generation IV International Forum, January 2014.

The ASME Presidential Task Force on Response to Japan Nuclear Power Plant Events. *Forging a New Nuclear Safety Construct.* ASME, 2012.

"the westinghouse [sic] pressurized water reactor and nuclear power plant." Westinghouse Electric Corporation, Water Reactor Divisions, 1984.

Tretiyakov, I.T. et al. "MBIR, a Reactor Facility for Validation of Innovative Designs." Moscow: NIKIET, 2014.

Van Tichelen, K. "MYRRHA Multipurpose hYbrid Research Reactor for High-tech Applications." Kraftwerkstechnisches Kolloquium 2014: Kernenergestisches Symposium Dresden, Germany, 14-15 October, 2014.

Waltar, A. E. et al. *Fast Spectrum Reactors*. Springer Science & Business Media, 2011.

Wang, Chang et al. "Parametric optimization of steam cycle in PWR power plant using improved genetic-simplex algorithm." *Applied Thermal Engineering,* Volume 125, 2017, Pages 830-845.

Wilson, Gary E and Boyack, Brent E. "The role of the PIRT process in experiments, code development and code applications associated with reactor safety analysis." *Nuclear Engineering and Design,* Volume 186, Issues 1-2, November 1998, Pages 23-37.

Xoubi, N. and Primm, R. T. III. "Modeling of the High Flux Isotope Reactor Cycle 400." Oak Ridge: Oak Ridge National Laboratory. ORNL/TM-2004/251. 2005.

APPENDIX

This section contains the literature review for some of the heuristics and a historical example of nuclear system design. This literature review provides the background and proof for the heuristics given in the main dissertation. The historical example outlines how the levels of abstraction were implicit in the decisions made by the engineers. The safety heuristics are generally applicable to all reactor types while specific heuristics are given for sodium fires in SFRs. Reliability oriented engineering design schemes are compared with Robust Design. The utility of Axiomatic Design is explored first with respect to a fast spectrum MTR (presented in Chapter 4) and then with respect to PWRs and BWRs. Axiomatic Design appears useful for nuclear system design but not for core design.

A.1 The Development of BWR Pressure Suppression Containment

The proposed method is outlined here in a historical example of nuclear system design. There are four actors in this story: Pacific Gas and Electric (PG & E) which will own the reactor being designed and has most of the financial responsibility, General Electric (GE) which developed the BWR and will be providing design work and fuel, Bechtel which is the construction contractor, and Atomic Energy Commission (AEC) which is responsible for licensing the reactor. Design work commenced in 1958 and the final design was approved by the AEC in 1960. All the constraints of the design were not known at the outset of the problem but emerged in conversations between the various entities. The design can be interpreted as occurring at the same level of abstraction or at different levels. The engineers took a limited approach and planned for multiple stages of design and generated concepts which were attacked by the other entities who added different constraints to the design. If these constraints could have been known from the outset then the design could have proceeded more smoothly. However, the multiple levels and multiple stakeholders ensured that all of the constraints were implemented. Basic physics about steam in large pools of water was not known at the outset so essential physics phenomenon would have to be investigated before any design work could be finalized. As the licensing body, the AEC was ultimately responsible for the final design decisions.

A containment is a large structure that surrounds a nuclear reactor vessel and is designed to prevent the release of radionuclides in the event of an accident. In the event of a break in the primary system of a LWR, large volumes of steam and water would be released pressurizing the containment. For this reason, the PWR uses a large steel lined reinforced concrete building to accommodate pressure spikes by providing enough air volume for the steam to expand without breaching the containment. The thickness of the wall is determined by the size of the break (which is determined by regulations) and the total volume of the containment. A thicker but smaller containment proved more expensive than a thinner but larger containment. PG & E and GE were interested in reducing the capital cost of a BWR for their proposed 50 MWe Humboldt Bay reactor, just north of San Francisco. In an informal meeting between a GE executive and a PG & E executive, the GE executive informed the PG & E executive about the pressure suppression concept which reduces the size of a BWR containment without increasing its thickness. PG & E were very excited about this concept as they were largely responsible for financing the reactor. Pressure suppression works by condensing steam into a large pool of cool water which absorbs the high pressure steam, heating slightly in the process. This reduces the pressure spike upon a break in the primary piping, enabling a smaller containment volume. A smaller containment volume results in considerable cost savings despite the need to build large water tanks. If located around the reactor, this water could be used as shielding saving money on concrete. Fission products are likely to be stored in the water in the event of an accident. This water also provides a convenient source for flooding the core in case of an accident. A smaller containment would enable the core to be built below ground, making it look more like a conventional power plant and potentially easing the public's concerns with nuclear power. While all of these attributes encouraged the development of the pressure suppression concept, PG & E focused mainly on the reduction in capital cost. PG & E had decided to contract with Bechtel to design the reactor and their inputs would become important in this story.

PG & E requested that GE submit a proposal outlining the development of the pressure suppression containment. The proposal sought to determine the maximum credible accident, appropriate design to ensure that the steam was injected and condensed in the pool, the degree to which the pool trapped fission products, and the use of the pool as a source of emergency water. GE broke up the research into three phases each lasting several months. At the end of each phase, progress would be evaluated and a decision made about proceeding to the next. This proposal was sent back to PG & E which approved it and added additional concerns relating to the cost savings, AEC final approval, ease of construction, and plausibility of the design scheme itself. Bechtel, which had already been informed about the pressure suppression containment, critiqued the proposed GE designs that were included in their R&D proposal. Bechtel contended that the designs did not consider refueling nor the placement of control rods. Bechtel also felt that GE's designs would be difficult to construct and might be vulnerable to overturning in the event of an earthquake. PG & E had developed their own designs and Bechtel critiqued theirs as well, noting that their design used a great deal of concrete and would be difficult to refuel. Bechtel also provided a cost estimate much higher than PG & E had originally predicted and recommended their own design. After the various rounds of meetings between all the participants, all participants agreed to adopt Bechtel's concept and proceed with Phase 1 of GE's proposed R&D scheme. At this stage, the pressure suppression pool would be completely separate from the reactor building but connected by a series of high pressure concrete conduits for steam to flow in. The containment had an odd shape being in two connected pieces to enclose the pool and reactor.

The first stage of GE's research included a basic demonstration of the idea with both experiments and simulations. Steam was injected into a large pool of water through a box with many holes. The holes broke up the steam and encouraged bubble formation. At some point, the engineers placed a 1.5 inch diameter pipe in the pool and injected a steam jet directly into the pool. The large diameter jet and the sparge box with small holes performed equally well. The large diameter pipe was much simpler to implement and has less risk of being clogged so the sparge box with many small holes was quickly abandoned. The steam jet did not form large bubbles but was completely condensed within one pipe diameter of the pipe exit. The pipe exit must be located 10 pipe diameters below the water surface otherwise air would be sucked into the exit pipe as condensation in a pool creates a partial vacuum. The pool was noted to be well mixed at high flow rates but would stratify at low flow rates. In parallel to experimental research, a simple transient analysis was programmed to predict the peak pressure rise in containment. Blast effects were also conservatively estimated. Upon review of this research, PG & E decided to proceed with Phase 2. Phase 2 began with an estimation of the maximum credible accident the system should be designed against. Brittle failure of the vessel was ruled out because the reactor operated above the critical failure temperature. Ultimately, a break in the largest pipe leaving the reactor pressure vessel was deemed to be the design basis accident (to use a modern parlance). GE constructed a transient test facility at their plant in San Jose CA. The reactor design only existed in the most general form so the engineers responsible for the design of the transient test facility had to guess at its design. The transient facility data matched the simulation results and both showed that the pool could heat up to 120 °F without any safety consequence.

GE generated a report about the research conducted during Phase 2 and sent it to PG & E and Bechtel. With the dissemination of the test data from Phase 2, engineers at GE and PG & E began the design of the containment in Phase 3. Although GE was responsible for the final design, managers at PG & E liked to anticipate the needs of the contractors. While there had been interest within PG & E in building the entire reactor below ground, some engineers within the company expressed problems with the idea. While earth is a useful structural material, building the entire structure underground increases the cost of construction with a minor increase in safety obviating any cost savings by the use of the pressure suppression concept. The final containment design was proposed by Bechtel, which generated nine design concepts and sent them to PG & E for approval. PG & E ultimately decided on a reactor system using pressure suppression and a concentric pool around the reactor. The operating floor would be at grade level meaning the reactor is below ground and the building is above ground. The reactor building was of a conventional design, was 1 foot thick, and was gas tight. This design had the added benefit of being quickly and reasonably cheaply reconfigurable for a conventional containment in the event the AEC did not approve the pressure suppression concept. GE also approved of the design concept and all three companies approached the AEC for final approval of the concept. The AEC did not feel that the Phase 2 experiments were comprehensive enough to ensure the safe operation of the pressure suppression concept. PG & E conducted a 1/48 scale test to demonstrate the steady state and transient behavior of the proposed reactor. A 1/48 scale reactor vessel, 1/48 scale dry well, one full size vent pipe, and a full size segment of the suppression chamber and pool were built and subjected to accident conditions. Steam was generated at the design pressure of the vessel (1250 psig) and injected into the dry well. Peak pressure in the suppression chamber was well predicted, but the peak pressure in the dry well was significantly less than predicted. This was due to very conservative assumptions about the orifice coefficient for a steam water mixture. The proposed design had quite large tolerances compared to its design specifications. Pool temperatures 60 °F higher than design specifications would not affect the peak pressures in the system. The break flow area could be twice that of the design accident and the vent tubes could accommodate mass flow rates three times their design specifications without exceeding the design pressure. After these test, the AEC approved the pressure suppression concept for the Humboldt Bay reactor which was used on all other GE BWRs. The Mark I BWR containment with the familiar toroidal shaped pressure was first proposed by a draftsmen after the Humboldt reactor was developed. As a result, the Humboldt Bay reactor used a concentric pool but BWRs built afterwards used the doughnut and upside down light bulb containment. Later versions of the BWR changed the containment to resemble something closer to the Humboldt Bay reactor.

