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Work Domain Analysis of a Predecessor Sodium-Cooled Reactor as Baseline for AdvSMR Operational Concepts

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March 2014

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ABSTRACT

This report presents the results of the Work Domain Analysis for the Experimental Breeder Reactor (EBR-II). This is part of the phase of the research designed to incorporate Cognitive Work Analysis in the development of a framework for the formalization of an Operational Concept (OpsCon) for Advanced Small Modular Reactors (AdvSMRs). For a new AdvSMR design, information obtained through Cognitive Work Analysis, combined with human performance criteria, can and should be used in during the operational phase of a plant to assess the crew performance aspects associated with identified AdvSMR operational concepts.

The main objective of this phase was to develop an analytical and descriptive framework that will help systems and human factors engineers to understand the design and operational requirements of the emerging generation of small, advanced, multi-modular reactors. Using EBR-II as a predecessor to emerging sodium-cooled reactor designs required the application of a method suitable to the structured and systematic analysis of the plant to assist in identifying key features of the work associated with it and to clarify the operational and other constraints.

The analysis included the identification and description of operating scenarios that were considered characteristic of this type of nuclear power plant. This is an invaluable aspect of Operational Concept development since it typically reveals aspects of future plant configurations that will have an impact on operations. These include, for example, the effect of core design, different coolants, reactor-to-power conversion unit ratios, modular plant layout, modular versus central control rooms, plant siting, and many more. Multi-modular plants in particular are expected to have a significant impact on overall OpsCon in general, and human performance in particular. To support unconventional modes of operation, the modern control room of a multi-module plant would typically require advanced HSIs that would provide sophisticated operational information visualization, coupled with adaptive automation schemes and operator support systems to reduce complexity. These all have to be mapped at some point to human performance requirements.

The EBR-II results will be used as a baseline that will be extrapolated in the extended Cognitive Work Analysis phase to the analysis of a selected advanced sodium-cooled SMR design as a way to establish non-conventional operational concepts. The Work Domain Analysis results achieved during this phase have not only established an organizing and analytical framework for describing existing sociotechnical systems, but have also indicated that the method is particularly suited to the analysis of prospective and immature designs. The results of the EBR-II Work Domain Analysis have indicated that the methodology is scientifically sound and generalizable to any operating environment.

NOTE: It is assumed that readers are familiar with previous milestone reports, hence some important background information that would have added substantially to the bulk of this report has been omitted. References to previous reports are included where relevant.

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ACRONYMS

ADF	Abstraction-Decomposition Framework
AdvSMR	Advanced Small Modular Reactors
AH	Abstraction Hierarchy
ALMR	Advanced Liquid Metal Reactor
ANL	Argonne National Laboratory
ANSI	American National Standards Institute
AOO	Anticipated operational occurrence
AdvSMR	Advanced small modular reactor (non-light water reactor technology)
CRBRP	Clinch River Breeder Reactor Project
CWA	Cognitive Work Analysis
DC	Direct current
DiD	Defense-in-Depth
DOE	U.S. Department of Energy
EBR	Experimental Breeder Reactor
EOP	Emergency Operating Procedure
FFTF	Fast Flux Test Facility
FOAK	First-of-a-Kind
FUM	Fuel Unloading Machine
GE	General Electric
HFE	Human factors engineering
HRA	Human reliability analysis
HSI	Human-system interface
HVAC	Heating, ventilation, and air conditioning
IBC	Interbuilding Coffin
ICHMI	Instrumentation and Control and Human Machine Interface
I&C	Instrumentation and control
IAEA	International Atomic Energy Agency
IEEE	International Electrical and Electronics Engineers
INCOSE	International Council on Systems Engineering
LOCA	Loss of Coolant Accident
INL	Idaho National Laboratory
LMR	Liquid metal reactor
LWR	Light water reactor
MCR	Main control room
MWe	Megawatts electricity
MWt	Megawatts thermal
NEA	Nuclear Energy Agency
NOP	Normal operating procedure
NPP	Nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
O&M	Operations and maintenance (costs)
OCC	Outage control center

OCD	Operational concept document
OpsCon	Operational Concept
PCU	Power conversion unit
PPE	Personal Protective Equipment
PRA	Probabilistic risk analysis
SEMP	Systems Engineering Management Plan
SEP	Systems Engineering Process
SSC	Structures, systems, and components
WDA	Work domain analysis

Work Domain Analysis of a Predecessor Sodium-cooled Reactor as Baseline for AdvSMR Operational Concepts

1 Introduction and Background

The new generation of small, modular nuclear reactors currently being designed around the world and also in the US are all expected to be simpler, safer, and cost-competitive compared to other energy options. These plants will be characterized by unique structural and functional designs, such as multiple reactor modules controlled via high levels of automation from a single control room. One of the more challenging aspects of the introduction of Advanced Small Modular Reactors (AdvSMRs) into the US nuclear fleet involves the detailed description of how these plants will be operated and by whom.

A previous milestone report (Hugo et al., 2013b) described how the operation and associated design requirements of a complex socio-technical system could be described by means of an Operational Concept document. It was described how the Operational Concept (referred to as OpsCon in the rest of this report) identifies and defines the characteristics of a proposed system from the viewpoint of an individual who will use that system. In the course of a new engineering project it is used to communicate the quantitative and qualitative system characteristics of the plant to all stakeholders, including system owners, managers, designers, engineers, probabilistic risk analysts, human factors engineers (HFE), utility owners, etc. It also provides the basis for procedures (e.g. training, operating, maintenance, etc.) and serves as input to functional requirements analysis, function allocation, task analysis, staffing, human-system interface (HSI) design, automation design, and control room design.

Literature reviewed for this project illustrated that advanced operational concepts are not well documented, if at all. It was difficult to find clear information on the organizational and operational impact associated with implementing these new reactors, specifically the expected impacts on engineering, operations, instrumentation and control (I&C), and maintenance functions. This is especially true for multiple-purpose hybrid energy plants where the boundaries between processes may intersect and personnel may have dual roles. As reported previously, no information is currently available on the type of safety critical operational scenarios that might include the interaction of nuclear power processes with other processes, such as hydrogen production. It is almost guaranteed that operators of these new plants will be faced with new tasks due to the increased ability of multi-modular plants to load-follow, to distribute load demand among multiple units, and to transition among different product streams. It seems obvious that much of this will be achieved through higher levels of automation, advanced HSI technologies, computerized procedures, and on-line maintenance of multiple reactor units. All engineering disciplines involved in first-of-a-kind (FOAK) design will be affected, and particularly I&C and HFE. The advanced features envisaged will mean that designers cannot simply rely on operating experience or design practices based on legacy designs. The definition of new operational concepts, procedures, staffing and training will require up-to-date methods and tools.

As described in Hugo et al. (2013b), a new plant requires a comprehensive OpsCon document that describes the plant's structure, systems and their functions, and how operating personnel will interact with the system to achieve the plant goals. All engineering groups will require detailed definition of the unique operating scenarios that will influence the design of systems and procedures, and also how humans and the operating context will interact under different operational conditions to achieve the plant's operational and safety goals.

The technical basis for operational choices will be achieved through a detailed evaluation of operational concepts that would include high levels of automation, advanced HSI technologies, computerized procedures, and on-line maintenance of multiple reactor units. All of the features enabled by advanced technology will result in new challenges for the definition of plant concepts of operations, systems design, and staffing and training. For example, it is expected that operational sequences will include failure phenomena such as high temperature excursions and other types of disturbances not associated with light water reactor (LWR) designs. Past research has shown that the new generation of reactors will include an extensive list of human performance issues associated with such conditions that have not been empirically evaluated in detail (O'Hara et al. 2011, 2012; Hugo et al. 2013b).

Although there is no AdvSMR operating experience that completely informs the development of Operational Concepts, a considerable amount of conceptual design information in published literature is available, along with operating experience from predecessor plants, such as the Experimental Breeder Reactor-II (EBR-II) and the PRISM Advanced Liquid Metal Reactor Design. Given these sources of information, it was possible to make significant progress in developing preliminary operational concepts for AdvSMRs. For example, from an examination of this information it was possible to anticipate the impacts of various AdvSMR designs on the human aspects of operations, such as workload, situation awareness, human reliability, staffing levels, and the appropriate allocation of functions between the crew and various plant systems that are likely to be highly automated. Nevertheless, these impacts are largely uncertain and will remain uncertain until empirical research data become available to support the development of sound technical bases. Given these uncertainties and other issues, it has become critical that new operational concepts must be researched, developed, and held up to analysis for this next generation of nuclear power plants (NPPs).

Current HFE and systems engineering methods practiced in the nuclear industry do not offer specific analytic tools for deciding how to design the operation of a new plant, including the application of new types of automation. A comprehensive, systematic process that would be formally called out, for example, as part of the system life cycle described in the INCOSE (International Council on Systems Engineering) Systems Engineering Handbook 3.2.2, is needed to address that shortcoming and produce reliable information for the design of robust and resilient systems that allow dynamic collaboration between operators and plant systems.

This report summarizes the progress to date on efforts to develop a framework for the analysis and definition of operational concepts. It also describes how a Cognitive Work Analysis (CWA) framework was applied to the definition of operational concepts of a predecessor sodium-cooled reactor. The report also describes the roles of human and system agents for that plant, and the systems involved in a variety of operational conditions.

The report concludes with a description of how the baseline results will be extrapolated to the analysis of a selected advanced sodium-cooled SMR design as a way to establish non-conventional operational concepts.

2 Project Objectives and Scope

The report INL/EXT-13-30117 described the importance of conducting research to provide technical bases for designers to incorporate these principles in their designs. Since it almost guaranteed that AdvSMRs will require non-traditional operational concepts and requirements to match new processes and technologies like advanced automation systems. Particular emphasis was placed on the critical importance of integrating human factors considerations into the systems engineering process throughout the project lifecycle. Incorporating all the engineering and human factors elements into an integrated systems engineering process is not a trivial undertaking and representing this process visually is difficult. For this reason it was important to adopt a methodology that is suited to the analysis and description of work domains where all engineering and human elements need to be identified very early in the project life cycle, especially for first-of-a-kind projects that lack operating experience of sufficient information from predecessor plants.

An end goal of this phase of the research was therefore to incorporate CWA in the development of a framework for the formalization of an Operational Concept for AdvSMRs. For a new AdvSMR design, information obtained through CWA, combined with human performance criteria, could also be used in during the operational phase of a plant to assess the crew performance aspects associated with identified AdvSMR operational concepts.

2.1 Significance

The extensive literature on CWA and recent results for the AdvSMR OpsCon project at Idaho National Laboratory (INL) suggest that the nuclear industry is in serious need of a methodological makeover, especially with regard to the way operating concepts are designed and the roles of humans are defined.

While CWA may not be a magic bullet in Systems Engineering, our experience and results to date strongly suggest that the CWA approach will add significant value in all new-build projects, especially those dealing with advanced reactor and automation technologies.

2.2 Assumptions and Constraints

The constraint described in Hugo et al. (2013a) must be reiterated here: the only predecessor plant that could provide useful design information and operating experience for the purpose of this project was the EBR-II, shut down in 1994. Although not all EBR-II operational information was archived, sufficient information was obtained to allow an extensive analysis of the work domain and its operational concepts.

3 Methodology

3.1 Cognitive Work Analysis Methodology Background

This report is concerned with the analysis and description of a complex socio-technical system. Specifically, the aim of the report is to develop an analytical and descriptive framework that will help systems and human factors engineers to understand the design and operational requirements of the emerging generation of small, advanced, multi-modular reactors.

To analyze a system like this requires the application of a method that would assist in identifying key features of the work associated with it and to clarify the operational and other constraints. The original CWA methodology was developed by Jens Rasmussen to analyze complex socio-technical systems such as those found in nuclear power generation (Rasmussen, 1974, 1987). This was mainly in response to the realization by systems and human factors engineers that conventional engineering and human factors methods were inadequate to achieve an in-depth understanding of the interrelations of social and technical systems and how constraints act upon the working of system functions (for example the activities in the main control room of a NPP.

A complex socio-technical system is made up of numerous interacting parts, both human and non-human, operating in dynamic, ambiguous and often safety critical-domains. The complex relationships and constraints found in systems like these present significant challenges for analysis and design. Most traditional methods are not well suited to capture the complexity of these interrelations. The semi-structured, formative framework presented within CWA helps to guide the analyst through considerations of the various levels of constraints acting on systems and the effects that can have upon the way in which work can be carried out. As a formative approach, CWA focuses on constraints within the work domain, which provides the analyst and engineer a far greater understanding of how the functions and tasks would impact the work domain and vice-versa.

Although CWA has a long history in research in the military and some specialized domains, it has never been applied on a large scale in the nuclear industry. This is not surprising, since no major nuclear engineering projects have been undertaken since construction was started in 1973 on the last nuclear power plant in the US, Watts Bar. New construction projects started only after the Energy Policy Act 2005, which provided new stimulus for investment in electricity infrastructure, including nuclear power. New reactor construction got under way from about 2012, with first concrete on two units in March 2013. However, these new projects are for LWRs, and although they will make use of more advanced materials and systems and also be more automated than legacy plants, they are essentially known technology, which means designers can rely on well-established engineering methods. The same will not be true for AdvSMRs. Here designers will be confronted by challenges, many of which will lack proof-of-concept experience. This is where CWA will provide a proven and verifiable way to reduce uncertainty and risk associated with FOAK projects.

However, it should be emphasized that the aim of this phase of the AdvSMR OpsCon project is not to develop requirements for the design of systems, but rather to identify the principles and constraints that would influence the development of operational concepts for a new plant. Only once these concepts can be verified would it be possible to translate them into system design requirements.

3.2 CWA model components

As shown in previous reports, CWA can be broken down into 5 phases, each with a defined outcome that serves as input to the next phase:

Table 1: Cognitive Work Analysis Phases

CWA Phase	Product
Work Domain Analysis	Abstraction Hierarchy and Abstraction-Decomposition Framework
Contextual Activity Analysis	Decision Ladders
Strategies Analysis	Course of Action, Information Flow Map
Social Organization and Cooperation Analysis	Combination of previous
Worker Competencies Analysis	Skills, Rules, Knowledge Inventory, high-level function allocations

Stanton and Bessell (2013) explain how the five main phases of CWA focus on different constraint sets and how this presents different perspectives on the system. These five phases shown in Table 1 were described briefly in Hugo et al., 2013b).

Wilson (2012, cited in Stanton and Bessell, 2013), describe the key characteristics of the methodology:

- *Systems focus*: CWA captures the whole socio-technical system in the five different perspectives and does not favor one system over the other.
- *Context*: CWA analyzes system behavior within the work domain using observations in context with input from Subject Matter Experts. System boundaries are defined by the subject matter of interest as well as by emergent findings during the course of the analysis, such as operational scenarios, explained later in this report.
- *Interactions*: The interacting parts of the system are revealed in the Social Organization and Cooperation Analysis phases, in which functions are constrained by the relationships between situations and actors, decisions and actors, and strategies and actors (in the Contextual Activity Analysis and Strategies Analysis, described later in this report).
- *Holism*: The system analysis is based on the whole system, from the Abstraction Hierarchy (AH) and throughout all of the phases and representations. This is a significant strength of CWA.
- *Emergence*: The emergent properties of the system are revealed through different phases of analysis, particularly in the Social Organization and Cooperation Analysis phase where all of the previous phases are assigned agents. (This is the equivalent of the conventional Function Allocation process described in the previous milestone report). It serves to bring the systems analysis together. The different perspectives offered by the analyses reveal the combined systems properties and their associated emergent properties.

- *Embedding*: Since the method lends itself to be integrated into the systems engineering discipline, it offers familiarity to organizations wishing to develop operational concepts and subsequent technical designs for an existing or new socio-technical system. It has the benefit of representing the system in diagrammatical form, which supports communication with other engineers.

As pointed out by all leading CWA authors and practitioners, it is important that the first phase (construction of the WDA) be particularly thorough because the level of accuracy at this stage will determine the accuracy and validity of the other phases (Wilson, 2012; cited in Stanton and Bessell, 2013).

WDA identifies the functional constraints on the activity of the system. As such WDA provides the foundation for all of the subsequent phases. As Bennett and Flach (2012) point out, the ultimate aim of the methodology is to inform design. The design challenge is to ensure that the semantic content of a design is communicated so effectively to operators that it will enhance their situation awareness and ensure that situational constraints on operator actions are well-specified and congruous with the demands of a work ecology. The CWA approach is therefore to design functional and visual representations so that there is an explicit mapping between the patterns in the representation and the situational constraints. Inevitably, this requires an analysis of operational conditions in the work domain to identify these constraints, as well as an understanding of what functional and communication elements contribute to awareness and to know what patterns are likely to be salient. "*We believe that the quality of representing the work domain constraints will ultimately determine effectivity and efficiency; that is, it will determine the quality of performance and the level of effort required*." (Bennett and Flach, 2012).

3.3 The Work Domain Analysis Method

Hugo et al. (2013b) explained how CWA provides a structured framework for the analysis and development of complex socio-technical systems. The framework helps the analyst to consider the environment within which operational functions and tasks take place and the effect of the imposed constraints on the system's ability to perform its purpose. This is a formative methodology that guides the analyst through the process of answering the question of *why* the system exists; *what* activities are conducted within the domain as well as *how* this activity is achieved and *who* is performing it (Jenkins et al., 2008).

The emphasis in this part of the project is on the first phase of CWA, Work Domain Analysis. This is the foundation upon which all subsequent analyses for the development of operational concepts is built. As explained in previous reports, CWA, and especially WDA are now beginning to be accepted as standard practice in many industries, especially for FOAK engineering such as the design of multi-unit AdvSMRs. Many practitioners regard this as superior to more conventional human factors engineering practices. This is because analysis of the work domain and its functional and structural characteristics identifies a fundamental set of constraints on the actions of any human or system agent, thus providing a solid foundation for subsequent analysis and design phases. The goals and functions of the work domain impose constraints on workers by specifying the purposes that the work system must fulfill, the values and priorities that the work system must satisfy, and the functions that the work system must perform (Naikar et al. 2005, Naikar, 2013). Therefore, the work system environment within which the task is conducted has the potential to significantly affect the task and ultimately the entire plant operation. This report will demonstrate how CWA, and specifically WDA, is particularly suitable as an organizing framework for analysis of the key principles of AdvSMR operation.

In combination, the goals and purposes of the work domain define the fundamental problem space of workers and include the values, priorities, and functions that must be achieved by a work system with a given set of physical resources. However, within these constraints, workers have many options or possibilities for action in the work domain. This becomes the basis for the further allocation of functions to humans or systems, the analysis of tasks, determination of skills, rules and knowledge involved in those tasks, the definition of operating principles and requirements, and ultimately the design of human-system interaction tools to enable operators to perform the identified tasks effectively, efficiently and safely.

The main aim of WDA for AdvSMR Concepts of Operations is thus to model the constraints that relate to the functional and physical context within which workers of a new generation of NPPs will perform their tasks. For example, the environmental, physical and functional requirements of an advanced plant will impose physical as well as mental constraints on workers. These constraints will determine the physical objects that must be available to the operators to perform their tasks as well as the functional capabilities and limitations of those objects.

As explained in previous milestone reports (Hugo et al., 2012, 2013a, 2013b), WDA has never been applied in the development or analysis of concepts of operation in the nuclear industry. Previous efforts in various industries to standardize the format and the process of developing concept of operations documents have also not paid much attention to human factors issues. Nevertheless, examples in the literature (Roth, Patterson and Mumaw, 2012; Bisantz and Vicente, 1994) suggest that WDA is the most systematic and structured method for this purpose.

This report demonstrates first of all that the methodology is a powerful way to analyze existing systems, and, secondly, that WDA of a predecessor system can produce a valid baseline for the analysis of a related, but more advanced design. In fact, CWA in general will enable HFE to cross the chasm between the old, often ineffective paradigms, and advanced design approaches that are not just different, but add significant value in both human and technological terms, especially with regard to the way operational concepts are designed and the roles of humans are defined.

Two additional CWA models are included in the methodology adapted for this project: Contextual Activity Analysis and Strategies analysis. These two models are described below.

3.3.1 Abstraction Hierarchy

The product of a WDA is known as an Abstraction Hierarchy (AH) and its main purpose is to identify and describe the purpose, structure, interdependencies and functions of the systems. As the main product of WDA, the AH typically models the system at five levels of abstraction; at the highest level the overall functional purpose of the system is considered, at the lowest level the individual components within the system are described. The five levels of abstraction are as follows:

1. *Functional Purposes*

   These purposes describe the reasons for the existence of the system. The purposes are independent of time and exist for the lifetime of the system.

2. *Values and Priority Measures*

   These are measures that determine how well the system is achieving its functional purposes. The way that the system is configured to meet these needs is contextually dependent and thus also describes the constraints on the functional purposes.

3. *Purpose-Related Functions*

   These are the general functions of the work system that are necessary for achieving the functional purposes. The functions have the ability to influence one or more of the values and priority measures and they link the purpose-independent processes with the object-independent functions.

4. *Object-Related Processes*

   These processes describe the processes that are performed by the physical objects in order to perform purpose-related functions. In particular, they capture the functional capabilities and limitations that enable or inhibit the purpose-related functions of physical objects in the work system.

5. *Physical Objects*

   The boundaries of an analysis will limit these objects to the systems of direct relevance to the chosen operational scenario, rather than every single object in the plant. The boundaries of the analysis will thus indicate the levels of fidelity applied. In an attempt to keep the analysis manageable, the boundary will omit individual "non-operable" components. In principle it would be possible to decompose most of the listed objects into their component parts and describe their affordances more concisely; however, this will not help to describe the constraints within the scenario more accurately, as will be demonstrated in the analyses in the next section.

However, the real strength of the AH is in identifying the constraints upon the system and its functions. Constraints can be either *causal* (that is, determined by physical or natural laws), or intentional (determined by social laws, conventions, policies or values). For example, the structure, functions and dependencies of a complex sociotechnical system like a nuclear power plant are influenced by the properties of the thermohydraulic process, materials and specific technologies. It is also influenced by regulations, company policy, market requirements, design conventions and many other intangible constraints. The analysis of a system therefore depends upon the degree to which the behavior of the human and system agents within the system is influenced by the relationship and interaction between causal and intentional constraints. Causality and intentionality lie on a continuum from high to low, which leads to the definitions of *tightly-coupled causal systems* and *user-driven intentional systems* (Naikar, 2013, p.38). This relationship can be illustrated as follows:

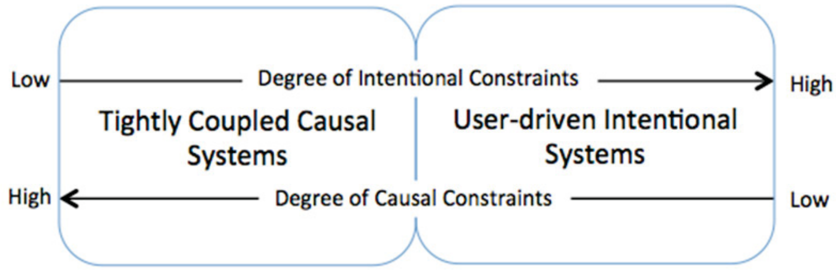

Figure 1: Characterizing system constraints (Adapted from Naikar, 2013)

Nuclear power plants and industrial processes are typical examples of tightly coupled causal systems, whereas typical examples of user-driven intentional systems are research institutes, educational institutes, public information systems, the World Wide Web, and so on. The NPP as a tightly coupled causal system would be characterized, for example, by causal (physical) constraints such as the requirements for reactor cooling, conversion of fission heat to mechanical energy, conversion of mechanical energy to electrical energy, containment of fission products, requirements for environmental control, and many more. Intentional constraints would be governmental regulations (such as staffing requirements and safety regulations), licensing requirements, company policy, and so on. All of these constraints have an effect on the way the plant is operated, and must therefore be taken into consideration when operational concepts are developed.

