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Prototypic Enhanced Risk Monitor Framework and Evaluation

Advanced Reactor Technology Milestone:
M3AT-15PN2301054

September 2015

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Executive Summary

The challenging environments in Advanced Reactors (ARs) increase the possibility of degradation of safety-critical active and passive components and pose a challenge for deployment and extended operation of ARs. Information on component condition and failure probability in these reactor concepts will be critical to maintaining adequate safety margins and avoiding unplanned shutdowns, both of which have regulatory and economic consequences. Technologies that help characterize real-time risk to safe and economic operation can ensure affordability of ARs through optimized operation planning and maintenance scheduling by:

- Maximizing generation through assessment of the potential impact of taking key components offline for testing or maintenance,
- Supporting reduced staffing needs by assessing the contribution of individual components to changes in risk and using this information to optimize inspection and maintenance activities, and
- Enabling real-time decisions on stress-relief for risk-significant equipment susceptible to degradation and damage, thereby supporting optimized lifetime management.

As described in previous reports in this series, probabilistic risk assessment (PRA) provides a static representation of risk associated with operation and maintenance (O&M) of nuclear power plants. Technologies for characterizing real-time risk (so called Enhanced Risk Monitors or ERMs) take into account plant-specific normal, abnormal, and deteriorating states of systems, structures, and components (SSCs) in the estimation of current and future risk to safe and economic operation. Additionally, technologies for characterizing real-time risk provide a mechanism for compensating for the relatively small amount of long-term reliability data from AR systems, structures, and components. The ability to monitor performance and characterize changes in operational risk in real-time can reduce the level of dependence on such performance data. Proactively establishing a viable ERM methodology before AR component design specifications are established also supports: (i) building in opportunities for automated monitoring (on-line and off-line) of those components for optimizing performance with respect to anticipated demands on these reactors; and (ii) improving the maintainability of components from the perspective of time-to-repair and component cost.

This research report summaries the development and evaluation of a prototypic ERM methodology (framework) that includes alternative risk metrics and uncertainty analysis. This updated ERM methodology accounts for uncertainty in the equipment condition assessment (ECA), the prognostic result, and the PRA model. It is anticipated that the ability to characterize uncertainty in the estimated risk and update the risk estimates in real-time based on ECA will provide a mechanism for optimizing plant performance while staying within specified safety margins.

The report provides an overview of the methodology for integrating time-dependent failure probabilities into risk monitors. This prototypic ERM methodology was evaluated using a hypothetical PRA model, generated using a simplified design of a liquid-metal-cooled AR. Component failure data from industry compilation of failures of components similar to those in the simplified AR model were used to initialize the PRA model. By using time-dependent probability of failure (POF) that grows from the initial probability when equipment is in like-new condition to a maximum POF, which occurs before a scheduled maintenance action that restores or repairs the component to “as-new” condition, the changes in core damage frequency (CDF) over time were computed and analyzed.

The project results indicate that, when using the proposed methodology for ERM, as the failure probabilities and failure rates change over time, the CDF changes over time. Repairs or replacements (bringing the components to as-new condition) reduce the risk, although aging of other components may still drive the overall risk higher.

Uncertainty analysis indicated that the ability to propagate uncertainties in various inputs to the ERM provides useful information. Specifically, the uncertainty bounds in the ERM output can have an impact on the ability to perform quantitative assessments of the changes in O&M and safety risk metrics due to component degradation. Improved quantification of the sources of uncertainty will be needed to improve the ability to perform these kinds of trade-off analyses.

In addition, a study on alternative risk metrics for ARs was conducted. Risk metrics that quantify the normalized cost of repairs, replacements, or other O&M actions were defined and used, along with an economic model, to compute the likely economic risk of future actions such as deferred maintenance based on the anticipated change in CDF due to current component condition and future anticipated degradation. Such integration of conventional-risk metrics with alternate-risk metrics provides a convenient mechanism for assessing the impact of O&M decisions on safety and economics of the plant. It is expected that, when integrated with supervisory control algorithms, such integrated-risk monitors will provide a mechanism for real-time control decision-making that ensure safety margins are maintained while operating the plant in an economically viable manner.

Acknowledgments

The work described in this report was sponsored by the Advanced Reactor Technology Research and Development Program of the U.S. Department of Energy Office of Nuclear Energy. The authors gratefully acknowledge Ms. Kay Hass and Ms. Lisa Staudinger for their invaluable assistance in the technical editing and formatting of this report. The authors also thank the technical peer reviewers for their feedback and assistance in improving this report.

Acronyms and Abbreviations

AC	alternating current
AdvSMR	advanced small modular reactor
AFI	aging fractional increase
AST	aging start time
CAFTA	Computer Aided Fault Tree Analysis (system)
CCF	common cause failure
CDF	core damage frequency
CREDO	Centralized Reliability Data Organization (component reliability database)
DOE	U.S. Department of Energy
ECA	equipment condition assessment
EM	electromagnetic
EPRI	Electric Power Research Institute
ERM	enhanced risk monitor
FFTF	Fast Flux Test Facility
F-V	Fussell-Vesely (importance measure derived from probabilistic risk assessments)
HCR	horizontal control rod
HPP	homogenous Poisson process
HRA	human reliability analysis
HTGR	high-temperature gas reactor
ICHMI	instrumentation, control, and human-machine interface
IHX	intermediate heat exchanger
JCS	Job Control System
LMR	liquid metal reactor
LWR	light-water-cooled reactor
MOV	motor operated valve
MTBF	mean time between failures
MTTF	mean time to first failure
NHPP	non-homogeneous Poisson process
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
O&M	operation and maintenance
ORNL	U.S. Department of Energy, Oak Ridge National Laboratory
PDF	probability density function
PHM	prognostics and health management
POF	probability of failure
PRA	probabilistic risk assessment

PSM	planning and scheduling module
RAW	risk achievement worth (importance measure derived from probabilistic risk assessments)
ROCOF	rate of occurrence of failures
RRW	risk reduction worth
RVACS	reactor vessel auxiliary cooling system
SCRAM	an emergency shutdown of a nuclear reactor
SFR	sodium-cooled fast reactor
SGL	steam generator louver
SMR	small modular reactor
SSCs	systems, structures, and components
SWRPRS	sodium-water-reaction pressure relief system
USWL	unplanned shutdown work list

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1.0 Introduction

Advanced reactors (ARs) and advanced small modular reactors (AdvSMRs; based on modularization of advanced reactor concepts) may provide a longer-term alternative to traditional light-water-cooled reactor (LWR) concepts, given their passive safety features and the ability to incrementally add modules over time. Enhancing affordability of ARs which include sodium-cooled fast reactors (SFRs) and high-temperature gas reactors (HTGRs) (Abram and Ion 2008), will be critical to ensuring wider deployment. Critical to this effort will be the management of operation and maintenance (O&M) costs, a significant component of which is the management and mitigation of degradation of components due to their impact on planning for maintenance activities and staffing levels.

The challenging environments in ARs increase the possibility of degradation of safety-critical active and passive components and pose a challenge for deployment and extended operation of ARs. Harsh environments in ARs within the primary and intermediate loops include high temperatures (in excess of 500°C), potential for fast spectrum neutrons, and corrosive coolant chemistry.

The relatively lower level of operational experience with AR concepts (when compared with LWRs) and the consequent limited knowledge of physics-of-failure (POF) mechanisms of materials and components in AR environments, when combined with the potential for increased degradation rates, point to the need for enhanced situational awareness with respect to critical systems. Information on component condition and failure probability is considered critical to maintaining adequate safety margins and avoiding unplanned shutdowns, both of which have regulatory and economic consequences.

Traditional approaches to detecting and managing degradation such as periodic in-service inspections may have limited applicability to ARs, given the expectation of longer operating periods and potential difficulties with inspection access to critical components because of integrated and compact designs. Addressing the need for O&M decision support based on enhanced situational awareness will require techniques to integrate advanced plant configuration information, equipment condition information, and predictive risk monitors are needed to support real-time decisions on O&M (Coble et al. 2013b).

Technologies that help characterize real-time risk to safe and economic operation can ensure affordability of ARs through optimized operation planning and maintenance scheduling by:

- Maximizing generation through assessment of the potential impact of taking key components offline for testing or maintenance,
- Supporting reduced staffing needs by assessing the contribution of individual components to changes in risk and using this information to optimize inspection and maintenance activities, and
- Enabling real-time decisions on stress-relief for risk-significant equipment susceptible to degradation and damage, thereby supporting optimized lifetime management.

Risk monitors are used in current nuclear power plants to provide a point-in-time estimate of the system risk given the current plant configuration (e.g., equipment availability, operational regime, and environmental conditions). However, current risk monitors are unable to support the capability requirements listed above as they do not take into account plant-specific normal, abnormal, and deteriorating states of systems, structures, and components (SSCs). Additionally, technologies for characterizing real-time risk (so called Enhanced Risk Monitors or ERMs) provide a mechanism for compensating for the relatively small amount of long-term reliability data from AR components. Such information was primarily collected from components used in test reactors over a number of years, and is not easily accessible presently (Ramuhalli et al. 2014).

The ability to monitor performance and characterize changes in operational risk in real-time can reduce the level of dependence on such performance data. In parallel, proactively establishing a viable ERM methodology before AR component design specifications are established supports: (i) building in opportunities for automated monitoring (on-line and off-line) of those components for optimizing performance with respect to anticipated demands on these reactors; and (ii) improving the maintainability of components from the perspective of time-to-repair and component cost.

Essentially, ERMs are risk monitors that incorporate the time-dependent failure probabilities from prognostic health management (PHM) systems to dynamically update the risk metric of interest. In this, the ERM methodology differs from other approaches that incorporate aging models for key components. Rather than include generic aging models (for example linear aging models where the failure probability increases linearly over time), the ERM approach uses condition of the component to calculate the failure probability. Such systems may be applied at several levels in the hierarchy of AR or AdvSMR (advanced small modular reactor) systems. For example, component-level PHM systems may be applied to assess the condition of components or subsystems, such as the intermediate heat exchanger. The use of multiple PHM modules provides increased opportunity to monitor the health of critical subsystems within the plant. However, it increases the amount of information that must be aggregated prior to use with risk monitors and in plant supervisory control actions. Figure 1.1 shows a possible scenario for the aggregation; where each PHM module is associated with a risk monitor resulting in predictive estimates of the subsystem health and the associated risk metrics. This information is used to augment data used for supervisory control and plant-wide coordination of multiple modules by providing the incremental risk incurred due to aging and demands placed on components that support mission requirements.

1.1 Research Objectives

The objectives of the research described in this report are:

- Develop and evaluate the ability to augment ERM to include uncertainty bounds and new risk metrics; validate using simulations and experimental data.
- Evaluate the ability to dynamically update the ERM calculation based on real-time updates to information on equipment condition, and evaluate the potential for utilizing these calculations for increasing surveillance intervals for components.
- Examine the potential for tradeoffs between O&M-based risk metrics while staying within allowable safety margins during operation of the AR.
- Provide recommendations for integrating the prototypic ERM framework with likely advanced reactor O&M practices.

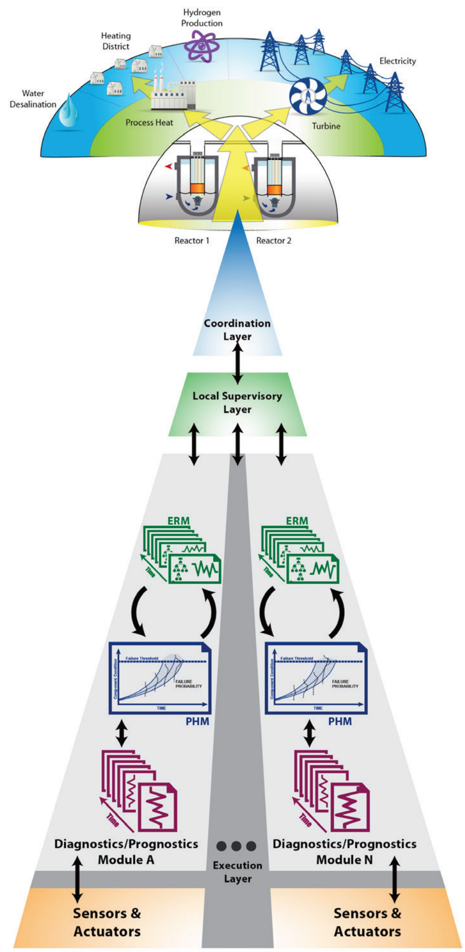

Figure 1.1. Schematic Showing the Integration of PHM Systems with Enhanced Risk Monitors, and Their Location within the Hierarchy of Supervisory Control Algorithms for AdvSMRs

1.2 Objectives of this Report

This research report summaries the development and evaluation of a prototypic ERM methodology (framework) that includes alternative risk metrics and uncertainty analysis. This updated ERM methodology accounts for uncertainty in the equipment condition assessment (ECA), the prognostic result, and the probabilistic risk assessment (PRA) model. It is anticipated that the ability to characterize uncertainty in the estimated risk and update the risk estimates in real-time based on ECA will provide a mechanism for optimizing plant performance while staying within specified safety margins.

Previous reports in this series described gaps and requirements assessments for ERM, an initial methodology for ERM and its evaluation using a simplified model of a two-reactor advanced reactor module (Coble et al. 2013b; Ramuhalli et al. 2013; Ramuhalli et al. 2014). Uncertainty analysis, relative to CDF, conducted using this model was also reported in previous reports. The results described in this report (based on impacting active component O&M using real-time equipment condition information) assess alternative risk metrics and their use in O&M decision-making, that if integrated with AR supervisory plant control systems, can help control O&M costs and improve affordability of ARs.

1.3 Organization of Report

This technical report is organized as follows. Section 2 provides background information for the ERM, including a summary of benefits, requirements and assumptions. Section 3 briefly describes the prototypic ERM framework. Section 4 provides the results to date of the evaluation of the ERM framework and documents simple scenarios wherein multiple risk metrics may be used for O&M decision making. Section 5 summarizes the research to date, and planned future activities are discussed in Section 6. Additional details about the PRA models used, equipment condition assessment methods, component failure data, and advanced reactor simulation models and potential O&M practices are provided in Appendices A through E.

2.0 Background

The vast majority of nuclear power plant (NPP) operating experience involves LWRs and includes small LWRs. ARs generally encompass all non–LWR concepts, and are being considered as a longer-term option for meeting electrical generation and process heat needs in the United States (Abram and Ion 2008). Among the concepts being considered are sodium-cooled fast reactors (SFRs) and HTGRs, both of which have some operational history in the United States and elsewhere. A detailed description of these concepts is available in previous reports in this series (Meyer et al. 2013a). Additional details of AR concepts as they apply to AdvSMRs and likely O&M approaches are provided in the previous reports in this series associated with AdvSMR prognostics and ERM research (Coble et al. 2013b; Meyer et al. 2013a; Ramuhalli et al. 2013; Ramuhalli et al. 2014).

There is some experience with select AR concepts, which may be used to identify potential faults and failure modes for key components in AR concepts. Some of these issues are expected to be resolved in new AR designs (e.g., moisture intrusion through water-lubricated bearings may potentially be avoided by using sealed magnetic bearings), while other issues may still be relevant (though relevant data may not be easily accessible). These issues are likely to drive inspection and maintenance requirements for ARs.

Below, we briefly summarize information from previous reports (Coble et al. 2013b; Meyer et al. 2013a; Ramuhalli et al. 2013; Ramuhalli et al. 2014) in this series on ERM needs (including ECA and PRA) for nuclear power applications, technical assumptions that bound the research described in the rest of this document, and the approach taken to evaluate the ERM methodology.

2.1 Overview of Enhanced Risk Monitors (ERMs) for Advanced Reactors

Advanced plant configuration information, equipment condition information, and risk monitors are needed to support frequently changing plant configurations (Yoshikawa et al. 2011). To utilize these three, often disparate pieces of information in making real-time decisions on O&M, two separate technologies need to be integrated:

- Risk monitors (that currently are based on PRA models)
- Diagnostic and prognostic technologies for determining, based on the operational history and current configuration of the unit and its components, the present state of the component (for instance, “likely to continue operating within specifications,” or “likely to fail soon with some probability,” etc.) and its probability of failure over a given time horizon.

The integration of these two technologies results in ERMs that use the real-time information on equipment condition to provide real-time updates to risk metrics. Essentially, ERMs would incorporate the time-dependent failure probabilities from PHM systems to dynamically update the risk metric of interest. These envisioned PHM systems would provide condition indicators for relevant equipment using online, in-situ sensors and measurements to support detection and identification of incipient failure and to reflect evolving degradation. In this, the ERM methodology differs from other approaches that incorporate aging models for key components. Rather than include generic aging models (for example, linear aging models where the failure probability increases linearly over time), the ERM approach uses condition of the component to calculate the failure probability. Details of the ERM methodology are provided in Coble et al. (2013b); Ramuhalli et al. (2013); Ramuhalli et al. (2014).

Relevant SSCs are generally those that are considered risk-significant, although this list can change as the plant configurations and operational conditions change. It is important to ensure that in determining relevancy such factors are considered. These key SSCs are then candidates for ECA.

The ability to predict (or estimate for future times) the POF based on ECAs and incorporate these in ERM may also help compensate for a relative lack of knowledge about the long-term component behavior of some components that are being proposed for ARs.

2.1.1 Equipment Condition Assessment (ECA)

ECA process measurements (e.g., flow, temperature, and pressure) or performance measurements (e.g., pump efficiency) are used as input to the ECA. Generally speaking, ECA methods rely on change detection techniques (Coble et al. 2013b) to identify departure from normal operation and characterize the condition in terms of various condition indices. Challenges from the harsh environments in ARs, including AdvSMRs, may necessitate novel measurement methods, such as optical (Anheier et al. 2013) measurements of process parameters, or the use of sensors tolerant to these conditions (Daw et al. 2012).

Health monitoring would provide condition indicators for key equipment using online, in-situ sensors and measurements to support detection and identification of incipient failure and to reflect evolving degradation. This is particularly important for SSCs proposed for use in AR designs that differ significantly from those used in the operating fleet of LWRs (or even in LWR-based SMR designs), as operational characteristics for these SSCs may not be fully available.

As discussed in Ramuhalli et al. (2013), the risk significance of active components in AdvSMRs may increase in spite of the greater reliance on passive mechanisms for safety goals. In combination with the potential for reduced access for testing and maintenance of in-vessel or in-containment components, this points to the need for greater condition monitoring of select active components with the goal of obtaining equipment condition in near real-time. Determining whether available condition monitoring techniques may be applicable to these components is a necessary step to leveraging existing technologies to the fullest extent possible.

Given that components with similar functionality are anticipated in proposed AR and AdvSMR concepts, the ability to monitor performance and characterize changes in operational risk in real-time can reduce the level of dependence on such performance data, while providing the technical basis for optimizing (with respect to economics) operations and maintenance actions and at the same time maintaining adequate safety margins (Ramuhalli et al. 2014).

2.1.2 Probabilistic Risk Assessment (PRA)

Current risk monitors use PRA techniques that have been used in U.S. nuclear power plants to assess the risks associated with operation since the 1980s (Wu and Apostolakis 1992). PRA systematically combines event probability and POF for key components to determine the hazard probability for subsystems and the overall system (Kafka 2008). In general, PRA models use a static estimate for event probability and POF, typically based on historic observations and engineering judgment. These methods do not take into account the current condition of the components, and are susceptible to error in probabilistic estimates. More recently, time-based POF values have been used (Vesely and Wolford 1988; Arjas and Holmberg 1995); however, these are derived from operating experience and traditional reliability analysis and are usually not specific to the operating component.

While conventional risk metrics (specifically core damage frequency or CDF) may be utilized in this framework, it is likely that the real value of ERMs is with respect to alternative risk metrics that address risk from an O&M perspective. These alternative risk metrics can be characterized by non-safety risk measures such as availability, productivity, ability to meet demand, and probability of mission completion. However, O&M-based risk metrics will need to be balanced with safety metrics to ensure that plant performance and maintenance schedules can be optimized to reduce cost while staying within specified safety margins.

2.1.3 Role of ERMs in Advanced Reactor Control and Coordination

Proactively establishing a viable ERM methodology before AR component design specifications are established supports: (i) building in opportunities for automated monitoring (on-line and off line) of those components for optimizing performance with respect to anticipated demands on these reactors, and (ii) improving the maintainability of components from the perspective of time-to-repair and component cost.

Further challenging existing O&M practices is the expectation that ARs will operate in regimes that are removed from the current base-load generation regime. Thus load-following, reactor run-backs and load-balancing in multi-module reactors are all likely operational regimes for ARs. U.S. experience with these modes is limited and overseas operating experience suggests that these modes may result in added, potentially unanticipated, wear and tear on several components (such as control rod drive motors).

In SMR/ICHMI/ORNL/TR-2013/03, Cetiner et al. (2012) describe the rationale for designing a supervisory control system for AdvSMR plants, based on the financial incentive to reduce staffing requirements and to enhance plant availability, and on the more complex operating regime expected for AdvSMR plants. This rationale also holds for AR plants. However, supervisory control systems in this report are for non–safety-related systems, independent of reactor protection systems, although it is required to not interfere with safety systems.

AR plants are generally expected to have more than one reactor per plant; supervisory control can simplify the operator’s work load in managing startup, load changes, and shut down of the individual reactors. Supervisory control may allow automatic load following and transfer of heat load between electric generation and process heat. Beyond operations, the supervisory control can provide for automated diagnosis of failed/failing components and automated plant response to isolate the failure, and monitor condition of equipment to allow operations and maintenance personnel to avoid overloading failing equipment and to make preparations for repair. The supervisory control system design in Cetiner et al. (2012); Cetiner et al. (2013) is intended not only for failure diagnosis but also for providing a continuously updated estimate of the condition of at least the most important plant components. This technology has been demonstrated in non-nuclear applications, particularly in aerospace, and is likely to be of value in the operation of ARs.

Previous work on this project (Coble et al. 2013b; Ramuhalli et al. 2013; Ramuhalli et al. 2014) has demonstrated the ability to provide estimates of predictive risk based on the actual condition, as well as the ability to update these estimates as new information about the condition of individual SSCs becomes available. Uncertainty (from various sources) plays a role in the predictive risk estimates, and methods for uncertainty quantification and propagation were studied. While the majority of the work has focused on CDF as the risk metric of interest, alternative risk metrics (including economic metrics) are being developed and evaluated.

Given the possibility of frequently changing demands on ARs, techniques to integrate advanced plant configuration information and predictive risk monitors are needed to support real-time decisions on plant operations (Coble et al. 2013b). Such information may be applied at several levels in the hierarchy of AR systems. For example, component-level PHM systems may be applied to assess the condition of components or sub-systems, such as the intermediate heat exchanger. The use of multiple PHM modules provides increased opportunity to monitor the health of critical sub-systems within the plant. However, it increases the amount of information that must be aggregated prior to use with risk monitors and in plant supervisory control actions. The example in Figure 1.1, taken from SMR/ICHMI/PNNL/TR-2014/01 (Ramuhalli et al. 2014), shows a possible scenario for the aggregation of multiple PHM modules and ERMs; where each PHM module is associated with a risk monitor resulting in predictive estimates of the subsystem health and the associated risk metrics. This information is used to augment data used for supervisory control and plant-wide coordination of multiple modules by providing the incremental risk incurred because of aging components and demands placed on those components to support mission requirements.

While these studies have demonstrated the feasibility of the ERM, the ability to incorporate it into O&M practices will be vital to its deployment in ARs. Information provided by the ERM can significantly simplify O&M planning through increased situational awareness, and compensate for uncertainties about component reliability in ARs. Determining the requirements for integrating the ERM into O&M practices will be key. A series of recommendations for this purpose is described in this document, including in Section 2.2 herein.

2.1.4 Risk-Informed Surveillance and Surveillance Test Interval Extension

In recent years, industry and the regulators in the United States have initiated and implemented a series of initiatives that use the contribution to plant risk to determine the timing of specific maintenance actions. A major element of this shift has been the Surveillance Test Interval Extension, whereby the set of components for testing and inspection was moved from the plant's technical specifications to licensee control. Under this Risk Management Technical Specifications Initiative 5b, Surveillance Frequency Control Program (NRC 2011), the licensee is committed to the following actions:

- Retain surveillance requirements (SR) in Tech Specs
- Relocate SR frequencies/test intervals to licensee-controlled document (i.e., Technical Requirements Manual)
- Surveillance test interval adjustment
- Change interval based on risk-informed process

The approach is tempered by performance and commitments, and requires the development of a detailed series of process steps.

2.2 Benefits of Prototypic Enhanced Risk Monitoring Technologies for Advanced Reactors

A primary challenge to wide deployment of ARs is the relatively lower level of operational experience with AR concepts (when compared with LWRs), and the consequent limited knowledge of POF mechanisms in advanced reactor environments. Information on AR active and passive (Ramuhalli et al. 2014) component condition and failure probability is considered critical to maintaining adequate safety margins and avoiding unplanned shutdowns (which have regulatory and economic consequences), and for providing sufficient lead-time for planning O&M activities. Technologies that provide improved awareness of system condition, when integrated during design of the AR, can provide the tools necessary for quantifying and maintaining the operational envelope for safe economic O&M. Given the possibility of frequently changing configurations in AR concepts, including AdvSMRs, to meet multiple mission goals, and the aforementioned relative lack of reliability data, techniques to integrate advanced plant configuration information, equipment condition information, and predictive risk monitors are needed to support real-time decisions on O&M (Coble et al. 2013b).

In their use of real-time component condition, ERM technologies differ from conventional risk monitors (Wu and Apostolakis 1992; Kafka 2008) that use a static estimate for event probability and POF, typically based on historical observations and engineering judgment. More recently, time-based POF values derived from operating experience and traditional reliability analysis have been used (Vesely and Wolford 1988; Arjas and Holmberg 1995); however, these are usually not specific to the component. Critical to the ERMs is a predictive estimate of POF of the component, which is precisely what PHM provides(Coble et al. 2012). As a result, PHM technologies are likely to be applicable to achieving enhanced risk monitoring to obtain a realistic assessment of dynamic risk that is unit-specific and accounts for the operational history of the component (Ramuhalli et al. 2013). Therefore, ERM systems are expected to play a vital role in AR operations specifically by incorporating real-time component condition into the calculation of plant risk [usually measured in terms of CDF or some other safety-related risk metric (Coble et al. 2013b; Ramuhalli et al. 2014)].

The anticipated objective of interfacing enhanced risk monitoring to the supervisory control modules is to provide the control logic with a series of options that account for plant equipment condition and the risk to mission of operating the reactor given the current equipment condition, and predictive estimates of remaining life and POF provided to the ERMs from prognostic modules. In general, we expect that the ERM output will be utilized by the supervisory control logic as well as by the Planning and Scheduling Module (PSM) (for scheduling maintenance actions). The PSM modules are expected to generate a partial schedule, work list, and parts list needed to restore the SSC during the next outage. In addition, prognostic information that indicates the rate of degradation of SSC may be needed by the supervisory control algorithms to indicate an automatic trip of equipment that is suffering fast degradation (though this may be better handled by trip devices on SSCs). The supervisory control logic may utilize additional information (such as diagnostic information from the ECA) in its decision making process.

At a minimum, the information provided by the ERM should be sufficient for each entity to perform its function. Given the likely needs from AR plant O&M practices (Ramuhalli et al. 2014), and the potential outputs available from the ERM, the following recommendations may be made relative to requirements for ERM to integrate with O&M practices:

- <u>Support Initiatives for Risk-informed Maintenance/Operations</u>: Given the increased emphasis on risk calculations to inform maintenance actions for current reactors, it is likely that as ARs come online, similar initiatives will be applied from both licensees and regulatory agencies. The ERM is one possible option for improving the accuracy of calculations to support risk-informed initiatives such as extension of surveillance test intervals. The ERM keeps track of the dynamic value of component failure rates, and can forecast failure rates associated with component failure and unavailability over time. Further, the extent to which in-service inspection extension contributes to CDF is predicted dynamically by the ERM.
- <u>Support Multiple Operational Options</u>: As SSCs age and degrade, normal plant operation with these SSCs is likely to increase the risk to mission. Under these circumstances, supervisory control logic for AR will need to support operational modes that reduce demands on SSC experiencing degraded condition, at least until the next available opportunity for maintenance and return to serviceable condition. This will require the ERM to provide the predictive risk calculations for multiple

operational options (such as full power operation vs. operating at 80% of rated power). This information will enable the supervisory control module and/or the PSM to evaluate the appropriate tradeoffs (for instance, revenue generated by running the plant in a degraded state vs. incremental risk in terms of cost and safety metrics).

- Support Multiple Missions and Fluctuating Generation Demands: As discussed in Coble et al. (2013b), ERMs must be able to account for frequently changing demands and load mismatch from multiple interconnected modules, as well as changing balance-of-plant configurations to meet multiple mission needs. This assessment of predictive risk over some time horizon for multiple missions will need to be integrated with the computation of predictive risk for multiple operational options because of degraded SSC.
- Support Cost-Benefit Analysis: Using the information on incremental risk for multiple operational options, the ERM and/or the PSM needs to evaluate not only success probability of operating particular SSCs until the next scheduled outage, but also (if possible) to calculate the probability of doing more expensive-to-repair damage to SSC by continuing to run them. This is effectively a cost-benefit analysis. For example, minor failure of a bearing can frequently be repaired by just replacing the bearing material itself, without having to rework the shaft. Continued operation at loose clearances can damage the shaft itself, considerably complicating repair and ratcheting cost upwards.
- Identify Key Contributors to Predicted Risk: Importance measures are one approach to identifying key contributors to plant risk. For the ERM to be useful in assisting with updating and optimizing O&M schedules, methods to extend importance measures (or related quantities) in a predictive manner are needed. The result is a clear understanding of the major contributors to plant risk, and an assessment of how these change as SSCs age, degrade, and change the incremental predicted risk.
- Support Risk Evaluation over Multiple Time Horizons: Ideally, the operation of the plant with components in their current (likely degraded) state can be extended to the next scheduled maintenance outage. However, if there is no option that will make it to the next planned outage (above some predefined minimum success probability), then other time periods will need to be considered. A useful practical minimum for remaining run-time might be to run long enough to obtain parts and materials for repair. The information provided by the ERM to the PSM/supervisory control modules will need to be a function of managing O&M events (e.g., maintenance planning and scheduling, outage planning and execution, etc.).
- Unconventional Predictive Risk Metrics: Traditional risk metrics such as CDF provide a measure of the risk associated with safety-related consequences of the plant configuration. However, alternative risk metrics that address risk from an O&M perspective, such as the risk of plant trip and unplanned outage frequency, will be required when attempting to balance competing needs in O&M. O&M-based risk metrics will need to be balanced with safety metrics to ensure that plant performance and maintenance schedules can be optimized to reduce cost while staying within specified safety margins. It is proposed that a focus be put on maintenance schedules and component state of health. Using ERM to gather real-time component status, accurate plant operational condition can be determined with minimal uncertainty. Using the currently known component condition, estimates of predicted failure can be made and aligned with O&M expectations and goals. This will allow better organization of resources for both unplanned and planned outages as key areas of maintenance can be pinpointed ahead of time.
- Provide On-demand and Periodic Outputs: The ERM needs to provide output periodically (say hourly or daily) during steady state, as well as “on demand” when plant power or power split (e.g., electrical to thermal) changes by some threshold amount. In addition, it would be useful to the operations and maintenance staff for the PSM to provide risk (success probability and/or PRA impact) on-demand for proposed on-line maintenance.

- Verification of Correct Operation: A mechanism for self-test or test by the supervisory control module might be useful to verify that the ERM module(s) are operating properly, and are not suggesting counterproductive or unnecessary action. However, it is unclear at this stage how such verification may be accomplished.
- Uncertainty Quantification: Methods for identifying sources of uncertainty in the ERM and quantifying and propagating these uncertainties to the ERM output will be needed. The total uncertainty in the predicted risk plays a key role in determining the incremental risk from SSC degradation.

2.3 Requirements and Assumptions

2.3.1 Summary of Technical Requirements for ERM

SMR/ICHMI/PNNL/TR-2013/02 (Coble et al. 2013b) focused on the technical gaps in development of ERMs for active components in AdvSMR designs by integrating real-time information about equipment condition and POF into the risk monitor framework. This included defining a number of requirements for enhanced risk monitors that integrate real-time estimates of equipment condition. These requirements were derived from expected operational characteristics of proposed AdvSMRs and include the ability to:

- integrate online, real-time ECA
- apply to multiple, interconnected modules and generation blocks
- evaluate risk over multiple time horizons
- apply condition-specific fault trees, event trees, and success criteria
- support reconfigurable balance-of-plant and fluctuating generation demands
- evaluate multiple risk measures
- meet runtime requirements for control and O&M planning

Follow-up technical reports (Ramuhalli et al. 2013) proposed a preliminary methodology for ERMs with ECA to address some of the technical gaps highlighted earlier. This ERM methodology addresses changes (i.e., degradation) in the failure rate of a component that might be expected to normally occur over the component life, and begins by defining PRA models that include all relevant components (based on failure modes and effects analysis that accounts for all potential operating conditions) and interdependencies between different modules of AdvSMRs. Subsequent realignment of this research within the DOE Advanced Reactor Technologies (ART) Program expanded the considerations for the deployment of this ERM methodology to ARs in general.

This report describes progress towards increasing the realism of the ERM models for ARs through incorporation of uncertainty at several levels, particularly as available POF data is updated (nominally through the use of real-time condition assessment of key components).

2.3.2 Potential Risk Metrics to Support O&M Optimization

Risk metrics in PRA modeling are intended to capture the frequency of occurrence of undesirable consequences (e.g., reactor unavailability, core damage, release of radioactivity). A common metric in Level 1 PRA models [see Figure 3.1 in Ramuhalli et al. (2014) for definitions of PRA levels] is the frequency of accidents that can result in core damage (i.e., CDF).

While this is a useful metric for AdvSMRs and is used in this assessment, the increased reliance of these designs on passive safety features is likely to result in very low CDF values that reduce the utility of this particular metric. Instead, given the need to reduce O&M costs, metrics that capture the risk of plant unavailability to meet its mission needs (whether electrical generation or process heat or some combination of the two) are likely to be of more relevance. In particular, such metrics provide a quantitative mechanism for understanding the impact to mission of the probability of component failure and consequent unavailability.

2.3.3 Technical Assumptions

Several key assumptions are made in the development of the preliminary methodology for ERM that integrates time-dependent failure probabilities that are specific to the unit and the component condition. These are described in Ramuhalli et al. (2013), and are summarized below for convenience.

- The key aspects of the ERM methodology may be developed and initially assessed using a simplified model of an AR. In particular, we assume that the simplified model is of a liquid-metal-cooled AR.
- The focus of the ERM methodology described in this report is on active components in ARs that are included in risk monitors.
- Effective ECA techniques are assumed to be available for key active components and systems, including identification of the measurements necessary to perform ECA.
- Sensors for making the measurements needed for effective ECA are assumed to exist. These include measurements that are sensitive to component condition (such as vibration or current/voltage) as well as measurements of the operational environment (stressors). Ongoing research into sensors [such as that documented in SMR/ICHMI/PNNL/TR-2013/04 Anheier et al. (2013) and Daw et al. (2012)] will be leveraged where possible.
- We assume that existing prognostic algorithms will provide accurate extrapolation of equipment condition through future operation, as well as confidence bounds on the extrapolation; new approaches to prognosis are not a focus of this research. Investigations into PHM including risk assessment of passive components are covered separately as summarized in the report on prototypic prognostic techniques for AR passive components (Meyer et al. 2013b). Developments in this area, with appropriate modifications to address active components, will be leveraged as needed.
- For the initial assessment of the ERM methodology, POF estimates at future time instants for the components identified in the simplified AR design (Appendix A) are assumed to be available; however, the specific ECA technique and prognostic algorithm are not defined at this stage.

The development of the ERM methodology was also driven by the functional requirements for ERMs [details are in Coble et al. (2013b)]. However, the preliminary methodology addresses only a sub-set of these requirements, with additional development necessary to address the other requirements.

2.4 Evaluation Approach for Prototypic ERM Methodology

The evaluation of the ERM methodology uses PRA analysis of the simplified two-module liquid-metal cooled reactor concept described in Appendix A. The example design is defined to provide a simple level of abstraction but contains enough resolution and specific design elements to inform the development of a PRA model that, when quantified, produces a cogent set of results. A simplified PRA model is used for computing the safety-related risk metric (e.g., CDF); while a related economic model is used to generate the data for evaluating the alternate (economic) risk metrics. For the analysis conducted to date, nominal values for the inputs (such as failure probabilities of equipment, cost of equipment replacement, and fraction of equipment replaced) are being assumed, with the assumption that accurate values for these (and other quantities) will be available as AR concepts get closer to operational deployment. Where possible, these values are updated with available component failure data (such as from NUREG/CR-6928 (Eide et al. 2007)). While the safety risk metric calculations are conducted using the usual procedures utilized within the nuclear industry, the assessment of economic risk requires a change in the way that cut-sets are identified within the risk monitor. In this case, instead of identifying event sequences that can lead to core damage, the new assessment may require the identification of event sequences that can lead to economically undesirable consequences, such as plant shutdown. The results of the two risk metric calculations are integrated to enable risk-informed decision making. Evaluation criteria are qualitative at this stage, and are based on whether the incorporation of ERM decreases the projected cost of O&M decisions while enabling the plant to maintain adequate safety margins.

3.0 Prototypic ERM Framework Summary

This section briefly describes the methodology for prototypic enhanced risk monitors that integrate equipment condition assessment for dynamic characterization of system risk. Details of the methodology have been documented in previous reports (Coble et al. 2013a; Coble et al. 2013b; Ramuhalli et al. 2013; Ramuhalli et al. 2014) and are summarized here for completeness.

ECA is a requirement for ERM, and as discussed in Section 2.0, techniques for ECA are assumed to exist for the selected components of an AR. Thus, the state-of-the-art for ECA constrains the ability to deploy the ERM methodology and a better understanding of the state-of-the-art for ECA is needed before research needs for ECA of AR components may be defined.

This section begins by briefly describing the ERM methodology for the sake of completeness, including the general approach to integrating ECA/prognostics results with risk monitors. Factors impacting the ability to accurately assess risk in the ERM are then discussed.

3.1 ERM Methodology

As described earlier, ERMs require integration of two sets of technologies—risk monitors and ECA/prognostics. In this section, we provide an overview of the approach to this integration.

Time-independence of component failures is assumed in traditional PRA modeling, and PRA component failure rates are typically assumed to be static over the life of the component. Changes (i.e., degradation) in the failure rate of a component that might be expected to occur normally over the component life are not explicitly represented.

However, experience has shown that aging of components generally results in time-dependent failure rates (Vesely and Wolford 1988). In reliability engineering, the failure probability is often defined to be a "bathtub" curve (Ramuhalli et al. 2013; Ramuhalli et al. 2014, Figure 3.1), though linear aging models are common in the risk monitor arena (Ramuhalli et al. 2014).

The ERM methodology removes the fundamental assumption of static failure rates in risk monitors by integrating component-specific time-dependent failure probabilities that are calculated based on the current condition of the equipment.

We begin by defining PRA models that include all relevant components, as well as interdependencies between different modules of ARs. Component relevancy is determined by performing a failure-modes-and-effects analysis that takes into account all potential operating conditions (e.g., full power steady-state operation, load-following, and reactor run-back). This information is used in the development of fault-trees and event-trees of the PRA model. These are solved to identify the cutsets that contribute most to risk.

For each of the relevant components, ECA methods are deployed to monitor the condition of the equipment and the surrounding environment. This information is used by a prognostic algorithm to predict the POF at a specified future time given the current condition of the component. As additional measurements become available (for instance at successive time instants) the predictions may be improved by making use of updated condition information.