The original concept of a pressure suppression containment proved successful aside from use of jet pipes instead of sparger boxes. The containment configuration fluctuated wildly in the design process. The initial designs submerged the vessel in the pool and did not have a conventional reactor building. With the input of Bechtel, the design changed to a more conventional reactor building with jet pipes leading to a pool separated from the foundations of the reactor building. The next iteration located the suppression pool inside the reactor building in a concentric pool around the vessel, while the final Humboldt Bay reactor configuration located the suppression pool to a concentric chamber below the vessel. The shaped of the suppression pool was changed to a torus for later BWR designs, then modified to a different shaped for later generations of BWR. Such dramatic changes in design are perhaps reasonable given a first of a kind design. These changes did not hamper the construction process so did not result in a substantial loss of income. The initial concept, the vessel in the pool, was discarded because it would have complicated the placement of control rods, refueling, and operations. The separate pool away from the reactor building would have caused differential settling between the pool and the reactor although this configuration offered the most protection against projectiles. The design constraints for the containment can be stated as follows: regulatory minimum peak and steady state pressures resulting from the design basis accident in the containment; easy refueling; easy access to the reactor; easy construction; protection of the reactor in case of an accident while refueling; no differential settling of the foundations; adequate missile protection; adequate shielding. These constraints were not stated from the outset despite experience with nuclear systems on the part of all concerned. Bechtel was the source for some of the essential constraints befitting their experience with reactor designs both commercial and naval.

The design of the containment could be a single level of abstraction where the aforementioned constraints and the scientific data would be considered together. The engineers in some ways adopted this viewpoint, drafting full preliminary sketches of the containment even with partial information. These sketches furthered the development and discussion although they could be considered levels of abstraction. The back and forth between the various participants is essential in the design process and could have not been approved. The tight coupling between the constraints means that containment design should occur at a single level of abstraction. The three R&D phases can be understood as levels of abstraction. The first level established the basic scientific principles. Using the principles, a preliminary containment design was developed and a test facility built to study transient behavior. Phase three finalized the design concept and proposed a design to the AEC. As an addendum to phase three, a 1/8 scale facility was built to verify correct system behavior.

A.2 Safety Heuristics

The recommendations of the ASME Presidential Task Force listed in “Forging a New Nuclear Safety Construct” are incorporated as heuristics. Firstly, beyond design basis accidents must be considered. Secondly, PRA of all accident scenarios is recommended but must consider the difficulty of predicting the likelihood of severe natural phenomena. The consequences of a nuclear disaster can be extreme. Whether or not Japan should have shut down all operating nuclear reactors after the Fukushima NPP accident is beside the point; they have done so and nuclear engineers must take note. The societal consequences of a nuclear disaster can be far greater than originally foreseen. Traditional estimates for nuclear safety focus on latent cancer risk to the population. While this is good, nuclear accidents do far more damage to society beyond cancer deaths. In addition, an event which was deemed beyond the design basis has occurred. The total number of reactor years of operation is ~16,000 at time of writing. Three loss of offsite power accidents with a common initiating event occurred with a frequency of 2E-4 per reactor year (3 events in 16000 years), a likelihood far in excess of what was expected.

Thirdly, the reliability of nuclear power as a non-greenhouse gas emitting technology appears to be its strongest attribute. For this reason, any plant design must have a high capacity factor. Design criteria are extensive and currently being rewritten to accommodate advanced reactors, especially non-LWRs. When written, such criteria should be used to guide the design process (fourth heuristic). Criteria from other nations or the IAEA can be used. A common theme among the various design criteria is an emphasis on loss of offsite power (fifth heuristic). Changes to the design criteria can be quite extensive. The traditional five layers of defense in depth can be satisfied in different ways. The SFR fulfills them in the normal order (fuel, cladding, coolant, vessel, containment) while the VHTR fulfills them with fuel, fission product barrier, graphite matrix, coolant, and vessel. The reactor building does not protect against fission product release. The regulatory framework differs from country to country. However, the safety design analysis would appear to be the same for all countries and systems. Sixthly, safety analysis must begin at the conceptual stage. Ideally it would use the same code packages as licensing, but simplifications are allowed. It must be conservative and should analyze the most limiting cases.

Forging a New Nuclear Safety Construct, ASME 2012.

Nuclear safety must include the enormous societal costs of nuclear accidents in addition to traditional metrics of societal risk i.e. latent cancer risk. Extremely low probability events previously ignored or only given cursory examination must be analyzed. This would entail a complete loss of offsite power as in the Fukushima accident. It is noted that an accident previously thought of as beyond the design basis happened in Fukushima Dai-Ichi. The ability to estimate the probability of such an occurrence is called into question. This underlies the focus on beyond design basis accidents and their inclusion into considered accident scenarios. The traditional tools and methods associated with nuclear power are not challenged. It is suggested that the metric used to judge the safety of nuclear energy in the past (core damage frequency of less than 1E-5 per reactor year) is insufficient in light of the extremely adverse effects of the accident on the Japanese people and the inability of the Japanese authorities to predict the tsunami. Deficiencies in human performance and accident management are noted in the Fukushima accident. The document also traces the history of the ASME boiler code and safety analyses of nuclear reactors. Emphasis is placed on current US regulations and what needs to be altered. Regulations concerning the Emergency Planning Zone were based on WASH-1400 and are noted as being unduly conservative; it is suggested that they be updated. The scientific problems associated with the nonlinear threshold of radiation effects are noted. The ASME task force foresees a time when this model is no longer used to estimate radiation risk but does not recommend that a different model be used at the present time. Better crisis communication and community outreach are recommended. While a new safety construct is proposed, it is possible that the US response to the 9/11 attacks is sufficient to meet it. Responsibility for plant safety rests on the owners and operators; the limitations of regulatory agencies are noted [The ASME, 2012].

Safety of Nuclear Power Plants: Design Specific Safety Requirements No. SSR-2/1 (Rev. 1) IAEA 2016.

The requirements of nuclear safety in the design of a nuclear system are summarized in this document. Other documents in this set pertain to the overall safety philosophy, nuclear security, safety of fuel cycle plants, etc. This document is a comprehensive overview of the requirements of nuclear safety touching on all major aspects at multiple levels of the design space. The requirements begin with the most general and gradually becomes more specific. Although this document is intended for LWRs only, it can be modified to other systems. The requirements do not differ substantially from those found in 10 CFR 50 Appendix A except for an emphasis placed on loss of offsite power accidents. Several requirements are given for the emergency power supply [Safety, 2016].

10 CFR 50 Appendix A

General Design Criteria are provided from which the designers and operators of a proposed nuclear facility must develop principal design criteria in their application for a construction permit or operating license to the USNRC. Simply stated, the general design criteria provide a minimum basis for safety considerations for commercial nuclear power plants. As acknowledged within the document, they are based on experience with LWRs and have limited applicability beyond them. For that reason, the USNRC and USDOE are writing modifications to the general design criteria for non-LWR systems. The general design criteria are grouped into six classes. There are five overall requirements in Appendix A: quality assurance of all components of the systems, the establishment of appropriate design bases for natural phenomena, assuring adequate fire protection, the establishment of appropriate design bases for internal accidents, and the prohibition of multiple nuclear reactors sharing safety systems. The defense in depth strategy relating to multiple fission product barriers is explicitly stated in the document. Redundancy is called for in numerous criteria, especially in the emergency core cooling, residual heat removal, and reactivity control criteria [10 CFR 50 Appendix A].

Advanced Reactor Design Criteria (ARDC) Development Process, USDOE 2015. Presentation at public meeting January 21, 2015.

In 2012, both the USNRC and USDOE released studies on the prospects and needs of advanced reactors. Both noted the need for a regulatory framework for advanced reactors. Towards that end, the USNRC and USDOE agreed to work together in the development of the new design criteria specific to advanced reactors. The DOE would generate a report and send it to the NRC, which would then use that report to generate new NRC guidelines. The DOE study began with a literature review of previous AEC and NRC safety analyses, national laboratory reports, ANS guidelines, etc. The literature review focused on the SFR, FHR, and HTGR. The DOE requested insight from various companies involved in advanced reactors and UC-Berkeley. The general design criteria in 10 CFR 50 Appendix A were classified into four categories: those that are generic and applicable to advanced reactors, those that need minor modification to be applicable to advanced reactors, those that need to be rewritten to be applicable to advanced reactors, and those that are not applicable. The study also specified new design criteria for advanced reactors based on stakeholder input. ANS 54.1 is specifically mentioned as being valuable in the creation of new design criteria. The remainder of the presentation discusses the meetings, methods, input, considerations, and example criteria [Advanced, 2015].

Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors, USDOE 2014.

This report gives recommended design criteria for advanced reactors stemming from a joint USDOE and USNRC project. The USNRC uses the recommendations as a starting point for the development of design criteria which will be included in the regulations. Different sets of design criteria were developed. The first set is general to all advanced reactor types, while the other two sets are relevant to the SFR and HTGR types, respectively. The SFR and HTGR design criteria were developed first, and the general design criteria where developed from them. The SFR and HTGR were chosen because of the diversity in their design basis accidents, how the design basis accidents were mitigated, and the volume of literature available. Input from the other four types was included as the general advanced reactor design criteria were developed. As the proposed design criteria are based off of those in 10 CFR 50 Appendix A, they follow the same numbering and organization as the original design criteria. Criteria relating to redundancy of electrical equipment were altered as advanced reactors make use of passive systems for decay heat removal. The same heat removal system is used during shutdown and in emergencies in many types of advanced reactor, necessitating a change to the criteria. Criteria relating to containment were also changed as the HTGR does not rely on a containment as the final fission product barrier. The need for containment isolation is also reduced, as it is possible to design a system where such systems could compromise the severe accident response.

