A Methodology for Analyzing the Consequences of Accidents in Sodium-Cooled Fast Reactors

THESIS

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ABSTRACT

The objective of this research is to develop methods to analyze the offsite consequences of specific sodium fast reactor (SFR) licensing basis events as used to characterize the safety of a nuclear plant design in a license application. An algorithm has been developed which assesses pool heatup, containment load, radionuclide transport, and release from containment during accident scenarios. To analyze temperature transients, the SFR pool has been divided into two, well-mixed regions separated by a metal divider referred to as a redan. Heatup due to fission power or decay heat in the core and heat removal by either the Intermediate Heat Exchanger (IHX) or passive heat removal system drive the energy balance. A reactor kinetics model is also included in this analysis. Radionuclide transport from the cover gas region to containment, as well as the heat and pressure load on containment are analyzed using MELCOR. Subsequent releases from containment to the environment are then analyzed using WinMACCS or Regulatory Guide 1.145 to calculate offsite consequences. Both WinMACCS and Regulatory Guide 1.145 employ a Gaussian dispersion model and meteorological data from an Eastern U.S. site.

Several accident scenarios are examined in the radionuclide release and transport analysis: varying sizes of sodium spills inside containment, arrested melt scenarios, an energetic recriticality event, core uncovery and a case of sustained sodium vaporization. The scenarios will be analyzed for both containment intact and containment failed, where
the containment has a design basis leak rate of 1.0 v/o per day. Resulting offsite consequences are then compared with the NRC’S frequency-consequence (F-C) curve in the Technology Neutral Framework (TNF) for compliance. Calculations are also performed to examine the implications of satisfying the F-C curve and its relationship to satisfaction of the NRC’s Quantitative Health Objectives.
Dedicated to my family.
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CHAPTER 1:

INTRODUCTION

The nation’s growing concern about greenhouse gas production in electricity generation and the issues of nonproliferation and nuclear waste reduction all drive the nuclear energy industry to implement a new generation of reactors which would satisfy stringent economic, safety and non-proliferation design objectives. For several decades, multiple designs of a Sodium Fast Reactor (SFR) have been investigated as a means to more effectively use the world’s uranium resources in the role of breeder reactors. More recently, consideration has also been given to the use of SFRs as actinide burners as a way to reduce the amount of very long lived radionuclides that would be placed in a geologic waste repository that are being produced by our current fleet of Light Water Reactors (LWR). In either role, a high level of safety and reliability are also desirable qualities.

The existing regulations for the design and operation of nuclear power plants were developed for LWRs. With a new generation of reactors involving substantially different design features, a new approach to safety analysis is required. Historically the basis on which a nuclear power plant’s safety was judged was through the analysis of a set of deterministic Design Basis Accidents. New reactor designs incorporating metallic fuel,
liquid metal coolant and passive safety systems significantly alter the criteria by which one designs and assesses accident response. While one can never design for prevention or mitigation of all accidents, the safety analyst must identify a spectrum of accident scenarios, and through analysis demonstrate that an adequate level of safety protection is provided. When the existing LWRs were licensed, deterministic criteria were used to determine what constituted an adequate level of safety. Today, the NRC is examining risk-informed approaches to the licensing of new reactor designs, while maintaining the defense-in-depth aspects of deterministic criteria.

In light of the need for revised regulations, the Nuclear Regulatory Commission (NRC) has developed a draft technology neutral framework (TNF) as a possible risk-informed approach to the regulation of new plant designs [1]. By regulating the licensing requirements in a technology-neutral manner, the design process and licensing phase can be made more consistent between design-types. Despite the necessity of a new approach to safety analysis, the ultimate design goals for a reactor remain unchanged with regards to offsite consequences. The NRC maintains Quantitative Health Objectives which place limits on the risk to the public [2].

This research is an element of a three year DOE-NERI project titled “Risk-Informed Balancing of Safety, Non-proliferation and Economics for the Sodium-Cooled Fast Reactor.” The overall objective of this project is to develop a methodology for minimizing the economic cost of an SFR while satisfying constraints on acceptable safety
and acceptable protection against proliferation. The draft TNF provides a quantitative measure of acceptable safety. In order to support the assessment of design alternatives, this research has assisted in the development of a methodology which analyzes several accident scenarios critical to SFR safety in accordance with the draft TNF. In the case of an LWR, the analysis of transient behavior and release to the environment could be performed entirely by MELCOR, a severe accident analysis code developed by Sandia National Laboratories [3]. No comparable computer code exists for SFR accident analysis. MELCOR currently does not contain models for sodium reactor fuel degradation and failure or for the sodium equation of state. MELCOR does, however, have a generic treatment of containment thermal-hydraulic response and for aerosol transport and deposition mechanisms in containment. Thus, at this time, it is necessary to use an integrated suite of codes for the analysis of the entire accident scenario. This suite contains multiple codes with the following responsibilities: analysis of reactor behavior during the transient, analysis of pool heatup, analysis of containment load and radionuclide transport and release, and analysis of offsite consequences. The first code used in the analysis of SFR accidents is SAS4A. This code was developed by Argonne National Laboratory [4] to perform severe accident analyses in liquid metal reactors. SAS4A is similar to MELCOR in that it has the ability to deterministically analyze the thermal, neutronic, hydraulic and mechanical phenomena observed during steady-state and transient conditions to the point of fuel failure. While SAS4A contains well-developed models of fast reactor kinetics, a complete model of fuel melting does not exist for SAS4A. To continue the analysis from initiation of fuel failure, pool heatup is
analyzed using a C++ code called RCS developed as part of this thesis. Containment load and radionuclide transport and release analyses are performed using MELCOR. Lastly, offsite consequences are calculated using a combination of WinMACCS and an algorithm based on Regulatory Guide 1.145.

Seven different accident scenarios are considered in this research: a large and small sodium spill inside containment, an arrested melt scenario which fails the primary system and vaporizes sodium, an arrested melt scenario that does not fail the primary system, an energetic event, a core uncovery scenario and a case of sustained sodium vaporization. All cases, with the exception of extended sodium vaporization, will also be analyzed with both the containment intact and failed, creating thirteen total cases analyzed in this research.

To analyze the consequences of the above scenarios, the heatup of the sodium pool following the transient is first examined. Within this algorithm based on energy balances, the pool is treated as two well-mixed volumes; effects of the core and Intermediate Heat Exchanger (IHX) are accounted for in the energy balance of each pool. The effects of reactor kinetics and decay heat are also considered. Sodium density and specific enthalpy, both as a function of temperature, as well as temperature as a function of enthalpy are represented by analytic expressions in this algorithm.
Following the pool analysis, MELCOR 1.86 is then utilized to examine both the temperature and pressure load on containment. MELCOR, is currently the industry standard for the analysis of severe accident progression in LWRs. MELCOR includes models for the thermal hydraulic response of coolant systems, core melting and subsequent degradation, among other things. In this research however, MELCOR will be solely utilized to examine loads on containment, as well as radionuclide release and transport inside containment, and their eventual release. Estimations of radionuclide release from the fuel and their rise through the sodium pool provide a source term for analysis of radionuclide transport inside containment. The MELCOR model consists of a simplified version of the SFR and includes volumes for the containment and the environment, and a heat source below containment used to simulate the hot sodium pool. Flow paths from the containment to the outside environment are also included.

Radionuclide release from containment will then be analyzed using two tools: WinMACCS and an algorithm based on Regulatory Guide 1.145. WinMACCS, developed by Sandia National Laboratories, is an offsite consequence analysis tool used primarily to evaluate the offsite impacts of severe accidents [5]. This software assesses radionuclide transport and dispersion in the environment accounting for changes in meteorology. Atmospheric transport and dispersion (ATD) are calculated based on a Gaussian-plume model and includes plume meander. Stochastic and deterministic health effects are calculated based on the coefficients of Federal Guidance Report 13 [6]. Offsite radionuclide concentration is analyzed using the algorithm based on atmospheric
dispersion prescribed in RG 1.145, and also includes treatment of plume meander. The final output of these two tools is then compared to the draft TNF to demonstrate compliance with the frequency-consequence curve. For the case of an energetic event, in addition to comparison with the TNF, the results will also be compared with the NRC’s QHOs to show that if the TNF’s frequency-consequence curve is satisfied, the QHOs are also satisfied by a large margin. The QHOs assign a risk limit to the population surrounding a nuclear power plant, where the risk due to plant operations must be a negligible fraction of risk due to all other sources.
CHAPTER 2:

BACKGROUND

2.1 DOE-NERI Project Objectives

The Ohio State University is currently collaborating with Idaho State University and The Massachusetts Institute of Technology on a Department of Energy’s Nuclear Energy Research Initiative (DOE-NERI) project titled “Risk-Informed Balancing of Safety, Non-proliferation and Economics for the Sodium-Cooled Fast Reactor.” The overall goal of the DOE NERI program is to integrate educational institutions with the DOE’s research objectives while addressing the challenges of applied engineering research and development and advancing our nation’s nuclear industry and technology. This particular project falls under the Generation IV Nuclear Energy Systems Initiative and specifically focuses on Design and Evaluation Methods for a Generation IV (Gen IV) reactor. The Gen IV reactors are intended to address our country’s longer term energy needs; the goals of these new designs were outlined by the Gen IV International Forum (GIF) and include safety and proliferation resistance improvements and resource utilization and cost reductions [7]. The DOE program recognizes that the evaluation and validation of Gen IV designs require new modeling and analytical methodologies as well as computer codes and information databases.
The objective of this specific DOE-NERI project is to establish a risk-informed design methodology and subsequent evaluation tools for the Sodium Fast Reactor (SFR). Issues in which design alternatives are considered are for reducing power generation cost within constraints of acceptable risk and acceptable resistance to proliferation.

2.2 SFR Design Characteristics

The SFR is one of several advanced designs proposed among the alternative Generation IV reactors. This reactor operates on a fast spectrum with liquid sodium as a coolant and is intended to operate in the near term on a closed-loop, actinide burning fuel cycle with the possibility in the long term of utilizing a breeding fuel cycle. Two design alternatives are currently under investigation: a compact loop arrangement or an integral pool-type (Figure 2.1); the latter design is the focus of this research. In the pool design, the core, intermediate heat exchangers (IHX), and primary pumps are housed in a sodium pool contained within the reactor vessel. The sodium pool is isolated from the containment atmosphere by a deck which closes the primary system; while there are no penetrations in the reactor vessel itself, all penetrations for fuel access and monitoring are located in the deck. Within the pool, the redan separates the hot sodium exiting the core from the cold sodium entering the core; the only additional penetration of the redan is the outlet of the IHX which removes heat from the hot pool and deposits cold sodium into the
lower cold pool. The IHX loop maintains the secondary system barrier. An additional tertiary loop receives heat from the secondary system and drives the steam generators.

Figure 2.1: Plant layout for the SFR [8].

2.3 SFR Severe Accident Characteristics

The approach to reactor safety in the nuclear industry utilizes the concept of defense-in-depth, which requires multiple barriers to protect the public from radioactive releases resulting from an accident. A key advantage to the SFR’s design is its primary sodium pool in which the core is immersed. This pool provides several safety functions, most of them passive and inherent to the design. Sodium’s thermophysical properties include a high heat transfer capability and a significant margin to boiling at the reactor’s
operating temperature and pressure. Given the large volume of the pool, there is a significant amount of thermal inertia which increases the response time of the primary and secondary coolant. Decay heat removal is also achieved passively due to natural circulation of the pool. The choice of metallic fuel also includes inherent benefits over the alternative oxide fuel; metallic fuel is highly conductive and has a reduced Doppler response during cooldown transients.

In part because of the passive safety features of the SFR design, many initiating events that can lead to severe accident progression in LWRs have reduced safety significance in SFRs. Single failures of the SFR components generally lead to scenarios that can be mitigated by inherent design of the reactor rather than relying on the operation of standby safety systems. Much of LWR safety is associated with the prevention and mitigation of loss of coolant accidents. Due largely to the SFR’s low operating pressure, SFR accidents rarely involve scenarios in which the primary system boundary is in a failed state when core degradation begins. Core damage events in SFRs typically involve multiple failures and fall into three categories which envelope reactor transients: a change in coolant flow, a change in coolant inlet temperature, and a change in reactivity.

A change in coolant flow may result from failure of the primary or secondary sodium pumps or a blockage in the core. In order to result in fuel damage, typically multiple pump failures with rapid flow coastdown and failure to scram are required. Increase in coolant inlet temperature can result from a loss of the heat sink, be it a failure
in the IHX or the loss of the ultimate heat sink in the tertiary loop. Both events, a reduction in coolant flow and increase in coolant inlet temperature would reduce the primary coolant’s capacity to remove heat from the fuel, potentially leading to fuel heatup and possible fuel melting. An insertion of positive reactivity, such as an uncontrolled rod withdrawal event, can also lead to a temperature excursion in the fuel. If there is a large reactivity excursion with failure to scram there is a potential for fuel melting and clad failure. Sodium boiling in an excursion is a particular concern in SFRs because of the positive sodium void coefficient.

Unlike LWR core designs in which the configuration of fuel and moderation is near optimal, geometry change in a fast reactor is of concern since fuel is not in its most reactive configuration. Fuel melting could lead to two mechanisms of positive reactivity insertion. If molten fuel collects in the fuel channel and moves downward to form a more compact configuration, a positive insertion of reactivity would occur. Furthermore, because of the positive sodium void coefficient near the center of the core, the ejection of liquid sodium from fuel channels could result in a large positive reactivity insertion. Hypothetically, a resulting super prompt critical condition could result in fuel vaporization, a rapid expansion of a bubble of fuel and sodium vapor, destruction of the core, failure of the reactor primary system boundary, the production of missiles and failure of the containment. This scenario is referred to as the hypothetical core disruptive accident (HCDA).
Whether HCDAs are potentially real events that must be considered in establishing design bases for the containment, very low probability events that can be eliminated from design basis considerations, or mechanistically unrealistic fantasies of creative analysts has been hotly debated. The answer may be design dependent. An argument has been made by advocates of metallic-fueled SFR designs that in scenarios of interest, fuel melting within a fuel pin results in an axial movement that decreases reactivity, fuel failure when it occurs would likely be above the active core region, and that the fuel debris would be swept out of the core region. This removal of fuel from the core will most likely shut down the reactor before fuel vaporization becomes a possibility [9].

2.4 Technology Neutral Framework Safety Constraints

The NRC’s Office of Nuclear Regulatory Research recently issued a report describing an approach to the licensing of future reactors that would be independent of reactor design type. The historic approach to reactor licensing involved identification of design basis accidents and the demonstration of compliance with deterministic criteria. The Technology Neutral Framework (TNF), NUREG-1860, provides a more structured approach to the identification of licensing basis events (the analog to design basis accidents) based on PRA accident scenarios and the use of frequency dependent consequence limits.
Future reactor designs are expected to adhere to the NRC’s risk objectives by providing enhanced margins of safety, complying with the Commission’s Safety Goal Policy Statement, and presenting negligible risk to the public. The level of safety, or risk acceptance, of a plant is typically represented by a three-region approach (Figure 2.2). The result of NUREG-1150 indicated that current operating reactors satisfy the safety goals with substantial margin, at least for internally-initiated accidents. Nevertheless, the Commission has indicated that the risk of future plant designs should be even safer than existing plants.

Figure 2.2: Three-region approach to risk profiling [1].

The TNF also proposes a new measure of safety acceptability called the frequency-consequence (F-C) curve (Figure 2.3). The intention of this curve is to establish the maximum allowable consequence for a given sequence frequency. Low frequency sequences are permitted to have higher offsite consequences than high
frequency events. The specific consequences selected as limits have historical regulatory precedence, such as the 25 rem limit for site-dose calculations, the 5 rem limit for protective action guidelines and the 0.1 rem for annual exposures to members of the public. At the high end of the frequency scale, the limits represent integrated exposures over a time period of one year rather than the limits for a single event. The specific limits are tabulated in Table 2.1.

Figure 2.3 : Frequency-consequence curve based on the TNF.
Table 2.1: Dose range and limiting frequency for FC Curve in TNF.

<table>
<thead>
<tr>
<th>Dose Range (rem, TEDE)</th>
<th>Frequency (yr⁻¹)</th>
<th>Source/Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.001 – 0.005</td>
<td>1.0</td>
<td>10 CFR 50 App 1, ALARA</td>
</tr>
<tr>
<td>0.005 – 0.1</td>
<td>1 x 10⁻⁴</td>
<td>10 CFR 20, licensed operation</td>
</tr>
<tr>
<td>0.1 – 1.0</td>
<td>1 x 10⁻³</td>
<td>EPA Protective Action Guidelines</td>
</tr>
<tr>
<td>1 – 25</td>
<td>1 x 10⁻⁴</td>
<td>10 CFR 50.34a and 10 CFR 100</td>
</tr>
<tr>
<td>25 – 100</td>
<td>1 x 10⁻⁵</td>
<td>Deterministic effects possible at 50 rem</td>
</tr>
<tr>
<td>100 – 300</td>
<td>1 x 10⁻⁶</td>
<td>Exceedance of early fatality threshold</td>
</tr>
<tr>
<td>300 – 500</td>
<td>5 x 10⁻⁷</td>
<td>Early fatality likely</td>
</tr>
<tr>
<td>&gt;500</td>
<td>1 x 10⁻⁷</td>
<td>Curve capped</td>
</tr>
</tbody>
</table>

The current LWR fleet satisfies the NRC’s safety goals by complying with risk objectives that act as a surrogate for the QHOs. These surrogate risk objectives are design specific and relate the frequency of core damage and the likelihood of a large release of radioactivity, i.e. core damage frequency (CDF) and large early release frequency (LERF), respectively, to the QHOs. While these metrics are adequate comparisons for LWRs, they are not necessarily applicable to new reactor designs. For example, in a gas-cooled, graphite-moderated reactor, the concept of core damage is less clearly defined than in an LWR or SFR in which fuel melting represents a threshold for substantial release of radioactive material from the fuel. The TNF as discussed in NUREG-1860 addresses consequence limits for individual accident scenarios (LBES) that are characteristic of classes of accident scenarios. Because the approach does not integrate over the class of scenarios, the TNF as developed does not actually impose a risk limit that can be compared with the QHOs.
The TNF also outlines several safety fundamentals which are to be utilized in developing regulations for future plant designs. These fundamentals consist of five protective strategies that will ensure the safety of the public with significant margin: physical protection (i.e. against illicit activities), stable operation, protective systems (i.e. accident mitigation), barrier integrity and protective actions. These approaches to safety encompass all engineering aspects by which a plant may be challenged.

2.5 Severe Accident Analysis Approach

Currently in the nuclear industry, the risk profile of a power plant is identified by carrying out a probabilistic risk assessment (PRA). Various software tools are utilized for Level-specific analyses (i.e. Level I, Level II, Level III). This thesis specifically covers Levels II and III of a PRA, where Level II analysis includes severe accident progression, containment response and source term estimation, and Level III encompasses offsite consequence analysis.

In the NERI project, reactor transient behavior leading to fuel failure is analyzed using SAS4A. SAS4A analyzes the transient thermal-hydraulic behavior of the reactor coolant system and balance of plant. It also analyzes the thermal response of fuel rods in some detail including the prediction of clad failure. Specific accident scenarios are analyzed with SAS4A given particular reactivity insertions and states of the plant (i.e.
loss of heat sink, primary pump failure with coastdown, etc.). The timing and extent of fuel damage obtained from SAS4A provide input to the radionuclide release and transport analyses. While SAS4A is analogous to MELCOR in that it has the capability to deterministically analyze thermal, neutronic and mechanical phenomena during transients, SAS4A does not calculate the release and transport of radionuclides. Transients are analyzed with SAS4A up to the point of fuel failure, but time at temperature information gathered in SAS4A is subsequently in an external analysis of the source term. Arrhenius curves, volatility estimates and EBR-II data are used in the source term model. The modeling of the release from fuel was not part of this thesis. A limited analysis of the sodium pool’s effect on radionuclide retention was also used in the source term model.