The component-specific time-dependent failure information (POF and confidence bounds as a function of time) is then integrated into the PRA model and the PRA model is solved to provide a time-dependent risk measure (such as the change in CDF with time).

3.2 Importance Measures for CDF

Existing importance measures are based on the use of static failure rates, and may be less useful when applied to a model where failure rates and the calculated CDF change over time. A primary reason for this is the manner in which traditional importance analysis is generally performed; that is, through the use of ratios. This may be understood using a simple example where risk achievement worth (RAW) is expressed as the ratio of the risk calculated with the element (e.g., basic event) being in a failed state or otherwise unavailable and the baseline risk (Vesely et al. 1983). In the case where the baseline CDF changes with time (as does the POF), assuming a component is fully available does not change the time-dependency of the CDF (because other components are still assumed to have time-dependent POF values), although the values may be different from the baseline case. Because of division by small numbers; taking ratios under these circumstances may result in large excursions in the risk reduction worth (RRW) that mask important details. Consequently, a failure event with a high-importance value at a given point in time might not be as important as a lower importance value at another point in time.

A more useful measure of importance must include consideration of the relative importance of the event to the total CDF as well as the value of total CDF itself. We proposed a new importance measure in Ramuhalli et al. (2013), in which the component failure of interest is set to a value of 1.0 (i.e., the component is assumed unavailable), the total CDF recalculated, and ratio of the CDF to a target CDF is calculated. This approach examines the relative increase in risk over the time-horizon of interest (when compared to a static or time-independent risk profile) due to the unavailability of a component. Other options for importance analysis may also be of relevance and need investigation.

3.3 Uncertainty Estimation in ERMs

Uncertainty in PRA modeling arises from a number of sources that are typically divided into aleatory variability and epistemic uncertainty (EPRI 2011). Aleatory variability is related to the statistical confidence we have in failure probability data, while epistemic uncertainty is related to the uncertainty in the accident sequences used to develop the PRA model. Epistemic uncertainty is dealt with by developing event- and fault-trees as complete as possible, identifying keys sources of uncertainty, and performing sensitivity analyses. The aleatory variability is addressed explicitly by propagation of parametric data uncertainty for initiating basic event data. Uncertainty analysis is performed through a sampling strategy (e.g., Monte Carlo sampling) over some number of observations.

When incorporated in an ERM framework, several sources of uncertainty exist that directly impact the uncertainty analysis performed with PRA models. These include but are not limited to:

- Measurement uncertainty, including factors such as calibration tolerances, sensor and instrumentation precision, external factors such as poor coupling, etc.
- Stochastic variability in stressors
- Manufacturing variability, leading to variability in failure rates
- Manufacturing defects that can lead to rapid failure of components (so-called infant mortality)
- Variability in degradation levels at which components fail

These sources of variability result in uncertainty in both the ECA and the predicted POF. In turn, these uncertainties are expected to impact the predicted risk estimates from the ERM. In order to utilize the ERM results in a meaningful manner, the various uncertainties will need to be propagated through the ERM methodology to produce estimates of uncertainty in the ERM output.

Methods exist to account for uncertainty in conventional risk monitors. However, as with component failure rates, these uncertainties are generally static and when propagated through the PRA models, result in static estimates of uncertainty.

A number of other mechanisms exist that can help study uncertainty propagation. Many of these methods, such as Latin Hypercube, are based on statistical sampling mechanisms. These techniques utilize models of the data that relate one or more explanatory variables to the observed data, and use probabilistic sampling mechanisms to propagate uncertainty.

In Ramuhalli et al. (2013), an initial assessment of sensitivity of ERM outputs to variation in component failure rates was performed. This was a limited assessment, with a small variability in the initial failure rates for one component was assumed and propagated forward through the ERM methodology. The approach assumed that the aging rate (or equivalently the rate at which the probability of failure of components increases with time) was unchanged. Results appeared to indicate that the small variation in initial failure rates resulted in a relatively small change in the CDF at future times, though the exact sensitivity was not quantified.

In this study, we explore this further. Specifically, we assume that mechanisms to quantify uncertainty in the various inputs exist and can be leveraged for uncertainty quantification in the component condition (based on the available measurements). Further, we assume that prognostic techniques can also utilize statistical sampling approaches to quantify the uncertainty in predicted POFs. An example of this approach is particle filter-based prognostics, which can be used to estimate the uncertainty bounds for the predicted time to failure.

We then systematically vary the input uncertainty bounds and examine the impact on the predicted risk. To simplify the problem, we assume that the uncertainties are compounded over time (simulating the effect of increasing uncertainty in the POF with time), with the exact behavior of uncertainty with time based on prognostic calculations. Figure 3.1 shows an example of this calculation (using the CDF as the risk metric) using the PRA model described in Appendix A. The horizontal red line represents the nominal goal for CDF beyond which the plant is assumed to be outside its safety margins. As seen in this example, the uncertainty associated with the predicted risk plays a decisive role in determining when the risk metric is considered to exceed the safety goal. As seen from this plot, and results presented previously (Ramuhalli et al. 2013; Ramuhalli et al. 2014), O&M decisions based on periodic condition monitoring and updates to the predicted risk are also likely to be dependent on the uncertainty bounds.

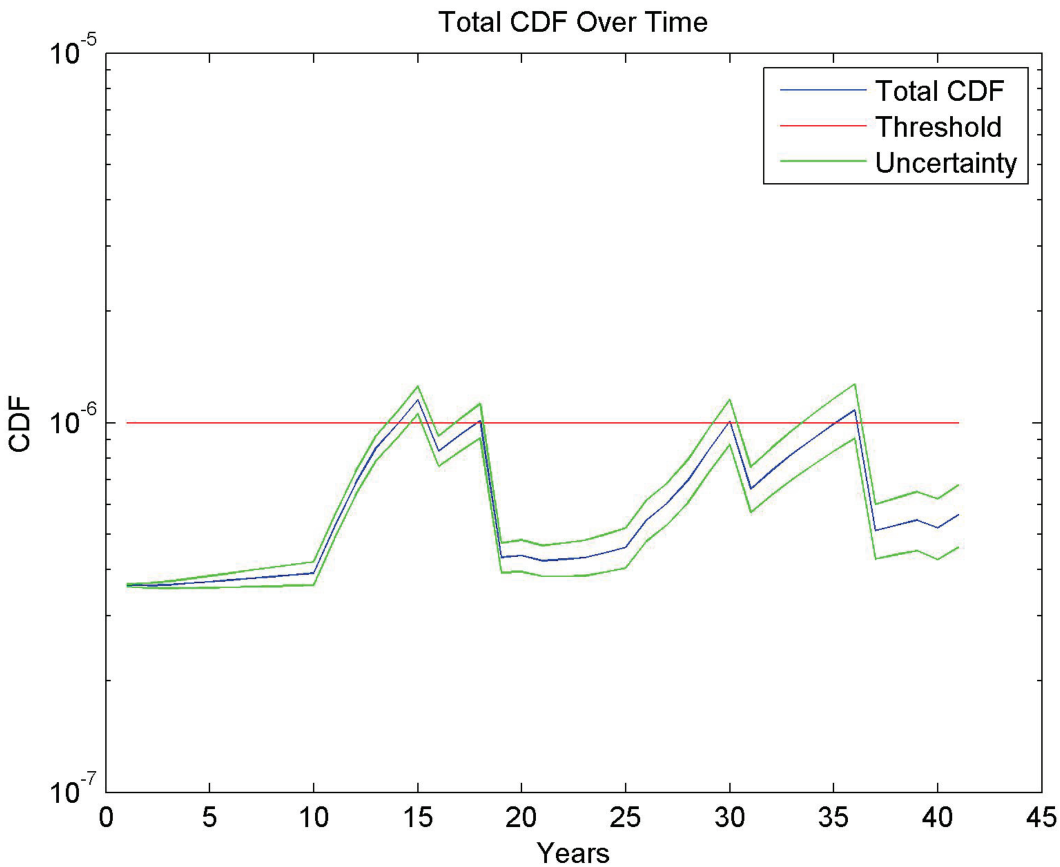

Figure 3.1. Aging Failure Rate Model for a Component

3.4 Risk Metrics for ERMs

Advanced reactors, and AdvSMRs (based on modularization of AR concepts) may provide a longer-term alternative to traditional LWR concepts. Information on component condition and failure probability in these reactor concepts will be critical to maintaining adequate safety margins and avoiding unplanned shutdowns, both of which have regulatory and economic consequences. In particular, information on component condition will be needed for characterizing the risk (in terms of both safety and economic metrics) to optimize O&M planning and control O&M costs by

- Maximizing generation through assessment of the potential impact of taking key components offline for testing or maintenance.
- Supporting reduced staffing needs by assessing the contribution of individual components to changes in risk and using this information to optimize inspection and maintenance activities.
- Enabling real-time decisions on stress-relief for risk-significant equipment susceptible to degradation and damage, thereby enabling lifetime management.

System risk in current nuclear power plants is computed using risk monitors that provide a point-in-time estimate of risk given the current plant configuration (e.g., equipment availability, operational regime, and environmental conditions). Traditional risk metrics such as CDF provide a measure of the risk associated with safety-related consequences of the plant configuration. However, alternative risk metrics that address risk from an O&M perspective may also be of value, particularly in ARs and AdvSMRs. O&M-based risk metrics will need to be balanced with safety metrics to ensure that plant performance and maintenance schedules can be optimized to reduce cost while staying within specified safety margins.

Unplanned outages are assumed to be mitigated for components whose condition can be monitored. The ERM measurements are translated into a failure rate and plugged in to PRA model to derive the CDF estimate. A trend recognition and prognostic algorithm together predict the time of next unplanned outage. The top components that significantly contribute towards CDF in excess of the safety goal are accordingly scheduled for repair during planned outages at regular intervals. However, there is still a possibility of unplanned outages owing to conditions beyond ERM control such as random failures.

An importance ranking mechanism, such as the Fussell-Vesely importance measure, is used to rank components that are likely to cause safety goal to be breached sometime within the reactor's lifetime in the future. These, say *n* components are scheduled for regular maintenance activity, say every *k* years coinciding with a planned outage.

As discussed in earlier reports (Ramuhalli et al. 2013), the initial analysis indicated that, using available component failure data, the overall CDF in the example PRA model was orders of magnitude smaller than those generally accepted for currently operating reactors. This is likely because of the small number of key components used in the PRA modeling as well as the use of passive safety features in ARs and AdvSMRs. However, this is an expected feature in advanced reactor PRA modeling, as typical risk measures such as CDF are expected to be lowered because of the inclusion of passive safety mechanisms. Although the CDF was seen to increase over time (and eventually exceeds levels generally accepted for operating reactors), it is because of a potentially inflated rate of change of the POF over time.

Along with identifying appropriate risk measures, criteria need to be established to assess the acceptability of plant configurations based on risk results (Puglia and Atefi 1995). Establishing acceptance criteria for different risk measures is an operational issue that will be considered in conjunction with the development of supervisory control and O&M planning algorithms, although site-specific acceptance criteria will likely need to be developed by utilities and regulators.

4.0 Evaluation of Prototypic ERM Framework

4.1 Preliminary Assessment of Prototypic ERM Methodology for Advanced Reactors

The assessment of the ERM methodology uses PRA analysis of the simplified-model AR design depicted in Figure 2.3 of Ramuhalli et al. (2014). The design only shows frontline components and supporting systems such as alternating current (AC) and direct current electrical power systems, instrumentation, and the details of the reactor trip system are not shown in the figure. Unlike the assessments described in previous reports (Ramuhalli et al. 2013), the PRA model used in the evaluations in this document incorporates additional details, including supporting systems and instrumentation as well as the reactor trip system. Details of the PRA model used in this assessment are provided in Appendix A.

There are a number of ways a plant might choose to manage equipment failure and replacement. The cases identified below are discussed in detail in the following as a way to demonstrate how ERM could be used to evaluate equipment management options while maintaining adequate safety margins:

- Case 1: Run to failure; replacement during unplanned outage. Let the system run to failure and replace the affected components during an unplanned outage.
- Case 2: Replace at End-of-service-life during Subsequent Planned Outage. Replace the components that have reached their end-of-service-life during a planned outage when replacement of the component is scheduled.
- Case 3: Equipment Replacement Using Population-based Aging Rate Model. Utilize an aging rate model derived from population statistics to inform decisions about equipment replacement.
- Case 4: ERM-based Component Replacement. Use equipment condition monitoring to gauge component health so that components are replaced just in advance of failure.
- Case 5: ERM to Avoid Unplanned Outages. Use condition monitoring regime to predict unplanned outages. In combination with preventive maintenance, schedule any necessary predictive maintenance to avoid unplanned outages.

Each of the cases is discussed and compared in rest of this section.

4.1.1 Case 1: Run to Failure; Replacement during Unplanned Outage

Reactive/corrective maintenance strategies where a system is run-to-failure and then repaired allow maximal utilization of constituent component life. However, this strategy may create extensive downtime and increased economic loss due to lost power production. Irrecoverable damages lead to extended outages adding further burden on operational economy. Literature such as Yoshikawa et al. (2014) explain why run-to-failure is not necessarily the optimal reliability-centered maintenance strategy.

4.1.2 Case 2: Replace at End-of-service-life during Subsequent Planned Outage

Each component has a prescribed expected design-service-life as estimated by its manufacturer. In order to evaluate the safety and economic effectiveness of this Case, we let the components reach their end-of-service-life and assume a component replacement during the subsequent planned outage. To illustrate this scheme, we study the effect of replacement on safety (CDF) and economic metrics.

The safety metric was assumed to follow current industry practices where CDF is evaluated once during plant design and assumed to remain unchanged unless there is plant reconfiguration. For the economic analysis, a planned outage was assumed at an interval of two years (cycle). A prescribed cost was assigned for repairs assuming a probabilistic unplanned outage based on PRA cutsets. The average cost of such repairs and replacement costs at end-of-service-life along with lost power production during the outages are reflected in the economic analysis. A hypothetical time-varying failure rate was also constructed to analyze if preventive scheduled replacement has its merit when maintaining safety limits is also on the agenda. The failure rate of each component was assumed to follow a two-part piecewise linear aging model (Higgins et al. 1988) where the failure rate remains constant until an aging start time (AST) and then linearly increases with time based on a notional aging fractional increase (AFI) estimated from a sample of components whose failures were observed in a population of similar components.

These time dependent failure rates were then used in the PRA model for each time instant to construct a time dependent CDF curve (Figure 4.1). This will be referred to as the base-Case from now on. The components listed in Table 4.1 were assumed to be the ones whose condition could be monitored using ERM. The parameters assumed and adapted from NUREG-5502 (Higgins et al. 1988) report are also tabulated.

Table 4.1. Components Aging Characteristics Adapted from NUREG-5502 (Higgins et al. 1988)

Component	AFI Adapted from (Higgins et al. 1988)	Multiplicative factor on AFI used for base case	Multiplicative factor on AFI used for ERM case	AST (years) Adapted from (Higgins et al. 1988)
Electromagnetic Pump	0.28	2	5 for EMP1A 3 for EMP2A	9.2 18 for EMP2A in ERM case
Intermediate Loop Pump	0.28	2	1E-2	9.2
Feedwater Pump	0.28	2	1E-2	9.2
Main Condensate Pump	0.28	2	1E-2	9.2
Main Feedwater Pump	0.28	2	1E-2	9.2
Turbine Bypass Valve	0.02	10	10	1
Intermediate Loop Isolation Valve	0.02	10	10	1

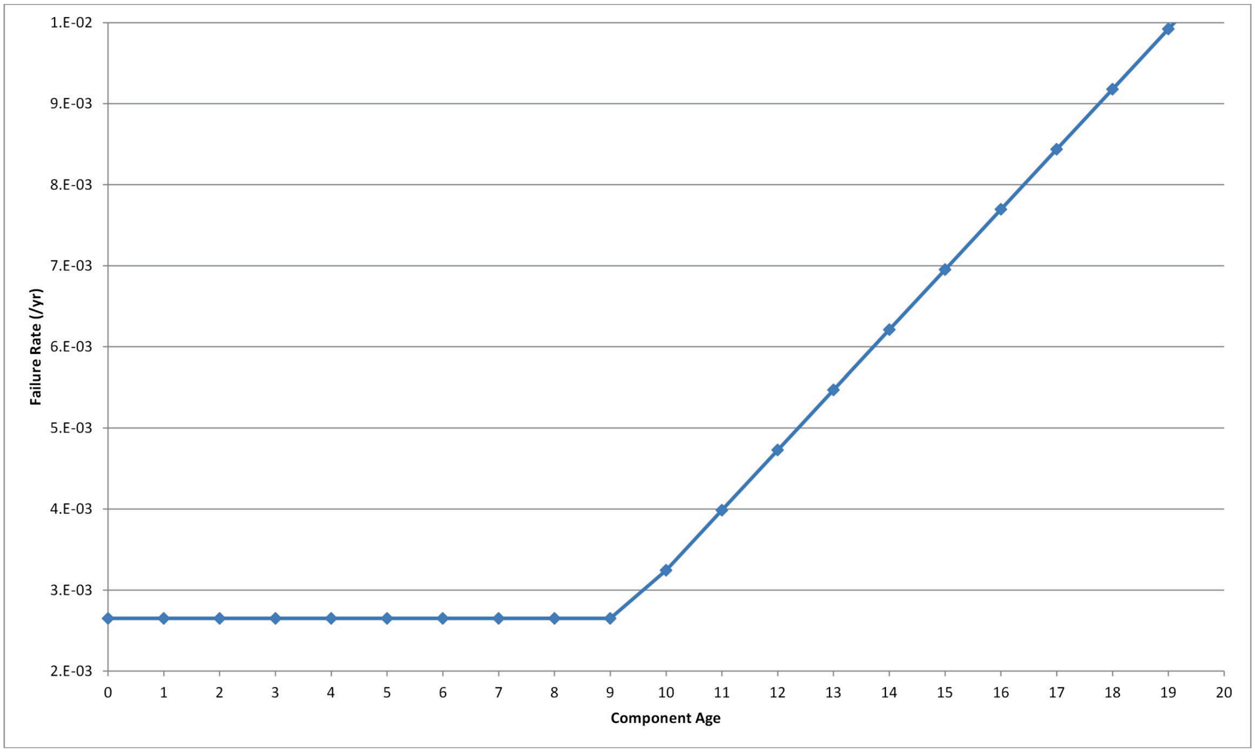

Figure 4.1. Aging Failure Rate Model for a Component with AFI=0.28, AST=9.2 Years and a Base Failure Rate of 2.65E-03

For this particular option of scheduled replacement, the failure rates were reset to base failure rate when a component replacement was done at the end-of-service-life. A graphic of the projected CDF and cost over the 40 year life of this strategy is presented in Figure 4.2 along with the red-line showing the non-time-varying CDF (static) as estimated per current industry practices. The static CDF coincides with the time-varying CDF at time 0. An assumed acceptable risk limit of 1.0E-06 was assumed which is also shown on the chart (horizontal green line). The dips in the safety metric at various time instants reflect reset of one or more component failure rates following a planned replacement or repair/maintenance that results in the component reaching an as-new condition. Similarly, peaks in the economic metric correspond to increase in maintenance costs when components reach their end-of-service-life. Often times, the CDF is well within the safety goal, but the replacement policy drives anticipated, but major economic costs leading to seeking for other maintenance policy alternatives. For example, at time 26 years, the CDF is well below the threshold limit not requiring any economic investment at all. However, there is a spike in maintenance cost owing to the time-based replacement policy. On the contrary, at 12 years, the acceptable risk limits are expected to be violated based on a population based generic aging failure rate model. However, there is no investment undertaken to maintain safety levels in anticipation of the threshold breach. These anomalies can be better managed if ERM technology based maintenance policy is established.

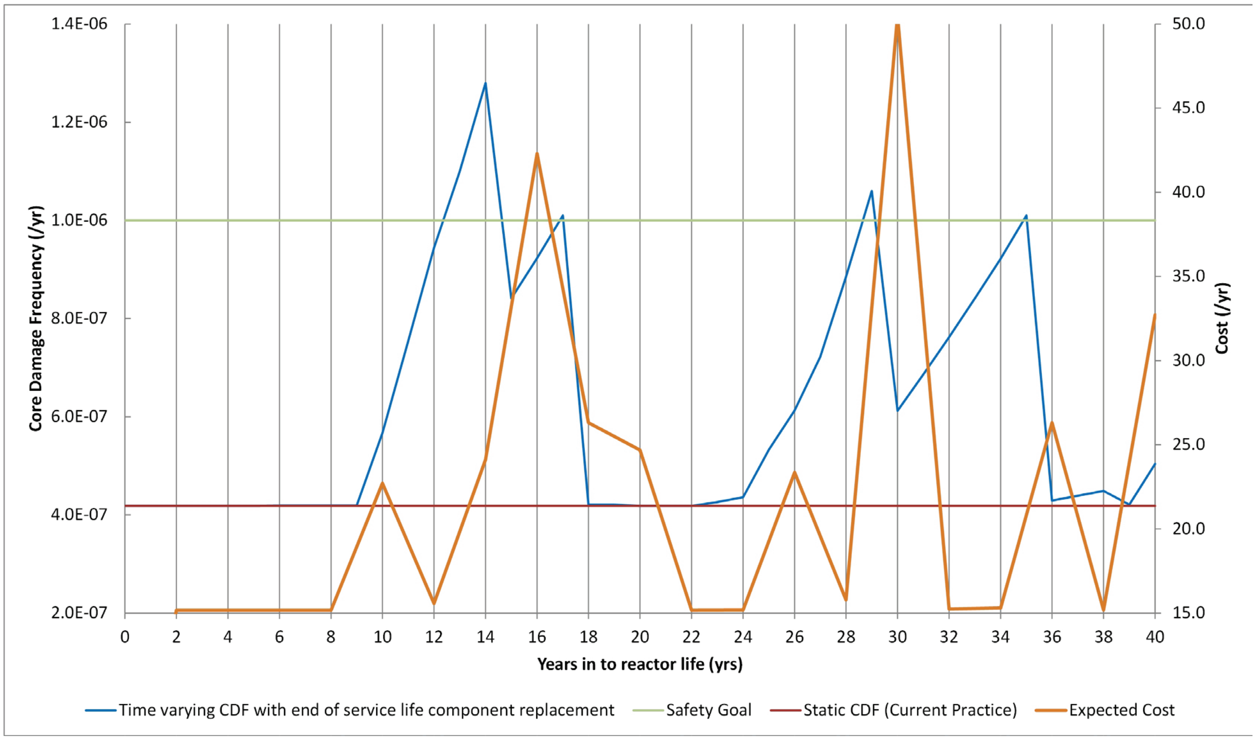

Figure 4.2. Safety and Economic Effectiveness of Replacing Components at the End of Component Service Life

4.1.3 Case 3: Equipment Replacement Using Population-based Aging-Rate Model

In this alternative maintenance strategy, a component's failure rate is predicted based on historical observations on a population of similar components as was discussed for Case 2 (i.e., the base-Case) in constructing a time-varying CDF. As seen in Figure 4.2, the CDF is expected to cross the safety goal at 12 years and hence, one could schedule a repair or replacement activity on components of importance in anticipation at 10 years during a planned outage. However, a repair may result in no significant change in component condition due to one or more factors (such as human error). On the other hand, a replacement is not necessarily the best strategy because the anticipated spike in the component failure rate is based on a population of components and not the specific condition of the component. Given these logical observations, we now move on to exploring ECA and ERM based strategy alternatives.

4.1.4 Case 4: ERM-based Component Replacement

The prime benefit of ERM is the ability to interrogate the present condition of each ECA-enabled component and plan maintenance strategies just in time to mitigate its adverse impact on the goal.

In this scheme, we assume that there exists a suitable functional model that receives component condition measurements and translates them to an equivalent failure rate. One strategy is to assume an industry generic failure rate at the onset and then update it based on new component specific measurements obtained at optimal intervals in time. This enables coherent platform for both ECA-enabled and other components to be plugged in to the same PRA model. These updated failure rates for each component are then used in the PRA model to determine time-sensitive CDF family of profiles with associated measurement and model uncertainties. A prognostic algorithm fits a suitable trend to the generated CDF and predicts an upcoming unplanned outage, preferably with a time-window due to inherent uncertainties. This window of opportunity allows importance ranked components to be replaced strategically to avoid unplanned outages. The graphic presented in Figure 4.3 presents the benefits of condition-based replacement, if hypothetical measurements were assumed for each of the components considered measureable. Measurements were assumed to be taken on all ECA-enabled components. The hypothetical measurements were based on proportionally reduced the aging fractional increase (AFI) for all pumps relative to base case discussed under the base-Case. AFI remained the same for non-pumps relative to base-Case. The CDF thus estimated based on condition measurement is shown in Figure 4.3. The decision-maker is faced with various time alternatives to replace components based on current trends as seen through the dotted lines in Figure 4.3. A Fussell-Vesely importance measure (ranking) was employed to analyze top contributors to the safety indicator.

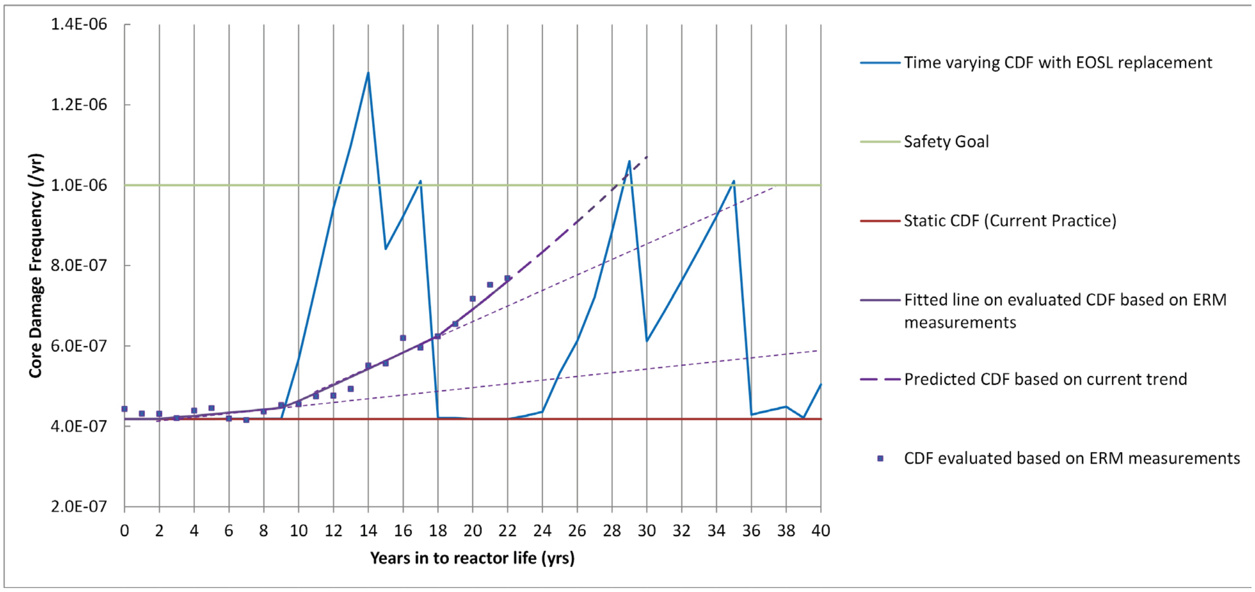

Figure 4.3. An ECA-based CDF Profile Illustrating Decision Points at Various Points in Time

At time 0, the evaluated CDF is 4.18E-07 per year and the top contributors to the CDF are intermediate loop pumps as shown in Table 4.2. However, the CDF is within the preset safety goal of 1E-06 and does not require any risk reduction measures.

As seen in Figure 4.3, around 12 years there is an increasing trend in the CDF. The underlying causal factor based on importance analysis (see Table 4.3) demonstrates a marked increase in the AFI of electromagnetic pump EMP1A. The CDF is still within the safety limits, but is predicted to cross it around 37 years when EMP1A will have to be replaced.

Table 4.2. Fussell-Vesely Event Importance at Time Zero-years into Reactor Life

Event	Description	Fussell-Vesely Importance
%IMHP1A	Intermediate Loop Pump	0.07
%IMHP1B	Intermediate Loop Pump	0.07
%IMHP2A	Intermediate Loop Pump	0.07
%IMHP2B	Intermediate Loop Pump	0.07
%EMP1A	Electromagnetic Pump	0.03
%EMP1B	Electromagnetic Pump	0.03

Table 4.3. Fussell-Vesely Event Importance at 12 Years into Reactor Life

Event	Fussell-Vesely Importance
%EMP1A	0.10
%IMHP1A	0.06
%IMHP1B	0.06
%IMHP2A	0.06
%IMHP2B	0.06
%EMP1B	0.02

Around 22 years, there is a further visible steepness in the CDF as seen in Figure 4.3. An importance analysis (see Table 4.4) indicates electromagnetic pump "%EMP2A" also contributing to the CDF more than the intermediate loop pumps. The CDF is acceptable, but is now predicted to cross the limits around 29 years with a CDF of 1.03E-6. A replacement during planned outage prior to exceeding the threshold at 28 years would be recommended. In order to study the number of components to be replaced, a further probe and decision-impact analysis needs to be conducted. Table 4.4 shows the impact on CDF of replacing each additional component. Replacement of electromagnetic pump "%EMP1A" alone is adequate to reduce CDF to below the safety levels as seen in Table 4.4, with the CDF predicted to remain under the safety goal until the end-of-reactor-life.

Table 4.4. Fussell-Vesely Event Importance at 22 and 28 Years into Reactor Life

Name	FV Importance at 22 years	CDF after Risk Reduction at 28 Years
%EMP1A	0.27	6.84E-07
%EMP2A	0.07	5.65E-07
%IMHP1A	0.04	5.64E-07
%IMHP1B	0.04	5.60E-07
%IMHP2A	0.04	5.58E-07
%IMHP2B	0.04	5.57E-07

This condition-based predictive replacement seems beneficial over the other discussed strategies both in terms of maintaining safety goals and from economic perspective (Figure 4.4). However, an even more strategic alternative is to perform periodic maintenance at optimal intervals based on prognostics rather than waiting close to compromising the safety goal and then replacing the component at higher costs. This is illustrated in the next case.

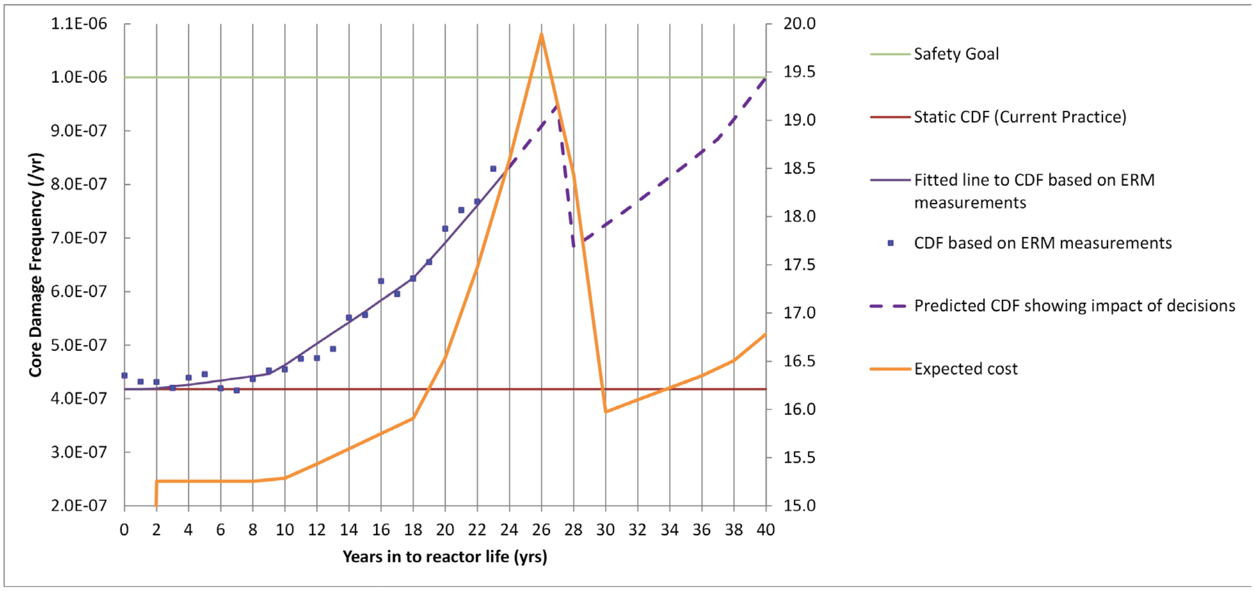

Figure 4.4. An ERM-based CDF Profile Illustrating Impacts of Decision Taken Just Before a Predicted Unplanned Outage

4.1.5 Case 5: ERM to Avoid Unplanned Outages

The strategy for this Case relies on conducting regular maintenance activities early on in the lifecycle of a component when predicted trends in the safety metric show signs of compromising the acceptable risk limits anytime within the reactor's expected service-life.

In order to illustrate this strategy, we assumed similar configuration and condition assessment as shown in Figure 4.3 which showed only the decision points without considering impacts of any particular decision. The electromagnetic pump "%EMP1A" showed confirmed signs of deterioration around 13 years. A hypothetical decision was taken to conduct a preventive maintenance involving minimal repair that returns the component to as-good-as-new condition. The preventive maintenance schedule was set to every five years following a renewed aging start-time from then on. This decision is reflected in Figure 4.5 with dips in the CDF visible at 14 and 28 years. An additional plunge in the CDF around 24 years is attributed to a second decision to establish preventive maintenance of electromagnetic pump "%EMP2A". However, there is an increasing trend in the CDF owing to aging in other components that presumably do not impact to the extent of reaching unacceptable safety limits. If those components were to contribute significantly, importance ranking and predictive analysis would flag them at an appropriate time for proactive decision-making. This strategy clearly demonstrates that ERM-based preventive maintenance achieves optimal economic benefits and safety objectives compared to other decision-making strategies.

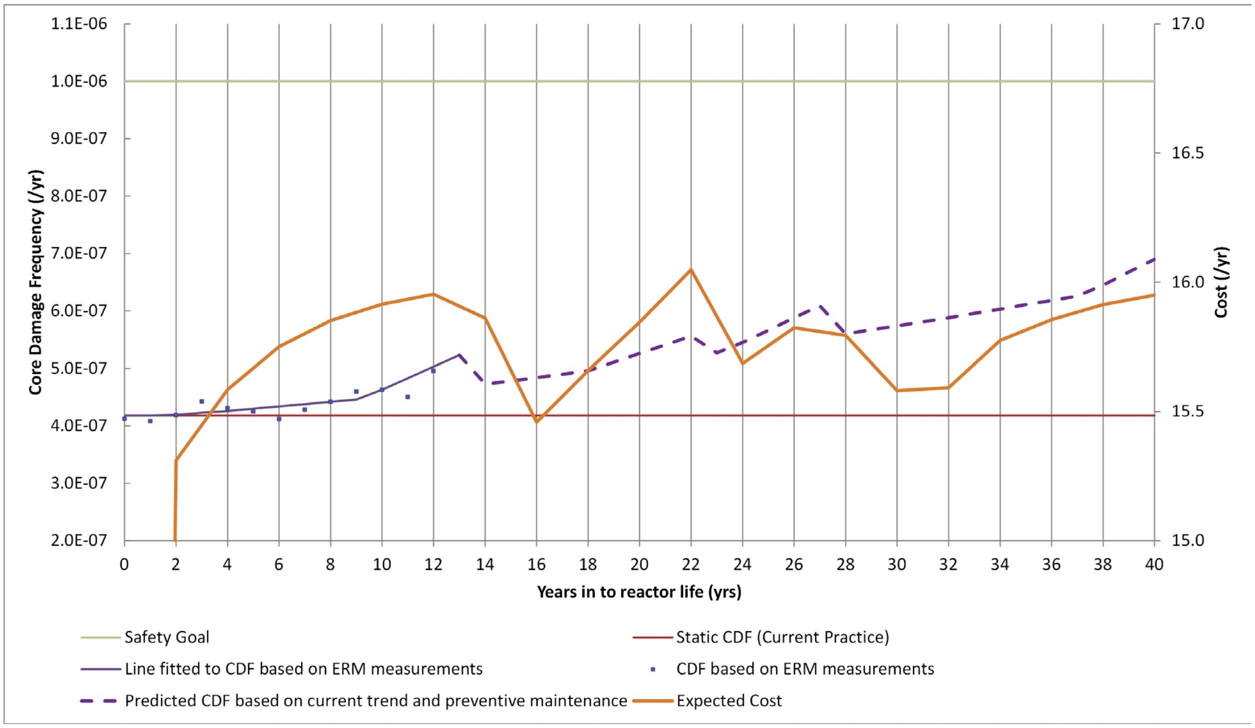

Figure 4.5. An ERM-based CDF Profile Illustrating Impact of Conducting Preventive Maintenance at Regular Intervals

4.2 Discussion

The results presented in the previous sections show that, of the cases studied and with the failure data used, the ERM-related cases (Case 4 and Case 5) provide an advantage when it comes to determining which components might require a maintenance action and the timing of such action. Even within these two Cases, the use of a condition monitoring regime in a predictive mode (i.e., Case 5) to avoid unplanned outages appears to provide optimal economic benefits when compared to the other Cases.

Note that the results presented in this section do not include the uncertainty bounds. This was primarily for the sake of simplicity in displaying the predicted risk. Clearly, the costs and associated decisions will be impacted by the uncertainty in the predicted risk, particularly if the uncertainty bounds are large. This is particularly likely when there is a large uncertainty in the predicted POFs used in the ERM-based calculations. In this situation, it is debatable whether the benefits of using ERM outweigh the costs associated with deploying an ERM-based solution. Even in cases where the uncertainties in the predicted POFs are not large, there is a (generally small) likelihood that the estimates of predicted risk and the predicted cost of the O&M options are incorrect. As a result, there is a need to perform a cost-benefit analysis for the ERM-based solutions under realistic scenarios prior to deployment of the technology.

Clearly, a challenge in the application of ECA and ERM for this purpose is the need to instrument several components for condition monitoring, and applying the resulting data to one or more prognostic techniques for predicting condition, remaining life, and likely POFs. It is possible that the ERM methodology may also be used to prioritize the deployment of the condition monitoring technology, by selecting components that are considered to be risk-significant. Such risk significance may be determined through, for instance, importance ranking (as done in Case 4) though the importance rankings (measures) themselves are dependent on the data used to construct the models (which is a function of the components for which condition-monitoring is deployed).

To avoid such circular reasoning, an alternative may be to use a statistical sampling approach wherein the risk significance is computed over a statistically valid sample of failure data, drawn from appropriate failure distributions based on population statistics. An effort was undertaken to extract information about component failure rates from previously operational reactors (Appendix B and Appendix C), though the finding from these efforts was that the available data was sparse and its use in the ERM methodology was challenging. An alternative approach is to use simulation-based tools to determine the impact of one or more components as they degrade. For this purpose, a tool was developed that simulated the operational characteristics of a prototypic SFR (Appendix D). Degradation of several components was simulated and the resulting impact on plant performance computed. Such data might provide the necessary information to understand risk significance of specific components, and this study is ongoing.

5.0 Summary

Enhanced risk monitors that integrate ECA and prognostics information to calculate time- and condition-dependent failure probabilities have the potential to enable real-time decisions about stress relief for susceptible equipment while supporting effective maintenance planning. As a result, ERMs are expected to improve the safety, availability, and affordability of ARs.

An initial ERM methodology for integrating time-dependent failure probabilities into risk monitors was developed. This prototypic methodology was evaluated using a hypothetical PRA model from a simplified model of a liquid-metal-cooled AR. Component failure data from industry compilation of failures of components similar to those in the simplified AR model were used to initialize the PRA model. By using time-dependent POF that grows from the initial probability when equipment is in like-new condition to a maximum POF, which occurs before a scheduled maintenance action that restores or repairs the component to "as-new" condition, the changes in CDF over time were computed and analyzed.