The SFR required five additional criteria. The intermediate loop must be chemically compatible with the primary loop and radioactive material must not leak into the intermediate. Sodium purity must be maintained. Sodium must not be allowed to freeze, which is possible in extended shutdown and in small sampling lines. Sodium-air chemical reactions must be mitigated by leak detection and through dedicated safety systems. Sodium-water reactions are more energetic than air interactions, and must be mitigated by adequate design, leak detection, and dedicated safety systems. In contrast to LWRs and SFRs, the HTGR safety regulations do not reference fuel limits. A holistic requirement, the core radioactive release design limit is used instead. This defines an acceptable offsite and onsite release that the multiple fission product barriers act in concert to achieve. Three new criteria were added. The first mandates that the design of the reactor vessel and system ensures the timely insertion of neutron absorbers and the integrity of the passive safety system. The second ensures that the reactor building maintains the passive heat removal systems and allow for the release of pressure in case of a break in the primary coolant loop. Primary coolant is not needed for severe accident mitigation not is a reactor containment. Thirdly, the reactor building must be designed to be periodically inspected and surveilled [Guidance, 2014].

NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness, USNRC 2016.

While the NRC states that it is capable of licensing a non-LWR with current regulations and licensing frameworks, it acknowledges that significant inefficiencies would result as the existing LWR based regulatory framework would have to be modified during the licensing process. This would cause the license to take longer and be more expensive than it otherwise would be. Towards that end, the NRC is implementing a scheme to efficiently handle non-LWRs. Part of this project is to develop or modify design criteria to handle advanced reactors. The NRC recognizes that it needs additional technical knowledge, skills, and tools to efficiently handle non-LWRs. The NRC also recognizes that it needs to clearly communicate its requirements and expectation to the potential licensee. The NRC will also need to review how it allocates its resources when licensing a non-LWR. The NRC divides its strategies into near-term (0-5 years), mid-term (5-10 years), and long-term (>10 years) sections. In the near-term, the NRC will focus on its knowledge base, skills, codes/tools, and standards. It will devise a communication scheme for stakeholders and create a flexible regulatory review process for non-LWRs. In the mid-term, it will continue and finalize the activities of the near-term. The NRC may decide to create a new non-LWR regulatory framework; this would be implemented after information is gathered and analyzed in the near-term. This new framework would be implemented in the long-term if needed [NRC, 2016].

An Overview of Differences in Nuclear Safety Regulatory Approaches and Requirements Between United States and Other Countries, USNRC 2004.

The regulatory agencies of other countries resemble those of the US due to the early leadership of the US in the development and exporting of nuclear technology. As this document was written in 2004, some of the observations in the document may no longer be relevant. Towards that end, the findings must be judged according to whether or not they might still be pertinent. Perhaps the greatest difference between the US and other countries is the requirement of periodic safety reviews. Every country listed in the document except Canada and the US had safety reviews every 10 years. Canada has license renewals every 2-5 years which accomplish a similar purpose. The depth and breadth of these safety reviews is different between the counties. Japan's are of a limited scope mainly concerned with plant aging while Switzerland requires comprehensive safety reviews with an emphasis on the state-of-the-art in science and technology. Most countries in the world do not have fixed term licenses, instead relying on the periodic safety reviews to ensure continuing safety in the face of an aging plant. Canada has 2-5 year license extensions, Spain has 5-10 year extensions, Finland has 10-20, and Mexico has 30 years license extensions. The others are technically infinite with approval of the safety reviews. Much discussion pertains to the use of PRA in safety analyses. It is mentioned that PRA is incorporated in the US more so than in other countries, but given the amount of time since writing this may no longer be true. Similarly, severe accident responses and modeling techniques are discussed at length but are not mentioned here because of the time elapsed since publication [Nourbakhsh, 2004].

A comparison of US and Japanese regulatory requirements in effect at the time of the Fukushima accident, USNRC 2013.

The Japanese nuclear regulatory system has been greatly modified after the Fukushima accident, so the critiques mentioned in this document may no longer be applicable. This fact is mentioned within the document. That being said, the Japanese regulatory agencies did not consider natural phenomena when devising severe accident management techniques. Delayed containment venting was also practiced in Japan in contrast to the US. This technique does not vent containment until it reaches the design pressure. Hydrogen leaked out of the primary containment, allowing hydrogen to build up in the reactor building where it exploded. After 9/11, the US adopted Extensive Damage Mitigation Guidelines, which improved the severe accident mitigation strategies in the US. These had not been adopted in Japan. The Japanese regulators did not consider beyond design basis accidents such as station blackouts, anticipated transient without SCRAM, nor terrorist attacks. No regulatory guidelines existed for tsunamis and design basis floods. Japanese regulators has great faith in the safety of their plants, believing that severe accidents were so unlikely as to be unforeseeable. The document cautions against any feelings of superiority on the part of the NRC, as it states that US regulations may not have been sufficient to prevent the accident if they been in place. It is mentioned that the NRC did issue new regulations in light of the Fukushima accident [A comparison, 2013].

A Comparison of International Regulatory Organizations and Licensing Procedures for New Nuclear Power Plant, EPRG University of Cambridge 2007.

Nuclear power plant safety regulation and licensing are examined in Canada, US, UK, France, Germany, Switzerland, and Japan. All steps of the licensing process are discussed including site selection, design approval, construction oversight, etc. The conclusions are briefly summarized here. In contrast to the IAEA, the authors propose that final decisions made by a nuclear regulatory agency be subject to elected governments to increase democratic feedback in the process. It is believed this will ameliorate the concerns of environmentalists. They recommend reducing the number of laws which nuclear operators must comply with. A single nuclear regulator for the EU is proposed. It is also recommended that the licensing process be split up into three different steps with licenses for each: design, siting, and construction/commissioning/operation. Time limited licenses with a renewal period of 10 years are recommended, partly to ensure public participation. While recommendations about the regulatory framework are not strictly relevant to the dissertation, the emphasis on public participation suggests a concern about severe accidents which parallels commentary from other literature. It is important to note that this was published before Fukushima [Bredimas, 2007].

Comparison of Canadian NPP Design Requirements with those of Foreign Regulators, CNSC 2011.

The Canadian Nuclear Safety Commission compares the regulations in RD-337 and RD-310, Design of New Nuclear Power Plants and Safety Analysis for Nuclear Power Plants, to similar regulations in the US, Finland, UK, France, and Western European Nuclear Regulator's Association to identify areas where the Canadian regulations need further elaboration. It should be noted that the pertinent Canadian regulations only concern new LWRs, not non-LWR advanced reactors. This review will focus on the comparison to US regulations and the overall conclusions of the investigation. Many regulatory differences are noted; only those of significance are mentioned. Commonly, the same overall regulatory objective is meet with different regulatory measures, causing no substantial discrepancy. The dose limit in the US is an annual dose of 0.05 mSv per reactor during normal operations and expected occurrences, in contrast to 0.5 mSv used for anticipated occurrences in Canadian plants. The document then states that the 25 rem (250 mSv) dose limit for 2 hours at the site boundary during a postulated fission product release is used when examining severe accidents, not the 0.05 mSv limit. The Canadians allege that the NRC is not completely following regulations. Without reading the NRC licensing applications, it can be difficult to judge whether the NRC is incorrectly using the 25 rem rule. It should be noted that the 0.05 mSv limit isn't relevant to severe accidents, and the 25 rem rule was used to establish the exclusion zone in the event of a large release of radioactivity, almost certainly in the event of a severe accident. On this basis alone, it would appear that the NRC is correctly using the regulations by forcing licensees to demonstrate that severe accident release does not jeopardize the underlying rational behind the exclusion zone and that the Canadian regulations for dose limits during anticipated occurrences are higher than the US regulations. The core damage frequencies for the US reactors at 1E-4 per reactor-year whereas the Canadian limit is 1E-5 per reactor year. However, the concept of large early release is used in the US which has no direct analog in Canada. The large early release frequency in the US is less than 1E-6 per reactor year. The Canadian regulations do include a limit on the frequency of large releases of ^{131}I, as well as a requirement that "containments allow sufficient time for implementation of off-site emergency barriers." It is also noted that the Canadian regulations do not explicitly mention station blackouts as a potential severe accident. 100% of the water and metal are assumed to chemically react when designing the hydrogen mitigation systems in US plants.

The Canadians point out that their regulations are more technology neutral than others which would explain the less specific requirements for system behavior. The document lists the main safety features of new reactors as: a focus on severe accidents especially in regard to the design basis, more passive systems, increased reliability, less reliance on operator actions, minimization of waste, ease of decommissioning, and reduction of worker doses. The Canadians state that the regulatory practices of other countries come close to mandating certain design choices, which could favor certain design choices over others. The Canadians did not examine how the other regulatory agencies analyzed license applications or applied their respective criteria, a fact noted as severely limiting any conclusions. That being said, the different dose limits are given as a potential discrepancy. Severe accident behavior of new nuclear designs is commonly analyzed with respect to offsite dose limits in PRA. In the judgment of the author, the advanced reactor designer must have either design specific deterministic criteria or design general PRA criteria with plenty of details concerning the proper use of PRA [Comparison, 2011].