The timing of fuel failure, extent of fuel failure and source term analysis are then input to a computer code called RCS. Within this thesis research, RCS was written to provide an interface between the fuel behavior analysis with SAS4A and the containment behavior calculated with MELCOR. The primary intention of RCS was to perform long-term transient analyses of neutronic and thermal behavior, and radionuclide release (i.e. pool scrubbing); an example scenario would be a failure to scram without shutdown for several days. RCS assesses sodium pool processes including an energy balance on the pool and reactor kinetics in the later stages of an accident. In follow-on work to this thesis, radionuclide release to the pool and transport to the containment will be assessed. It should be noted that problem time starts at time zero with the initiation of the RCS.
analysis; the remainder of the codes used in this methodology continue in time sequentially.

Thermal and pressure loads on containment and radionuclide transport and release from containment are analyzed using MELCOR 1.86. Lastly, the offsite consequences of the source term gathered from MELCOR analyses are calculated using an algorithm based on Regulatory Guide 1.145. Perspective on the ability of SFRs to satisfy the NRC’s QHOs is also obtained using WinMACCS to determine individual risk in the vicinity of the plant. Not all portions of the methodology in this section (Figure 2.4) were developed in this thesis research. Specifically, the writing of the RCS computer code, the development of the MELCOR model, the development of the offsite consequence analysis model, and the development of the oxidation release model were performed within this thesis. In addition, a number of accident scenarios were analyzed to develop estimates of offsite doses to support the tradeoff studies in the NERI project. A bounding calculation was also performed to determine the degree of conservatism in the TNF with respect to satisfying the NRC’s Quantitative Health Objectives. Analyses with SAS4A, the development of the radionuclide release model, and the pool scrubbing analysis were not performed in this thesis.
2.5.1 Code Suites Available

Since WASH-1400, the first major risk assessment of a nuclear power plant [10], there have been several evolutions of severe accident analysis codes. These codes enable
analysts to examine a reactor’s response to a range of accident scenarios. The MELCOR/MACCS computer codes were developed by Sandia National Laboratories with funding from the NRC to describe severe accident behavior in LWRs. Although the level of fidelity of some of the models in MELCOR is crude, the scope of severe accident phenomena addressed by MELCOR is comprehensive and permits an integrated analysis to be performed. A combined MELCOR/MACCS analysis can begin with analyzing the thermal-hydraulic behavior of an initial transient and end with the calculation of dose to offsite individuals. MELCOR’s capabilities include, but are not limited to, thermal hydraulic analysis of volumes and flow paths (including coolant systems, vessels and buildings), core degradation and relocation, and fission product release from fuel and its subsequent transport through volumes and flow paths. MELCOR operates on a unified framework which inherently contains multiple phenomenology and program control packages that drive the overall code response. It should also be noted that MELCOR is restricted to use water as the working fluid, and is not yet capable of using liquid sodium in its analysis.

Plant designs in MELCOR are strictly user-defined and consist of heat structures and control volumes connected by single-direction flow paths. The desired level of detail in the analysis therefore is controlled by the user’s chosen nodalization through the use of control volumes. The control volume and flow path packages serve as the backbone of all MELCOR analyses, and all other packages are dependent on this information; boundary conditions for all other packages are calculated based on control volume and
flow path information. To advance the simulation in time, an algorithm is applied to all control volumes and flow paths, and all equations developed by the algorithm are simultaneously solved. Within control volumes resides all material, including fuel, radionuclides and water, which may be modeled as a liquid, vapor or mixture of both (where conventional notation for the liquid phase of water is pool, atmosphere for water in the vapor phase, and fog for a mixture). In the interest of efficiency, MELCOR utilizes lumped-parameters within control volumes and assigns a single energy to the volume if it is declared to be in equilibrium (as is the assumption for this research); substances within the volume, such as atmosphere or pool, are assigned their own temperature. A single pressure for the volume exists, and is referenced at the interface between pool and atmosphere. Flow paths in MELCOR do not contain any material or energy, but the flow of mass and energy is permitted through the path.

Most calculations in MELCOR are based on mechanistic models; however some models still follow a parametric approach where phenomenological behavior is not well understood. For control volume analysis, the governing equations of state include the conservation of mass, energy and momentum. Correlations for quantities such as specific heat, density, and enthalpy are provided by correlations inherent in each package.

The transport of radionuclides is handled by a specific package within MELCOR. To appropriately account for the release and transport of fission products, all other aerosols present in the model which may affect fission product transport must be
accounted for; examples of this may include water vapor or hydrogen, both of which may not be radioactive. Individual isotopes of radioactive material are not tracked in MELCOR, instead the mass of an element is summed over all isotopes. In addition to this summing, elements and their nontrivial molecular forms are lumped into classes which consist of similar chemical characteristics. There are fifteen default classes in MELCOR accounting for the majority of elements relative to fission product release. Aerosol dynamics are modeled within control volumes based on models from the multisection, multicomponent aerosol dynamics code, MAEROS; these dynamics include agglomeration, condensation, advection and deposition, although the MAEROS models are not employed for condensation. Intervolume radionuclide transport is included in MELCOR and is associated with advection from the hydrodynamic calculations of the control volume package. MELCOR does have the capability to track the intra- and intervolume transport of individual radionuclide classes and their size distribution.

With the growth of the nuclear energy industry and events such as Three Mile Island in 1979 came the development of reactor safety analysis and PRA. One of the components included in PRA is the aspect of Level III analysis, which dictates the probability and severity of offsite consequences. Given the ability to model severe accident progression using tools such as MELCOR, it is often necessary to also analyze offsite consequences. WinMACCS is one of several codes capable of performing this analysis. Developed primarily by Sandia National Labs, WinMACCS is the graphical user interface version of MACCS2, the MELCOR Accident Consequence Code System,
Version 2. Phenomena modeled in MACCS2 include atmospheric transport and dispersion (ATD) for specific meteorological data input, exposure to offsite personnel by a variety of pathways, offsite doses, stochastic and deterministic health effects, and economic impacts.

Radionuclide transport is calculated based on a Gaussian-plume model, where transport is largely affected by the prevailing wind. MACCS2 has the capability to include real meteorological data including information on wind direction, wind speed and precipitation; there are also several weather sampling methods available in the code, allowing the user to use randomly selected meteorological data or constant conditions. Transport and deposition may also be affected by building wake, buoyant rise and wet and dry deposition. Dispersion coefficients are based on Pasquill-Gifford dispersion coefficients [11]. Source terms may be modeled by several plumes if necessary; radioactive decay before release and during suspension in the atmosphere is also accounted for. Plume meander is modeled as prescribed by Regulatory Guide 1.145. Precipitation may also be modeled, as this affects fallout from the cloud. In addition to cloud concentration, MACCS2 tracks the ground concentration of radionuclides.

Consequences may be calculated at any radial location from the plume origin. Doses and other health effects are calculated assuming five pathways: direct external exposure from cloudshine, direct external exposure from groundshine, cloud inhalation, inhalation from resuspension and skin deposition. Cloudshine exposure and cloud
inhalation are limited to the duration of the passing plume. Two types of doses are calculated: acute doses affecting early fatalities and lifetime doses leading to latent cancers. All doses are converted to health effects in accordance with coefficients from Federal Guidance Report 13. In MACCS2, population information may be supplied by a site file which details populations radially based on sectors of the wind rose. MACCS2 also has the capability to model several emergency response phases and includes factors such as protective actions and evacuation, specified by route.

2.5.2 Software Inadequacies and Need for RCS Code

Traditionally for an LWR severe accident analysis, MELCOR would be employed to analyze the thermal hydraulic behavior of the reactor coolant system and the subsequent degradation of fuel. However, MELCOR currently does not have the capability for performing accident sequence analyses with sodium as a coolant. In addition, the fuel failure and core melt progression models are specifically based on LWR severe accident behavior. A version of MELCOR utilizing the sodium equation of state is currently under development at Sandia National Laboratory, but was not available for this research. Therefore, MELCOR is solely utilized in this research for assessing containment pressure and temperature loads, and the transport and deposition of radionuclides in containment. A simplified MELCOR model was created which consists only of a containment volume and an environment volume. Control volumes for the
cover gas region and deck space were not created, as MELCOR’s ability to model natural circulation is very limited.

A method for assessing the sodium pool heatup and radionuclide release and transport through the pool was required for this research. This algorithm, called RCS, is to act as a transition from analyses performed in SAS4A to MELCOR. Developed in C++, RCS gathers temperature data from SAS4A and performs an energy balance for the SFR pool given particular operating conditions. RCS also performs a reactor kinetics analysis to assess the long term heat source to the sodium pool. Because SAS4A and RCS are run in a serial manner, the RCS analysis needs to faithfully reproduce the conditions predicted by SAS4A for the initial stage of the accident. This is accomplished by using the net reactivity calculated by SAS4A as input to RCS for this phase. The results of the RCS calculation are then used in conjunction with estimates of fuel release rates to assess radionuclide release. These releases are then input to MELCOR, where radionuclide transport and release is assessed.

2.6 Offsite Consequence Analysis Using Regulatory Guide 1.145

The TNF prescribes a methodology for offsite consequence analysis which is dependent on dose location and duration. Three separate analysis categories have been designated for the offsite location of the calculated dose that are a function of accident frequency. The first category consists of the annual dose to an individual within the
Exclusion Area Boundary (EAB), typically located 1 km from the plant, and may be interpreted as the annual total effective dose equivalent (TEDE). The second category consists of 2-hour doses that result from a specific event, and are to be calculated at the EAB. Within the third category are rare events which occur with a frequency of less than $1 \times 10^{-5}$ per reactor year. The consequences of these events are to be analyzed at 1 mile from the EAB for a duration of 24 hours. For this research, all cases are treated as rare events and will be analyzed as described by the third category. Cases involving instantaneous releases of sodium will also be analyzed with the criteria described by the second category.

The methodology for evaluating radionuclide cloud concentration given site-specific meteorological data is outlined in the NRC’s Regulatory Guide 1.145. This document provides guidelines for licensees for the performance of site dose calculations for design basis accidents. Methods for calculating a 2-hour and an accident duration atmospheric concentration for both ground and stack releases are included. Wind speed, wind direction and atmospheric stability are required for the analysis. Maximum sector concentration and 95th percentile overall site concentration analysis methods are also presented, where the maximum sector concentration is the concentration value that is exceeded 0.5% of the total hours in the data set, and the 95th percentile overall site concentration is the concentration value that is exceeded 5% of the total number of hours in the data set. To evaluate a site’s acceptability, the larger of maximum sector or 95th percentile overall site concentration should be taken as the concentration value.
2.7 Nuclear Regulatory Commission’s Quantitative Health Objectives

The NRC commissioners issued a Safety Goal Policy Statement which qualitatively outlines the mean risk to the public from a nuclear power plant accident should “bear no significant additional risk to life and health,” and that societal risk should not exceed other societal risks [2]. In addition to these qualitative goals, the NRC has also established Quantitative Health Objectives (QHOs). These goals prescribe a limit to mean individual prompt fatality risk and mean population latent fatality risk resulting from a nuclear power plant accident. Individual prompt fatality risks should not exceed 0.1% of the sum of other prompt fatality risks within one mile of the plant to which the public is typically exposed; in terms of frequency, this limit is $5 \times 10^{-7}$ per year. Latent fatality risks should not exceed 0.1% of the sum of all other cancer risks within ten miles of the plant, or are limited to a frequency of $2 \times 10^{-6}$ per year. Although the latent fatality risk is referred to as a societal risk, it is actually the mean risk to individuals living in this region. It is affected by the distribution of people living within ten miles of the plant but is independent of the total population.
CHAPTER 3:

METHODOLOGY

3.1 RG 1.145 Approach to Development of 95th Percentile Meteorology

To assess the meteorological conditions leading to the top 95th percentile value of airborne concentration of the total data set for data at a site, Regulatory Guide 1.145 requires a complementary cumulative distribution function (CCDF) of the cloud concentration be constructed [12]. The cloud concentration is evaluated at a down-wind location at the center of the plume; for this research, cloud concentration was evaluated at the site boundary (1.0 km) and at one mile from the site boundary (2.6 km). The Guide prescribes the following equation for ground-level cloud concentration during non-fumigation conditions at the cloud centerline:

\[
\frac{\chi}{Q} = \frac{1}{\pi U \sigma_x \sigma_y} e^{\frac{h^2}{2\sigma_z^2}}
\]

(3.1)

where:

\[
\frac{\chi}{Q} = \text{cloud concentration s m}^{-3},
\]
$\bar{U}_s$ = mean wind speed at the release height (m/s),

$\sigma_y$ = lateral plume spread as a function of atmospheric stability (m),

$\sigma_z$ = vertical plume spread as a function of atmospheric stability (m),

$h_e = h_s - h_t$ = effective stack height (m),

$h_s$ = initial plume height above plant grade (m),

$h_t$ = maximum terrain height between release point and evaluation location (m). If $h_t > h_s$, then $h_s = 0$.

Given a specific meteorological data set and the appropriate cloud concentration, a cumulative probability distribution is then constructed. The meteorological data consist of wind speed, wind direction, stability class and rainfall for one year for a specific site. A probability of exceedance is assigned to each cloud concentration value based on the total number of data points. The cloud concentration value exceeding 5% of the total data set is then chosen as the cloud concentration value indicative of the overall site value for 95th percentile meteorology.

It should be noted that Regulatory Guide 1.145 also includes a methodology for calculating a sector-based cloud concentration value. The cloud concentrations for each sector are found using a methodology similar to the above process; however, the value exceeded 0.5% of the total data set is to be selected at the concentration representative of that sector, giving the analyst 16 separate values of cloud concentration. This method was not used for this research, as it is more likely to be characteristic of a specific site.
3.2 Source Term Modeling

Significant data currently exist for assessing radionuclide release from oxide fuel, however a similar database does not exist for metallic fuel. As this research focuses primarily on an SFR utilizing metallic fuel, fairly realistic source terms must be estimated to perform the offsite consequence analysis. It is expected that accidents involving severe fuel damage will dominate risk. These releases are also expected to be highly dependent on power level.

It is known that molten uranium and plutonium are very good solvents, and based on oxide data it is well understood that release rate from molten fuel is highly sensitive to temperature and oxidation. The chemical form of radionuclides, chemical volatilities as a function of temperature, concentration, and duration at particular temperatures are assessed to derive the source term for metallic fuel. The release from the fuel as a function of temperature is typically expressed in the form of an Arrhenius equation, which is very sensitive to temperature. Metallic fuel will form a eutectic with the steel clad, which results in a very low melting temperature in comparison with oxide fuel. Thus, relative to oxide fuel it is likely that release rates of radionuclides will be low. This, of course, also depends on the chemical form of the radionuclides. Considering the low melting temperature of metallic fuel and the temperature of the coolant, the molten
release will not remain at high temperatures for an extended period of time; any expelled fuel will most likely assume refreeze upon exiting the fuel pin.

Mechanism of release and release fractions for noble gases in metallic fuel was gathered from EBR-II data [13]. A large percentage of noble gases (50 to 75%) is released from the fuel to the gas plenum at the top of the fuel pin during operation; other volatile fission products may also be released from the fuel during operation but are likely to be in solution in the sodium bond. Upon cladding failure, effectively all of the noble gases within the plenum will be released to the coolant. The sodium bond material is also expected to be entrained in the flow of escaping noble gases. Any radionuclides contained in the bonded sodium will be carried to the sodium pool. The noble gases are expected to travel as a bubble to the cover gas region, given that the solubility of noble gases is negligible. The Henry’s Law constant for sodium at 773 K is $8.5 \times 10^{-9}$ atom fraction in solution per atmosphere of gas pressure [14]. Given a sodium pool of $1.9 \times 10^{6}$ kg, cover gas region of $78.5 \text{ m}^3$ and assuming one mole of krypton is released, the fraction of krypton in solution would be $1 \times 10^{-4}$; a similar analysis performed for xenon will also show a negligible solubility component.

It is well understood that iodine, cesium, strontium and barium have significant effects on offsite consequences in LWR accident scenarios. Iodine may be released in the vapor phase as $\text{I}_2$, sodium iodide or cesium iodide; however, iodine also reacts with uranium to form the non-volatile $\text{UI}_3$, greatly reducing the potential contribution of iodine
in the vapor phase to the source term. Given that significantly more uranium and sodium are present in the fuel pin compared to cesium, and given the high boiling points of UI₃ and NaI (on the order of 1300 K and higher), it is projected that the iodine component of the source term will be extremely small. Cesium, which is also typically considered a volatile fission product, is also expected to have a limited contribution to the source term. Because uranium is such an excellent solvent, the activity coefficient for cesium is expected to be very small and the amount released from the fuel will be substantially suppressed. Cesium can react with iodine and other fission products. There is also some potential for cesium released in elemental form from the fuel to react with sodium. However, at temperatures above 370 K, the cesium would be in solution in the sodium pool. The contribution of strontium and barium to the source term are expected to be very small given their high boiling temperatures of 1655 K and 2170 K, respectively; it is unlikely that significant quantities of either of these elements will be released in the vapor phase. It is expected that the majority of volatiles and actinides released into the sodium will remain trapped in the sodium pool because of their very low concentrations.

A source term consisting of activated sodium should also be considered, particularly for accident sequences involving a sodium fire. Once the sodium has been activated in the core, it mixes uniformly in the pool; the activated sodium is then recycled through the core. Preliminary calculations estimate the activity concentration of $^{22}$Na to be on the order of $1 \times 10^3$ Ci/kg, and $^{24}$Na to be approximately 11 Ci/kg. However, it should be noted that the half-life of $^{24}$Na is 15 hr, while the half-life of $^{22}$Na is 2.6 yr. For
a 24-hour dose, $^{24}$Na is the more significant contributor to the source term. The radiological impact of sodium on offsite consequence will be reduced, however, for cases where release to the environment is delayed due to the rapid decay of this isotope.

For this research, the reference design of the SFR includes a 1000 MW$_t$ reactor utilizing metallic fuel with a composition of 15% plutonium, 12% zirconium, and 73% uranium. The SFR employs a U-Pu-Zr alloy, for which oxidation data has not been gathered; however, extensive databases exist for the oxidation of uranium and plutonium in metallic form. The corrosion kinetics of uranium and plutonium are largely affected by the temperature, oxygen pressure and water pressure, but the oxidation of these elements will only be discussed for dry air.

Based on Arrhenius curves, a release rate based on temperature may be derived. The oxidation of uranium in dry air may be described by two separate Arrhenius curves which are limited by temperature. For temperatures ranging from 40°C to 350°C, the presence of nitrogen does not affect the release, and the activation energy ($E_a$) is 16.0 kcal mol$^{-1}$ [15]. The release rate, $R$ (g U cm$^{-2}$ min$^{-1}$), is described by the equation:

$$\ln R = 6.19 - \frac{8077}{T}$$

(3.2)

where $T$ is temperature. However, for the temperature range 415°C to 1440°C, the release rate from the oxidation of uranium is described by the least squares equation:
\[
\ln R = 1.21 - \frac{4655}{T}.
\] (3.3)

The activation energy for the temperature range described by equation 3.5 is derived to be 9.3 kcal mol\(^{-1}\), which is significantly lower than the \(E_a\) of the temperature range of equation 3.4. While equation 3.5 does accurately describe corrosion, it fails to properly account for autothermic effects in the 500°C to 600°C range; autothermic reactions are those in which the temperature of the process is maintained by the heat of the reaction alone. Accounting for these effects, the activation energy may be reevaluated and is then found to be 11.0 kcal mol\(^{-1}\), which more closely matches the \(E_a\) for the 40°C to 350°C temperature range [15].