The results indicate that, using the proposed methodology for ERM, as the failure probabilities and failure rates change over time, the CDF changes over time. Repairs or replacements (bringing the components to as-new condition) reduce the risk, although aging of other components may still drive the overall risk higher.

Uncertainty analysis indicated that the ability to propagate uncertainties in various inputs to the ERM provides useful information. Specifically, the uncertainty bounds in the ERM output can have an impact on the ability to perform quantitative assessments of the changes in O&M and safety risk metrics due to component degradation. Improved quantification of the sources of uncertainty will be needed to improve the ability to perform these kinds of trade-off analyses.

In addition, a study on alternative risk metrics for ARs was conducted. Risk metrics that quantify the normalized cost of repairs, replacements, or other O&M actions were defined and used, along with an economic model, to compute the likely economic risk of future actions such as deferred maintenance based on the anticipated change in CDF due to current component condition and future anticipated degradation. Such integration of conventional risk metrics with alternate risk metrics provides a convenient mechanism for assessing the impact of O&M decisions on safety and economics of the plant. It is expected that, when integrated with supervisory control algorithms, such integrated risk monitors will provide a mechanism for real-time control decision-making that ensure safety margins are maintained while operating the plant in an economically viable manner.

Planned research activities are focused on integrating the ERM methodology with plant supervisory control algorithms within a simulation platform that simulates AR module plant operation. This integration will enable the evaluation of the ERM within a plant operation framework and provide specific information on the ability to make risk-informed decisions that impact plant control. Quantitative information on plant performance impact due to the ERM technology is anticipated as a result of this integration effort. In addition, we will continue to explore the possibility of evaluations using experimental data, and to this end, will continue to evaluate sources of relevant reliability data, including data from test reactors, and available test-beds.

6.0 Future Research Opportunities

Beyond the planned activities described in the previous section, a number of other open questions remain with respect to the ability to deploy the ERM. These questions remain targets for future research, and are briefly documented in this section.

One of the easiest ways to provide real-time situational awareness is either graphically or in dynamic tables. These are common methods used throughout many industries for facility/grid O&M, and such displays can be readily adapted to integrating information from the ERM, supervisory control, and PSM. Using displays similar to load forecasting and contingency analysis currently used in facility and power grid O&M can help an operator more efficiently and quickly understand the current operational status with an outlook on future status with some uncertainty. Any planned maintenance can also be shown on the graphical interface to indicate any alignment or lack thereof with current plant operations and component health. Quantities that are likely to determine impact deployment of the ERM include those listed in the following sections.

6.1 Computational Complexity

The needs described above may lead to substantial computational complexity of both the supervisory control and ERM modules. To reduce the computational demand on the overall system (supervisory control and ERM) it may be useful to

- Limit the number of power reduction options to limit the number of choices the supervisory control has to make. An example may be options that reduce the power output in integral multiples of 5 percent (95%, 90%, 85%, etc.), at least initially, to simplify the needed ERM calculations to account for the opportunity cost of operating the plant for an extended period of time below 100 percent power. Note that the cost of running the plant for a fixed period of time is (roughly) the same regardless of plant power, while generation revenue is proportional to plant power.
- Suppress operational options with success probabilities below some threshold, to help focus the supervisory control algorithm to only viable options. Determination of the minimum viable success probability will need to be done carefully, and would involve an assessment of the various possible risk metrics to better understand the tradeoffs involved.

6.2 Cost-benefit Analysis for ERM

Integration of ERM and ECA into plant O&M practices is likely to occur if a clear benefit can be shown, given the added cost of implementing and applying the ERM.

6.3 Operator Interfaces

Any ERM output that triggers action by the supervisory control modules needs to be available to the (human) operator for review in an easy to comprehend format. As previously stated, current metrics for plant O&M rely on models using static inputs or inputs based on operational experience and statistical component metrics. These methods are lacking in providing the operator with a real-time situational awareness that can accurately represent the real condition of plant O&M. These factors are becoming more important as newer designs are incorporating in-vessel and in-containment components that are not readily accessible for maintenance or observation along with new component designs with no historical operational data to provide statistical operational characteristics. In order to safely and efficiently implement new designs, good situational awareness provided to the operator will be necessary.

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Appendix A

Generic Advanced Reactor PRA Model Description

Appendix A

Generic Advanced Reactor PRA Model Description

This appendix contains a description of the generic advanced reactor PRA models that were used in evaluating the ERM concept.

A.1 Generic Advanced Reactor Description

A simplified-model AdvSMR (power block) design is used in the development of the PRA model used for the research that supported the development of a framework for ERMs. This simplified model is shown in Figure A.1. This hypothetical design is intended to be prototypical and resembles proposed liquid metal-cooled SMR designs. The example design is defined to provide a simple level of abstraction but contains enough resolution and specific design elements to inform the development of a PRA model that, when quantified, produces a cogent set of results.

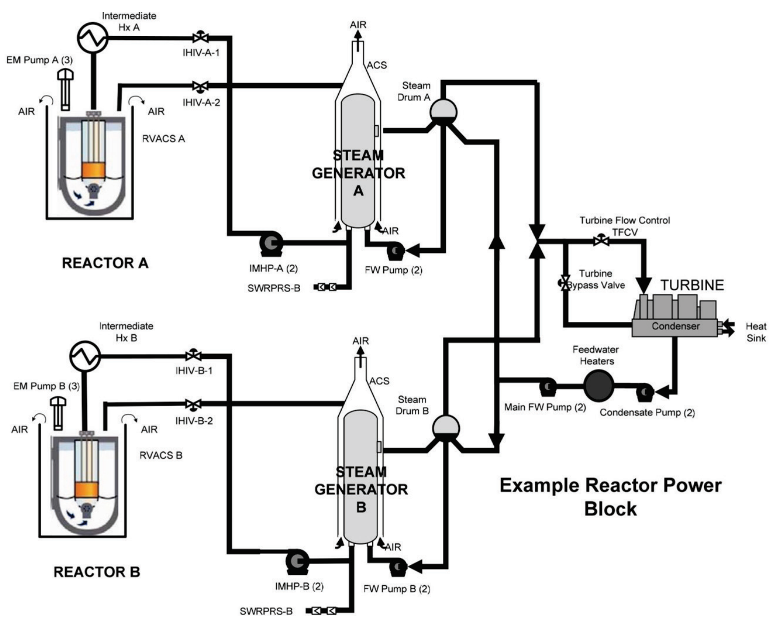

Figure A.1. One-Line Diagram of Simplified-Model AdvSMR

The simplified-model AdvSMR design in Figure A.1 is a small, modular, pool-type, liquid-metal-cooled reactor assumed to be producing 200 to 500 MWt[(1)] of power. The plant design consists of an unspecified number of identical power blocks, with each power block comprised of two reactor modules. Each module is connected to its own intermediate heat exchange system and steam generator. The secondary side (i.e., steam side) equipment is located in a different building and connects two modules to form a power block. A power block feeds a single variable capacity turbine generator. (*Note: While a greater number of reactor modules in a power block are possible, two modules provide sufficient complexity to develop and demonstrate a methodology for ERM.*)

[1]The electrical output of a reactor depends on the efficiency of the power conversion process.

A.1.1 Key Components in the Simplified-Model AdvSMR Design

The components defined for modeling in the example reactor power block are:

- Electromagnetic pumps (3 per reactor module)
- Reactor vessel auxiliary cooling system, RVACS (1 per reactor module)
- Intermediate heat exchangers (1 per reactor module)
- Intermediate loop isolation valves (2 per reactor module)
- Intermediate loop pumps (2 per reactor module)
- Steam generators (1 per reactor module)
- Sodium-water-reaction pressure relief system, SWRPRS (1 per reactor module)
- Steam drum (1 per reactor module)
- Feedwater pumps (2 per reactor module)
- Passive steam generator cooling system (1 per reactor module)
- Turbine generator (1 per power block)
- Turbine bypass valve (1 per power block)
- Turbine flow control valve (1 per power block)
- Main feedwater pumps (2 per power block)
- Main feedwater heater (1 per power block)
- Main condensate pumps (2 per power block)
- Emergency diesel generator (1 per power block)

The primary features of the simplified design are the primary cooling loop, intermediate cooling loop, secondary system including the steam generators, and residual heat removal systems consisting of a passive RVACS and passive steam generator cooling system.

The primary loop is contained entirely within the reactor vessel. Liquid sodium is pumped by electromagnetic pumps up through the reactor core and out through the top. Flow is then forced back down through the space (annulus) between the outer wall and reactor core past two intermediate heat exchangers. The electromagnetic pumps are suspended into the reactor pool from above. Because electromagnetic pumps have no moving parts and therefore there is no associated “flywheel effect,” a synchronous coast-down function is designed into pumps to provide coast-down upon loss of power.

The intermediate loop transfers heat to the secondary system via two steam generators. The primary components of this system are the steam generator, the intermediate cooling pumps, and the intermediate loop isolation valves. The intermediate cooling pumps force flow of heated liquid sodium from the intermediate heat exchangers to the steam generators during both normal and upset conditions. The isolation valves close to isolate the reactor from a pressure increase resulting from a sodium-water interaction that would occur in the event of a steam generator tube rupture event. The signal to close these isolation valves is based on opening of passive pressure relief valves connected directly to the steam generators. Together the isolation and pressure relief valves constitute part of the SWRPRS.

The secondary system consists of a steam generator and a steam drum for each reactor module connected to a single turbine generator. The secondary system delivers steam from the steam generators to the inlet of the turbine. Turbine steam exhaust flows through the condensers and then to main condensers and feed-water pumps back to the reactor module steam drums where it can be pumped by the reactor module feed-water to the steam generators. The turbine bypass valves allow steam to flow past the turbine and directly into the condenser when required. This allows a means of residual heat removal from the reactor modules during reactor shutdown and startup, and provides a flow path that will be needed in case of load rejection and some event that trips the turbine. Each steam generator has a sodium-water reaction pressure-relief system that relieves pressure in the event of a generator tube rupture. This is a passive system and provides a path for the increased steam pressure that would occur from sodium-water reaction.

The residual heat removal system consists of RVACS and the passive steam generator cooling system. The passive steam generator cooling system removes heat by air circulation past the steam generators. This airflow is initiated by remote manual opening of louvers at the inlet and outlet of the shroud around the steam generators. In this mode, heat is removed by natural convection to the air. This system can operate with forces or natural circulation of intermediate cooling loop sodium. If operators are unsuccessful at opening louvers to initiate convective cooling or if the intermediate cooling flow or inventory is lost, then residual heat can by removed by natural air circulation around the containment vessel that surrounds the reactor vessel via the RVACS. Heat will be transferred from the reactor vessel to the containment vessel by radiative heat transfer and then to the air around the containment vessel and ultimately the atmosphere via convective heat transfer. A key design feature of RVACS is that no components or operator actions are required to initiate RVACS, because it is continually operating during normal power operation and is designed to be able to accommodate residual heat transfer after reactor shutdown.

A.2 Probabilistic Risk Assessment (PRA)

In general, *Risk* can be defined as the product of the frequency of an event and its consequence:

$$Risk = Frequency \times Consequence$$

where *Consequence* refers to undesirable outcomes (reactor core damage, release frequency of radionuclides, cancer deaths, etc.) and *Frequency* is the likelihood of the consequence per unit time. In the nuclear industry, risk is typically evaluated for events that have consequences related to public health and safety.

The assessment of risk with respect to nuclear power plants (NPPs) is intended to achieve the following general objectives (Fulwood and Hall 1988):

- Identify initiating events and event sequences that might contribute significantly to risk;
- Provide realistic quantitative measures of the likelihood of the risk contributors;
- Provide a realistic evaluation of the potential consequences associated with hypothetical accidents; and
- Provide a reasonable risk-based framework for making decisions regarding nuclear plant design, operation, and siting.

PRA is a systematic safety analysis methodology that (Haasl et al. 1988; Apostolakis 2000) begins by identifying undesirable consequences (e.g., reactor unavailability, core damage, release of radioactivity) and initiating events that can lead to these consequences. This is followed by systematically identifying accident sequences [defined by event-trees (Papazoglou 1998) and fault-trees (Vesely et al. 1981)] through which the facility can move from the initiating event to the undesired consequence. The PRA model then calculates the probability of occurrence for each accident sequence and ranks the accident sequences according to probability of occurrence (or, alternatively, contribution to the undesirable event) to manage the major contributors to risk.

Three levels of PRA, designated by the type of risk being assessed, have been considered for NPPs (NRC 2012). Level 1 PRA estimates the frequency of accidents that cause core damage (commonly called core damage frequency or CDF); Level 2 PRA, the frequency of radioactive release from the NPP (assuming that the core is damaged); and Level 3, the consequences to the public and environment outside the NPP from Level 2 radioactive releases. The ultimate result of the PRA is the probability of each undesirable consequence (e.g., core damage, radioactive release) and a list of the major contributors to its occurrence.

A full PRA model consists primarily of event-tree and fault-tree models that, when solved, produce cutsets representing the combinations of failures that result in an accident sequence and define the likelihood of those failures (EPRI 2011). Fault-trees and event-trees define Boolean relationships among fault events that cause the top event to occur. Event-trees define logic among fault-trees in a way that accident sequences can be translated entirely into an equivalent set of Boolean equations. This logic can be reduced to an expression of cutsets. The list of cutsets for an accident sequence represents all combination failures leading to that accident sequence. The dominant cutsets represent the most important combinations along with the frequency or probability of those failures.

An event-tree is a diagram that defines accident sequences. Each horizontal "pathway" running from left to right through an event-tree defines an accident sequence beginning with an initiating event, followed by a series of top events (i.e., the systems and/or actions needed to mitigate the initiating event), and finishing at a particular plant end state (e.g., plant damage). Each branch point of the event-tree represents a question asked about the status or condition of a system. Traditionally, the up branches indicate success while the down branches indicate failure. Figure A.2 shows an example event-tree.

Fault-trees are graphic models depicting the various fault combinations that will result in the occurrence of an undesired (i.e., top) event. A simple fault-tree is presented in Figure A.3. Fault-tree analysis is an analytical technique, whereby an undesired state of the system is specified, and the system is analyzed in the context of its environment and operation to find all credible ways in which the undesired event can occur (Vesely et al. 1981).

Both passive and active components may be included in fault-trees and event-trees. Typical active component failures include: 1) failure to run, 2) failure to start, 3) failure to open or close or operate, and 4) unavailability because of test or maintenance. Typical passive component failures include: 1) rupture, 2) plugging, 3) failure to remain open or closed, and 4) cold or hot short of power or instrument cables.

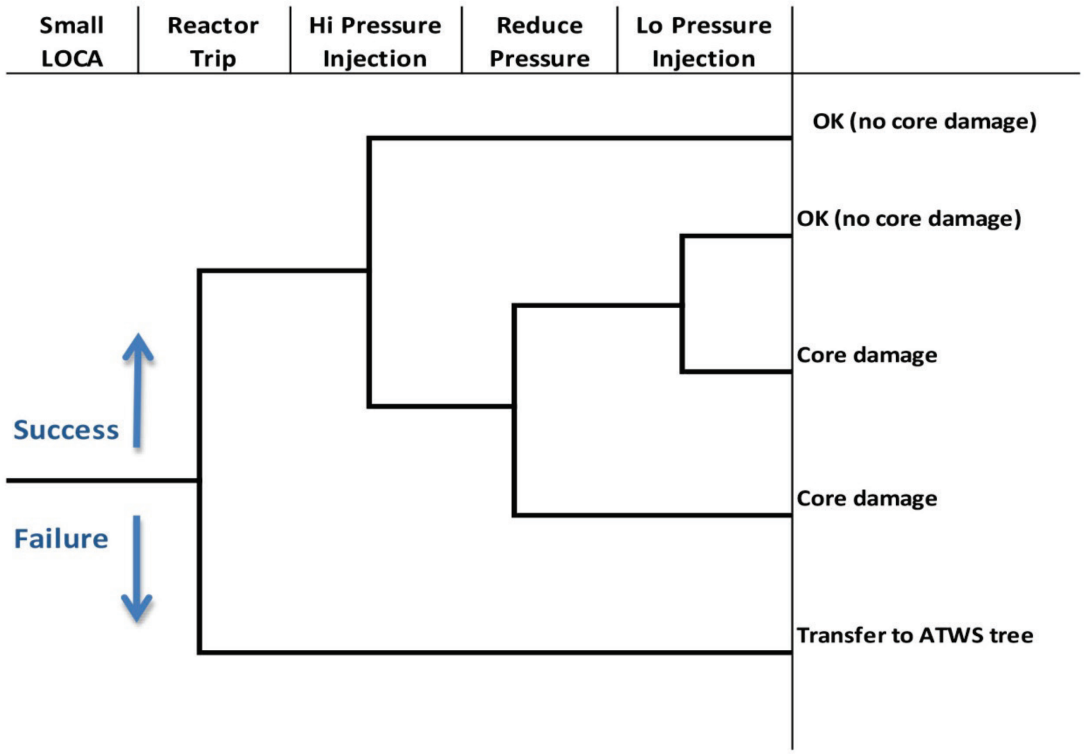

Figure A.2. Simplified Reactor PRA Event (NRC Undated)

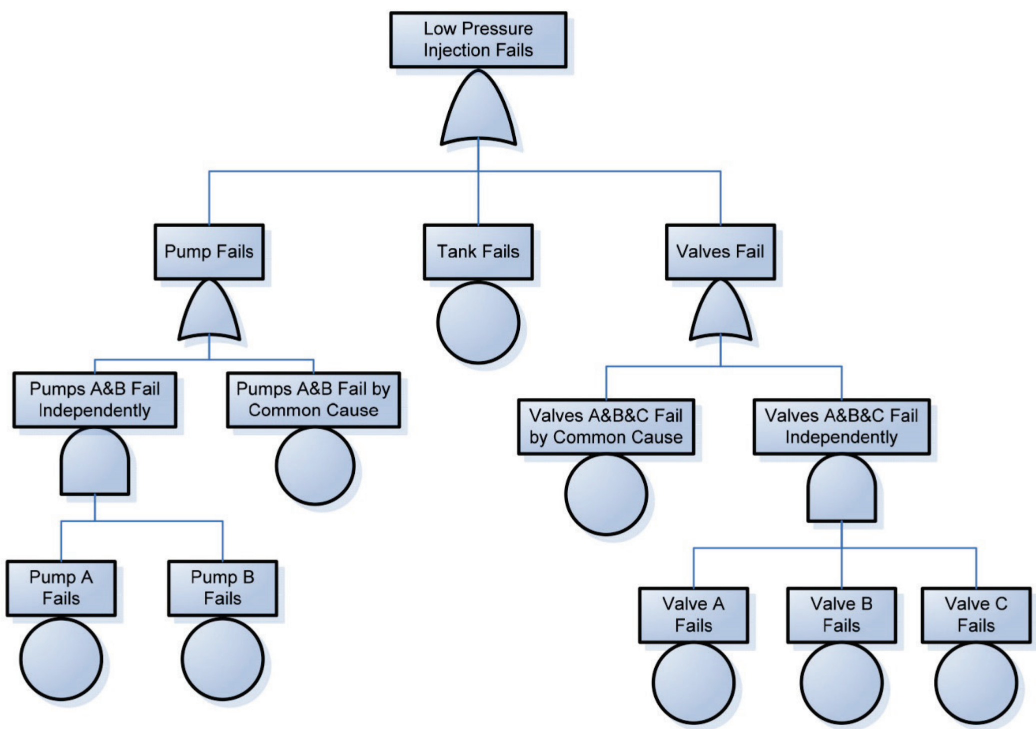

Figure A.3. Simplified Example Fault Tree (NRC Undated)

Each failure event in the fault-tree is called a basic event and has a component failure or human error probability associated with it. Component failures are typically demand- or time-related (e.g., valve fails to close on demand, or pump fails to run for 24 hours). Data for component failure rates and failure probabilities comes from generic sources, plant-specific sources, or a combination of the two (as when generic data is adjusted using plant-specific data by performing a Bayesian update). Aging-related failure data, if included, typically utilizes reliability models (Vesely and Wolford 1988; Smith et al. 2001). Human error probabilities are generally compiled using human reliability analysis (HRA) that is based on research done in NPP control rooms and simulators. HRA is an important part of PRA, and considers such performance-shaping factors as stress level, crew resources, cues, and timing.

Importance analysis is typically performed on the results of a PRA and provides a quantitative perspective on risk and sensitivity of risk to changes in input values (Vesely et al. 1983). Three commonly encountered importance analyses (measures) are determination of risk achievement worth (RAW), risk reduction worth (RRW), and Fussell-Vesely (F-V). These analyses produce different kinds of measures of basic or initiating event importance, such as determining the ratio of the total CDF produced when a particular basic event is set to either one or zero to the baseline CDF produced when the basic or initiating event is set to its nominal value. For instance, RRW analysis uses the ratio of the baseline risk to the reduced risk calculated by assuming a component is completely reliable (i.e., no failures) (Vesely et al. 1983). Importance measures are valuable in sorting out the most important component failure modes.

Uncertainty in PRA modeling arises from a number of sources that are typically divided into aleatory variability and epistemic uncertainty (EPRI 2011). Aleatory variability is related to the statistical confidence we have in failure probability data, while epistemic uncertainty is related to the uncertainty in the accident sequences used to develop the PRA model. Epistemic uncertainty is dealt with by developing event- and fault-trees as complete as possible, identifying keys sources of uncertainty, and performing sensitivity analyses. The aleatory variability is addressed explicitly by propagation of parametric data uncertainty for initiating basic event data. Uncertainty analysis is performed through a sampling strategy (e.g., Monte Carlo sampling) over some number of observations.

As PRA models are integrated into plant management, they have become living models that reflect the as-modified and as-operated plant configuration and are able to estimate the changing likelihood of undesired events. Risk monitors extend the PRA framework by incorporating the actual and dynamic plant configuration (e.g., equipment availability, operating regimes, and environmental conditions) into the risk assessment, although failure data on equipment is based on operational experience and reliability analysis, and unit-specific failure information is generally not used.

A.2.1 PRA for Simplified-Model AdvSMR

The PRA model developed for the simplified-model AdvSMR (Figure A.1) is capable of modeling fault (or accident) sequences that could occur, induced by a perturbation (or initiating event) in the system, and of identifying the combinations of system failures, support system failures and human errors that could lead to core damage. The general framework for the PRA model discussed herein includes the following analyses, each of which are discussed in detail below:

- Initiating Event Analysis
- Accident Sequence Analysis
- Systems Analysis
- Data Analysis
- Common Human Reliability Analysis

- Cause Failure Analysis
- Quantification

Table A.1 presents the initiating event and system component failure probabilities used to initialize the model (i.e., the failure probabilities when the components are as-built). Some components in this listing actually represent systems, such as RVACS, while others represent components.

Table A.1. Initiating Event Frequencies and Component/System Failure Rates used in the Model

Component and Failure Mode	Failure Rate	Initiator or System Failure	Assumption/Comments
Electromagnetic Pump (Failure to Run)	3.00E-05/hr	Both	Assumed unproven for NPP use. Failure rate somewhat higher than average.
Electromagnetic Pump (Failure to Start)	3.34E-03/dmd	System Failure	Assumed unproven for NPP use. Failure rate somewhat higher than average.
RVACS (Failure to Operate)	5.00E-07/hr	Both	Recovery of RVACS given it plugs was assumed to be 1E-1.
Intermediate Heat Exchanger (Tube Rupture)	8.70E-03/yr	Initiator	Assumed unproven for NPP use. Failure rate much higher than average.
Intermediate Loop Isolation Valve (Failure to Close)	7.00E-03/dmd	System Failure	Assumed to somewhat higher than NPP average. Motive power undefined.
Intermediate Loop Pump (Failure To Run)	2.00E-05/hr	Both	Failure rate assumed to be near NPP average for motor driven pumps.
Steam Generator (Tube Rupture)	8.76E-04/yr	Initiator	Assumed to be proven for NPP use. Failure rate lower than average.
SWR Pressure Relief System (Failure to Operate)	2.00E-04/dmd	System Failure	Failure rate assumed to be near NPP average for pressure relief systems.
Steam Drum	-	-	Failure of this passive component not modeled. Assumed to be small contributor to risk.
Feedwater Pump (Failure To Run)	1.00E-05/hr	Both	Failure rate assumed to be near NPP average for motor-driven pumps.
Steam Generator Louver (Failure To Open)	5.00E-02/hr	System Failure	Bounded by operator failure to open steam generator air flow louvers.
Turbine Generator	-	-	Assumed to be encompassed by reactor transient trip events.
Turbine Bypass Valve (Failure To Open)	1.00E-03/hr	System Failure	Failure rate assumed to be near NPP average.
Turbine Flow Control Valve	-	-	Assumed to be encompassed by reactor transient trip events.
Main Feedwater Pump (Failure To Run)	1.00E-05/hr	Both	Failure rate assumed to be near NPP average for motor-driven pumps.
Main Feedwater Heater	-	-	Assumed to be encompassed by reactor transient trip events.
Main Condensate Pump (Failure To Run)	1.00E-05/hr	Both	Failure rate assumed to be near NPP average for motor-driven pumps.
Emergency Diesel Generator (Failure To Start)	4.53E-03/ dmd	System Failure	Failure rate assumed to be near NPP average for emergency diesel generators.
Control Rod Drive Mechanism (Independent Failure)	5.78E-06/ dmd	System Failure	Failure rate assumed to be near NPP average for control rod drive mechanisms.
Trip Sensor (Independent Failure)	2.00E-15/ dmd	System Failure	Failure rate assumed to be near NPP average for trip sensors.

Component and Failure Mode	Failure Rate	Initiator or System Failure	Assumption/Comments
Trip Circuit Breaker (Independent Failure)	2.00E-16/ dmd	System Failure	Failure rate assumed to be near NPP average for trip circuit breakers.
Trip Setpoint (Independent Failure)	3.00E-15/ dmd	System Failure	Failure rate assumed to be near NPP average for trip setpoints.
Emergency Diesel Generator (Failure To Run During First Hour)	2.90E-03/ dmd	System Failure	Failure rate assumed to be near NPP average for emergency diesel generators.
Emergency Diesel Generator (Failure To Run)	8.48E-04/ dmd	System Failure	Failure rate assumed to be near NPP average for emergency diesel generators.
Motor Control Center (Failure to Operate)	4.34E-07/hr	System Failure	Failure rate assumed to be near NPP average for motor control centers.
Electrical Bus (Failure to Operate)	4.34E-07/hr	System Failure	Failure rate assumed to be near NPP average for electrical busses.
Circuit Breaker (Failure to Open/Close)	2.55E-03/hr	System Failure	Failure rate assumed to be near NPP average for circuit breakers.
Circuit Breaker (Spurious Operation)	1.71E-07/hr	System Failure	Failure rate assumed to be near NPP average for circuit breakers.
Motor-Operated Valve (Spurious Operation)	4.45E-08/hr	System Failure	Failure rate assumed to be near NPP average for motor-operated valves.
Reactor Transient (Trip)	2.50E-01/yr	Initiator	Failure rate assumed to be below average for NPP trips.

Note: Adapted from NUREG/CR-6928, Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants (Eide et al. 2007).

A list of the dominant cutsets that account for over 97% of the total CDF (calculated using the analyses listed above and assuming static POF) is shown in Section A.3.8. The full list is used as the input to the ERM model.

The following success criteria are implicit to the defined cutsets:

- Four out of six control rod units, one out of four trip sensors, one out of four SCRAM breakers, and accurate trip set-points are required for each module.
- The turbine bypass valve is required to open for one or two modules.
- One out of two main feed-water pumps is required for one or two modules. Both pumps are assumed to be running.
- One out of two module feed-water pumps is required for each module. Both pumps are assumed to be running.
- One out of two main condensate pumps is required for one or two modules. Both pumps are assumed to be running.
- One out of three electromagnetic pumps is required for each module. Two pumps are assumed to be running, and one is assumed to be in standby.
- One out of two intermediate loop pumps is required for each module. Two are required in case of an intermediate heat exchanger tube rupture. Both pumps are assumed to be running.
- The steam generator louvers (SGLs) are required to open for each module.

- In case of a steam generator tube rupture, either both the intermediate loop isolation valves or the SWRPRS is required to prevent a loss of coolant accident from the reactor vessel, which would make RVACSs ineffective.
- Sufficient heat (i.e., to prevent core damage) must be transferred from the reactor vessel to the containment vessel by radiative heat transfer and then to the air around the containment vessel and ultimately the atmosphere via convective heat transfer.
- For failure of RVACS caused by external events such as high winds, the opportunity for recovery (e.g., unplug radiating fins) by plant operators was assumed to be possible.

These success criteria are summarized in Table A.2.

Table A.2. Success Criteria for the Simplified-Model AR PRA Model

Key System	Description of System Failure	Success Criteria	Plant Function Supported
RSS/RPP	The reactor shutdown or protection system (RSS/RPP) fails to trip the reactor and maintain reactivity control.	Four out of six control rod units, one out of four trip sensors, one out of four SCRAM breakers, and accurate trip set-points are required for each module.	Reactivity Control
TB	The turbine bypass (TB) system fails to allow steam to flow past the turbine and directly into the condenser when required (e.g., in case of load rejection and some event that trips the turbine).	The turbine bypass valve is required to open for one or two modules.	Condenser Cooling
MFW	The main feed-water (MFW) system fails to provide feed-water to module steam drums to establish decay heat removal via the condenser.	One out of two MFW pumps is required for one or two modules. Both pumps are assumed to be running.	Condenser Cooling
FW	The feed-water (FW) system fails to provide feed-water to module steam generators to establish decay heat removal via the condenser.	One out of two module feed-water pumps is required for each module. Both pumps are assumed to be running.	Condenser Cooling
CD	The condensate (CD) system fails to remove decay heat via the condenser.	One out of two main condensate pumps is required for one or two modules. Both pumps are assumed to be running.	Condenser Cooling
PTHS	The primary heat transport system (PHTS) fails to maintain flow of sodium through the reactor vessel and consequently remove decay heat via the intermediate heat exchangers.	One out of three electromagnetic pumps is required for each module. Two pumps are assumed to be running, and one is assumed to be in standby.	Condenser and Passive Steam Generator Cooling
ITHS	The intermediate heat transport system (IHTS) fails to transfer heat via the intermediate heat exchangers to the secondary system for decay heat removal through the steam generator.	One out of two intermediate loop pumps is required for each module. Two are required in case of an intermediate heat exchanger tube rupture. Both pumps are assumed to be running.	Condenser and Passive Steam Generator Cooling

Key System	Description of System Failure	Success Criteria	Plant Function Supported
Passive Steam Generator Cooling	The passive steam generator cooling system fails to remove heat by air circulation past the steam generators. This airflow is initiated by remote manual opening of louvers (SGLs) at the inlet and outlet of the shroud around the steam generators. In this mode, heat is removed by natural convection to the air.	The SGLs are required to open for each module.	Passive Steam Generator Cooling
SWRPRS	The SWRPRS fails to isolate a SGTR-initiated sodium-water reaction that subsequently fails the IHTS and PHTS by means of an unrecoverable loss of sodium.	In case of a steam generator tube rupture, either both the intermediate loop isolation valves or the SWRPRS is required to prevent a loss of coolant accident from the reactor vessel, which would make RVACS ineffective.	Passive Cooling
RVACS	Residual heat cannot be removed by natural air circulation around the containment vessel that surrounds the reactor vessel via the RVACS.	Sufficient heat (i.e., to prevent core damage) must be transferred from the reactor vessel to the containment vessel by radiative heat transfer and then to the air around the containment vessel and ultimately the atmosphere via convective heat transfer. For failure of RVACS caused by external events such as high winds, the opportunity for recovery (e.g., unplug radiating fins) by plant operators was assumed to be possible.	Passive Cooling

For this preliminary analysis, where available, industry documented failure data (Eide et al. 2007) was used to define initiating event and component failure likelihoods for the key components in the simplified-model AdvSMR design. The first-year values were set to be compatible to mean industry failure rates presented in NUREG/CR-6928; however, latitude was taken in adjusting these values for the example. Specifically, for components where such data is not readily available, assumed failure data was used based on available operational experience and like-kind components.

Initial evaluation of the ERM incorporated assumed time-based event and failure probabilities for each of the initiating events and key components failures of our example AdvSMR power block (Figure A.1). These time-based likelihoods assume that the probability of failure increases from the initial probability when equipment is in like-new condition to a maximum probability of failure from component aging, until a scheduled maintenance action is taken. Periodic maintenance intervals are staggered for each component to reflect different operating lifetimes.

A.2.2 Initiating Event Analysis

An initiating event is an event that could lead directly to core damage (e.g., reactor vessel rupture) or that challenges normal operation and requires successful mitigation using safety or non-safety systems to prevent core damage. Identifying initiating events is the first step in the development of plant accident sequences, which are discussed further in Section A.2.3. The identification of initiating events applicable to a plant system is an iterative process that requires feedback from other PRA elements, such as system analysis, and review of plant or generic industry experience/data. The initiating events considered for the simplified-model AdvSMR are outlined in Table A.3.

Table A.3. Initiating Events for Simplified-Model AdvSMR

Loss of Electromagnetic Pump	Loss of Main Feed-water Pump
Loss of Feed-water Pump	Reactor Transient (Trip)
Loss of Intermediate Loop Pump	Plug or Failure of RVACS due to External Event
Intermediate Heat Exchanger Tube Rupture	Steam Generator Tube Rupture
Loss of Offsite Power	Anticipated Transient Without SCRAM
Loss of Main Condensate Pump	Loss of Main Feed-water Pump

A.2.3 Accident Sequence Analysis

Conceptually, each accident sequence can be thought of as a combination of an initiating event, which triggers a series of plant system and/or operator responses, with a certain combination of successes and/or failures of these responses that lead to a core damage state. The fault-tree linking approach, which involves a combination of event-trees and fault-trees, was used to identify and analyze the plant functions required to respond to each identified initiating event to prevent core damage. Event-trees are developed to outline the broad characteristics of the accident sequences that start from the initiating event and, depending on the success or failure of each defined plant function, lead to a successful outcome or to damage to a core damage event. Fault-trees are then used to model the failure of the key and supporting systems that are deemed necessary to carry out each plant function. Initiating events that require the same or similar plant response may be grouped into categories that each uses a single event-tree. The resulting event-tree for simplified-model AdvSMR is presented in Figure A.4.

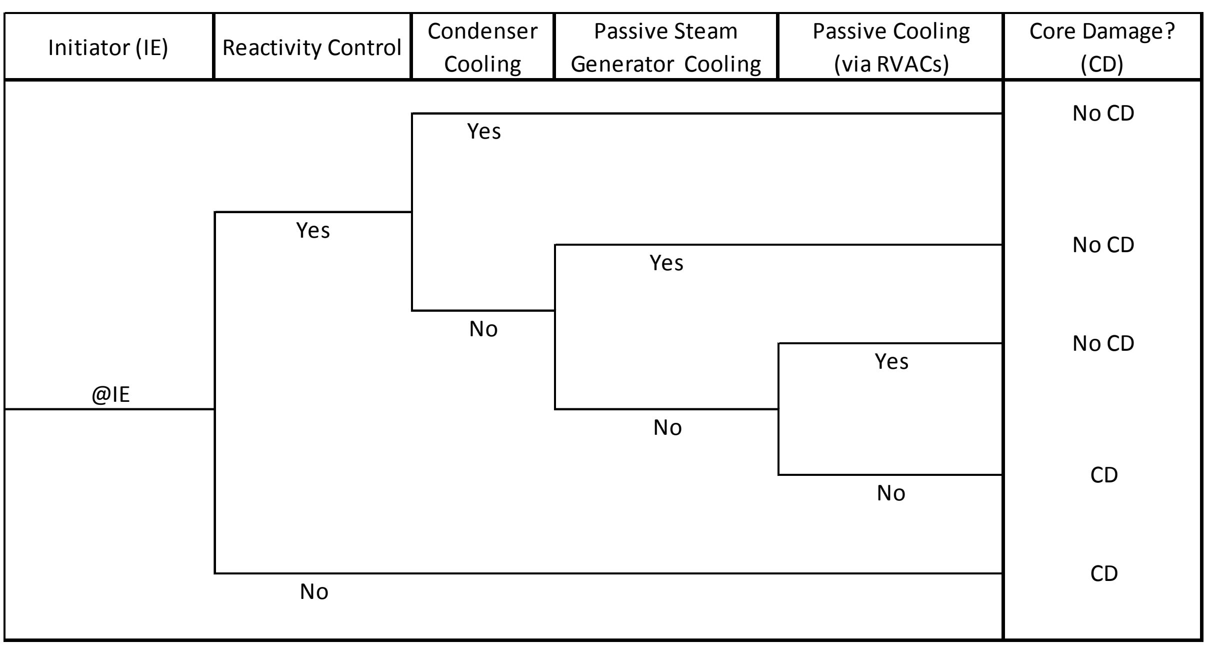

Figure A.4. Event-Tree for Simplified-Model AdvSMR

A.2.4 Systems Analysis

To model the system failures that are identified in the accident sequence analysis outlined in Section A.2.3, a system analysis is performed on each key and supporting system deemed necessary to carry out the functions delineated in the event-tree. This is done by means of fault-tree analysis, which extends down to the level of individual basic events that include component failures (e.g., failures of pumps, valves, diesel generators, etc.), unavailability of components during periods of maintenance or testing, common cause failures of redundant components and human failure events that represent the impact of human errors.

The overall mission time assumed by the PRA model is 24 hours. The mission times assumed for each component and/or system vary according to characteristics of available failure data as well as the time period over which each plant function is defined. The failure criteria for each key and supporting system are represented by the logical inverse of the accident sequence success criteria. Table A.2 presents the assumptions implicit to the success criteria defined for key systems modeled within the PRA developed for the simplified-model AdvSMR (Figure A.1); each system is also cross-referenced to the plant functions defined in Figure A.4.

For key systems with more than one train (or additional form of redundancy) available, the PRA model uses a nomenclature that assigns a module identifier (i.e., A or B) as well as a train identifier (i.e., 1, 2 or 3, as applicable) to each component. The system analysis performed on support systems and the resulting success criteria are limited to power dependencies modeled within the PRA according to the simplified electrical arrangement presented in Figure A.5. For key systems with single level of redundancy, each train is assumed to be dependent on a single electrical division (i.e., Division A or B) for power, whereas for those systems with an additional layer of redundancy, the third train may be fed from either electrical division. Two standby emergency diesel generators are assumed for the power block; however, one is assumed to be sufficient for required shutdown loads. A typical fault tree logic model that results from the system analysis, in this case for the three electromagnetic pumps that support the primary heat transport system, is shown in Figure A.6.