The French Connection: Comparing French and American Civilian Nuclear Energy Programs, Stanford Journal of International Relations 2010.

The authors of this document, released before Fukushima and the drop in natural gas and petroleum prices, propose that the US adopt some of the nuclear policies of France.

The French regulatory system is regarded as simpler with no input for public opinion. This allows projects to move faster, reducing the cost of nuclear power. There is only one utility, one reactor supplier, and one turbine-generator supplier in France which also speeds up regulatory action. Since the French nuclear system is supported by the government, it can actively promote and force the growth of nuclear power. The US government does not have a similar role. Mentions of nuclear safety in the pre-Fukushima era must be recalibrated, as well as mentions of cost effectiveness against currently lower natural gas, petroleum, wind, and solar prices. The incident at Davis-Besse in 2002 is mentioned as a critique of the NRC, but the flood at the Blayais nuclear power plant is not mentioned. The authors recommend that the US government should begin promoting nuclear power to the public and subsidizing it including some form of carbon cap-and-trade. The changes to the NRC licensing requirement are approved of, but the authors suggest that the NRC should designate just four reactor designs as templates to speed up regulatory oversite. Severe accident behavior, assumed safe before March 2011, must be given greater emphasis than suggested in this document. Limitations on the number of potential reactor designs would not be possible as it would unnecessarily restrict nuclear innovation. Given the proposed NRC changes mentioned elsewhere in this dissertation, artificial restrictions on nuclear concepts would appear unnecessary [Sastry, 2010].

NUCLEAR SAFETY: Countries' Regulatory Bodies Have Made Changes in Response to the Fukushima Daiichi Accident, USGAO 2014.

The US Government Accountability Office reviewed the nuclear regulatory bodies of 16 nations (including the US) to examine changes made by regulatory bodies since the Fukushima accident and study the extent to which automated systems might relay data about plant conditions during an accident. The USGAO ultimately recommended that the US enhance its system and that the IAEA play a larger role in advising the NRC. Stress tests with a special emphasis on flooding were performed in many nations after the accident revealing deficiencies in existing systems. Sweden is considering changing its exclusion zone policy. Regulators are reexamining their accident response strategies when multiple reactors at the same site suffer severe core damage. Regulators are also considering more beyond design basis accidents. Regulators have required larger amounts and more diverse emergency equipment to be stored on site. Hydrogen mitigation strategies were examined in the majority of countries; some countries made regulatory changes. Filtered venting, necessary to minimize radioactive releases to the environment while reducing containment pressure, was also examined by the regulatory agencies. Japan restructured its regulatory agencies after the USNRC. The Emergency Response Data System (ERDS) was developed in response to Three Mile Island II and collects plant data during accidents. Officials at the NRC regional offices can then take the appropriate accident mitigation strategy. When the document was written, the system could not operate during a loss of offsite power accident. A minority of other countries had such a system, as some of the countries without such a system were considering one [Nuclear, 2014].

Deterministic Safety Analysis for Nuclear Power Plant Specific Safety Guide No. SSG-2, IAEA 2009.

Deterministic safety analyses are used in the nuclear industry to certify regulatory safety requirements. Analyses fall into two general classes, conservative and best estimate. Best estimate with uncertainty analysis is the preferred method, wherein the most realistic prediction of the plant's behavior is made with a thorough study of the code's uncertainty, uncertainty of the inputs, and uncertainty of the availability of the plant's systems. The single failure criterion is often used in both conservative and best estimate analysis. Conservative analyses were conducted in the past due in part to a deficiency of thermal hydraulic behavior and the limitations of computers. Conservative analyses often make erroneous predictions and can lead operators to make poor decisions during an actual accident. If plant response is close to the safety limits, then conservative analysis can predict breaching of safety limits while in fact the plant would be safe. Conservative analyses for a few bounding cases are recommended for safety analysis during design. These will be supplemented with best estimate analyses for the development of emergency procedures, safety analysis reports, licensing requirements, plant modifications, etc. [Deterministic, 2009].

Accident Analysis for Nuclear Power Plants Safety Report Series No. 23, IAEA 2002.

While all aspects of accident analysis are covered, only those relating to design are summarized here. The same computer codes used in licensing are recommended for safety analysis at the design stage. Conservative assumptions about plant response and initial conditions are used. The plant will not be fully characterized, so realistic estimates for plant behavior are used. Later, those estimates become the requirements of the plant systems. Significant safety margins should be implemented early in the design as safety margins tend to decrease as the design progresses. Constraints imposed by cost, research and development results, operating experience, and model refinement can erode safety margins. The most damaging incidents are analyzed at the design level [Accident, 2002].

Nuclear Power in the UK, National Audit Office 2016.

The government of the UK foresees a tremendous building program of power plants as coal plants are phased out in favor of low greenhouse gas emitting generating stations. Electricity consumption in the UK is expected to increase due to population growth, economic growth, and the use of electricity for transport and public heating. The levelized cost of wind and solar is greater than or comparable to that of nuclear currently, but is predicted to be lower than the cost of nuclear power by 2025. Despite this, the intermittent nature of those energy sources is considered a liability and nuclear plants are desired to combat this issue. The UK government is encouraging private investment in nuclear power by ensuring a set price for electricity. The intermittent nature of wind and solar provide a niche nuclear power must fill. Any new nuclear design must ensure the reliability of the plant [Comptroller, 2016].

A.3 Sodium fires in SFRs

Firstly, sodium fires in sodium fast reactors are likely to occur at least once in the plant's lifetime and should be regarded as an anticipated operational occurrence. Secondly, sodium leaks are a threat to the operations of the plant and to the personnel, but do not threaten primary system integrity. Thirdly, literature recommends various steam generator configurations but the absence of a problem free system in historical fast reactors prevents any guarantees about their performance. It is noted that EBR-II did not have any significant sodium leaks. It is difficult to derive any meaningful conclusion from this fact as other small reactors did not have any significant sodium leaks while all large reactors did have extensive leaks. The steam generator is particularly prone to fires. The large difference in pressure on either side of the tubes, the corrosive nature of water at elevated temperatures, and the chemical reactivity of hot sodium and water exacerbate the preexisting problems with steam generators. Pinhole leaks can quickly turn into tube ruptures, the energy released by the reaction causing other tubes to fail. For this reason, steam generators must be protected by methods proposed in literature. Leaks in the primary side are mitigated by double walled piping, seismic proofing the vessel, and a guard vessel. Intermediate sodium leaks are more common. Standard techniques including steel sheeting around concrete, leak detection, fire suppression, dividing up the rooms containing the piping, pressure relief valves/seals, etc. are found in literature. It should be noted that sodium leaks do not pose a threat to fuel integrity, although a primary sodium leak could cause a pressure rise in the containment. Aerosols could damage equipment in the primary containment if the secondary sodium were to leak, assuming the containment uses air. It would be prudent if multiple independent steam generators were used, continuing to operate if one of them were to suffer a fire.

Fast Breeder Reactor Programs: History and Status. IFPM 2010.

This document by the International Panel on Fissile Materials was released in 2010 and summarizes the various fast breeder reactor programs in Russia/USSR, Japan, India, France, UK, and USA. The document is highly critical of fast breeder reactor programs as expensive, unsafe, and proliferating. Its summaries of the fast reactor programs in each country place special emphasis on sodium fires. The difficulties with Superphenix, Monju, Dounreay Fast Reactor, BN-350, and BN-600 are listed in detail. No sodium leaks or fires are listed for USA reactors, although the partial fuel meltdown in Fermi-1 is mentioned. It is interesting to note that FFTF or EBR-II did not experience large sodium leaks leading to significant shut down times. This systems were actually noted for their reliability at the time. While the document disparages sodium fast reactors, the lack of sodium fires in USA reactors is an interesting fact [Cochran, 2010].

Metal Fire Implications for Advanced Reactor Parts 1 and 2. SNL 2007 and 2008.

These documents summarize an investigation into sodium fires in sodium fast reactors. The investigation was in two parts, a literature review and a PIRT study. The literature review focused on incidents at Monju, BN-600, Almeria Solar Plant, and ILONA. Each accident had a different cause and fire propagation scenario. These accidents are given as situations which the plant must be designed to overcome. Corrosion is the root cause behind leaks in water cooled reactors, whereas corrosion in sodium cooled reactors is negligible if the sodium is purified. This has been demonstrated in a number of reactors. Disastrous sodium leaks from the reactor vessel are considered rare because of the lower internal pressure, extensive seismic isolation, and use of a guard vessel. Smoke from sodium fires is capable of entering the control room and preventing operators from administering the accident. The 1975 Browns Ferry fire severed all normal core cooling functions, which is a worst case scenario for a sodium fire in the secondary side. Sodium leaks and fires in steam generators can remove the ultimate heat sink in some designs. Design and manufacturing defects are regarded as containing the greatest risk to sodium leaks. Pipes, welds, and steam generator tubes are the most likely to fail. Sodium leaks can also be caused by human error in following procedures and poor procedures themselves. An extensive literature review is presented in part two concerning analytical and experimental studies of sodium leaks and fires. The PRIT study focused on the Advanced Burner Test Reactor. Two fires where studied; a leak in the hot leg of the intermediate loop causing a pool fire and a leak in the cold leg of the intermediate loop causing a spray fire. Spray and aerosol dynamics were highlighted as needing more research, as is the role the sodium oxide solid precipitate plays. Hydrogen production in concrete was also highlighted, as was the radiation loss from the sodium crust. Most of the PRIT document gives the curriculum vita's of the individuals involved in the PIRT [Olivier, 2007] and [Olivier, 2008].

Approaches to Resolve Safety Issues Related to Sodium as a Fast Reactor Coolant, Second Joint GIF-IAEA/INPRO Workshop 2011. I. Pakhamov.