The particle size distribution of oxidized uranium has been shown to depend strongly on the oxidation method itself. Quantitative analysis providing consistent results has been performed for the oxidation of uranium in fuel-fires. The respirable (< 3 µm) and airborne (< 10 µm) mass fractions of oxide for fuel-fires has been shown to be \(1 \times 10^{-3}\) and \(1.5 \times 10^{-2}\), respectively [15].

The oxidation of plutonium in dry air may be described by a single Arrhenium relationship which holds for temperatures greater than 200°C. In this region, the release rate appears to be independent of moisture. The release rate, \(R\) (mg Pu cm\(^{-2}\) min\(^{-1}\)) for plutonium [16] is described by the equation:
A study of the oxidation of plutonium under static conditions has also been performed. In this analysis, plutonium was heated with an arc weld until ignition. It was then placed in a chimney where it was subjected to a natural updraft of 5.3 m/s controlled by a fan. Under these conditions, the release rate, \( R \) (g Pu cm\(^{-2}\) min\(^{-1}\)), described by the Arrhenius equation [17] over the 500°C to 925°C temperature range is:

\[
\ln R = 13.68 - \frac{9010 \pm 280}{T}.
\]  

(3.4)

Release rates ranging from 0.1 g Pu cm\(^{-2}\) min\(^{-1}\) to 0.2 g Pu cm\(^{-2}\) min\(^{-1}\) have also been observed in fuel-fire and air atmospheres for temperatures ranging from 500°C to 925°C [18].

Similar to uranium, the particle size distribution of plutonium is dependent on the mechanism of oxidation. However, compared to uranium, the respirable fraction of oxidized plutonium is significantly less than the estimated value of 0.1% for oxidized uranium [17]. The amount of material released is typically a small fraction and is a function of the velocity of air flowing over the burning metal. The airborne release fraction (ARF) may be represented by the relationship:

\[
\ln R = -0.9 + \frac{1180 \pm 760}{T}.
\]  

(3.5)
\[ ARF = 4 \times 10^{-5} v \]  \hspace{1cm} (3.6)

where \( v \) (m/s) is the velocity of air [19].

The DOE Handbook 3010-94 (Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities, Vol 1) [20] provides a detailed compilation of available data and assessment of ARFs and respirable fractions (RFs) for plutonium metal fires. The 95\(^{th}\) percentile values recommended by the Handbook are \( 5 \times 10^{-4} \) for the ARF and 0.5 for the RF for plutonium. For uranium, the Handbook recommends a value of \( 1 \times 10^{-3} \) for the ARF and 1.0 for the RF. To perform a conservative analysis, the RF is chosen to be \( 1 \times 10^{-3} \); the larger release fraction is chosen since it is not clear how the U-Pu-Zr fuel alloy will behave relative to the behavior of uranium or plutonium during an accident scenario.

For this research, the inventory of radionuclides was calculated using the ORIGEN2 code. ORIGEN was developed by Oak Ridge National Laboratory and calculates the buildup, decay and processing of radionuclides using a matrix exponential method [21]. Radionuclide inventory is calculated as a function of burnup using generic fast reactor cross section data.
Larger source terms could be expected for energetic accident sequences involving fuel vaporization. Loss of long term heat removal resulting in core uncovering in an oxidizing environment could also result in larger source terms. For conditions where fuel vaporization is possible, it is likely that other fission products present in the fuel pin will also be vaporized and then condense to form aerosols. The loss of long term heat removal will lead to sodium boiling and eventual primary system failure, releasing sodium. Sodium will evaporate until the core is uncovered, leading to fuel melting which may continue until vessel penetration. In the scenario involving primary system failure, air may enter the system and lead to fuel oxidation and an enhanced source term.

### 3.3 RCS Code Model

Currently the available version of MELCOR does not have the ability to treat sodium as a working fluid, preventing a MELCOR analysis of pool heatup and cooldown during transients for the SFR. Because of this, the RCS code was developed to predict the sodium pool temperature and to account for radionuclide release and transport within the reactor coolant system (Appendix A). Effects of reactor kinetics and decay heat are also accounted for.

To analyze the temperature of sodium, the SFR pool has been divided into two regions, separated by the redan: the upper pool and lower pool. Each pool is treated as well-mixed. An energy balance for the whole system is performed at each time step of
the code, with heat loss due to the IHX (or passive decay heat removal system) and sodium evaporation and heat up due to fission power and decay heat. Several analytic relationships [22], such as density as a function of temperature (equation 3.7), specific enthalpy as a function of temperature (equation 3.8) and temperature as a function of enthalpy (equation 3.9) are utilized in these balance equations and are given below:

\[
\rho(T) = \rho_c + 275.32 \left(1 - \frac{T}{T_c}\right) + 511.58 \left(1 - \frac{T}{T_c}\right)^{0.5}
\]  
\(\text{(3.7)}\)

\[
h(T) = -365.77 + 1.6582T - 4.2375 \times 10^{-4}T^2 + 1.4847 \times 10^{-7} + \frac{2992.6}{T}
\]  
\(\text{(3.8)}\)

\[
T(h) = 234.9 + 0.6275h + 2.08 \times 10^{-4}h^2 - 9.35 \times 10^{-8}h^3 + 1.04 \times 10^{-11}h^4
\]  
\(\text{(3.9)}\)

where:
\(\rho = \) density of sodium (kg m\(^{-3}\)),
\(\rho_c = \) critical density of sodium (kg m\(^{-3}\)),
\(T = \) temperature of sodium (K),
\(T_c = \) critical temperature of sodium (K),
\(h = \) enthalpy of sodium (kJ kg\(^{-1}\)).
The energy balance for the upper pool (equation 3.11) includes enthalpy flow into the pool from the core outlet (equation 3.10) and enthalpy flow out of the pool into the IHX. For the lower pool energy balance (equation 3.13), the calculation includes enthalpy flow out the IHX (equation 3.12) into the pool and enthalpy flow into the core at the lower pool temperature. Pool temperatures are calculated using equation 3.9 and are a function of the enthalpies calculated using the energy balance equations below:

\[ h_{c,\text{out}}(t') = h(T_c(t)) + \frac{Q(t)}{\dot{m}} \]  

(3.10)

\[ H_u(t') = H_u(t) + \Delta t\dot{m} \left( -h(T_u(t)) + h_{c,\text{out}}(t') \right) \]  

(3.11)

\[ h_{\text{IHX,\text{out}}}(t') = h(T_u(t)) - \frac{Q_{\text{IHX}}(t)}{\dot{m}} \]  

(3.12)

\[ H_l(t') = H_l(t) + \Delta t\dot{m} \left( -h(T_l(t)) + h_{\text{IHX,\text{out}}}(t') \right) \]  

(3.13)

where:

- \( h_{c,\text{out}} = \) specific enthalpy exiting the core (kJ),
- \( T_l = \) lower pool temperature (K),
- \( Q = \) core power (kW),
- \( \dot{m} = \) mass flow rate (kg s\(^{-1}\)).
$H_u$ = enthalpy of the upper pool (kJ),

$T_u$ = upper pool temperature (K),

$h_{\text{IHX, out}}$ = specific enthalpy exiting the IHX (kJ),

$Q_{\text{IHX}}$ = IHX power (kw),

$H_l$ = enthalpy of the lower pool (kJ),

$\Delta t$ = time step,

$t'$ = current problem time (s),

$t = t' - \Delta t$ (s).

Given the upper and lower pool enthalpies, the temperatures of both pools can then be calculated at the end of the time step using the following equation:

$$T_p(t') = T(H_p(t'))$$

(3.14)

where:

subscript $p$ denotes either the upper or lower pool.

The transient response to kinetic perturbations is analyzed in the RCS code by utilizing an integral solution of the reactor kinetic equations. The effect of reactor kinetics on core neutronic power [23] is taken into account using the equation:
\[
Q_n(t') = \frac{0.934Q_0 + \sum_{j=0}^{6} A_j e^{S_j m\Delta t} \sum_{l'=0}^{m-1} e^{-S_j l'\Delta t} R_j Q_j \Delta t}{1 - \sum_{j=0}^{6} A_j R_j \Delta t}
\]  

(3.15)

where:

\( Q_n \) = neutronic power (kW),

\( Q_0 \) = operating power (kW),

\( S_j \) = characteristics roots,

\( m \) = current time step,

\( R \) = total reactivity,

\( l' = m - 1 \).

\( A_j \) is defined as:

\[
A_j = \left( \frac{1 - \beta}{\Lambda} \right) B_j + R_j
\]  

(3.16)

where:

\( \beta \) = total delayed neutron fraction,

\( \Lambda \) = prompt neutron lifetime (s),

\( B_j, R_j \) = characteristic roots corresponding to \( A \).
In equation 3.14, the time step is limited to values less than:

$$\left[ \sum_{j=0}^{6} A_j R_m \right]^{-1} = \frac{I}{R_m}. \quad (3.17)$$

Reactivity is calculated as both a function of inserted reactivity (e.g. control rod withdrawal) and temperature feedback:

$$R_{ins} = R_a + R_b m \Delta t \quad (3.18)$$

$$R_{fb} = R_c T_{c,avg} \quad (3.19)$$

$$R = R_{ins} + R_{fb} \quad (3.20)$$

where:

- $R_{ins}$ = inserted reactivity,
- $R_a, R_b$ = user inputs used to generate a ramp and/or step insertion,
- $R_{fb}$ = reactivity due to temperature feedback,
- $R_c$ = reactivity temperature coefficient,
- $T_{c,avg}$ = average core temperature (K).
For this research, the total delayed neutron fraction, $\beta$, was based on ANL analyses for the ABR-1000 design [24]. The reactivity temperature coefficient, $R_c$, was calculated as an effective coolant temperature coefficient by inserting reactivity into a SAS4A calculation and then by finding the increase in average coolant temperature within the core. Typically, the RCS code will be run by inputting the net reactivity, determined by SAS4A, as a function of time as a tabular function within the RCS code. In this thesis, $\beta = 0.002$ and $R_c = -2.2 \times 10^{-5}$.

For the purpose of determining decay heat for a few days of accident time, it is reasonable to approximate the irradiation history as infinite. However, for scenarios in which there is failure to scram, it is necessary to consider the contribution to decay heat that arises from powered operation following initiation of the accident. The contribution of decay to total core power is also considered in the RCS code. The effects of decay heat due to infinite power history and decay heat post problem start are treated separately. For the case of infinite irradiation time, the standard Wigner-Way equation holds:

$$ q(t) = a t^{-1.2}. \quad (3.21) $$

Given the general form that is equation 3.21, and integrating from $t' = -\infty$ to $t' = 0$ with a constant power of $Q_0$ and assuming the reactor scrams at $t = 0$, the following equation is derived:
\[ \frac{P(t)}{P_0} = 0.066t^{-0.2} \]  \hfill (3.22)

where:

\( P(t) \) = core power at time \( t \),

\( P_0 \) = initial core power.

An additional integral is necessary for the case where core power does not go to zero at time zero:

\[ P(t) = \int_{-\infty}^{0} Q(a) (t-t')^{-1.2} dt' + \int_{0}^{t} Q(t') a (t-t')^{-1.2} dt'. \]  \hfill (3.23)

In this case, the contribution of decay heat to core power may be represented by the equation:

\[ Q_{DCH}(t) = 0.066Q_0t^{-0.2} + \sum_{i} 0.2 \times 0.066Q_i (t-t_d)^{-1.2} \Delta t_d \]  \hfill (3.24)

where:

\( Q_{DCH} \) = decay heat contribution to core power (kw),

\( \Delta t_d \) = time step for decay heat calculation,

\( t_d \) = decay heat problem time (s).
In equation 3.24, the first term represents decay heat due to infinite irradiation time. The second term represents the decay heat contribution due to power generation after the initiation of the transient without scram. Given the power contribution from the neutronic calculation (equation 3.16) and the power contribution from decay heat (equation 3.24), the total power is then represented by:

\[ Q(t) = Q_n(t) + Q_{DCH}(t) \]  

(3.25)

### 3.3.1 C++ and Compiler

The RCS code was written in C++, a general purpose language with an object-oriented structure. C++ is considered a mid-level language as it is comprised of a combination of high-level and low-level language features. An enhancement on the C language, C++ was developed by Bjarne Stroustrup of Bell Labs in 1979 [25]. As a static language, C++ was developed to be both portable and efficient, and operates without a sophisticated programming environment. C++ also features an extensive Standard Library, which provides built-in ready-made classes enabling commonly desired functionality, such as output printing or math functions.

Several compilers are widely available for C++. For this research, the RCS code was compiled using two different platforms. The RCS code was initially compiled using
Microsoft Visual C++ 2008 Express Edition in Microsoft Windows XP, a commercial independent development environment (IDE) created by Microsoft. The IDE includes the ability to debug as well as develop code. Apple’s Xcode was used in Mac OS X. Similar to Visual C++, Xcode is an IDE with development and debugging capabilities. Xcode also includes a modified version of GNU Compiler Collection (GCC), which also has the capability to compile C++. GCC was developed for the GNU Project, which is a free-software, mass-collaboration project with the goal to create efficient, functional and free software that is compatible with a wide variety of other software. For this research, the RCS code was largely run from the Terminal in OS X utilizing GCC as the compiler, as this provided the cleanest, most transparent build.

3.4 MELCOR Model

For this research, a MELCOR model was created to examine aerosol transport within containment and release from containment during accident scenarios (Appendix B). MELCOR permits user-defined control volumes, flow paths and heat structures, allowing for the creation of a model which closely resembles the SFR layout. A simplified, two-volume model was created which consists of the containment, the environment, their associated heat structures and flow paths, and a heat structure simulating the sodium pool (Figure 3.1).
This model includes two control volumes, one representing the containment and the other representing the environment. The containment is 50.6 m tall with a volume of $2.86 \times 10^4$ m$^3$. Thermodynamic initial conditions include initialization at non-equilibrium, fog enabled (meaning a mixture of pool and atmosphere may develop) and the state within containment to be calculated based on the conservation equations. Initially, only atmosphere is present, and is initialized in the superheated state. The control volume representing the environment has a height of 70.0 m and a total volume of $1 \times 10^4$, however it should be noted that the total volume of the environment is arbitrary. Thermodynamic initial conditions for the environment include a non-equilibrium state.
initialization, fog enabled and a time-independent state. Similar to the containment, only atmosphere is present, and is superheated. Initial conditions for containment and the environment are a total pressure of 1.0 atm and an atmosphere temperature at room temperature, 300 K. Air composition for both the containment and environment is initialized as 80% nitrogen and 20% oxygen.

Three heat structures are associated with the containment. The containment walls are initialized at steady-state and have a cylindrical geometry with vertical orientation. The heat structure is meshed as concrete with five nodes. Heat transfer is set to be external, with coefficients set to be calculated by the heat structure (HS) package in MELCOR. The inner surface is convective to the containment control volume, while the outer surface is convective to the environment. The containment ceiling is represented by a separate heat structure, and is initialized at steady-state and has a rectangular geometry with horizontal orientation. Similar to the containment walls, the ceiling is meshed as concrete and includes five nodes. The inner surface convects to the containment control volume while the outer surface is convective to the environment, with external heat transfer and coefficients calculated by the HS package in MELCOR. The containment walls and ceiling both have a thickness of 2.0 m. A heat structure representing both the deck above the sodium pool and the containment floor is also included. Properties of the containment floor/deck, including dimensions, initialization state, meshing and inside boundary condition match those of the containment ceiling. However, the outside boundary of the floor/deck is insulated, and the floor/deck has a thickness of 15.24 cm.
All areas and lengths of the heat structures associated with containment match the areas and lengths of the containment control volume.

The sodium pool is represented as a heat structure located 0.02 m below the containment control volume. The area of the heat structure is 153.94 m$^2$, representative of the area of the pool surface. The characteristic length, axial length and thickness of the heat slab are 7.0 m, 14.0 m and 0.02 m, respectively. The boundary condition of the bottom surface of the heat structure is represented by a time-dependent tabular function (TF). This TF represents the pool temperature as a function of time, a relationship calculated using the RCS code. The top surface of the heat slab is convective to the containment control volume, with boundary coefficients calculated by the MELCOR HS package. Both the bottom and top surfaces of the heat structure are initialized with a boundary condition of internal flow across the boundary.

An additional heat structure was created to represent the environment floor. The primary purpose of this heat structure is for deposition, as each control volume must have an associated horizontal heat structure for deposition purposes. This heat structure is located at the reference axis (0 m) and mirrors the physical properties of the sodium pool heat structure. The primary differences are the boundary conditions of the heat structure; the top surface is convective to the environment, while the bottom surface is an insulated boundary.
One flow path was created to represent flow exiting containment. The flow path is located at a height of 25.0 m and has a horizontal orientation. Sample runs were performed in MELCOR to calculate the appropriate hole size which represents a 1.0 v/o per day leak rate from containment. The containment was set to a pressure of 1.0 atm gauge and a temperature of 373 K; energy sources, such as the heat from the sodium pool or the energy from sodium vaporization, were removed from the model for the leak rate tests. Flow rates exiting containment were observed for various hole sizes, and the appropriate hole size was chosen. The area of the flow path was found to be 3.862 x 10⁻⁵ m² with a length of 1.0 m, corresponding to a 1.0 v/o per day leak rate from containment. The form loss coefficients for this flow path, representing the form loss pressure drop across the flow path, are 9.0 for both forward and reverse flow. The choked flow discharge coefficients for this flow path are 1000 for both forward and reverse flow.

The RN package is used to insert radionuclides into the model. Radionuclides are input to containment as either a fission product vapor mass or as an aerosol mass. The aerosols are defined as having a lognormal size distribution with a mass median diameter and 2.26 µm and a standard deviation of 2. By default, MELCOR bounds aerosol size distributions by 0.1 µm and 50 µm; based on these boundaries, a lognormal distribution was constructed consisting of 10 bins for the mass median diameter and masses for all radionuclides were binned appropriately. For the cases studied in this thesis, all releases in the model occurred instantaneously at problem time zero, with the exception of sodium, which is inserted into containment using a time-dependent tabular function when
appropriate. Radionuclide masses were calculated from an ORIGEN run based on metallic fuel operated on a fast spectrum.

The RN package is also used to activate or deactivate heat structures as deposition surfaces. Since neither significant thermophoresis nor diffusiophoresis is expected in the MELCOR calculations, both sides of the containment walls, both sides of the containment ceiling, the bottom surface of the sodium pool heat structure and the bottom surface of the environment floor heat structure are deactivated as deposition surfaces. The top surface of the sodium pool heat structure was also deactivated as a deposition surface because it is expected that minimal deposition will occur on the pool surface as it is shielded by the deck structure. Only two surfaces are available for deposition: the top sides of the containment floor and environment floor heat structures.

For cases involving sodium, a control function is included in the model to account for the energy input to containment from sodium reacting with oxygen and the sensible heat from vaporization. Given that the specific heat, \( c_v \), of sodium is 1.23 kJ kg\(^{-1}\) K\(^{-1}\) and the pool temperature is 780 K, the sensible heat of sodium due to sodium thermally equilibrating with the containment atmosphere is calculated to be 16.4 kW. Two sodium-oxygen reactions are accounted for in this model: the formation of sodium oxide (\( \text{Na}_2\text{O} \)) and sodium peroxide (\( \text{Na}_2\text{O}_2 \)). Preliminary calculations show that for a sodium vaporization rate of 100 kg hr\(^{-1}\), if only the peroxide reaction is considered all oxygen will be consumed within 27 hours. To simplify the model, for cases with sodium
vaporization the peroxide reaction is assumed to dominate for the first 20 hours of the simulation; the oxide reaction then dominates until the oxygen is consumed. Given that the standard enthalpy of formation ($\Delta H_f^\circ$) of Na$_2$O is $-414.2$ kJ mol$^{-1}$ and the $\Delta H_f^\circ$ of Na$_2$O$_2$ is $-513$ kJ mol$^{-1}$, the heat input from the vaporization of 100 kg hr$^{-1}$ of sodium is 500 kW for the Na$_2$O reaction and 620 kW for the Na$_2$O$_2$ reaction [26]. For Cases 1 and 2, where masses of sodium are released over a duration of 0.5 hr, the heat input to containment is 62 MW and 0.64 MW for Case 1 and Case 2, respectively.