Typical causal and intentional constraints identified in complex systems include constraints imposed upon the behavior or activities of people due to:

1. purpose or mission of a system (causal as well as intentional);
2. the physical situation, including environmental conditions (causal);
3. specific operational situations, such as operator response time, accuracy or situation awareness (causal);
4. the available means by which activities can be performed (causal and intentional);
5. organizational structures, specific actor roles and definitions (intentional);
6. human capabilities and limitations, such as physical strength, mental workload or fatigue (causal);

Similarly, causal as well as intentional constraints may be imposed on the ability of systems to perform a function due to:

1. physical limitations imposed by material properties;

2. reliability limits under adverse operational conditions, such as temperature, pressure, speed, etc.;

3. inability of operators to act quickly enough to either initiate or stop a function;

4. spatial limitations in plant layout, preventing systems to be located in optimal positions;

5. limits in production capacity of a system or component, such as flow rate, volume, tolerance, heat conductivity or dissipation, speed, etc.;

6. limits in the control capability of the automation system;

7. limits imposed by industry codes or regulations for reliability, safety or quality;

3.3.2 Means-Ends Links

The use of means-ends-links and the utility of the AH can be described with an example from the figure below which deals with a generic abnormal condition:

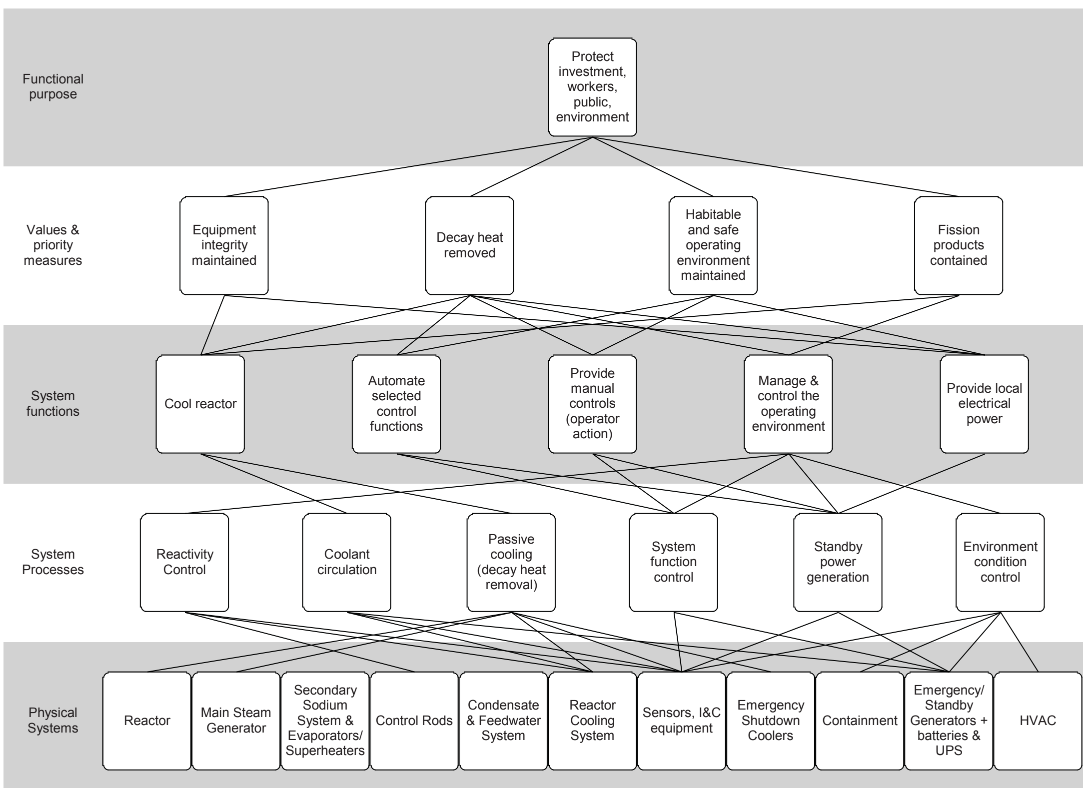

Figure 2: AH for Generic Abnormal Operations

Consider the node 'Habitable and safe operating environment maintained' in the Values and Priority Measures level: following the link from the top of this node, it answers the question 'why is this needed?', in this case to 'Protect investment, workers, public, environment'. Following the links down from the 'Habitable and safe operating environment maintained' it is possible to answer the question 'how can this be achieved?', in this case, by cooling the reactor, automating selected control functions, providing manual controls for operator action, managing and controlling the operating environment, and by providing local electrical power.

The same method can be applied either top-down or bottom up, for example, by focusing on an element in any level and then determining why that element is needed and what is needed to ensure that it will serve that purpose.

3.3.3 Abstraction-Decomposition Framework

The abstraction-decomposition framework, or ADF (also called the abstraction-decomposition space by some authors) is the main tool for modeling the purposive and physical work context or problem space of workers. This is essentially extends the AH by considering the part-whole architecture of structures, systems and components for each of the abstraction levels. As shown below, this is typically represented in a table where the abstraction dimension is shown along the vertical axis of the ADF, and the decomposition dimension is shown on the horizontal axis of the ADF.

	Whole System	Subsystems	Components
Functional Purposes			
Values and Priority Measures			
Purpose-related Functions			
Object-related Processes			
Physical Objects			

Figure 3: Abstraction-Decomposition Framework Format

In practice the ADF is similar to a conventional system breakdown structure, except that it is associated with the five levels of abstraction described above. An example of a conventional system breakdown structure for EBR-II is shown below.

3. Primary Sodium Cooling System
3.1 Primary Sodium Pumps (2)
3.2 Flow Meters
3.3 Primary Auxiliary Electromagnetic Pump (and Battery)
3.4 Ball Joint Connector
3.5 Throttle Valves (High and Low Pressure)
3.6 Grid Plenum Assembly
3.7 Intermediate Heat Exchanger (IHX)
3.8 Shutdown Cooling System
3.8.1 Primary Sodium to NaK Heat Exchanger
3.8.2 NaK to Air Heat Exchanger
3.8.3 Louvers
3.8.4 Flow Meter
3.9 Primary Sodium Purification System
3.9.1 Sodium Surge Tank
3.9.2 Vacuum Pump
3.9.3 DC-EM Pump
3.9.4 Economizer
3.9.5 NaK AC-EM Pump
3.9.6 Nuclide Trap
3.10 Silicone Coolant System

Figure 4: Conventional System Breakdown Structure

3.3.4 Contextual Activity Analysis

Contextual Activity Analysis (CAA) models the so-called Control Tasks for a particular scenario. This is concerned with the constraints associated with what needs to be accomplished in a system. These activities may be characterized as a set of recurring work situations, plant conditions, work functions or control tasks (Naikar, 2013). For example, these contexts may be specific operational conditions within a defined scenario, such as upset conditions or transitions.

The following table format (Figure 5) is typically used to identify the correlation between plant functions and operational conditions:

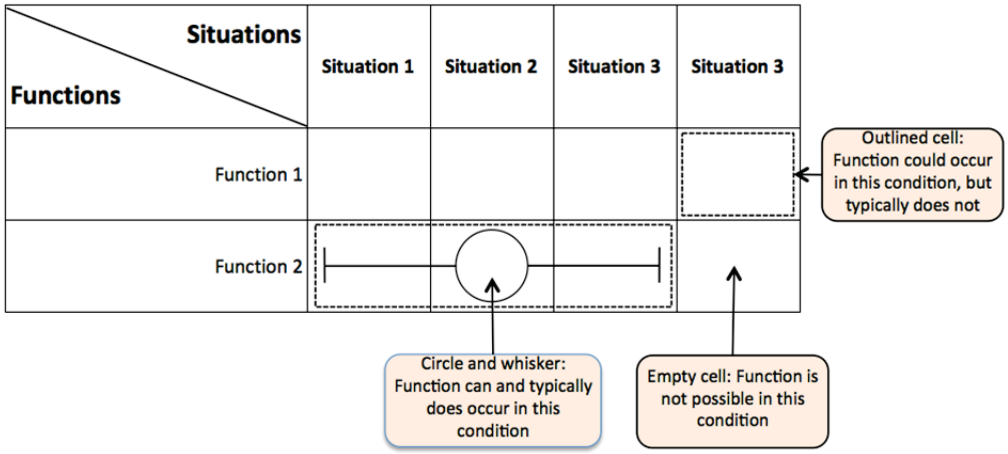

Figure 5: Contextual Activity Template

The CAA leads directly to two subsequence models: strategies analysis and decision ladder analysis. The former is discussed below, but decision ladders form a detailed part of CWA and will only be included in a later report.

3.3.5 Strategies Analysis

The Strategies Analysis phase models the alternative pathways from one system state to another, which could include ways to mitigate the consequences of an adverse condition. The strategies adopted under a particular situation may vary significantly, depending on the constraints within that situation. For example, either a human or an automated agent could perform an activity, or each may perform different parts of an activity. Also, different agents may perform tasks in different ways, and the same agent (either human or nonhuman) might perform the same task in a variety of different ways. The strategy that the agent selects will be dependent on many variables, for example, their experience, training, workload, familiarity with the current situation, available of tools, environmental conditions and accessibility of work areas, familiarity with procedures, and many more. It is assumed that activities in an environment like a NPP are mandated by standard as well as emergency operating procedures. This means that often such procedures will limit the strategies available to an agent. However, the CWA analyst must also consider the applicability of procedures and allowable workarounds.

It was pointed out earlier that EBR-II had minimal automation and therefore the strategies analyses presented in this report focused only on human operators. It is expected however that in future AdvSMR designs there will be significant differences in these strategies in the presence of advanced automation systems; this will be examined in more detail in the CWA for an advanced design during the next project phase.

The following diagram format is typically used to identify different strategies to transition from one system state to another. In this example, the condition starts with a failure of the control rod drive mechanisms, which means that the reactor does not scram when it is supposed to. To handle this situation, four alternative strategies may be effective; the simplest one is a manual scram and the most complex strategy involves several inspection, diagnostic and manual actions to actuate the control rod drive mechanisms.

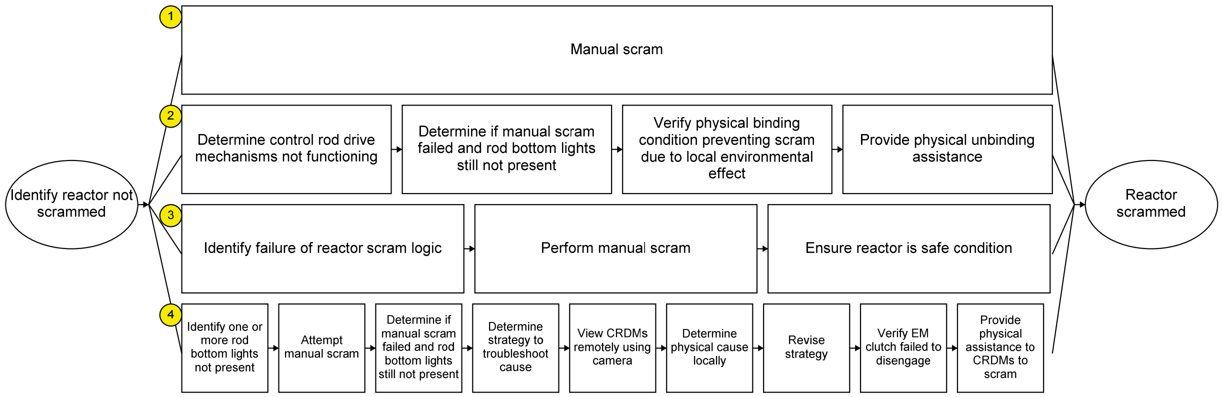

Figure 6: Strategies Analysis Diagram

In combination, the AH, ADF, CAA and strategies analysis provide early insights into how the physical and functional architecture of the sociotechnical system could influence strategies to handle specific operational conditions. In particular, the CAA provides the "course of action" basis for many subsequent decision regarding operating practices and procedures and the associated control and monitoring artifacts. The remaining phases of CWA, described briefly below, form the bridge between high-level operational principles and design requirements for the plant's operational concepts as well as the role of humans.

3.4 Remaining CWA models

The two models remaining models in the typical CWA framework are Social Organization and Cooperation Analysis and Worker Competencies Analysis. These two models (analyses?) are not included in this phase of the project, but will be included in the further analysis for an advanced design.

These two models are briefly described:

- Social Organization and Cooperation Analysis

  This model (analysis?) is concerned with how work is allocated among human workers and intelligent agents. The objective is to determine how the social and technical factors in a sociotechnical system can work together in a way that enhances the performance of the system as a whole (Jenkins et al., 2012, Stanton and Bessell, 2013). In earlier milestone reports for this project this was referred to as Function Allocation. The difference in the CWA method is that explicit attention is given to the relationships to the work domain, contextual activities and operational strategies, or course-of-action. For example, the strategy of turbine startup and synchronizing to the grid might be distributed across human workers and automation; here the operator provides the ramp-up rate and the automation system responds with the steps, checks and holdpoints and waits for the operator's permission. The AH, ADF and Contextual Activity Analysis can all be used as templates in deciding how to allocate functions.

- Worker Competencies Analysis

  This is the final phase of the CWA framework. It involves identifying the competencies that operators require to perform the required activity within the system under analysis. It is only at this phase of CWA that the particular constraints of human workers are considered, because the constraints identified in the initial phases of CWA will affect the analysis of worker competencies. For example, if a control task is allocated purely to automation, then human workers will not require skills for performing this control task. The primary goal of Worker Competencies Analysis is to identify psychological constraints applicable to systems design. As the final phase of the CWA framework, Worker Competencies Analysis inherits all requirements identified through the four previous phases. The typical output of this analysis is a table detailing the skills, rules and knowledge elements for specific operational strategies (Rasmussen, 1986).

It is important to note that the output of this part of the method is not a finished design. Instead, the entire CWA process provides information on constraints and considerations for developing information requirements used in subsequent interface design activities.

4 Work Domain Analysis Results for EBR-II

4.1 EBR-II as a Predecessor Design

Liquid-metal reactors like sodium/potassium- or lead/bismuth-cooled designs currently seem to be the most prominent AdvSMR designs and the designs most likely to be licensed within the next ten to fifteen years. However, at the time of writing, none of the reactor designs currently in progress (e.g., Toshiba 4S, GE PRISM, TerraPower TWR, etc.) was mature enough to have design information available for analysis. It was therefore decided to conduct an exploratory exercise using information from a predecessor sodium-cooled reactor.

The subject matter and relevant information was derived from the design of the Experimental Breeder Reactor (EBR-II), which was the successor of EBR-I. The EBR-II reactor design is the basis for several of the current AdvSMR sodium fast reactor designs, in spite of its 1970s-era technology, including analog I&C. In considering emerging sodium reactor designs for the next phase of this project, we will assume that across all operating scenarios there will be a high degree of automation, including the likelihood that operators will be able to take manual control of components, systems, and processes when necessary or appropriate. We will also assume that the control rooms will employ advanced, digital I&C and HSIs.

A subject matter expert who was an operator on EBR-II and a senior EBR-II engineer have assisted with the analysis. The results have established a baseline and will be used to extrapolate EBR-II operating principles to a more modern sodium-cooled reactor design, based upon various assumptions, such as modularity, plant layout, and higher levels of automation. The ultimate aim is twofold: a) to produce a set of unique AdvSMR operational scenarios for at least one candidate design and b) to formalize and document the methodology for future application by AdvSMR designers and human factors analysts.

At EBR-II, automation existed only at the component level, and manual control of systems and processes was the operational norm. Such limitations were part of the original design and determined primarily by the limited automation and digital control capabilities at the time of construction. In AdvSMR sodium-cooled designs, it is expected that automation will be at the system and/or process levels. The automatic control of systems and processes will most likely be the norm, though dual control capability and the capability for manual control will be required (e.g., for off-normal or emergency events). Examples of the systems we expect to be under automatic control include reactivity control (automatic control rod drive system), primary and secondary sodium systems, steam plant systems, turbine control and fuel handling operations.

The following diagram illustrates the as-built-layout of EBR-II:

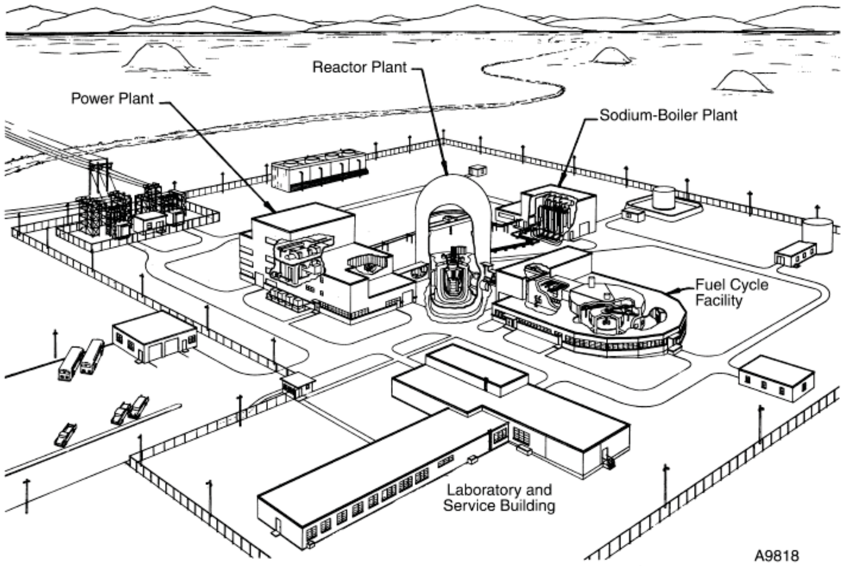

Figure 7: EBR-II Plant Layout

The following flow diagram of a sodium fast reactor serves as a simplified model to illustrate some key performance parameters of EBR-II:

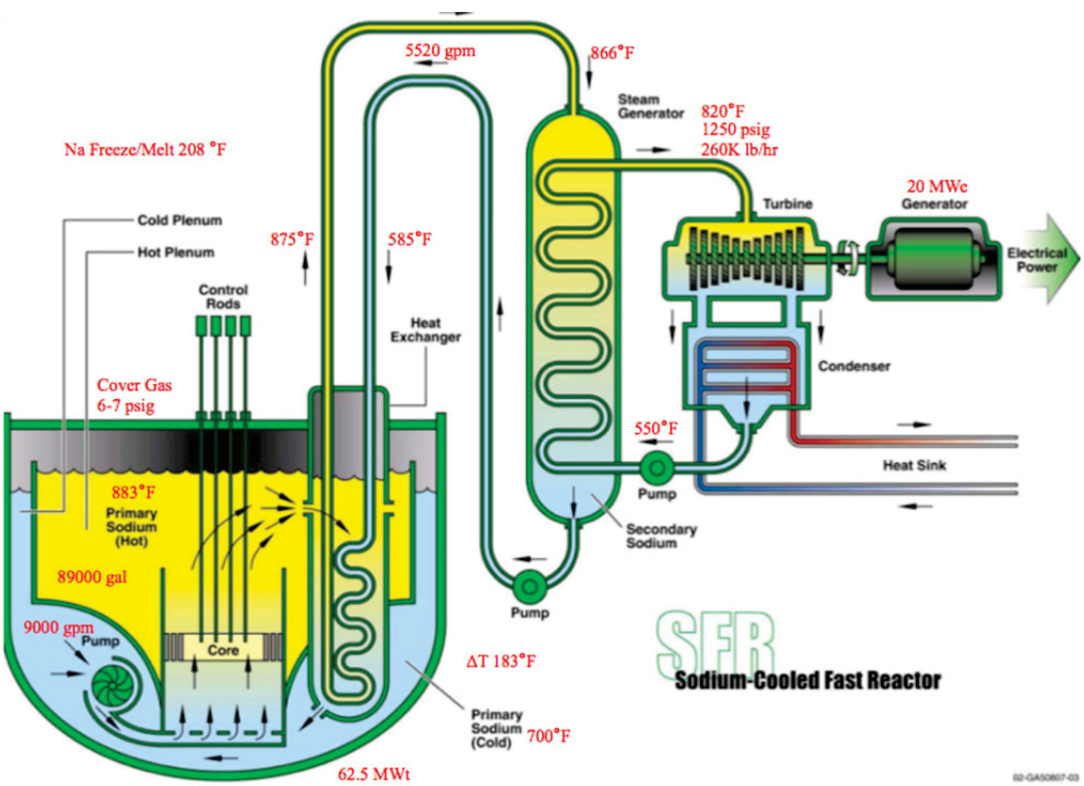

Figure 8: Simplified EBR-II Flow

The overall flow of the primary and secondary systems is presented in the following Functional Flow Block Diagram. This diagram reflects the majority of the system decomposition described in section Figure 11.

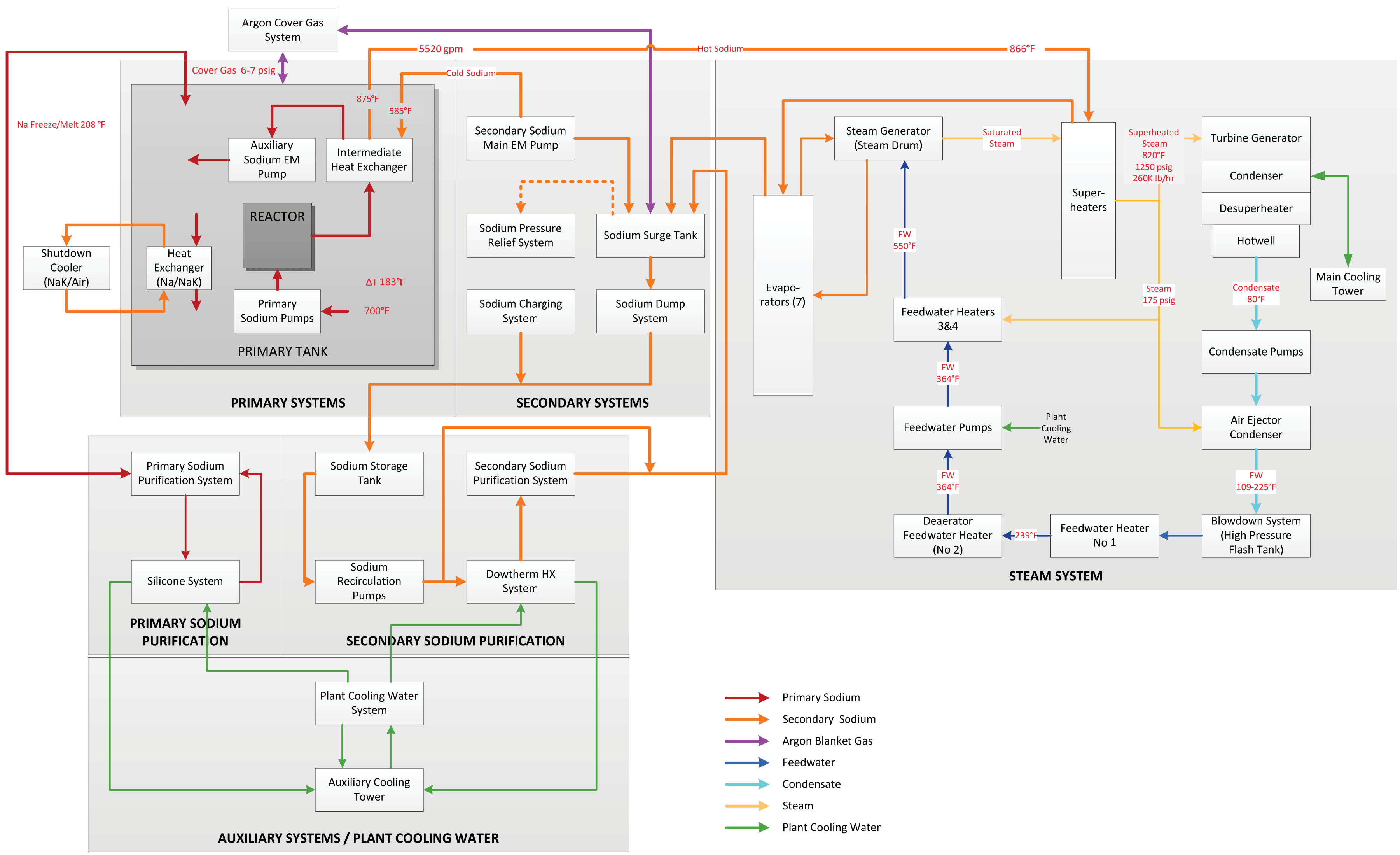

Figure 9: EBR-II Functional Flow and Relationships – Primary, Secondary and Steam Systems

The reasons why EBR-II is regarded as a valid predecessor for future sodium-cooled reactors are included in the summary of its operational characteristics, in section 4.2.