This presentation posits a crack in piping which causes a small DBA or a large crack which causes a BDBA where most of the sodium leaks out. The crack is assumed to propagate slowly before a catastrophic break. A series of electroheaters surround the piping which will short circuit when sodium leaks. The pipe is also encased in a thermal insulation surrounded by another steel casing. Smoke detection systems are employed. Radioactivity is also measured in the air. The sodium systems are isolated into smaller rooms that are lined with steel, thus preventing the sodium from interacting with the concrete. Each room is sealed from the others. Exhaust systems remove air from the rooms if a fire occurs. Steam generator tube ruptures are dealt with in a similar manner. The steam generator is isolated when a leak occurs. Numerous safety systems are employed to detect or mitigate the consequences of a sodium leak. Chemistry control systems that control the oxygen and hydrogen concentrations in the sodium are used. When a leak occurs, the sodium is quickly drained out of the steam generator and an inert gas is used to fill the steam generator [Pakhamov, 2011].

Approaches to Resolve Safety Issues Related to Sodium as a Fast Reactor Coolant, Second Joint GIF-IAEA/INPRO Workshop 2011. S. Kubo.

This presentation was given during the same workshop as the previous review in this literature review. The author emphasize double walls around the piping and in the steam generator. Systems for detecting leaks are mentioned but not elaborated on. Steel lined concrete was also emphasized. A large scale leak stemming from a common fault in a single or multiple lines was regarded as the worst case scenario, not smaller leaks as in the other presentation [Kubo, 2011].

Design Features and Operating Experience of Experimental Fast Reactors. IAEA NP-T-1.9 2013.

Liquid metal properties are reviewed including thermo-hydraulic properties, neutronics properties, chemical compatibility with air and water, corrosion behavior, and induced radioactivity. The history of liquid metal technology is reviewed including various thermodynamic cycles. The document focuses on Clementine, EBR-I, LAMPRE, BR-5, Dounreay Fast Reactor, EBR-II, Enrico Fermi Fast Breeder Reactor, and Rapsodie. FFTF is not mentioned. The basic operating features and notable events in their lifetimes are recorded. Mercury, lithium, and NaK coolants are not favored over sodium. Lead or Pb-Bi suffer ^{210}Po contamination and corrode steel if the oxygen content is too low. Lead forms PbO is the presence of water, which can block coolant channels in the reactor in the event of a break in steam generator tubes. Sodium and Pb/Pb-Bi purification techniques are discussed [Design, 2013].

Liquid Metal Cooled Reactors: Experience in Design and Operation. IAEA TECDOC 1569, 2007.

This 272 page document summarizes experience with Prototype Fast Reactor, Phenix, Superphenix, BN-350, BN-600, Dounreay Fast Reactor and the Pb-Bi cooled ship reactors. The document discusses Rapsodie, EBR-II, and FFTF in relation to their safety tests. Rapsodie and EBR-II underwent a loss of flow without SCRAM accident, while EBR-II also underwent a loss of heat sink without SCRAM accident. FFTF was subjected to similar tests but at 50% power. Positive sodium void coefficients are mitigated by radial and axial expansion. Rapsodie did not have an appreciable Doppler coefficient, one of the largest drawbacks of oxide fueled reactors. Therefore, using the Rapsodie test to demonstrate the safety of all oxide fueled SFRs is incorrect as will be demonstrated later in this work. Metallic fuels have much lower temperatures and lower Doppler Coefficients, so can withstand transients more easily. The last portion of this document discusses the safety performance of the PRISM which uses metallic fuel. The benefits of metallic fuel are stated, although not compared to oxide fuel. The last portion of this document also gives concise advice for the design of a steam generator and primary loop based on experience in the BN-600 and Superphenix. The primary sodium purification technology can be located in the vessel, along with the fresh and used fuel if B4C shielding assemblies are utilized. Titanium stabilized steels are not recommended for use in sodium piping despite their mechanical strength. A single vessel steam generator with long straight tubes in recommended as it minimizes the number of welds and complex tubing [Liquid, 2007].

Status of Fast Reactor Research and Technology Development. IAEA TECDOC 1691, 2012.

This 846 page document contains a great deal of information. Sodium leaks and fires will be discussed first. No significant sodium leaks are mentioned for EBR-I, Enrico Fermi

Fast Breeder Reactor, JOYO, or the FBTR. BR-5/10 has experienced 19 sodium leaks in its operation, although none since 1986. 7 of the leaks stemmed from sodium valve failures and 6 of the leaks stemmed from failures of the level gauges in sodium storage tanks. EBR-II did not experience any leaks in the steam generators. Their performance, both in heat transfer and operation, is attributed to the robust duplex tube design of the superheaters and evaporators. During its peak years, EBR-II's capacity factor exceeded 70% and reached 81% while being used for extensive fuel, materials, and system testing. EBR-II experienced a sodium leak and fire in an intermediate loop sampling line that was confined to the boiler room. It was caused by a faulty weld during regular maintenance and maintenance procedures were modified to ameliorate the situation. The only significant maintenance problem was the seal around the rotating shield plugs. A Sn-Bi eutectic was used as a seal and had a tendency to become contaminated and block up. This was mitigated by a thorough cleaning procedure, but could be permanently fixed in future designs by an inflatable seal. The seal prevented the cover gas in the primary circuit from mixing with the air on the reactor building. There have been no sodium leaks in the operation of BOR-60. BN-350 experienced a number of difficulties with its steam generators stemming from cold stamping of the Field's tubes.

Phenix experienced a number of sodium leakage events during its operation. In all, faults with the intermediate heat exchanger and steam generators accounted for ~41% of the total lost energy production from 1974-1990. During this period, the average load factor was 60%. Sodium leaks not including the intermediate heat exchanger or steam generator accounted for a much smaller 2.6% of the total lost energy production. Other significant losses of production included scheduled works (15.1%), refueling (11.2%), and negative reactivity events (8.1). In all, sodium leaks were the single greatest cause of loss of electricity production; 31 leaks occurred, most of them very minor. There were 5 sodium water interaction events. 4 of them were due to thermal fatigue at welds that caused erosion, piercing, or wasting of the other tubes or shell. The last event was due to a manufacturing defect. The intermediate heat exchangers have 11 events, not all of them leaks. As these leaks are not chemically exothermic, they are of lesser concern than faults in steam generators although such leaks do allow radioactivity to contaminate the intermediate loop. A leak in the intermediate loop led to sodium aerosol formation and a partial jamming of the control rod drivers. Beginning around 1999, a series of safety improvements were made to Phenix. Seismic reinforcement of the reactor was complete by 2000, with special emphasis on the steam generators. The steam generator building and associated piping was subdivided into steel rooms to prevent sodium leaks from spreading all over the building and to reduce the severity of sodium concrete interactions. The steam generators were also repaired by 2002 after cracks were found. Three sodium leaks occurred in 2003; the first in a valve bellow of the intermediate loop sodium purification circuit and the second in an electromagnetic pump of the steam generator hydrogen detection circuit. The third and minor leak occurred late in 2003, although none of these leaks posed a danger to the systems nor resulted in shutdowns of significant time. A small leak occurred in the intermediate loop in 2007 necessitating a brief shutdown.

Prototype Fast Reactor or PFR experienced a similarly large number of sodium leaks as Phenix. The plant spent most the vast majority of its time shutdown during the first ten years mostly due to leaks in the tube-tubeplate welds in the superheaters, reheaters, and evaporators. 44 leaks occurred in total, with 41 of them in the evaporators. The austenitic stainless steel was found to be highly unsuitable for the application and new 9Cr1Mo ferritic stainless steel set of tube bundles was used for the superheaters and reheaters. The tube-tubeplate interface was sleeved in those new bundles. The evaporator tube-tubeplate weld was also sleeved with 9Cr1Mo which eliminated problems in the evaporator. Prior to that, the bores of the tube-tubeplate welds had been shotpeened which was unsuccessful. Initial cracks were bypassed by simply blocking the tubes, but it became apparent that more drastic measures were needed after cracks were found on many welds. A small sodium fire happened in the intermediate loop cell after a leak from the hydrogen detection system in a superheater. The decay heat rejection loops suffered several leaks over the lifetime of the plant, but none as serious as the leaks in the steam generators. Bearing oil leaked from the primary pumps into the sodium, but this did not pose any threat although the reactor was shutdown for 6.5 months. The last significant fire in the steam generators happened in 1987 from a flow induced vibration in the austenitic tubes, not at a crack. The failure of a single tube caused 39 other tubes to fail. All of the superheaters were replaced as the design of the tubes was the same in each device. Cracks were also discovered in the outer vessels of the superheaters and reheaters, although they were repaired in situ.

KNK-II experienced a single sodium-water interaction at the beginning of life due to a faulty weld. A few minor sodium leaks occurred in the intermediate loop around cracks, but never in the main piping. 12 leaks have occurred in the steam generator units of BN-600, although no leaks have occurred since 1991. Half of these leaks occurred in the first year due to manufacturing defects. A total of 27 sodium leaks have occurred since 2010. Some causes, in order from most common to least include; valves, cracks, joints, procedural errors, manufacturing defects, holes made by personnel. It should be noted that the last leak took place in 1994. FFTF operated without significant sodium leaks, but it lacked steam generators. No significant leaks occurred in the sodium to air heat exchangers nor in the intermediate loops. In contrast to Phenix, Superphenix experienced far fewer problems sodium leaks during its operation. 3 minor leaks of no consequence and 1 major leak from the fuel storage drum. The storage drum was found to be irreparable. While few sodium leaks occurred, the overall capacity factor was 41.5%; 101 abnormal events were recorded and 100 of them were ‘safety significant.’ Roughly one third of the problems relate to design or first use and occur at the beginning of the plants life. One third of the problems are equipment failures or human error, and the other third are not plant specific. After the 1986 Almeria plant fire, the rooms containing the intermediate loop piping were subdivided and lined in stainless steel. Pressure release flaps were made in the walls of the containment building to allow hot gases to escape in the event of a fire. As is well documented in other sources, MONJU experienced a serious sodium leak and fire that was unsuccessfully covered up. An extensive safety review was conducted, but the reactor was never restarted baring massive public opposition [Status, 2012].