### 3.5 WinMACCS Model

A WinMACCS model was created to examine offsite consequences and compare these results with the QHOs. Physical details of the plant are not modeled in WinMACCS; only the weather, population data and source term characteristics are modeled in WinMACCS. Emergency response actions, such as evacuation, protective actions, and a KI model (simulating the administering of potassium-iodide tablets for the purpose of preventing radioactive iodine uptake) are available for modeling in WinMACCS, but were not utilized in this research.

Meteorological data, including wind speed, wind direction, rainfall and stability class, are supplied by a data file based on actual weather data from an Eastern U.S. site. The meteorological file consists of one year of data with meteorological conditions for each hour of the day. For the calculations, a stratified random sampling method was
used. Each day of the year was divided into four equal time periods and a random weather sequence was selected from each partition for each day. Each weather sequence was considered to have the same probability of occurrence. This sampling method lowers the variance in the entire population of data [27]; each day is divided into four eight-hour segments (i.e. morning or evening) in which meteorological conditions are more likely to be similar.

WinMACCS utilizes a Gaussian plume model which is fairly simple and computationally efficient. Gaussian models assume that the dispersion of a plume may be modeled as a random walk, generating a plume concentration with a normal distribution in all directions. When the plume is not limited by the ground or inversion layer, the cloud concentration is represented by:

\[
\chi(x, y, z) = \frac{Q}{2\pi \bar{u} \sigma_y \sigma_z} e^{-\frac{1}{2} \left( \frac{y}{\sigma_y} \right)^2} e^{-\frac{1}{2} \left( \frac{z-h}{\sigma_z} \right)^2}
\]

(3.26)

where:

\(\chi(x, y, z)\) = time-integrated cloud concentration as a function of downwind distance (Bq-s m\(^3\)),

\(Q\) = activity (Bq),

\(\bar{u}\) = mean wind speed (m s\(^{-1}\)),

\(\sigma_y\) = lateral plume spread as a function of atmospheric stability (m),
$\sigma_z = \text{vertical plume spread as a function of atmospheric stability (m),}$

$(x,y,z) = \text{downwind distance (m),}$

$h = \text{release height (m).}$

For the case where vertical plume expansion has caused plume capping, the ground and inversion layer must be treated as reflective boundaries. Identical sources are added above the mixing layer and below the ground level to model the reflective boundaries:

$$\chi(x,y,z) = \frac{Q}{2\pi\bar{u}\sigma_y\sigma_z} \left[ e^{-\frac{1}{2} \left( \frac{z-H}{\sigma_z} \right)^2} + e^{-\frac{1}{2} \left( \frac{z+H}{\sigma_z} \right)^2} + e^{-\frac{1}{2} \left( \frac{z+H+2nL}{\sigma_z} \right)^2} + e^{-\frac{1}{2} \left( \frac{z+H-2nL}{\sigma_z} \right)^2} \right]$$

where:

$\chi(x,y,z) = \text{time-integrated cloud concentration as a function of downwind distance (Bq-s m}^3),$

$Q = \text{activity (Bq),}$

$\bar{u} = \text{mean wind speed (m s}^{-1}),$

$\sigma_y = \text{lateral plume spread as a function of atmospheric stability (m),}$

$\sigma_z = \text{vertical plume spread as a function of atmospheric stability (m),}$

$(x,y,z) = \text{downwind distance (m),}$
H = h + Δh = plume centerline height (m),
Δh = plume rise (m),
h = release height (m),
L = inversion layer height (m).

Atmospheric dispersion is a result of atmospheric turbulence causing plume expansion in all directions; the rate of plume expansion increases with turbulence. Vertical expansion is affected by surface roughness and limited by the ground and inversion layer, while lateral expansion is unconstrained. Plume dispersion coefficients \( \sigma_y \) and \( \sigma_z \), were calculated by WinMACCS using lookup tables based on Pasquill-Gifford (PG) coefficients, and are a function of weather stability. These dispersion coefficients describe the lateral and vertical spread of the Gaussian-modeled plume. While the PG coefficients may typically be referenced from a figure, Tadmor and Gur [28] have developed analytic expressions representing the lateral and vertical plume spread:

\[
\sigma_{yi} = a_i x^{b_i} \\
\sigma_{zi} = c_i x^{d_i}
\]  

where:
subscript \( i \) denotes the PG stability class,
constants \( a_i, b_i, c_i, d_i \) are referenced in Table 3.1.
Table 3.1: Constants for analytic expressions representing plume spread.

<table>
<thead>
<tr>
<th>Stability Class</th>
<th>Constant</th>
<th>Constant</th>
<th>Constant</th>
<th>Constant</th>
</tr>
</thead>
<tbody>
<tr>
<td>PG</td>
<td>i</td>
<td>a_i</td>
<td>b_i</td>
<td>c_i</td>
</tr>
<tr>
<td>A</td>
<td>1</td>
<td>0.3658</td>
<td>0.9031</td>
<td>0.00025</td>
</tr>
<tr>
<td>B</td>
<td>2</td>
<td>0.2751</td>
<td>0.9031</td>
<td>0.0019</td>
</tr>
<tr>
<td>C</td>
<td>3</td>
<td>0.2089</td>
<td>0.9031</td>
<td>0.2</td>
</tr>
<tr>
<td>D</td>
<td>4</td>
<td>0.1474</td>
<td>0.9031</td>
<td>0.3</td>
</tr>
<tr>
<td>E</td>
<td>5</td>
<td>0.1046</td>
<td>0.9031</td>
<td>0.4</td>
</tr>
<tr>
<td>F</td>
<td>6</td>
<td>0.0722</td>
<td>0.9031</td>
<td>0.2</td>
</tr>
</tbody>
</table>

In WinMACCS, plume meander is based on guidance provided by Regulatory Guide 1.145. Default coefficients supplied by WinMACCS were used in all calculations. These values include: (a) the wind speed where the meander factor changes from a constant value to a linearly decreasing function of wind speed (2 m/s), (b) wind speed where meander factor reaches unity (6 m/s), (c) downwind distance where the effect of meander begins to diminish (800 m), (d) and the meander factor used to calculate correction of \( \sigma_y \), where the factors correspond to the atmospheric stability classes A through F, respectively (1, 1, 1, 2, 3, 4). Plume rise occurs when the sensible heat of the plume exceeds a limiting buoyancy criterion; this rise is inhibited however when the prevailing wind speed exceeds the liftoff criterion, and is calculated in WinMACCS using the Brigg’s model. For this work, plume heat was set to zero to negate plume rise and mixing. Building wake effects were also negated by setting the building height to the minimum value of 1 m. Plumes for the bounding case were modeled as a puff; the entire inventory is assumed to be released over a duration of 60 s with no delay (i.e. the inventory is released at problem time zero).
Wet deposition may be calculated by WinMACCS; this deposition is dependent on rain duration and rain intensity. The fraction of radionuclides remaining in the plume is based on the washout model and is represented by the equation:

\[ f = e^{-c_1 r_d + c_2 r_i} \]  

(3.30)

where:

- \( f \) = fraction of radionuclides remaining in plume,
- \( c_1 \) = linear coefficient (s\(^{-1}\)),
- \( c_2 \) = exponential coefficient,
- \( r_d \) = rain duration (s),
- \( r_i \) = rain intensity (mm hr\(^{-1}\)).

Dry deposition is calculated in WinMACCS using the source depletion method [29]. The rate at which materials are deposited onto the ground is calculated as a product of the integrated ground level cloud concentration and the deposition velocity, \( v_d \). WinMACCS simplifies this model by assuming that deposition does not distort the plume shape and the distribution remains Gaussian. In this model, default values supplied by WinMACCS were used to model wet and dry deposition. The following values were used: \( c_1 = 9.5 \times 10^{-3} \text{ s}^{-1} \), \( c_2 = 0.8 \), \( v_d = 0.01 \text{ m/s} \). All radionuclide groups, excepting noble gases, were subject to dry deposition.
WinMACCS allows user-defined radionuclide groups (Table 3.2) enabling variables such as release fractions, dry deposition velocities and particle size distributions to be defined for each group. To simplify this model, a particle size distribution was not assigned to the groups. All particles within a radionuclide group are assigned to a single distribution group.

Table 3.2: Radionuclide groups and their respective elements.

<table>
<thead>
<tr>
<th>Group Name</th>
<th>Element</th>
</tr>
</thead>
<tbody>
<tr>
<td>Noble Gases</td>
<td>Kr, Xe</td>
</tr>
<tr>
<td>Iodine &amp; Cesium</td>
<td>I, Cs</td>
</tr>
<tr>
<td>All Others</td>
<td>Am, Ba, La, Ru, Sr, Te, U, Pu, Zr</td>
</tr>
<tr>
<td>Sodium</td>
<td>Na</td>
</tr>
</tbody>
</table>

The site model used in this research consists of population data for the Eastern U.S. site from which meteorological data was used. This site data consists of 26 spatial regions radiating from the plant site. Spatial intervals range from 0.16 km to $1.6 \times 10^3$ km. This site data also contains watershed, crop and economic information for the use of food contamination models and economic impact; while WinMACCS has the capability to calculate the impacts of these factors in response to radionuclide release from a site, these calculations were not performed.
The WinMACCS model uses the Linear No Threshold (LNT) dose model. The LNT model assumes no minimum exposure below which linear effects are not applicable. This is in contrast to other models which assume a harmless minimum threshold of exposure. Dose conversion factors are based on Federal Guidance Report (FGR) No. 13, Cancer Risk Coefficients for Environmental Exposure to Radionuclides [28]. This database includes dose and risk coefficients for all significant organs as well as a whole body conversion factor. Coefficients are also a function of the exposure pathway: external (i.e. submersion), ingestion and inhalation. Exposure pathways in WinMACCS utilized in this research include external (cloudshine, groundshine, skin deposition) and inhalation. It was also assumed that the entire population was equally susceptible to early effects and latent cancers.

3.6 Dose Calculations

To assess the offsite consequences of the accident scenarios analyzed in this research, the whole body dose is calculated as the total effective dose equivalent (TEDE), which includes the committed effective dose equivalent (CEDE) resulting from internal sources and the deep dose equivalent (DDE) resulting from external sources. All radionuclides are assumed to contribute to both the CEDE and DDE, with the exception of noble gases. Since noble gases are released as fission product vapors, they contribute only to the DDE and therefore only have a submersion dose component; while fission product vapors are respirable, the noble gases do not react with the tissue in the
respiratory system. For this reason, they are not retained in the body and do not contribute to the internal (inhalation) dose.

For this research, the submersion dose and inhalation dose from isotope \( n \) are calculated as described by the following equations:

\[
D_{Sub,n} = Q_n \cdot DCF_{Sub,n} \cdot \frac{X}{Q}
\]  
(3.31)

where:

\( D_{Sub,n} \) = submersion dose from isotope \( n \) (rem),

\( Q_n \) = inventory of isotope \( n \) (Ci),

\( DCF_{Sub,n} \) = submersion dose conversion factor (rem m\(^3\) Ci\(^{-1}\) s\(^{-1}\)),

\( \frac{X}{Q} \) = cloud concentration representative of the meteorology and dose location (s m\(^{-3}\)).

\[
D_{Inh,n} = Q_n \cdot DCF_{Inh,n} \cdot \frac{X}{Q} \cdot R
\]  
(3.32)

where:

\( D_{Inh,n} \) = inhalation dose from isotope \( n \) (rem),

\( Q_n \) = inventory of isotope \( n \) (Ci),

\( DCF_{Inh,n} \) = inhalation dose conversion factor for isotope \( n \) (rem/Ci),
\( \frac{Z}{Q} \) = cloud concentration representative of the meteorology and dose location (s m\(^{-3}\)),

\( R \) = volumetric breathing rate (m\(^3\) s\(^{-1}\)).

All dose conversion factors were taken from FGR No. 13 for whole body dose. In the case of the inhalation dose, the dose conversion factors for particulate type F (i.e. fast dissolution and high level of absorption to blood) were used to achieve conservative results. The volumetric breathing rate, \( R \), for all calculations is 3.47 x 10\(^{-4}\) m\(^3\) s\(^{-1}\).

For cases where containment is failed, the amount released to the containment is assumed to be released to the environment in its entirety. Isotopes are decayed to 0.1 hr and then dose resulting from exposure to the isotopes is calculated:

\[
Q_a (0.1\ hr) = Q_i,a e^{-\lambda_i (0.1\ hr)}
\]  \( (3.33) \)

For cases where containment remains intact, results from MELCOR are used to calculate release fractions as a function of time. The released inventory is then decayed as a function of time:

\[
Q_a (t) = F_a (t) Q_i,a e^{-\lambda_i t}
\]  \( (3.34) \)

where:
\( Q_n(t) \) = inventory of isotope \( n \) at time \( t \) (Ci),

\( F_n(t) \) = release fraction of isotope \( n \) at time \( t \),

\( Q_{i,n} \) = initial inventory in containment of isotope \( n \) (Ci),

\( \lambda_n \) = decay constant for isotope \( n \) (s\(^{-1}\)),

\( t \) = time (s).
CHAPTER 4:

RESULTS AND ANALYSES

4.1 Determination of 95th Percentile Meteorology

Given meteorological data for an Eastern U.S. site, the cloud concentration representing the 95th percentile meteorology was calculated. Cloud concentration values, \( \chi/Q \), are calculated at the site boundary (1.0 km) and 1.0 mile from the site boundary (2.6 km). Using the methodology described in Section 3.1, a complementary cumulative distribution function was constructed and the cloud concentration exceeded 5% of the total data set was found. The \( \chi/Q \) for 1.0 km was found to be \( 9.614 \times 10^{-4} \) s m\(^{-3} \), the \( \chi/Q \) for 2.6 km was found to be \( 2.282 \times 10^{-4} \) s m\(^{-3} \).
Figure 4.1: CCDF representing cloud concentration exceedence for 1.0 km.

Figure 4.2: CCDF representing cloud concentration exceedence for 2.6 km.
4.2 Identification of Key Severe Accident Phenomena

Several mechanisms have been identified which are expected to affect severe accident progression and the subsequent radionuclide release. Failure of the four primary barriers in the SFR are required for a radioactive release from the core; these barriers include the fuel matrix, fuel cladding, the primary system and containment. It is assumed that sodium in the primary system is activated from travel through the core. Therefore, there are only two barriers preventing sodium release, where sodium would contribute to the source term. It should be recognized however that the sodium pool acts as a barrier in that most radionuclides of significance in offsite consequences tend to be dissolved in, dispersed in, or reacted with the sodium.

As discussed in Section 3, the behavior of oxide fuels under severe accident conditions has been studied more extensively for oxide than metallic fuels. Because of the low temperature at which melting occurs and the chemical forms of the radionuclides, the release of radionuclides from molten metallic fuel is expected to be comparatively small. However, fuel oxidation could significantly increase the release of radioactive material. An oxidizing environment may be created by either a loss of coolant accident in which both the vessel and guard vessel fail, or by a total loss of heat sink which includes the loss of the passive heat sink. The likelihood of these conditions is quite small, however.
The location of fuel failure, i.e. middle or top of pin, will have a significant impact on the potential for an HCDA. Such a hypothetical energetic event could potentially occur if after the fuel melts, it moves toward a more compact, criticality-favorable geometry. SAS4A analyses indicate that prior to clad failure, the motion of the fuel away from the center of the core results in a decrease in reactivity. If fuel failure occurred near the mid-height of the core, fuel motion could lead to a reactivity increase. Furthermore, if fuel failure occurs in such a location that molten fuel has the opportunity to collect, an energetic event could occur, potentially leading to primary system or even containment failure. It should be noted that that conditions leading to a HCDA are expected to have a very low frequency of occurrence.

If fuel debris accumulates on the lower head of the vessel and is uncoolable, the debris could remelt and lead to the failure of the primary or guard vessel. This would allow the molten fuel to interact with the concrete of the containment floor. This would lead to an oxidizing environment where releases would be significant. Primary vessel failure may also result from an energetic event or creep rupture following loss of heat removal and overheating of the structure. In the case of an energetic event or loss of heat sink, releases are also expected to be substantial.

Failure of the primary system is of significant concern when the failure occurs in the cover gas region. This cover gas is expected to retain all noble gases released from the fuel and also will contain any other radionuclides which were not dissolved in the
sodium pool or did not interact with sodium. Failure at the upper primary system boundary may result from heating and pressurization by noble gases and any other vapors present in the region.

Over-pressurization of the cover gas region would lead to a release to the containment, creating a pathway for radionuclide transport and the transfer of heat. This may result in over-heating or over-pressurization of containment. Failure of containment may also result from a bypass event or a failure to isolate the containment. Extremely energetic events could result in missile generation or a spray fire that could threaten containment integrity. External threats, such as aircraft crashes or seismic events, could result in containment failure and initiate conditions leading to severe core damage.

4.3 Identification of Generic Severe Accident Scenarios

To examine the behavior of the SFR during accident sequences and to analyze offsite consequences, thirteen scenarios have been identified. These accident scenarios represent a credible spectrum of events that could result in offsite consequences. These accident scenarios do not include a thorough mechanistic analysis with uncertainty calculations, which would be required to demonstrate compliance with a performance-based metric. Instead, these scenarios represent one product of the methodology being demonstrated; this methodology is the objective of this research and the DOE-NERI project which is following the TNF’s generic guidelines. Best estimate assumptions have
been used to generate realistic source terms associated with these scenarios. If the primary system and containment system remain intact in a scenario, the offsite consequences would be extremely small regardless of the extent of fuel damage. Thus, in all of the analyzed scenarios, it is assumed that the primary system fails. Consequences are then produced for scenarios in which the containment remains intact and for scenarios in which the containment is also failed, with the exception of the sustained sodium vaporization case.

The first case represents an unmitigated failure of the primary system in which a large leak and sodium spray fire occurs; 5,000 kg of primary sodium is released to containment in a fire and containment is intact at accident initiation. Case 2 represents a smaller sodium release characteristic of a release from a primary coolant sampling system; 50 kg of sodium is released to an intact containment in this scenario. For the purpose of modeling in MELCOR, the releases in Cases 1 and 2 occur over a period of 0.5 hr. Case 3 and Case 4 are identical to Case 1 and Case 2, except that containment is failed (Table 4.1).
Table 4.1: Accident scenarios and release masses for instantaneous releases of sodium.

<table>
<thead>
<tr>
<th>Case #</th>
<th>Description</th>
<th>Containment</th>
<th>Element/Group</th>
<th>Release Mass</th>
<th>Calculation Type</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Large Na Release</td>
<td>Intact</td>
<td>Na</td>
<td>5,000 kg</td>
<td>MELCOR R.G. 1.145</td>
</tr>
<tr>
<td>2</td>
<td>Small Na Release</td>
<td>Intact</td>
<td>Na</td>
<td>50 kg</td>
<td>MELCOR R.G. 1.145</td>
</tr>
<tr>
<td>3</td>
<td>Large Na Release</td>
<td>Failed</td>
<td>Na</td>
<td>5,000 kg</td>
<td>R.G. 1.145</td>
</tr>
<tr>
<td>4</td>
<td>Small Na Release</td>
<td>Failed</td>
<td>Na</td>
<td>50 kg</td>
<td>R.G. 1.145</td>
</tr>
</tbody>
</table>

The fifth case represents accidents in which 30% fuel failure occurs but melting is arrested. No credit is taken for sodium scrubbing and there is no extended period of release of iodine or cesium from the pool. In case 5, the primary system is failed, and sodium is vaporized at a rate of 100 kg/hr. The noble gases, cesium and iodine are released from the cover gas region into the intact containment. Case 6 is identical to case 5, except that there is no sustained release of sodium vapor from the pool. This is a sensitivity study to examine the effects of sodium vaporization from the pool. Case 7 and Case 8 are identical to Case 5 and Case 6, respectively, except containment is failed (Table 4.2).
Table 4.2 : Severe accident scenarios and the appropriate release fractions and rates for scenarios involving arrested melt progression.