Following the analysis of literature on CWA, the research team collected and analyzed a large amount of technical and procedure documentation from the EBR-II archives. These documents were mentioned in the previous milestone report, but are repeated here for reference purposes:

- Operating Instructions, Vol 1 – 7
- Technical Specifications
- System Design Descriptions – Primary and Secondary Systems
- Emergency Procedures
- Probabilistic Risk Assessment – Sections 1 – 14
- Reactor System Training Manuals Vol 1 - 5

These documents enabled the analysts to develop a comprehensive WDA for a number of scenarios, described in a later section.

4.2 EBR-II Operational Characterization

The following table summarizes the key characteristics of EBR-II that distinguish it from a light water reactor plant. In the next phase of the project these characteristics will be assessed from the perspective of a new design that may make use of new materials, different plant configuration, higher levels of automation, and many more potential differences.

Table 2: EBR-II Operational and Design Characteristics

Characteristic	Description
Plant Mission	1. Proof-of-concept – viability of sodium cooled fast reactor, reactor safety, fast breeder (fuel production), fuel performance, fuel recycling and reprocessing, waste reduction, and electric generation. 2. Irradiation and fuel testing for future fast reactors (Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor Project (CRBRP)). 3. Operational safety testing - subjecting fuel and the plant to off normal conditions such as operation of fuel with breached cladding and ultimately, the reactor inherent-safety demonstration tests, which subjected the reactor to anticipated transients without scram. 4. Integral Fast Reactor prototype, including demonstration of non-aqueous reprocessing and fuel recycling technology. 5. Decommissioning, yielding important information about the technology of sodium processing for disposal.
Physical plant configuration	Single reactor, single steam generator, single power generator. Reactor Building separated from the Power Plant, Sodium Boiler Building and Fuel Reprocessing Facility.
Operating pressures and temperatures	Reactor coolant is at atmospheric pressure - bulk sodium temperature at 695-705°F. Liquid sodium does not need to be kept at high pressure, even at very high temperatures. Sodium also has a high boiling point (> 1640°F at atmospheric pressure). The reactor is characterized by low-pressure primary systems that

Characteristic	Description
	eliminate fast events associated with water-cooled reactors, such as loss of coolant accidents (LOCAs).
Engineered Safety Features	No penetrations through wall of primary tank; all penetrations are through the top. A guard tank surrounded the primary tank with an annulus between them, which allows for detection of sodium leakage. The guard tank is in turn surrounded by concrete shielding which acts as a final containment vessel. Were leakage to occur in both the primary and guard tank, the core would not be uncovered and would be adequately cooled. The pool-type primary system also provides distinct advantages. Its large thermal capacity limits the severity of thermal transients and therefore limits stress on the primary tank and components submerged within it. The piped pool configuration allows for the majority of the primary sodium to be at reactor inlet temperature, further increasing the capacity of the sodium to absorb heat in the event of an upset. All primary system components are submerged in this relatively cold pool of sodium, which proved to be beneficial for their operating reliability and ease of removal for maintenance or repair. It also minimizes the potential for leakage of primary sodium, since all penetrations are through the top of the vessel. Decay heat can be removed from the primary sodium by shutdown cooler thimbles immersed in the primary sodium and filled with sodium-potassium, which removes decay heat by natural convection, and forces flow of air through the annulus between the primary tank and its guard tank. Because decay heat removal does not depend on the secondary sodium loop, sodium in that loop can be drained to a storage tank for maintenance or in the event of a sodium leak. The secondary sodium loop is designed such that a severe reaction between sodium and steam would not endanger the reactor. The steam generators are double-wall tubes to minimize the potential for leakage. The tube sheets are configured such that there is a plenum between the two tube sheets at each end, which provides a path for sodium or steam to travel if one of the tubes were to fail, facilitating detection.
Cooling systems	• No active safety injection system is required. The reactor is designed for passive cooling and rejects heat by conduction and convection. Heat rejection to an external water-cooled heat sink is not required. • No safety-related pumps for accident mitigation. No need for sumps and protection of their suction supply. Core cooling can be maintained using natural circulation. • Automatic reactor shutdown system upon loss of electric power. Reduced need for manual scram. • No operator intervention required for safety injection. No need for manual or mental calculations. Fewer emergency operating procedures (EOPs) required. Passive safety systems ability to remove core heat, leads to significant safety enhancement. • No operator action required for coolant pump failures or leaks. No need for leak calculations.
Fuel	Uranium alloy metal fuel pins in a subassembly of 91 pins in hexagonal stainless steel cladding tubes, providing a versatile and “forgiving” fuel design, able to accommodate a wide range of compositions and providing a large degree of self-protection in response to off-normal events.
Breeder Reactor Concept	The Breeder Reactor concept is based on the premise that nuclear power should be produced by “burning” natural uranium (depleted uranium as long as it is available) and is accomplished by converting uranium-238 to plutonium in fast

Characteristic	Description
	reactors. In this process, plutonium functions as the catalyst for consuming uranium-238 and the reactor is fueled with uranium-238.
Fast Reactor Concept	From the beginning, it was recognized that fast reactors would be significantly different from thermal (neutron moderated) reactors. The much smaller neutron cross-sections, both fission and capture, led to an entirely different geometry and design of these reactors. The fast reactor required high fuel density and relatively high fuel enrichment to achieve criticality in a fast neutron environment in which the fission cross-section is small. On the other hand, a broad choice of materials was permissible for reactor structures and coolant, also because of the small neutron capture cross-sections. Fast reactors are relatively insensitive to fission product buildup and their effect on reactivity of the reactor. This unique characteristic of fast reactors to tolerate fission product buildup made it feasible to incorporate into the EBR-II a fuel recycle process that did not remove all of the fission products. Although these fission products capture neutrons, the impact on the total neutron balance is insignificant.
Fuel Handling	Fuel Handling processes are divided into two broad categories: • *Unrestricted* operations: those that are performed with the reactor shut down. It involves the movement of subassemblies between the reactor and intermediate storage in the storage rack with the reactor shutdown, and the control rods disconnected from their drives. This includes fuel transfer operation and fuel transport operation. All of these operations are performed in the primary tank with the subassemblies submerged in, and cooled by, sodium. • *Restricted* operations: those that were performed with the reactor in operation. It involves transfer of subassemblies between the storage rack and the inter-building coffin (IBC), using the fuel-unloading machine (FUM), including the transition from sodium as the coolant, to argon gas as the coolant for spent assemblies, and the transition from gas to sodium for new or reprocessed subassemblies. These transitions occur as the subassemblies are transferred between the sodium environment in the primary tank and the inert gas environment in the fuel unloading machine and vice versa. Fuel transport operations involves the transport of subassemblies between the IBC transfer station in the Reactor Plant and the air cell in the Fuel Cycle Facility. It includes the transport through the equipment air lock between the buildings, which maintains the containment integrity of the reactor containment building. Both the fuel transfer and fuel transport operations could be performed while the reactor was operating. It was intended that these operations be performed as needed to meet the requirements of the external fuel cycle (i.e., recycle or storage/disposal).
Containment	• No pressurized containment was required, therefore no operator action was required to activate containment systems to reduce steam pressure or to remove radioiodine from containment. • The containment design reduces the level of challenge to the containment vessels and building by means of an integral primary system. By integrating the reactor and primary cooling systems in the containment, catastrophic LOCAs are eliminated.

Characteristic	Description
	• Access to the containment building is not restricted during operations, unlike LWR designs.
Non-proliferation	The use of on-site fuel recycle as pioneered by EBR-II, eliminates the shipment of irradiated material, thus making it far less accessible to would-be proliferators, and decreases the opportunity for theft or misappropriation. The absence of the need for shipment through public space eliminates the public safety concerns about shipping highly-radioactive or toxic materials.
Shutdown requirements	Negative temperature coefficient of reactivity and lower shutdown margin help to avoid power excursions, which means less need for active safety systems, and enables the reactor to shut down without operator action in the unlikely event of a LOCA and resulting rise in temperature.
Staffing and Crew roles and responsibilities	Crew consists of Shift Manager, Reactor Operator, Secondary Sodium Operator, Panel Coolant Operator, Field Coolant Operator, Power Plant Operator (includes Electric Plant), and Chemistry Technician
I&C and human-system interaction	Predominantly analog and hardwired systems used for I&C. A small number of digital instruments.
Normal Operations	Reactor at full power, primary and secondary cooling systems fully operational, argon cover gas system functional, primary tank heaters deenergized, generator power output nominal 20MWe, emergency diesel generators in standby. The reactor would load follow easily, responding through reactivity feedback as inlet coolant temperature changed in response to changes in power demand. No operator action is required in such a case. Metal fuel is not adversely affected by cyclic changes in power and temperature, which coupled with its strong tendency to maintain a constant average core temperature, greatly facilitates its ability to load follow. EBR-II could be easily controlled by fixing control rod position and controlling power demand at the steam turbine.
Abnormal operations	Typical Anticipated Operational Occurrences: reactor scram (manual or automatic), secondary sodium system water-to-sodium leak, minor earthquake, major leak in reactor outlet piping, loss of normal power.
Procedures	Normal and emergency operator procedures conducted as prescribed by plant policies.
Hazards	The only significant hazards identified in the PRA include external events such as fire, floods and earthquake. The dominant internal event is a sodium leak leaking to water/aid interaction. The PRA found that EBR-II was relatively insensitive to human error and few important human actors for nuclear safety are identified. For example, there is little or no requirement for electrical power to prevent fuel damage. The majority of human actions are focused on plant asset protection. It was shown that risks associated with EBR-II operation were substantially lower than typical LWR plants, an order of magnitude less. The EBR-II risks would have been lower still except for its seismic response. (Subsequent plants employ seismic isolation to mitigate even this risk).
Maintenance	Maintenance techniques were proven, with exposure to personnel less than 10% of that for a comparable Light Water Reactor (LWR). Sodium also has a high boiling point (> 1640°F at atmospheric pressure) allowing the primary and secondary systems to be low-pressure. Consequently, there is no potential for high-pressure ejection of coolant. This feature is important for maintenance activities and is a major reason that there were no injuries from sodium leakage over the course of EBR-II operation.

Characteristic	Description
Safety Testing	The EBR-II plant was subjected to all of the Anticipated Transients without Scram (ATWS) events without damage to fuel and SSCs, demonstrating the self-protecting characteristics of a metal-fueled fast reactor. The first of these tests was loss of all pumping power with failure to scram, simulating a station blackout with failure to scram. The second test subjected the reactor to loss-of-heat-sink (secondary sodium flow stopped) without scram.

4.3 EBR-II Operating Scenarios

The operation of all NPPs can be described in terms of operating scenarios. A scenario could describe a specific condition, its evolution over time, specific performance parameters of entities involved during the condition, start and end states of various entities, and the state of the system overall upon termination of the scenario. Scenarios are therefore useful in describing normal operational conditions, but particularly abnormal and emergency conditions where decisions made to mitigate the situation and the corresponding procedures are dependent upon the availability and comprehensibility of a lot of detailed information.

As described earlier, CWA focuses on identifying properties of the work environment and of the workers that define possible boundaries on the ways that human-system interaction might reasonably proceed, without explicitly identifying specific sequences of actions. This is called formative modeling and it is usually conducted as part of preparation for a new design, which could be instantiated in the form of, for example, a control room or HSIs. The application of CWA to the development of operational concepts for a new proposed design is therefore slightly unconventional, but nevertheless regarded as a powerful method to add value to early system engineering design efforts. In fact, we now consider it to be an essential part of our design formulation and approach.

The use of operating scenarios for CWA deviates from normal practice, but this is an essential adaptation because it serves as a graded approach to the analysis of the sociotechnical system. Because of the structural and functional complexity of the power plant as a whole, the scenario-based approach helps to identify clear boundaries for an analysis. In particular, it enables the analyst to include only the systems and functions involved in the scenario. At the same time, it helps to identify the constraints only in that part of the whole system. In addition, by developing a high-level AH for the whole system, it becomes much easier to identify the relationships between individual scenarios and the structure of the overall system.

An important part of operational scenarios is the specific performance parameters of systems during the different operational conditions described in the Contextual Activity Analysis. These parameters are also considered in the “values and priority measures” of the AH and could be qualitative or quantitative measures (see Figure 2 and Table 4 and 5 for examples). The following sections describe the basis for the identification of pertinent operating scenarios.

4.3.1 Mode and State Analysis

The design characteristics of AdvSMRs will inevitably require the definition of operational modes and states, some of which may be similar to those for light water reactors, but also new ones that are unfamiliar, such as unplanned shutdowns with degraded conditions in one module that may affect other modules, off-normal conditions at more than one module, running adjacent units while others undergo refueling operations, or adjusting module power levels to enable load following. These modes will require new operator tasks such as: load following operations, managing non-LWR processes and reactivity, and novel refueling methods. New plant modes and tasks will inevitably create complexities and require innovative treatments in the design and use of appropriate HSIs. Ultimately, all of these conditions will require development of a new family of normal and emergency operating procedures for multi-unit disturbances. CWA is an excellent tool for supporting this development.

The following state-transition diagram (Figure 10) was developed and used in conjunction with the documents listed above as the basis for identifying important normal and abnormal operating conditions. More detailed descriptions of the modes are provided below the figure.

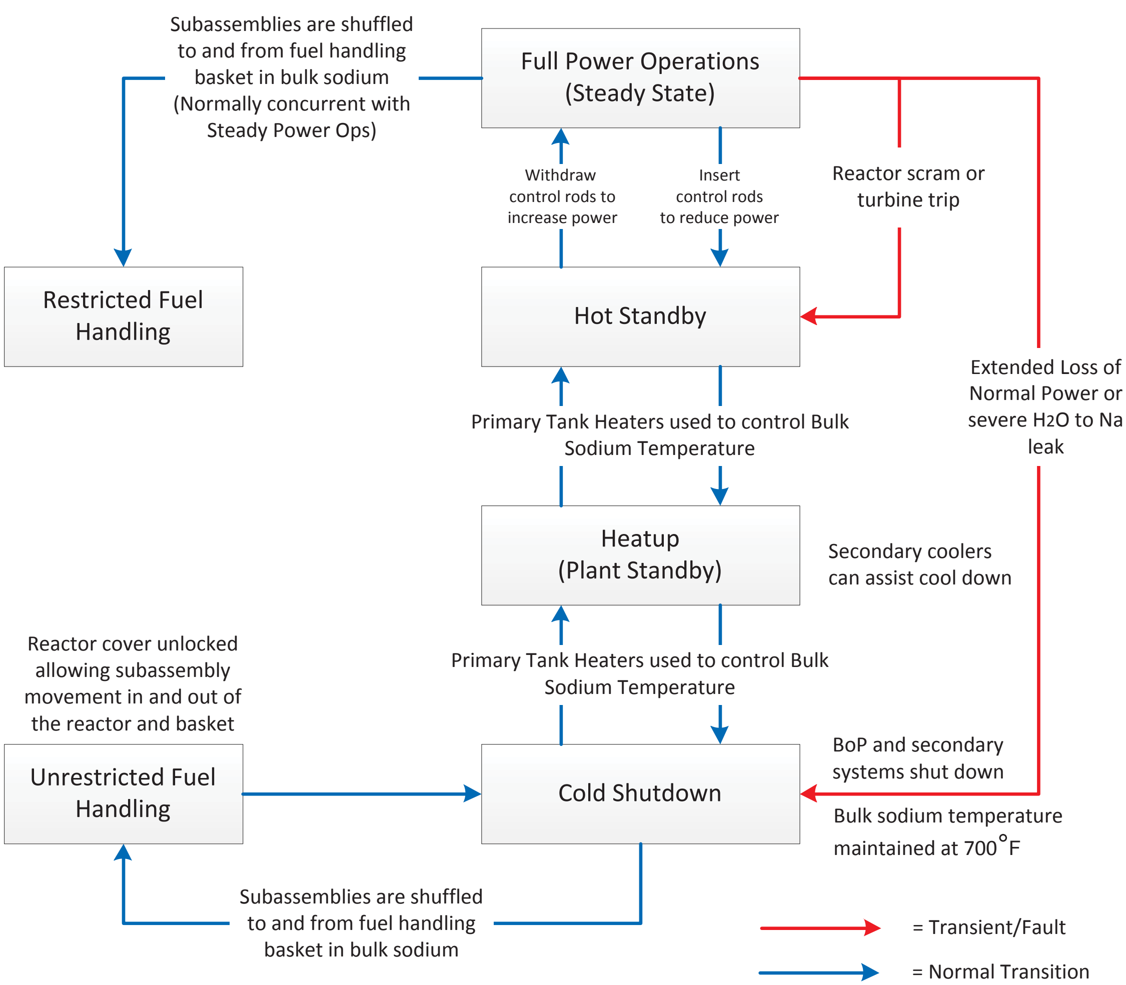

Figure 10: EBR-II State Transition Diagram

Mode 4: Plant Shutdown (Cold Shutdown)

The reactor is shutdown by normal procedures and plant system standby conditions are established and maintained.

Mode 4a: Unrestricted Fuel Handling

The mode of operation involves the reloading of the reactor by transferring subassemblies between the reactor and the storage basket. The operations described in Restricted Fuel Handling are also allowed during this mode. This can be done only with the reactor shut down and the plant systems are in standby, with the primary bulk sodium temperature is in the range of 580-730^{0}F.

Mode 3: Plant Standby (Heatup)

In Plant Standby mode, the reactor is shut down, the primary system, secondary system, and major portions of the steam system are maintained at temperature levels as close as practical to those that will permit reactor startup and increase to power.

Mode 2: Plant Startup (Hot Standby)

This mode of operation requires that Plant Standby conditions exist, the reactor will be started up, and power will be increased to steady power level specified by the Reactor Run Plan and Authorization.

Mode 1: Power Operations (Steady State)

The reactor and plant systems are operating at steady power level condition within the range of 1.6% (1 MWt) to maximum authorized power of 100% (62.5 MWt). The turbine generator may or may not be operated with reactor power level less than 18 MWt.

Mode 1a: Restricted Fuel Handling

The transfer of subassemblies between the storage basket and the fuel-unloading machine. This can be done at any time provided the primary bulk sodium temperature is in the range of 580-730^{0}F.

Special Modes of Operation

Certain modes that are not routine in the startup, shutdown, or normal steady-state power operation of the plant are classified as special modes of operation. These non-routine modes are:

- Manual or Automatic Reactor Scram (rapid insertion of control rods)
- Anticipatory Reactor Shutdown (unplanned reactor shutdown as a result of a malfunction)
- Plant Heatup (described in more detail below)
- Plant Cooldown (described in more detail below)
- Secondary System Partial Drain and Recovery

Plant Heatup

This procedure is used whenever the plant systems (excluding the primary sodium system) have been cooled to ambient temperature and they are to be heated to plant standby conditions. The secondary system can be drained to the storage tank and cooled to ambient temperature, but the sodium in the storage tank is normally maintained at 350 ^{0}F.

Plant Cooldown

The plant cooldown mode encompasses several plant system cooldown conditions:

- Cooldown of the primary system from 700^{0}F to 580^{0}F with the secondary and steam systems at standby conditions.
- Cooldown of the primary, secondary, and steam systems from 580^{0}F to 350^{0}F.
- Cooldown of the secondary and steam systems from 350^{0}F to ambient.

Secondary System Partial Drain and Recovery

This is a special procedure that includes the following evolutions:

- Partial drain of the secondary system
- Maintenance during partial drain condition
- Recovery from partial drain condition

4.3.2 State Matrices

This state transition diagram was further elaborated in a state matrix for normal and abnormal (fault mode) operating conditions (See Table 4 and Table 5 in the Appendix). These matrices describe the operating conditions for the following major systems:

- Reactor cooling system
- Control rods and safety rods
- Secondary sodium system
- Emergency shutdown coolers
- Turbine generator
- Condensate and feedwater system
- Main steam system
- Evaporators and superheaters
- Fuel handling and associated equipment
- Electric plant

4.3.3 Scenario Analysis

Five operational scenarios were analyzed in detail. The scenarios are described according to thirty-two criteria that are deemed necessary to define all pertinent human and system performance parameters. This includes the systems involved, the primary roles of operators, start and end conditions, and specific system performance parameters.

In addition to four design basis events, normal operating conditions are also described to obtain a steady state baseline for the WDA. The scenarios are as follows:

1. Power Operation (Steady Power and Initial Condition for All Events)
2. Secondary Sodium Systems – Water-to-Sodium Leak
3. Earthquake
4. Reactor Scram (Manual or Automatic)
5. Loss of Offsite Power

The analysis illustrated that EBR-II fuel handling processes were complex and operators had a large amount of activities and safety-related requirements that demanded a high degree of situation awareness. The design required a significant amount of manual fuel handling and monitoring of sodium temperatures. Performance requirements for manual use of the crane and control of fuel assembly movement in and out of the fuel basket during restricted fuel operations contributed significantly to the physical and cognitive workload. For example, the operator depended to a great extent on haptic senses (that is, tactile feedback) to verify that there has been a positive capture of the fuel subassembly.

It was seen that the EBR-II control systems essentially required the plant to be operated at a component level. This imposed many manual tasks on the operator that are likely to be automated for a modern sodium-cooled reactor design. The PSF part of the analysis also revealed that the high workload and potential for error that characterized this older EBR-II function will be sufficient justification for future automation. For example, it is possible that with future designs the automation will monitor system parameters and alarms and provide integrated data, trends, and displays to the operator. The automation systems will also perform data analysis, diagnostics and prognostics, and provide more integrated data and diagnosis information to the crew. This means that the operator's role is to anticipate required automatic actions, monitor automation, and verify necessary recovery actions occurred as expected and required.