A.4 Reliability in Engineering Design

In this discussion, reliability in engineering is approached from two related viewpoints: from the viewpoint of the organization in charge of managing the system

(High Reliability Organization and STAMO), and from the viewpoint of the design itself (Robust Design). High reliability organizations are defined as organizations that operate in a high risk environment with few accidents. Nuclear power is regarded by researchers in this field as a high reliability organization [La Porte, 1996]. The author notes that high reliability organizations are often under financial pressure to devote fewer resources to safety especially if they have been reasonably successful in the past or if financial pressure suddenly increases. The Fukushima accident has brought renewed emphasis on nuclear safety and it is unlikely operators and designers of nuclear power plants will devote less effort to safety even in difficult financial situations. High reliability organizations have a strong concern within the organization itself and in the larger society about the performance of the system. This may be likened to the culture of safety prevalent in the nuclear field. High degrees of technical competence are noted in high reliability organizations, maintained by organizational visibility and status. Access to senior management and promotion to senior management for those maintaining high reliability are also prevalent. High operational performance is attained by extensive quality assurance programs and databases of equipment behavior and maintenance scheduling. High reliability organizations are often quite flexible in personnel scheduling or demands, have sufficient process overlap to accommodate failures, and sufficient independence between incompatible processes. While hierarchical organizational structures dominate, high reliability organizations have a greater preponderance of collegial working environments than other organizations. Roles within organizations may actually reverse depending on the situation. Certain actions and roles are scripted in accidents, functioning alongside more collaborative forms of problem solving. In time sensitive scenarios, solutions are quickly implemented without extensive review. This puts an emphasis on correct solutions. High reliability organizations also have extensive internal oversite, often rewarding error finding and reporting. Strong external regulators help enforce the reliability culture and practices within high reliability organizations. Employees in high reliability organizations are highly proactive in problem solving. Peer pressure can be intense, often resulting in employees going beyond their job roles to fix mistakes. Employees often take immense pride in their work, which is enhanced by official rewards from management for problem solving and error finding. Employees have great discretion in solving time sensitive problems, often leading to a mentality wherein employees view themselves as masters of their respective domains. Finally, the authors noted friction between operators and engineers stemming from the lack of first-hand experience in engineers. This friction can be aggravated by the other traits in the organization, namely high discretion and results focused management.

High Reliability Organization has been criticized for being too broad in its generalizations, lacking in firm definitions, and approached from a perspective other than engineering [Leveson, 2009]. It has also been noted that reliability and safety are not related in some systems and can be mutually exclusive. The overall methodology of High Reliability Organization, wherein general observations are abstracted from certain organizations and applied to others, has been criticized as sampling from too few data points, over generalizing, and applying inferences from one field to another field without any consideration of the differences between them. Towards this end, the authors in

[Leveson, 2009] recommend STAMP (Systems-Theoretic Accident Modeling and Processes) [Leveson, 2004] as the corrective to High Reliability Organization. Organizations are modeled with control theory, wherein the familiar logic of control systems is used. Stated briefly, control systems have a controller which receives input data, issues a signal to the system, and then measures the output of the system with a sensor which sends data back to the controller. This is called a closed loop control system; open loop systems are also used. The organization is modeled as a series of these controllers, systems, and sensors in a block diagram. While this approach can more clearly identify faults with organizations than High Reliability Organization, it is asking fundamentally different questions than High Reliability Organization. High Reliability Organization seeks to identify cultural/social traits within organizations that lead to better performance while STAMP seeks to identify specific technical faults within the organization. STAMP also incorporates cultural/social factors, which could account for some of the features in High Reliability Organization. However, these traits would have to be derived from observations from high reliability organizations. While insights from both fields are relevant to organizational reliability, this dissertation concerns engineering design and both fields must be reinterpreted in that light. STAMP, or control theory more generally, would appear to be far more useful in engineering design that High Reliability Organization. Nuclear safety is a highly evolved subject considered both deterministic and probabilistic risk management as mentioned elsewhere in this document. The relevance of control theory to describe nuclear safety must be made against the tried and tested methods of the wider nuclear field. That research is beyond the scope of the present endeavor. High

Reliability Organizations focuses on the internal motivations of the operators of nuclear plants. Likewise, the engineers responsible for the design of these systems must also inculcate nuclear safety culture.

The field of Robust Design comprises a wide variety of methods centered on a single core concept. First developed by Genichi Taguchi in the 1950's [Allen, 2006], Robust Design seeks to ensure greater product performance in the field by reducing the sensitivity of the product's performance to variation in its operating conditions. While originally conceived as an alternative to tighter and tighter tolerances in manufacturing, Robust Design has evolved to consider variations in object manufacturing, environment, inputs, object performance differing from expectation, and other aspects. The most robust product may not be the one which performs best under ideal conditions, but the one who performs adequately under a wide range of conditions. Three aspects of the product are studied: the input variables, noise variables, and the system which generates the output [Sleeper, 2007]. Input variables are more commonly called design variables and are modified by the engineer to attain a given purposed. Noise variables stem from the situation the product is used in. Standard manufacturing tolerances comprise variation in design variable while probability distribution functions of the environment often comprise the noise variables. The actual operations of the product can be subject to variability. More common early in the design process is discrepancy between the model of the product's behavior and its actual behavior. Type-I Robust Design only considers noise factor variation while Type-II Robust Design considers variations in the design variables themselves. Type-II can also encompass approximate design variables used early in the design process that satisfy a variety of situations before the design is finalized. Type-III Robust Design considers variability in the system's performance. Type-I is most commonly studied, followed by Type-II then Type-III. A wide variety of statistical processes have been proposed and are briefly summarized in [Allen, 2006]. Early in the design process the final noise variables, manufacturing tolerances, and system behavior are not known. Ranges of control variables can be chosen to accommodate variability based on perceived behavior, but this is system dependent. The recommended methods in [Allen, 2006] are design dependent and could be consulted for specific methods. While the various methods mentioned in this section are useful in their respective, their utility as heuristics is somewhat more limited. Sociological considerations are not relevant while Robust Design is focused on statistical methods to study various systems and scenarios. The delineation of the various kinds of uncertainties is useful to keep in mind when performing early design studies. Control Theory could be studied in more detail especially in relation to risk management.

A.5 Issues with Axiomatic Design

In this section the utility of Axiomatic Design is examined. It is shown throughout this section that the highly coupled nature of reactor design (especially neutronic and neutronic/thermal hydraulic) mitigates the utility of Axiomatic Design. However, in non-neutronic nuclear system problems Axiomatic Design could be of use. Axiomatic Design is first studied using the fast spectrum MTR example from Chapter 4. The utility of Axiomatic Design is used in each level of abstraction, before the entire design is reconceptualized in a form more conducive to Axiomatic Design. The nuclear system is examined at its broadest level with Axiomatic Design, and promise is shown in solving non-neutronic nuclear design problems. The fourth level is presented first because if contained the most variables, constraints, and objectives of the levels. After it is shown that Axiomatic Design would not be much use for the fourth level, it is shown that Axiomatic Design would be of little use in the first, second, third, and fifth levels. Axiomatic Design uses a process called zig-zagging which decomposes higher level functional requirements and design parameters into lower level aspects. Similar in form to levels of abstraction, it was decided to perform the reactor design process using Axiomatic Design without recourse to any other design methodology. Axiomatic Design is used to study nuclear systems without reference to the fast spectrum MSR. It appears that Axiomatic Design has some utility in nuclear system design outside of the reactor core. This could not explored in greater detail due to time limitations.

Axiomatic Design uses a few concepts which will be briefly restated; they are elaborated on in section 1.5. The independence axiom of Axiomatic Design seeks to minimize the relationships of functional requirements; this is achieved by picking design parameters so that at least one design parameter can be fixed. At least one other functional requirement requires only two design parameters; the one that was fixed and an undetermined design parameter. The undetermined design parameter is fixed by using the determined design parameter and associated functional requirement. The solution continues in this fashion, using determined design parameters to find the undetermined ones. The ideal design has each functional requirement satisfied by a single design parameter. The information axiom merely states that the best design has the highest probability of satisfying the functional requirements; this is analogous to Robust Design. The independence axiom could not be satisfied in any examined situation, while the information axiom is not unique to Axiomatic Design.