<table>
<thead>
<tr>
<th>Case #</th>
<th>Description</th>
<th>Containment</th>
<th>Element/Group</th>
<th>Release Fraction/Rate</th>
<th>Calculation Type</th>
</tr>
</thead>
<tbody>
<tr>
<td>5</td>
<td>Arrested Melt - 30% Fuel Failure</td>
<td>Intact</td>
<td>Na</td>
<td>100 kg/hr</td>
<td>MELCOR R.G. 1.145</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Noble gases</td>
<td>0.3</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>1 x 10^{-4}</td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>Arrested Melt - 30% Fuel Failure</td>
<td>Intact</td>
<td>Noble gases</td>
<td>0.3</td>
<td>MELCOR R.G. 1.145</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>1 x 10^{-4}</td>
<td></td>
</tr>
<tr>
<td>7</td>
<td>Arrested Melt - 30% Fuel Failure</td>
<td>Failed</td>
<td>Na</td>
<td>100 kg/hr</td>
<td>R.G. 1.145</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Noble gases</td>
<td>0.3</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>1 x 10^{-4}</td>
<td></td>
</tr>
<tr>
<td>8</td>
<td>Arrested Melt - 30% Fuel Failure</td>
<td>Failed</td>
<td>Noble gases</td>
<td>0.3</td>
<td>R.G. 1.145</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>1 x 10^{-4}</td>
<td></td>
</tr>
</tbody>
</table>

The ninth case represents an energetic event with 63% fuel failure and primary system failure that does not fail containment. In addition to the radionuclides released in the previous scenarios, there is an additional release of volatiles and actinides as a result of fuel vaporization. The cesium and iodine release increases relative to the previous cases because of the more extensive fuel damage, higher fuel temperatures and likelihood that the vapor bubble formed is so large that pool scrubbing has no effect. Case 10 is identical to Case 9 except containment is failed. Case 11 represents a core uncovery scenario where 63% fuel failure occurs with primary system failure; containment remains...
intact. Based on oxidation analyses, the source term is largely affected by the conditions associated with the core uncovery. In this case, longer durations at higher temperatures enhance the cesium and iodine component of the source term. Case 12 is identical to Case 11, except containment is failed. Case 13 represents the failure of the primary system in which sustained sodium vaporization occurs with containment intact. No fuel failure occurs, so sodium is the only contributor to the source term (Table 4.3).
Table 4.3: Severe accident scenarios and appropriate release fractions and rates for accident scenarios involving energetic events and core uncovery.

<table>
<thead>
<tr>
<th>Case #</th>
<th>Description</th>
<th>Containment</th>
<th>Element/Group</th>
<th>Release Fraction/Rate</th>
<th>Calculation Type</th>
</tr>
</thead>
<tbody>
<tr>
<td>9</td>
<td>Energetic Event - Fuel Vaporization</td>
<td>Intact</td>
<td>Na</td>
<td>100 kg/hr</td>
<td>0.63</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Noble gases</td>
<td></td>
<td>MELCOR</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>3 x 10^4</td>
<td>R.G. 1.145</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>All Others</td>
<td>1 x 10^5</td>
<td></td>
</tr>
<tr>
<td>10</td>
<td>Energetic Event - Fuel Vaporization</td>
<td>Failed</td>
<td>Na</td>
<td>100 kg/hr</td>
<td>0.63</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Noble gases</td>
<td></td>
<td>R.G. 1.145</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>3 x 10^4</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>All Others</td>
<td>1 x 10^5</td>
<td></td>
</tr>
<tr>
<td>11</td>
<td>Core Uncovery</td>
<td>Intact</td>
<td>Na</td>
<td>100 kg/hr</td>
<td>0.63</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Noble gases</td>
<td></td>
<td>MELCOR</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>5 x 10^4</td>
<td>R.G. 1.145</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>All Others</td>
<td>1 x 10^5</td>
<td></td>
</tr>
<tr>
<td>12</td>
<td>Core Uncovery</td>
<td>Failed</td>
<td>Na</td>
<td>100 kg/hr</td>
<td>0.63</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Noble gases</td>
<td></td>
<td>R.G. 1.145</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>5 x 10^4</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>All Others</td>
<td>1 x 10^5</td>
<td></td>
</tr>
<tr>
<td>13</td>
<td>Sustained Na Vaporization</td>
<td>Intact</td>
<td>Na</td>
<td>100 kg/hr</td>
<td></td>
</tr>
</tbody>
</table>
4.4 RCS Validation Results

Before the RCS code could be used for this research, results provided by the code required validation. To achieve this, two separate comparisons were made. A stand-alone kinetics version of the code was run for two reactivity insertion cases and the resulting period was observed. These reactor period results were then compared with tabulated data from Keepin [23], from which the kinetics equations for the RCS code were derived. The full version of the code was then run for two different cases of reactivity insertion. Pool temperature and core power results were then compared with SAS4A results from identical calculations.

4.4.1 Kinetics Validation Results

The effects of reactor kinetics were taken into account in the RCS code by using the reactor kinetic equations. These equations are numerically challenging to solve because of the difference in stiffness of the prompt neutron behavior and delayed neutron release. For this work, these kinetic equations were transformed into an integral solution developed by Keepin which was utilized numerically. In order to assure that the form of the kinetics solution incorporated into the RCS code is correct and consistent with the SAS4A code, validation calculations were performed.
To validate the neutronics component of the code, the kinetics portion was isolated and two runs were performed. In these runs, a specific amount of reactivity was inserted and the resulting period was observed. These periods were then compared to data supplied by Keepin for agreement.

In the first run, 30 cents of reactivity was inserted into the reactor. For the second run, 60 cents of reactivity was inserted. No feedback mechanisms were modeled. For both calculations, reactivity was inserted over a time period of 5 s and was not withdrawn once fully inserted. For a 30 cent insertion, a reactor period of approximately 20 s was expected, where for a 60 cent insertion, a 3.5 s period was expected. Results show fairly good agreement for the 30 cent case, where a period of approximately 20.5 s was observed (Figure 4.3) after the initial power increase due to the prompt jump in neutron population at approximately 0.5 s. For the case of 60 cents of inserted reactivity, results show very good agreement, where a period of approximately 3.7 s was observed (Figure 4.4).
Figure 4.3: Core power as a function of time for a 30 cent reactivity insertion.

Figure 4.4: Core power as a function of time for a 60 cent reactivity insertion.
4.4.2 SAS4A Validation Results

In the RCS code, pool temperature is calculated by performing an energy balance between pool volumes. This energy balance is based on fundamental concepts, but the code’s response to transients still required validation. To accomplish this, two cases were analyzed using the RCS code and the results were compared with results from SAS4A for two identical cases. The first case involves a 1 cent per second insertion over 40 s and the second case involves a 2 cent per second insertion over 60 s; in both cases, the reactivity was not removed, representing an unexpected reactivity insertion such as control rod withdrawal. For both the RCS code and SAS4A, core power is initially 1000 MW and the IHX is set to remove a constant 1000 MW from the primary system.

For the first case, the results for upper and lower pool asymptotic temperatures show fairly good agreement (Figure 4.5). The RCS under predicted the pool temperatures by 0.7%. Results for normalized core power show that the RCS code over predicted core power by approximately 4% (Figure 4.6). Disagreement between the RCS code and SAS4A arises from the fact that SAS4A separately considers multiple feedback mechanisms including Doppler feedback and axial and radial expansion of the fuel, while the RCS code relies solely on the point kinetics equations and its inherent consideration of lumped feedback. Peak normalized core power in the RCS code occurs approximately 20 s after the reactivity insertion is completed, whereas in SAS4A, the peak in core
power occurs at the time of completion of the reactivity insertion. In accordance with this, there is also disagreement among the codes on the net reactivity, but overall the reactivity transients tend to follow the same trends (Figure 4.7).

An additional peak in net reactivity is observed in the SAS4A results. In SAS4A, there is a significant drop in net reactivity after 40 s since the fuel and coolant are still hot, representing the effects of Doppler feedback and fuel expansion on reactivity. However, the fuel tends to cool off fairly quickly compared to the sodium, thereby reducing the negative reactivity caused by Doppler feedback and fuel expansion. At approximately 50 s, the positive reactivity insertion caused by the hot coolant exceeds the negative reactivity resulting from phenomena caused by the cooling fuel.

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1 In work performed subsequent to this thesis, the net reactivity from SAS4A was introduced into RCS to examine the consistency of the kinetics and heat balances. The agreement was found to be very good.
Figure 4.5: Pool temperatures versus time for the RCS code and SAS4A for the case of a reactivity insertion of 1.0 cent per second over 40 s.

Figure 4.6: Normalized core power as a function of time for the case of a reactivity insertion of 1.0 cent per second over 40 s.
Results from both the RCS code and SAS4A show fairly good agreement for the second case of a 2.0 cent per second insertion over 60 s. The RCS code again under predicts the upper and lower pool temperatures by approximately 1.5% in this case (Figure 4.8). The over prediction of peak core power in the RCS code is larger in this case; peak normalized core power in the RCS code is 13% larger than the core power calculated by SAS4A (Figure 4.9). Similarly, there is significant difference in the amount of net reactivity inserted, but the results follow the same overall trend (Figure 4.10). These disagreements are again the result of the respective code’s handling of feedback mechanisms. In this case, disagreement in results is exaggerated since this case relates to a larger reactivity insertion where the results are largely affected by fast-acting feedback mechanisms such as Doppler feedback and fuel expansion.
Figure 4.8: Plots of pool temperatures versus time for the case of 2.0 cent per second insertion over 60 s.

Figure 4.9: Normalized power as a function of time for the case of 2.0 cents per second insertion over 60 s.
4.5 MELCOR Results

The MELCOR model described in Section 3.4 was run for the generic severe accident scenarios with containment intact described in Section 4.3. Each case was run for a simulation time of 24 hours. Results from the code are plotted to show temperature, pressure and aerosol behavior over the entire period of the simulation. Generally, the results show that for cases with a constant vaporization rate of sodium, the temperature, pressure and aerosol behaviors are very similar among cases. All cases show tolerable peak containment pressure and structure temperatures that would not fail the boundary or compromise internal structures, with the exception of Case 1.

Figure 4.10: Net reactivity versus time for the case of a reactivity insertion of 2.0 cents per second over 60 s.
In Case 1, due to the large release of sodium over a short period of time and its heat input to containment, there is a significant peak in containment pressure of approximately 4.0 atm gauge, which is four times the design pressure. The typical conservatism in design pressure relative to failure pressure is somewhere between a factor of two and a factor of three. While the exact tolerance of the containment for the SFR would be highly design dependent, it is likely that an increase of this magnitude would lead to containment failure. The temperature of the atmosphere in containment also experiences a significant increase of 1100 K, with the maximum temperature reaching approximately 1400 K. Peak pressure and atmosphere temperature occur at 0.5 hr, which coincides with the end of sodium release to containment. After the peak at 0.5 hr, containment pressure and atmosphere temperature quickly return to their asymptotic values (Figure 4.11). Given that the containment volume is constant with a fairly small leak rate, results demonstrate, as expected, that the pressure increase in containment mirrors the temperature increase.

Given that the peak atmosphere temperature is not sustained for a significant period of time and that convection is the dominant heat transfer mechanism in this model, the majority of the structures in containment (considering those contained in this model) do not experience an increase in temperature which would compromise their integrity, with the exception of the deck surface (Figure 4.12). The deck structure in this model is an artificial structure used to represent the convective heat transfer from the open sodium pool surface to the containment atmosphere; it does not have a direct physical meaning.
The results of aerosol behavior, such as sodium deposition and agglomeration, for Case 1 are not shown. Due to the large mass of sodium released effectively instantaneously, the sodium aerosols agglomerate very quickly and therefore deposit before a significant amount can be released to the environment. More than half of the sodium deposits within 5.0 min and 99% is deposited within 2.0 hrs.

Figure 4.11: Containment atmosphere temperature and pressure for Case 1.
Figure 4.12: Containment structure temperatures for Case 1.

Results for Case 2 show a minor peak in containment atmosphere temperature and pressure. The peak atmosphere temperature in Case 2 is 338 K, and the peak pressure is approximately 1.13 atm; temperature and pressure increase by 38 K and 0.13 atm, respectively. As with Case 1, the peak in temperature and pressure occurs at 0.5 hr, which coincides with the end of sodium release (Figure 4.13). In this case however, it is unlikely that the pressure spike will fail containment. For Case 2, the structures in containment do not experience as dramatic of an initial increase in temperature as they do in Case 1. Where the deck underwent a 250 degree increase in temperature in approximately one-half hour in Case 1, the deck sees a four degree jump in temperature in one-half hour in Case 2 (Figure 4.14). Structure temperatures in containment continue to linearly increase over the 24-hour simulation at a rate of one-half degree per hour.
This steady increase in temperature is a result of the heat input to containment from the sodium pool which convects directly to the containment atmosphere. The diminished containment response in Case 2 relative to the results of Case 1 is a result of the reduced energy input to containment from sodium combustion; as expected, the smaller mass of released sodium has a lessened effect on containment behavior.

![Containment Temperature and Pressure - Case 2](image)

Figure 4.13: Containment atmosphere temperature and pressure for Case 2.
The results of aerosol behavior for Case 2 are shown to demonstrate sodium aerosol agglomeration and deposition. Compared to Case 1, deposition occurs much more slowly; however, by the end of the simulation, nearly all the sodium airborne in containment is deposited on the floor/deck due to agglomeration and gravitational settling (Figure 4.15).

The effects of agglomeration on the aerosol size distribution are shown in Figure 4.16. While the mass per section is not vital to the conclusion of this research, the overall agglomeration trend is of interest. Where Section 1 represents particles with the smallest diameter and Section 10 represents the largest, the results show that aerosols from
Section 1 quickly bond with other particles, increasing their diameter so that they are now accounted for in higher sections. Also, particles from the largest size groups (Section 9 and Section 10) quickly fall out and deposit onto the containment floor/deck. Sections in the middle of the size distribution see an increase in sodium mass as aerosols agglomerate and essentially “migrate” into higher sections.

Figure 4.15 : Sodium suspension and deposition for Case 2.
For the remaining cases, containment temperature and pressure results are largely similar. The only exception is with Case 6, in which no sodium is released to containment. For cases with a constant rate of sodium vaporization, the containment atmosphere temperature peaks at approximately 370 K at 20.0 hr, a 70 K increase; the sudden decrease in atmosphere temperature is a result of the decrease in heat of reaction as the sodium oxide reaction begins to dominate (Figure 4.17). For Case 6, which does not include the constant rate of sodium vaporization, the atmosphere temperature peaks at 330 K at the end of the simulation, only a 30 K increase; the increase in atmosphere temperature in this case is due to the convection of heat from the sodium pool surface. 
For Cases 5, 6, 9, 11 and 13, the increase in atmosphere temperature is not likely to challenge the integrity of any structures within containment.
Similar to the temperature results, the response of containment pressure for Cases 5, 9, 11 and 13 are identical. These cases all include a constant rate of sodium vaporization which adds an additional heat of reaction beyond the sensible heat from sodium. For these cases, containment pressure peaks at approximately 0.22 atm gauge at 20.0 hr. As with the temperature, the decrease in pressure at 20.0 hr is due to the decrease in the heat of reaction as the sodium oxide reaction begins to dominate at that time. For Case 6, containment pressure linearly increases throughout the simulation and peaks at approximately 0.1 atm, gauge. The results show that for Cases 5, 6, 9, 11 and 13 it is unlikely that these accident scenarios will fail containment and the driving force for leakage is substantially smaller than the design leak rate (based on the leak rate at 1.0 atm gauge).
Figure 4.17: Containment atmosphere temperature for scenarios with sodium vaporization.

Figure 4.18: Containment pressure for scenarios with sodium vaporization.
The effects of sodium on cesium and iodine deposition are shown in Figure 4.19 and Figure 4.20. Case 5 and Case 6 have identical noble gas, cesium and iodine releases; the only difference in these cases is the lack of the release of sodium in Case 6. For this reason, this makes these two cases ideal for the comparison of the effects of sodium on deposition. In Case 5, the entire cesium and iodine inventory within containment is deposited within 6 hours. Compared to Case 5, only 40% of the inventory initially in containment is deposited within 6 hours for Case 6. By the end of the simulation, the normalized deposition goes to only 50%.

This decrease in deposition for Case 6 means that more airborne radionuclides are available for release to the environment. Accordingly, Figure 4.20 shows that for Case 6, 0.2% of the inventory initially in containment is airborne in the environment and will contribute to the dose. Conversely, in Case 5, 0.05% of the cesium and iodine initially in containment is airborne in the environment; this peak fraction is achieved within 5.0 hours, as the majority of sodium within containment is deposited and is not available for release.
Figure 4.19: Cesium iodine suspension and deposition in containment for Case 5 and Case 6.

Figure 4.20: Cesium and iodine suspension and deposition in the environment for Case 5 and Case 6.
The sectional mass showing the size distribution of aerosols in containment for Case 5 and Case 6 is shown in Figure 4.21 and Figure 4.22, respectively. For Case 5, where sodium is present, the aerosols quickly agglomerate and deposit inside containment. This is in agreement with the suspension and deposition results shown in Figure 4.19, where the majority of the cesium and iodine within containment are deposited within 6 hours. Mass per group for Sections 4, 5, and 6 show peaks near 1 hour because aerosols from sections of smaller size (i.e. Sections 1, 2, and 3) are agglomerating, thereby increasing their diameter which changes their size group classification. These sections with peaks eventually decrease to approximately zero as the aerosols continue to agglomerate, “migrate” into larger sections and eventually deposit.

In contrast of Case 5, the sectional aerosol behavior within containment of Case 6, shown in Figure 4.22, is very different in the absence of sodium. The aerosols slowly agglomerate over the course of the entire simulation. Aerosols in groups with the largest diameters, i.e. Sections 8, 9 and 10, quickly fall out and deposit inside containment. Groups with medium to large diameters, i.e. Sections 5, 6, and 7, undergo a decrease in mass much more slowly than the largest two sections, due to the very slow rate of agglomeration in the absence of large aerosols. Aerosols in the smallest group, i.e. Section 1, also show a decrease in mass, albeit more slowly than the largest sections. The
smallest aerosols are agglomerating and forming aerosols with larger diameters. These growing aerosols contribute to the increase in mass exhibited in Sections 2 and 3.

Figure 4.21: Sectional mass of cesium and iodine for Case 5.
The sectional mass for sodium is shown in Figure 4.23; while the data were taken from results for Case 5, this aerosol behavior is characteristic of all cases with a constant rate of sodium vaporization. Within 5 hours, the mass per section of sodium goes to an asymptotic value and remains constant during the rest of the simulation. The constant mass per section after 5 hours shows that the sodium aerosols have reached an equilibrium state; while agglomeration and fallout are definitely occurring, sodium aerosols are being introduced to containment at such a rate that aerosols that do deposit are replaced effectively instantaneously.
Figure 4.23: Sectional mass of sodium for Case 5.

For the remaining cases, the sectional mass, suspension and deposition trends are not shown because the results are very similar to those of Case 5. The aerosol behavior for Cases 5, 9, 11 and 13 is effectively identical in those cases; the majority of radionuclides released from the fuel quickly deposit within containment before they can be released to the environment. Also, as stated, the behavior of sodium aerosols is effectively identical for Cases 5, 9, 11 and 13 where the constant rate of sodium vaporization is included in the model.