These findings were a clear indication of the value of CWA, because this kind of information is very difficult to discover without a systematic analysis of systems, processes, functions, measures and purposes. The impact of automation on future advanced designs, including the importance of data visualization and the practicality of computational performance modeling as a tool to support design, will be further examined in the next phase of this project.

The detailed scenario analyses are included in Appendix B: Operating Scenarios

4.3.4 Operator Performance Shaping Factors

As indicated previously, Worker Competencies Analysis is the last phase of CWA. That part of the analysis would typically include consideration of human performance issues in general, and performance shaping factors (PSFs) in particular. However, in addition to the identification and description of technical and operational characteristics for the selected scenarios, it quickly became apparent that human performance considerations need to be included at least at a high level during this phase of the WDA. This would support the development of a baseline for the further analysis of AdvSMR designs.

PSF analysis is an important aspect of any human reliability analysis (HRA) method and is applicable to WDA and the identification and determination of sound operational concepts for most any environment including AdvSMRs. Research conducted by Hallbert and Kolaczkowski (2007) demonstrated an approach to address the effect of PSFs in human performance by obtaining empirical information about performance shaping and contextual factors. Advanced statistical techniques (such as linear modeling and factor analysis) were then used to find relationships between PSFs and to determine if PSFs are predictive of important aspects of operator performance. In particular, one of the key objectives was to determine if the PSF responses obtained could be used to identify and characterize systematic performance interactions. Since EBR-II is no longer operational, and operational experience reviews for human factors were not performed during its operational lifetime, we implemented PSF analysis based on expert opinion of subject matter experts and HRA and human factors expertise.

This work is considered an important preparation for the remaining WDA analysis for AdvSMRs. The results of this analysis are described in Appendix D: EBR-II Performance Shaping Factors and a brief summary of the PSFs for the selected scenarios are included as Item 15 (“Potential Performance Shaping Factors”) in the Scenario summaries in Appendix B: Operating Scenarios.

4.3.5 WDA products

Interviews with previous EBR-II operators were conducted to determine major functions associated with EBR-II. The results were integrated with the review of plant schematics, emergency procedures, and detailed normal operational procedures. The AH and the contextual activity analysis performed for normal operations identified the following high-level functions that must be accomplished in normal operations:

1. Drive the turbo generator (function to convert mechanical energy to electrical energy)
2. Maintain fast fission (functions required to convert potential energy to nuclear energy)
3. Maintain reactor cooling (functions to utilize sodium coolant to remove and transfer? reactor heat)
4. Manage and control plant operations (functions to manage the allocation of plant functions to operators and control systems)

Based upon an analysis of the plant’s physical architecture, a set of high-level AH diagrams, abstraction-decomposition frameworks, and contextual activity templates were developed for the selected normal and abnormal operating scenarios. The diagrams included in this report show the abstraction (that is, described bottom-up in decreasing levels of detail) of the EBR-II work domain in terms of physical objects (systems and components) at the lowest level, object related processes, purpose-related functions, values and priority measures, and functional purpose at the highest level. It also shows the “why-what-how” means-ends links described in an earlier report.

The diagrams developed include the following:

1. AH for EBR-II Normal Operations.

2. Contextual Activity diagram for Anticipated Operating Occurrences.
3. AHs, Contextual Activities, and Strategies Analysis for the four scenarios described in Appendix B: Operating Scenarios.

See Figure 12 through Figure 28 in Appendix C: WDA Diagrams and Discussion

4.3.6 Abstraction-Decomposition Framework and System Breakdown Structure

When all the elements described above are broken down (decomposed) in terms of Total System (also called “Structures”), Subsystem and Components, the AH and Abstraction-Decomposition Framework (in some literature also called the ‘abstraction-decomposition space”) is produced.

Figure 11 below illustrates a small section of the three top levels of the plant system decomposition. The remainder of the decomposition is omitted from this report because it was found to add only limited value. All relevant systems were however included in the analysis and those systems form the bottom level (“Physical Components”) of the abstraction hierarchies included in this report.

EBR-II System Decomposition

Level 1 Systems
- *Level 2 Systems*
  - *Level 3 Systems*

- **Reactor**
  - Control and Safety Rod Drive Mechanisms
    - Control Rod Gripper Mechanism, Reactor Vessel Cover Assembly, Control Rods, Safety Rods, Control Rod Drive Mechanism
- **Primary Sodium Cooling System**
  - Shutdown Cooling System
    - Primary Na-NaK Hx, NaK-Air Hx, Louvers, Flow Meter
  - Radioactive Sodium Chemistry Loop (RSCL)
    - Flow Meter, RSCL Pump, Expansion Compensator, Control Valves, Trace Heating Systems, Siphon Break, Leak Detection System, Smoke Detection System, Sodium-24 Monitor, RSCL Met-L-X System
  - Sodium Recirculation System
    - Sodium Surge Tank, Sodium Recirc Pumps, Sodium Storage Tank, AC Linear Induction Pump, Charging Tank, Cold Trap, Dowtherm Pumps, Dowtherm Expansion Tank, Dowtherm-to -water Hx
  - Sodium Dump System
    - Sodium Dump Tank, Sodium Storage Tank
- **Secondary Sodium System**
  - Secondary Sodium Purification System
    - Economizer, Cold Trap, Dowtherm Pumps, Dowtherm Expansion Tank, Dowtherm-to -water Hx
  - Secondary Argon System
    - Flow Meter, Liquid Argon Supply Tank, Filter, Pressure Regulators
  - Sodium Pressure Relief System
    - Pressure Relief Tank, High and Low Pressure Rupture Discs, Sodium Collection Tank
  - Secondary Argon Blanket Gas System
    - Liquid Argon Supply Tank, Pressure and Flow Control Valves
- **Steam Plant System**
  - Steam Generation Plant
    - Steam Drum, Evaporators, Superheaters
  - Main Steam System
    - Main Turbine, Dearating Feedwater Heater, Feedwater Heaters, Main Air Ejectors, Startup Hogging Air Ejector, Aux Steam System, Desuperheater and Steam Bypass System
  - Steam Generator Water Dump System
    - Catch Tank, Dump Valve
  - Steam Drum Blowdown System

Figure 11: EBR-II System Decomposition (Extract)

5 Basis for Future Operational Concepts

The identification and description of operating scenarios is an invaluable aspect of OpsCon development. There are many aspects of future plant configurations that will have an impact on operations. These include, for example, the effect of core design, different coolants, reactor-to-power conversion unit ratios, modular plant layout, modular versus central control rooms, plant siting, and many more. Multi-modular plants in particular are expected to have a significant impact on overall OpsCon in general, and human performance in particular. For example, in one case with advanced automation, a single operator may simultaneously monitor and control multiple modules. In another case, multiple operators may be required to manage multiple modules from the same control room. To support these modes of operation, the modern control room of a multi-module plant would typically require advanced HSIs that would provide sophisticated operational information visualization, coupled with adaptive automation schemes and operator support systems to reduce complexity. These all have to be mapped at some point to human performance requirements.

Several other factors need to be considered in the operational design of multi-modular plants:

- **Operating states -** Different modules in a multi-modular plant can be at different power levels or different states, such as shutdown, startup, transients, accidents, refueling and various conditions of maintenance and testing.
- **Operational Coordination of multiple units** – Future designs will place emphasis on passive safety and on coordination of multiple modules, each of which may be in a different operational state, to meet demands of single grid or energy system.
- **Human-Automation Collaboration** – New approaches are needed to integrate personnel and automation to maximize productive safe operations of AdvSMRs. (This is the topic of the current Human-Automation Collaboration project).
- **Control systems for multiple units -** Control systems may need to be able to manage multiple units in an integrated fashion. This could include systems that the modules share in common, such as for secondary plant cooling water or the ultimate heat sink for removing decay heat, and also systems for instrument air, service-water cooling and electrical distribution. It may also include common control of systems that are similar but not shared between modules, such as balance-of-plant systems. The integrated control of multiple modules and their shared systems would thus be not only an automation challenge, but also an operational challenge. The demand placed on operators to maintain situation awareness, not only of the plant overall, but of individual units and shared systems, may impose a severe cognitive workload. In fact, without evidence to the contrary, this may challenge current assumptions that workload in AdvSMR control rooms will be lower, thus theoretically allowing operators to handle more than one module at a time.
- **HSI for multiple units -** The detailed design of HSIs (alarms, displays, and controls) to enable a single operator to effectively manage one or more modules will be an additional challenge. HSIs must enable monitoring the overall status of multiple units (or simply multiple reactors), as well as easy retrieval of detailed information on an individual reactor module.

Table 3 below provides a summary of operational concepts for three types of reactors that will be investigated in the next phase of this project, with specific reference to the lessons learned from the EBR-II analysis. Note however that the scope of the continuing research will be determined by the availability of information on lead-bismuth reactors.

Table 3: Advanced SMR Operating Concept Summary

Reactor Type	Fuel	Operation (Reactivity control, Coolant, Thermal/Electrical Generation	Innovations (Safety features, Waste reduction, Non-proliferation)	Operating Experience and commercial outlook	Advantages	Disadvantages	Operations and Human Factors Impact
Liquid Metal (Integral Fast Reactor) (NaK or Na)	• Uranium alloy core with liquid metal coolant • Metallic fuel, stainless steel cladding	• Liquid metal (NaK - sodium/potassium) or other liquid metal coolant such as lead-bismuth (Pb-Bi) • Integral heat exchanger • Heat transferred to secondary salt circuit by intermediate heat exchanger and then to steam by steam generator • Low pressure, high temperature	• Automatic power regulation is achieved due to negative reactivity feedback	• Argonne National Laboratory has significant experience - EBR I & II, Fast Flux Test Facility (FFTF) demonstrated passive shutdown • Reprocessing techniques have been demonstrated • Considered by Japan, China, Korea, Russia (others?) • 50 MWe version under development	• Modular off-site construction • Allow higher power density • Possible long term cooling possible - impose design requirements • Pool or loop configurations available • Fast reactor, small size • Very low coolant corrosion of steel vessels and piping • Fuel reprocessing possible - ability to burn long-lived actinides • Could use U→Pu or Th cycle • Refueling interval of ~20 years	• Violent reaction of Na and K with water - produces hydrogen (explosion hazard), neutron activation of sodium • Exothermic reaction on contact of Na with air (fire hazard) • Must heat piping for metal coolant to remain liquid	• Lower staffing needs, due to level of automation, long refueling cycle

Fast Liquid Metal Reactor (Pb-Bi eutectic)	• 15-20% enriched and possibly uranium nitride - UN, (U,Pu)N, (U,transuranic) N, U-Zr, or (U,Pu)Zr • Could use U→Pu or Th cycle • No moderator	• Pb-Bi eutectic alloy coolant • Heat transferred to secondary salt circuit by intermediate heat exchanger and then to steam by steam generator • Operate at near-atmospheric pressure • High neutron flux • Pool or loop configurations available	• Installed below ground	• Russian Alfa submarines in 1960s - several sank, but Russians are pursuing it again	• Automatic power regulation due to reactivity feedback (loss of coolant flow leads to higher core temperature which slows the reaction) • Passive safety features • Facilities available at universities • Could use U→Pu or Th cycle • Process heat available for cogeneration, desalination	• Expensive • Steam generator tube leaks produce insoluble lead oxides - can plug cores, unless intermediate heat exchanger is used • Long-lived Pb isotopes complicate decommissioning	•
Molten Salt Reactor	• Molten mixture of lithium and beryllium fluoride salts • Core is unclad graphite moderator • Could use U→Pu or Th cycle • Fission products dissolve in the salt and are removed continuously in an on-line reprocessing loop and replaced with Th-232 or U-	• Heat transferred to secondary salt circuit by intermediate heat exchanger and then to steam by steam generator	• Multiple units per control room - experience on LWRs should help • Reduced proliferation concerns	• Small prototype operated at Oak Ridge. Early studies (1950 - 1960) were encouraging. Early design had fuel circulating through RCS – (unthinkable nowadays) • Renewed interest in Japan, Russia, France, USA	• Low pressure primary coolant, high temperature - reduces cost, weight & size of pressure vessel & piping • High temperature reduces cooling water requirements • Fluoride salts relatively friendly to steel • Passively cooled design - salt freezes at ~480°C	• Expensive • If TRISO fuel is used - extremely difficult to reprocess	•

	238.				• High temperatures reduces waste heat • Desalination possible from waste heat		

6 Conclusion

This report described the results of the process for identifying the structural and functional characteristics and the human performance requirements in AdvSMR sodium reactor designs. Because no multi-unit AdvSMR designs have been commissioned to date, only limited information was available for review of their design features and operating experience. This phase of the project has therefore relied on subject matter experts, including former EBR-II operators, open source literature, operating procedures from EBR-II, and limited information from emerging advanced FSR designs such as Toshiba's 4S, GE-Hitachi's PRISM, other emerging AdvSMR designs such as TerraPower, and the SMRs designs currently under construction in India. It is expected that more information will become available in the near future and could then be included in the detailed analyses.

Thanks to the availability of EBR-II information, important progress was made. In particular, the research in this phase considered some instances where systems did not function as intended by designers, and also instances where system design allowed certain work-arounds. Using design basis events and procedures from the EBR-II archives, we have documented the key operator responsibilities at EBR-II for those operating scenarios and demonstrated how this would be extrapolated to future AdvSMR designs with higher degrees of automation and advanced HSIs.

In reviewing the normal and abnormal/emergency operating scenarios and developing the WDA described above, our interim findings for AdvSMR designs compared to EBR-II include the following:

a) New concepts in physical and functional plant layout lead to dramatic changes in operational and maintenance procedures. For example, different plant configurations such as multiple power conversion units per reactor will require special attention to I&C and HSI design, as well as unconventional operating procedures. Compact plant footprints will make it difficult for field operators and technicians to reach certain areas; this will require increased attention to remote monitoring, surveillance, diagnostics and control.
b) The biggest change in the role of the operator will be a shift from many manual tasks to monitoring and supervising highly automated systems. Automation systems will allow operators to manually intervene in automated process in many cases, but this will be the exception to the rule. It is expected that an adaptive automation system may even monitor the operator's performance and take over when a potential error is detected.
c) Load following will involve reduced manual tasks, but increased communication with grid operators;
d) Different modules may be in different operational states at the same time, which will also require special HSIs and procedures.
e) The need to manage different product streams (i.e. electrical power and process heat) will have design, licensing, and procedural implications that are still unclear and require additional study.

The results to date confirm that CWA has value in creating a structured approach to OpsCon design that accounts for human performance, even for FOAK systems such as AdvSMRs. It must be emphasized that CWA is not only a way to facilitate systems engineering activities by showing how to incorporate cognitive work in systems engineering. CWA also can aid project managers by describing the systematic integration of human functions in the systems engineering process. Above all, an integrated CWA approach helps the design team understand the human requirements of work and how technology may help or hinder operators, technicians, engineers and managers in meeting those requirements. CWA practitioners facilitate design discussions by describing the impact of various design choices on the execution of human work. No one else on the team has this capability or responsibility. The designers are usually charged with shaping the system itself and thus do not always appreciate how users will interact with the system. The appropriate use of CWA in the design life cycle reduces the risk of additional iterations, project cancellations or rejected deliverables, reduces the time associated with trial and error approaches. Successful integration of CWA into existing project activities such as modeling, simulation, and prototyping can mitigate rising project costs. CWA supports the design process at different levels and also support licensing (review time, iterations, and ultimate acceptance).

In practice, WDA, as the dominant component of CWA, can happen concurrently with automation system design, and in an ideal world there will be a lot of iteration, coordination and integration between the processes. Although the conduct of WDA can be considered to be consistent with the spirit of NUREG-0711, it has never before been applied (with the exception of small-scale academic studies, e.g. Bisantz and Vicente, 1994; Jamieson et al., 2007) in the development or analysis of concepts of operation in the nuclear industry. Examples in the literature (Roth, Patterson, & Mumaw, 2012; Bisantz & Vicente, 1994) suggest that WDA is the most systematic and structured method for this purpose. With time, we expect its use in the nuclear industry to become more widespread.

The WDA results achieved during this phase have not only established an organizing and analytical framework for describing existing sociotechnical systems, but have also indicated that the method is particularly suited to the analysis of prospective and immature designs. At this stage is generally assumed that there are sufficient similarities between EBR-II and prospective sodium-cooled reactor designs (for example, same basic reactor design, same coolant and therefore similar basic thermohydraulic processes), to justify the use of the baseline WDA for analysis of more advanced designs. However, the researchers are aware that the differences between EBR-II and future AdvSMRs should not be underestimated. These differences would essentially be due to new materials and new components, but especially due to advanced automation systems, digital I&C, and advanced HSI, different plant configurations, and even different, concurrent product streams. Nevertheless, the EBR-II WDA has indicated that the methodology is scientifically sound and generalizable to any operating environment. This means that it will be easy to identify the differences and exceptions and adapt the existing analysis to a new design.

During the next phase some assumptions will be made about automation, modular design, concurrent processes, and more. Data from other industries for operator response to failures in automation will be reviewed in future with the intent of being able to predict operator response to failed automation for sodium-cooled SMRs. For example, the expected dynamic nature of the interaction between humans and systems in future plants will be a direct result of the design and architecture of distributed control systems, but will also be influenced by advanced design concepts resulting from new materials, multiple product streams, modular plant layout, and more. A large part of automation system design will be beyond the influence of human factors considerations. The reasons for this will be found in the reliability, accuracy and controllability requirements of certain physical processes. It is the purpose of the WDA to also identify those functions that are clearly beyond human capability. We will also look at how different approaches to OpsCon and specifically human-automation interaction concepts, can lead to different expectancies regarding operator performance and ultimately the impact of operations and maintenance. In going from an older design to a new FOAK design, the Contextual Activity Analysis part of the WDA outlined by Naikar (2013) will prove to be particularly valuable in helping to identify and characterize expected differences in responsibilities, roles, and automation likely to be present for normal and abnormal operations. Key assumptions, such as modularity, plant layout, and higher levels of automation will be included in the analysis. For a new AdvSMR design, information like this, combined with human performance criteria, could also be used in forthcoming work to assess the crew performance aspects associated with identified AdvSMR operational concepts.

The EBR-II baseline WDA includes detailed descriptions of operating scenarios (derived from the Contextual Activity and Strategies Analysis), and state matrices for the systems identified from the abstraction-decomposition for those scenarios. This baseline WDA (and further phases of CWA) will produce input for the extended analysis of a reference AdvSMR design. Once the WDA and the rest of the CWA for the reference design is complete, there will be sufficient information to test the framework's process steps and methods, which will allow power plant designers to develop operational concepts with greater confidence.

7 Further Work Planned for FY 2014

The continuing work for FY14 would consist of the following two activities:

7.1 Human Performance Requirements for AdvSMR Designs

August 2014: *Develop application guidance for Human Performance considerations for AdvSMRs.*

As described in a previous milestone report, human performance requirements are complementary to the WDA process and reflect function allocations that were assigned by system designers and implementers.

This task will include, where necessary, an update of the September 2013 information on FA and Human Performance and will focus on the application of the concepts to advanced designs.

7.2 Refinement and Extension of Work Domain Analysis for an Advanced Sodium-Cooled Reactor Design

March 2015: *Develop a complete WDA for a selected or generic FSR design.*

During FY 2014 this research project will focus on the extrapolation of the baseline WDA and other relevant phases of CWA to advanced designs. The role of the operator in relation to the operational requirements of multiple modules in the presence of higher levels of automation will be a particular emphasis. The researchers will also identify ways in which changes in control philosophy will lead to differences in AdvSMR operator performance requirements.

8 References

[1] Bailey, R. W. (1982). *Human performance engineering*. Englewood Cliffs, NJ: Prentice Hall.

[2] Bennett, K. and Flach, J. (2012). *Display and Interface Design: Subtle Science, Exact Art*. CRC Press, Taylor & Francis: New York

[3] Bisantz, A. M., and Vicente, K. J. (1994). Making the abstraction hierarchy concrete. *International Journal of Human-Computer Studies*, *40*, 83-117.

[4] Hallbert, B. and Kolaczkowski , A. (2007). Modeling Causal Mechanisms and Context in HRA, in *The Employment of Empirical Data and Bayesian Methods in Human Reliability Analysis: A Feasibility Study, NUREG/CR-6949*, U.S. NRC

[5] Hassall, M. and Sanderson, P. (2012). A formative approach to the Strategies Analysis phase of Cognitive Work Analysis. *Theoretical Issues in Ergonomics Science*. Boca Raton, FL: Taylor &

[6] Hugo, J., Boring, R., Forester, J., Gertman, D., Joe, J, Medema, H., Persensky, J., and Whaley, A. (2012). *Draft Framework for Advanced SMR Concepts of Operations Research*. (INL/EXT-12-27299/SMR/ICHMI/INL/TR-12/01). Idaho Falls, ID: Idaho National Laboratory.

[7] Hugo, J., Forester, J., Gertman, D., Joe, J., Medema, H., Persensky, J., and Whaley, A. (2013a). *Draft Function Allocation Framework and Preliminary Technical Basis for Advanced SMR Concept of Operations*. (INL/EXT-13-28601/SMR/ICHMI/INL/TR-2013/02). Idaho Falls, ID: Idaho National Laboratory.

[8] Hugo, J., Gertman, D., Joe, J., Medema, H., Whaley, A. and Farris, R. (2013b). Development of a Technical Basis for Advanced SMR Function Allocation. (INL/EXT-13-30117/SMR/ICHMI/INL/TR-2013/04). Idaho Falls, ID: Idaho National Laboratory.

[9] Hugo. J. (2012). Towards a Unified HFE Process for the Nuclear Industry. *8th International Topical Meeting on Nuclear Plant Instrumentation, Control and Human Machine Interface Technologies, NPIC&HMIT 2012*, San Diego, CA, July 22- 26.

[10] Jenkins D.P., Stanton N.A, Salmon, P.M., Walker, G.H., and Young, M.S. (2008) Using Cognitive Work Analysis to explore system flexibility. *Theoretical Issues in Ergonomics Science.* Volume 11, Issue 3, 2010. Boca Raton, FL: Taylor & Francis.

[11] Naikar, N., Hopcroft, R. and Moylan, A. (2005). *Work Domain Analysis: Theoretical Concepts and Methodology*. Air Operations Division. Defence Science and Technology Organisation, Australia.

[12] Naikar. N. (2013) *Work Domain Analysis: Concepts, Guidelines and Cases*. Boca Raton:CRC Press.

[13] NUREG-0711, Revision 3 (2012). O'Hara, J., Higgins, J.C., Fleger, S.A and Pieringer, P.A. *Human Factors Engineering Program Review Model*, US Nuclear Regulatory Commission.

[14] O'Hara J., Higgins, J. and Pena M. (2011) *Human-Performance Issues Related to the Design and Operation of Small Modular Reactors* (NUREG/CR-7126). Washington DC: U.S. Nuclear Regulatory Commission.

[15] O'Hara, J., Higgins, J, and D'Agostino, A. (2012). Multi-unit Operations in Non-nuclear Systems: Lessons Learned for Small Modular Reactors. *Proceedings of the Eighth American Nuclear Society International Topical Meeting on Nuclear Plant Instrumentation, Control and Human-Machine Interface Technologies NPIC&HMIT 2012*, San Diego, CA, July 22–26, 2012.