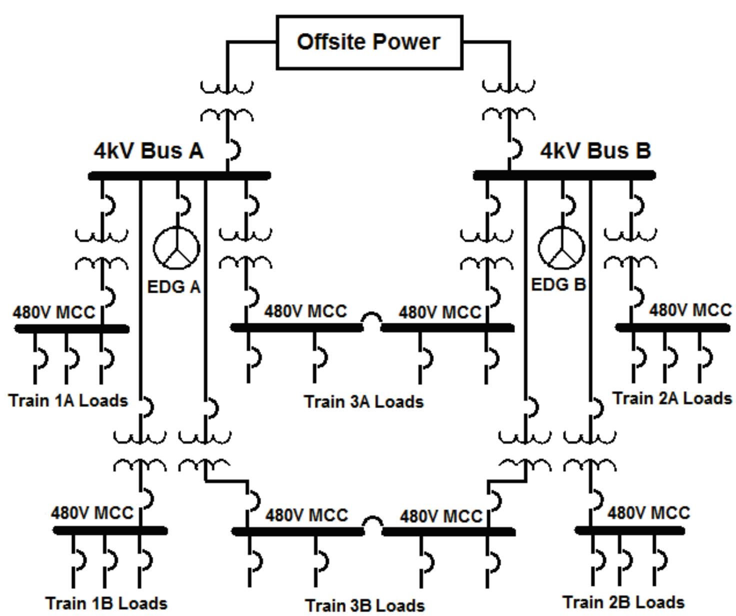

Figure A.5. Simplified One-Line Diagram of Electrical System

A.2.5 Data Analysis

Table A.4 presents the initiating event and system component failure probabilities used within the baseline PRA for the simplified-model AdvSMR (i.e., the failure probabilities when the components are as-built). Note that some components in this listing actually represent system-level failures, such as RVACS. Supporting systems, such as electrical power systems, instrumentation, and the reactor trip system, are also reflected in this list.

For this analysis, where available, industry documented failure data (Eide et al. 2007) was used to define initiating event and component failure likelihoods for the key components in the simplified-model AdvSMR design. Note, however, that some latitude was taken in adjusting these values for the simplified-model AdvSMR design, specifically for components where such data is not readily available. In these cases, assumed failure data was based on available operational experience and like-kind components.

Unavailability of components during periods of maintenance or testing was assumed to occur at intervals of 0.5, 1, and 3 weeks per year, based on the risk significance of the component and level of system and/or functional redundancy available.

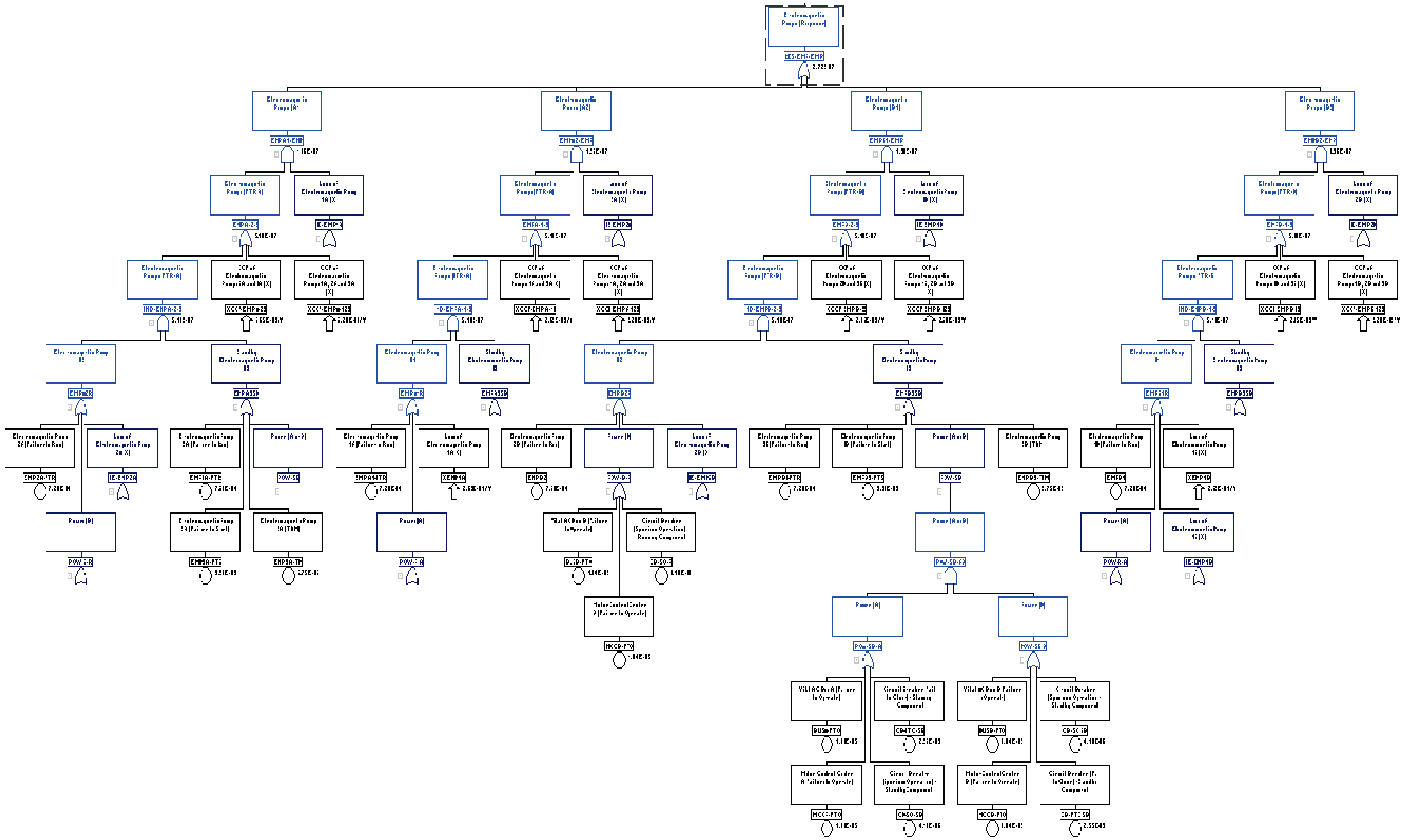

Figure A.6. System Response Model for Electromagnetic Pumps

Table A.4. Initiating Event Frequencies and Component/System Failure Rates used by the Model

Component and Failure Mode	Failure Rate	Initiator or System Failure	Assumption/Comments
Electromagnetic Pump (Failure to Run)	3.00E-05/hr	Both	Assumed unproven for NPP use. Failure rate somewhat higher than average.
Electromagnetic Pump (Failure to Start)	3.34E-03/dmd	System Failure	Assumed unproven for NPP use. Failure rate somewhat higher than average.
RVACS (Failure to Operate)	5.00E-07/hr	Both	Recovery of RVACS given it plugs was assumed to be 1E-1.
Intermediate Heat Exchanger (Tube Rupture)	8.70E-03/yr	Initiator	Assumed unproven for NPP use. Failure rate much higher than average.
Intermediate Loop Isolation Valve (Failure to Close)	7.00E-03/dmd	System Failure	Assumed to somewhat higher than NPP average. Motive power undefined.
Intermediate Loop Pump (Failure To Run)	2.00E-05/hr	Both	Failure rate assumed to be near NPP average for motor driven pumps.
Steam Generator (Tube Rupture)	8.76E-04/yr	Initiator	Assumed to be proven for NPP use. Failure rate lower than average.
SWR Pressure Relief System (Failure to Operate)	2.00E-04/dmd	System Failure	Failure rate assumed to be near NPP average for pressure relief systems.
Steam Drum	-	-	Failure of this passive component not modeled. Assumed to be small contributor to risk.
Feedwater Pump (Failure To Run)	1.00E-05/hr	Both	Failure rate assumed to be near NPP average for motor-driven pumps.
Steam Generator Louver (Failure To Open)	5.00E-02/hr	System Failure	Bounded by operator failure to open steam generator air flow louvers.
Turbine Generator	-	-	Assumed to be encompassed by reactor transient trip events.
Turbine Bypass Valve (Failure To Open)	1.00E-03/hr	System Failure	Failure rate assumed to be near NPP average.
Turbine Flow Control Valve	-	-	Assumed to be encompassed by reactor transient trip events.
Main Feed-water Pump (Failure To Run)	1.00E-05/hr	Both	Failure rate assumed to be near NPP average for motor-driven pumps.
Main Feed-water Heater	-	-	Assumed to be encompassed by reactor transient trip events.
Main Condensate Pump (Failure To Run)	1.00E-05/hr	Both	Failure rate assumed to be near NPP average for motor-driven pumps.
Emergency Diesel Generator (Failure To Start)	4.53E-03/ dmd	System Failure	Failure rate assumed to be near NPP average for emergency diesel generators.
Control Rod Drive Mechanism (Independent Failure)	5.78E-06/ dmd	System Failure	Failure rate assumed to be near NPP average for control rod drive mechanisms.
Trip Sensor (Independent Failure)	2.00E-15/ dmd	System Failure	Failure rate assumed to be near NPP average for trip sensors.
Trip Circuit Breaker (Independent Failure)	2.00E-16/ dmd	System Failure	Failure rate assumed to be near NPP average for trip circuit breakers.
Trip Set-point (Independent Failure)	3.00E-15/ dmd	System Failure	Failure rate assumed to be near NPP average for trip set-points.

Component and Failure Mode	Failure Rate	Initiator or System Failure	Assumption/Comments
Emergency Diesel Generator (Failure To Run During First Hour)	2.90E-03/ dmd	System Failure	Failure rate assumed to be near NPP average for emergency diesel generators.
Emergency Diesel Generator (Failure To Run)	8.48E-04/ dmd	System Failure	Failure rate assumed to be near NPP average for emergency diesel generators.
Motor Control Center (Failure to Operate)	4.34E-07/hr	System Failure	Failure rate assumed to be near NPP average for motor control centers.
Electrical Bus (Failure to Operate)	4.34E-07/hr	System Failure	Failure rate assumed to be near NPP average for electrical busses.
Circuit Breaker (Failure to Open/Close)	2.55E-03/hr	System Failure	Failure rate assumed to be near NPP average for circuit breakers.
Circuit Breaker (Spurious Operation)	1.71E-07/hr	System Failure	Failure rate assumed to be near NPP average for circuit breakers.
Motor-Operated Valve (Spurious Operation)	4.45E-08/hr	System Failure	Failure rate assumed to be near NPP average for motor-operated valves.
Reactor Transient (Trip)	2.50E-01/yr	Initiator	Failure rate assumed to be below average for NPP trips.

Note: Adapted from NUREG/CR-6928, Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants (Eide et al. 2007).

A.2.6 Common Cause Failure Analysis

Common cause failures (CCF) occur when multiple (usually identical) components fail due to shared causes. A CCF event consists of component failures that meet four criteria:

1. Two or more individual components fail or are degraded, including failures during demand, in-service testing or deficiencies that would have resulted in a failure if a demand signal had been received
2. Components fail within a selected period of time such that success of the PRA mission would be uncertain
3. Component failures result from a single shared cause and coupling mechanism
4. A component failure occurs within the established component boundary

The coupling mechanism classification generally consists of three major classes:

- Hardware-based
- Operation-based
- Environment-based

In the PRA model developed for the simplified-model AdvSMR design, a parametric model known as the Alpha Factor model was used to model most CCF events. This model is a multi-parameter model that can handle any redundancy level and is based on ratios of failure rates, which make the assessment of its parameters easier when no statistical data are available. Alpha factors used by the PRA model are presented by component type and failure model in Table A.5.

Table A.5. Common Cause Parameters used by the Model

Component and Failure Mode	CCCG	Alpha Factors
Electromagnetic Pump (Failure to Run)	3	α_1=9.72E-01 α_2=1.96E-02 α_3=8.44E-03
Intermediate Loop Isolation Valve (Failure to Close)	2	α_1=8.61E-01 α_2=1.39E-01
Feed-water Pump (Failure To Run)	2	α_1=8.80E-01 α_2=1.20E-01
Main Condensate Pump (Failure To Run)	2	α_1=8.80E-01 α_2=1.20E-01
Intermediate Loop Pump (Failure To Run)	2	α_1=9.90E-01 α_2=1.00E-02
Emergency Diesel Generator (Failure To Run)	2	α_1=9.60E-01 α_2=4.01E-02
Emergency Diesel Generator (Failure To Run During First Hour)	2	α_1=9.60E-01 α_2=4.01E-02
Emergency Diesel Generator (Failure To Start)	2	α_1=9.69E-01 α_2=3.12E-02

Note: Adapted from NUREG/CR-5497, Common-Cause Failure Parameter Estimations (Marshall et al. 2007).

Point value estimates representing the totality of common cause failure modes for trip circuit breakers, control rod drive mechanisms, trip set-points, and trip sensors were used in modeling the reactor shutdown or protection system.

A.2.7 Human Reliability Analysis (HRA)

HRA is a structured approach used to identify potential human failure events and to systematically estimate the probability of those events using data, models, or expert judgment. Types of human errors considered in a PRA include:

- Type A errors are made before the occurrence of the initiating event and have the potential to lead to the failure or unavailability of safety related equipment or systems.
- Type B errors that could lead to an initiating event.
- Type C errors are made during the performance of the critical actions that need to be carried out by plant operators after the occurrence of an initiating event.

For the PRA model developed for the simplified-model AdvSMR design (Figure A.1), all safety and support systems are assumed to actuate and disengage automatically and as needed through use of a highly reliable supervisory control system. As a result, a detailed HRA was not performed; however, given a failure of RVACS caused by external events such as high winds, the opportunity for recovery (e.g., unplug radiating fins) by plant operators was assumed to be possible and modeled using a conservative screening human error probability.

A.2.8 Quantification

The resulting PRA model, which consists of a single fault-tree logic model that characterizes all relevant accident sequences identified in Section A.2.3, was quantified using a fault-tree software package (EPRI 2013) used extensively within the U.S. nuclear power industry. The Boolean expressions represented by the fault-tree are reduced to arrive at the smallest combination of basic failure events (i.e., minimal cutsets) that result in a core damage event. The overall CDF for the simplified-model AdvSMR design was determined to be approximately 4.18E-07/yr based on 1358 cutsets for the power block, which consists of two modules, or 2.09E-07/yr based on 679 cutsets for an individual model. The top 100 cutsets, ranked according to CDF, are presented in Table A.6 for the power block and account for approximately 97% of the total CDF. The descriptions of all basic events that are modeled in the PRA and thus form the cutsets shown in Table A.6 are provided in Table A.7. The overall contribution to the overall CDF from each initiator identified in Table A.4 is shown in Figure A.7. Note, however, that some initiating events contribute negligibly to overall CDF and are therefore not presented in this figure.

In addition, an analysis was performed to determine the relative importance of each initiating or basic event to the overall CDF. This importance analysis considered the following four importance measures:

- RAW, which represents the relative risk increase assuming failure;
- RRW, which represents the relative risk reduction assuming perfect performance;
- F-V, which represents the fractional reduction in risk assuming perfect performance; and
- Birnbaum, which represents the difference in risk between perfect performance and assumed failure.

As discussed in Section 3.2, importance measures are valuable in sorting out the most important component failure modes and initiating events. The results of this analysis are presented in Table A.7.

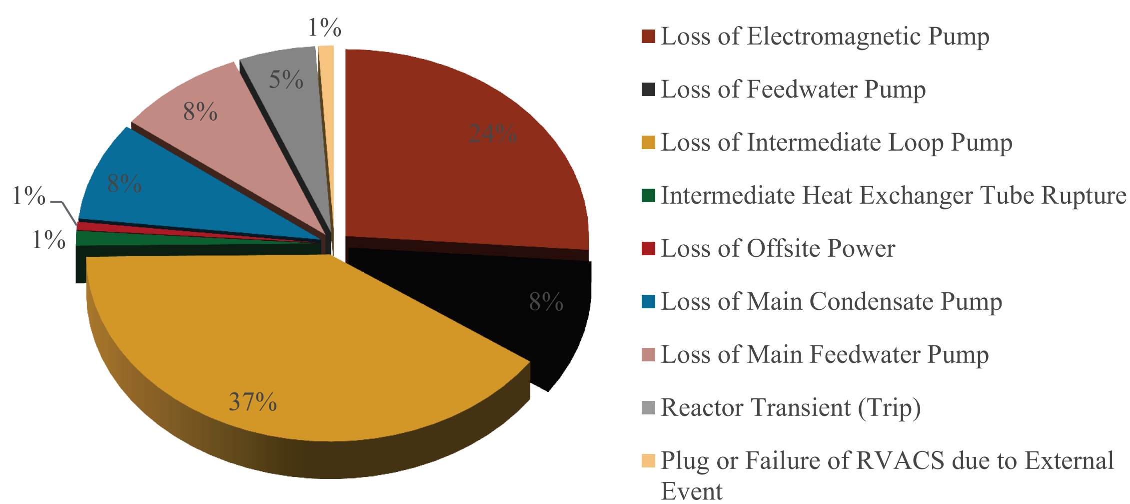

Figure A.7. Contribution of Each Initiating Event to Overall CDF

Table A.6. Dominant Cutsets for Simplified-Model AdvSMR PRA Model (Modules A and B)

CDF	Initiating Event	Subsequent Component/System Failures	
2.74E-08	%CCF-EMPA-123	RVACSA	
2.74E-08	%CCF-EMPB-123	RVACSB	
2.12E-08	%CCF-IMHPA-12	RVACSA	
2.12E-08	%CCF-IMHPB-12	RVACSB	
2.02E-08	%IMHP1A	IMHP2A-TM	RVACSA
2.02E-08	%IMHP1B	IMHP2B-TM	RVACSB
2.02E-08	%IMHP2A	IMHP1A-TM	RVACSA
2.02E-08	%IMHP2B	IMHP1B-TM	RVACSB
1.05E-08	%SGA	RVACSA	
1.05E-08	%SGB	RVACSB	
9.12E-09	%EMP1A	CCF-CDRM-RSS	
9.12E-09	%EMP1B	CCF-CDRM-RSS	
9.12E-09	%EMP2A	CCF-CDRM-RSS	
9.12E-09	%EMP2B	CCF-CDRM-RSS	
8.68E-09	%RTTA	CCF-CDRM-RSS	
8.68E-09	%RTTB	CCF-CDRM-RSS	
7.17E-09	%CCF-FWPA-12	RVACSA	SGLVA
7.17E-09	%CCF-FWPB-12	RVACSB	SGLVB
7.17E-09	%CCF-MCPA-12	RVACSA	SGLVA
7.17E-09	%CCF-MCPB-12	RVACSB	SGLVB
7.17E-09	%CCF-MFWPA-12	RVACSA	SGLVA
7.17E-09	%CCF-MFWPB-12	RVACSB	SGLVB
6.08E-09	%IMHP1A	CCF-CDRM-RSS	
6.08E-09	%IMHP1B	CCF-CDRM-RSS	
6.08E-09	%IMHP2A	CCF-CDRM-RSS	
6.08E-09	%IMHP2B	CCF-CDRM-RSS	
3.04E-09	%FWP1A	CCF-CDRM-RSS	
3.04E-09	%FWP1B	CCF-CDRM-RSS	
3.04E-09	%FWP2A	CCF-CDRM-RSS	
3.04E-09	%FWP2B	CCF-CDRM-RSS	
3.04E-09	%MCP1A	CCF-CDRM-RSS	
3.04E-09	%MCP1B	CCF-CDRM-RSS	
3.04E-09	%MCP2A	CCF-CDRM-RSS	
3.04E-09	%MCP2B	CCF-CDRM-RSS	
3.04E-09	%MFWP1A	CCF-CDRM-RSS	
3.04E-09	%MFWP1B	CCF-CDRM-RSS	
3.04E-09	%MFWP2A	CCF-CDRM-RSS	
3.04E-09	%MFWP2B	CCF-CDRM-RSS	
2.00E-09	%RVACSEE	HR-RVACSA	SGLVA
2.00E-09	%RVACSEE	HR-RVACSB	SGLVB
1.83E-09	%CCF-EMPA-12	EMP3A-TM	RVACSA
1.83E-09	%CCF-EMPB-12	EMPB3-T&M	RVACSB
1.25E-09	%LOP	CCF-CDRM-RSS	
1.22E-09	%SGA	IHIVA1	SWRPPSA1
1.22E-09	%SGA	IHIVA1	SWRPPSA2
1.22E-09	%SGA	IHIVA2	SWRPPSA1
1.22E-09	%SGA	IHIVA2	SWRPPSA2
1.22E-09	%SGB	IHIVB1	SWRPPSB1
1.22E-09	%SGB	IHIVB1	SWRPPSB2
1.22E-09	%SGB	IHIVB2	SWRPPSB1
1.22E-09	%SGB	IHIVB2	SWRPPSB2
1.01E-09	%IMHXA	IMHP1A-TM	RVACSA
1.01E-09	%IMHXA	IMHP2A-TM	RVACSA

CDF	Initiating Event	Subsequent Component/System Failures		
1.01E-09	%IMHXB	IMHP1B-TM	RVACSB	
1.01E-09	%IMHXB	IMHP2B-TM	RVACSB	
1.01E-09	%FWP1A	FWP2A-TM	RVACSA	SGLVA
1.01E-09	%FWP1B	FWP2B-TM	RVACSB	SGLVB
1.01E-09	%FWP2A	FWP1A-TM	RVACSA	SGLVA
1.01E-09	%FWP2B	FWP1B-TM	RVACSB	SGLVB
1.01E-09	%MCP1A	MCP2A-TM	RVACSA	SGLVA
1.01E-09	%MCP1B	MCP2B-TM	RVACSB	SGLVB
1.01E-09	%MCP2A	MCP1A-TM	RVACSA	SGLVA
1.01E-09	%MCP2B	MCP1B-TM	RVACSB	SGLVB
1.01E-09	%MFWP1A	MFWP2A-TM	RVACSA	SGLVA
1.01E-09	%MFWP1B	MFWP2B-TM	RVACSB	SGLVB
1.01E-09	%MFWP2A	MFWP1A-TM	RVACSA	SGLVA
1.01E-09	%MFWP2B	MFWP1B-TM	RVACSB	SGLVB
1.01E-09	%IMHP1A	IMHP2A	RVACSA	
1.01E-09	%IMHP1B	IMHP2B	RVACSB	
1.01E-09	%IMHP2A	IMHP1A	RVACSA	
1.01E-09	%IMHP2B	IMHP1B	RVACSB	
6.04E-10	%EMP1A	DEP-TSP-RPS		
6.04E-10	%EMP1B	DEP-TSP-RPS		
6.04E-10	%EMP2A	DEP-TSP-RPS		
6.04E-10	%EMP2B	DEP-TSP-RPS		
5.75E-10	%RTTA	DEP-TSP-RPS		
5.75E-10	%RTTB	DEP-TSP-RPS		
4.14E-10	%CCF-FWPA-12	CCF-CDRM-RSS		
4.14E-10	%CCF-FWPB-12	CCF-CDRM-RSS		
4.14E-10	%CCF-MCPA-12	CCF-CDRM-RSS		
4.14E-10	%CCF-MCPB-12	CCF-CDRM-RSS		
4.14E-10	%CCF-MFWPA-12	CCF-CDRM-RSS		
4.14E-10	%CCF-MFWPB-12	CCF-CDRM-RSS		
4.03E-10	%IMHP1A	DEP-TSP-RPS		
4.03E-10	%IMHP1B	DEP-TSP-RPS		
4.03E-10	%IMHP2A	DEP-TSP-RPS		
4.03E-10	%IMHP2B	DEP-TSP-RPS		
3.04E-10	%IMHXA	CCF-CDRM-RSS		
3.04E-10	%IMHXB	CCF-CDRM-RSS		
2.68E-10	%LOP	CCF-EDG-FTR	RVACSA	
2.68E-10	%LOP	CCF-EDG-FTR	RVACSB	
2.63E-10	%EMP1A	CCF-TSENS-RPS		
2.63E-10	%EMP1B	CCF-TSENS-RPS		
2.63E-10	%EMP2A	CCF-TSENS-RPS		
2.63E-10	%EMP2B	CCF-TSENS-RPS		
2.50E-10	%RTTA	CCF-TSENS-RPS		
2.50E-10	%RTTB	CCF-TSENS-RPS		
2.01E-10	%FWP1A	DEP-TSP-RPS		
2.01E-10	%FWP1B	DEP-TSP-RPS		
2.01E-10	%FWP2A	DEP-TSP-RPS		
4.06E-07[(a)]				

(a) Total CDF of the top 100 cutsets.

Table A.7. Basic Events Descriptions and Importance Analysis Results

Basic Event Name	Probability	F-V	Birnbaum	RRW	RAW	Basic Event Description
%CCF-EMPA-12	2.65E-03	5.13E-03	8.08E-07	1.01E+00	2.93E+00	CCF of Electromagnetic Pumps 1A and 2A (Initiating Event)
%CCF-EMPA-123	2.28E-03	6.58E-02	1.20E-05	1.07E+00	2.98E+01	CCF of Electromagnetic Pumps 1A, 2A and 3A (Initiating Event)
%CCF-EMPA-13	2.65E-03	2.50E-04	3.93E-08	1.00E+00	1.09E+00	CCF of Electromagnetic Pumps 1A and 3A (Initiating Event)
%CCF-EMPA-23	2.65E-03	2.50E-04	3.93E-08	1.00E+00	1.09E+00	CCF of Electromagnetic Pumps 2A and 3A (Initiating Event)
%CCF-EMPB-12	2.65E-03	5.13E-03	8.08E-07	1.01E+00	2.93E+00	CCF of Electromagnetic Pumps 1B and 2B (Initiating Event)
%CCF-EMPB-123	2.28E-03	6.58E-02	1.20E-05	1.07E+00	2.98E+01	CCF of Electromagnetic Pumps 1B, 2B and 3B (Initiating Event)
%CCF-EMPB-13	2.65E-03	2.50E-04	3.93E-08	1.00E+00	1.09E+00	CCF of Electromagnetic Pumps 1B and 3B (Initiating Event)
%CCF-EMPB-23	2.65E-03	2.50E-04	3.93E-08	1.00E+00	1.09E+00	CCF of Electromagnetic Pumps 2B and 3B (Initiating Event)
%CCF-FWPA-12	1.19E-02	1.83E-02	6.39E-07	1.02E+00	2.51E+00	CCF Involving Loss of Feed-water Pumps 1A and 2A (Initiating Event)
%CCF-FWPB-12	1.19E-02	1.83E-02	6.39E-07	1.02E+00	2.51E+00	CCF Involving Loss of Feed-water Pumps 1A and 2A (Initiating Event)
%CCF-IMHPA-12	1.77E-03	5.10E-02	1.20E-05	1.05E+00	2.98E+01	CCF Involving Loss of Intermediate Loop Pump 1A and 2A (Initiating Event)
%CCF-IMHPB-12	1.77E-03	5.10E-02	1.20E-05	1.05E+00	2.98E+01	CCF Involving Loss of Intermediate Loop Pump 1B and 2B (Initiating Event)
%CCF-MCPA-12	1.19E-02	1.83E-02	6.39E-07	1.02E+00	2.51E+00	CCF Involving Loss of Main Condensate Pumps 1A and 2A (Initiating Event)
%CCF-MCPB-12	1.19E-02	1.83E-02	6.39E-07	1.02E+00	2.51E+00	CCF Involving Loss of Main Condensate Pumps 1B and 2B (Initiating Event)
%CCF-MFWPA-12	1.19E-02	1.83E-02	6.39E-07	1.02E+00	2.51E+00	CCF Involving Loss of Main Feed-water Pumps 1A and 2A (Initiating Event)

Basic Event Name	Probability	F-V	Birnbaum	RRW	RAW	Basic Event Description
%CCF-MFWPB-12	1.19E-02	1.83E-02	6.39E-07	1.02E+00	2.51E+00	CCF Involving Loss of Main Feed-water Pumps 1B and 2B (Initiating Event)
%EMP1A	2.63E-01	2.47E-02	3.93E-08	1.03E+00	1.07E+00	Loss of Electromagnetic Pump 1A (Initiating Event)
%EMP1B	2.63E-01	2.47E-02	3.93E-08	1.03E+00	1.07E+00	Loss of Electromagnetic Pump 1B (Initiating Event)
%EMP2A	2.63E-01	2.47E-02	3.93E-08	1.03E+00	1.07E+00	Loss of Electromagnetic Pump 2A (Initiating Event)
%EMP2B	2.63E-01	2.47E-02	3.93E-08	1.03E+00	1.07E+00	Loss of Electromagnetic Pump 2B (Initiating Event)
%FWP1A	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Feed-water Pump 1A (Initiating Event)
%FWP1B	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Feed-water Pump 1B (Initiating Event)
%FWP2A	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Feedwater Pump 2A (Initiating Event)
%FWP2B	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Feed-water Pump 2B (Initiating Event)
%IMHP1A	1.75E-01	6.72E-02	1.60E-07	1.07E+00	1.32E+00	Loss of Intermediate Loop Pump 1A (Initiating Event)
%IMHP1B	1.75E-01	6.72E-02	1.60E-07	1.07E+00	1.32E+00	Loss of Intermediate Loop Pump 1B (Initiating Event)
%IMHP2A	1.75E-01	6.72E-02	1.60E-07	1.07E+00	1.32E+00	Loss of Intermediate Loop Pump 2A (Initiating Event)
%IMHP2B	1.75E-01	6.72E-02	1.60E-07	1.07E+00	1.32E+00	Loss of Intermediate Loop Pump 2B (Initiating Event)
%IMHXA	8.76E-03	5.91E-03	2.82E-07	1.01E+00	1.67E+00	Intermediate Heat Exchanger Tube Rupture on Module A (Initiating Event)
%IMHXB	8.76E-03	5.91E-03	2.82E-07	1.01E+00	1.67E+00	Intermediate Heat Exchanger Tube Rupture on Module B (Initiating Event)
%LOP	3.59E-02	6.58E-03	7.65E-08	1.01E+00	1.18E+00	Loss of Offsite Power (Initiating Event)
%MCP1A	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Main Condensate Pump 1A (Initiating Event)

Basic Event Name	Probability	F-V	Birnbaum	RRW	RAW	Basic Event Description
%MCP1B	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Main Condensate Pump 1B (Initiating Event)
%MCP2A	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Main Condensate Pump 2A (Initiating Event)
%MCP2B	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Main Condensate Pump 2B (Initiating Event)
%MFWP1A	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Main Feed-water Pump 1A (Initiating Event)
%MFWP1B	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Main Feedwater Pump 1B (Initiating Event)
%MFWP2A	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Main Feed-water Pump 2A (Initiating Event)
%MFWP2B	8.76E-02	1.06E-02	5.04E-08	1.01E+00	1.11E+00	Loss of Main Feed-water Pump 2B (Initiating Event)
%RTTA	2.50E-01	2.35E-02	3.93E-08	1.02E+00	1.07E+00	Reactor Transient Trip on Module A (Initiating Event)
%RTTB	2.50E-01	2.35E-02	3.93E-08	1.02E+00	1.07E+00	Reactor Transient Trip on Module B (Initiating Event)
%RVACSEE	4.00E-07	9.56E-03	9.98E-03	1.01E+00	2.39E+04	Plug or Failure of RVACS on Modules A and B due to External Event (Initiating Event)
%SGA	8.76E-04	3.80E-02	1.81E-05	1.04E+00	4.43E+01	Steam Generator Tube Rupture on Module A (Initiating Event)
%SGB	8.76E-04	3.80E-02	1.81E-05	1.04E+00	4.43E+01	Steam Generator Tube Rupture on Module B (Initiating Event)
BUSA-FTO	1.04E-05	1.50E-04	5.84E-06	1.00E+00	1.50E+01	Vital AC Bus A (Failure to Operate)
BUSB-FTO	1.04E-05	1.50E-04	5.84E-06	1.00E+00	1.50E+01	Vital AC Bus B (Failure to Operate)
CB-FTC-SB	2.55E-03	4.50E-04	7.31E-08	1.00E+00	1.18E+00	Circuit Breaker (Fail to Close) - Standby Component
CB-SO-R	4.10E-06	1.10E-04	1.08E-05	1.00E+00	2.68E+01	Circuit Breaker (Spurious Operation) - Running Component
CB-SO-SB	4.10E-06	0.00E+00	7.31E-08	1.00E+00	1.18E+00	Circuit Breaker (Spurious Operation) - Standby Component
CCF-CDRM-RSS	3.47E-08	2.87E-01	3.45E+00	1.40E+00	8.27E+06	Control Rod Drive Mechanisms (Common Cause Failure)

Basic Event Name	Probability	F-V	Birnbaum	RRW	RAW	Basic Event Description
CCF-EDG-FTR	6.22E-04	1.28E-03	8.62E-07	1.00E+00	3.06E+00	CCF Involving EDGs (Failure to Run)
CCF-EDG-FTR-1HR	9.33E-05	1.90E-04	8.62E-07	1.00E+00	3.06E+00	CCF Involving EDGs (Failure to Run during First Hour)
CCF-EDG-FTS	1.46E-04	3.00E-04	8.62E-07	1.00E+00	3.06E+00	CCF Involving EDGs (Failure to Start)
CCF-IHIVA	1.13E-03	9.50E-04	3.50E-07	1.00E+00	1.84E+00	Intermediate Isolation Valves 1A and 2A (Failure to Close)
CCF-IHIVB	1.13E-03	9.50E-04	3.50E-07	1.00E+00	1.84E+00	Intermediate Isolation Valves 1B and 2B (Failure to Close)
CCF-TSENS-RPS	1.00E-09	8.27E-03	3.45E+00	1.01E+00	8.27E+06	Trip Sensors (Common Cause Failure)
CDRM1-RSS	5.78E-06	1.38E-03	9.98E-05	1.00E+00	2.40E+02	Control Rod Drive Mechanism 1 (Failure to Insert)
CDRM2-RSS	5.78E-06	1.38E-03	9.98E-05	1.00E+00	2.40E+02	Control Rod Drive Mechanism 2 (Failure to Insert)
CDRM3-RSS	5.78E-06	1.38E-03	9.98E-05	1.00E+00	2.40E+02	Control Rod Drive Mechanism 3 (Failure to Insert)
CDRM4-RSS	5.78E-06	1.38E-03	9.98E-05	1.00E+00	2.40E+02	Control Rod Drive Mechanism 4 (Failure to Insert)
CDRM5-RSS	5.78E-06	1.38E-03	9.98E-05	1.00E+00	2.40E+02	Control Rod Drive Mechanism 5 (Failure to Insert)
CDRM6-RSS	5.78E-06	1.38E-03	9.98E-05	1.00E+00	2.40E+02	Control Rod Drive Mechanism 6 (Failure to Insert)
DEP-TCB-RPS	2.00E-10	1.65E-03	3.45E+00	1.00E+00	8.27E+06	Trip Circuit Breakers (Common Cause Failure)
DEP-TSP-RPS	2.30E-09	1.90E-02	3.45E+00	1.02E+00	8.27E+06	Trip Setpoints (Common Cause Failure)
EDGA-FTR	1.93E-02	1.07E-03	2.30E-08	1.00E+00	1.05E+00	Emergency Diesel A (Fails to Run after First Hour)
EDGA-FTR-1HR	2.90E-03	1.60E-04	2.30E-08	1.00E+00	1.06E+00	Emergency Diesel A (Failure to Load and Run during First Hour)
EDGA-FTS	4.52E-03	2.50E-04	2.30E-08	1.00E+00	1.06E+00	Emergency Diesel A (Failure to Start)
EDGB-FTR	1.93E-02	1.07E-03	2.30E-08	1.00E+00	1.05E+00	Emergency Diesel B (Fails to Run after First Hour)
EDGB-FTR-1HR	2.90E-03	1.60E-04	2.30E-08	1.00E+00	1.06E+00	Emergency Diesel B (Failure to Load and Run during First Hour)
EDGB-FTS	4.52E-03	2.50E-04	2.30E-08	1.00E+00	1.06E+00	Emergency Diesel B (Failure to Start)

Basic Event Name	Probability	F-V	Birnbaum	RRW	RAW	Basic Event Description
EMP2A-FTR	7.20E-04	3.50E-04	2.04E-07	1.00E+00	1.49E+00	Electromagnetic Pump 2A (Failure to Run)
EMP3A-FTR	7.20E-04	6.00E-05	3.65E-08	1.00E+00	1.09E+00	Electromagnetic Pump 3A (Failure to Run)
EMP3A-FTS	3.33E-03	2.90E-04	3.65E-08	1.00E+00	1.09E+00	Electromagnetic Pump 3A (Failure to Start)
EMP3A-TM	5.75E-02	5.03E-03	3.65E-08	1.01E+00	1.08E+00	Electromagnetic Pump 3A (T&M)
EMPA1-FTR	7.20E-04	3.50E-04	2.04E-07	1.00E+00	1.49E+00	Electromagnetic Pump 1A (Failure to Run)
EMPB1	7.20E-04	3.50E-04	2.04E-07	1.00E+00	1.49E+00	Electromagnetic Pump 1B (Failure to Run)
EMPB2	7.20E-04	3.50E-04	2.04E-07	1.00E+00	1.49E+00	Electromagnetic Pump 2B (Failure to Run)
EMPB3-FTR	7.20E-04	6.00E-05	3.65E-08	1.00E+00	1.09E+00	Electromagnetic Pump 3B (Failure to Run)
EMPB3-FTS	3.33E-03	2.90E-04	3.65E-08	1.00E+00	1.09E+00	Electromagnetic Pump 3B (Failure to Start)
EMPB3-T&M	5.75E-02	5.03E-03	3.65E-08	1.01E+00	1.08E+00	Electromagnetic Pump 3B (T&M)
FWP1A	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Feed-water Pump 1A (Failure to Run)
FWP1A-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Feed-water Pump 1A (T&M)
FWP1B	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Feed-water Pump 1B (Failure to Run)
FWP1B-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Feed-water Pump 1B (T&M)
FWP2A	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Feed-water Pump 2A (Failure to Run)
FWP2A-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Feed-water Pump 2A (T&M)
FWP2B	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Feed-water Pump 2B (Failure to Run)
FWP2B-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Feed-water Pump 2B (T&M)
HR-RVACSA	1.00E-01	4.79E-03	1.99E-08	1.00E+00	1.04E+00	Op Action - Failure to Recover RVACS
HR-RVACSB	1.00E-01	4.79E-03	1.99E-08	1.00E+00	1.04E+00	Op Action - Failure to Recover RVACS
IHIVA-SO1	1.07E-06	0.00E+00	3.50E-07	1.00E+00	1.84E+00	Intermediate Isolation Valve 1A (Sprious Operation)
IHIVA-SO2	1.07E-06	0.00E+00	3.50E-07	1.00E+00	1.84E+00	Intermediate Isolation Valve 2A (Spurious Operation)
IHIVA1	6.98E-03	5.85E-03	3.50E-07	1.01E+00	1.83E+00	Intermediate Isolation Valve 1A (Failure to Close)
IHIVA2	6.98E-03	5.85E-03	3.50E-07	1.01E+00	1.83E+00	Intermediate Isolation Valve 2A (Failure to Close)

Basic Event Name	Probability	F-V	Birnbaum	RRW	RAW	Basic Event Description
IHIVB-SO1	1.07E-06	0.00E+00	3.50E-07	1.00E+00	1.84E+00	Intermediate Isolation Valve 1B (Spurious Operation)
IHIVB-SO2	1.07E-06	0.00E+00	3.50E-07	1.00E+00	1.84E+00	Intermediate Isolation Valve 2B (Spurious Operation)
IHIVB1	6.98E-03	5.85E-03	3.50E-07	1.01E+00	1.83E+00	Intermediate Isolation Valve 1B (Failure to Close)
IHIVB2	6.98E-03	5.85E-03	3.50E-07	1.01E+00	1.83E+00	Intermediate Isolation Valve 2B (Failure to Close)
IMHP1A	4.80E-04	2.54E-03	2.21E-06	1.00E+00	6.29E+00	Intermediate Loop Pump 1A (Failure to Run)
IMHP1A-TM	9.62E-03	5.08E-02	2.21E-06	1.05E+00	6.24E+00	Intermediate Loop Pump 1A (T&M)
IMHP1B	4.80E-04	2.54E-03	2.21E-06	1.00E+00	6.29E+00	Intermediate Loop Pump 1B (Failure to Run)
IMHP1B-TM	9.62E-03	5.08E-02	2.21E-06	1.05E+00	6.24E+00	Intermediate Loop Pump 1B (T&M)
IMHP2A	4.80E-04	2.54E-03	2.21E-06	1.00E+00	6.29E+00	Intermediate Loop Pump 2A (Failure to Run)
IMHP2A-TM	9.62E-03	5.08E-02	2.21E-06	1.05E+00	6.24E+00	Intermediate Loop Pump 2A (T&M)
IMHP2B	4.80E-04	2.54E-03	2.21E-06	1.00E+00	6.29E+00	Intermediate Loop Pump 2B (Failure to Run)
IMHP2B-TM	9.62E-03	5.08E-02	2.21E-06	1.05E+00	6.24E+00	Intermediate Loop Pump 2B (T&M)
IND-TCB-RPS	2.00E-16	0.00E+00	3.45E+00	1.00E+00	8.27E+06	Trip Circuit Breakers (Independent Failure)
IND-TSENS-RPS	2.00E-15	0.00E+00	3.45E+00	1.00E+00	8.27E+06	Trip Sensors (Independent Failure)
IND-TSPS-RPS	3.00E-15	0.00E+00	3.45E+00	1.00E+00	8.27E+06	Trip Set-points (Independent Failure)
MCCA-FTO	1.04E-05	1.50E-04	5.84E-06	1.00E+00	1.50E+01	Motor Control Center A (Failure to Operate)
MCCB-FTO	1.04E-05	1.50E-04	5.84E-06	1.00E+00	1.50E+01	Motor Control Center B (Failure to Operate)
MCP1A	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Loss of Main Condensate Pump 1A (Failure to Run)
MCP1A-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Loss of Main Condensate Pump 1A (T&M)
MCP1B	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Loss of Main Condensate Pump 1B (Failure to Run)
MCP1B-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Loss of Main Condensate Pump 1B (T&M)
MCP2A	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Loss of Main Condensate Pump 2A (Failure to Run)

Basic Event Name	Probability	F-V	Birnbaum	RRW	RAW	Basic Event Description
MCP2A-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Loss of Main Condensate Pump 2A (T&M)
MCP2B	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Loss of Main Condensate Pump 2B (Failure to Run)
MCP2B-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Loss of Main Condensate Pump 2B (T&M)
MFWP1A	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Main Feed-water Pump 1A (Failure to Run)
MFWP1A-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Main Feed-water Pump 1A (T&M)
MFWP1B	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Main Feed-water Pump 1B (Failure to Run)
MFWP1B-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Main Feed-water Pump 1B (T&M)
MFWP2A	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Main Feed-water Pump 2A (Failure to Run)
MFWP2A-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Main Feed-water Pump 2A (T&M)
MFWP2B	2.40E-04	3.00E-05	5.26E-08	1.00E+00	1.13E+00	Main Feed-water Pump 2B (Failure to Run)
MFWP2B-TM	1.92E-02	2.42E-03	5.26E-08	1.00E+00	1.12E+00	Main Feed-water Pump 2B (T&M)
RVACSA	1.20E-05	3.22E-01	1.12E-02	1.48E+00	2.69E+04	Reactor Vessel Auxiliary Cooling System A (Failure to Operate)
RVACSB	1.20E-05	3.22E-01	1.12E-02	1.48E+00	2.69E+04	Reactor Vessel Auxiliary Cooling System B (Failure to Operate)
SGLVA	5.00E-02	7.14E-02	5.96E-07	1.08E+00	2.36E+00	Steam Generator Louvers A (Failure to Open)
SGLVB	5.00E-02	7.14E-02	5.96E-07	1.08E+00	2.36E+00	Steam Generator Louvers B (Failure to Open)
SWRPPSA1	2.00E-04	6.35E-03	1.32E-05	1.01E+00	3.26E+01	Sodium-Water-Reaction Pressure Relief Valve 1A (Failure to Open)
SWRPPSA2	2.00E-04	6.35E-03	1.32E-05	1.01E+00	3.26E+01	Sodium-Water-Reaction Pressure Relief Valve 2A (Failure to Open)
SWRPPSB1	2.00E-04	6.35E-03	1.32E-05	1.01E+00	3.26E+01	Sodium-Water-Reaction Pressure Relief Valve 1B (Failure to Open)
SWRPPSB2	2.00E-04	6.35E-03	1.32E-05	1.01E+00	3.26E+01	Sodium-Water-Reaction Pressure Relief Valve 2B (Failure to Open)
TBVFTO	1.00E-03	7.20E-04	3.00E-07	1.00E+00	1.72E+00	Turbine Bypass Valve (Failure to Open)

A.3 Alternate Risk Metrics for Nuclear Power Reactors

Advanced reactors, and AdvSMRs (based on modularization of advanced reactor concepts) may provide a longer-term alternative to traditional LWR concepts. Information on component condition and failure probability in these reactor concepts will be critical to maintaining adequate safety margins and avoiding unplanned shutdowns, both of which have regulatory and economic consequences. In particular, information on component condition will be needed for characterizing the risk (in terms of both safety and economic metrics) to optimize operations and maintenance (O&M) planning, and controlling O&M costs, by:

- Maximizing generation through assessment of the potential impact of taking key components offline for testing or maintenance,
- Supporting reduced staffing needs by assessing the contribution of individual components to changes in risk and using this information to optimize inspection and maintenance activities, and
- Enabling real-time decisions on stress-relief for risk-significant equipment susceptible to degradation and damage, thereby enabling lifetime management.