Functional requirements for the fourth level of abstraction of the fast spectrum MTR are shown below. They represent the objectives and constraints. The design parameters commonly used in Axiomatic Design are presented after the functional requirements. As can be seen, there are eight functional requirements and ten design parameters. This is a redundant design unless some of the design parameters can be fixed. This is impossible so the design will not satisfy the independence axiom and will be coupled. The coupled nature of the fourth level of abstraction is due to the physics of nuclear reactors, as will be shown. In Axiomatic Design functional requirements are those features of the design which must be satisfied. Design parameters are those elements which are used to achieve the features (design requirements). The functional requirements and design parameters are vectors. In this example, the functional requirements are an 8 x 1 vector while the design parameters are a 10 x 1 vector. An 8 x 10 matrix is multiplied by the 10 x 1 design parameter vector to fulfill the functional requirements. This matrix, shown below, is filled with either a 0 or an X. If a 0 is used, then the corresponding design parameter has no effect on the corresponding functional requirement. If an X is used, then the corresponding design parameters has some effect on the corresponding design parameter. The relationship between the design parameters and the functional requirements can be of any form. The matrix was derived based on physical consideration without reference to the final design. Any design would have those same relationships.

$$[FR] = \begin{bmatrix} High\ fast\ flux \\ High\ core\ lifetime \\ Low\ core\ power \\ Low\ core\ \Delta P \\ Peak\ vel. \\ \frac{H}{D} = 1 \\ Peak\ LP \\ Indep.assemblies \end{bmatrix}$$

$$[DP] = \begin{bmatrix} Fuel\ height \\ Reflector\ height \\ Reflector\ material \\ Coolant\ flow\ rate \\ Coolant\ temps. \\ Control\ rod\ material \\ Control\ rod\ design \\ Moderator\ material \\ Moderator\ design \\ Irradiation\ positions \end{bmatrix}$$

$$\begin{bmatrix} High\ fast\ flux \\ High\ lifetime \\ Low\ power \\ Low\ \Delta P \\ Low\ v \\ H/D = 1 \\ Peak\ LP \\ Indep.ass. \end{bmatrix} = \begin{bmatrix} X & X & X & X & X & 0 & 0 & X & X & X \\ X & X & X & X & X & X & X & X & X & X \\ X & X & X & X & X & 0 & 0 & X & X & 0 \\ X & X & 0 & X & X & 0 & 0 & X & X & 0 \\ 0 & 0 & 0 & X & X & 0 & 0 & X & X & 0 \\ X & X & 0 & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ X & X & X & 0 & 0 & 0 & 0 & X & X & 0 \\ 0 & 0 & 0 & 0 & 0 & 0 & X & 0 & X & X \end{bmatrix} \begin{bmatrix} Fuel\ height \\ Reflector\ height \\ Reflector\ mat. \\ Cool.flow\ rate \\ Coolant\ temps. \\ Control\ rod\ mat. \\ Control\ rod\ design \\ Mod.material \\ Mod.design \\ Irradiation\ pos. \end{bmatrix}$$

The first three design requirements (high fast flux, high lifetime, and low power) are dependent on most of the design parameters. This is because all three parameters are integral quantities, determined from the interactions of all the design parameters. While these parameter are integral, the weaker dependencies can be excluded by careful choice of design parameters. Flux is dependent on every aspect of the core design, especially the core power. While core power is not a design parameter (perhaps it should be), the coolant flow rate and coolant temperatures depend on the core power. The core power, coolant flow rates, and coolant temperatures are tightly coupled with respect to the allowable coolant and fuel temperatures. Varying any one or two of them requires changing the others. Although flux is not explicitly a function of coolant conditions, if the core design is held constant and either coolant temperatures or mass flow rate are changed, core power will change to accommodate the maximum coolant/fuel temperatures. The flux is measured in the irradiation positions, so their location is quite important. While the control systems must affect flux (from their changes in reactivity), control systems are designed so that they have a minimal effect on flux. Lifetime is dependent on the k_{eff} of the system and flux (through which power density is calculated, determining the loss of fissile material). Power density depends on core power and the core layout, so coolant temperatures and mass flow rate are also relevant design parameters. Minimizing the core power depends on all of the design parameters for the same reasons. Lifetime is strongly dependent on the design of the control systems and the number of irradiation positions. Each irradiation position added to the core removes fuel, increasing burnup (for the same core power) and reducing lifetime.

The remaining functional requirements are also dependent on all of the design parameters, but only weakly. For example, the core pressure drop is dependent on the flow rate through each assembly, which can be controlled with orifices. By orificing the assembly inlets, more flow can be directed away from the low power assemblies and towards the high power assemblies. Less orificing is needed when the power profile across the core is flatter; all of the design parameters in some way affect the power profile which would affect the pressure drop. Weaker relationships have been removed from the matrix. Fuel height, reflector height, mass flow rate, and temperatures obviously affect pressure drop, and the moderator design does affect the power profile and consequently the mass flow rate in the hottest assemblies. From this the core pressure drop is deduced. The same is true of the maximum velocity, which is not dependent on the heights. The control systems and irradiation positions would be designed to have minimal effect on the power profile or to flatten it so their effect on the core pressure drop is quite weak. The requirement that height and diameter be identical is only dependent on the core layout and height. Core layout is not explicitly a design parameter (although perhaps it should be) so the heights are the only design parameters relevant to that functional requirement. Linear power limits are tied to every decision that affects core power. In principle, every design parameter should be relevant to the linear power limits; however the irradiation positions and control systems will be designed so that they have minimal effect on the power profile. Therefore only those design parameters which greatly affect the linear power limits are relevant. Maintaining the independence of all the assemblies affects the control systems, moderator assembly design, and irradiation positions.

Axiomatic Design with hierarchical systems can be used to decompose the system from higher levels of abstraction into lower levels of abstraction. This feature of Axiomatic Design will be examined with respect to all of the levels of abstraction. The first level of abstraction concerned whether or not to design a new reactor. This isn't really a design, so Axiomatic Design would not be a suitable tool for the first level of abstraction. While it is possible to specify functional requirements, defining design parameters is difficult as the functional as there are only two options; either construct a new reactor or use an existing one. The second level of abstraction selected the fuel vector, fuel type, and coolant type. There were three fuel vectors: HEU, LEU, and Pu. There were five fuel types: oxide, zirconium, carbide, aluminum, and dispersal. There were five coolant types: sodium, light water, heavy water, lead, and helium. The functional requirements are: providing a fast flux, a near term design, and a safe concept. Four possibilities are presented. No design satisfies the independence axiom and all are redundant. In fact the least safe concept (using helium and UO_2 is unsafe) is the least coupled. The UC and helium design uses a graphite moderator and is VHTR. While the VHTR and LWR concepts are near term and safe concepts, they cannot be utilized because they do not have high fast fluxes. The fast flux is determined almost exclusively by the coolant type and moderator type. The moderator type is commonly included when selecting coolant type. Safety analyses must consider the fuel type and coolant type. The technical feasibility of the concept requires analysis of all the design parameters. The information axiom clearly separates the concepts. The first two design concepts differ only in their fuel type and have the same relationships between the functional requirements and design parameters. As mentioned in section 4.3, metallic fueled sodium reactors have significant safety advantages over oxide fueled sodium reactors. Oxide fuel is less safe but more technically feasible. Oxide fuel is manufactured in the US while metallic fuel has not been manufactured for a number of years.

$$\begin{bmatrix} Fast\ flux \\ Near\ term \\ Safety \end{bmatrix} = \begin{bmatrix} X & X & X \\ X & X & X \\ 0 & X & X \end{bmatrix} \begin{bmatrix} LEU \\ U - Zr \\ Sodium \end{bmatrix}; fast\ spectrum$$

$$\begin{bmatrix} Fast\ flux \\ Near\ term \\ Safety \end{bmatrix} = \begin{bmatrix} X & X & X \\ X & X & X \\ 0 & X & X \end{bmatrix} \begin{bmatrix} LEU \\ UO_2 \\ Sodium \end{bmatrix}; fast\ spectrum$$

$$\begin{bmatrix} Fast\ flux \\ Near\ term \\ Safety \end{bmatrix} = \begin{bmatrix} X & X & 0 \\ X & X & X \\ 0 & X & X \end{bmatrix} \begin{bmatrix} LEU \\ UO_2 \\ Water \end{bmatrix}; thermal\ spectrum$$

$$\begin{bmatrix} Fast\ flux \\ Near\ term \\ Safety \end{bmatrix} = \begin{bmatrix} X & X & X \\ X & X & X \\ 0 & X & 0 \end{bmatrix} \begin{bmatrix} LEU \\ UO_2 \\ Helium \end{bmatrix}; fast\ spectrum$$

$$\begin{bmatrix} Fast\ flux \\ Near\ term \\ Safety \end{bmatrix} = \begin{bmatrix} X & X & 0 \\ X & X & X \\ 0 & X & X \end{bmatrix} \begin{bmatrix} LEU \\ UC \\ Helium \end{bmatrix}; thermal\ spectrum$$

The third level of abstraction determined the assembly and overall core layout. Safety constraints were based on core and system level behavior in addition to fuel type as in the second level of abstraction. Such behavior is difficult to model, so it was decided to mimic the design of a preexisting design. EBR-II was chosen for its proven safety record. The core was designed using a deterministic method which minimized the number of changes to the EBR-II design. The considerations in this level are especially unsuited to Axiomatic Design, while the fifth level of abstraction contains more constraints and design variables. In the fifth level of abstraction, the core layout is assumed while the irradiation positions are located within the core. The total volume available for irradiation depends on the number of assemblies and their design. Assuring the independence of the irradiation assemblies only depends on the irradiation assembly design, while minimizing the water content within the assembly only depends on the moderator design. Flux, k_{eff}, and linear power are neutronically coupled and cannot be teased apart as mentioned in the discussion about the fourth level of abstraction.

$$\begin{bmatrix} High\ flux \\ High\ k_{eff} \\ Volume \\ Peak\ LP \\ Indep.ass. \\ Low\ water \end{bmatrix} = \begin{bmatrix} X & X & X & X & X \\ X & X & X & X & X \\ 0 & X & X & 0 & 0 \\ X & X & X & X & X \\ 0 & 0 & X & 0 & 0 \\ 0 & 0 & 0 & 0 & X \end{bmatrix} \begin{bmatrix} Irr.locations \\ Irr.number \\ Irr.layout \\ Absorber\ mat. \\ Mod.material \end{bmatrix}$$

While Axiomatic Design did not appear very useful within the framework of level of abstraction, it was decided to use Axiomatic Design by itself. The constraints and objectives from the completed design were used to frame the problem but the all were translated into forms more conducive to Axiomatic Design. Zig-zagging will be used to decompose the design parameters and functional requirements. The simplest available scheme is shown below. The functional requirements are divided between those necessary for good reactor performance (Excellent ops.) and those necessary for good materials testing characteristics (Excellent irr.). The first design parameter contains all core and system parameters while the second contains all irradiation parameters. The system is still tightly coupled at this stage. A metallic fueled SFR with moderator located on the core periphery minimizes the information content of the system for reasons listed in Chapter 4. The functional requirements for this system were expanded as shown below.

$$\begin{bmatrix} Excellent\ ops. \\ Excellent\ irr. \end{bmatrix} = \begin{bmatrix} X & X \\ X & X \end{bmatrix} \begin{bmatrix} Coolant + fuel + layout \\ Irradiation\ design + layout \end{bmatrix}$$

Irradiation characteristics can be expanded into flux and linear power requirements. It is also desired that all irradiation positions be independent of each other.