[16] Rasmussen, J. (1974*). The human data processor as a system component: Bits and pieces of a model* (Report No. Risø-M-1722). Roskilde, Denmark: Danish Atomic Energy Commission.

[17] Rasmussen, J., and Goodstein, L. P. (1987). Decision support in supervisory control of high-risk industrial systems. Automatica, 23(5), 663-671.

[18] Roth, E. M., Patterson, E.S. and Mumaw, R. J. (2002). Cognitive Engineering: Issues in User-Centered System Design. In *J. J. Marciniak (Ed.), Encyclopedia of Software Engineering, 2nd Edition,* 163-179. New York: Wiley-Interscience, John Wiley and Sons.

[19] Sanderson, P., Naikar, N., Lintern, G. and Goss, S. (2012). Use of Cognitive Work Analysis Across the System Life Cycle: From Requirements to Decommissioning. *Proceedings of the Human Factors and Ergonomics Society Annual Meeting*. September 2012, 56(1).

[20] Stanton, N.A. and Bessell, K. (2013). How a submarine returns to periscope depth: Analysing complex socio-technical systems using Cognitive Work Analysis. *Applied Ergonomics* 45 (2014) 110-125.

[21] Vicente, K. J. (1999). *Cognitive Work Analysis: Toward Safe, Productive, and Healthy Computer-Based Work*. Mahwah, NJ: Lawrence Erlbaum Associates, Inc.

[22] Woods, D. D., and Hollnagel, E. (2006). *Joint cognitive systems: Patterns in cognitive systems engineering*. Boca Raton, FL: CRC Press.

9 APPENDICES

9.1 Appendix A: EBR-II State Matrices

The state transition diagram shown in Figure 10 was elaborated in a state matrix for normal (Table 4) and abnormal (fault modes - Table 5) operating conditions. The State Matrices for EBR-II shown on the following pages first appeared in the September 2013 milestone report. The information for primary systems and normal as well as abnormal conditions was reviewed and verified for this phase and included here for reference purposes.

These matrices describe the operating conditions for the following major systems:

- Reactor cooling system
- Control rods and safety rods
- Secondary sodium system
- Emergency shutdown coolers
- Turbine generator
- Condensate and feedwater system
- Main steam system
- Evaporators and superheaters
- Fuel handling and associated equipment
- Electric plant

Table 4: EBR-II State Matrix for Primary Systems and Normal Operations

EBR-II - Normal Operations State Matrix

System/ Mode	**Unrestricted Fuel Handling**	**Plant (Cold) Shutdown**	**Plant (Heatup) Startup**	**Plant (Hot) Standby**	**Power Operation (Steady Power)**	**Restricted Fuel Handling**
	(This is typical fuel shuffling with the reactor shutdown)	**(Typical plant shutdown for maintenance)**		**(A condition of readiness – ready to go to power)**		**(Steady Power Operations – Allows for subassemblies to be removed from/placed into the fuel basket only)**
Reactor Cooling System	RCPs: S/D Aux Pump: S/D Primary Flow: 0 gpm Bulk Na Temp: Core cooling: N/A Rx Power: 0% Cover Gas: 6-7 psig Primary Tank Heaters:	RCPs: S/D Aux Pump: S/D Primary Flow: 0 gpm Bulk Na Temp: 350-695 F Core cooling: S/D Coolers Rx Power: 0% Cover Gas: 6-7 psig Primary Tank Heaters:	RCPs: 32 – 98% Flow Aux Pump: 100% (575 gpm) Primary Flow: 100% (9000 gpm) Bulk Na Temp: 350-695 F Delta Temp: 2 F/min Rate (max) Core cooling: IHX Rx Power: 0-100% Cover Gas: 6-7 psig	RCPs: 100 % Aux Pump: 100% (575 gpm) Primary Flow: 100% (9000 gpm) Bulk Na Temp: 695 – 705 F Delta Temp: 183 F Core cooling: Rx Power: 0% Cover Gas: 6-7 psig	RCPs: 100% Flow Aux Pump: 100% (575 gpm) Primary Flow: 100 % (9000 gpm) Bulk Na Temp: 695 – 705 F Delta Temp: 183 F Core cooling: IHX Rx Power: 100% (62.5 MWt) Cover Gas: 6-7 psig	RCPs: 100% Flow Aux Pump: 100% (575 gpm) Primary Flow: 100 % (9000 gpm) Bulk Na Temp: 695 – 705 F Delta Temp: 183 F Core cooling: IHX Rx Power: 100% (62.5 MWt) Cover Gas: 6-7 psig

EBR-II - Normal Operations State Matrix

System/ Mode	Unrestricted Fuel Handling	Plant (Cold) Shutdown	Plant (Heatup) Startup	Plant (Hot) Standby	Power Operation (Steady Power)	Restricted Fuel Handling
	Energized	Energized	Primary Tank Heaters: Deenergized at 700 F Bulk Na Temp Limitations: RCS H/U Rate 10 F/Hr Max	Primary Tank Heaters: Energized – Variable output based on Bulk Na Temp	Primary Tank Heaters: Deenergized	Primary Tank Heaters: Deenergized
Secondary (Sec.) Sodium (Na) System	Sec. Na Pump: S/D Sec. Na Temp: N/A Press: N/A Argon Press: N/A	Sec. Na Pump: S/D Sec. Na Temp: N/A Press: N/A Argon Press: N/A Note: If system is drained for maintenance both Primary and Secondary Na temp should be less than 350 F	Sec. Na Pump: Variable gpm to maintain bulk Na temp 695-705 F Argon Press: 6-7 psig	Sec. Na Pump: Variable gpm to maintain bulk Na temp 695-705 F Sec. Na Temp to Steam Drum: 866 F Sec. Na Temp From Steam Drum: 588 F Argon Press: 6-7 psig	Sec. Na Pump: 86 % / 5160 gpm Sec. Na Temp to Steam Drum: 866 F Sec. Na Temp From Steam Drum: 590 F Argon Press: 6-7 psig	Sec. Na Pump: 86 % / 5160 gpm Sec. Na Temp to Steam Drum: 866 F Sec. Na Temp From Steam Drum: 590 F Argon Press: 6-7 psig
Emergency Shutdown Coolers	Operationally Ready	Operationally Ready	Operationally Ready	Operationally Ready	Operationally Ready	Operationally Ready

EBR-II - Normal Operations State Matrix

System/ Mode	Unrestricted Fuel Handling	Plant (Cold) Shutdown	Plant (Heatup) Startup	Plant (Hot) Standby	Power Operation (Steady Power)	Restricted Fuel Handling
	Note: S/D Coolers can be used to assist plant cool down (dampers open)	Note: S/D Coolers can be used to assist plant cool down (dampers open)				
Turbine Generator	Offline	Offline Note: Turbine Generator will be unloaded at Rx Power 18-20 MWt	Turbine warm-up in progress	Condenser providing steam dump	Turbine Generator: Loaded	Turbine Generator: Loaded
		Condenser providing steam dump			Power Output: 20 MWe	Power Output: 20 MWe
Condensate and Feedwater System	Offline. Typically drained for work on much of condensate and feedwater system	Offline. Typically drained for work on much of condensate and feedwater system	Filled and vented. Perhaps being recirculated by operating condensate pumps to clean piping and water to acceptable purity and low oxygen levels Startup Feed Pump or Emergency Feedwater Pump maintaining steam drum level.	Filled and vented. Being recirculated by operating condensate pump Startup Feed Pump or Emergency Feedwater Pump maintaining steam drum level and circulating yard lines. Feedwater Temp. 320-330 F	Condensate and Feedwater Systems in normal operational modes Hotwell Level: 26-29 in. Feedwater Temp: 550 F	Condensate and Feedwater Systems in normal operational modes Hotwell Level: 26-29 in. Feedwater Temp: 550 F

EBR-II - Normal Operations State Matrix

System/ Mode	Unrestricted Fuel Handling	Plant (Cold) Shutdown	Plant (Heatup) Startup	Plant (Hot) Standby	Power Operation (Steady Power)	Restricted Fuel Handling
					Condensate Pumps: 1 Pump in Run / 1 Stby Feedwater Pumps: 1 Pump in Run / 1 Stby	Condensate Pumps: 1 Pump in Run / 1 Stby Feedwater Pumps: 1 Pump in Run / 1 Stby
Main Steam	Atmospheric Pressure	Atmospheric Pressure	Steam being dumped to condenser	Main steam lines being heated. Steam through Atmospheric Dump Valves or to condenser via turbine bypass system	Steam Pressure: 1250 psig Steam Temp: 820 F	Steam Pressure: 1250 psig Steam Temp: 820 F
Evaporators/ Superheaters	Offline. Typically drained for work on much of condensate and feedwater system.	Offline. Typically drained for work on much of condensate and feedwater system.	Maintain Superheater and Evaporator Shell Temps 0-50 F > Tube Temp	Maintain Superheater and Evaporator Shell Temps 0-50 F > Tube Temp	Fully Operational Na Temp from Evaporators: 590 F Na Temp to Superheaters: 866 F	Fully Operational
Fuel Handling & Associated Equipment	Small / Large Rotating Plugs: Heated to allow rotation Rx Vessel Cover: Up/Unlocked	Reactor Vessel Cover: Depends on work to be performed	Reactor Vessel Cover: Down/Locked	Reactor Vessel Cover: Down/Locked	Reactor Vessel Cover: Down/Locked	Small / Large Rotating Plugs: Frozen Reactor Vessel Cover: Down/Locked

Table 5: EBR-II State Matrix for Primary Systems and Abnormal Conditions

EBR-II - Abnormal Operations State Matrix

System/ Mode	Power Operation (Steady Power) Initial Condition for All Events	Sec. Na – Water to Na Leak	Earthquake (Minor)	Major Na Leak in Rx Outlet Piping	Reactor (Rx) Scram (Manual or Automatic)	Loss Of Normal Electric Power
Initial Indications	N/A	Hydrogen Meter Leak Detector (HMLD) alarm Cover Gas Meter Leak Detector (CHMLD) alarm Increasing Hydrogen (H_2) level Sec. Na High Sec. Na Press Leak Probe Detector alarms Sec. Na Relief Header flow alarm Sec. Na System Rupture Disk alarm SBB Evacuation	All rods on bottom (rod bottom lights lit) Earthquake alarms Full Rx Building Isolation Rx Building Evacuation SBB Evacuation	Increasing Bulk Na Temp Increasing primary tank cover gas temp. Decreasing IHX Sec. Na outlet temp Decreasing Rx Power Decreasing Rx Delta Temp Rx Building Evacuation	All rods on bottom (rod bottom lights lit)	Auto Rx Scram RCPs deenergized T-G Trips Sec. Na Pump Trips Sec. Na Recirc Pumps Trip EDGs auto start Primary and Sec. Heating lost Majority of electric pumps in plant deenergized

EBR-II - Abnormal Operations State Matrix

System/ Mode	Power Operation (Steady Power) Initial Condition for All Events	Sec. Na – Water to Na Leak	Earthquake (Minor)	Major Na Leak in Rx Outlet Piping	Reactor (Rx) Scram (Manual or Automatic)	Loss Of Normal Electric Power
Control Rods and Safety Rods All rods on bottom (rod bottom lights lit)	Normal position for full power operations	Manual Rx Scram	Auto Rx Scram All rods on bottom (rod bottom lights lit) **Note:** Validating rods scrammed and did not bind /stick is extremely important	**Initial:** Bulk Na Temp = 720 F Anticipatory Rx S/D **Final:** If Bulk Na Temp = 725 F Manual Rx Scram	Auto or Manual Rx Scram All rods on bottom (rod bottom lights lit)	Auto Rx Scram All rods on bottom (rod bottom lights lit)
Rx Cooling System	Rx Coolant Pump (RCPs): 100% Flow Aux Pump: 100% (575 gpm) Primary Flow: 100 % (9000 gpm) Bulk Na Temp: 695 – 705 F Delta Temp: 183 F	RCPs: 100% Flow Aux Pump: 100% (575 gpm) Primary Flow: 100 % (9000 gpm) Bulk Na Temp: 695 – 705 F Delta Temp: 183 F and decreasing	RCPs: 100 % Aux Pump: 100% (575 gpm) Primary Flow: 100% (9000 gpm) Bulk Na Temp: 695 – 705 F Delta Temp: 183 F	**Initial:** RCPs: min. flow for 30 min. Aux Pump: 100% (575 gpm) Bulk Na Temp: 695 – 705 F and increasing Delta Temp: 183 F and decreasing	RCPs: 100% Flow Aux Pump: 100% (575 gpm) Primary Flow: 100 % (9000 gpm) Bulk Na Temp: 695 – 705 F Delta Temp: 183 F and decreasing	RCPs: deenergized Aux Pump: 100% (575 gpm) on battery power Primary Flow: Aux Pump only (575 gpm) Bulk Na Temp: 695 – 705 F and increasing Delta Temp: 183 F and rapidly decreasing

EBR-II - Abnormal Operations State Matrix

System/ Mode	Power Operation (Steady Power) Initial Condition for All Events	Sec. Na – Water to Na Leak	Earthquake (Minor)	Major Na Leak in Rx Outlet Piping	Reactor (Rx) Scram (Manual or Automatic)	Loss Of Normal Electric Power
	Core cooling: IHX Rx Power: 100% (62.5 MWt) Cover Gas: 6-7 psig Primary Tank Heaters: Deenergized	Core cooling: IHX Rx Power: 0% Cover Gas: 6-7 psig Primary Tank Heaters: Deenergized	Core cooling: Rx Power: 0% Cover Gas: 6-7 psig Primary Tank Heaters: Energized – Variable output based on Bulk Na Temp	Core cooling: IHX and S/D Coolers Rx Power: 100% (62.5 MWt) and decreasing Cover Gas: 6-7 psig Primary Tank Heaters: Deenergized **Final:** 30 Min. after Rx S/D or Scram secure RCPs	Core cooling: IHX Rx Power: 0% Cover Gas: 6-7 psig Primary Tank Heaters: Deenergized	Core cooling: IHX and S/D Coolers Rx Power: 0% (0 MWt) Cover Gas: 6-7 psig Primary Tank Heaters: Deenergized
Secondary (Sec.) Sodium (Na) System	Sec. Na Pump: 86 % / 5160 gpm Sec. Na Temp to Steam Drum: 866 F Sec. Na Temp From Steam Drum: 590 F	**Initial:** Press Sodium Boiler Building (SBB) Fire Push Button (P/B) Fire P/B cause trips Sec. Na Pump and Recirc. Pumps	Sec. Na Pump: Variable gpm to maintain bulk Na temp 695-705 F Sec. Na Temp to Steam Drum: 866 F Sec. Na Temp From Steam Drum: 588 F	**Initial:** Sec. Na Pump: Pump tripped – off (Scram) **Final:**	**Initial:** Sec. Na Pump: Pump tripped – off **Final:**	**Steam Drum Level Indicated:** Sec. Na Pump: 0.1-0.3 % on Alternate Pwr Argon Press: 6-7 psig

EBR-II - Abnormal Operations State Matrix

System/ Mode	Power Operation (Steady Power) Initial Condition for All Events	Sec. Na – Water to Na Leak	Earthquake (Minor)	Major Na Leak in Rx Outlet Piping	Reactor (Rx) Scram (Manual or Automatic)	Loss Of Normal Electric Power
	Argon Press: 6-7 psig	Sec. Na Pump: Deenergized (not restarted) **Final:** Dump Sec. Na to Sec. Na Drain Tank and allow to cool to ambient	Argon Press: 6-7 psig	Restart Sec. Na Pump (alternate power) when Bulk Na Temp = 690 F Sec. Na Pump Variable gpm to maintain bulk Na temp 695-705 F Argon Press: 6-7 psig	Restart Sec. Na Pump (alternate power) when Bulk Na Temp = 690 F Sec. Na Pump Variable gpm to maintain bulk Na temp 695-705 F Argon Press: 6-7 psig	**Steam Drum Level <u>Not</u> Indicated:** Sec. Na Pump: Deenergized Sec. Argon Sys: Vented Dump Sec. Na to Sec. Na Drain Tank and allowed to cool to ambient
Emergency Shutdown (S/D) Coolers	Operationally Ready	Operationally Ready	Operationally Ready	S/D Cooler Louvers Open when Bulk Na Temp = 710 F	Operationally Ready	S/D Cooler Louvers Open when Bulk Na Temp = 710 F

EBR-II - Abnormal Operations State Matrix

System/ Mode	Power Operation (Steady Power) Initial Condition for All Events	Sec. Na – Water to Na Leak	Earthquake (Minor)	Major Na Leak in Rx Outlet Piping	Reactor (Rx) Scram (Manual or Automatic)	Loss Of Normal Electric Power
						Shut S/D Louvers if Bulk Na Temp < 690 F
Turbine Generator (T-G)	T-G: Loaded Power Output: 20 MWe	T-G: Tripped Power Output: 0 MWe	Condenser providing steam dump	T-G: Tripped Power Output: 0 MWe	T-G: Tripped Power Output: 0 MWe	T-G: Tripped Power Output: 0 MWe
Condensate and Feedwater System	Condensate and Feedwater Systems in normal operational modes Hotwell Level: 26-29 in. Feedwater Temp: 550 F	FWPs: Deenergized Condensate and Feedwater System being S/D	Filled and vented. Being recirculated by operating condensate pump Startup Feed Pump or Emergency Feedwater Pump maintaining steam drum level and circulating yard lines.	**Initial:** Feedwater System in manual control to maintain steam drum level 30 inches Condensate Pumps: 1 Run / 1 Stby Feedwater Pumps (FWP): 1 Run / 1 Stby	**Initial:** Feedwater System in manual control to maintain steam drum level 30 inches Condensate Pumps: 1 Run / 1 Stby Feedwater Pumps (FWP): 1 Run / 1 Stby	**Steam Drum Level Indicated:** Emergency Feedwater Pump maintaining steam drum level and circulating yard lines. **Steam Drum Level Not Indicated:**

EBR-II - Abnormal Operations State Matrix

System/ Mode	**Power Operation (Steady Power)** **Initial Condition for All Events**	**Sec. Na – Water to Na Leak**	**Earthquake (Minor)**	**Major Na Leak in Rx Outlet Piping**	**Reactor (Rx) Scram** **(Manual or Automatic)**	**Loss Of Normal Electric Power**
	Condensate Pumps: 1 Run / 1 Stby Feedwater Pumps: 1 Run / 1 Stby		Feedwater Temp. 320-330 F	**Final:** Condensate Pumps: 1 Run / 1 Stby FWPs: Secured	**Final:** Condensate Pumps: 1 Run / 1 Stby FWPs: Secured	Emergency Feedwater Pump: deenergized
Main Steam	Steam Pressure: 1250 psig Steam Temp: 820 F	Steam drum isolated on FW and Steam side Steam Dump Valve actuated: immediately drains Steam Drum to Steam Generator Water Dump System (SGWDS)	Main steam lines being heated. Steam through Atmospheric Dump Valves or to condenser via turbine bypass system	Steam Pressure: 1250 psig and rapidly lowering Steam Temp: 820 F and rapidly lowering	Steam Pressure: 1250 psig and rapidly lowering Steam Temp: 820 F and rapidly lowering	**Steam Drum Level Indicated:** Main Steam Stop shut **Steam Drum Level <u>Not</u> Indicated:** Steam drum isolated on FW and Steam side

EBR-II - Abnormal Operations State Matrix

System/ Mode	Power Operation (Steady Power) Initial Condition for All Events	Sec. Na – Water to Na Leak	Earthquake (Minor)	Major Na Leak in Rx Outlet Piping	Reactor (Rx) Scram (Manual or Automatic)	Loss Of Normal Electric Power
	Generator Power: 20 MWe	Fill Steam Drum (water side) with Argon gas				Steam Dump Valve actuated: immediately drains Steam Drum to Steam Generator Water Dump System (SGWDS)
Evaporators/	Fully Operational	Rapid drain on Na and Water sides	Maintain Superheater and Evaporator Shell Temps 0-50 F > Tube Temp	Fully Operational	Fully Operational	**Steam Drum Level <u>Not</u> Indicated:** Rapid drain on Na and Water sides
Superheaters	Na Temp from Evaporators: 590 F Na Temp to Superheaters: 866 F	Cooling towards ambient temperature		Na Temp from Evaporators: 590 F and lowering Na Temp to Superheaters: 866 F and lowering	Na Temp from Evaporators: 590 F and lowering Na Temp to Superheaters: 866 F and lowering	Cooling towards ambient temperature Dry layup
Fuel Handling & Associated Equipment	Rx Vessel Cover: Down/Locked Large & Small Plug seals frozen	Rx Vessel Cover: Down/Locked Large & Small Plug seals frozen	Rx Vessel Cover: Down/Locked Large & Small Plug seals frozen	Rx Vessel Cover: Down/Locked Large & Small Plug seals frozen	Rx Vessel Cover: Down/Locked Large & Small Plug seals frozen	Rx Vessel Cover: Down/Locked Large & Small Plug seal heaters power supply switched to emergency power

EBR-II - Abnormal Operations State Matrix

System/ Mode	Power Operation (Steady Power) Initial Condition for All Events	Sec. Na – Water to Na Leak	Earthquake (Minor)	Major Na Leak in Rx Outlet Piping	Reactor (Rx) Scram (Manual or Automatic)	Loss Of Normal Electric Power
Electric Plant	Emergency Diesel Generators (EDGs): In Stby	EDGs: In Stby	EDGs: In Stby	EDGs: In Stby	EDGs: In Stby	EDGs: In Run and loaded supplying emergency electrical distribution panels

9.2 Appendix B: Operating Scenarios

9.2.1 Secondary Na System: Water to Na Leak (Emergency Procedure EP 3-8)

Item	Item Name	Item Description
1	**Scenario ID**	**Event 1**
2	**Name**	**Secondary Na System: Water to Na Leak (EP 3-8)**
3	**Type**	Transient/Fault mode
4	**Scenario Description**	The steps taken for a major sodium leak are taken into consideration for personnel safety and equipment protection. The use of the Fire Pushbutton and Steam Generator Water Dump causes rapid dumping of the feedwater, steam, and secondary sodium systems. Draining the secondary sodium system accomplishes the following: 1) removes the source of the sodium leaking 2) reduces further water-sodium reactions. Depressing the Steam Generator Water Dump causes rapid removal of water and steam from the steam generation system further reducing the possibility of severe perturbation. De-energizing the induction and resistance heating will allow the evaporator and superheater shells to cool down with minimum thermal stress. The Fire Pushbutton is actuated outside the SBB to protect personnel in the event of an explosion from severe water-sodium reaction. When a water leak occurs in the steam one of the evaporators or superheaters - alarms will be received on the Hydrogen Meter Leak Detector (HMLD) or Secondary Cover Gas Hydrogen Leak Detector (CHMLD) hydrogen level or rate-of-rise data on the plant computer. This will cause trend data for hydrogen to rise. All operating leak detectors should show upward trends in 5 minutes at full power operations. Other actions to validate actual water-to-sodium leak are: 1) Verify plant computer is functioning properly 2) Observe the HMLD and CHMLD membrane temperature and flow data for any unusual upsets 3) Validate no system perturbations have occurred such as reactor startup or changes in secondary cold trap operations that could account for an increase in secondary hydrogen level 4) Perform equilibrium pressure testing of HMLDs 5) Start a plugging run 6) Inspect secondary chromatographs charts for increase in hydrogen and perform a span-gas check to verify hydrogen readers if needed 7) Check secondary cover gas pressure reading, pressure may increase if water-to-sodium leak is large enough. A water-to-sodium leak results in a reaction, which generates heat and liberates hydrogen, causing rapidly increasing temperature and pressure in the affected evaporator or superheater. The magnitude of the temperature or pressure increase depends on the size of the leak. Localized high temperature and pressure has the potential to create failure of the affected evaporator or superheater. A number of these steps may be performed in parallel. To achieve these 7 steps above assumes a crew of seven personnel as detailed