System risk in current NPPs is computed using risk monitors that provide a point-in-time estimate of risk given the current plant configuration (e.g., equipment availability, operational regime, and environmental conditions). Traditional risk metrics such as CDF provide a measure of the risk associated with safety-related consequences of the plant configuration.

While this is a useful metric for ARs and AdvSMRs, the increased reliance of these designs on passive safety features is likely to result in very low CDF values that reduce the utility of this particular metric. Instead, given the need to reduce O&M costs and provide a predictive estimate of future risk, metrics that capture the risk of plant unavailability to meet its mission needs (whether electrical generation or process heat or some combination of the two) are likely to be of more relevance. In particular, such metrics provide a quantitative mechanism for understanding the impact to mission of the probability of component failure and consequent unavailability.

The development of alternative metrics of risk may be informed by applying lessons learned from the LWR community. Operational experience from the LWR community suggests that optimization of nuclear power plant outages and equipment life management can significantly impact plant availability, operations, safety, and other costs associated with a nuclear power plant. As a result, extensive efforts are directed towards comprehensive planning of outages to minimize outage extensions, radiation exposure, and plant unreliability (IAEA 2002) while maximize worker safety. Given the competition from increased production of natural gas, the economics of plant life management have become a crucial factor in being successful in competitive electricity markets (OECD 2000), and the price of excessive outage extension or inadequate plant equipment management can lead to a nuclear power facility that is no longer cost-effective to operate.

The development of alternative risk metrics will require a study of the direct inputs to the ERM and the factors that influence these direct inputs. These direct inputs to ERM are:

- Component failure rate
- Component service life
- Component mission time in an accident
- Component test interval

- Component repair time
- Component failure thresholds. This is a quantitative relationship between the component condition, and failure to perform its mission.

Data from test reactors indicate that the component failure rate is highly variable, and may be somewhat cyclic. However, the cyclic nature of the failure rate may be partly due to the replacement or repair of failing components, which will skew the distribution somewhat. Besides the maintenance practices and maintenance intervals, other factors that have the potential to impact these direct inputs include:

- Component aging management policies (note that these may be informed by regulatory guidance). This includes inspection and test intervals, dictating the frequency with which failing equipment may be identified prior to failure. Equipment condition monitoring is an alternative to periodic inspection/testing, providing information on equipment condition in near-real-time.
- Spare parts inventory and lead time for obtaining spare parts. This dictates how far in advance of maintenance or repair action equipment condition will need to be determined.

These influential factors represent the tradeoffs that define the O&M decision making, and therefore represent targets against which suitable risk-based metrics may be developed. Such metrics could include an array of concepts (quantities in parentheses indicate measurement units in terms of cost and/or time offline):

- <u>Loss of generation capacity</u>. This is related to the potential for unanticipated shutdown of the plant due to aging or degraded components. This may be measured in terms of cost, and number of days the generation will be offline. This metric may be computed in a predictive fashion (i.e., risk of lost generation capacity as the component ages over time).
- <u>Extended or permanent shutdown of the plant</u>. This is related to loss of generation capacity (metric above), and is a measure of the total time that generation capacity may be lost.
- <u>Expected number of outages due to equipment failure</u>. Note that this metric is a measure of the cost of not changing maintenance schedules, given the equipment condition and projected failure probability.
- <u>Extended outage time because of equipment failure</u>. This quantity is a measure of the potential for extended outages given the condition of certain equipment, if they are not on the outage plan for maintenance or repair.
- <u>Deferred equipment maintenance</u>. This quantity provides a measure of the change in cost due to deferred maintenance of equipment, given their condition and the projected probabilities of failure. In some instances (when the condition of the equipment is better than anticipated, for instance), the change in cost may be negative indicating a net benefit.
- <u>Regulatory compliance</u> with required testing and maintenance actions, and the consequence of failures due to changed testing/inspection schedules. This is likely to be difficult to quantify, though additional cost required to move back into regulatory compliance may be a mechanism for quantifying this metric.

Some of these metrics may need to be normalized appropriately (for instance, with respect to normal operations) to ensure that the metrics are appropriately bounded. A number of open questions will need answering before these metrics may be evaluated or used routinely. These include:

- Fraction of equipment replaced (perhaps unnecessarily) or repaired before it becomes a problem.
- Cost ($, hours for maintenance, etc.) of equipment replacement/repair/maintenance during planned outages.

- Approaches to prioritizing equipment for replacement/repair/maintenance during an outage.
- Frequency of preventive maintenance and effort spent on preventive maintenance. Note that preventive maintenance may not necessarily need plant shutdown.
- Consequences of equipment failures (such as unplanned shutdowns), and the impact of mitigation strategies. For example, not all equipment failures may lead to unplanned shutdowns and there may be cases where the plant can bring online a spare to continue generating while repairs/replacements are made. In the latter case, however, there is likely to be additional effort (cost, resources) spent on replacement/repairs. Note that this may not show up as lost generation or loss of capacity factor although there is a cost associated with this.
- Average duration and cost of unplanned outages.

Appendix B

Equipment Condition Assessment (ECA) and Component Reliability

Appendix B

Equipment Condition Assessment (ECA) and Component Reliability

B.1 Component Reliability Information from Existing Databases

As discussed earlier, ARs and AdvSMRs are expected to utilize components that, functionally, are similar to those used in currently operational and test reactors. For the purposes of calculating risk as a function of component degradation, baseline data on component failure rates are useful to bound the initial POF as well as expected failure rates as the component ages.

In the case of components on the secondary side, reliability data from currently operational plants may be used for this purpose, assuming that the secondary side of AdvSMRs is likely to serve similar functions (electrical generation, rejection of excess heat). However, several components are likely to be unique to advanced reactor concepts—components such as electromagnet (EM) pumps and intermediate heat exchangers. Reliability data on these components is limited. These data sets were primarily generated through the operation of a few test reactors, and with most of these test reactors no longer in operation, the accessibility of these data sets is greatly reduced.

To address this issue, two steps were taken. First, we began a systematic search of component reliability data that may be relevant to the generic liquid-metal cooled AdvSMR (Figure A.1) that is being used as a case-study for the enhanced risk monitor (ERM). Data that may be relevant from the Fast Flux Test Facility (FFTF) and EBR-II operation were collated into a database (Centralized Reliability Data Organization [CREDO] database) and efforts were initiated to assess the availability and relevance of the data. In parallel, similar data from other test reactors (such as N-reactor on the Hanford site) were also examined for availability and applicability. Details of these data sets, and the status of searches for the data, are described in Appendix C for FFTF. In the interim, component failure rates from published literature (where available) were used to initialize the ERM for the generic AdvSMR design (Appendix A), and where unavailable, augmented with failure rates from like-kind components.

B.2 Component Failure Trends Using Maintenance Records

Current probabilistic risk assessments (PRA) on nuclear reactors and the risk monitors associated with them are based on average failure rates that do not account for the instantaneous condition of individual equipment or components. Having time-based information about component failures can help inform ERMs. Using over 700 work authorizations from the Hanford N Reactor (in Richland, Washington) between 1973 and 1987, trend analysis was performed for the exhaust fans, dampers, horizontal control rods, drive turbines, circulating raw water pumps and batteries to determine how failure rates change over time. The results showed cyclic failure rates slowly decreasing over the 15 year period, with inflection points caused by replacement or refurbishment of deteriorating components. Statistical analysis showed a generally high confidence interval for the negative slope of the trend-line. This more nuanced component failure information can be used to generate a more dynamic risk profile to inform operations and maintenance activities for nuclear power plants.

B.2.1 Introduction

One major concern about reliance on nuclear energy is the risks associated with maintaining a nuclear plant. In light of nuclear disasters such as Fukushima and Chernobyl, accurate risk profiles and assessments become pivotal to advances in nuclear technology, and are one motivation behind a model that the U.S. Department of Energy's Pacific Northwest National Laboratory (PNNL) is developing (Hore-Lacy 2012). A second motivation is the high cost incurred by the failure of components that lead to the shutdown of a nuclear power plant. Because of the compact design and lengthened operating hours of AdvSMRs, the inspections that conventionally evaluate component condition are less practical, and additional risk assessment tools must be employed. Traditional PRA models assume average component failure rates, even though it is understood that failures rates can change over time. As a result, researchers at PNNL are developing an ERM concept which integrates component condition into failure estimates for a dynamic approach to the PRA (Ramuhalli et al. 2014). This model captures how failure rates over time. Using these real-time models to inform plant decision-making and maintenance planning is expected to make current and future AdvSMRs safer and more affordable.

B.2.2 Purpose of this Study

The purpose of this investigation was to generate insight about component aging in order to inform how condition monitoring might best be integrated into an ERM. Data from maintenance records and work histories of previously operational reactors can provide this information about the time-dependence of failure rates. Extensive work authorizations have been archived for one such reactor, the Hanford N Reactor, and are distinct from industry data in their accessibility and level of detail. This data was compiled electronically and analyzed to determine how components age.

B.2.3 Background

In 1990, trend analyses of 17 groups of components from the Hanford N Reactor were conducted as part of a Level 1 PRA. The trend analysis consisted of plotting the number of failures per year for each component against time and creating a linear regression to determine the slope of the trend-line. It was predicted that aging would impede component function, causing an upward trend with a positive slope. However, failure rates increased in only two systems: the Low Pressure Injection System and the Ball Safety System. These two systems were analyzed in more detail, and it was determined that the majority of Ball Safety System failures were due to air supply failures, which increased over time, while all other Ball System failures decreased. The 1990 PRA Final Report (Zentner et al. 1990, Section 1.0, pp. 27-53) concluded that "aging effects are not highly significant at N Reactor…due in some measure to… N Reactor programs implemented to counteract the onset of aging." These programs included preventative maintenance in safety-related structures, an in-service inspection program, upgrades to industry-wide engineered safety features, and two programs to restore components to their original conditions (Zentner et al. 1990).

With the goal of informing ERMs with real-time component condition information, the trend analysis of the 1990 PRA was re-visited with several differences in trending method. Six components that were not analyzed in 1990 and that are of particular interest to AdvSMR technologies were chosen and trended across the 15 year period from 1973-1987. Though the 1990 PRA (Zentner et al. 1990) only trended yearly failures and cumulative failures, this study surveyed the data further based on failure mode and severity, which are important in understanding a potential correlation between time, the state of the reactor, and the condition of the component. In addition to searching for an overall increase or decrease in component failure rates, this study surveyed the effects of time on component failure rates at the monthly and yearly scale. Lastly, instead of trending the number of failures, this analysis standardized the data using the component type population, generating a per-component failure rate that can be compared to other component types. A more comprehensive analysis of six particularly relevant components (below in Table B.1) affords a clearer picture of how failure rates change over time.

Table B.1. Component Population Table

Component	Population
Dampers	23
Exhaust Fans	22
Horizontal Control Rods	107
Drive Turbines	6
Batteries	10
Circulating Raw Water Pumps	16

B.2.4 Data Compilation Activity

The data from the Hanford N Reactor exists in the form of work authorizations, such as the example shown in Figure B.1, which were hand-written forms filled out to request maintenance in a particular area of the plant. Listed on each work authorization are the date, component name, job title, urgency level and description of the failure for each maintenance request. Data from work authorizations were compiled for six components: exhaust fans, dampers, horizontal control rods (HCRs), primary drive turbines, circulating raw water pumps and 125 volt batteries, totaling 723 work authorizations. These components were selected in part because the number of work authorizations on file provided a large data pool for comprehensive analysis.

Component failures were categorized as either demand-related or time-related, based on the description of the failure included on each work authorization. A demand-related failure is defined as the failure of a component to respond to a specific command given by an operator, such as the failure of a valve to open. Time-related failure is the failure of a component due to sustained operation, such as a pump that continues to run. The distinction is significant because mathematically, average failure rates are calculated on a per-demand or per-operating hour basis, depending on the failure mode. Because the 125 volt batteries were always in operation, all battery failures were considered time-related.

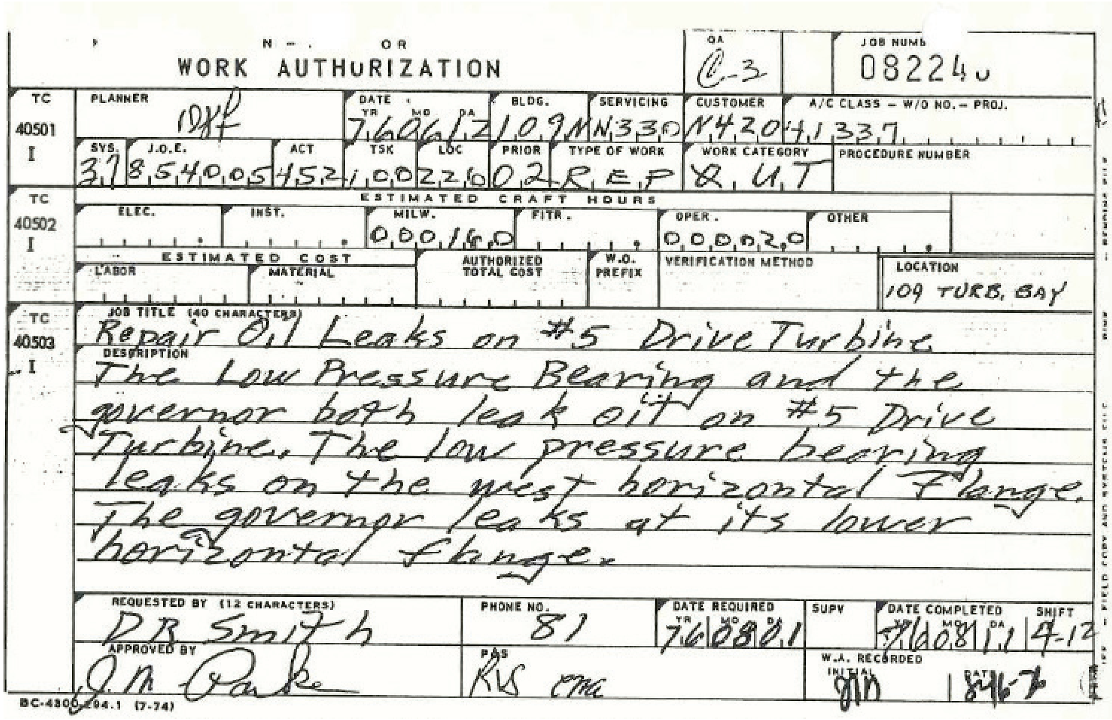
WORK AUTHORIZATION
QA: C-3
JOB NUMBER: 08224
TC 40501 I
PLANNER: DKF | DATE YR MO DA: 76 06 17 | BLDG.: 109 | SERVICING: MN330 | CUSTOMER: N420 | A/C CLASS – W/O NO. – PROJ.: 4 1337
SYS.: 3 | J.O.E.: 18540 | ACT: 0545 | TSK: 21 | LOC: 0022 | PRIOR: 02 | TYPE OF WORK: REP | WORK CATEGORY: Q, U, T | PROCEDURE NUMBER
TC 40502 I
ESTIMATED CRAFT HOURS
ELEC. | INST. | MILW.: 0,00,16,0 | FITR. | OPER.: 0,0,0,0,2,0 | OTHER
ESTIMATED COST: LABOR | MATERIAL | AUTHORIZED TOTAL COST | W.O. PREFIX | VERIFICATION METHOD | LOCATION: 109 TURB. BAY
TC 40503 I
JOB TITLE (40 CHARACTERS): Repair Oil Leaks on #5 Drive Turbine
DESCRIPTION: The Low Pressure Bearing and the governor both leak oil on #5 Drive Turbine. The low pressure bearing leaks on the west horizontal Flange. The governor leaks at its lower horizontal flange.
REQUESTED BY (12 CHARACTERS): D R Smith | PHONE NO.: 87 | DATE REQUIRED YR MO DA: 760801 | SUPV | DATE COMPLETED: 760811 | SHIFT: 4-12
APPROVED BY: J N Parke | PAS: RLS CMC | W.A. RECORDED INITIAL: JNP | DATE: 8-16-76
BC-4300-294.1 (7-74)

Figure B.1. Example of a Work Authorization from the N Reactor in 1976

To identify any correlation between elapsed time and failure severity, failures were categorized as catastrophic, degraded, or incipient based on the severities defined in *The Pump Handbook* (Krutzsch and Cooper 1976). An incipient failure is characterized by a component that performs within its capability but exhibits problems that could escalate into a degraded failure. A degraded failure is a component that operates at a less than specified performance level, whereas a catastrophic failure is a completely inoperable component (Krutzsch and Cooper 1976). This distinction is important because of the potential for correlation between incipient, degraded and catastrophic trends. If, for example, high incipient and degraded failure rates caused increased catastrophic failure rates in subsequent years, the correlation of component failure severities may also be useful information for ERMs.

B.2.5 Analysis and Results

Once separated based on component, failure mode and severity, the failures were ordered chronologically by year and plotted against time. The results (see Figure B.2) show that the number of failures per year for each component was generally cyclic, rising and falling periodically as time progressed. For HCRs, the number of failures trended downward, with time-related and demand-related failures reaching maxima in 1974 and 1980, respectively, before approaching zero in 1987 Figure B.3. A similar trend was observed in the number of time-related and demand-related damper failures (Figure B.4) which reached maxima in 1976 and 1980 respectively, and then decreased gradually to their minima in the early 1980s. Likewise, both the demand-related and time-related drive turbine failures reached maxima in the mid-1970s before decreasing sharply from 1976 to 1980 (Figure B.5). In all three of these cases, the demand and time-related failures rose and fell in tandem, following the same downward trend. In contrast, the number of demand-related CRW pump failures increased slightly and the number of time-related failures decreased gradually from 1973-1985 (Figure B.6). More drastically, the number of time-related exhaust fan failures trended significantly upward from 1 failure in 1974 to 8 in 1982, while the number of demand failures was moderate, ranging from 0 to 1 failures across the same period (Figure B.7). Lastly, the number of time-related battery failures also rose and fell cyclically from 1973-1984, but as shown by the trend in Figure B.3, the overall trend was upward, reaching an absolute minimum in 1974 and an absolute maximum in 1982 (Figure B.8).

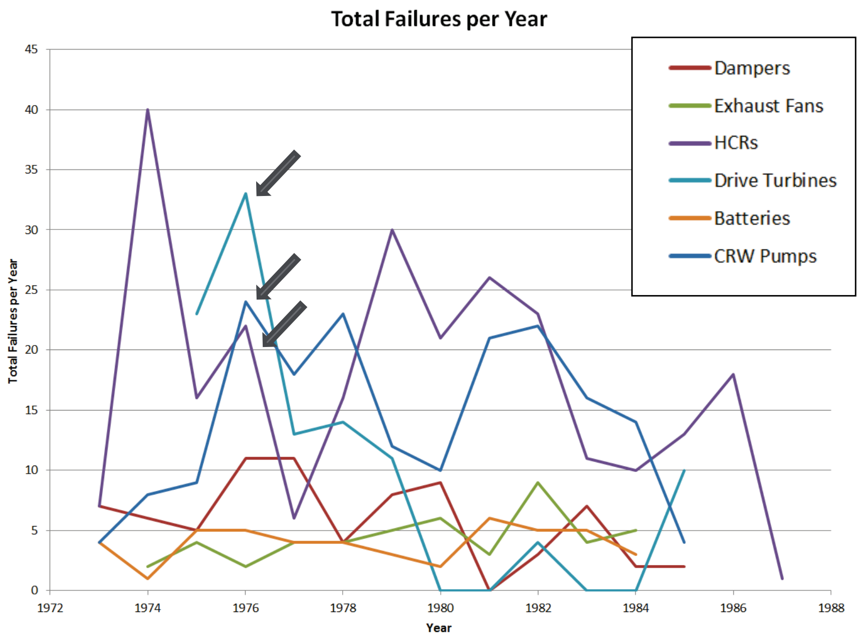

Figure B.2. Trend Analysis of the Total Number of Failures per Year for the Six Components, with Special Attention Directed to the Significant Reduction in Failure Rates from 1976 to 1977

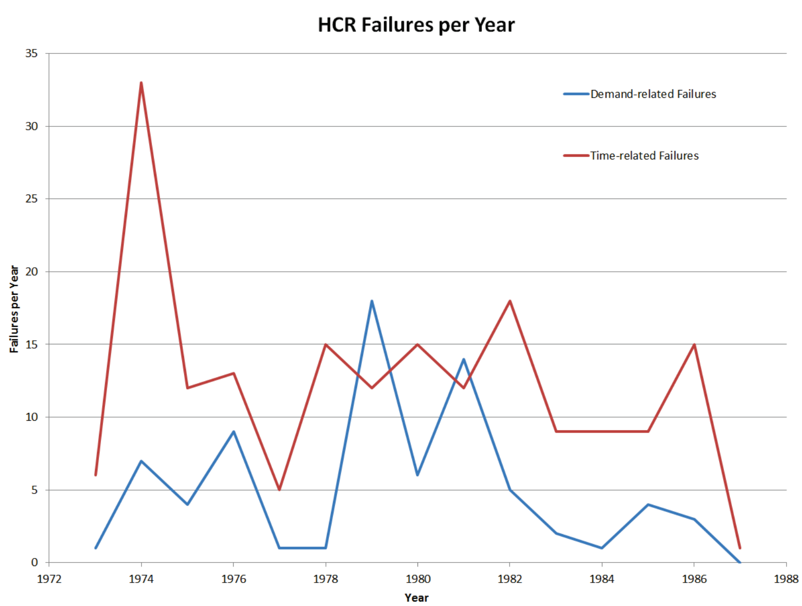

Figure B.3. Demand-Related and Time-Related Failure Trends for HCR

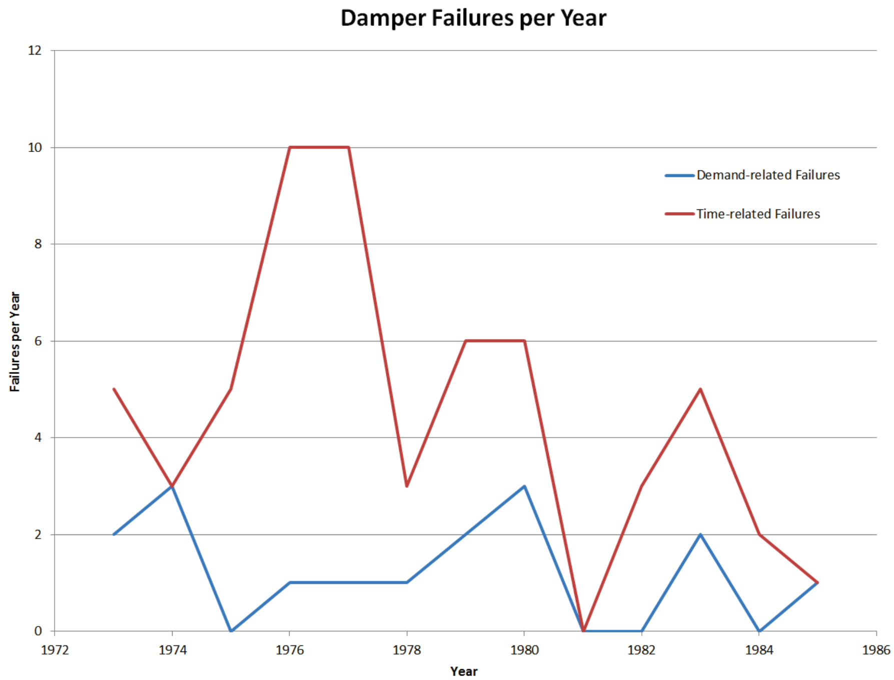

Figure B.4. Demand- versus Time-Related Failures, 1972-1986

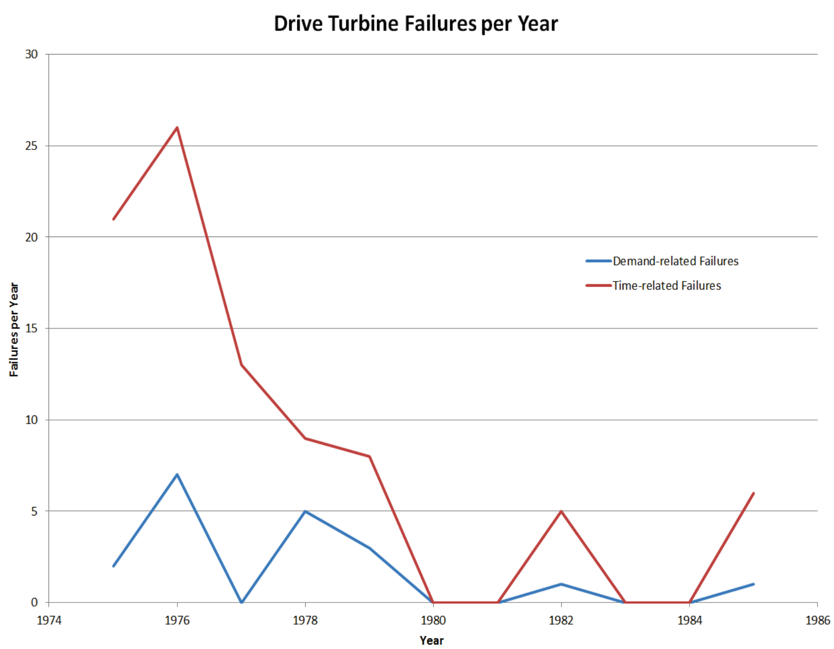

Figure B.5. Drive Turbine Failures per Year

Though the work authorizations were categorized and trended based on failure severity, no correlation was found between catastrophic, degraded or incipient failures. It was predicted that degraded failures led to catastrophic failures, which would be manifested graphically by a peak in degraded failure rates followed immediately by an increase in catastrophic failure rates. However, the catastrophic, degraded and incipient failures generally trended together, showing no evidence that a high number of lower severity failures led to an increase in progressively more severe failures.

In order to better compare each component's failures, the population of each component type was used to calculate a yearly failure rate for each component, according to the equation below:

$$Yearly\ Failure\ Rate\ per\ Component = \frac{Number\ of\ Failures\ per\ Year}{Component\ Population}$$

The population is important to consider because of the variability across components. For example, the Hanford N Reactor had only six drive turbines but over 100 HCRs. As such, comparing six components that failed 108 times to 107 components that failed 260 times over the same period is misleading. Using the component populations in Table B.1, the yearly failure rates for each component were calculated and plotted against time for each component. The result was an interesting comparison of the failure rates per year per component. As shown by the chart in Figure B.8, all components except the drive turbines ranged from zero failures per component to 1.5 failures per component, with the CRW pumps consistently having slightly higher failure rates than the other components. The drive turbines were a clear anomaly as the purple line shows, reaching a maximum of 5.5 failures per turbine in 1976, then decreasing sharply to 2.33 failures per turbine in 1978 and bottoming out at zero failures per turbine in 1980. This trend provoked questions about the unusually high failure rate in 1976 and the distinctly negative trend of the drive turbine failures.

To better identify the inflection points of the drive turbine trend, the work authorizations for drive turbines were grouped by month. This more clearly isolated the time of the unusually high failure rate in 1976. As shown by the absolute maximum of the bar graph in Figure B.9, the drive turbines failed six (6) times in June 1976. This was preceded by a gradual buildup of failures between August 1975 and June 1976. Following a rapid decrease in failures between June 1976 and October 1977, the graph then peaked at four (4) failures in December 1977. With no failures between February 1978 and July 1978, the number of failures maxed out again at five (5) failures in January 1979. The number of failures then bottomed out almost immediately. Grouping the work authorizations in this way generated a clearer timeline of the drive turbine failure history and provoked questions about the maintenance conducted in June 1976 and January 1979 and the rapid decrease in failure rates following these periods.

Because of the high fiscal cost of reactor shutdowns, a possible correlation between the peaks of the monthly failure rate trend and reactor scrams was of interest and therefore investigated. Using archived outage reports that logged the dates of every reactor scram from 1973-1985, the monthly component failure trends were compared to the dates of reactor shutdowns. Only four (4) of the 723 work authorizations surveyed shared the same date as a reactor scram. Based on the available data, no correlation between the maxima of the failure rate trends and reactor shutdowns was identified.

In attempt to explain the cause of the high drive turbine failure rates, the failures were further subdivided into the individual six drive turbines (DT 1-6) and timelines of each turbine's failures were reconstructed from the work authorizations. Each drive turbine's failures were trended separately to determine if one particular turbine was consistently faulty and resultantly pulled up the failure rate for all turbines. As shown in Figure B.10, although DT4 and DT5 had slightly higher failure rates over the 15-year period, all six turbines trended roughly together, reaching maximums in either 1976 or 1977 and all settling to zero by 1980. Thus, the results showed that the majority of drive turbine failures were caused by DT4 and DT5, with the other four turbines trending in roughly the same direction. Using the "Description of Job" section of the work authorizations, timelines of when failures occurred for each drive turbine were reconstructed, such shown in the example for DT 5 presented in Figure B.11. According to these failure sequences, repeated oil leaks and governor control failures contributed to the high failure rates in 1975 and 1976, which were repaired during a 1976 summer outage. In 1977, the number of failures fell 60%

from 33 failures in 1976 to 13 in 1977. These results suggest that the restoration of deteriorating components one year caused a reduction in failure rates the following year.