Numerical results for peak containment pressure, peak atmosphere temperature in containment, initial radionuclide masses in containment and fractional release to the environment are tabulated in Table 4.4. Peak containment pressure occurs in Case 1, the
large sodium spray fire scenario. Peak atmosphere temperature in containment also occurs in this case. In terms of fractional release, Case 5 exhibits the largest fractional release of the initial inventory, however, this does not guarantee that the largest dose will be observed in Case 5. The additional release of volatiles and actinides in Case 9 and Case 11 will prove to have a significant effect on the offsite dose.
Table 4.4: Numerical results for MELCOR simulations.

<table>
<thead>
<tr>
<th>Case #</th>
<th>Peak Pressure (atm)</th>
<th>Peak Temp. (K)</th>
<th>RN Group</th>
<th>Initial Inventory in Containment (kg)</th>
<th>Fraction of Inventory Released to Environment</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>4.80</td>
<td>1440</td>
<td>Na</td>
<td>5,000</td>
<td>9.95 x 10^{-5}</td>
</tr>
<tr>
<td>2</td>
<td>1.13</td>
<td>338</td>
<td>Na</td>
<td>50</td>
<td>9.53 x 10^{-4}</td>
</tr>
<tr>
<td>5</td>
<td>1.22</td>
<td>367</td>
<td>Noble gas</td>
<td>2.48 x 10^{-1}</td>
<td>5.52 x 10^{-3}</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>1.04 x 10^{-2}</td>
<td>4.51 x 10^{-4}</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Na</td>
<td>100 kg/hr</td>
<td>3.37 x 10^{-4}</td>
</tr>
<tr>
<td>6</td>
<td>1.09</td>
<td>328</td>
<td>Noble gas</td>
<td>2.48 x 10^{-1}</td>
<td>3.54 x 10^{-3}</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>1.04 x 10^{-2}</td>
<td>2.08 x 10^{-3}</td>
</tr>
<tr>
<td>9</td>
<td>1.22</td>
<td>367</td>
<td>Noble gas</td>
<td>5.21 x 10^{-1}</td>
<td>5.53 x 10^{-3}</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>3.14 x 10^{-2}</td>
<td>4.51 x 10^{-4}</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>All others</td>
<td>2.89 x 10^{-2}</td>
<td>4.51 x 10^{-4}</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Na</td>
<td>100 kg/hr</td>
<td>3.37 x 10^{-4}</td>
</tr>
<tr>
<td>11</td>
<td>1.22</td>
<td>367</td>
<td>Noble gas</td>
<td>5.21 x 10^{-1}</td>
<td>5.53 x 10^{-3}</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cs &amp; I</td>
<td>5.23 x 10^{-2}</td>
<td>4.51 x 10^{-4}</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>All others</td>
<td>2.89 x 10^{-2}</td>
<td>4.51 x 10^{-4}</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Na</td>
<td>100 kg/hr</td>
<td>3.37 x 10^{-4}</td>
</tr>
<tr>
<td>13</td>
<td>1.22</td>
<td>367</td>
<td>Na</td>
<td>100 kg/hr</td>
<td>3.37 x 10^{-4}</td>
</tr>
</tbody>
</table>
4.6 Dose Results

Doses for the generic severe accident scenarios were calculated based on the methodology described in Section 3.6. The 24-hour dose was calculated at one mile from the site boundary based on the methodology in Regulatory Guide 1.145. Doses were calculated as the whole body dose using dose conversion factors based on FGR-13. As expected, the results show that the largest doses result from cases where containment is failed and the largest amount of radionuclides are released. In all cases, the largest contributors to the dose are the noble gases and plutonium.

The 24-hour whole body dose for cases involving sodium only are presented in Table 4.5. Cases with the containment intact show to have the lowest doses, where cases with containment failed have the largest doses. More specifically, the scenario with the large sodium fire and containment failure, Case 3, has the largest dose at 14.1 rem. The case where containment is intact and there is a small sodium release has the smallest dose at $1.28 \times 10^{-3}$ rem. Another trend to note is the similarity in dose between Cases 1 and 13. Even though the amount of sodium released in Case 1 is twice that in Case 13, the dose of Case 1 is larger by only $6.6 \times 10^{-3}$ rem.
Table 4.5: Dose results (24 hr) for scenarios containing sodium only.

<table>
<thead>
<tr>
<th>Case #</th>
<th>Containment State</th>
<th>Dose (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Intact</td>
<td>$1.96 \times 10^{-2}$</td>
</tr>
<tr>
<td>2</td>
<td>Intact</td>
<td>$1.28 \times 10^{-3}$</td>
</tr>
<tr>
<td>3</td>
<td>Failed</td>
<td>$1.41 \times 10^1$</td>
</tr>
<tr>
<td>4</td>
<td>Failed</td>
<td>$1.41 \times 10^1$</td>
</tr>
<tr>
<td>13</td>
<td>Intact</td>
<td>$1.30 \times 10^{-2}$</td>
</tr>
</tbody>
</table>

The 2-hour dose at the EAB was also calculated for the cases with a small release of sodium. Compared to the 24-hour dose, the 2-hour dose is smaller for Case 2 at $2.03 \times 10^{-4}$ rem. Despite the larger cloud concentration, only two hours of exposure contribute to the dose as opposed to 24 hours, making the 2-hour dose smaller. The 2-hour dose at the EAB for Case 4 is $5.94 \times 10^{-1}$ rem, which is approximately a factor of four larger than the 24-hour dose at one mile. The FC curve of the TNF would impose a requirement of less than $1 \times 10^{-3}$ per year for an LBE with the characteristics of a 50 kg release of primary sodium in a failed containment, such as could potentially occur in a refueling accident if the containment was not isolated at the time of the release.
Table 4.6: Dose results (2 hr) for the small sodium release cases.

<table>
<thead>
<tr>
<th>Case</th>
<th>Containment State</th>
<th>Dose (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>2</td>
<td>Intact</td>
<td>2.03 x 10^{-4}</td>
</tr>
<tr>
<td>4</td>
<td>Failed</td>
<td>5.94 x 10^{-1}</td>
</tr>
</tbody>
</table>

The results for 24-hour whole body dose for cases involving arrested melt progression are shown in Table 4.7. Of these cases, the largest dose is from scenarios where containment is failed; for Cases 7 and 8, the dose is 225 rem. Despite the absence of sodium in Case 8, the doses are identical since sodium has a negligible contribution to the total dose. For arrested melt scenarios with containment intact, Cases 5 and 6, the largest dose is 1.98 rem in Case 5. Given the effect of sodium on aerosol deposition, it would have been expected that Case 6 would have exhibited the largest dose. However, in the absence of sodium, containment pressure is decreased; it is the pressurization of containment that drives the increased flow rate exiting containment and thereby increases the release to the environment. Accordingly, doses from individual elements are increased in Case 5 relative to Case 6.

With regards to the whole body dose from noble gases, in all cases, the isotope $^{133}$Xe has the largest inventory of all noble gases by two orders of magnitude. As expected from the inventory of $^{133}$Xe, in scenarios where containment is intact, the dose contribution from $^{133}$Xe alone is larger than the dose contribution of all krypton isotopes.
combined, primarily due to relatively long half life of $^{133}$Xe, which is 5.24 days; it should be noted that the half life of most krypton isotopes is on the order of hours, not days. However, considering only the noble gases, krypton has the larger dose contribution in early containment failure scenarios because of isotope $^{88}$Kr. In these scenarios, the dose is received immediately upon release from fuel when $^{88}$Kr has the largest inventory of all krypton isotopes and the largest dose conversion factor of all noble gases. Given that the dose conversion factor is proportional to the energy imparted to the tissue, it follows that isotopes with the largest energy release during decay will have the largest dose conversion factor. Accordingly, $^{88}$Kr undergoes gamma emission with an energy of 2391 keV, far larger than the decay energy of any other noble gas isotope [30].
Table 4.7: Dose results (24 hr) for arrested melt scenarios.

<table>
<thead>
<tr>
<th>Case #</th>
<th>Containment State</th>
<th>RN Group</th>
<th>Dose (rem)</th>
<th>Total Dose from All RNs (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>5</td>
<td>Intact</td>
<td>Xe</td>
<td>1.32</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Kr</td>
<td>5.6 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>I</td>
<td>7.11 x 10^{-2}</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Cs</td>
<td>1.24 x 10^{-2}</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Na</td>
<td>1.31 x 10^{-2}</td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>Intact</td>
<td>Xe</td>
<td>8.47 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Kr</td>
<td>3.62 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>I</td>
<td>1.81 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Cs</td>
<td>3.29 x 10^{-2}</td>
<td></td>
</tr>
<tr>
<td>7</td>
<td>Failed</td>
<td>Xe</td>
<td>4.30 x 10^{1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Kr</td>
<td>1.72 x 10^{2}</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>I</td>
<td>8.77</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Cs</td>
<td>1.26</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Na</td>
<td>2.87 x 10^{-2}</td>
<td></td>
</tr>
<tr>
<td>8</td>
<td>Failed</td>
<td>Xe</td>
<td>4.30 x 10^{1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Kr</td>
<td>1.72 x 10^{2}</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>I</td>
<td>8.77</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Cs</td>
<td>1.26</td>
<td></td>
</tr>
</tbody>
</table>

Similar to the previous results, the largest dose for scenarios involving an energetic event occur when containment is failed (Table 4.8). With containment failure, the offsite whole body dose at one mile from the site boundary is 1.15 x 10^3 rem. The
largest contributors to the dose in this case are plutonium and krypton. When containment is intact, the offsite whole body dose is $1.11 \times 10^1$ rem; in this case, the largest contributors to the dose are plutonium and xenon.

For the case of core uncovery, the 24-hour total dose is very similar to the dose from the energetic event scenarios (Table 4.9). As expected, the dose contribution from each element remains unchanged, with the exception of iodine and cesium which have larger release fractions from fuel for the core uncovery scenario. Again, the largest total whole body dose results when containment is failed; the dose in this case is $1.17 \times 10^3$ rem, with the largest contributors again being plutonium and krypton. For the case where containment is intact, the whole body dose is $1.13 \times 10^1$ rem, with the largest contributors being plutonium and xenon.
Table 4.8: Dose results (24 hr) for scenarios involving an energetic event.

<table>
<thead>
<tr>
<th>Description</th>
<th>RN Group</th>
<th>Dose (rem)</th>
<th>Total Dose from All RNs (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case 9</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Containment</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Intact</td>
<td>Xe</td>
<td>2.78</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Kr</td>
<td>1.18</td>
<td></td>
</tr>
<tr>
<td></td>
<td>I</td>
<td>2.13 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Cs</td>
<td>3.73 x 10^{2}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Te</td>
<td>1.85 x 10^{3}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Sr</td>
<td>1.47 x 10^{3}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Ba</td>
<td>1.09 x 10^{3}</td>
<td>1.11 x 10^{1}</td>
</tr>
<tr>
<td></td>
<td>Ru</td>
<td>7.09 x 10^{3}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Pu</td>
<td>5.78</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Am</td>
<td>9.84 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>U</td>
<td>4.28 x 10^{-5}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>La</td>
<td>8.00 x 10^{3}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Zr</td>
<td>1.04 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Na</td>
<td>1.31 x 10^{-2}</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Description</th>
<th>RN Group</th>
<th>Dose (rem)</th>
<th>Total Dose from All RNs (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case 10</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Containment</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Failed</td>
<td>Xe</td>
<td>9.03 x 10^{1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Kr</td>
<td>3.63 x 10^{2}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>I</td>
<td>2.63 x 10^{1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Cs</td>
<td>3.78</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Te</td>
<td>2.10 x 10^{1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Sr</td>
<td>1.59 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Ba</td>
<td>1.11 x 10^{1}</td>
<td>1.15 x 10^{3}</td>
</tr>
<tr>
<td></td>
<td>Ru</td>
<td>7.03 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Pu</td>
<td>5.63 x 10^{2}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Am</td>
<td>9.62 x 10^{1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>U</td>
<td>5.13 x 10^{-3}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>La</td>
<td>1.19</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Zr</td>
<td>1.08 x 10^{1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Na</td>
<td>2.87 x 10^{-2}</td>
<td></td>
</tr>
</tbody>
</table>
Table 4.9: Dose results (24 hr) for scenarios involving core uncovery.

<table>
<thead>
<tr>
<th>Description</th>
<th>RN Group</th>
<th>Dose (rem)</th>
<th>Total Dose from All RNs (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case 11</td>
<td>Xe</td>
<td>2.78</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Kr</td>
<td>1.18</td>
<td></td>
</tr>
<tr>
<td></td>
<td>I</td>
<td>3.55 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Cs</td>
<td>6.22 x 10^{-2}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Te</td>
<td>1.85 x 10^{-3}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Sr</td>
<td>1.47 x 10^{-3}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Ba</td>
<td>1.09 x 10^{-3}</td>
<td>1.13 x 10^{1}</td>
</tr>
<tr>
<td></td>
<td>Ru</td>
<td>7.10 x 10^{-3}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Pu</td>
<td>5.78</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Am</td>
<td>9.84 x 10^{-4}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>U</td>
<td>4.28 x 10^{-5}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>La</td>
<td>8.00 x 10^{-3}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Zr</td>
<td>1.04 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Na</td>
<td>1.31 x 10^{-2}</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Description</th>
<th>RN Group</th>
<th>Dose (rem)</th>
<th>Total Dose from All RNs (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case 12</td>
<td>Xe</td>
<td>9.03 x 10^{1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Kr</td>
<td>3.62 x 10^{2}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>I</td>
<td>4.39 x 10^{1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Cs</td>
<td>6.3</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Te</td>
<td>2.10 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Sr</td>
<td>1.59 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Ba</td>
<td>1.11 x 10^{-1}</td>
<td>1.17 x 10^{3}</td>
</tr>
<tr>
<td></td>
<td>Ru</td>
<td>7.03 x 10^{-1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Pu</td>
<td>5.63 x 10^{2}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Am</td>
<td>9.62 x 10^{1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>U</td>
<td>5.13 x 10^{-3}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>La</td>
<td>1.19</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Zr</td>
<td>1.08 x 10^{1}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Na</td>
<td>2.87 x 10^{-2}</td>
<td></td>
</tr>
</tbody>
</table>
Returning to the FC curve, it can be seen that the whole body dose results for all cases do satisfy the curve with the exception of Cases 10 and 12. The dose results from these two cases exceed the curve in that an offsite dose of this magnitude is not acceptable for any event sequence frequency, even those less than $1 \times 10^{-7}$ per year. For Cases 7 and 8, the arrested melt scenario with containment failure, the offsite whole body dose satisfies the FC curve if the sequence frequency is no greater than $1 \times 10^{-6}$ per year. For all other cases, the offsite dose results will satisfy the FC curve if the sequences have a frequency no greater than $1 \times 10^{-4}$ per year. The current fleet of LWRs deployed in our country satisfy this event frequency for severe accidents, so it is expected that the SFR will also satisfy the $1 \times 10^{-4}$ per year limit.

### 4.7 Comparison of TNF with QHOs

Two calculations were performed in WinMACCS in order to compare the constraints of the TNF with the NRC’s QHOs: Case 12 and a bounding case in which the source terms are greater than what appears to be mechanistically achievable. For the bounding case, it is assumed that containment is failed and the release fractions from fuel are substantially larger than best estimates (Table 4.10). In WinMACCS, the source term is released as a plume at problem time zero with a duration of 60 s. The bounding case and Case 12 have been assigned an event frequency of $1 \times 10^{-7}$ per year, which is the highest frequency for this accident that would be within the constraints of the TNF. For
comparison, the results of NUREG-1150 have also been shown to demonstrate the margin to the QHO limits that is achieved by existing plants.

Table 4.10: Release fractions for the bounding case.

<table>
<thead>
<tr>
<th>Group</th>
<th>Release Fraction or Rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>Noble Gases</td>
<td>1.0</td>
</tr>
<tr>
<td>Cs &amp; I</td>
<td>0.1</td>
</tr>
<tr>
<td>All Others</td>
<td>$1 \times 10^{-5}$</td>
</tr>
<tr>
<td>Sodium</td>
<td>100 kg/hr</td>
</tr>
</tbody>
</table>

The WinMACCS results show that the mean early fatality risk to an individual within 1 mile of the plant and the mean latent cancer fatality risk to an individual within 10 miles of plant for the energetic case and the bounding case satisfy the QHOs with a large margin (Figure 4.24). For Case 12 (the core uncovery case) the early fatality risk is $8.7 \times 10^{-13}$ per year; the risk of fatality from latent cancer to an individual within 10 miles of the plant is $2.2 \times 10^{-12}$ per year. For the bounding case, the early fatality risk is $5.4 \times 10^{-9}$ and the latent cancer fatality risk is $7.4 \times 10^{-11}$ (Table 4.11). Comparing these results to the NRC’s QHOs, the WinMACCS results for offsite risk in Case 12 satisfy the QHOs by an extremely large margin (at least five orders of magnitude). The results of the bounding case satisfy the QHOs by a smaller margin of two orders of magnitude in the case of early fatalities. However, relative to the mean individual risk results from NUREG-1150, the early fatality risk in the bounding case is nearly an order of magnitude smaller than the largest NUREG-1150 result. In summary, the methods prescribed by the TNF satisfy the NRC’s QHOs with substantial margin even when considering a non-mechanistically large bounding accident scenario. The TNF limit of $1 \times 10^{-7}$ for large
consequence events could be made an order of magnitude larger and still provide high assurance that the QHOs would be satisfied.

Table 4.11: Tabular results of offsite risk.

<table>
<thead>
<tr>
<th>Reactor/Case</th>
<th>Early Fatality Risk (yr⁻¹)</th>
<th>Latent Cancer Fatality Risk (yr⁻¹)</th>
</tr>
</thead>
<tbody>
<tr>
<td>NUREG-1150</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Surry</td>
<td>2 x 10⁻⁸</td>
<td>2 x 10⁻⁹</td>
</tr>
<tr>
<td>Peach Bottom</td>
<td>5 x 10⁻¹¹</td>
<td>9.5 x 10⁻¹⁰</td>
</tr>
<tr>
<td>Sequoyah</td>
<td>1.5 x 10⁻⁸</td>
<td>1.5 x 10⁻⁸</td>
</tr>
<tr>
<td>Grand Gulf</td>
<td>3 x 10⁻¹¹</td>
<td>3.5 x 10⁻¹⁰</td>
</tr>
<tr>
<td>Zion</td>
<td>1 x 10⁻⁸</td>
<td>1 x 10⁻⁸</td>
</tr>
<tr>
<td>SFR with Largest Mechanistic Source Term</td>
<td>9 x 10⁻¹³</td>
<td>2 x 10⁻¹²</td>
</tr>
<tr>
<td>SFR with Bounding Source Term</td>
<td>5 x 10⁻⁹</td>
<td>7 x 10⁻¹¹</td>
</tr>
<tr>
<td>QHO Limit</td>
<td>5 x 10⁻⁷</td>
<td>2 x 10⁻⁶</td>
</tr>
</tbody>
</table>
Figure 4.24: Mean individual risk results for early fatality and latent cancer fatality.
CHAPTER 5:

CONCLUSIONS & FUTURE WORK

In this research, a methodology was demonstrated which can be used to calculate the offsite consequences and risk from severe accidents in an SFR for comparison with the TNF limit curve. The RCS code was developed to analyze the heatup of sodium and the reactor kinetics associated with a transient. The kinetics portion of the code was validated against other kinetics solutions and was shown to obtain the same asymptotic period without feedback. Comparisons were also made with the SFR-specific severe analysis code SAS4A; reasonable agreement was found between RCS and SAS4A. Additional feedback mechanisms would have to be included in RCS, such as axial and radial expansion in the fuel, to obtain better agreement. The added complexity of the model is unwarranted. During the time frame in which the SAS4A and RCS overlap, the net reactivity predicted by SAS4A will be used as input to the RCS analysis.