Item	Item Name	Item Description
		in the action flow described in the following paragraphs. Initial operating crew response to the alarms will be to determine the validity of the alarms by checking with the Chemistry Technician if any actions such as planned maintenance activities or other actions may have created false alarms. The SRO will immediately notify the SM of the situation to include alarm trends and actions taken. If a relief header alarm indicating extreme high pressure due to water-sodium reaction that is also received the SM will direct the SRO to immediately scram the reactor if not already scrammed. SM will direct SSO or CO to evacuate the SBB by actuating the SBB Evacuation Alarm from just outside the SBB entrance. This alarms directly any personnel in the SBB to immediately evacuate without hesitation for any reason. SM will direct the SSO or available operator to depress the SBB Fire Push Button (located in several locations outside and inside the SBB not in MCR). This action causes following automatic actions: Trips the Secondary sodium recirculating pumps and sec. sodium pump. SM will direct the PPO (if not already done) to secure the Feedwater Pumps (FWPs), shut feedwater and main steam isolation valves to stop all water and steam flow to and from the steam generation system. SM then directs SSO (if not already done) to dump Steam Generator (SG) water and drain the secondary sodium system to reduce the chance of more severe reactions. The SSO or CO will also depress the SG water dump valve pushbutton in MCR on the Secondary Sodium Panel, -observe- the open indication light, and - depress the Sodium Vent Valves Open pushbutton. This last action opens sodium-argon vent valves allowing secondary sodium to vent while draining without creating vapor lock. After a six minute waiting period the SSO will shut the vent valves and SG water dump valve. The combination of these actions should reduce the possibility of continued water and sodium reactions, the chance of an explosion causing significant damage and fire in the SBB, as well as limit or prevent personnel injury.
5	**Related function**	Reactor Scrams for other events. Secondary Sodium Pump Leak event. Sodium Leak in the Secondary Sodium System event
6	**Mode/state**	**Initial** - Full Power Operations **Final -** Scram, Cold Shutdown, and drained
7	**Initiating conditions**	• Hydrogen Meter Leak Detector (HMLD) alarm • Compact Hydrogen Meter Leak Detector (CHMLD) alarm • Increasing Hydrogen (H2) level Secondary Sodium • High Secondary Sodium Press • Leak Probe Detector alarms

Item	Item Name	Item Description
		• Secondary Sodium Relief Header flow alarm • Secondary Sodium System Rupture Disk alarm

Item	Item Name	Item Description
8	**Start state(s)**	**Reactor:** - Control & Safety Rods positioned for full power operations - Reactor Power: 100% (62.5 MWt) **Electric Plant:** - Normal Power available - Emergency Diesel Generators: In Standby - Turbine Generator: Supplying the electrical grid @ 20 MWe **Primary Sodium Systems:** - Rx Coolant Pump (RCPs): 100% Flow - Aux Pump: 100% (575 gpm) - Primary Flow: 100% (9000 gpm) - Bulk Na Temp: 695 – 705 F - Delta Temp: 183 F - Core cooling: Intermediate Heat Exchanger to Secondary Sodium - Primary Cover Gas: 6-7 psig - Primary Tank Heaters: Deenergized **Emergency Shutdown Coolers:** Operationally ready **Secondary Sodium Systems:** - Secondary Sodium Pump: 86 % (5160 gpm) - Secondary Sodium Recirculating Pumps operating normally - Secondary Sodium Temp to Steam Drum: 866 F - Secondary Sodium Temp From Steam Drum: 590 F - Secondary Argon Press: 6-7 psig - Sodium Boiler Building Fire Pushbutton depressed: Operationally Ready - Secondary Sodium Drain Tank: Operationally ready

Item	Item Name	Item Description
9	**End state(s)**	**Reactor:** - Control & Safety Rods full down (manual reactor scram) - Reactor Power: 0% (0 MWt) **Electric Plant:** - Normal Power available - Emergency Diesel Generators: In Standby - Turbine Generator: Tripped by manual scram @ 0 MWe **Primary Sodium Systems:** - Rx Coolant Pump (RCPs): 100% Flow - Aux Pump: 100% (575 gpm) - Primary Flow: 100% (9000 gpm) - Bulk Na Temp: 695 – 705 F - Delta Temp: <183 F and decreasing rapidly toward 0 F as decay heat dissipates - Core cooling: Ambient heat loses through primary tank - Primary Cover Gas: 6-7 psig - Primary Tank Heaters: De-energized, to be energized as needed when primary temperature decreases **Emergency Shutdown Coolers:** Operationally ready **Secondary Sodium Systems:** - Secondary Sodium Pump: 0 % (0 gpm) deenergized (not restarted) - Secondary Sodium Recirculating Pumps deenergized (not restarted)

Item	Item Name	Item Description
		- Secondary Sodium Temp to Steam Drum: rapidly decreasing to ambient - Secondary Sodium Temp From Steam Drum: rapidly decreasing to ambient - Secondary Argon Press: 6-7 psig - Sodium Boiler Building Fire Pushbutton depressed: Trips Secondary Sodium and Recirculating Pumps - Secondary Sodium dumped to Drain Tank and allowed to cool to ambient **Condensate and Feedwater Systems -** cold shutdown mode - Hotwell Level: N/A - Feedwater Temp: Ambient - Condensate Pumps: shutdown - Feedwater Pumps: shutdown **Main Steam System:** - Steam Pressure: 0 psig - Steam Temp: 0 F **Evaporators & Superheaters:** - Drained on both sodium and water sides - Sodium Temp from Evaporators: rapidly decreasing temperature - Sodium Temp to Superheaters: rapidly decreasing temperature **Fuel Handling and Associated Equipment:** - Rx Vessel Cover: Down/Locked - Large & Small Plug seals frozen

Item	Item Name	Item Description
10	**Related system**	Reactor Safety System
11	**Personnel Involved**	Shift Manager (SM), Reactor Operator (RO), Secondary Sodium Operator (SSO), Panel Coolant Operator (PCO), Field Coolant Operator (FCO), Power Plant Operator (PPO)
12	**Operator role**	Detection, diagnosis, and all responses require operators to take direct actions, no actions are automated For all events the following applies: • SM typically monitors plant response and initiates outside communications • The Shift Foreman (SF) typically monitors or directs communication between crew members
13	**Task Location**	Main Control Room, Just outside of the Secondary Sodium Building, Power Plant Building
14	**Operator main functions**	HMLD, CHMLD, and plant computer alarm monitoring. Diagnosis of alarms. Immediate actions: **Shift Manager (SM) -** Monitor Secondary Cover Gas Pressure (rising pressure will occur if valid). Validate plant computer operating correctly. Note: If a relief header-flow alarm is received during leak verification and validation period, immediately (SSO) drain the steam drum and drain (dump) the secondary sodium system (see actions below). **Chemistry Technician** - Observe HMLD and CHMLD data on plant computer for additional alarms or upward trends that would occur following the initial alarm(s). Note: If initial alarm is valid all operating leak detectors should show upward trends within 5 minutes at full power. - Observe HMLD and CHMLD membrane temperature data and flow data for any unusual upsets that might indicate false readings. - Verify if any perturbations have occurred or change in Secondary Cold Trap. Note: If steps above confirm valid leak alarm have SRO scram and continue with immediate actions below, if not continue to validate leak. - Start an equilibrium pressure measurement on HMLDs with pressure gages.

Item	Item Name	Item Description
		- Start a plugging run. - Inspect secondary gas chromatograph charts for increase in hydrogen. Perform a span-gas check to verify the hydrogen readings are accurate. Do not perform span-gas checks on both chromatographs simultaneously. - Observe the secondary cover gas pressure reading. (The pressure may increase if water-to-sodium leak is large enough.) If all indications point to water-to-sodium leak perform following; **Senior Reactor Operator (RO) -** Scrams Rx. and validates rod bottom lights indicate all rods on the bottom **Panel Coolant Operator (PCO) -** Actuate Sodium Boiler Building (SBB) Evacuation Alarm from MCR or **Secondary Sodium Operator (SSO) -** actuates SBB Evacuation Alarm from just outside the SBB entrance. **SM -** directs SSO or available operator to depress the SBB Fire Push Button (located in several locations outside and inside the SBB not in MCR). This action causes following automatic actions: Trips the Secondary sodium recirculating pumps and secondary sodium pump. **Power Plant Operator (PPO) -** secure Feedwater Pumps (FWPs), shut feedwater and main steam isolation valves. **SM -** directs SSO to dump Steam Generator (SG) water and drain the secondary sodium system. **SSO -** depresses the SG water dump valve pushbutton in MCR on Secondary Sodium Panel and observes open indication light. Depresses the Sodium Vent Valves Open pushbutton this opens sodium-argon vent valves allowing secondary sodium to vent while draining without creating vapor lock. Shuts vent valves after waiting six minutes. Shuts SG water dump valve. **Subsequent Actions:** **SM** – Notify the Operations Manager **SSO** – Maintain the recirculating pumps, cold trap, and associated piping and valves at 350 F by performing the following: - Place the control switches for heaters 91-61, 69, 71 and 72) on the

Item	Item Name	Item Description
		induction/resistance heater panel in SBB in off position. - Reset the Fire relays, using the Fire relay reset key switch located on the SBB control panel. - Close the 240-V Resistance Heating Main Breaker. - Close the 120-V Resistance Heating Main Breaker. - Restart the recirculating pump and flow sodium through the cold trap to cool the sodium storage tank to 350 F. (See SOP) **SSO -** After the sodium has cooled to 350 F, keep heating circuits 62 and 63 energized to maintain storage tank sodium at 350 F. **SSO** – If directed by Operations Manager, fill the water side of the steam generator with argon. (See SOP) **SM** – have the Environmental Compliance Representative sample the water in the catch tank to ensure that it meets environmental regulations for discharge. Plant Services – pump the water from the steam generator water dump system catch tank. (See SOP)
15	**Potential Performance Shaping Factors**	Stress, time available, workload, procedures, training, etc.
16	**Sub-functions**	Emergency Procedure compliance, alarm and indicator diagnosis, crew coordination and action sequencing, and place keeping
17	**Execution/Performance requirements**	In-depth knowledge and understanding of severe consequences of water-sodium reaction. Memorization of immediate actions and familiarity with subsequent actions.
18	**Timing**	Immediate Rx Scram is necessary to minimize time needed to drain Sec. Sodium System and potential water and sodium reactionary forces of plant systems. Also see timing instructions in section 14 associated with vent and dump valves
19	**Sequence up/down**	Immediate Down power - reactor scram
20	**Information from system**	Trend data from CHMLD, HMLD, Secondary Chromatographs, and plant computer
21	**Information**	Alarms come in on plant computer screen, CHMLD or HMLD alarms in

Item	Item Name	Item Description
	transmittal method	Sodium Boiler Builder main alarm panel or MCR alarm panel
22	**Termination indications**	Secondary Na System drained and at ambient temperature. SG feedwater drained and argon blanket placed on feedwater side of heat exchangers.
23	**Potential Errors**	Misdiagnoses of alarms such as plant computer malfunctions providing false alarms or failed hydrogen leak detectors providing false alarms from flow and temperature upsets in leak detection system (these false alarms are more likely during Rx startup or shutdown). Additionally, Changes in Secondary Cold Trap operations can also account for an increase in hydrogen level.
24	**Source documents**	Applicable Emergency Procedure
25	**Cues to the Operator (for commencement of the action)**	Alarms are received on the Hydrogen Meter Leak Detector (HMLD) or Secondary Cover Gas Hydrogen Leak Detector (CHMLD) hydrogen level or rate-of-rise data on the plant computer. The trend data for hydrogen will be on the rise. All operating leak detectors should show upward trends in 5 minutes at full power operations.
26	**Diagnosis Required**	Alarm validity and indication of faulty hydrogen detector or false information from plant computer malfunctions
27	**Control and Display Sufficiency**	In this instance the controls used are Rx Scram button, Sec. Sodium Vent Valve pushbutton and SG Water Dump Valve pushbutton.
28	**Feedback on the Operation**	Scram - rod bottom lights will illuminate SBB Evacuation Pushbutton - loud audible alarm in the SBB Fire Pushbutton - Level rise in sodium drain tank indicating sodium has dumped to the tank, zero flow indication on secondary pump and secondary sodium recirculation pumps Secondary Sodium Cover Gas Pressure - may be on the rise if leak is severe Trending data from various secondary sodium system monitors that indicate leak Temperature Indicators - rapid decrease in secondary sodium system temperature, evaporator and superheater temperature indicators
29	**Recovery Opportunities if**	N/A

Item	Item Name	Item Description
	Omitted	
30	**Consequences of Failure/Non-recovery**	A water-to-sodium leak results in a reaction that generates heat and liberates hydrogen, causing rapidly increasing temperature and pressure in the affected Evaporator or Superheater. The magnitude of the temperature and pressure increase depends on the size of the leak. Localized high temperature and pressure can cause failure of Evaporators or Superheaters.
31	**Technology Recommendations**	• Potential automation including Rx scram to be more conservative given the significance of such an event. • Consideration given to pushbutton actions to be automated upon hydrogen detection alarms (i.e. 3 out 5 sequence) with an alarm that is followed by automatic dumping of the sec. sodium system and SG water if operator action is not taken within a given time frame. • Provide a dedicated Water-Sodium Leak alarm/indication screen or panel that also provides trending data.
32	**Operating Experience**	**History of Sodium Leakage in Fast Reactors** • All operating fast reactors have experienced sodium leaks. • One of the most famous occurred at the MONJU reactor in 1995 when approximately 640Kg of sodium leaked from the secondary sodium system with a resulting fire. • A major sodium leak occurred at EBR-II in 1965 when a frozen sodium plug in sodium piping melted during maintenance, releasing approximately100Kg of secondary sodium. • As significant as these events were, no injuries resulted nor have any injuries resulted from leaks at any other operating fast reactors. • The low pressure of the coolant limits the rate of leakage and improved understanding of designs increasingly prevent and detect leakage. **Steam Generator Failures** • Sodium-steam interaction due to failure of steam generator tubing has also occurred, the most serious at the BN-350 reactor in its early operation. All reactors have experienced steam-sodium leakage with the exception of Superphoenix and FBTR. • In all cases, the failures were traced to poor welds, fabrication or design. • Much has been learned about prevention, detection and mitigation of

Ite m	Item Name	Item Description
		the consequences of steam-sodium leakage but perhaps the most important is that such leaks are not catastrophic. • As with other sodium leaks, no injuries have resulted from failure of steam generators. In addition, physical damage to the plants has been minor. J. I. Sackett, C. Grandy, "International Experience with Fast Reactor Operation & Testing", International Conference on Fast Reactors and Related Fuel Cycles, Paris, France - March 4-7, 2013

9.2.2 Reactor Scram (Manual or Automatic) – Generic (Emergency Procedure EP 2-1)

Item	Item Name	Item Description
1	**Scenario ID**	**Event 2**
2	**Name**	**Reactor Scram (Manual or Automatic) – Generic (EP 2-1)**
3	**Type**	Transient/Fault mode
4	**Scenario Description**	**Emergency Procedure Overview** The purpose of this emergency procedure (EP) is to provide corrective actions following a manual or automatic reactor scram that would put the reactor and plant systems in a safe configuration until the Shift Manager can evaluate the situation and direct appropriate follow-up corrective actions. It is impossible to foresee all situations that could result in a reactor scram; so the steps/actions of this EP should be followed by the specific EP needed to mitigate for the actuating event that lead to the reactor scram. Manual reactor scrams are performed when plant parameters, alarms or combination of the two indicate a need to take conservative actions to mitigate for plant damage, protect equipment (including the reactor core), environmental damage, a release of radioactive material, or personnel injury or death. Complex and/or compound events are not evaluated or addressed by written EPs, as a result operators are trained to be conservative and scram the reactor if it is deemed necessary for any of the reasons listed above. This may or may not be the case with advanced sodium reactor designs where real-time monitoring and diagnostics, smart controllers and computerized cross-referencing procedures may be available to the crew. With EBR-II design the automatic scrams occur when the safety string logic experiences a 2 out 3 sequence leading to de-energizing of the control rod drive mechanisms. **Control and Safety Drive Systems** Twelve control rods controlled the operation of the reactor. Each rod was independently driven by an electrical-mechanical drive mechanism. The drives were identical and were so arranged that only one drive could be operated at a time, with the exception of scram when all 12 operated simultaneously. Operating control was achieved by a 14-inch vertical motion of the control rods that was provided by a rack and pinion-type drive with constant-speed electric motors, therefore, only one speed of movement was possible. The control rods were disconnected from their drives during fuel handling operations. The disconnect was made with the control rods in their down or least reactive position. The control rods remained in this position during fuel handling operations. A reactor scram is a rapid insertion of the reactor control rods. This action removes fuel

Item	Item Name	Item Description
		from the reactor core and drives a neutron absorber section of the control rods into the core region taking the reactor from a critical state to a sub-critical state. No further heat production could occur within the reactor with the exception of decay heat. **Mechanics of a Reactor Scram** Upon a scram signal, the magnetic clutch was de-energized, releasing the shaft from the drive rack and driving the control rod down, out of the reactor core. Scram could occur at any position in the operating stroke of the control rod and was automatically actuated by a power failure, which de-energized the magnetic clutch. This was accomplished in a release time of 0.008 second, including the time elapsed between actuating the scram signal and beginning of shaft motion. To ensure the compressed air supply to the air cylinder, accumulator tanks were provided, which in turn were supplied by an air compressor. Check valves were provided in the connecting lines between the accumulator tanks and the air cylinders, and between the air compressor and the accumulator tanks, to prevent loss of compressed air in the event of line failure. Pressure actuated switches scrammed the reactor in the event of failure of the air supply. The compressed air available in the cylinder or in the accumulator tanks was sufficient to insure pressure to assist during a scram, in addition to the force of gravity. Deceleration of the scram stroke was accomplished by a hydraulic shock absorber connected to the air cylinder. The shock absorber was actuated during the lower 5 inches of travel.
5	**Related function**	
6	**Mode/state**	**Start:** Normal Power Operations **Final**: Shutdown, Zero Power
7	**Initiating conditions**	Any number of abnormal or emergency conditions or combinations (compound event) can lead to a manual or automatic scram. Automatic scrams are initiated by a shutdown logic string with a 2 out 3 logic requirement except for loss of power that was single logic. Automatic scram initiators (Reactor Shutdown System): Earthquake, High Reactor Power (110%), Wide-Range Period, Primary Pump Low Primary Flow, Total Coolant Low Flow, Subassembly Outlet Temperature High, and Loss of Normal Power (2400 Voltage Bus - Low Voltage)

Ite m	Item Name	Item Description
8	**Start state(s)**	**Reactor:** - Control & Safety Rods positioned for full power operations - Reactor Power: 100% (62.5 MWt) **Electric Plant:** - Normal Power available - Emergency Diesel Generators: In Standby - Turbine Generator: Supplying the electrical grid @ 20 MWe **Primary Sodium Systems:** - Rx Coolant Pump (RCPs): 100% Flow - Aux Pump: 100% (575 gpm) - Primary Flow: 100% (9000 gpm) - Bulk Na Temp: 695 – 705°F - Delta Temp: 183°F - Core cooling: Intermediate Heat Exchanger to Secondary Sodium - Primary Cover Gas: 6-7 psig - Primary Tank Heaters: Deenergized **Emergency Shutdown Coolers**: Operationally ready **Secondary Sodium Systems:** - Secondary Sodium Pump: 86 % (5160 gpm) - Secondary Sodium Recirculating Pumps operating normally - Secondary Sodium Temp to Steam Drum: 866°F

Item	Item Name	Item Description
		- Secondary Sodium Temp From Steam Drum: 590°F - Secondary Argon Press: 6-7 psig - Sodium Boiler Building Fire Pushbutton depressed: Operationally Ready - Secondary Sodium Drain Tank: Operationally ready **Condensate and Feedwater Systems –** Normal Full Power mode - Hotwell Level: 26-29 inches - Feedwater Temp: 550°F - Condensate Pumps: 1 Run / 1 Standby - Feedwater Pumps: 1 Run / 1 Standby **Main Steam System:** - Steam Pressure: 1250 psig - Steam Temp: 820°F **Evaporators & Superheaters:** - Fully operational - Sodium Temp from Evaporators: 590°F - Sodium Temp to Superheaters: 866°F **Fuel Handling and Associated Equipment:** - Rx Vessel Cover: Down/Locked - Large & Small Plug seals frozen
9	**End state(s)**	**Reactor:** - Control & Safety Rods full down (Automatic or manual reactor scram)

Item	Item Name	Item Description
		- Reactor Power: 0% (0 MWt) **Electric Plant:** - Normal Power available (Except in the case of Loss Of Off-site Power or partial loss of normal power systems) - Emergency Diesel Generators: In Standby - Turbine Generator: Tripped by automatic or manual scram @ 0 MWe **Primary Sodium Systems:** - Rx Coolant Pump (RCPs): 100% Flow (except loss of RCPs or Normal Off-site power) - Aux Pump: 100% (575 gpm) - Primary Flow: 100% (9000 gpm) (except loss of RCPs or Normal Off-site power) - Bulk Na Temp: 695 – 705 F - Delta Temp: <183 F and decreasing rapidly toward 0 F as decay heat dissipates - Core cooling: Ambient heat loses through primary tank - Primary Cover Gas: 6-7 psig - Primary Tank Heaters: Deenergized, to be energized as needed when primary temperature decreases **Emergency Shutdown Coolers**: Operationally ready **Secondary Sodium Systems:** Initial State - Secondary Sodium Pump: 0 % (0 gpm) deenergized Final State - Secondary Sodium Pump: Restart pump on alternate power

Item	Item Name	Item Description
		(Variable gpm to control bulk Na Temp. 695-705 F) - Secondary Sodium Temp to Steam Drum: rapidly decreasing - Secondary Sodium Temp From Steam Drum: rapidly decreasing - Secondary Argon Press: 6-7 psig - Sodium Boiler Building Fire Pushbutton: Operationally ready - Secondary Sodium Drain Tank: Operationally ready **Condensate and Feedwater Systems:** - Feedwater System in manual control to maintain steam drum level 30 inches - Hotwell Level: 26-29 inches - Feedwater Temp: decreasing - Condensate Pumps: 1 Run / 1 Standby - Feedwater Pumps: Initial - 1 Run / 1 Standby Final - secured **Main Steam System:** - Steam Pressure: rapidly decreasing - Steam Temp: rapidly decreasing **Evaporators & Superheaters:** - Fully operational - Sodium Temp from Evaporators: rapidly decreasing - Sodium Temp to Superheaters: rapidly decreasing **Fuel Handling and Associated Equipment:** - Rx Vessel Cover: Down/Locked