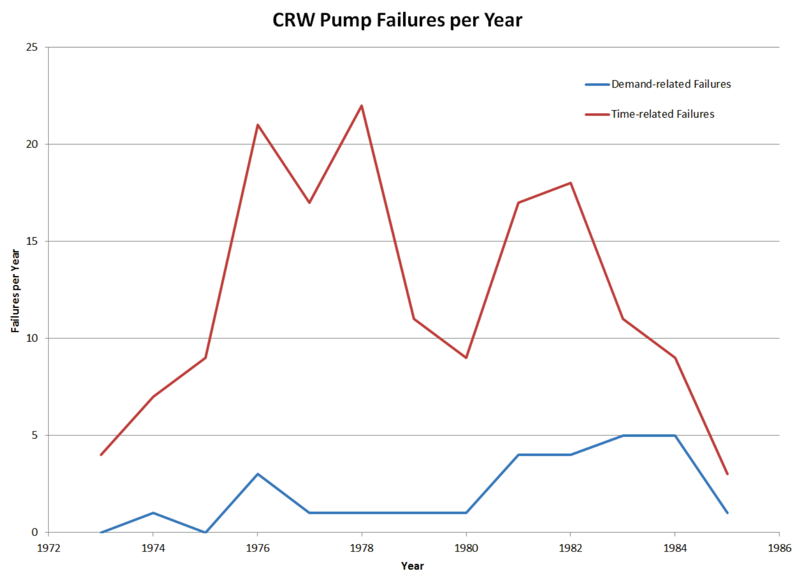

Figure B.6. CRW Pump Failure per Year

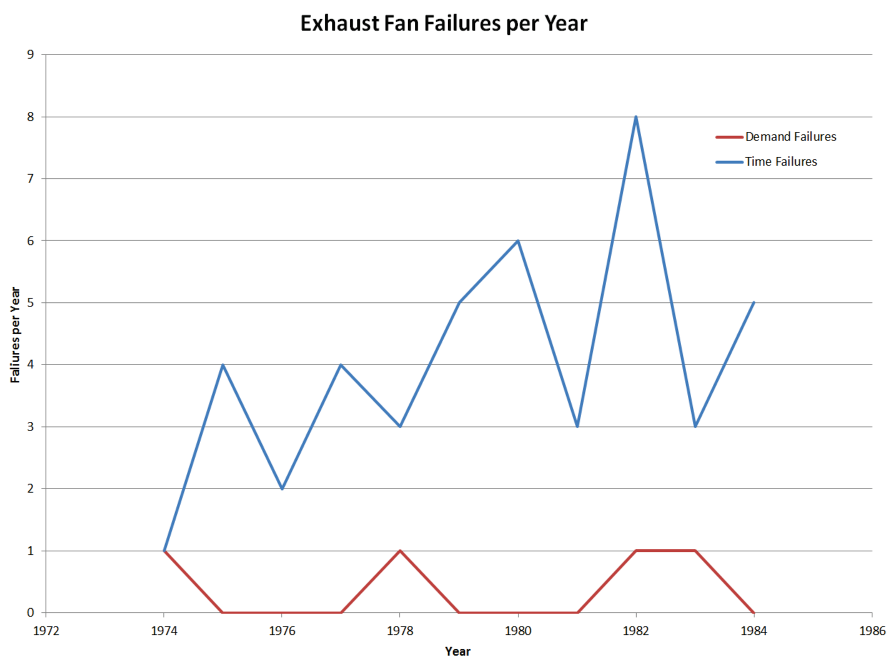

Figure B.7. Exhaust Fan Failures per Year

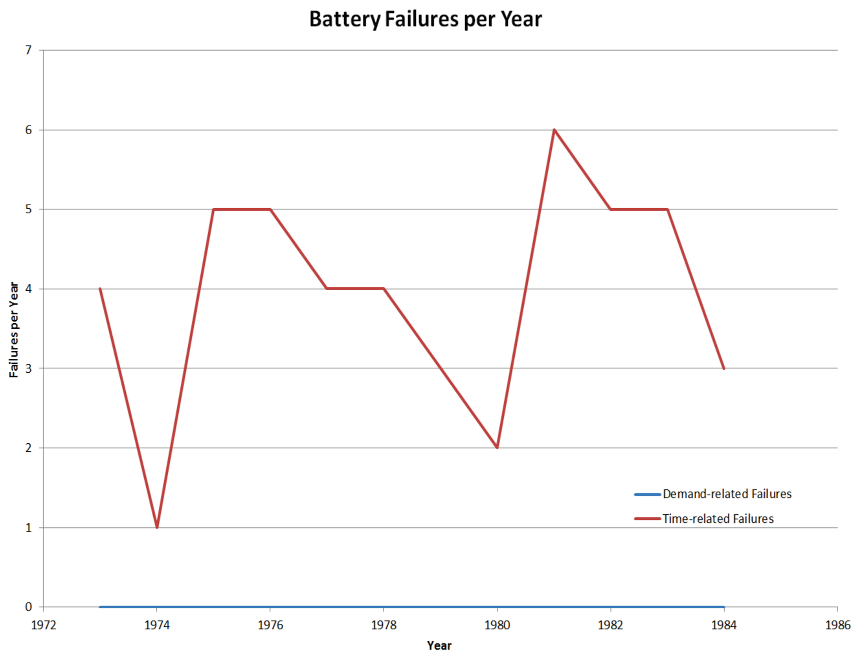

Figure B.8. Battery Failures per Year

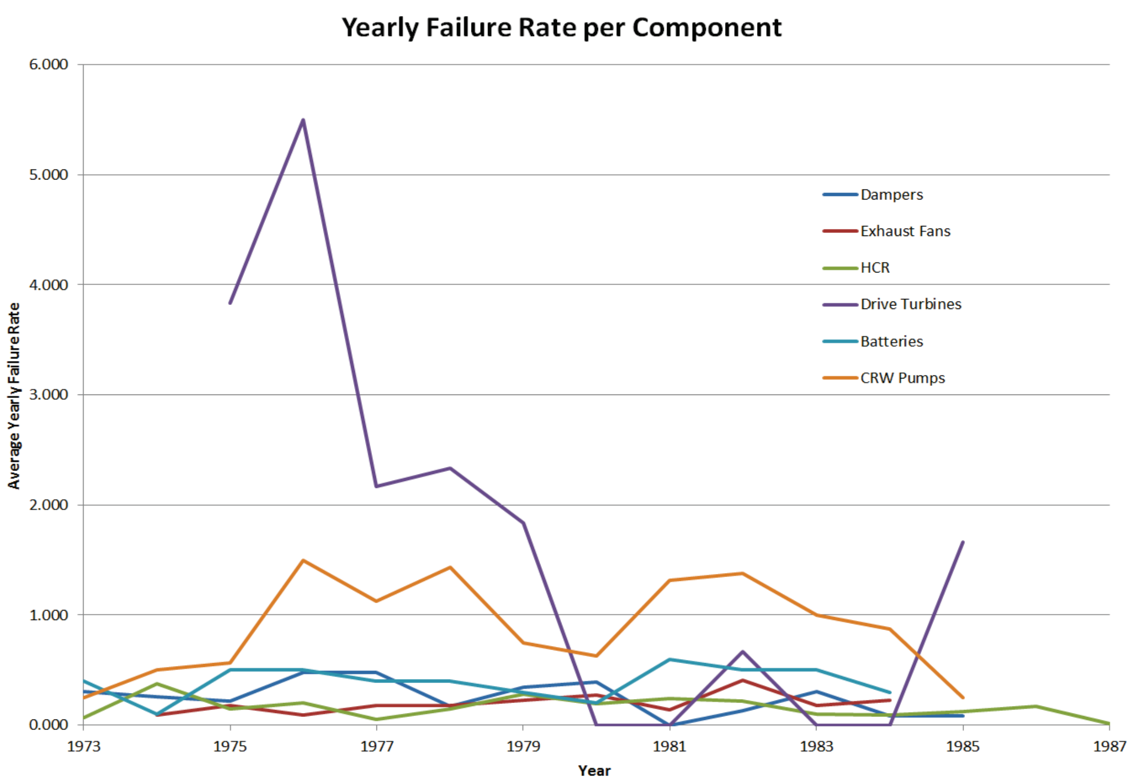

Figure B.9. Yearly Failure Rate per Component

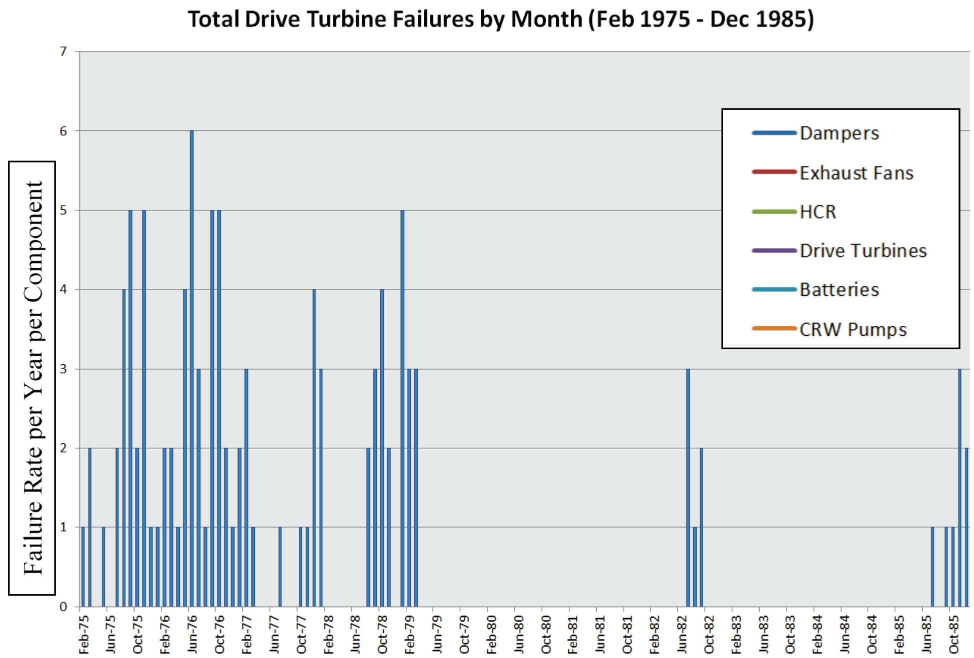

Figure B.10. Drive Turbine Failures by Month

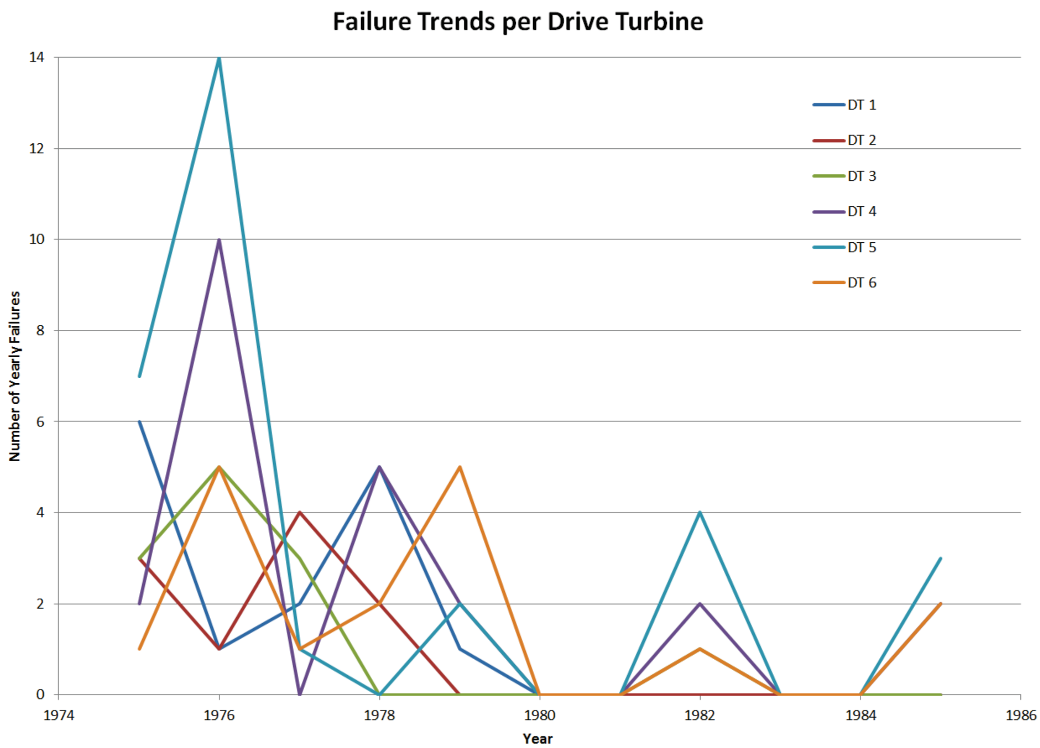

Figure B.11. Failure Trends for Each Drive Turbine (DT 1-6)

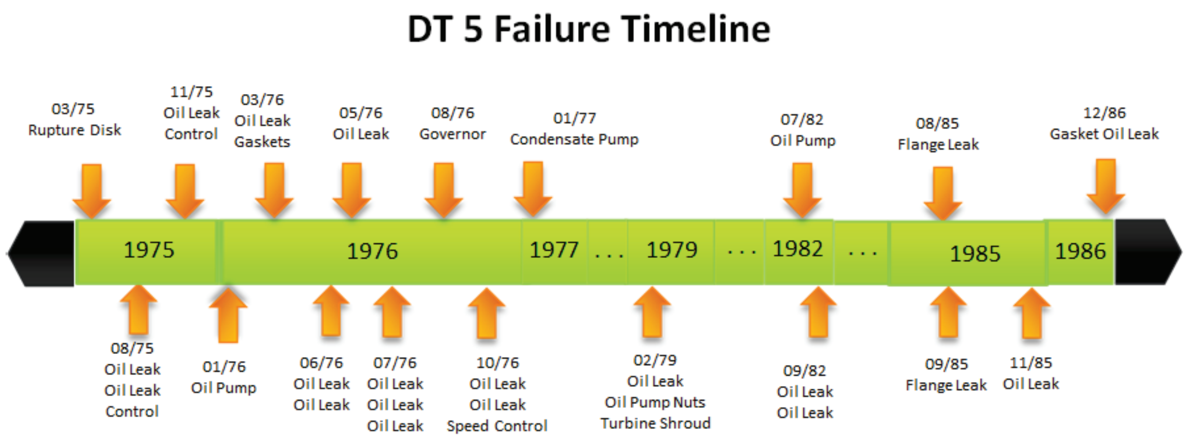

Figure B.12. Example Failure Timeline from Drive Turbine 5

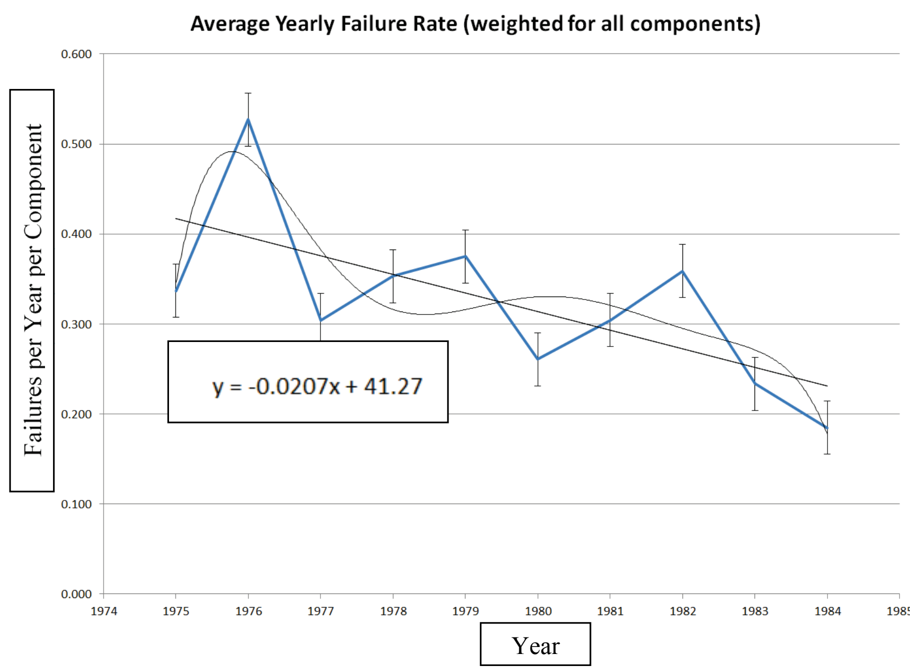

Figure B.13. Trend Analysis of the Weighted Average Failure Rate for All Components

Notably, a similar decrease in failures was observed between 1976 and 1977 in other components as well. As denoted by the arrows in Figure B.12, CRW pumps, HCRs, and drive turbines all saw substantial decreases in the number of failures from 1976 to 1977. A possible explanation for this trend is that N Reactor had a summer outage in 1976 during which maintenance was conducted on the circulating raw water pumps, drive turbine governor controls and oil leaks were repaired, and HCR hydraulic leaks were corrected. Subsequently, each of these component types saw significant reductions in failure rates the following year. Like the drive turbines, CRW pump failures fell 25% from 24 failures in 1976 to 18 in 1977 and HCR failures fell 72% from 22 failures in 1976 to 6 in 1977. This suggests that the maintenance and repairs made during the 1976 outage contributed to the decreased failure rate of drive turbines, HCRs, and CRW pumps in 1977.

B.2.6 Conclusions

Considering the component trends more holistically, the trend analysis clearly showed that failure rates are: cyclic, rising and falling from year to year. Using regression techniques, this was mathematically proven by the fact that high-order polynomials with multiple inflection points were the best estimates of the component failure trends, as demonstrated by high R-squared values. For example, when fit with a sixth order polynomial, the drive turbine time-related failures trend had an R-squared value of 0.9307, which is relatively close to its ideal value of one (1). This shows that component failure rates follow a cyclic pattern of maxima and minima, which are best modeled using a regression with curvature and inflection points. This conclusion is an addition to the 1990 PRA (Zentner et al. 1990), which only modeled the trends linearly, and is important for informing the ERM by showing an important time-related trend.

Despite consistent observations of cyclic failure trends, the overall direction of the component failure rate trends varied. When fitted with linear trend-lines, the drive turbines, HCRs and dampers all showed decreasing failure rates over time, as shown by the negative slope of their respective linear regressions. Contrastingly CRW pumps, exhaust fans, and batteries all showed increasing failure rates over time, as shown by the positive slopes of their linear trend-lines. Holistically, failure rates for all 168 components analyzed decreased over time, as shown in Figure B.13 by the negative slope of the linear trend-line for the weighted average failure rate curve. This curve represents the number of failures per year per component for all six component types analyzed in this study. The slope of the trend-line was -0.0274 with a 95% confidence interval of -0.0484 to -0.0063. Therefore, this study suggests that for this set of components, the trend-line of the overall weighted average is, in fact negative and that collectively, component failure rates decreased over time. However, because three component failure rates trended positively and the other three trended negatively, this overall trend is secondary to the discovery that failure rates have a cyclic nature.

Conclusions about two separate one-off trend analyses revealed that for this data set, there was no correlation between reactor shutdowns and the maxima of the monthly component failure trend. Also, no correlation was found between the trends of incipient, degraded and catastrophic failures.

In summary, the trend analyses conducted for these six component types revealed the cyclic nature of component failure rates, which are shown to reach multiple maxima and minima across a period of time. Given that three of the components analyzed showed increasing failure rates and the other three showed decreasing failure rates over the 15-year period, further study of additional components would be necessary to make generalizations about component failure rates increasing or decreasing over time. However, the discovery that failure rates cyclically increase and decrease from year-to-year is useful in informing the ERM about how components age. These trends not only demonstrate the time-dependence of component failure rates, but also show on a smaller scale that maintenance periods such as the summer outage of 1976 affect component failure rates the following year. This component aging information lends insight to how component condition monitoring might best be integrated into an ERM.

Appendix C

FFTF Component Reliability Effort

Appendix C

FFTF Component Reliability Effort

An initial methodology for enhanced risk monitors (ERMs) is described in the main document that integrates real-time information about equipment condition and probability of failure into risk monitors to provide an assessment of dynamic risk as plant equipment ages. An important aspect of ERM is the inclusion of uncertainty within the ERM framework. Several sources of uncertainty exist when estimating the probability of failure, including uncertainty regarding the specific condition of the component, uncertainty in the probability of failure, and uncertainty in the time-to-failure. One way to address these sources of ERM uncertainty is through evaluation of real plant data.

The Fast Flux Test Facility (FFTF) was the most recent liquid metal reactor (LMR) to operate in the United States. The FFTF was located on the U.S. Government's Department of Energy (DOE) Hanford Site near Richland, Washington, and was operated successfully from 1982 to 1992. Safe, reliable, and economic operation of the FFTF was achieved through administrative controls, technical specifications, and operating procedures, even with a demanding test schedule as a liquid metal irradiation test reactor. The high level of operating efficiency of FFTF is a potential source of vital data on the performance of liquid sodium as a safe and efficient heat transport medium that confirms the reliability of many of its large-scale components. The ten years of successful operation of the FFTF provided a very useful framework that could potentially be used for determining the reliability of LMR technology components. A potential advantage of raw data sources like FFTF is the ability to track component reliability over time. FFTF sources of reliability data are being compiled and evaluated for applicability. Efforts to recover FFTF data useful for verifying ERM methodology have focused on locating the FFTF input to the Component Reliability Data Organization (CREDO) database records.

Processed CREDO component failure rate information has been identified as a source of information for developing the simplified ERM framework AdvSMR PRA model. A subset of several hundred significant events collected and categorized during a preliminary FFTF PRA effort has been recovered and is being evaluated for component reliability information. Such component reliability data is being evaluated as a way to validate the proposed methodology for ERM.

C.1 Background on FFTF

Conceptual design of the FFTF began in 1965, followed by a period of construction and acceptance testing that ended with first cycle operations in 1982. FFTF operations extended for a decade until it was shut down in 1992. FFTF was the most instrumented reactor in the world and had an excellent data monitoring and acquisition system. DOE investment in the design and operation of FFTF easily exceeds $10B.

The plan to build FFTF began to take shape in 1967 when the Hanford site at Richland, Washington, was chosen as the home of the first large-scale liquid metal test reactor. The plan was culminated on April 30, 1982, with dedication of the FFTF. In 1970, the DOE selected Westinghouse Hanford Company, a wholly owned subsidiary of the Westinghouse Electric Corporation, to manage the design, construction, and operation of the FFTF as part of the Hanford Engineering Development Laboratory. The Advanced Reactors Division of the Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, was the reactor designer; and Bechtel Power Corporation, San Francisco, California, was the architect engineer and construction manager. In addition, more than 300 companies across the nation provided components, materials, and fuel for the FFTF.

The primary mission of the FFTF was to test full-size nuclear fuels and components typical of those to be found in a commercial liquid metal reactor. To accomplish this mission, the DOE established two fundamental objectives. First, the reactor plant technology would support the liquid metal reactor industry by developing fuel assemblies, control rods, and other core components whose lifespans could be proven to be economical in commercial power-generating applications. Second, the reliability of the FFTF would be proven by matching or exceeding the operational performance of commercial light water plants. Safe, reliable, and economic operation of the FFTF was achieved through administrative controls, technical specifications, and operating procedures. The high level of operating efficiency of FFTF provided vital data on the performance of liquid sodium as a safe and efficient heat transport medium and confirmed the reliability of many of its large-scale components.

The FFTF plant was an 86,103 sq. ft. complex of buildings and equipment arranged around a reactor containment building. The reactor was located in a shielded cell in the center of the containment building. Heat was removed from the reactor by liquid sodium circulating under low pressure through three primary coolant loops. (This is in contrast to conventional reactor plants that use water circulated under high pressure.) An intermediate heat exchanger separated radioactive sodium in the primary system from nonradioactive sodium in the secondary system. Three secondary sodium loops transported reactor heat from the intermediate heat exchangers to the air-cooled tubes of the twelve dump heat exchangers. Instrumentation and control equipment provided monitoring and automatic control of the reactor and heat removal facilities; automatic reactor shutdown (SCRAM) if preset limits are exceeded; and computerized collection, handling, retrieval, and processing of operating and test data. Onsite utilities and services included emergency generation of electrical power, heating and ventilation, radiation monitoring, fire protection, and auxiliary cooling systems for plant equipment and components. The FFTF was the only U.S. liquid metal reactor built and maintained to American Society of Mechanical Engineers codes. Complementary standards were also developed for safety, testing, and quality assurance issues involved in liquid metal reactor technology. Facilities were included for receiving, conditioning, storing, and installing core components and test assemblies as well as examining and packaging for offsite shipment and radioactive waste disposal.

A picture of the FFTF plant and its location at the Hanford site in Washington State is shown in Figure C.1. Figure C.2 provides a diagram of the FFTF reactor plant and key parameters are listed in Table C.1. A cutaway of the reactor is shown in Figure C.3. Schematics of the primary and secondary coolant systems are shown in Figure C.4. Because it was designed as a flexible test reactor, the FFTF did not have steam generators but included dump heat exchangers. It was designed to provide a prototypic test bed with respect to temperature, neutron flux level, and gamma ray spectra for fast reactor fuels and materials testing. The FFTF was designed as the most extensively instrumented fast spectrum test reactor in the world, with proximity instrumentation of temperature and flow rate for each core component as well as contact instrumentation and gas and electrical connections for special test positions. Figure C.5 shows an FFTF instrumented test assembly.

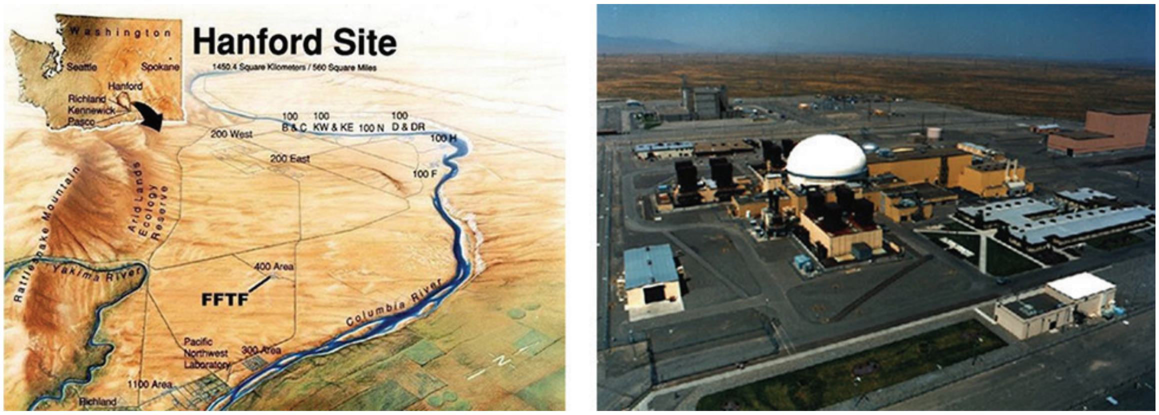

Figure C.1. FFTF at the Hanford Site

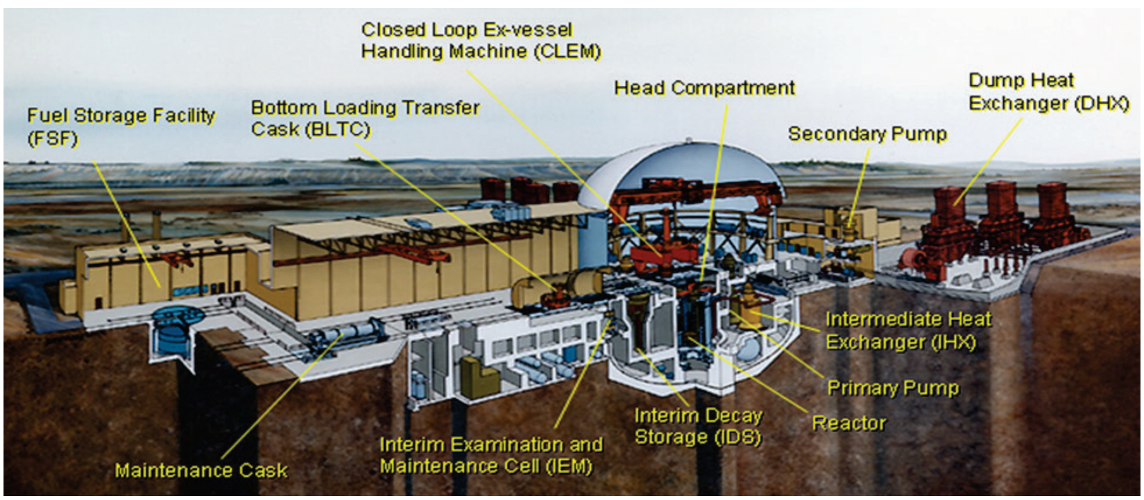

Figure C.2. FFTF Reactor Plant

Table C.1. FFTF Parameters

Parameter	Value
Thermal Power	400 MW
Coolant	Sodium
Coolant Inlet/Outlet Temperatures	360/526 C
Coolant Loops	3
Driver Fuel Material	$(Pu\text{-}U)O_2$
Enrichment Zones	2
Core Height	91.4 cm
Core Diameter	120 cm
In core Driver, Test Locations	82
Instrumented Through Head	8
Piping Length	64 km
Wiring Length	300 km
Instruments and Sensors	>20,000

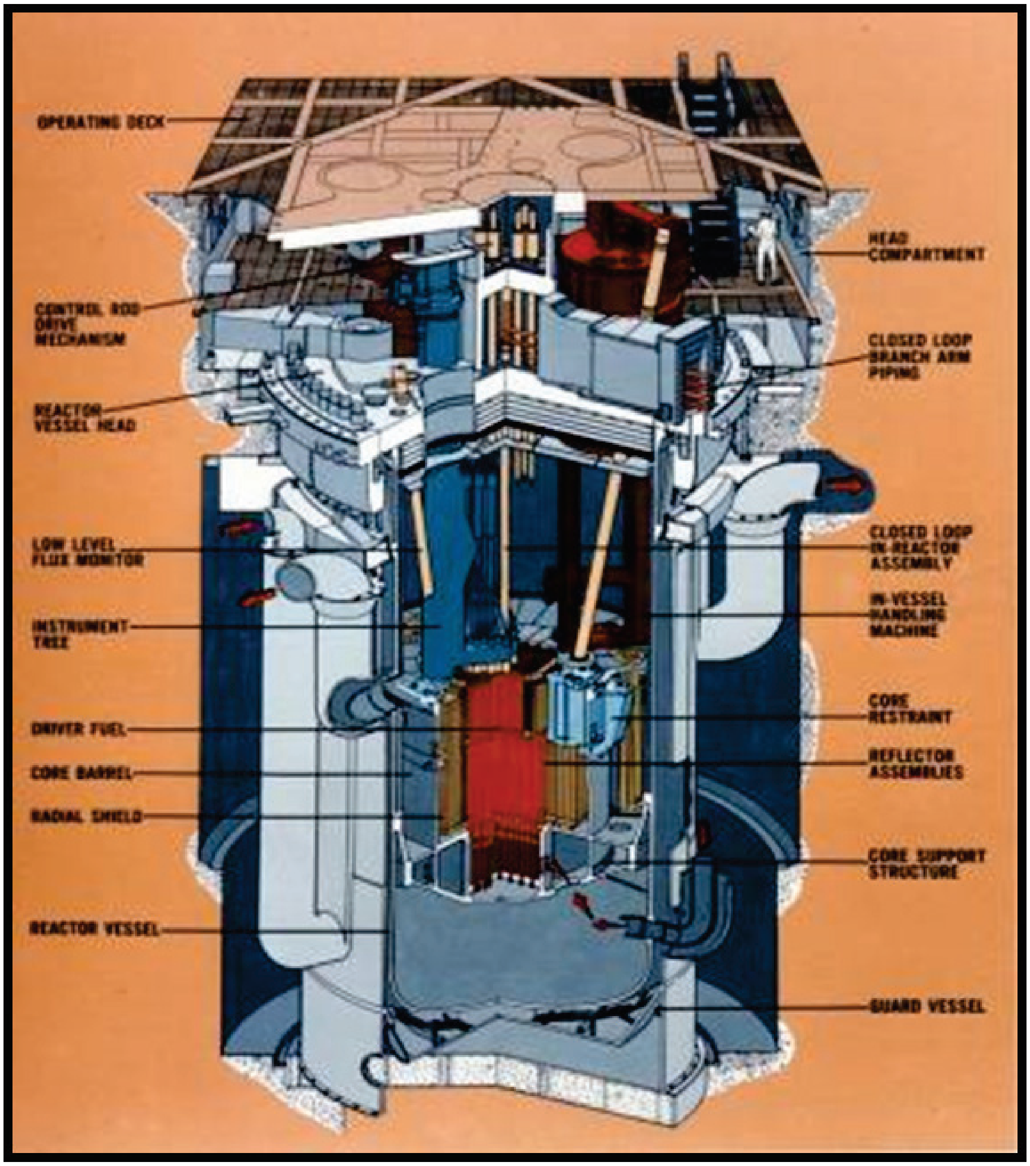

Figure C.3. FFTF Reactor

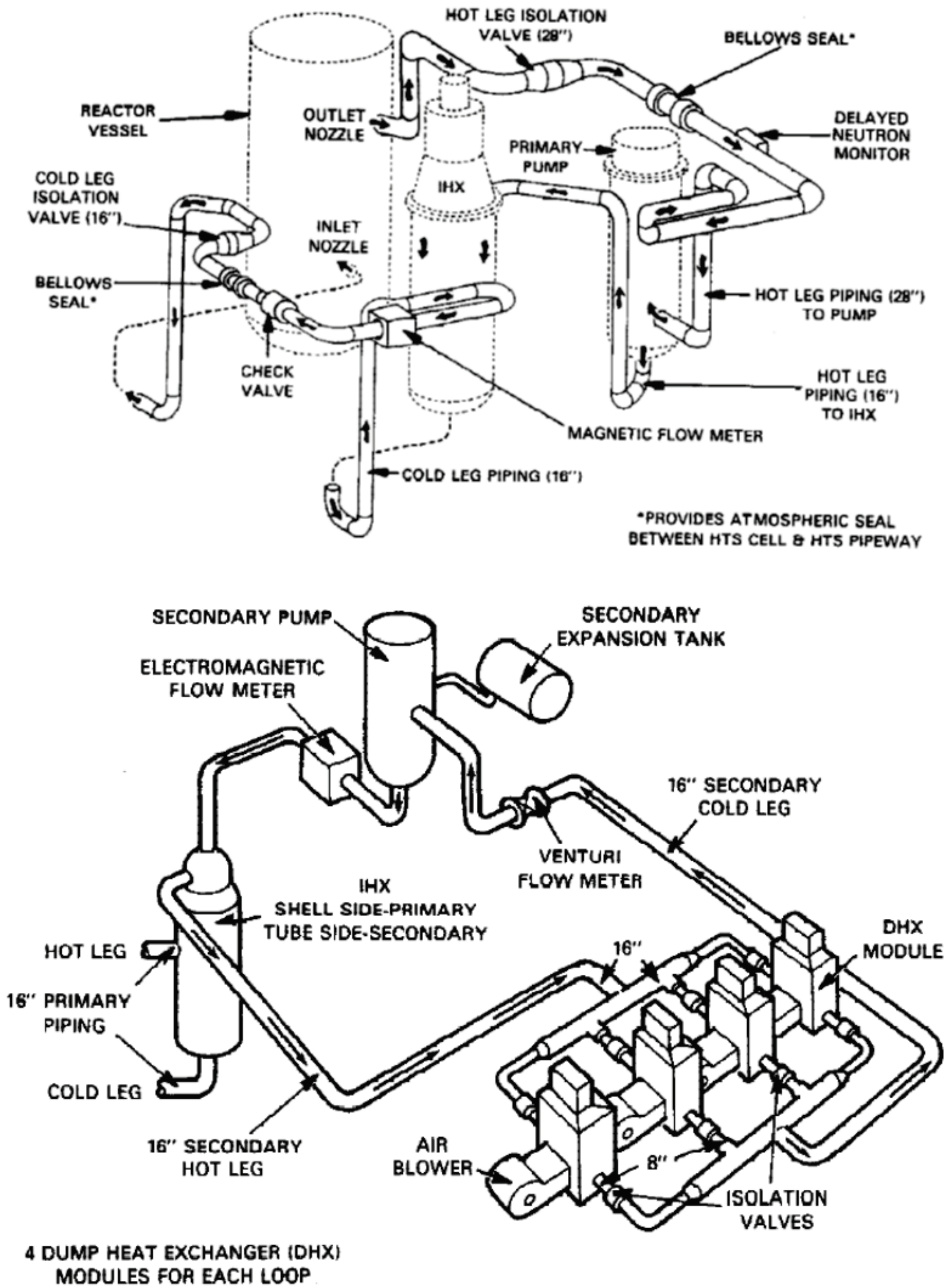

Figure C.4. FFTF Primary and Secondary Loop Schematics

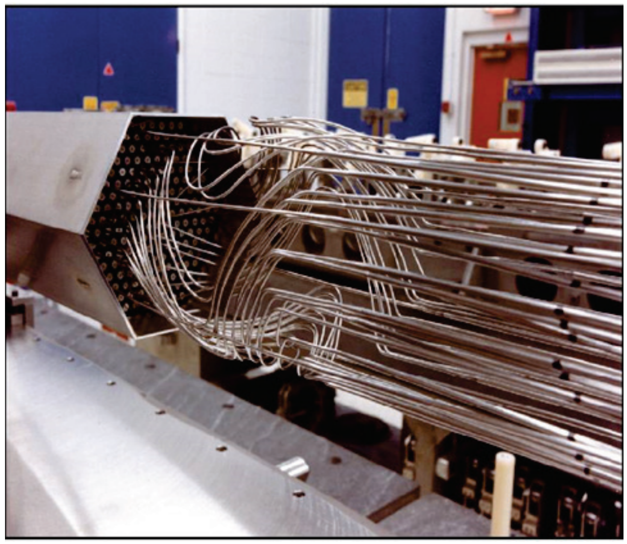

Figure C.5. Instrumented FFTF Test

Figure C.6 shows a timeline from the beginning of conceptual design to the first operating cycle. It is notable that during the 1960s and 1970s a substantial effort was expended in the development and testing of liquid metal reactor components. Figure C.7 shows the major activities during the twelve cycles of reactor operation. Safe, reliable, and economic operation of the FFTF was achieved through administrative controls, technical specifications, and operating procedures even with a demanding test schedule as a liquid metal irradiation test reactor. The high level of operating efficiency of FFTF provided vital data on the performance of liquid sodium as a safe and efficient heat transport medium and confirmed the reliability of many of its large-scale components.

FFTF was the most instrumented reactor in the world, with proximity instrumentation of temperature and flow rate for each core component as well as contact instrumentation and gas and electrical connections for special test positions. Detailed plant data acquired during operations and testing, such as assembly outlet temperatures and flow rates, coolant system temperatures and flow rates, and reactor vessel temperatures, were recorded on magnetic tapes by the plant data acquisition systems at frequencies up to once per second. During the years of operation, the FFTF plant data system systematically recorded over 1300 instrument variables. FFTF data measurement features include:

- Primary and secondary loop hot and cold leg temperatures and flow rates, neutron detectors, pump speed indicators
- Thermocouples with a response time of minutes were used to monitor assembly outlet temperatures for each core location
- Fast response thermocouples for measuring assembly outlet temperatures with a response time of seconds were used for two core locations during selected tests
- Two fuel tests included high response wire wrap thermocouples on fuel pins and were used during tests at startup
- The plant data system recorded >1300 variables at 0.1–60 second intervals.

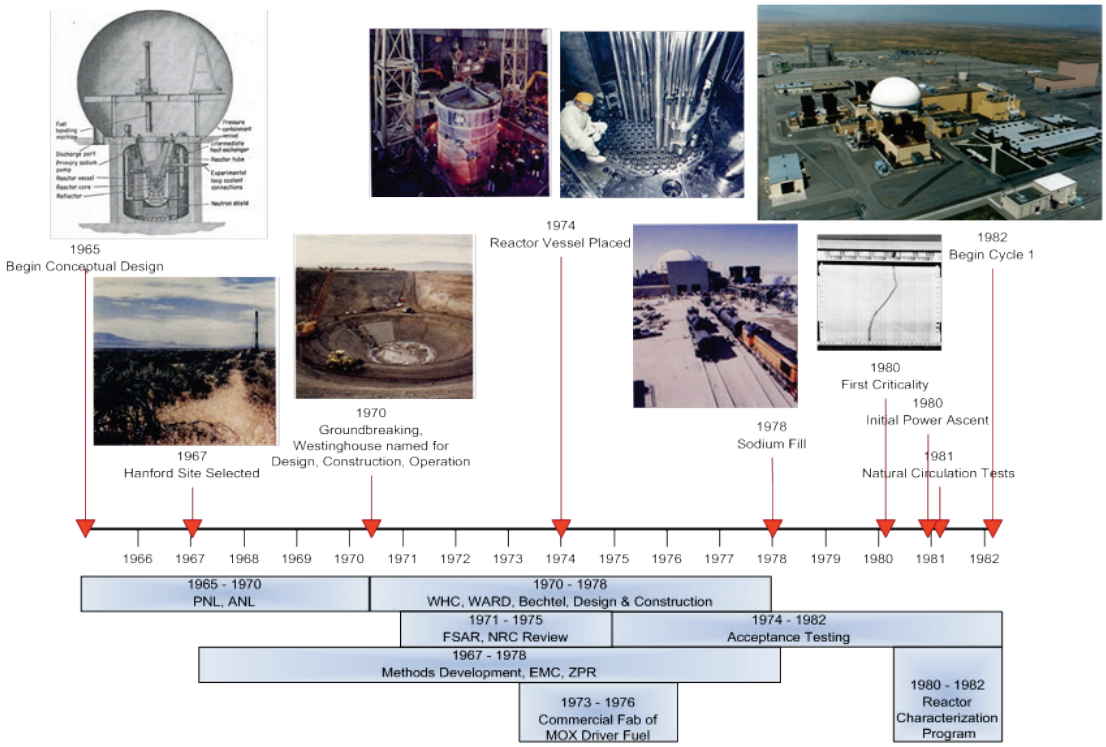

Figure C.6. FFTF History Prior to First Operating Cycle

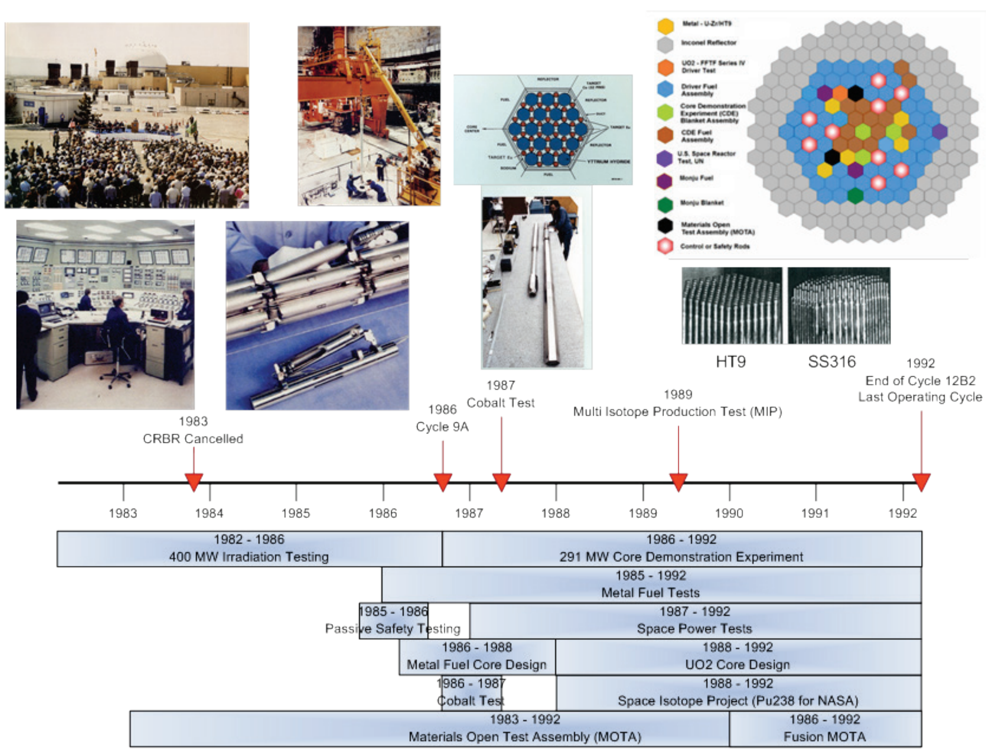

Figure C.7. FFTF Operating History

Documentation of the rigorous and successful design, construction, testing, and operational experience at FFTF was thorough and immense, with official records routinely archived. Efforts are currently directed at locating, extracting, and processing FFTF records of potential relevance to AdvSMR enhanced risk monitoring. Engineering knowledge from the design, construction, and operation of FFTF and other fast reactors represents a huge investment. Tapping this knowledge base is potentially worth billions of dollars, and at any valuation, will contribute to advanced fuel cycle designs. However, the FFTF information will not be useful if it is not accessible in a form that is useful and can be interpreted correctly. In order to ensure the FFTF information is useful, it is important to capture the tacit knowledge surrounding the documents and data. This tacit knowledge goes beyond what is printed on the pages of documents and includes the understanding of how the documents and data relate to one another historically, programmatically, and technically. Understanding of the context is important in navigating the collection of documents and data, recognizing the importance of specific data. Such tacit knowledge is not reproducible from electronic scans and knowledge must be captured from actual experts involved at the time.