A MELCOR model was also designed which assessed the loads on containment during a severe accident, the transport and deposition of radionuclides in the containment, and their release to the environment. Containment behavior during the identified severe accident scenarios showed to be generally non-threatening to the integrity of containment and its structures, with the exception of a large sodium fire. For a sufficiently large spray
fire, such an event could fail containment. Until a sodium-based version of MELCOR is released, this portion of the research could be improved upon by creating a more complete containment model with vents, filters and more detailed internal structures. If a sodium version of MELCOR is released, the RCS portion of this research could be performed entirely within MELCOR if realistic models for fuel failure and release from failed fuel were included.

Initial inventories released from fuel and release rates to the environment from MELCOR were used to calculate the offsite consequences for the identified accident scenarios. Results showed that, for most accidents, the offsite doses would be less than the 500 rem boundary of the TNF limit curve. The energetic event and core uncovery event with containment failure were found to exceed the 500 rem offsite dose. Thus, the frequency of these events would have to be less than $1 \times 10^{-7}$ per year with high confidence to satisfy the TNF criteria. For the other severe accidents, it is expected that for the reference SFR design their event frequencies would be low enough such that they would satisfy the FC curve with very large margin.

The parametric analysis of the effects of sodium on containment response and aerosol behavior showed that sodium vaporization does have noticeable effects on containment pressure, radionuclide release and aerosol deposition. In the cases studied in this research, the energy input to containment due to sodium vaporization increases containment pressure by approximately 0.01 MPa; similar effects on containment
atmosphere temperature are observed. With the additional containment pressurization, there is also an increase in release of noble gases to the environment when sodium vaporization occurs. Environmental release of radionuclides, other than noble gases, decreases when sodium is present due to the effect of sodium aerosols on the deposition of the fission product aerosols; when sodium vaporization occurs, the majority of radionuclides deposit before they can be released to the environment. Nevertheless, the energy input to the containment when sodium is being vaporized from the pool results in an increase in offsite doses because of the effect of the additional pressurization of the containment on the rate of leakage of the noble gases.

Lastly, in order to evaluate whether the TNF is unnecessarily restrictive, the offsite risk was analyzed for an assumed plant design in which the risk is dominated by an event with a dose larger than the 500 rem limit of the TNF limit curve and an event frequency of $1 \times 10^{-7}$ per year, which is the largest acceptable frequency for such an event. Two types of source term were considered for this limiting condition, the largest mechanistic source term obtained in the analysis and a non-mechanistic source term comparable to the Chernobyl accident but with early failure of the containment and a prompt release to the environment. The early and latent fatality risks in the proximity of the plant were compared with the NRC’s QHOS. The results showed that the large mechanistic event results satisfied the QHOS by a very large margin. The bounding case also satisfies the QHOS by a substantial but smaller margin; the early fatality risk in the
Bounding case is less than the largest early fatality risk result of NUREG-1150 and two orders of magnitude less than the QHO.
LIST OF REFERENCES


APPENDIX A: RCS Code

Kinetics Version

/* RCS Program - Calculates the core kinetics of an SFR RCS during an accident */

#include <iostream>
#include <cmath>
#include <fstream>
#include <string>
using namespace std;
using std::cout;
using std::log10;

//Declare subroutines
double log10(double time);
double rho(double temp);
double enthalpy (double temp);
double QDH (double time);
double invtemp (double enth);

int main()
{
    //User Input
    double Tmax = 100; //Max problem time (s)
double Rfbc = 0; //Reactivity temp fb coeff -2E-5
double Ra =  12E-4; //Reactivity ramp/step input
double Rb = 0;
double Tinp = 5; //Time for reactivity insertion (s)
double Tendr = 0;

    //Initialize vbls
    double DT = 0.00001; //Time step (s)
double QIHX; //IHX power (kw)
double QP;
double Q[2];
double power = 1E6; //Core power (kw)
    int Indexm = 1 + Tmax/DT;
int Tindex = 0;
double time = 0;

//Characteristic roots and coeffs
double Ametal[7] = {3.24575E1, 2.7453, 2.8278E1, 3.0789E1, 4.8407E1, 4.2248E1, 9.9663E6}; //Keepin data

string Coeff;
if (Smetal[6] == -2.1E4){
    Coeff = "Keepin";
} else{
    Coeff = "SAS";
}

//Feedback coeffs
double Rins[2]; //Reactivity inserted
Rins[0] = 0;
double Rtot[2]; //Total reactivity
Rtot[0] = 0;

double del[7];
double sig3 = 0;
double sigt = 0;

//Initialize del[j] with 0s
for (int j = 0; j < 7; j++){
    del[j] = 0;
}

//Print data

//Set QP and IHX to core power
QP=power;
QIHX = power;
Q[0] = power;

ofstream datafile; //Create and open txt file
ofstream paramfile;
datafile.open ("RCSPlotData.txt");
paramfile.open("RCSPrams.txt");

120
paramfile << "Rfbc:," << Rfbc << 
"Ra:," << Ra << 
"Rb:," << Rb 
"Tinp:," << Tinp << 
"Power (kw):," << power << 
"DT:," << DT 
"Data Source:," << Coeff << "\n";

datafile << "Time (s),Q (kw),Rtot\n";
datafile << time << "," << Q[0] << "," << Rtot[0] << "\n";

for(0; Tindex<Indexm;++Tindex){
time = time + DT; //Increment time

if (QP < QDH(time)) //Define power
    Q[Tindex] = QDH(time);
//else Qold[Tindex] = QP;

QIHX = power;

// Reactivity Step Insertion
if (Tinp >= (time)){
    Rins[1] = Ra + Rb*time;
}
else Rins[1] = Ra + Rb*Tinp;

if (Tendr != 0 && time > Tendr){
    Rins[1] = 0; //Used to withdraw reactivity
}


//Power calcs based on reactivity
//Summations
for (int j = 0; j < 7; j++){

del[j]=del[j]*(exp(Smetal[j]*DT)) + Ametal[j]*(exp(Smetal[j]*DT))*Rtot[0]*Q[0]*
DT;

    sig3 += Ametal[j]*(Rtot[1])*DT;
    sigt += del[j];
}

Q[1] = (power + sigt)/(1-sig3);
// Print results
if (time > 220){
    if (Tindex % 500000 == 0){
        cout << "\nTime = " << time << "\n";
    }
}

if (Tindex % 50000 == 0){
    //if (time > 126){
    //    cout << "\nTime = " << time << "\n";
    //    cout << "Tupper = " << Tupper[1] << "\n";
    //    cout << "Tlower = " << Tlower[1] << "\n";
    //    cout << "Core outlet temp = " << Tout << "\n";
    //    cout << "IHX outlet temp = " << TIHXout << "\n";
    //    cout << "Pvapor = " << Pvapor[Tindex+1] << "\n";
    //    cout << "Pool Height = " << Pheight << "\n";
    //}
}

Rtot[0] = Rtot[1];
Rins[0] = Rins[1];
Q[0] = Q[1];
sig3 = 0;
sigt = 0;
}
datafile.close();
paramfile.close();

   return 0;
}

// Function rho returns density of sodium at temperature temp (K)
double rho(double temp){
    double answer;
    //if (temp > 2503.7){
    //    //answer = 219 + 275.32*(1-temp/2503.7)) + 511.58*(sqrt(abs((1-
    //    (temp/2503.7))));
    //    //temp = 2503.7;
    //}
    //else{

    answer = 219 + 275.32*(1-temp/2503.7)) + 511.58*pow(1-(temp/2503.7),0.5);
/} return answer; } // Returns enthalpy (kJ/kg) of sodium at temperature temp (K). double enthalpy (double temp) { double answer; answer = -365.77 + 1.6582*temp - 4.2375E-4*pow(temp,2) + 1.4847E-7*pow(temp,3) + 2992.6/temp; return answer; } // Returns decay heat as function of time (s) double QDH (double time) { double answer; double x; x = log10(time); answer = -7.02E-5*pow(x,4) + 1.1E-3*pow(x,3) - 4.65E-3*pow(x,2) - 4.31E-3*x + 5.11E-2; return answer; } double invtemp (double enth) { double answer; answer = 234.93 + 0.62746*enth + 2.0799E-4*pow(enth,2) - 9.3544E-8*pow(enth,3) + 1.0401E-11*pow(enth,4); return answer; }
/* RCS Program - Calculates the heatup of an SFR RCS during an accident */

#include <iostream>
#include <cmath>
#include <fstream>
#include <string>

using namespace std;

using std::cout;
using std::log10;

//Declare subroutines
double log10(double time);
double rho(double temp); //Density
double enthalpy (double temp); //Enthalpy as a fn of temp
double QDH (double time); //Decay heat
double invtemp (double enth); //Temp as a fn of enthalpy

int main()
{
    //User Input
    double Tmax = 2500; //Max problem time (s)
double DT = 0.00001; //Time step (s)
double Rfbc = -2.2E-5; //Reactivity temp fb coeff (-1.1E-2 $/degree
    from SAS)
    double Ra = 0; //Reactivity step input
double Rb = 2.E-5; //Reactivity ramp input
double Tinp = 40; //Time for reactivity insertion (s)
double Tendr = 0; //Time to remove reactivity (use 0 if not removing)
double power = 1E6; //Core power (kw)
double prfr = 5; //Print results every prfr sec
double DCHfr = 1; //Calculate DCH every DCHfr sec
    int printfreq = prfr/DT +1; //Printfreq based on Tindex
    int DCHfreq = DCHfr/DT +1; //DCHfreq based on Tindex

    //Initialize vbls
    double Mevap = 0; //Mass evaporated (kg)
double Mupper = 0; //Mass in upper pool (kg)
double Mlower = 0; //Mass in lower pool (kg)
double M0 = 1.278E6; //Tot. mass (kg)
double mdot; //Mass flow rate (kg/s)
double Vlower = 812; //Vol. of lower pool (m**3)
double Vupper; //Vol of Upper pool (m**3)
double w = 5.86; //Vol. flow rate (m**3/s)
double Pheight = 11.5; //Pool height (m)
double QIHX; //IHX power (kw)

double Qn[11000]; //Kinetics Q for DCH calc
double Q[2]; //Kinetics Q
double DCH; //DCH based on finite history
double Qtot[2]; //Total power
double DCHinf = 0; //Decay heat based on infinite history at full power
double DCHtot[2]; //Total decay heat (DCHinf + DCH)
double DCHsum[11000]; //Decay heat post t=0

double TPavg = 705; //Avg pool temp (K)
double Tupper[2]; //Upper pool temp (K)
double Tlower[2]; //Lower pool temp (K)
double Tout = 0; //Core exit temp (K)
double TIHXout = 0; //IHX outlet temp (K)
double TCavg = 704; //Core avg temp (K)
double deltCavg = 0;

time = 0;
double DCHtime = 0;
int DCHTindex = 0;

Hupper[2];
Hupper[0] = 4.25E8; //Initial upper pool enth.
Hupper[1] = 0;
double Hlower[2];
Hlower[0] = 3.87E8; //Init. lower pool enth.
Hlower[1] = 0;
double Uhout = 0;
double Lhout = 0;
//Characteristic roots and coeffs (!Beta_eff = 0.002!)
double Ametal[7] = {3.2475E1, 2.7453, 2.8277E1, 3.0789E1, 4.8418E1, 4.2276E1, 2.7303E6}; // (SAS data)

// Feedback coeffs
double Rfb[2]; // Total temp fb
Rfb[0] = 0;
double Rins[2]; // Reactivity inserted
Rins[0] = 0;
double Rtot[2]; // Total reactivity
Rtot[0] = 0;

double del[7];
double sig3 = 0;
double sigt = 0;

// Initialize del[j] with 0s
for (int j = 0; j < 7; j++){
del[j] = 0;
}

// Inital Core Avg Temp
double TCavgo = (Tupper[0]+Tlower[0])/2;

cout << "RCS Computer Program \n";

// Set Q and QIHX to core power, declare decay heat fraction
QIHX = power;
Qtot[0] = power;
Q[0] = 0.934*power;
Qn[0] = 0.934*power;
DCH = 0.066*power;
DCHtot[0] = 0.066*power;

mdot=w*rho(Tlower[0]); // Calculate mass flow rate

ofstream datafile; // Create and open txt file
ofstream paramfile;
datafile.open ("RCSPlotData.txt");
paramfile.open("RCSParams.txt");
paramfile << "Rfbc:," << Rfbc << "," << Ra << "n" << "Rb:," << Rb << "," << Tinp: (s)," << Tinp << "n" << "Tendr: (s)," << Tendr << "n" << "Power (kw):," << power << "n" << "DT:," << DT << "n" << "QIHX: (kw)," << QIHX << "n" << "DCH Calc. Freq (s):," << DCHfr << "n";

datafile << "Time (s),Tupper (K),Tlower (K),TCavg (K), TPavg (K),Tout (K),TIHXout (K),Qtot (kw),Q (kw),QIHX (kw),DCH (kw),DCHinf (kw),DCHtot[t] (kw),Rins,Rfb,Rtot,Pheight (m),Pvapor (atm),DCHTindex,tn, Qn(DCTindex),DCHTime\n";

datafile << time << "," << Tupper[0] << "," << Tlower[0] << "," << TCavg << "," << TPavg << "," << Tout << "," << TIHXout << "," << Qtot[0] << "," << Q[0] << "," << QIHX << "," << DCH << "," << DCHinf << "," << DCHtot[0] << "," << Rins[0] << "," << Rfb[0] << "," << Rtot[0] << "," << Pheight << "," << Pvapor[0] << "," << DCHTindex << "\n";

for(0; Tindex<=Indexm;++Tindex){

time = time + DT; //Increment time
DCHtime = Tindex*DT;

if(Tindex%DCHfreq == 0){
    DCHTindex = (DCHtime-1)/DCHfr;
}

// Upper pool temperature
Uhout = enthalpy(Tlower[0]) + Qtot[0]/mdot;
Tout = invtemp(Uhout); //Core outlet temp as a fn of enth
Hupper[1] = Hupper[0] + (DT*mdot*(-enthalpy(Tupper[0]) + Uhout));

// Lower pool temperature
Lhout = enthalpy(Tupper[0]) - QIHX/mdot;
TIHXout = invtemp(Lhout); //IHX outlet temp as fn of enth
Hlower[1] = Hlower[0] + (DT*mdot*(-enthalpy(Tlower[0]) + Lhout));

// Mass balance & Pool Height
Mlower = rho(Tlower[0])*Vlower;
Mupper = M0 - Mlower - Mevap;
Vupper = Mupper/rho(Tupper[0]);
Pheight = (4*Vupper)/(3.141592*pow(14.0,2.0)) + 7.03;

// Reactivity Step Insertion
if (Tinp >= (time)){
    Rins[1] = Ra + Rb*time;
}
else Rins[1] = Ra + Rb*Tinp;

if (Tendr != 0 && time > Tendr){
    Rins[1] = 0;
    //Used to withdraw reactivity
}

// Reactivity Temp Feedback Calcs
TCavg = (Tlower[0] + Tout)/2;
Rfb[1] = Rfbc*(TCavg - TCavgo); //704
delTCavg = TCavg - TCavgo;


//Kinetics calculations
for (int j = 0; j < 7; j++){

del[j] = del[j]*(exp(Smetal[j]*DT)) + Ametal[j]*(exp(Smetal[j]*DT))*Rtot[0]*Q[0]*DT;

    sig3 += Ametal[j]*(Rtot[1])*DT;
    sigt += del[j];
}

Q[1] = (0.934*power + sigt)/(1-sig3);

//Decay Heat calcs
if (Tindex % DCHfreq == 0){
    DCH = 0;
}

//Declare Qn for DCH calc
if (Tindex % DCHfreq == 0){
    Qn[DCHTindex] = Q[1];
}
double tn = DCHfr;

if (Tindex % DCHfreq == 0){
    //double tn = 0;
    DCHinf = 0.066*power*(pow(time,-0.2));
    for (int n=1; n <= DCHTindex; n++) {
        128
DCHsum[n] = 0.2*0.066*Q[n]*(pow(DCHtime-tn,-1.2))*DCHfr;
DCH += DCHsum[n];

} }  

DCHtot[1] = DCHinf + DCH;


// Calculate temperature at end of time step
Tlower[1] = invtemp(Hlower[1]/Mlower);
Tupper[1] = invtemp(Hupper[1]/Mupper);
TPavg = (Tlower[1]+Tupper[1])/2;

// Calculate vapor space pressure (Ar only!)
Pvapor[1] = (1.67625*.0821*Tupper[1])/(3.141592*(14.0^2)*(12.2-Pheight))/4;

// Print results
//if (time > 1.9955){
if (Tindex % printfreq == 0){

cout << "\nTime = " << Tindex*DT << "\n; \\
}  
}

if (Tindex % printfreq == 0){
//if (time > 1.999){
  //cout << "\nTime = " << time << "\n; \\
  //cout << "Tupper = " << Tupper[1] << "\n; \\
  //cout << "Tlower = " << Tlower[1] << "\n; \\
  //cout << "Core outlet temp = " << Tout << "\n; \\
  //cout << "IHX outlet temp = " << TIHXout << "\n; \\
  //cout << "Pvapor = " << Pvapor[Tindex+1] << "\n; \\
  //cout << "Pool Height = " << Pheight << "\n; \\
  //    
//  }
}  
}
// Prepare for next timestep
Hlower[0] = Hlower[1];
Hupper[0] = Hupper[1];
mdot= w*rho(Tlower[1]);
Tupper[0] = Tupper[1];
Tlower[0] = Tlower[1];
Pvapor[0] = Pvapor[1];
Rtot[0] = Rtot[1];
Rins[0] = Rins[1];
Rfb[0] = Rfb[1];
Q[0] = Q[1];
Qtot[0] = Qtot[1];
DCHtot[0] = DCHtot[1];
sig3 = 0;
sigt = 0;
}
datafile.close();
paramfile.close();
return 0;

// Function rho returns density of sodium at temperature temp (K)
double rho(double temp) {
    double answer;
    answer = 219 + 275.32*(1 - (temp/2503.7)) + 511.58*pow(1 - (temp/2503.7), 0.5);
    return answer;
}

//Returns enthalpy (kJ/kg) of sodium at temperature temp (K).
double enthalpy (double temp) {
    double answer;
    answer = -365.77 + 1.6582*temp - 4.2375E-4*temp + 1.4847E-7*temp + 2992.6/temp;
    return answer;
}

//Returns decay heat as function of time (s)
double QDH (double time) {
    double answer;
    double x;
    x = log10(time);
answer = -7.02E-5*\text{pow}(x,4) + 1.1E-3*\text{pow}(x,3) - 4.65E-3*\text{pow}(x,2) - 4.31E-3*x + 5.11E-2;
    return answer;
}

double invtemp (double enth) {
    double answer;
    answer = 234.93 + 0.62746*enth + 2.0799E-4*\text{pow}(enth,2) - 9.3544E-8*\text{pow}(enth,3) + 1.0401E-11*\text{pow}(enth,4);
    return answer;
}
**SFR v9 Deck - Includes Na Enthalpy**  
**Case 5 - Arrested Fuel Melt - Fuel Failure w/ Sustained Na Vaporization**  
**Treating Na as Boron (to isolate Na group)**  
**Models aerosol transport & dispersion**  
**AB 5/11/10**

*eor* melgen  

```
********** MELGEN INPUT **********  
**********************************
```