Item	Item Name	Item Description
		- Large & Small Plug seals frozen
10	**Related system**	Reactor Safety System (Shutdown String) Control Rod Drive Mechanisms
11	**Personnel Involved**	Shift Manager (SM), Reactor Operator (RO), Secondary Sodium Operator (SSO), Panel Coolant Operator (PCO), Power Plant Operator (PPO), Electric Plant Operator (EPO) typically PPO and EPO are same person
12	**Operator role**	Detection, diagnosis, and all responses require operators to take direct actions, the exception is an automatic scram For all events the following applies: • SM typically monitors plant response and initiates outside communications • The Shift Foreman (SF) typically monitors or directs communication between crew members
13	**Task Location**	Immediate Actions: Main Control Room Subsequent Actions: Main Control Room and Power Plant
14	**Operator main functions**	Diagnosis of alarms and indications to support manual scram or understand/diagnose automatic scram **Immediate actions:** **Shift Manager (SM)**, verify crew response is appropriate for given indications, alarms, and emergency procedure immediate and subsequent actions. If reactor shutdown cannot be verified perform the following: Reactor Facility Evacuation and try to maintain reactor flow to level prior to scram if possible **Reactor Operator (RO) -** Scrams Rx. and validate rod bottom lights indicate all rods on the bottom, Channels A, B, and C Power Indications show decreasing power, and Channels A, B, and C Period Meter indicate negative period indications. For automatic scram SRO also pushes Scram Pushbutton and turns control keys to off position and removes them from the console. **Panel Coolant Operator (PCO) -** Verifies Secondary Electromagnetic Pump (EM) trips 6 seconds after scram **Power Plant Operator/Electric Plant Operator (PPO/EPO) -** Verify

Item	Item Name	Item Description
		turbine generator, the blowdown system, and 150 psig Auxiliary Steam supply have tripped. Place the pilot wire trip to off. Place the tap changers switch to Auto. Press the Off Pushbuttons for the low voltage and high MVAR trips. Manually control feedwater control valve to increase steam drum level to 30 inches. **Subsequent Actions:** **SRO -** Inform SM if not already aware of scram **PCO -** Allow secondary flow until primary sodium temperature reaches 690 F then restart secondary EM pump on alternate power (controller must be run to minimum first) and adjust flow to maintain 685-695 F primary temperature. **SSO -** Open secondary system vent valves using pushbutton in MRC or SBB control panel and deenergize secondary EM pump motor-generator set in SBB. **Crew -** determine cause of automatic scram **SRO -** lower the control rod racks **PPO -** Secure main cooling tower fans power plant panel in MCR, locally secure one condenser circulating water pump, locally route condenser cooling water flow to the main tower basin to maintain condenser cooling water temperature. Place steam plant in hot standby or as directed by SM.
15	**Potential Performance Shaping Factors**	
16	**Sub-functions**	Emergency Procedure compliance, alarm and indicator diagnosis, crew coordination and action sequencing, and place keeping
17	**Execution/Performance requirements**	In-depth knowledge and understanding of Reactor Shutdown logic, anticipated events, and unanticipated complex or compound events. Memorization of immediate actions and familiarity with subsequent actions.
18	**Timing**	Immediate response for reactor safety events to prevent core damage or damage to safety related systems
19	**Sequence up/down**	Immediate Down power - reactor scram

Item	Item Name	Item Description
20	**Information from system**	Trending data, plant alarms, and plant parameters from various sources
21	**Information transmittal method**	Alarms come in on plant computer screen and Alarm Annunciator Panels illuminate
22	**Termination indications**	• Control rod down (bottom) lights are on • Channels A, B, and C power channel indicators and recorders show decreasing reactor power • Channels A, B, and C period meter shows negative period indications • Secondary Electromagnetic Pump trip 6 seconds after scram • Turbine Generator trips
23	**Potential Errors**	Delay in scramming the reactor when complex or compounding events occur -making the diagnosis more difficult. Multiple alarms create difficulty in disseminating order alarms came in and cascading alarms create difficulty in diagnosis.
24	**Source documents**	EP and system description manuals
25	**Cues to the Operator (for commencement of the action)**	Control rod down lights are on
26	**Diagnosis Required**	Automatic Scram is self-revealing in nature (see item 28) Manual Scrams are a result of diagnosis and action decision
27	**Control and Display Sufficiency**	Control Rod position indications, reactor power, reactor period and alarms are easily viewed on reactor console, reactor panel in front of the console and the plant computer
28	**Feedback on the Operation**	Control rod down lights are on Channels A, B, and C power channel indicators and recorders show decreasing reactor power Channels A, B, and C period meter shows negative period indications Secondary Electromagnetic Pump trip 6 seconds after scram Turbine Generator trips Blowdown System trips Auxiliary Steam Cross Connect trips
29	**Recovery**	If reactor did not automatically scram but alarms and indications dictate

Item	Item Name	Item Description
	Opportunities if Omitted	scram, manual scram can easily be performed by pushing Scram Pushbutton on Reactor Console
30	**Consequences of Failure/Non-recovery**	See Operating Experience Item 31 - numerous test performed at EBR-II indicate in most cases that even without a scram the passive safety systems and self-protecting nature of the reactor design are sufficient to prevent core damage
31	**Recommendations**	Evaluate additional faults to be added to the Reactor Safety System logic and operator support systems to aid in rapid diagnosis of abnormal and emergency conditions
	Operating Experience	**Reactor Operation is Straightforward** • At EBR-II the reactor was operated with many different core configurations. In all configurations, the reactor was stable in its operation. • Another aspect is that operating procedures are straightforward, aided by the self-protecting nature of the reactors. • At EBR-II, extensive tests were conducted that not only included ATWS events associated with loss of flow, but also – single rod run-out, – primary pump control malfunctions, – load following and – steam system failures • These tests also demonstrated that EBR-II was tolerant of operators taking an improper control action. • These characteristics greatly reduced pressure on operators in the event of off-normal events. Rapid operator response was not required. **Safety Testing in EBR-II** • Fuel was extensively tested under off-normal conditions – Operation with breached cladding (oxide and metal) – Transient overpower (TOP) events (EBR-II TOPs complimented more severe TOPs in the TREAT reactor) • Inherently safe response of EBR-II was demonstrated after 12 years of extensive testing and analysis

Item	Item Name	Item Description
		– Loss-of-flow without scram (station blackout) – Loss-of-heat-sink without scram • A level one PRA was completed to quantify safety J. I. Sackett, C. Grandy, "International Experience with Fast Reactor Operation & Testing", International Conference on Fast Reactors and Related Fuel Cycles, Paris, France - March 4-7, 2013

9.2.3 Earthquake (minor to moderate) (Emergency Procedure EP 1-7)

Item	Item Name	Item Description
1	**Scenario ID**	**Event 3**
2	**Name**	**Earthquake (minor to moderate) (EP 1-7)**
3	**Type**	Transient/Fault mode
4	**Scenario Description**	**Hazards from an Earthquake** An earthquake presents two hazards, (1) reactor shutdown may not be possible if control and safety rods or their drive mechanisms bind, and (2) piping systems may fail, releasing water, sodium, or high-pressure high-temperature steam presenting danger to the plant equipment and personnel. The earthquake detection system is designed to scram the reactor prior the forces of the actual earthquake reaching EBR-II structures. Operating personnel responsibilities, therefore, involve ensuring that the reactor has scrammed and placing plant systems in a safe condition if possible. Depressurization and draining of some systems may be desirable, based on judgment of the Shift Manager (SM). **Additional Concerns** An earthquake may also cause a loss of electrical power to the entire site. If power is lost to the deep well pumps, water for fire protection would be limited to that available in the storage tanks. If the deep well pumps lose power, it will be necessary to eliminate unneeded use of water to conserve water in the storage tanks. Post-earthquake, debris and fires may make certain areas including the MCR uninhabitable, interfere with site and building evacuation requirements. **Earthquake Detection System** The earthquake detection system consists of three separate detectors, which are connected in a two-out-of-three coincident scram circuit. Each detector has a vertical, transverse, and longitudinal motion acceleration sensor. Any of the three sensors will trip the detector, causing annunciation on the primary systems panel in the Main Control Room (MCR). There are local lights that provide indication of which sensor tripped the detector. A trip of two or more detectors is annunciated on the reactor control console and results in a reactor scram. **Detector Locations** Earthquake Detector # 1, Sodium Boiler Building (SBB) west wing basement (electrical vault)

Item	Item Name	Item Description
		Earthquake Detector # 2, Power Plant Cable Tunnel East Earthquake Detector # 3, Power Plant Cable Tunnel West
5	**Related function**	Reactor Scram Full Rx Building Isolation Rx Building Evacuation SBB Evacuation Site Area Emergency
6	**Mode/state**	**Start:** Normal Power Operations **Final**: Shutdown, Zero Power
7	**Initiating conditions**	An earthquake that is strong enough to actuate the earthquake detection system causing a two-out-of-three coincident and resultant scram. Each detector has a vertical, transverse, and longitudinal motion acceleration sensor. Any of the three sensors will trip the detector.
8	**Start state(s)**	**Reactor:** - Control & Safety Rods positioned for full power operations - Reactor Power: 100% (62.5 MWt) **Electric Plant:** - Normal Power available - Emergency Diesel Generators: In Standby - Turbine Generator: Supplying the electrical grid @ 20 MWe **Primary Sodium Systems:** - Rx Coolant Pump (RCPs): 100% Flow - Aux Pump: 100% (575 gpm) - Primary Flow: 100% (9000 gpm) - Bulk Na Temp: 695 – 705 F

Item	Item Name	Item Description
		- Delta Temp: 183 F - Core cooling: Intermediate Heat Exchanger to Secondary Sodium - Primary Cover Gas: 6-7 psig - Primary Tank Heaters: Deenergized **Emergency Shutdown Coolers:** Operationally ready **Secondary Sodium Systems:** - Secondary Sodium Pump: 86 % (5160 gpm) - Secondary Sodium Recirculating Pumps operating normally - Secondary Sodium Temp to Steam Drum: 866 F - Secondary Sodium Temp From Steam Drum: 590 F - Secondary Argon Press: 6-7 psig - Sodium Boiler Building Fire Pushbutton depressed: Operationally Ready - Secondary Sodium Drain Tank: Operationally ready **Condensate and Feedwater Systems –** Normal Full Power mode - Hotwell Level: 26-29 inches - Feedwater Temp: 550 F - Condensate Pumps: 1 Run / 1 Standby - Feedwater Pumps: 1 Run / 1 Standby **Main Steam System:** - Steam Pressure: 1250 psig - Steam Temp: 820 F

Item	Item Name	Item Description
		Evaporators & Superheaters: - Fully operational - Sodium Temp from Evaporators: 590 F - Sodium Temp to Superheaters: 866 F **Fuel Handling and Associated Equipment:** - Rx Vessel Cover: Down/Locked - Large & Small Plug seals frozen
9	**End state(s)**	**Reactor:** - Control & Safety Rods full down (Automatic or manual reactor scram) - Reactor Power: 0% (0 MWt) **Electric Plant:** - Normal Power available - Emergency Diesel Generators: In Standby - Turbine Generator: Tripped by automatic or manual scram @ 0 MWe **Primary Sodium Systems:** - Rx Coolant Pump (RCPs): 100% Flow - Aux Pump: 100% (575 gpm) - Primary Flow: 100% (9000 gpm) - Bulk Na Temp: 695 – 705 F - Delta Temp: <183 F and decreasing rapidly toward 0 F as decay heat dissipates - Core cooling: Ambient heat loses through primary tank - Primary Cover Gas: 6-7 psig - Primary Tank Heaters: Deenergized, to be energized as needed when primary

Item	Item Name	Item Description
		temperature decreases **Emergency Shutdown Coolers:** Operationally ready **Secondary Sodium Systems:** Initial State - Secondary Sodium Pump: 0 % (0 gpm) deenergized Final State - Secondary Sodium Pump: Restart pump on alternate power (Variable gpm to control bulk Na Temp. 695-705 F) - Secondary Sodium Temp to Steam Drum: rapidly decreasing - Secondary Sodium Temp From Steam Drum: rapidly decreasing - Secondary Argon Press: 6-7 psig - Sodium Boiler Building Fire Pushbutton: Operationally ready - Secondary Sodium Drain Tank: Operationally ready **Condensate and Feedwater Systems:** - Feedwater System in manual control to maintain steam drum level 30 inches - Hotwell Level: 26-29 inches - Feedwater Temp: decreasing - Condensate Pumps: 1 Run / 1 Standby - Feedwater Pumps: Initial - 1 Run / 1 Standby Final - secured **Main Steam System:** - Steam Pressure: rapidly decreasing - Steam Temp: rapidly decreasing **Evaporators & Superheaters:**

Item	Item Name	Item Description
		- Fully operational - Sodium Temp from Evaporators: rapidly decreasing - Sodium Temp to Superheaters: rapidly decreasing **Fuel Handling and Associated Equipment:** - Rx Vessel Cover: Down/Locked - Large & Small Plug seals frozen
10	**Related system**	Reactor Safety System Reactor Building Isolation System
11	**Personnel Involved**	Entire Crew and potential emergency response personnel including fire department
12	**Operator role**	Operating personnel responsibilities involve ensuring that the reactor has scrammed - reactor shutdown may not be possible if control and safety rods or their drive mechanisms bind. The Crew is responsible for placing plant systems in a safe condition if possible. Depressurization and draining of some systems may be desirable, based on judgment of the Shift Manager (SM) - piping systems may fail, releasing water, sodium, or high-pressure high-temperature steam presenting danger to the plant equipment and personnel. For all events the following applies: • SM typically monitors plant response and initiates outside communications • The Shift Foreman (SF) typically monitors or directs communication between crew members
13	**Task Location**	All plant facilities including MCR
14	**Operator main functions**	**Immediate Actions:** **SM -** Direct evacuation of reactor building and SBB - make all call announcement **SRO -** Verify that the control and safety rods have scrammed (rod bottom lights) (see Scram Event Scenario)

Item	Item Name	Item Description
		PCO - Verify that a full reactor building isolation has occurred - Push reactor building evacuation pushbutton **PPO -** Place power plant systems in standby **SSO -** Push SBB evacuation pushbutton **-** Place secondary sodium systems in standby **Subsequent Actions:** **SM -** Direct an inspection of plant systems and determine the advisability of draining and/or depressurizing any system(s) that may have been damaged. **-** Inform Operations Manager **-** Contact Warning Communication Center and declare a site area emergency **-** Notify Emergency Action Manager **Crew -** complete system operations (shutdown, isolate, drain, depressurize, etc.) as directed by SM **I&C Technician -** coordinate with CO and reset tripped earthquake detectors upon SM orders
15	**Potential Performance Shaping Factors**	
16	**Sub-functions**	Notifications to emergency and state agencies, Emergency Procedure compliance, alarm and indicator diagnosis, crew coordination and action sequencing, and place keeping
17	**Execution/Performance requirements**	In-depth knowledge and understanding of Reactor Shutdown logic, anticipated events, and unanticipated complex or compound events. Memorization of immediate actions and familiarity with subsequent actions.
18	**Timing**	Immediate response for reactor safety events to prevent core damage or damage to safety related systems this especially true should binding occur that prevents one or more of the control or safety rods from scramming
19	**Sequence up/down**	Down
20	**Information from system**	Trending data, plant alarms, and plant parameters from various sources

Item	Item Name	Item Description
21	**Information transmittal method**	Alarms come in on plant computer screen and Alarm Annunciator Panels illuminate
22	**Termination indications**	All rods on bottom (rod bottom lights lit) Earthquake alarms Full Rx Building Isolation Rx Building Evacuation SBB Evacuation
23	**Potential Errors**	Delay in scramming should failure of earthquake detectors or reactor shutdown logic string does not scram the reactor or when complex or compounding events occur due to diagnosis difficulties that may be encountered due to the presence of fire, toxics, or hampered by ingress/egress issues. Communication errors due to large volume of communication and crew coordination required.
24	**Source documents**	EP and system description manuals
25	**Cues to the Operator (for commencement of the action)**	Earthquake Detector System annunciator on the reactor console annunciator panel Earthquake Detector # 1, Sodium Boiler Building (SBB) annunciator on the primary sodium panel Earthquake Detector # 2, Tunnel East annunciator on the primary sodium panel Earthquake Detector # 3, Tunnel West annunciator on the primary sodium panel
26	**Diagnosis Required**	Proper understanding of event sequence, multiple annunciator alarms that may mask initial cause if earthquake is not felt by personnel, full scram indication (all rod bottom lights), did plant activity such as movement of heavy equipment create fault (trip of detectors)
27	**Control and Display Sufficiency**	Control Rod position indications, reactor power, reactor period and alarms are easily viewed on reactor console, reactor panel in front of the console and the plant computer
28	**Feedback on the Operation**	Control rod down lights are on (scram) Reactor Building Isolation automatically occurs Channels A, B, and C power channel indicators and recorders show decreasing reactor power Channels A, B, and C period meter shows negative period indications

Item	Item Name	Item Description
		Secondary Electromagnetic Pump trip 6 seconds after scram Turbine Generator trips Blowdown System trips Auxiliary Steam Cross Connect trips
29	**Recovery Opportunities if Omitted**	If Earthquake Detectors illuminate (2 of 3) the SRO can still scram the reactor if actual earthquake is occurring
30	**Consequences of Failure/Non-recovery**	Failure to scram either automatically or manually could result in a loss of heat sink due to failure of systems that were damaged in the earthquake and are no longer functioning normally. This could result in high bulk sodium temperature. However, previous test results for a loss of secondary sodium (heat sink) proved that the reactor would shutdown with no operator action and no core damage.
31	**Recommendations**	In this event numerous cascading alarms will mask the more significant elements and areas of concern. An alarm management system that provides the operators with the key alarms (not subsequent alarms) would improve diagnosis and response times. Additional automation of systems such as automatic draining and depressurization of systems affected by earthquake might improve survivability of plant personnel and reduce the likely hood of additional systems failing or creating irrecoverable conditions due to catastrophic failures such as fires created by the initial or subsequent events.
32	**Operating Experience**	**Reactor Operation is Straightforward** • At EBR-II the reactor was operated with many different core configurations. In all configurations, the reactor was stable in its operation. • Another aspect is that operating procedures are straightforward, aided by the self-protecting nature of the reactors. • At EBR-II, extensive tests were conducted that not only included ATWS events associated with loss of flow, but also – single rod run-out, – primary pump control malfunctions, – load following and – steam system failures • These tests also demonstrated that EBR-II was tolerant of operators taking an improper control action. • These characteristics greatly reduced pressure on operators in the event of off-normal events. Rapid operator response was not required.

Item	Item Name	Item Description
		Safety Testing in EBR-II • Fuel was extensively tested under off-normal conditions – Operation with breached cladding (oxide and metal) – Transient overpower (TOP) events (EBR-II TOPs complimented more severe TOPs in the TREAT reactor) • Inherently safe response of EBR-II was demonstrated after 12 years of extensive testing and analysis – Loss-of-flow without scram (station blackout) – Loss-of-heat-sink without scram • A level one PRA was completed to quantify safety J. I. Sackett, C. Grandy, “International Experience with Fast Reactor Operation & Testing”, International Conference on Fast Reactors and Related Fuel Cycles, Paris, France - March 4-7, 2013 **Our review of operating experience revealed that an earthquake measuring 6.9 on the Richter scale happened in October 1983. The epicenter for the quake was in Mackey Idaho area. This is within a 100 miles of EBR-II. The EBR-II earthquake detection system functioned as designed resulting in a scram. This information has been factored into this analysis.

9.2.4 Loss of Normal Off-Site Power (not a Turbine Generator Fault) (Emergency Procedure EP 6-2)

Item	Item Name	Item Description
1	**Scenario ID**	**Event 4**
2	**Name**	**Loss of Normal Off-Site Power (not a Turbine Generator Fault) (EP 6-2)**
3	**Type**	Transient/Fault mode
4	**Scenario Description**	**Equipment Affected** All equipment that is not energized by power from emergency batteries or emergency diesel generators (EDGs) **General Overview** Loss of normal power is defined as the loss of power to the 13.8-kV bus. This is the bus supplied from two potential sources, the INL loop and/or the EBR-II turbine generator. This Event analysis only considers a loss of off-site power not a turbine generator fault that leads to a loss of normal power. In the event normal electric power is lost, many changes occur automatically. The most important are listed below: - The reactor scrams - Primary reactor coolant pumps (RCPs) stop - The turbine generator trips - Secondary sodium pump and secondary recirculating pumps stop - The 400 and 125 kW emergency diesel generators start and load - All electrical heating of the primary and secondary sodium systems is lost. - All motor-driven pumps in the power plant systems lose power except the turning-gear oil pump, ac and dc seal oil pumps, reactor auxiliary-cooling-water pumps, demineralizer pumps, and emergency feedwater charging pump. **Most Critical Systems** The most critical systems continue to operate supplied by emergency electric power and/or battery power. These include: - Shield and thimble cooling

Item	Item Name	Item Description
		- Primary auxiliary pump - Instrument Air - Continuous power supply - Emergency lighting - Auxiliary boiler steam - Secondary sodium pump (on alternate power when manually switched) - Turbine generator lube oil - Turbine generator seal oil - Demineralized water **Major Operational Objectives** The major operational objectives are conserving steam generation system water and ensuring that the emergency power systems operate properly. The primary bulk sodium can be effectively used as a heat sink for reactor decay heat, and the shutdown coolers can maintain the bulk sodium temperature well below the temperature where damage to the primary system and reactor components becomes a consideration. During loss of normal power, the primary and steam systems are not operational, as during normal plant standby. The important differences are: - No electrical heating is available for any system - The turbine generator condenser is isolated - Feedwater can only be added to the steam drum using the emergency feedwater charging pump (EFCP) When loss of power occurs, existing water in the steam drum represents a heat reservoir to be conserved as efficiently as possible. Excess steam pressure will be vented to the atmosphere through the main steam header pressurematic relief valve. Hence, to conserve the water in the steam drum, the secondary sodium pump should be reenergized on the alternate power supply to reduce natural circulation flow of the secondary sodium systems, and thus reduce the amount of steam production and loss of water inventory in the steam drum. On the loss of cooling water to the secondary sodium pump, the winding temperatures of the pump must be monitored closely. If any of the temperatures increase to 275 F, the plant cooling water system must be connected to the raw system to allow for supply of cooling

Item	Item Name	Item Description
		water. If feedwater is not added to the steam drum with the EFCP before the steam drum water is consumed by steam production, the steam generator must be voided of all of water (dumped) and the secondary sodium system must be drained to the secondary sodium drain tank to avoid undue thermal stress.
5	**Related function**	Turbine Generator Fault that leads to tripping the turbine generator and loss of normal power
6	**Mode/state**	**Start:** Normal Power Operations **Final**: Shutdown, Zero Power
7	**Initiating conditions**	Loss of Normal Off-Site Power due to internal or external events
8	**Start state(s)**	**Reactor:** - Control & Safety Rods positioned for full power operations - Reactor Power: 100% (62.5 MWt) **Electric Plant:** - Normal Power available - Emergency Diesel Generators: In Standby - Turbine Generator: Supplying the electrical grid @ 20 MWe **Primary Sodium Systems:** - Rx Coolant Pump (RCPs): 100% Flow - Aux Pump: 100% (575 gpm) - Primary Flow: 100% (9000 gpm) - Bulk Na Temp: 695 – 705°F - Delta Temp: 183°F - Core cooling: Intermediate Heat Exchanger to Secondary Sodium - Primary Cover Gas: 6-7 psig - Primary Tank Heaters: Deenergized