C.2 FFTF Contribution to Enhanced Risk Monitoring

C.2.1 FFTF Data Potentially Relevant to AdvSMR Enhanced Risk Monitoring

FFTF data that is of potential use in developing enhanced risk monitoring is shown in Table C.2. The information has been separated into design, operations, and safety categories. Design information includes fabrication and procurement specifications, system design descriptions, and as-built drawings that can be used to pinpoint specific details on components such as valves, breakers, instrumentation, etc. The QA program specifies the controlled parameters for acceptance and testing of components. Operations data includes recorded sensor data, CREDO event reports, logs/records, and scheduled/unscheduled maintenance. The FFTF Job Control System (JCS) contains records of all work done at the plant, which would include maintenance and repair of components. Cycle operating and outage reports include descriptions of important activities and also list unusual occurrences during each cycle or outage. Safety data includes the safety analyses assumptions in the FSAR and from interactions with the NRC prior to operation. It also includes information that was gathered for the incomplete FFTF PRA effort.

C.2.2 CREDO

In 1977 the DOE established a CREDO at Oak Ridge National Laboratory (ORNL) to provide a centralized computer-based source of information on the reliability of components utilized in advanced liquid metal cooled reactors. The data were collected from operating reactors (EBR-II, FFTF, Joyo) and liquid metal loop test facilities and entered into the CREDO database on the ORNL mainframe until the program was terminated in 1992. During the ten years of FFTF operation, data forms were compiled into reports on FFTF events that were transmitted to CREDO. FFTF prepared and transmitted hundreds of CREDO Event Data Reporting Forms to ORNL over life of plant. Transmittal letters from FFTF were entered into records but attachments were typically not included. The CREDO database was only maintained at ORNL and was only available by access through ORNL. FFTF did not have a copy of the CREDO database. Currently no records of the CREDO database can be found.

Table C.2. Relevant FFTF Data

Mode	Type
Design	Fabrication specifications Procurement specifications Technical specifications Quality Assurance Program System Design Descriptions As-built drawings
Operations	Plant Sensor data CREDO data event reports Operational logs/records Maintenance/JCS database
Safety	FSAR approach NRC interactions Partial PRA/CAFTA input

Specific actions in progress at PNNL related to CREDO include:

- The few CREDO transmittals from FFTF that included CREDO forms are being collected.
- FFTF plant operations letterbooks are being searched for because they might contain the CREDO transmittals.
- FFTF plant Quality Assurance (QA) Vault records are being searched for CREDO files, because CREDO reporting was a function of the FFTF QA organization.
- A draft report, *Handbook of Component Reliability*, was located that contains various measures of component reliability and failure information for 13 component classifications from the CREDO database. This report includes the number of events by type, and overall failure rate, but no time frequency information. The 13 components were cold and vapor traps, electric heaters, filters/strainers, heat exchangers, logic gates, mechanical pumps, motors, non-nuclear sensors, pipes and fittings, pressure vessels and tanks, signal modifiers, support and shock devices, and valves.

Such processed CREDO component failure rate information are being examined for utilization in the simplified ERM framework AdvSMR PRA model described in earlier in this report.

C.2.3 FFTF Event Descriptions Relevant to Component Reliability

During the ten years of FFTF operation, hundreds, maybe thousands, of events were recorded by FFTF operations and filed for every abnormal event that occurred. Efforts to locate a complete set of event fact sheets continue. Several records holding boxes containing FFTF operations files on occurrence reports with folders of histories of actions and resolutions related to the events have been located and are being examined for relevant component reliability information such as time frequency information for specific components and systems. The FFTF JCS contains records of all work done at the plant. Access to the FFTF JCS continues to be pursued. Once access is obtained, the intent is to search the JCS records for useful information.

During the late 1980s an effort was underway to prepare a PRA for FFTF. Part of that effort was to develop component failure rates by reviewing descriptions of events for that type of information. The FFTF PRA effort was terminated before it was complete, but resulted in over 200 event descriptions for significant events between 1980 and 1989 that were categorized into 18 internal event initiators, 6 internal leak locations, and external events for potential use in the preliminary FFTF PRA effort. This subset of event descriptions was retrieved and entered into a spreadsheet so that it could be searched for component reliability information. An example listing of a few of the events is shown in Table C.3. The FFTF PRA working files and system notebooks have also been located. These system notebooks and FFTF PRA information on specific components/systems are being used to guide the ERM PRA modeling. FFTF CAFTA working PRA input files were located on 5¼ inch floppy disks, but preliminary evaluation is that these files would be of little use in updating ERM methodology.

C.3 FFTF Summary

The ten years of successful operation of the FFTF provided a very useful framework that could potentially be used for determining the reliability of LMR technology components. Such component reliability data may be of increased importance to new designs after the events at Fukushima. Efforts to recover FFTF data useful for verifying ERM methodology have had limited success. FFTF CREDO database records have not been located. A subset of several hundred significant events collected and categorized during the preliminary FFTF PRA effort has been recovered. Efforts to extract component reliability information continue.

Table C.3. Example Listing of Preliminary FFTF PRA Events

Event Fact Sheet Number	Additional Documentation	Date	Nature of Problem	Location	Component	Cause	Explanation
80-003	HEDL 80-016	6/16/1980	Spurious Plant Protection System Trip	Control Room	Ratchet Puller Hoist	Maintenance Error	A ratchet puller hoist gave way, dropped detector, causing PPS shutdown signal
80-012		6/22/1980	Loss of Electrical Power	Control Room, RSS Panel C13DP	PPS System	Electrical Error	Ground located in PPS System during performance of SC-12-9
80-014		6/23/1980	Pump Failure	P-5 Pump Tower, cell	psi pressure controllers	Maintenance / Design Error	Supply reservoir went to 5 psi & cocked seal on secondary pump P-5 seal housing, causing oil to leak into lower seal leakage reservoir
80-015		6/24/1980	Inadvertent Sodium Leak	DHX - West	E-15 HV-43342	Electrical Error	Unexplained sodium flow from DHX E-15 drain valve HV-43342
80-018		6/25/1980	Cover Gas Pressure Transient	Control Room / Reactor Services Bldg	RAPS cold box	Operator / Design Error	RAPS cold box back pressurized due to reduced discharge path from CAPS maintenance
80-019		6/25/1980	Thermal Transient	DHX-East modules No. 2 and 4	pony motor gear box	Maintenance Error	Oil leakage from gear box sight glass led to high outlet temperature differential
80-023		7/4/1980	Inadvertent Valve Operation	Secondary Loop 1 drain piping	UV-43144	Electrical / Operator Error	Inadvertent opening of Secondary Loop 1 cold leg fill/drain valve resulting in transfer of Na to secondary drain header
80-024		7/5/1980	Pump Failure	Cell 556	P-52	Electrical Error	Overheating & improper heat up of P-52 Primary Sodium Sampling Pump
80-025		7/6/1980	Pump Failure	P-6 Pump Tower, Cell 461/435	Pump P-6 lower seal	Mechanical Error	P-6 lower seal cocked during routine shutdown of main motor; cause unknown
80-026		7/7/1980	Loss of Fire Protection System	C-1356, Zone 1	E-85 & E-86	Operator Error	Flow valves to detectors for E-85 & E-86 were isolated & hoses removed

Appendix D

Development of a Prototypical Advanced Reactor Model to Support the Enhanced Risk Monitor

Appendix D

Development of a Prototypical Advanced Reactor Model to Support the Enhanced Risk Monitor

D.1 Introduction

The development of advanced reactors faces significant technique hurdles to commercialization due to the unique features and characteristics inherent to their designs. The features may include new materials of construction, employment of modular fabrication techniques, and unique safety and instrumentation and control issues related to the potential multi-modular operation. These features, along with the lack of operating experience for many advanced reactor coolants and component designs, will challenge our ability to accurately characterize the evolving risk of operating advanced reactors. Current online risk monitors provide a point-in-time estimate of the system risk given the current plant configuration (e.g., equipment availability, operational regime, environmental conditions). However, these risk monitors do not account for plant-specific normal, abnormal, and deteriorating states of active components and systems. Incorporation of unit-specific estimates of the probability of failure (POF) of key components into dynamic probabilistic risk assessment (PRA) has the potential to enable real-time decisions about stress relief and to support effective maintenance planning while ensuring investment protection. Such enhanced risk monitors (ERMs) are expected to improve the safety, economy, and availability of advanced reactors (Coble et al. 2013b).

In order to demonstrate the efficacy of the Enhanced Risk Monitor (ERM) for evaluating the economic risks of advanced reactors, a Simulink-based model of a prototypical advanced reactor (PAR) has been developed at the University of Tennessee. ERMs will support the economic goals of advanced reactors by providing a tool for optimizing operations and maintenance activities. Asset optimization through ERMs will improve economics of advanced reactors by:

- Maximizing generation by assessing the potential impact of taking key components offline for testing or maintenance;
- Supporting reduced operations and maintenance staff by aiding in optimization of O&M planning; and
- Supporting potential remote siting by providing early warning of potential increases in plant risk.

The PAR model explicitly models the major reactor systems and numerically simulates the degradation of key components. The effect of component degradation on the overall power production of the PAR is simulated for integration in the ERM. This report summarizes the development of the PAR, including simulation of all major components, reactor control schemes, and component degradation models. Initial simulation results are presented for degradation of pumps in the primary and intermediate loops.

D.2 Prototypical Advanced Reactor

Work on the ERM thus far has focused on liquid metal reactors to demonstrate the efficacy of the approach, although the framework is generally applicable to any advanced reactor design. The notional PAR design is shown in Figure D.1 (Coble et al. 2013b). This power block features two reactor cores, each connected to a dedicated intermediate heat exchanger (IHX) and steam generator. The output of these two steam generators is then connected to a common balance of plant (BOP). BOP includes steam drums, turbine, condenser, feed-water pumps, and feed-water heaters. The key components identified in this power block that require physical models include: reactor core, IHX, steam generator, and BOP.

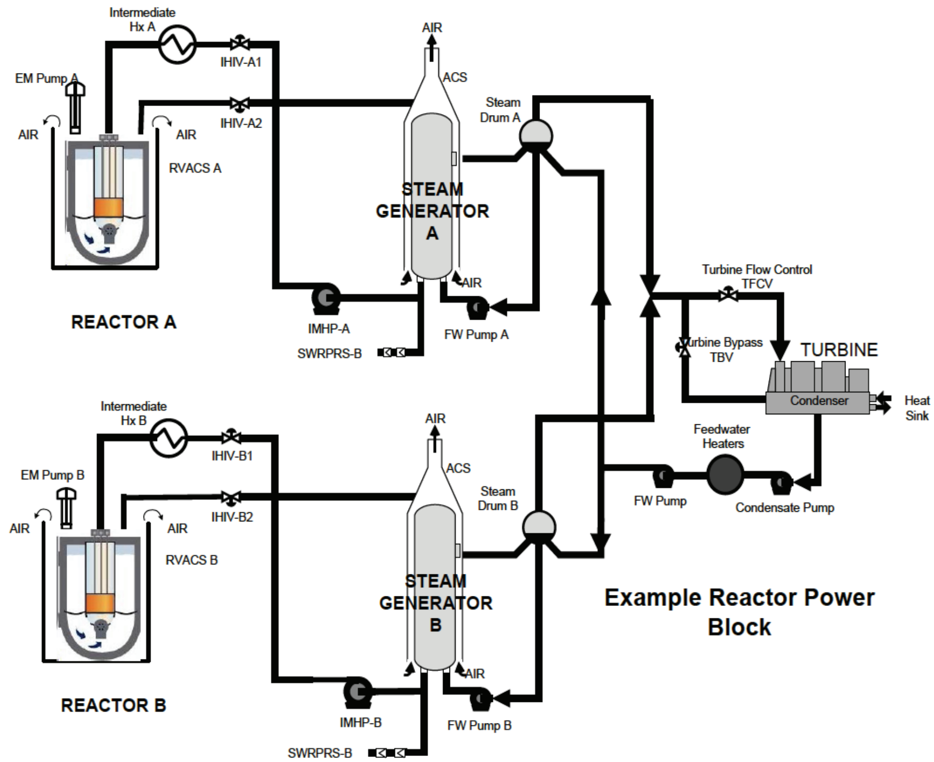

Figure D.1. Prototypical AdvSMR Power Block (Coble et al. 2013b)

The initial PRA model for the PAR includes the following components (Ramuhalli et al. 2013):

- Electromagnetic pumps
- RVACS
- Emergency diesel
- Steam generator (tube rupture)
- Liquid metal sodium pressure relief system
- Isolation valve
- Feed-water pump
- Steam generator louvers
- Intermediate sodium pump
- Condensate pump
- IHX tube rupture
- Turbine bypass valve

Modeling the degradation and failure of these components will be necessary to demonstrate the efficacy of the ERM. No physics-based models are planned for these components. Instead, the effects of evolving degradation and failure of key components on the overall system performance will be numerically modeled. For instance, a faulty valve may have slower response to control actions or may have a limited range of operation; a degraded pump may give reduced flow; and a fouled IHX may have reduced heat transfer coefficients. For this demonstration of the ERM, degradation of the primary and intermediate sodium pumps has been the primary degradation mode of interest.

The following subsections describe the modeling of major components and systems: primary system and IHX, steam generator, and BOP. The control strategy for the reactor power block is described. Finally, the numeric pump degradation model is described.

D.2.1.1 Primary system Modeling

The reactor core and IHX models are based on the Experimental Breeder Reactor (EBR)-II. EBR-II was a pool-type sodium-cooled fast reactor (SFR). EBR-II featured 62.5 MWt with 20 MWe output. The prototypical AdvSMR has two independent EBR-II cores connected to a common BOP, giving a total of 40-MWe output for the power block. Existing perturbation models of EBR-II core and IHX provide a starting point for modeling (Berkan and Upadhyaya 1988). These perturbation models are linearized at 100% nominal power. Nonlinear equations were derived from these models in order to support simulation of normal transient operation from 30 to 100% full power. This model does not support accident scenarios.

Description of EBR-II Primary System

The Experimental Breeder Reactor-II is a liquid-metal fast breeder reactor designed as an engineering test facility. The EBR-II tests and demonstrations play an important role in the development of future advanced reactors.

The nodal representation of the primary system, including the core, reflectors, plenum, and IHX, is given in Figure D.2 (Berkan and Upadhyaya 1988). The core model is a 25 node model, which includes the active core, inner and outer blankets, lower and upper reflectors, and piping. Core bowing and control rod expansion reactivity effects are neglected in the EBR-II models. The core bowing reactivity effect may become significant at high temperatures as the structural material inside the vessel expands radially. The thermal bowing cannot be handled without increasing the order of the model by introducing radial lumps. The contribution of thermal bowing to reactivity is small compared to other feedback effects. At steady state conditions, the control rod expansion reactivity effects are assumed to be negligible. For core heat transfer using Mann's model, the lower coolant lump outlet temperature is assumed to represent the average lump temperature, which is a coupling parameter for the heat transfer driving force between the metal and the coolant region. The assumptions for the piping region include: constant coolant density, no axial heat conduction, and no heat gain or loss in piping.

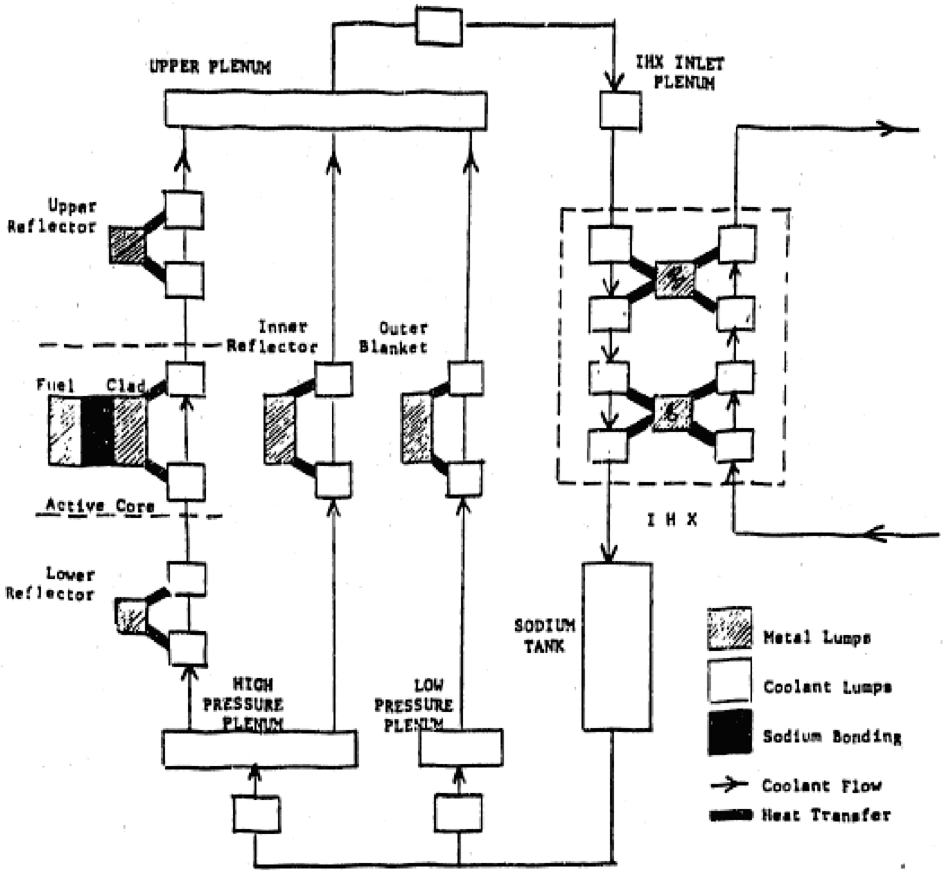

Figure D.2. Node Representation of EBR-II Primary System (Berkan and Upadhyaya 1988)

The intermediate heat exchanger is modeled using twelve state variables: ten nodes in the IHX, indicated in the dashed box in Figure D.2; the IHX inlet plenum on the primary side; and the sodium tank at the primary side outlet. The core and IHX models are coupled into a single model. The governing equations for each subsystem and definition of variables are presented in the following subsections. The values for model parameters are given in Berkan and Upadhyaya (1988) and Berkan et al. (1990).

Primary System Equations

1. Nonlinear reactor kinetics

The active core dynamics are described by the point reactor kinetics equations. Reactivity feedback effects modeled include Doppler feedback, steel expansion, core sodium expansion, reflector expansion, inner reflector expansion, and fuel expansion. Six precursor groups are summarized with average decay constant, delayed neutron fraction, and mean lifetime.

$$\dot{P}_c = \frac{-\beta_T}{\Lambda} P_c + \frac{\rho P_c}{\Lambda} + \bar{\lambda} C$$

$$\rho = \rho_{external} + \rho_{feedback}$$

$$\rho_{feedback} = \sum_i \alpha_i (T_i - T_{io})$$

$$\dot{C}_c = \frac{\beta_T}{\Lambda} P_c - \bar{\lambda} C$$
$$C = C_c + C_o$$
$$C_o = \frac{\beta_T P_{co}}{\Lambda \bar{\lambda}}$$

where:
P_c = Fractional Core Power
β_T = Total Delay Neutron Fraction
Λ = Mean Neutron Generation Time
ρ = Reactivity
$\bar{\lambda}$ = Precursor Average Decay Constant
C = Precursor Concentration
α_i = Temperature Reactivity Feedback Corresponding to Temperature T_i
T_i = Current Temperature in Channel *i*
T_{io} = Steady State Temperature for Channel *i* at 100% power.

2. Core heat transfer

The heat transfer in active core region is modeled using five differential equations corresponding to five lumps as shown in Figure D.3 (Berkan and Upadhyaya 1988). These nodes represent the fuel, sodium bond, cladding, inlet coolant, and outlet coolant regions. The heat transfer dynamics between cladding and coolant regions is represented using Mann's model. In Mann's model, the lower coolant lump outlet temperature is assumed to present the average lump temperature, which is a coupling parameter for the heat transfer driving force between the metal and the coolant regions.

$$\dot{T}_F = \frac{P_f P_o}{(MC_p)_F} P_c - \frac{1}{R_1 (MC_p)_F} \quad (T_F - T_B)$$

$$\dot{T}_B = \frac{1}{R_1 (MC_p)_B} \quad (T_F - T_B) - \frac{1}{R_2 (MC_p)_B} \quad (T_B - T_C)$$

$$\dot{T}_C = \frac{1}{R_2 (MC_p)_B} \quad (T_B - T_C) - \frac{1}{R_3 (MC_p)_\theta} \quad (T_B - \theta_1)$$

$$\dot{\theta}_1 = \frac{1}{R_2 (MC_p)_\theta} \quad (T_C - \theta_1) + \frac{2}{\tau}(\gamma_2 - \theta_1)$$

$$\dot{\theta}_2 = \frac{1}{R_3 (MC_p)_\theta} \quad (T_C - \theta_1) + \frac{2}{\tau}(\theta_1 - \theta_2)$$

where:
T_F = fuel temperature
T_B = sodium-bond temperature
T_C = fuel cladding temperature
θ_i = temperature of the *i*th coolant node
R_1, R_2, R_3 = heat transfer resistances,
γ_2 = lower axial-reflector coolant outlet temperature
τ = resident time of the coolant in the active core region
P_F = fraction of the power deposited in the fuel
$(C_p)_F$ = specific heat capacity of the fuel
$(C_p)_B$ = specific heat capacity of the blanket material

$(C_p)_\theta$ = specific heat capacity of the coolant
M_F = mass of the fuel
M_B = mass of the blanket material
M_θ = mass of the coolant

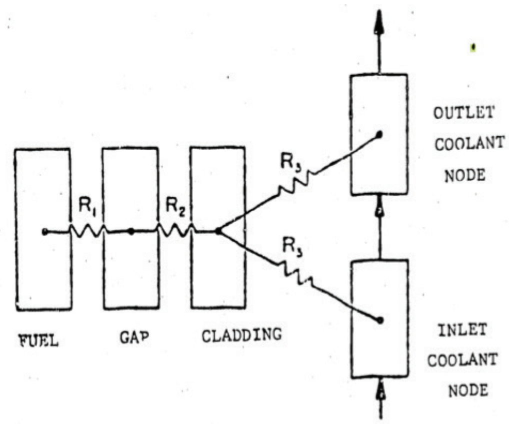

Figure D.3. Mann's Core Heat Transfer Model (Berkan and Upadhyaya 1988)

3. Reflector and blanket models

Reflectors and blankets surround the active core in the EBR-II primary system. The complete core model includes twelve additional nodes representing the axial and the radial reflector zones and radial blanket region. The same heat transfer principle is carried out in developing the state equations as in the core heat transfer model. The general equations for reflector and blanket regions are described by a set of three equations for each as shown below.

$$\dot{T}_M = \frac{P_i}{(MC_p)_M} P_c - \frac{U * A}{(MC_p)_M} (T_M - T_1)$$

$$\dot{T}_1 = \frac{U * A}{(MC_p)_T} (\delta T_M - \delta T_1) + \frac{2}{\tau}(\theta_{in} - T_1)$$

$$\dot{T}_2 = \frac{U * A}{(MC_p)_T} (\delta T_M - \delta T_1) + \frac{2}{\tau}(T_1 - T_2)$$

where:
T_M = temperature of the metal node
T_1 = temperature of the first region coolant node
T_2 = temperature of the second region coolant node
A = total heat transfer area
τ = residence time of the coolant in the reflector or the blanket region
U = metal to coolant heat transfer coefficient
θ_{in} = inlet coolant temperature
$(C_p)_M$ = specific heat capacity of the metal
$(C_p)_T$ = specific heat capacity of the coolant

4. Piping and plenum model

The model includes six lumps representing the low pressure plenums, the high pressure plenum, the upper plenum, and core inlet-outlet piping region. A first order transfer-lag has been assumed for all piping. The other assumptions are: (1) constant coolant density, (2) no axial heat conduction, and (3) no heat gain or loss in the piping.

$$\dot{T_U} = \frac{M_1 C_{p_2}}{(MC_p)_u}\gamma_4 + \frac{M_2 C_{p_2}}{(MC_p)_u}\gamma_6 + \frac{M_3 C_{p_3}}{(MC_p)_u}\gamma_8 - \left[\frac{M_1 C_{p_2}}{(MC_p)_u} + \frac{M_2 C_{p_2}}{(MC_p)_u} + \frac{M_3 C_{p_3}}{(MC_p)_u}\right] T_U$$

$$\dot{T_{out}} = \frac{1}{\tau_1} T_U - \frac{1}{\tau_1} T_{out}$$

$$\dot{T_{LI}} = \frac{1}{\tau_3}\theta_p - \frac{1}{\tau_3} T_{LI}$$

$$\dot{T_{HI}} = \frac{1}{\tau_4}\theta_p - \frac{1}{\tau_4} T_{HI}$$

$$\dot{T_H} = \frac{1}{\tau_5} T_{HI} - \frac{1}{\tau_5} T_H$$

$$\dot{T_L} = \frac{1}{\tau_6} T_{LI} - \frac{1}{\tau_6} T_L$$

where:
T_U = upper plenum temperature
T_{out} = reactor outlet temperature
θ_p = primary sodium tank temperature
T_{LI} = low-pressure plenum inlet temperature
T_{HI} = high-pressure plenum inlet temperature
T_H = high-pressure plenum temperature
γ_4 = upper reflector outlet temperature
γ_6 = inner reflector outlet temperature
γ_8 = blanket region outlet temperature
T_L = low-pressure plenum temperature
τ_1 = resident time of sodium in reactor outlet piping
τ_2 = resident time of sodium in the pot
τ_3 = resident time of sodium in the pot-to-reactor low-pressure piping
τ_4 = resident time of sodium in the pot-to-reactor high-pressure piping
τ_5 = resident time of sodium in the high-pressure plenum
τ_6 = resident time of sodium in the low-pressure

5. Intermediate Heat Transfer

The intermediate heat exchanger (IHX) is modeled by a 10-node lumped parameter approximation of a counter-flow heat exchanger; one half of the IHX is shown in Figure D.4 (Berkan and Upadhyaya 1988). The primary inlet plenum and the sodium tank are represented by first order transport-lag approximations. The heat transfer from the primary to the secondary sodium is modeled using Mann's technique. Primary and secondary nodes are numbered in the direction of flow.

$$\dot{P}_1 = \frac{2}{\tau_{HXP}} T_{HP} - \left(\frac{(UA)_P}{\left(MC_p\right)_P} + \frac{2}{\tau_{HXP}} \right) P_1 + \frac{(UA)_P}{\left(MC_p\right)_P} M_1$$

$$\dot{P}_2 = \left(\frac{2}{\tau_{HXP}} - \frac{(UA)_P}{\left(MC_p\right)_P} \right) P_1 - \frac{2}{\tau_{HXP}} P_2 + \frac{(UA)_P}{\left(MC_p\right)_P} M_1$$

$$\dot{M}_1 = \frac{(UA)_P}{\left(MC_p\right)_M} P_1 - \left(\frac{(UA)_P + (UA)_S}{\left(MC_p\right)_M} \right) M_1 + \frac{(UA)_S}{\left(MC_p\right)_M} S_3$$

$$\dot{S}_4 = \frac{(UA)_S}{\left(MC_p\right)_S} M_1 - \left(\frac{(UA)_S}{\left(MC_p\right)_S} - \frac{2}{\tau_{HXS}} \right) S_3 - \frac{2}{\tau_{HXS}} S_4$$

$$\dot{S}_3 = \frac{(UA)_S}{\left(MC_p\right)_S} M_1 - \left(\frac{(UA)_S}{\left(MC_p\right)_S} + \frac{2}{\tau_{HXS}} \right) S_3 + \frac{2}{\tau_{HXS}} S_2$$

$$\dot{P}_3 = \frac{2}{\tau_{\mathrm{HXP}}} P_2 - \left(\frac{(UA)_P}{\left(MC_p\right)_P} + \frac{2}{\tau_{\mathrm{XHP}}} \right) P_3 + \frac{(UA)_P}{\left(MC_p\right)_P} M_2$$

$$\dot{P}_4 = \left(\frac{2}{\tau_{\mathrm{HXP}}} - \frac{(UA)_P}{\left(MC_p\right)_P} \right) P_3 - \frac{2}{\tau_{\mathrm{XHP}}} P_4 + \frac{(UA)_P}{\left(MC_p\right)_P} M_2$$

$$\dot{M}_2 = \frac{(UA)_P}{\left(MC_p\right)_M} P_3 - \left(\frac{(UA)_P + (UA)_S}{\left(MC_p\right)_M} \right) M_2 + \frac{(UA)_S}{\left(MC_p\right)_M} S_1$$

$$\dot{S}_2 = \frac{(\mathrm{UA})_S}{\left(MC_p\right)_S} M_2 - \left(\frac{(UA)_S}{\left(MC_p\right)_S} - \frac{2}{\tau_{\mathrm{XHS}}} \right) S_1 - \frac{2}{\tau_{\mathrm{XHS}}} S_2$$

$$\dot{S}_1 = \frac{(UA)_S}{\left(MC_p\right)_S} M_2 - \left(\frac{(UA)_S}{\left(MC_p\right)_S} + \frac{2}{\tau_{\mathrm{HXS}}} \right) S_1 + \frac{2}{\tau_{\mathrm{HXS}}} S_{in}$$

$$\dot{P_{in}} = \frac{1}{\tau_7} T_{out} - \frac{1}{\tau_7} P_{in}$$

$$\dot{\theta}_P = \frac{1}{\tau_2} P_4 - \frac{1}{\tau_2} \theta_P$$

where:

P_1 = first primary node temperature
P_2 = second primary node temperature
M_1 = first (upper) tube wall temperature
S_4 = fourth secondary node temperature
S_3 = third secondary node temperature
P_3 = third primary node temperature

P_4 = fourth primary node temperature
M_2 = second (lower) tube wall temperature
S_2 = second secondary node temperature
S_1 = first secondary node temperature
P_{in} = primary inlet plenum temperature
T_{out} = reactor outlet temperature
S_{in} = secondary sodium inlet temperature
θ_p = sodium tank temperature
τ_{HXP} = resident time in primary nodes
τ_{HXS} = resident time in secondary nodes
τ_7 = resident time in primary outlet plenum

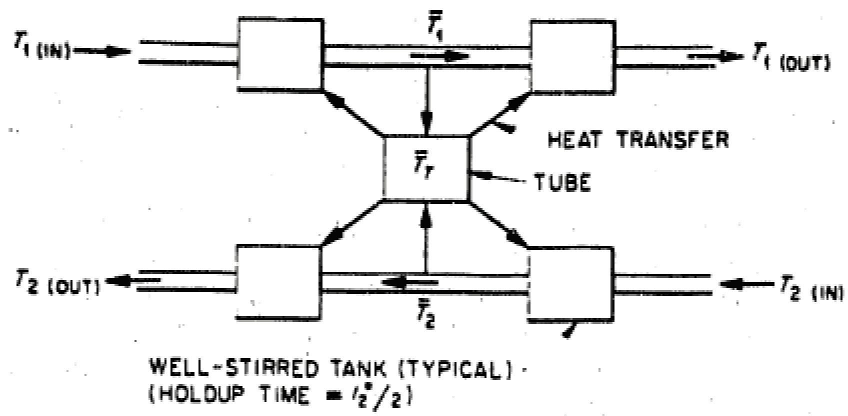

Figure D.4. Lumped-Parameter Approximation of a Counter-Flow Heat Exchanger (Berkan and Upadhyaya 1988)

6. Model validation

The response of the PAR model was compared to the perturbation model response reported in Berkan and Upadhyaya (1988) for both fractional core power and sodium tank temperature following a -5 cent reactivity insertion. Figure D.5 shows the reactor fractional power response to a -5 cent reactivity perturbation in the PAR model and the EBR-II model. The sodium tank temperature response of the PAR and EBR-II models is shown in Figure D.6. For the step reactivity perturbation of -5 cents, Figure D.6 indicates that the temperature response of the tank sodium settles down at about 2500s. This delayed temperature deviation will affect the core and reflector regions as the recycling sodium temperature reaches the tank temperature. The effect of the tank sodium temperature on the core power can be seen in Figure D.7. The time response of the primary system model is observed to be in three modes: the prompt jump (0 to 1s), the reactivity feedback settlement (1 to 200s), and delayed thermos-hydraulic effects (200 to 3000s). The results of the nonlinear PAR model match with the results of the original EBR-II perturbation model for this reactivity insertion.

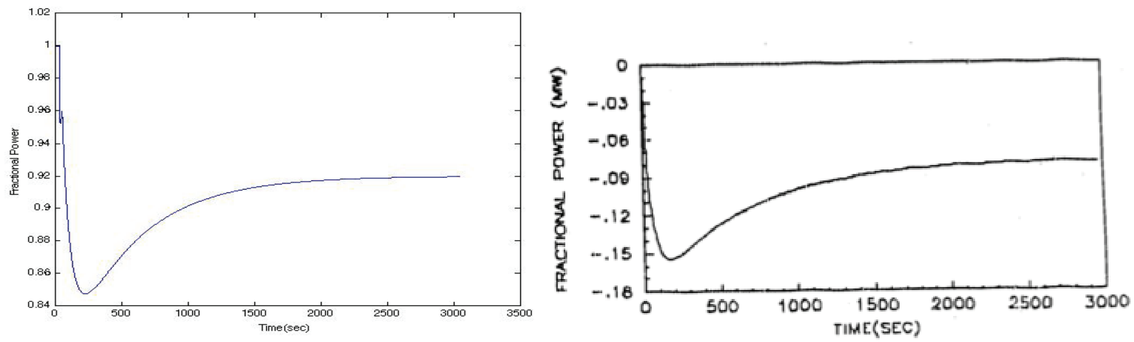

Figure D.5. Step Response of Fractional Reactor Power to a -5 Cents Reactivity Perturbation in (left) PAR Model and (right) EBR-II Model (Berkan and Upadhyaya 1988)

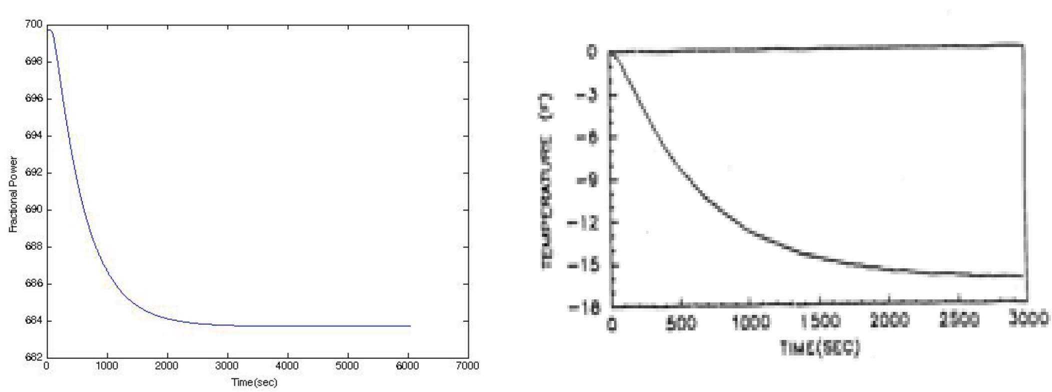

Figure D.6. Step Response of Sodium Tank Temperature to a -5 Cents Reactivity Perturbation in (Left) PAR Model and (Right) EBR-II Model (Berkan and Upadhyaya 1988)

Figure D.7 shows reactor fractional power response to different step reactivity insertions: -5, -10, and -15 cents. The model response follows expected behavior for these insertions, though no results were available for the EBR-II model for comparison for the larger reactivity insertions.

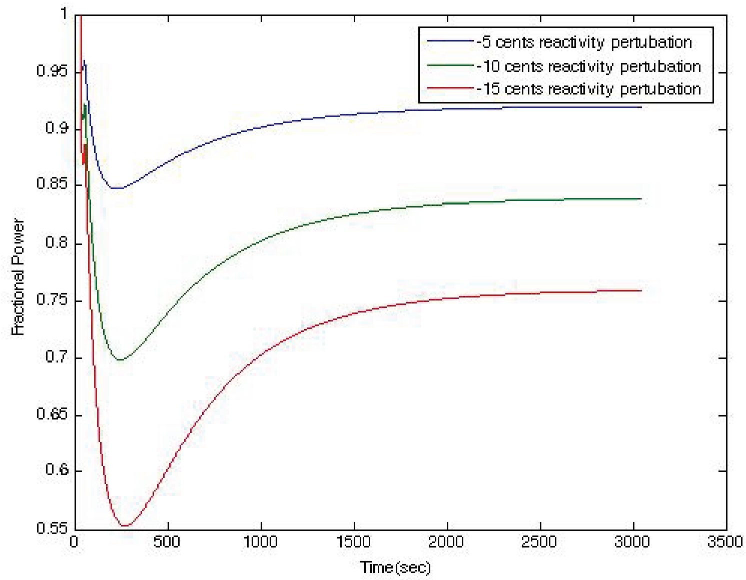

Figure D.7. Step Response of Fractional Reactor Power to Different Reactivity Perturbation

D.2.1.2 Steam Generator Modeling

Each reactor core/IHX model is connected to a dedicated steam generator. The EBR-II steam generator is a natural circulation system (Berkan and Upadhyaya 1988). The system is divided into thirteen lumps each representing average physical quantities. The nodal representation is shown in Figure D.8 (Berkan and Upadhyaya 1988**)**. The steam generator is represented by twenty differential equations using the state-space technique. The superheater model considers a single-phase heat transfer regime. The intermediate sodium flow is assumed to be constant. Three state variables of the superheater model are temperatures of the intermediate sodium, superheated steam, and the tube wall. The two remaining state variables are the control input and feedwater flow.

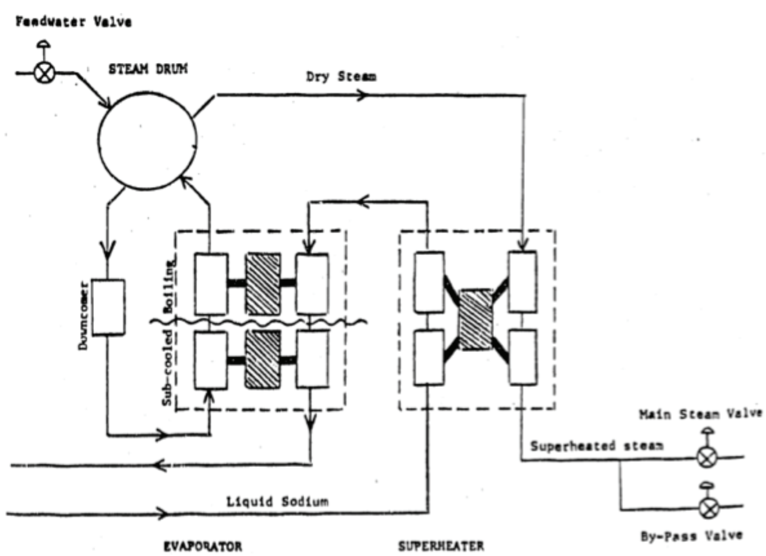

Figure D.8. Node Representation of Steam Generator (Berkan and Upadhyaya 1988)

1. Evaporator and Drum Balance Equations

On the evaporator side, the primary tube wall and the secondary lumps are divided with a moving boundary determined by the subcooled height. The system dynamics is a function of drum pressure and pressure inside the tubes of the evaporator. Thermodynamic properties are determined at these two pressures. The primary assumptions used in this model are:

- Phase equilibrium,
- No superheating in the boiling region,
- The separators are 100% effective, and
- Linear dependence between flow and enthalpy increase caused by the heat transfer into this region.