Lastly, internal volume of the irradiation positions should be maximized. The design parameters are the layout of the irradiation positions within the core, the internal design of the irradiation positions, the type of moderator, and the type of absorber within the irradiation positions. Linear power is a function of flux and will share the same dependencies. Flux and k_{eff} are tightly coupled are depend on all aspects of the core design. Flux and k_{eff} are the solutions to the neutron transport equation, flux being the eigenvector and k_{eff} being the eigenvalue. Maintaining independent assemblies and maximizing the volume follows the independence axiom with respect to irradiation external layout and irradiation internal design.

$$\begin{bmatrix} Excellent\ ops. \\ High\ flux \\ Peak\ LP \\ Indep.ass. \\ Volume \end{bmatrix} = \begin{bmatrix} X & X & 0 & X & 0 \\ X & X & X & X & X \\ X & X & X & X & X \\ 0 & 0 & X & 0 & 0 \\ 0 & X & X & 0 & 0 \end{bmatrix} \begin{bmatrix} Coolant + fuel + layout \\ Irradiation\ ext.layout \\ Irradiation\ int.layout \\ Moderator\ mat. \\ Absorber\ mat. \end{bmatrix}$$

For the sake of completeness, the operational requirements are divided and used to generate the following equation. The requirements are arranged so that they resemble the form of an Axiomatic Design that fulfills the independence axiom.

$$\begin{bmatrix} High\ fast\ flux \\ High\ lifetime \\ Low\ power \\ Safe \\ Near - term \\ High\ therm.flux \\ Mult.coolants \\ High\ Volume \\ Indep.ass. \end{bmatrix} = \begin{bmatrix} X & X & X & X & X & X & X & X \\ X & X & X & X & X & X & X & X \\ X & X & X & X & X & X & X & X \\ X & X & 0 & X & X & X & X & 0 \\ X & X & 0 & 0 & 0 & X & X & X \\ X & 0 & X & 0 & 0 & X & X & X \\ X & 0 & 0 & 0 & 0 & 0 & 0 & X \\ 0 & 0 & 0 & 0 & 0 & 0 & X & X \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & X \end{bmatrix} \begin{bmatrix} Reactor\ type \\ Fuel\ type \\ Reflector\ mat. \\ Power \\ CR\ design \\ Mod.material \\ Mod.design \\ Irradiation\ pos. \end{bmatrix}$$

While Axiomatic Design appears completely unusable with regard to neutronics aspects of core design, perhaps it will be more useful with regards to non-neutronics aspects. To examine this possibility, the following equation was created describing a common nuclear system. Each component of the nuclear reactor typically fulfills a single function.

$$\begin{bmatrix} Generate\ heat \\ Generate\ electricity \\ Safety\ in\ case\ of\ failure \end{bmatrix} = \begin{bmatrix} X & 0 & 0 \\ 0 & X & 0 \\ 0 & 0 & X \end{bmatrix} \begin{bmatrix} Reactor\ core \\ Turbine \\ RCCS + Containment \end{bmatrix}$$

The RCCS and containment are both integral to reactor safety; now they will be separated. The containment seeks to protect the reactor from external threats and serves as a last ditch protection against radioactive release to the public. The containment in Three Mile Island was critical in preventing the release of large amounts of radiation. The RCCS, or reactor core cooling system, is necessary in case normal core cooling fails. The phrase "in case of failure" refers to expected failure of the reactor core. If this phrase were not included, then the system would be more tightly coupled. The $A_{3,1}$ and $A_{4,1}$ positions would be X if the phrase "in case of failure" were to be omitted.

$$\begin{bmatrix} Generate\ heat \\ Generate\ electricity \\ Minimize\ leakage\ in\ case\ of\ failure \\ Minimize\ FP\ release\ in\ case\ of\ failure \end{bmatrix} = \begin{bmatrix} X & 0 & 0 & 0 \\ 0 & X & 0 & 0 \\ 0/X & 0 & X & 0 \\ 0/X & 0 & 0 & X \end{bmatrix} \begin{bmatrix} Reactor\ core \\ Turbine \\ Containment \\ RCCS \end{bmatrix}$$

The reactor core and turbine are different components and some additional systems must exist to transport heat between them. This results in a fifth design parameter and a fifth functional requirement, and the new equations meets the independence axiom.

$$\begin{bmatrix} Generate\ heat \\ Generate\ electricity \\ Coolant\ transport \\ Low\ leakage\ with\ failure \\ Low\ FP\ release\ with\ failure \end{bmatrix} = \begin{bmatrix} X & 0 & 0 & 0 & 0 \\ 0 & X & 0 & 0 & 0 \\ 0 & 0 & X & 0 & 0 \\ 0 & 0 & 0 & X & 0 \\ 0 & 0 & 0 & 0 & X \end{bmatrix} \begin{bmatrix} Reactor\ core \\ Turbine \\ Heat\ transport \\ Containment \\ RCCS \end{bmatrix}$$

If the proposed design is a PWR, then the Axiomatic Design layout could resemble that shown below. The reactor core generates subcooled water which exchanges heat with water at a lower pressure to generate steam. In the BWR, the core generates steam. System pressure is primarily regulated by the turbine set point valves and bypasses. The PWR equation is presented first while the BWR equation is presented second. The same functional requirements must be meet in both design but with fewer systems in the BWR. The desire to generate steam within the core necessitates some design changes. The largest changes are within the reactor system but outside the reactor core. The relocation of the steam separators/dryers to the reactor vessel and the development of a recirculation system change the design, but the overall number of design constraints is not significantly different from the PWR. The usage of the turbine bypass systems and valves for pressure regulation changes the design envelope. Such systems are included in a PWR but are of less significance. For these reasons, although at this stage in the design the BWR appears to be less promising than the PWR by the independence axiom, it does not pose any challenges greater than the PWR. While the PWR was developed first historically, the BWR was developed shortly thereafter without any hurdles.

$$\begin{bmatrix} Generate\ heat \\ Generate\ steam \\ Regulate\ pressure \\ Regulate\ flow\ rate \\ Generate\ electricity \\ Transport\ coolant \\ Low\ leakage\ w.failure \\ Low\ FP\ release\ w.failure \end{bmatrix} = \begin{bmatrix} X & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\ 0 & X & 0 & 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & X & 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & X & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & X & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 & X & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 & 0 & X & 0 \\ 0 & 0 & 0 & 0 & 0 & 0 & 0 & X \end{bmatrix} \begin{bmatrix} Reactor\ core \\ Steam\ generator \\ Pressurizer \\ Pumps \\ Turbine \\ Piping \\ Containment \\ RCCS \end{bmatrix}$$

$$\begin{bmatrix} Generate\ heat \\ Generate\ steam \\ Regulate\ pressure \\ Regulate\ flow\ rate \\ Generate\ electricity \\ Transport\ coolant \\ Low\ leakage\ w.failure \\ Low\ FP\ release\ w.failure \end{bmatrix} = \begin{bmatrix} X & 0 & 0 & 0 & 0 & 0 \\ X & 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & X & 0 & 0 & 0 \\ 0 & X & 0 & 0 & 0 & 0 \\ 0 & 0 & X & 0 & 0 & 0 \\ 0 & 0 & 0 & X & 0 & 0 \\ 0 & 0 & 0 & 0 & X & 0 \\ 0 & 0 & 0 & 0 & 0 & X \end{bmatrix} \begin{bmatrix} Reactor\ core \\ Pumps \\ Turbine \\ Piping \\ Containment \\ RCCS \end{bmatrix}$$

A.6 How to use PIRT for Engineering PIRT

As described in Section 1.1, PIRT can be used for engineering design. PIRT partitions the phenomena, systems, and time scales of interest into convenient lower level phenomena and systems. The knowledge and importance of each phenomena in each system and time scale is then qualitatively judged. The qualitative judgement is translated into a simple numerical scale and reported. Phenomena that is unknown and importance is then identified. With very few modifications this process can be adapted for nuclear system design. Common issues within nuclear engineering include the choice of coolant and choice of fuel. Objectives and constraints such as cost, feasibility, safety, etc. are provided and each concept is judged accordingly. Based on the numerical values for each category some fitness can be evaluated and the appropriate design choice identified. Table

4.6 provides a good example of the results of a PIRT although an overall fitness is not calculated. PIRT for engineering design does not need to emphasize the knowledge base of the potential designs. PIRT for engineering design does not need to partition the performance requirements. It is assumed that the objective and constraints are defined well enough that no further partitioning is needed. If partitioning is needed, then the constraints or objectives have been insufficiently designed and the engineer should reconsider all aspects of the level of abstraction.