* 
** TITLE 'SFRv9C5'**
* 
** DTTIME 0.001 * Initial time-step of 1 ms**
* 
** CRTOUT * 80-column output format**
* 
** FILES **
**
```
** OUTPUTFILE SFRv9C5.OUTG**
** DIAGFILE SFRv9C5.DIAG**
** RESTARTFILE SFRv9C5.RST**
*  
**************
* ** NCG INPUT **
**************
* 
** NCG001 N2 4 * N2 is material number 4**
** NCG002 O2 5 * O2 is material number 5**
*
* CONTROL VOLUMES *

* Containment

*Non-Eq, vertical, CVType

*Containment CVTYPE01 'Cont'

CV60000 CONTAINMENT 2 2 1

*Non-Eq, vertical, CVType

CV600A1 MLFR.4 .8 MLFR.5 .2

*N2=80%, O2=20%

CV600A2 PVOL 1.E5

*Pressure = 1atm

CV600A3 PH2O 0.0

*Partial pressure of water

CV600A4 TATM 300.

*Atmos Temp, K

CV600B0 0.0 0.0

*A/V Table, bottom at 0 m

CV600B1 50.6 2.86E4

*Total vol = 2.86E4 m**3

C600C0 ENERGY.A RATE TF.661

*Containtment Energy Source TF

*Assume that for t<20 hr, peroxide dominates

*For t>20 hr, oxide dominates

TF66100 'Cont ES' 5 1.0

*t < 20 hr, Rate = 636kW

TF66110 0.0 636000.

TF66111 72000. 636000.

*t > 20 hr, Rate = 16kW

20 < t < 27 hr, Rate = 516kW

TF66112 72001. 516000.

TF66113 97200. 516000.

TF66114 97201. 16000.

*Environment

*Non-Eq, vertical, CVType

CV70000 ENVIRONMENT 2 2 2

*N2=80%, O2=20%

CV700A1 MLFR.4 .8 MLFR.5 .2

*Pressure = 1atm

CV700A2 PVOL 1.E5

*Partial pressure of water

CV700A3 PH2O 0.0

*Atmos Temp, K

CV700A4 TATM 300.

*A/V Table, bottom at 0 m

CV700B0 0.0 0.0

*Total vol arbitrary

C700C0 ENERGY.A RATE TF.661

* FLOW PATHS *

* FLOW PATHS *


* From To From To

FL60100 CONT-LEAK 600 700 25.0 25.0

FL60101 3.8623E-5 1. 1.0 *A, L, Frac open

FL60102 3 0 0 0

FL60103 9.0 9.0 1.E3 1.E3 *Form loss coeffs

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*HEAT STRUCTURE INPUT*

*Containment Ceiling*
HS60002000 5 1 0  # nodes, Rect. geo, SS
HS60002001 'Containment Ceiling'
HS60002002 50.6 0.  # Elevation, horiz. orientation
HS60002100 -1 1 0.0  # 5th node at 2m (last node)
HS60002101 2.0 5
HS60002200 -1
HS60002201 'CONCRETE' 4  # All meshes concrete
HS60002300 0  # No internal power source
HS60002400 1 600 EXT 0.5 0.5  # Convective to Cont., ext. heat transfer, crit pool frac.
HS60002500 942.9 12.25 76.9  # A = 2*pi*r^2, CL = r, AL = pi*D
HS60002600 1 700 EXT 0.5 0.5
HS60002700 942.9 12.25 76.9

*Containment walls*
HS60003000 5 1 0  # nodes, Rect. geo, SS
HS60003001 'Containment Wall'
HS60003002 0.0 1.  # Elevation, vert. orientation
HS60003100 -1 1 0.0  # 5th node at 2m (last node)
HS60003101 2.0 5
HS60003200 -1
HS60003201 'CONCRETE' 4  # All meshes concrete
HS60003300 0  # No internal power source
HS60003400 1 600 EXT 0.5 0.5  # Convective to Cont., ext. heat transfer, crit pool frac.
HS60003500 3894.6 1. 50.6  # A = 2*pi*r*h
HS60003600 1 700 EXT 0.5 0.5
HS60003700 3894.6 1. 50.6

*Sodium Pool*
HS50001000 5 1 0  # nodes, Rect. geo, SS
HS50001001 'Heat Slab'
HS50001002 -0.02 -1.E-7  # Elevation, horiz. orientation
HS50001100 -1 1 0.0  # 5th node at 2cm (last node)
HS50001101 0.02 5
HS50001200 -1
HS50001201 'STAINLESS STEEL' 4  # All meshes SS
HS50001300 0
HS50001400 1 600 INT 1.0 1.0
pool frac.
HS50001500 153.94 7.0 14.0
HS50001600 2001 -1 EXT 1.0 1.0
HS50001700 153.94 7.0 14.0
*
*Environment Floor
HS70001000 5 1 0
HS70001001 'Enviro Floor'
HS70001002 0.0 -1.0E-7
HS70001100 -1 1 0.0
HS70001101 0.02 5
HS70001200 -1
HS70001201 'CONCRETE' 4
HS70001300 0
HS70001400 1 700 EXT 0.5 0.5
transfer, crit pool fracs
HS70001500 153.94 7.0 14.0
HS70001600 0 -700
*
*Deck Surface
HS60001000 5 1 0
HS60001001 'Deck Surface'
HS60001002 0.0 -1.0E-7
HS60001100 -1 1 0.0
HS60001101 0.1524 5
HS60001200 -1
HS60001201 'CONCRETE' 4
HS60001300 0
HS60001400 1 600 EXT 0.5 0.5
HS60001500 942.9 12.25 76.9
HS60001600 0 -600
*
={['No internal power source', 'Convective to Cont., int. heat transfer, crit pool frac.', 'A = \pi r^2, CL = r, AL = D', '# nodes, Rect. geo, SS', 'Elvelation, horiz. orientation', '5th node at 2 cm (last node)', 'All meshes concrete', 'No internal power source', 'Convective to environment, external heat transfer, crit pool fracs', 'Insulated boundary on RHS, not associated with any CV', 'Constant temp = 780 K']}

*CF/TF INPUT for Heat SLab*

* Constant temp = 780 K
************
* RN Input *  
*RN Input used to edit radionuclide releases
************
RN1000 0  
*Activate RN package
RN1001 10 4 16 14 13 1 0 16 6
RN1100 0.1E-6 5.0E-6 1000.
RNCC001 1 2 2 2 2 2 2 2 2 3 4 2 2
*Place NGs as vapors in atmosphere
RNVG001 600 1. 0.16947 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. *Xe
RNVG002 600 1. 0.35154 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. *Kr
*  
*Aerosol Sources - input to containment
*I & Cs
RNAG000 600 2 1.
RNAG001 6.14E-3 3.81E-3 4.99E-3 5.94E-3 6.42E-3 6.31E-3 5.64E-3 4.58E-3
RNAG002 3.37E-3 5.07E-3
*
*All Others
RNAG003 600 2 1.
RNAG004 3.4E-3 2.1E-3 2.76E-3 3.28E-3 3.55E-3 3.49E-3 3.12E-3 2.53E-3 1.87E-3
RNAG005 2.8E-3
*
*N
RNAS021 600 2 13 1.0 1. 621 3
RNAS022 .00326 .00202 .00264 .00315 .0034 .00335 .00299 .00243 .00179 .00269
******Aerosol Sources TFs******
*Na
TF62100 'NaSource' 1 1.0
TF62110 0.0 0.0277
*
*
*Deactivate Dep Surfaces
RNDS001 60002 LHS INACTIVE *Ceiling
RNDS002 60003 LHS INACTIVE *Wall
RNDS003 60003 RHS INACTIVE
RNDS004 60002 RHS INACTIVE
RNDS005 50001 LHS INACTIVE * Na pool surf.
*
*
****Add Plot Vbls****
*CVCLT-x-n-cv, x=group, n=total (1) or rad (2)
*CVCLT = mass (kg) excluding deposited
PLOTV2A RN1-CVCLT-2-1.600 * RNs in Cont.
PLOTV9A RN1-CVCLT-13-1.600 * Na in Cont.

136
*RNs in Environment
PLOTV2C RN1-CVCLT-2-1.700 * RNs in Cont.
PLOTV9C RN1-CVCLT-13-1.700 * Na in Cont.
*
*
*Total Aerosol mass (kg) deposited on HS
*ADEP-s-x-y-hs
*s=1=LHS
*y=1=total mass
PLOTDS2 RN1-ADEP-1-2-1.60001 * RNs on Slab
PLOTDS9 RN1-ADEP-1-13-1.60001 * Na on Slab
*
PLOTDF2 RN1-ADEP-1-2-1.70001 * RNs on enviro. floor
PLOTDF9 RN1-ADEP-1-13-1.70001 * Na on enviro. floor
*
*Sectional Plots
*AMG-w-x-y.cv, w=section, x=class, y=total mass
PLOTVS1 RN1-AMG-1-2-1.600 * Volatiles Section 1
PLOTVS2 RN1-AMG-2-2-1.600 * Volatiles Section 2
PLOTVS3 RN1-AMG-3-2-1.600 * Volatiles Section 3
PLOTVS4 RN1-AMG-4-2-1.600 * Volatiles Section 4
PLOTVS5 RN1-AMG-5-2-1.600 * Volatiles Section 5
PLOTVS6 RN1-AMG-6-2-1.600 * Volatiles Section 6
PLOTVS7 RN1-AMG-7-2-1.600 * Volatiles Section 7
PLOTVS8 RN1-AMG-8-2-1.600 * Volatiles Section 8
PLOTVS9 RN1-AMG-9-2-1.600 * Volatiles Section 9
PLOTV10 RN1-AMG-10-2-1.600 * Volatiles Section 10
*
PLOTNS1 RN1-AMG-1-13-1.600 * Na Section 1
PLOTNS2 RN1-AMG-2-13-1.600 * Na Section 2
PLOTNS3 RN1-AMG-3-13-1.600 * Na Section 3
PLOTNS4 RN1-AMG-4-13-1.600 * Na Section 4
PLOTNS5 RN1-AMG-5-13-1.600 * Na Section 5
PLOTNS6 RN1-AMG-6-13-1.600 * Na Section 6
PLOTNS7 RN1-AMG-7-13-1.600 * Na Section 7
PLOTNS8 RN1-AMG-8-13-1.600 * Na Section 8
PLOTNS9 RN1-AMG-9-13-1.600 * Na Section 9
PLOTNS10 RN1-AMG-10-13-1.600 * Na Section 10
.
*eor* melcor
*
******************************************************************************
********** MELCOR INPUT **********
******************************************************************************
* TITLE  'SFRv9C5'
* RESTART  0  * Restart from cycle 0
* TEND     86400.0
* CPULIM   1000.0
    CPULEFT  20.0
* CRTOUT         * 80-column output format
* *************
* FILES *
*************
* RESTARTFILE  SFRv9C5.RST
OUTPUTFILE   SFRv9C5.OUT
PLOTFILE     SFRv9C5.PTF
DIAGFILE     SFRv9C5.DIA
MESSAGEFILE  SFRv9C5.MES
* **********************
* TIME STEP AND EDIT CONTROL *
*******************************
* TIME DTMAX DTMIN DTEDT DTPLT DTRST
TIME1  0.0  1.0  0.001  500.0  250.0  1000.0
.
MELCOR Input without Sodium Energy

**SFR v9 Deck –
**Case 6 - Arrested Fuel Melt - Fuel Failure - no Na
**Models aerosol transport & dispersion
**AB 5/11/10
*
*eor* melgen
*

************ MELGEN INPUT ************

* TITLE 'SFRv9C6'
*
* DTIME 0.001 * Initial time-step of 1 ms
*
* CRTOUT * 80-column output format
*
* FILES *
************
* OUTPUTFILE SFRv9C6.OUTG
DIAGFILE SFRv9C6.DIAG
RESTARTFILE SFRv9C6.RST
*

************
* NCG INPUT *
************
*
NCG001 N2 4 * N2 is material number 4
NCG002 O2 5 * O2 is material number 5
*

************
* CONTROL VOLUMES *
************

*Containment
CVTYPE01 'Cont'
CV60000 CONTAINMENT 2 2 1 *Non-Eq, vertical, CVType
CV600A1 MLFR.4 .8 MLFR.5 .2 *N2=80%, O2=20%
CV600A2 PVOL 1.E5 *Pressure = 1atm
CV600A3 PH2O 0.0 *Partia l pressure of water
CV600A4 TATM 300. *Atmos Temp, K
CV600B0 0.0 0.0  *A/V Table, bottom at 0 m
CV600B1 50.6 2.86E4  *Total vol = 2.86E4 m**3

*Environment
CV70000 ENVIRONMENT 2 2 2  *Non-Eq, vertical, CVType
CV700A1 MLFR.4 .8 MLFR.5 .2  *N2=80%, O2=20%
CV700A2 PVOL 1.E5  *Pressure = 1atm
CV700A3 PH2O 0.0  *Partial pressure of water
CV700A4 TATM 300.  *Atmos Temp, K
CV700B0 0.0 0.0  *A/V Table, bottom at 0 m
CV700B1 70.0 1.E9  *Total vol arbitrary

*************************************
* FLOW PATHS *
*************************************

*  From  To  From  To
FL60100 CONT-LEAK 600 700 25.0 25.0  *A, L, Frac open
FL60101 3.8623E-5 1. 1.0  *Form loss coeffs
FL60102 3 0 0 0  *A, L, Hyd. diameter
FL60103 9.0 9.0 1.E3 1.E3
FL601S0 3.8623E-5 1. 0.007013

***************************************
* HEAT STRUCTURE INPUT *
***************************************

*Containment Ceiling
HS60002000 5 1 0  *# nodes, Rect. geo, SS
HS60002001 'Containment Ceiling'  *Elevation, horiz. orientation
HS60002002 50.6 0.  *5th node at 2m (last node)
HS60002100 -1 1 0.0  *All meshes concrete
HS60002101 2.0 5  *No internal power source
HS60002200 -1  *Convective to Cont., ext. heat transfer, crit
HS60002201 'CONCRETE' 4  pool frac.
HS60002300 0  *A = 2*pi*r^2, CL = r, AL = pi*D
HS60002400 1 600 EXT 0.5 0.5  *Conv. to enviro., ext HT, crit pool frac.
HS60002500 942.9 12.25 76.9
HS60002600 1 700 EXT 0.5 0.5
HS60002700 942.9 12.25 76.9
*
*Containment walls
HS60003000  5  1  0
HS60003001  'Containment Wall'
HS60003002  0.0  1.
HS60003100  -1  1  0.0
HS60003101  2.0  5
HS60003200  -1
HS60003201  'CONCRETE' 4
HS60003300  0
HS60003400  1  600  EXT  0.5  0.5
    pool frac.
HS60003500  3894.6  1.  50.6
HS60003600  1  700  EXT  0.5  0.5
HS60003700  3894.6  1.  50.6
*
*Sodium Pool
HS50001000  5  1  0
HS50001001  'Heat Slab'
HS50001002  -0.02  -1.E-7
HS50001100  -1  1  0.0
HS50001101  0.02  5
HS50001200  -1
HS50001201  'STAINLESS STEEL' 4
HS50001300  0
HS50001400  1  600  INT  1.0  1.0
    pool frac.
HS50001500  153.94  7.0  14.0
HS50001600  2001  -1  EXT  1.0  1.0
HS50001700  153.94  7.0  14.0
*
*Environment Floor
HS70001000  5  1  0
HS70001001  'Enviro Floor'
HS70001002  0.0  -1.0E-7
HS70001100  -1  1  0.0
HS70001101  0.02  5
HS70001200  -1
HS70001201  'CONCRETE' 4
HS70001300  0
HS70001400  1  700  EXT  0.5  0.5
    transfer, crit pool fracs
HS70001500  153.94  7.0  14.0
HS70001600  0 -700
    with any CV

*A = 2*pi*r*h
*Conv. to enviro., ext HT, crit pool frac.

*A = pi*r^2, CL = r, AL = D

*Insulated boundary on RHS, not associated
* Deck Surface
HS60001000 5 1 0
HS60001001 'Deck Surface'
HS60001002 0.0 -1.0E-7
HS60001100 -1 1 0.0
HS60001101 0.1524 5
HS60001200 -1
HS60001201 'CONCRETE' 4
HS60001300 0
HS60001400 1 600 EXT 0.5 0.5
HS60001500 942.9 12.25 76.9
HS60001600 0 -600

* #nodes, Rect. geo, SS
* Elevation, horiz. orientation
* 5th node at 15.24cm (6in), top of deck
* All meshes concrete
* No internal power source
* Conv. to cont., ext heat tr., crit pool fr.
* \( A = 2\pi r^2, CL = r, AL = \pi D \)
* Ins. bndry on RH, not assoc. w/any CV

* CF/TF INPUT for Heat Slab *

* TF00110 0.0 780.
* Constant temp = 780 K

* RN Input *
* RN input used to edit radionuclide releases

* Activate RN package
RN1000 0
RN1001 10 4 16 14 13 0 0 16 6
RN1100 0.1E-6 50.E-6 1000.
RNCC001 1 2 2 2 2 2 2 2 2 2 2 3 4 2 2
* Place NGs as vapors in atmosphere
RNVG001 600 1. 0.0807 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0.*Xe
RNVG002 600 1. 0.1674 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0. 0.*Kr

* Aerosol Sources - input to containment
* I & Cs
RNAG000 600 2 1.
RNAG001 .00123 .000761 .000997 .00119 .00128 .00126 .00113 .000915 .000675
RNAG002 0.00101

* Deactivate Dep Surfaces
RNDS001 60002 LHS INACTIVE * Ceiling
RNDS002 60003 LHS INACTIVE * Wall
RNDS003  60003 RHS INACTIVE
RNDS004  60002 RHS INACTIVE
RNDS005  50001 LHS INACTIVE  * Na pool surf.

****Add Plot Vbls****
*CVCLT-x-n-cv, x=group, n=total (1) or rad (2)
*CVCLT = mass (kg) excluding deposited
PLOTV2A  RN1-CVCLT-2-1.600  * RNs in Cont.
PLOTV9A  RN1-CVCLT-13-1.600  * Na in Cont.
*RNs in Environment
PLOTV2C  RN1-CVCLT-2-1.700  * RNs in Cont.
PLOTV9C  RN1-CVCLT-13-1.700  * Na in Cont.

*Total Aerosol mass (kg) deposited on HS
*ADEP-s-x-y-hs
*s=1=LHS
*y=1=total mass
PLOTDS2 RN1-ADEP-1-2-1.60001 * RNs on Slab
PLOTDS9 RN1-ADEP-1-13-1.60001 * Na on Slab

* PLOTDF2 RN1-ADEP-1-2-1.70001 * RNs on enviro. floor
PLOTDF9 RN1-ADEP-1-13-1.70001 * Na on enviro. floor

*Sectional Plots
*AMG-w-x-y.cv, w=section, x=class, y=total mass
PLOTVS1 RN1-AMG-1-2-1.600  * Volatiles Section 1
PLOTVS2 RN1-AMG-2-2-1.600  * Volatiles Section 2
PLOTVS3 RN1-AMG-3-2-1.600  * Volatiles Section 3
PLOTVS4 RN1-AMG-4-2-1.600  * Volatiles Section 4
PLOTVS5 RN1-AMG-5-2-1.600  * Volatiles Section 5
PLOTVS6 RN1-AMG-6-2-1.600  * Volatiles Section 6
PLOTVS7 RN1-AMG-7-2-1.600  * Volatiles Section 7
PLOTVS8 RN1-AMG-8-2-1.600  * Volatiles Section 8
PLOTVS9 RN1-AMG-9-2-1.600  * Volatiles Section 9
PLOTV10 RN1-AMG-10-2-1.600  * Volatiles Section 10

*cor* melcor

********************************************************************
************ MELCOR INPUT ************
********************************************************************
TITLE 'SFRv9C6'
* 
RESTART 0 * Restart from cycle 0 *
* 
TEND 86400.0 *
* 
CPULIM 1000.0 CPULEFT 20.0 *
* 
CRTOSU * 80-column output format *
* 
*********** * FILES * *********** *
* 
RESTARTFILE SFRv9C6.RST OUTPUTFILE SFRv9C6.OUT PLOTFILE SFRv9C6.PTF DIAGFILE SFRv9C6.DIA MESSAGEFILE SFRv9C6.MES *
* 
*********************************************************************** *
* TIME STEP AND EDIT CONTROL * 
*********************************************************************** *
* 
* TIME DTMAX DTMIN DTEDT DTPLT DTRST 
TIME1 0.0 1.0 0.001 500.0 250.0 1000.0 
."