The evaporator side consists of thirteen state variables including the downcomer and drum water temperature, drum and boiling region pressures, drum inlet steam quality, subcooled level and drum level, primary sodium and tube wall temperatures, and two flows for the downcomer and rising mixture in the boiling region.

2. Steam Drum

$$\frac{d\delta T_{ld}}{dt} = -\frac{h_{ld}A_d\rho_{ld}}{C_P M_{ld}} * \frac{d\delta L}{dt} + \frac{W_{fw}}{M_{ld}} * \delta T_{fw} + \frac{h_{fw}}{C_P M_{ld}} * \delta W_{fw} - \frac{W_{dc}T_{dc}}{M_{ld}} - \frac{h_{dc}}{C_P M_{ld}} * \delta W_{dc} + \frac{(1-X_e)W_{rm}\frac{\partial h_f}{\partial P_B}}{C_P M_{ld}} * \delta P_B + \frac{(1-X_e)h_f}{C_P M_{ld}} * \delta W_{rm} - \frac{W_{rm}h_f}{C_P M_{ld}} * \delta X_e$$

$$\frac{d\delta P_D}{dt} = \frac{X_e}{V_{SD}\frac{\partial \rho_{ST}}{\partial P_D}} * \delta W_{rm} + \frac{W_{rm}}{V_{SD}\frac{\partial \rho_{ST}}{\partial P_D}} * \delta X_e + \frac{C_L}{V_{SD}\frac{\partial \rho_{ST}}{\partial P_D}} * \delta P_D + \frac{P_D}{V_{SD}\frac{\partial \rho_{ST}}{\partial P_D}} * \delta C_L$$

where:
T_{ld} = temperature of liquid in the drum
T_{dc} = downcomer temperature
T_{fw} = feedwater temperature
h_{ld} = enthalpy of liquid in the drum
M_{ld} = mass of liquid in the drum
W_{fw} = feedwater mass flow rate
h_{fw} = enthalpy of the feedwater
L = level in the drum
ρ_{ld} = density of liquid in the drum
A_d = longitudinal area of the drum
W_{dc} = downcomer flow rate
h_f = saturation enthalpy of water
h_{dc} = downcomer water enthalpy
V_{SD} = volume of steam drum
P_D = pressure inside steam drum
ρ_{ST} = density of steam
C_L = steam valve coefficient
W_{rm} = rising water/steam mixture flow rate
X_e = steam exit quality

3. Boiling Region

$$\frac{d\delta X_e}{dt} = -\frac{A_B\rho_B Z_B\left(\frac{\partial h_f}{\partial P_B} + X_e\frac{h_{fg}}{\partial P_B}\right) + h_B A_B Z_B K_1}{A_B\rho Z_B\frac{h_{fg}}{2} - h_B A_B Z_B K_2} * \frac{d\delta P_B}{dt} + \frac{h_B\rho_B}{\rho Z_B\frac{h_{fg}}{2} - h_B Z_B K_2} * \frac{d\delta Z_{\ SC}}{dt}$$

$$+ \frac{U_{MS1}A_{MS1}}{A_B\rho Z_B\frac{h_{fg}}{2} - h_B A_B Z_B K_2} * \delta T_{M1}$$

$$+ \frac{W_2\frac{\partial h_f}{\partial P_B} - U_{MS1}A_{MS1}\frac{\partial T_{sat}}{\partial P_B} - W_{RM}X_e\frac{\partial h_{fg}}{\partial P_B} + WB3\ h_f}{A_B\rho Z_B\frac{h_{fg}}{2} - h_B A_B Z_B K_2} * \delta P_{\ B}$$

$$- \frac{U_{MS1}L_{MS}(T_{M1} - T_{sat}) - h_f WB4}{A_B\rho Z_B\frac{h_{fg}}{2} - h_B A_B Z_B K_2} * \delta Z_{SC} - \frac{h_{Xe}}{A_B\rho Z_B\frac{h_{fg}}{2} - h_B A_B Z_B K_2} * \delta W_{rm}$$

$$- \frac{W_{rm}h_{fg}}{A_B\rho Z_B\frac{h_{fg}}{2} - h_B A_B Z_B K_2} * \delta X_e + \frac{h_f WB1}{A_B\rho Z_B\frac{h_{fg}}{2} - h_B A_B Z_B K_2} * \delta T_{MZ}$$

$$+ \frac{h_f WB2}{A_B\rho Z_B\frac{h_{fg}}{2} - h_B A_B Z_B K_2} * \delta T_{DC}$$

$$h_B = h_f\left(1 - \frac{X_e}{2}\right)$$

$$\rho_B = \rho_f\left(1 - \frac{X_e}{2}\right) + \rho_g\frac{X_e}{2}$$

$$h_{Xe} = h_f + X_e h_{fg}$$

$$
\begin{aligned}
\frac{d\delta P_B}{dt} = &-\frac{\partial P_B}{\partial T_{sat}} * \frac{d\delta T_{\ DC}}{dt} - \frac{\partial P_B}{\partial T_{sat}} \frac{T_{DC} + T_{sat}}{Z_{SC}} * \frac{d\delta Z_{SC}}{dt} \\
&+ \frac{2}{A_{SC}\rho_{SC}Z_{SC}C_{PW}} \frac{\partial P_B}{\partial T_{sat}} (U_{MS1}L_{MS}Z_{SC} - C_{PW}T_{sat}WB1) * \delta T_{M2} \\
&- \frac{2}{A_{SC}\rho_{SC}Z_{SC}C_{PW}} \frac{\partial P_B}{\partial T_{sat}} (U_{MS1}L_{MS}Z_{SC} - W_{DC}C_{PW} + C_{PW}T_{sat}WB2) * \delta T_{DC} \\
&- \frac{2}{A_{SC}\rho_{SC}Z_{SC}C_{PW}} \left(U_{MS1}A_{MS1} + W_2C_{PW} + C_{PW}T_{sat}WB3 \frac{\partial P_B}{\partial T_{sat}} \right) \delta P \\
&+ \frac{2}{A_{SC}\rho_{SC}Z_{SC}C_{PW}} \frac{\partial P_B}{\partial T_{sat}} \left[U_{MS1}L_{MS} \frac{2T_{M1} - T_{DC} - T_{sat}}{2} - C_{PW}T_{sat}WB4 \right] * \delta Z_{SC} \\
&+ C_{PW}T_{DC} * \delta W_{DC}
\end{aligned}
$$

where:
X_e = steam exit quality
A_B = cross-sectional area of boiling region
Q_{MS1} = heat transfer rate between metal node 1 and boiling region
L_{MS} = unit heat transfer length between metal and secondary nodes
h_{fg} = latent heat of evaporation
Z_B = height of boiling region
h_f = enthalpy of fluid
WB1, WB2, WB3, WB4 = coefficients of approximated flow equation
K_1, K_2 = coefficients given in (Berkan and Upadhyaya 1988)
C_{PW} = specific heat capacity of subcooled water
M_{SC} = mass of subcooled water
W_{DC} = downcomer mass flow rate
W_2 = mass flow rate of water leaving subcooled region
ρ_{SC} = density of subcooled water
A_{SC} = cross sectional area of subcooled region

4. Primary Coolant and Tube Wall Nodes

$$
\frac{d\delta Z_{SC}}{dt} = \frac{1}{\rho_{SC}A_{SC}} * (W_{PE} - W_{P1})
$$

$$
\frac{d\delta\ T_{\ P1}}{\mathrm{dt}} = \frac{1}{\tau_{P1}} * \delta T_{PE} - \left(\frac{1}{\tau_{P1}} + \frac{U_{PM}A_{PM1}}{M_{P1}C_P} \right) * \delta T_{P1} + \frac{U_{PM}A_{PM1}}{M_{P1}C_P} * \delta T_{M1} + \frac{U_{PM}L_{PM}}{M_{P1}C_P} (T_{P1} - T_{M1}) * \delta Z_{SC}
$$

$$
\begin{aligned}
\frac{d\delta T_{P2}}{\mathrm{dt}} &+ \frac{T_{P1} - T_{P2}}{Z_{SC}} * \frac{d\delta Z_{SC}}{dt} \\
&= \frac{1}{\tau_{P2}} * \delta T_{P1} - \left(\frac{1}{\tau_{P2}} + \frac{U_{PM}A_{PM2}}{M_{P2}C_P} \right) * \delta T_{P2} + \frac{U_{PM}A_{PM2}}{M_{P2}C_P} * \delta T_{M2} - \frac{U_{PM}L_{PM}}{M_{P2}C_P} (T_{P2} - T_{M2}) \\
&* \delta Z_{SC}
\end{aligned}
$$

$$\frac{d\delta T_{M1}}{dt} - \frac{T_{M1} - T_{M2}}{2Z_{SC}} * \frac{d\delta Z_{SC}}{dt}$$
$$= \frac{U_{PM}A_{PM1}}{M_{M1}C_M} * \delta T_{P1} - \frac{U_{PM}A_{PM} + U_{MS1}A_{MS1}}{M_{M1}C_M} * \delta T_{M1} + \frac{U_{MS1}A_{MS1}}{M_{M1}C_M}\frac{\partial T_{sat}}{\partial P_B} * \delta P_B$$

$$\frac{d\delta T_{M2}}{dt} - \frac{T_{M1} - T_{M2}}{2Z_{SC}} * \frac{d\delta Z_{SC}}{dt}$$
$$= \frac{U_{PM}A_{PM2}}{M_{M2}C_M} * \delta T_{P2} - \frac{U_{PM}A_{PM2} + U_{MS2}A_{MS2}}{M_{M2}C_M} * \delta T_{M2} + \frac{U_{MS2}A_{MS2}}{2M_{M2}C_M}\frac{\partial T_{sat}}{\partial P_B} * \delta P_B$$
$$+ \frac{U_{MS2}A_{MS2}}{2M_{M2}C_M} * \delta T_{DC}$$

where:
Z_{SC} = Subcooled height
W_{PE} = mass flow rate at the entrance of the lump
W_{P1} = mass flow rate at the exit of the lump
ρ_P = density of primary sodium
A_P = flow area of primary sodium
T_{Pi} = Bulk mean temperature of primary coolant node *i*
T_{PE} = Entrance sodium temperature
U_{PM} = Overall heat transfer coefficient between primary and metal lumps
A_{PMi} = Heat transfer area between the metal and primary node *i* ($A_{PM1} = A_{PM2}$)
M_{Pi} = Mass of sodium in primary coolant node *i*
τ_{Pi} = residence time of sodium in primary coolant node *i*
L_{PM} = Unit heat transfer length between primary and metal nodes
T_{Mi} = average metal temperature in metal node *i*
U_{MSi} = heat transfer coefficient between metal and secondary node *i*
A_{MSi} = heat transfer area between metal and secondary node *i*
M_{Mi} = mass of tube metal in node *i*
T_{DC} = downcomer outlet temperature

5. Downcomer

$$\frac{d\delta W_{dc}}{dt} = \frac{g_c A_{dc}}{Z_{dc}} * \delta P_B - \frac{g_c A_{dc}}{Z_{dc}} * \delta P_d - \frac{f_{dc}W_{dc}}{D_{dc}A_{dc}\rho_{dc}} * \delta W_{dc}$$

$$\frac{d\delta T_{dc}}{dt} = \frac{1}{\tau_{dc}} * (\delta T_{ld} - \delta T_{dc})$$

where:
A_{dc} = cross sectional area of downcomer pipes
Z_{dc} = height of downcomer pipes
W_{dc} = mass flow rate in downcomer
ρ_{dc} = desnity of downcomer fluid
f_{dc} = friction factor in downcomer piping
D_{dc} = hydraulic diameter
g_c = gravitational constant
P_D = pressure inside steam drum

τ_{dc} = resident time in downcomer piping
T_{ld} = temperature of liquid in the drum
T_{dc} = downcomer temperature

6. Water/Steam Mixture

$$\frac{d\delta W_{rm}}{dt} = C_1 * \delta P_d + C_2 * \delta P_B + C_3 * \delta Z_{SC} + C_4 * \delta W_{rm}$$

$$C_1 = -\frac{g_c A_t}{Z_{ev}}$$
$$C_2 = \frac{g_c A_t}{Z_{ev}} - \frac{g_c A_t}{Z_{ev}}\left[Z_{sc}\left(\frac{\partial \rho_{sc}}{\partial P_B}\right) + Z_b\left(\frac{\partial \rho_b}{\partial P_B}\right)\right] + \frac{A_t \phi^2 fo}{2X_{ev} D_t}\left[\frac{f_{sc} Z_{sc}}{\rho_{sc}^2}\left(\frac{\partial \rho_{sc}}{\partial P_B}\right) + \frac{f_b Z_b}{\rho_f^2}\left(\frac{\partial \rho_f}{\partial P_B}\right)\right]$$
$$C_3 = (\rho_b - \rho_{sc})\frac{g_c A_t}{Z_{ev}} + \frac{A_t \phi^2 fo}{2Z_{ev} D_t}\left[\frac{f_b}{\rho_f} - \frac{f_{SC}}{\rho_{SC}}\right]$$
$$C_4 = -\frac{f_{sc} Z_{sc} W_{rm}}{A_t Z_{ev} D_t \rho_{SC}} - \frac{f_b Z_b W_{rm}}{A_t Z_{ev} D_t \rho_f}\phi^2$$

where:
W_{rm} = rising water/steam mixture flow rate
Z_b = boiling height
Z_{ev} = height of the evaporator
A_t = cross sectional area of duplex tubes
f_{sc} = friction factor through subcooled region
f_b = friction factor through boiling region
D_t = total hydraulic diameter
ϕ = integral two-phase friction multiplier, defined in Berkan and Upadhyaya (1988)
ρ_{SC} = density of subcooled region
ρ_b = density of boiling region
X_{ev} = quality of steam in the evaporator
Z_{sc} = height of the subcooled region

7. Superheater State Equations

The superheater model considers a single-phase heat transfer regime. Dry steam is heated by the primary sodium to 875°F at full power (Berkan et al. 1990). The superheater is modeled as a five-node counterflow single-phase heat exchanger, using the same equations as the IHX in the primary system model.

8. Control Design

The steam generator model responses to four different step perturbations:

- Feedwater Temperature
- Feedwater Flow
- Steam Valve Opening
- Inlet Sodium Temperature

These four perturbations are the forcing terms of the state-space model. The main control of the steam generator is performed by means of the steam drum level control. The controller accepts four analog signals: steam-drum level, feedwater flow, steam flow, and blowdown flow. The actuator is the feedwater valve. A PID controller is applied to control the steam pressure by assuming a linear relationship between the valve opening and the corresponding pressure drop. Figure D.9 shows the comparison between the plots generated by the EBR-II model (Berkan and Upadhyaya 1988) and the PAR model; the model responses match very closely, indicating that the PAR steam generator is performing as expected.

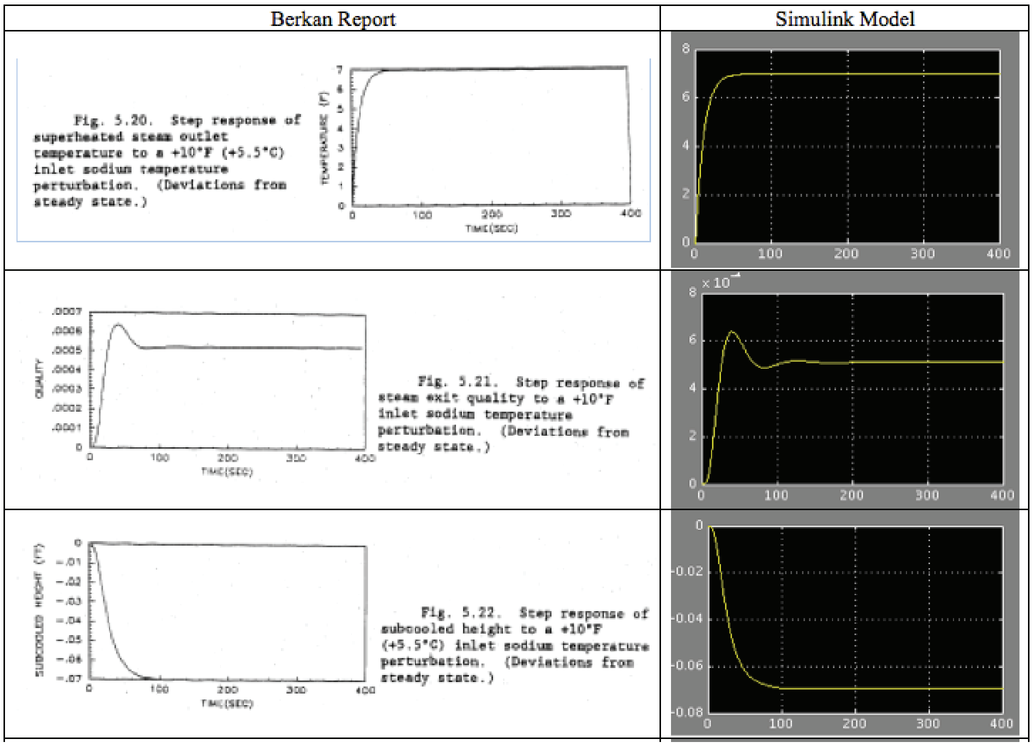

Figure D.9. The Comparison Between Berkan's Model and Simulink Model

D.2.1.3 Balance of Plant Modeling

The independent two steam generators are connected to a common balance of plant (BOP). An existing BOP model designed for a 180 MWe integral pressurized water reactor developed by Kapernick (2015), based on (Shankar 1977; Naghedolfeizi 1990; Dutta et al. 2008), was scaled to match the 40 MWe maximum output of the PAR reactor block. Figure D.10 shows the layout of the BOP (Shankar 1977). A portion of steam is routed from the steam header to the reheaters; the remainder is channeled to the nozzle chest, which regulates steam delivery to the high pressure turbine. An inline series of four turbines are connected to a single shaft, which is coupled to the generator. Moisture separator and reheater are between the high- and low-pressure turbines. Their function is to increase the enthalpy of steam from the high-pressure turbine outlet so that it may pass through the low-pressure turbines without inducing cavitation of the blades. The low-pressure turbine outlets are condensed into feedwater via the heat sink, then reheated and pumped back to the steam generator. The equations for the BOP model are given in (Kapernick 2015) and are not repeated here.

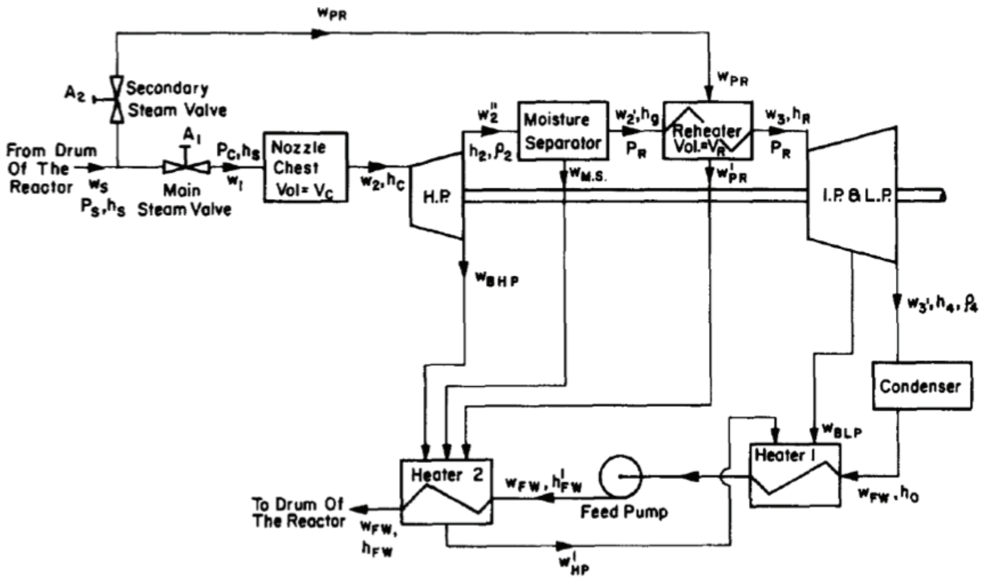

Figure D.10. Schematic of BOP system (Shankar 1977)

The steam output from the two steam generators is combined in a steam header before feeding into the BOP. Steam coming from both units is superheated and any pressure loss between the steam generator exit and the steam header is neglected. For the purpose of calculating the temperature and enthalpy of steam in the steam header, pressure of steam exiting the steam generators is assumed to remain constant at 1245 psig for the entire range of reactor operation. Steam mixture enthalpy at the steam header is calculated assuming constant steam pressure, balance of mass and steam properties, and is calculated as:

$$h_T(t) = \frac{h_1(t)\dot{m}_1 + h_2(t)\dot{m}_2}{\dot{m}_T}$$

$$\dot{m}_T = \dot{m}_1 + \dot{m}_2$$

where:

$h_T(t)$ = the temperature-dependent total enthalpy
$h_1(t)$ = module 1 temperature-dependent enthalpy
$h_2(t)$ = module 2 temperature-dependent enthalpy
$\dot{m}_T, \dot{m}_1, \dot{m}_2$= total, module1 and module 2 steam mass flow rates

The values of $h_T(t)$ obtained from the combined steam temperatures are then used to determine the temperature of the mixed steam at the corresponding superheated steam pressure of 1245 psig using a look-up table embedded in the Simulink model; this assumes that steam outlet pressure deviations can be neglected.

The BOP model provides total feedwater flow across both steam generators. The feedwater flow rate to each steam generator is proportional to the power output of that module. The feedwater temperature remains constant at 412 °F.

D.2.1.4 Multi-Modular Control Strategy

The EBR-II reactor was originally designed to operate with control rod motion to control reactor power, but constant primary and intermediate sodium flow (Sackett 2009). The reactor block is controlled to provide load following capabilities, with normal operation between 30% and 100% of full power output (12 to 40 MWe). A proposed load following scheme over the course of a day is shown in Figure D.11. The proposed control paradigm will preferentially produce power from module 1, while maintaining module 2 at a minimum of 30% power output (assuming no component degradation). This naïve approach to control is proposed only to provide data necessary to demonstrate the ERM. A fuzzy controller is currently being implemented to balance power production across the two modules as described to provide the total desired output power from the reactor block. The fuzzy rules are:

$$\begin{aligned}
&if\ [\epsilon > 0]\ AND\ [\rho_1 < 0]\ then\ \Delta\rho_1 > 0\\
&if\ [\epsilon > 0]\ AND\ [\rho_1 = 0]\ AND\ [\rho_2 < 0]\ then\ \Delta\rho_2 > 0\\
&if\ [\epsilon < 0]\ AND\ [\rho_2 > \rho_m]\ then\ \Delta\rho_2 < 0\\
&if\ [\epsilon < 0]\ AND\ [\rho_2 = \rho_m]\ AND\ [\rho_1 < \rho_m]\ then\ \Delta\rho_1 < 0
\end{aligned}$$

subject to the constraint $\rho_m \leq \rho_1, \rho_2 \leq 0$

where:

$\epsilon = TFP - (CFP_1 + CFP_2)$, the difference between the target power output and the actual power output

TFP = Total Fractional Power (power block output target)

CFPi = Core Fractional Power of module i

ρ_i = external reactivity for module i

ρ_m = minimum external reactivity (related to 30% module power)

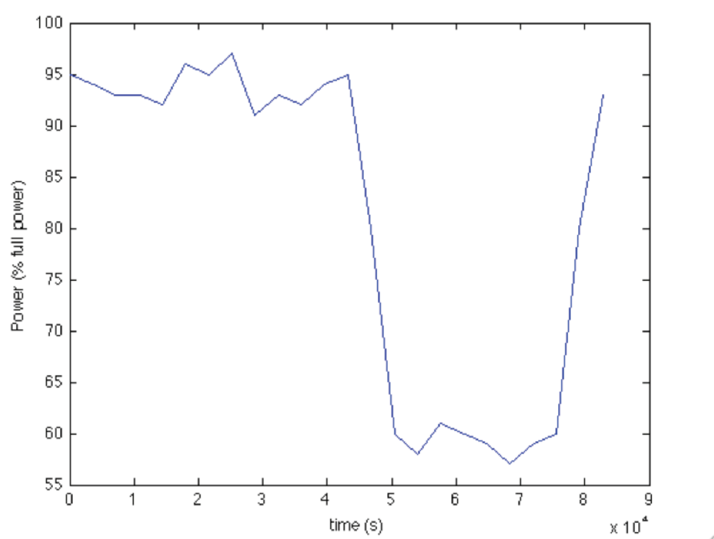

Figure D.11. A Proposed Daily Load Profile

D.2.1.5 Pump Degradation

Modeling the degradation of electromagnetic sodium pumps may prove difficult; no literature has been found to date that reports on the failure characteristics of these pumps. Centrifugal pump degradation due to cavitation can be modeled according to well-known pump curves (Grist 1998); models of degradation of mechanical pumps are employed for the purposes of demonstrating the ERM. This follows previous work in modeling pump cavitation in an integral pressurized water reactor (Hines et al. 2011). The degraded pump curves due to pump cavitation are shown in Figure D.12, where the pump curve is regenerated for each degradation level by making the following transform of the flow rate:

$$Q_p^* = \frac{Q}{p}$$

where p is the fraction of flowrate remaining, e.g. for the first degraded condition, where 99% of the flow is still available, $p = 0.99$. This effectively shifts the pump curves in along the flow variable, adjusting for the lost flow rate. By changing the flow rate in each loop, we simulate pump cavitation.

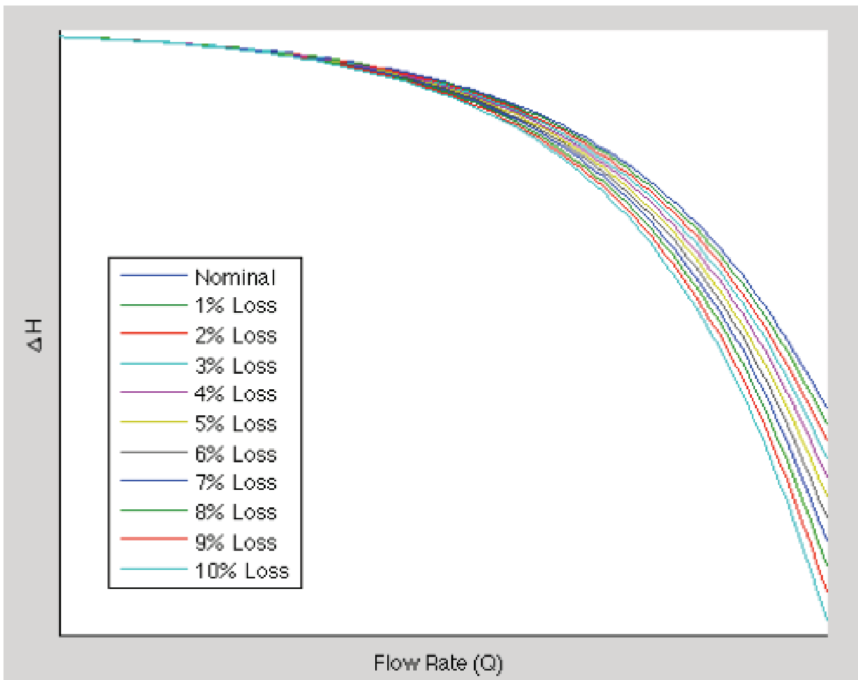

Figure D.12. Degraded Pump Curves (Hines et al. 2011)

D.3 Simulation Results

Due to negative temperature feedback effects, as the primary coolant temperature increases, fractional core power will decrease. The loss of flow in either primary or intermediate sodium loops, due to pump cavitation, will lead to an increase in coolant temperature and a corresponding decrease in core power. In the extreme case of zero flow in either case, the reactor will shut down, as shown in Figure D.13. In the case of complete loss of primary flow, the core power decreases to zero after ~100 seconds with no other action (e.g., control rod drop). Loss of secondary flow leads to core shut down in ~2200 seconds. However, degradation of the primary and intermediate sodium pumps, not complete failure, is of greater interest to the current research.

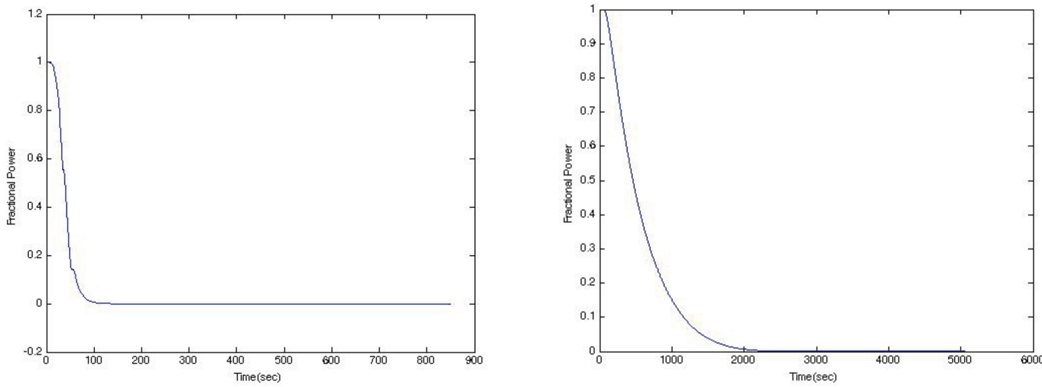

Figure D.13. Core Fractional Power Response to Complete Loss of Flow in (left) Primary and (right) Secondary Sodium

Table D.1 gives the steady state fractional power for reduced flow conditions in the primary or intermediate sodium. The results indicate that the core power decreases as pump cavitation leads to reduced flow in either primary or intermediate loops. The component condition and performance has a direct impact on overall plant performance.

Table D.1. Steady State Fractional Core Power with Degraded Primary or Intermediate Flow Conditions

Primary Sodium Flow Rate (gpm)	Intermediate Sodium Flow Rate (gpm)	Fractional Core Power
9000	5890	1.0
7000	5890	0.87
5000	5890	0.74
9000	3000	0.74
9000	2000	0.53

To date, the component degradation simulations have focused on degradation of a single pump (primary or intermediate) in a single module. After the power block control algorithm is implemented, the effect of degradation of multiple components across the two modules can be simulated. Degradation of additional components (valves, IHX, steam generator, RVACs, etc.) can easily be added to the simulation model as numeric representations of the effects of that degradation are available.

D.4 Summary and Future Work

This report focuses on the dynamic simulation of the nonlinear model of a multi-modular Prototypical Advanced Reactor using the EBR-II primary system for the reactor modules. Cavitation of primary and intermediate pumps was numerically simulated, and the effect on core fractional power was simulated. As sodium flow rate decreased due to pump degradation, power output decreased. This reduced capacity has a direct implication for the economic risk of operating the reactor block.

The described effort provides initial data to evaluate the ERM framework for advanced reactors. The developed model adequately simulates the full reactor power block under normal operation. However, in order to fully evaluate the ERM, additional degradation modes should be added beyond the current pump degradation capability. In addition, measurements that can be related to component performance (either direct measurements of performance or indicators inferred from process parameters) should be added in order to develop appropriate equipment condition assessment and prognostic models to provide the probability of failure information that the ERM requires to evaluate the operational risk.

The current fuzzy controller can be easily replaced with more advanced controllers or a risk-informed controller. Additional manipulated variables can also easily be added by augmenting the reactor equations. For instance, the primary and intermediate sodium flow rates can be used as manipulated variables to control key temperatures and power levels; currently, these flow rates are only related to the flow capacity of the appropriate pump.

Appendix E

Brief Overview of Reactor O&M Practices

Appendix E

Brief Overview of Reactor O&M Practices

E.1 Advanced Reactor O&M Concepts

Details of AR concepts that are likely to be adapted for AdvSMR concepts are available in the previous report for this project. Additional background on advanced reactor concepts and operational experience are available in previous reports in this series (Coble et al. 2013b; Meyer et al. 2013a). Of these, the greatest amount of operating experience comes from liquid-metal-cooled and gas-cooled reactors, although the amount of operating experience with these concepts is relatively small when compared to light-water reactors.

Given the possibility of frequently changing configurations in AR concepts to meet multiple mission goals, and the aforementioned relative lack of reliability data, techniques to integrate advanced plant configuration information, equipment condition information, and predictive risk monitors are needed to support real-time decisions on O&M (Coble et al. 2013b).

E.2 Reactor O&M Practices

In typical U.S. nuclear power plant watch-standing practice, operations concentrate more on determining actual equipment condition, looking for trends, or approach to a specific limit, rather than trying to predict remaining service life. If an equipment item gives signs of trouble, frequently the response is to bring on an installed spare, shut down the questionable unit, and contact maintenance for evaluation and any necessary repair. For example, in the case of typical small pumps found in the plant, full power operation only requires typically 2 out of 3 installed units. Generally, if any attempt to predict remaining life is made, it is made by maintenance personnel, usually by informal techniques, based on experience with the equipment in question and similar equipment.

In general, Planning and Scheduling Modules (PSM) are used in the planning of outage activities (Cetiner et al. 2013). PSM modules are expected to generate a partial schedule, work list, and parts list needed to restore the SSC during the next planned or unplanned outage. Because an unplanned outage can start with very little warning, the PSM needs to frequently update this unplanned shutdown work list (USWL) – a list of additional work that should be done if a given SSC failure causes a forced outage (additional work that can be done without interfering with the controlling path dictated by the "main" SSC repair work), but does not warrant shutting down the plant specifically to do this maintenance. This should include providing a list of materials and parts needed to do the work. An example could be a steam leak from a valve packing on the main steam lines to the turbine. The leak is small enough that it makes more sense economically to tolerate it and stay on-line, and personnel safety can be provided by warning signs and cordoning off the area. This will be put on the USWL, and a complete work package with repair procedure (including, most importantly, a careful estimate of time needed to do the repair), parts, and any needed special tools can be staged at a specific location, ready to take advantage of the opportunity provided by any unplanned shutdown. Then, assuming an unplanned shutdown does occur, and this valve packing leak can be worked on without interfering with the controlling path work, the repair will be made, improving plant condition and reducing the work load during the next planned shutdown.

Note that periodic inspections are performed on a subset of components to ensure their functionality, as part of the defense-in-depth philosophy for maintaining safety. For active components (in particular, components that are usually in stand-by mode), these inspections involve periodic testing, while passive components undergo nondestructive testing. Generally, the set of components selected for testing are based on a risk analysis, to identify components whose failure is likely to significantly increase the risk (based on CDF or other risk metrics).

E.3 ERM and Plant Supervisory Control – Preliminary Interface Recommendations

E.3.1 Brief Overview of Supervisory Control

The Supervisory Control for Multi-Modular SMR Plants Project is an effort led by Oak Ridge National Laboratory (ORNL) under the AdvSMR R&D Program to develop a new, state of the art overall control system intended to control O&M costs for multi-reactor plants to be in line with current LWR plant levels. Given the small output of each reactor, providing staff similar to current LWR practice would likely result in unsustainable O&M costs. The main overall goal is to allow operating the multi-reactor SMR plant with a staff size similar to a current generation LWR with similar total output; see SMR/ICHMI/ORNL/TR-2013/04 (Cetiner et al. 2013). The Supervisory Control system is planned for implementation as a non-safety system though it is required to not interfere with safety systems.

The output of the PSM is a schedule and work list for the next planned or unplanned outage (SMR/ICHMI/ORNL/TR-2013/04). Because an unplanned outage can start with very little warning, the PSM needs to frequently update this unplanned shutdown work list, a list of additional work that should be done if a given SSC failure causes a forced outage (additional work that can be done without interfering with the controlling path dictated by the "main" SSC repair work). This should include providing a list of materials and parts needed to do the work. It would be useful to the operations and maintenance staff for the PSM, on demand, to provide risk (success probability and/or PRA impact) for proposed on-line maintenance.

E.3.2 ERM Outputs

In general, the objective of interfacing the ERM to the supervisory control modules is to provide the control logic with a series of options that account for plant equipment condition and the risk to mission of operating the reactor given the current equipment condition. Equivalently, the risk to mission may be stated in terms of a probability of success ("success probability") given the current equipment condition.

In general, we expect that the ERM output will be utilized by the supervisory control logic as well as by the PSM module (for planning and scheduling maintenance actions). The ERM is expected to use predictive estimates of remaining life and POF from prognostic modules. At a minimum, the information provided by the ERM should be sufficient for each entity to perform its function. Specifically, the Supervisory Control logic should have sufficient information to be able to remove from service or reduce load on SSCs experiencing degraded condition, and the PSM should have sufficient information to generate a usable shutdown schedule, work list, and parts list. The output of the ERM to the Supervisory Control is likely a series of operational options with a PRA-based success probability for each. We anticipate that the ERM will generate a predictive output that quantifies the potential change in risk for a reduction in power (or other maneuver) in order to increase the success probability of avoiding an unplanned outage or decrease in safety margins.

The supervisory control logic may utilize additional information (such as diagnostic information from the ECA) in its decision making process. In addition, prognostic information that indicates the rate of degradation of SSC may be needed by the supervisory control algorithms to indicate an automatic trip of equipment that is suffering fast degradation (though this may be better handled by trip devices on SSC).

E.3.3 Some Observations

Given the likely needs from the Supervisory Control algorithms, and the potential outputs available from the ERM, the following observations may be made:

- The supervisory control logic needs to support operational modes that reduce demands on SSC experiencing degraded condition, at least until the next available opportunity for maintenance and return to serviceable condition. Note that this will require the ERM to provide information that enables the appropriate tradeoffs (revenue generated by running the plant in a degraded state vs. incremental risk in terms of cost and safety metrics). Ideally, the operation of the plant can be extended to the next maintenance outage. However, if there is no option that will make it to the next planned outage (above some predefined minimum success probability), then other time periods will need to be considered. A useful practical minimum for remaining run time might be to run long enough to obtain parts and materials for repair.
- The ERM needs to provide output periodically (say hourly or daily) to the Supervisory Control during steady state, as well as "on demand" when plant power or power split (electrical to thermal) changes by some threshold amount. On demand output should also be provided when significant SSC is diagnosed with a problem.
- Any ERM output that triggers action by the Supervisory Control modules needs to be available to the (human) operator for review in an easy to comprehend format.
- Ideally, the ERM and/or the PSM needs to evaluate not only success probability of operating particular SSC until the next scheduled outage, but also (if possible) to calculate the probability of doing more expensive to repair damage to SSC by continuing to run them. This is effectively a cost-benefit analysis. For example, minor failure of a bearing can frequently be repaired by just replacing the bearing material itself, without having to rework the shaft. Continued operation at loose clearances can damage the shaft itself, considerably complicating repair and ratcheting cost upwards.
- A mechanism for self-test or test by the Supervisory Control Module might be useful to verify that the ERM module(s) are operating properly, and are not suggesting counterproductive or unnecessary action.

These observations may lead to substantial computational complexity of both the supervisory control and ERM modules. To reduce the computational demand on the overall system (supervisory control and ERM):

- It may be useful to limit the number of power reduction options to limit the number of choices the Supervisory Control has to make. An example may be options that reduce the power output in integral multiples of 5% (95%, 90%, 85% etc.), at least initially. This is to simplify the needed ERM calculations to account for the opportunity cost of operating the plant for an extended period of time below 100% power. Note that the cost of running the plant for a fixed period of time is (roughly) the same regardless of plant power, while generation revenue is proportional to plant power.
- It may be useful to suppress operational options with success probabilities below some threshold, to help focus the supervisory control algorithm to only viable options. Determination of the minimum viable success probability will need to be done carefully, and would involve an assessment of the various possible risk metrics to better understand the tradeoffs involved.

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