Item	Item Name	Item Description
		Emergency Shutdown Coolers: Operationally ready **Secondary Sodium Systems:** - Secondary Sodium Pump: 86 % (5160 gpm) - Secondary Sodium Recirculating Pumps operating normally - Secondary Sodium Temp to Steam Drum: 866°F - Secondary Sodium Temp From Steam Drum: 590°F - Secondary Argon Press: 6-7 psig - Sodium Boiler Building Fire Pushbutton depressed: Operationally Ready - Secondary Sodium Drain Tank: Operationally ready **Condensate and Feedwater Systems** – Normal Full Power mode - Hotwell Level: 26-29 inches - Feedwater Temp: 550°F - Condensate Pumps: 1 Run / 1 Standby - Feedwater Pumps: 1 Run / 1 Standby **Main Steam System:** - Steam Pressure: 1250 psig - Steam Temp: 820°F **Evaporators & Superheaters:** - Fully operational - Sodium Temp from Evaporators: 590°F - Sodium Temp to Superheaters: 866°F

Item	Item Name	Item Description
		Fuel Handling and Associated Equipment: - Rx Vessel Cover: Down/Locked - Large & Small Plug seals frozen
9	**End state(s)**	**Reactor:** - Control & Safety Rods full down (Automatic or manual reactor scram) - Reactor Power: 0% (0 MWt) **Electric Plant:** - Normal Power not available - Emergency Diesel Generators: In Run (mode), supplying emergency distribution panels and associated loads - Turbine Generator: Tripped by automatic scram @ 0 MWe **Primary Sodium Systems:** - Rx Coolant Pump (RCPs): 0% Flow - Aux Pump: 100% (575 gpm)(supplied by Aux Pump battery) - Primary Flow: 575 gpm - Bulk Na Temp: 695 – 705 F - Delta Temp: <183 F and decreasing rapidly toward 0 F as decay heat dissipates - Core cooling: IHX (limited by Secondary Pump operations), Shutdown Coolers, and Ambient heat loses through primary tank - Primary Cover Gas: 6-7 psig - Primary Tank Heaters: Deenergized **Emergency Shutdown Coolers**: Operating when Bulk Sodium Temperature reaches 710 F

Item	Item Name	Item Description
		Secondary Sodium Systems: (see actions for Drum Level Indication below) Initial State - Secondary Sodium Pump: 0 % (0 gpm) deenergized - Secondary Sodium Temp to Steam Drum: rapidly decreasing - Secondary Sodium Temp From Steam Drum: rapidly decreasing - Secondary Argon Press: 6-7 psig - Sodium Boiler Building Fire Pushbutton: Operationally ready - Secondary Sodium Drain Tank: Operationally ready **Steam Drum Level Indicated:* Secondary Sodium Pump: Restart pump on alternate power (Variable gpm to control bulk Na Temp. 695-705 F) **Steam Drum Level Not Indicated (below sight-glass level):* Secondary Sodium Pump - to remain deenergized Secondary Argon Cover Gas - vented Secondary Sodium System - dumped to Secondary Sodium Drain Tank and allowed to cool to ambient **Condensate and Feedwater Systems:** - Feedwater System in manual control to maintain steam drum level 30 inches - Hotwell Level: 26-29 inches - Feedwater Temp: decreasing - Condensate Pumps: secured *Steam Drum Level Indicated *(below sight-glass level)*: - Emergency Feedwater Pump - run to maintain steam drum level and circulating yard lines

Item	Item Name	Item Description
		*Steam Drum Level Not Indicated: - Emergency Feedwater Pump - secured **Main Steam System:** - Steam Pressure: rapidly decreasing - Steam Temp: rapidly decreasing **Steam Drum Level Indicated:* Main Steam Stop shut **Steam Drum Level <u>Not</u> Indicated (below sight-glass level):* - Steam Drum isolated on feedwater and main steam side - Steam Dump Valve actuated: this immediately drains Steam Drum to Steam Generator Water Dump System **Evaporators & Superheaters: (unless system is drained to Secondary Sodium Drain Tank)** - Fully operational - Sodium Temp from Evaporators: rapidly decreasing - Sodium Temp to Superheaters: rapidly decreasing **Fuel Handling and Associated Equipment:** - Rx Vessel Cover: Down/Locked - Large & Small Plug seal heaters power supply manually switched to emergency power (in preparation for melting seals)
10	**Related system**	Reactor Safety System (shutdown logic string) Turbine generator control system
11	**Personnel Involved**	Entire crew

Item	Item Name	Item Description
12	**Operator role**	All responses require operators to take direct actions, the exception is an automatic scram For all events the following applies: • SM typically monitors plant response and initiates outside communications • The Shift Foreman (SF) typically monitors or directs communication between crew members
13	**Task Location**	Immediate Actions: Main Control Room Subsequent Actions: Main Control Room and Power Plant
14	**Operator main functions**	**Immediate actions:** **Shift Manager (SM) -** verify crew response is appropriate for given indications; alarms, and emergency procedure immediate and subsequent actions. Make the appropriate notification based on time of day and extent of event. **Reactor Operator (RO) -** Verify reactor scrammed and validate rod bottom lights indicate all rods on the bottom, Channels A, B, and C Power Indications show decreasing power, and Channels A, B, and C Period Meter indications show negative period indications. For all automatic scrams the SRO also pushes Scram Pushbutton and turns control keys to off position and removes them from the console. **Panel Coolant Operator (PCO) -** Verifies Secondary Electromagnetic Pump (EM) trips 6 seconds after scram. Note: The auxiliary pump rectifier will run to the minimum voltage position following loss of power. This will put a large drain on the auxiliary pump battery. - Return primary auxiliary pump rectifier voltage to normal as soon as possible following the 400 kW diesel generator has assumed emergency loads. - Verify the shutdown cooler dampers open when decay heat raises bulk sodium temperature to >710 F. **Power Plant Operator/Electric Plant Operator (PPO/EPO)** - Verify turbine generator has tripped and place the pilot wire trip to off. Place the tap changers to Auto. Press the Off Pushbuttons for the low voltage and high MVAR trips. - Verify emergency diesels start and load. - Verify the blowdown system, and 150 psig Auxiliary Steam supply have tripped. - Manually control feedwater control valve to increase steam drum level to 30 inches.

Item	Item Name	Item Description
		- Shut the main steam stop valve. *Steam Drum Level Indicated: **PPO -** Start the manually to maintain steam drum level, but do not decrease #2 Feedwater Heater level below the range of the local sight-glass. *Steam Drum Level Not Indicated: **PPO -** Verify all feedwater pumps are secured **-** Shut feedwater isolation valves **-** Verify main steam stop valve is shut **PCO -** Secondary Sodium Pump **-** to remain deenergized or deenergized when steam drum level is no longer indicated **Secondary Sodium Operator (SSO)** **-** With direction from the SM dump the steam generator water **-** Vent the secondary Argon Cover Gas by pushing the Sodium**-**Argon Vent Valve Open pushbutton **-** Drain the Secondary Sodium System **-** Press the drain pushbuttons system will be dumped to Secondary Sodium Drain Tank and allowed to cool to ambient **-** Deenergize the secondary sodium heating system **-** Place secondary sodium system in dry layout per standard operating instructions **Subsequent Actions:** **SM -** Determine the cause of the loss of normal power and if possible the expected duration of the power outage. **Field Coolant Operator (FCO)** • Determine if the shield and thimble cooling system are in operation • Switch the large rotation plug freeze seal heaters to emergency power (400 kW Diesel Generator) IAW Standard Operating Procedure (SOP) if normal power is not restored within one hour. • Note: If 400 kW is not operation, arrange for temporary power to the seal

Item	Item Name	Item Description
		heaters from the portable 125 kW Diesel Generator **Fuel Handling Crew -** if a subassembly requires cooling and is in either the Fuel Unloading Machine or an Interbuilding Coffin (IBC), verify that adequate cooling is being provided. If necessary, follow the applicable EOP for "Subassembly Cooling Emergency or IBC Cooling Emergency. **PPO/EPO** - energize the manually applied emergency loads as necessary for given plant state. Note: Do not overload the 400 kW Diesel. - The following are the loads most likely to be required: - Site Evacuation Siren - Turbine Generator turning gear - Reactor building air supply fan - Isolate condenser air ejector steam and blowdown system - Monitor condenser hotwell level. If level increases, isolate the makeup control valves **SSO** * If the secondary sodium system has not been drained perform the following: - Monitor the secondary sodium pump winding temperatures - If the winding temperatures increase to greater than 275 F, cross-connect the raw water system to the plant cooling water system to supply cooling water to the windings. - Admit cooling water to the secondary pump very slowly (3 gpm initially, then increase by no more than 3 gpm each minute until 20 gpm is obtained) to prevent thermal shock - Monitor the sodium level in the surge tank. If the level is decreasing make a system valve check to determine source of the leak and stop leakage if possible. Note: The recirculating pumps may be used to refill the surge tank by energizing them from emergency power. - If the temperature of any part of the secondary system (exclusive of drain and vents) reaches 300 F or any drain or vent reaches 275 F, then drain the secondary sodium system per previous drain steps **FCO**

Item	Item Name	Item Description
		If the bulk sodium temperature decreases to below 690 F, initiate the following steps: - Close the shutdown cooler dampers and take other steps (such as covering the dampers or closing off air leakage paths, etc.) to conserve primary tank heat - Investigate the practicality of reducing shield cooling air flow to conserve heat - If possible locate a backup portable generator if the power outage appears to be long lived, to back up the 400 kW generator as a power source for the large rotating plug seal heaters **PPO** Take steps to prevent freezing the feedwater, blowdown, and steam system yard lines if severe cold temperatures exist **Restoration Actions:** Note: Before restoring normal power perform the following steps: **PCO -** Press the Primary Pump Excitation Emergency Stop pushbutton on the primary panel in the MCR. - Verify that the primary pump speed controls have reset to minimum. **PPO** - Trip the following: - Feedwater pumps from steam system panel in the MCR - All the main cooling tower fans from the utilities panel in the MCR - Condensate pumps from the local control panel in the power plant - Both condenser cooling water pumps from the steam panel in the MCR - Open the disconnects for both chemical injection pumps for the main cooling tower and the feedwater system at local control panel in the power plant. - Place both silicon pump control selector switches in the off position on the local control panel in the power plant - When normal power returns restore the electrical switchgear to normal lineup per SOP
15	**Potential Performance**	

Item	Item Name	Item Description
	Shaping Factors	
16	**Sub-functions**	Emergency Procedure compliance, alarm and indicator diagnosis, crew coordination and action sequencing, and place keeping
17	**Execution/Performance requirements**	In-depth knowledge and understanding of Reactor Shutdown logic, anticipated events, and unanticipated complex or compound events. Memorization of immediate actions and familiarity with subsequent actions.
18	**Timing**	Immediate response for this reactor safety event to prevent core damage or damage to safety related systems. Immediate monitoring and response to lowering steam drum level to prevent loosing sight of the level. Immediately verify that the EDGs start and load.
19	**Sequence up/down**	Immediate Down power - reactor scram
20	**Information from system**	Trending data, plant alarms, sight-glasses, and plant parameters from various sources
21	**Information transmittal method**	Alarms come in on plant computer screen and Alarm Annunciator Panels illuminate. Visual local observations of sight-glasses. Standard MRC panel indications.
22	**Termination indications**	• Control rod down lights are on • Channels A,B, and C power channel indicators and recorders show zero reactor power • Channels A, B, and C period meter shows negative period indications • Secondary Electromagnetic Pump trip 6 seconds after scram • Instantaneous Turbine Generator trip
23	**Potential Errors**	Delay in scramming the reactor when complex or compounding events occur due to diagnosis difficulties
24	**Source documents**	EP and system description manuals
25	**Cues to the Operator (for commencement of the action)**	The following automatic actions that are a direct result of a loss of normal power are immediate indications to the crew of the event and what actions they need to follow in order to place the plant in a safe state: - The reactor scrams - Primary reactor coolant pumps (RCPs) stop - Secondary sodium pump and secondary recirculating pumps stop

Item	Item Name	Item Description
		- The 400 and 125 kW emergency diesel generators start and load - All electrical heating of the primary and secondary sodium systems is lost. - All motor-driven pumps in the power plant systems lose power except the turning-gear oil pump, ac and dc seal oil pumps, reactor auxiliary-cooling-water pumps, demineralizer pumps, and emergency feedwater charging pump.
26	**Diagnosis Required**	Automatic Scram is self revealing in nature (see 28) Manual Scrams are a result of diagnosis and action decision
27	**Control and Display Sufficiency**	Control Rod position indications, reactor power, reactor period and alarms are easily viewed on reactor console, reactor panel in front of the console and the plant computer. Power plant indications associated with the turbine generator and off-site power distribution are easily viewed on the electrical panel in the MCR. Some power plant pumps only have local indications requiring the PPO to go to local indications to validate correct system response.
28	**Feedback on the Operation**	• Control rod down lights are on • Channels A, B, and C power channel indicators and recorders show decreasing reactor power • Channels A, B, and C period meter shows negative period indications • Secondary Electromagnetic Pump trip 6 seconds after scram • Turbine Generator trips • Blowdown System trips • Auxiliary Steam Cross Connect trips
29	**Recovery Opportunities if Omitted**	If reactor did not automatically scram but alarms and indications dictate scram, manual scram can easily be performed by pushing Scram Pushbutton on Reactor Console. If Turbine Generator does not trip, PPO/EPO can manually trip from the electrical panel in the MCR. If the secondary sodium pump does not trip the PCO can trip it from the console.
30	**Consequences of Failure/Non-recovery**	See Operating Experience Item 31 - numerous test performed at EBR-II indicate in most cases that even without a scram the passive safety systems and self-protecting nature of the reactor design are sufficient to prevent core damage. However other plant assets will not be protected without operator intervention.
31	**Technology Recommendations**	• Evaluate additional failure modes to be added to the Reactor Safety System logic and operator support systems to aid in rapid diagnosis of abnormal and emergency conditions • Have digital display on computer monitor for all power plant systems and components. • Provide a computer display of off-site substations and disconnects, and diagnostic capabilities that enhance the ability of the EPO to predict a coming

Item	Item Name	Item Description
		fault on the off-site power distribution system. • Automate secondary sodium pump to place it on alternate power supply. • Provide remote indications and control for both secondary and primary sodium piping heating systems. • Provide safety critical systems (see section 4) monitoring on one display on plant computer.
	Operating Experience	**Reactor Operation is Straightforward** • At EBR-II the reactor was operated with many different core configurations. In all configurations, the reactor was stable in its operation. • Another aspect is that operating procedures are straightforward, aided by the self-protecting nature of the reactors. • At EBR-II, extensive tests were conducted that not only included ATWS events associated with loss of flow, but also – single rod run-out, – primary pump control malfunctions, – load following and – steam system failures • These tests also demonstrated that EBR-II was tolerant of operators taking an improper control action. • These characteristics greatly reduced pressure on operators in the event of off-normal events. Rapid operator response was not required. **Safety Testing in EBR-II** • Fuel was extensively tested under off-normal conditions – Operation with breached cladding (oxide and metal) – Transient overpower (TOP) events (EBR-II TOPs complimented more severe TOPs in the TREAT reactor) • Inherently safe response of EBR-II was demonstrated after 12 years of extensive testing and analysis – Loss-of-flow without scram (station blackout) – Loss-of-heat-sink without scram • A level one PRA was completed to quantify safety

Item	Item Name	Item Description
		J. I. Sackett, C. Grandy, "International Experience with Fast Reactor Operation & Testing", International Conference on Fast Reactors and Related Fuel Cycles, Paris, France - March 4-7, 2013

9.3 Appendix C: WDA Diagrams and Discussion

This section contains the graphical results as well as the descriptions of the Abstraction Hierarchies, Contextual Activities and Strategies for the scenarios described in Appendix B: Operating Scenarios.

9.3.1 Normal Operations

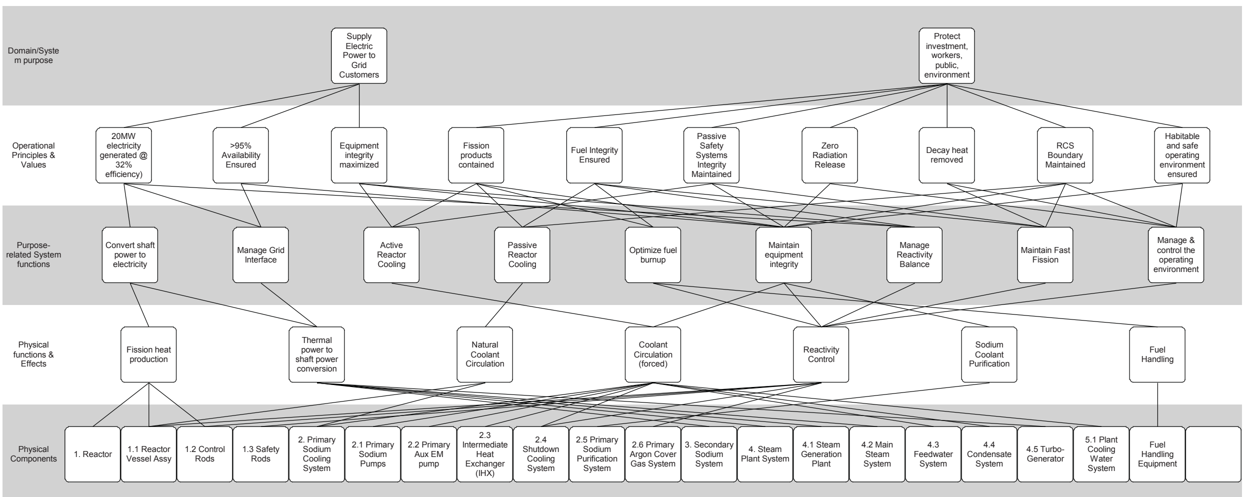

Figure 12: Abstraction Hierarchy - Normal Operations

The Abstraction Hierarchy (AH) in Figure 1 shows the minimum primary and secondary systems that are required to perform the following physical processes:

- Fission Heat Production
- Reactivity Control
- Thermal power to shaft power conversion
- Coolant Circulation
- Sodium Coolant Purification
- Fuel Handling

The AH also shows the most important functions that are supported by the physical processes. The top level of the hierarchy shows that two primary missions are served during normal operations – the production mission (supplying electric power to the grid), and the safety mission (protecting the assets, workers, public and environment). The means to measure whether those high-level goals or purposes were achieved are indicated in the second level – where possible the measures are indicated in quantitative terms, but most measures are shown as qualitative values.

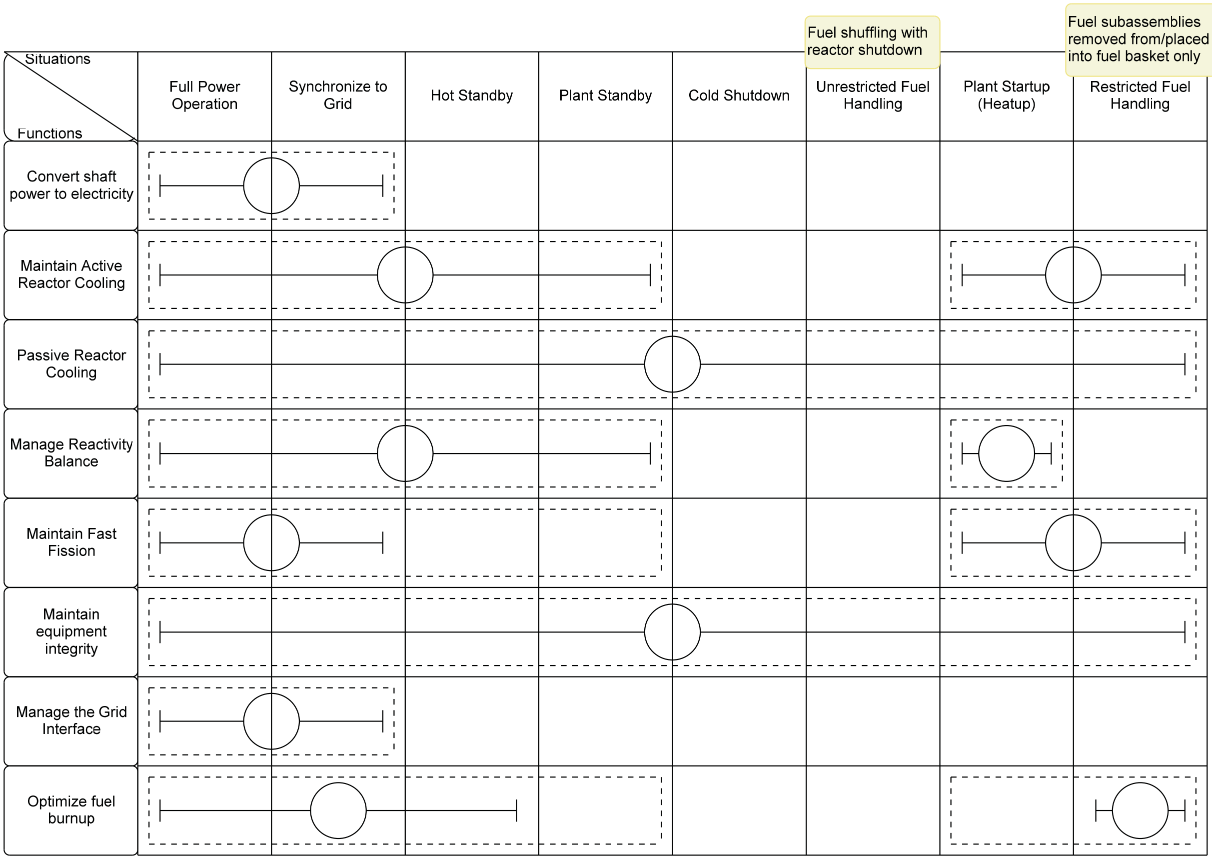

Figure 13: Contextual Activities - Normal Operational Modes

Figure 3 describes the functions that apply during the identified operational conditions. The rows in the diagram show the applicable functions and the columns show the different operational conditions that are possible as the plant transitions from the lowest operational state (for example, "Unrestricted Fuel Handling") to the highest state (for example, "Full Power Operation"). The marked cells in the grid indicate the correspondence between functions and conditions, as follows:

- An empty cell indicates that the function is not possible in that condition.
- A cell outlined with a dotted line indicates that the function could occur in that condition, but typically does not.
- A cell marked with a circle and whisker indicates that the function can and typically does occur in that condition. The circle/whisker can span several cells, indicating all conditions that apply.

Examples:

Function	Explanation
Convert shaft power to electricity	Electricity is only produced during full power operation and when the generator is synchronized to the grid.
Maintain Active Reactor Cooling	Active cooling is required during all conditions, except during Cold Shutdown and Unrestricted Fuel Handling.
Maintain Equipment Integrity	This function is always required, regardless of the

Function	Explanation
	condition.
Optimize Fuel Burnup	Optimal fuel burnup is not possible during Cold Shutdown or Unrestricted Fuel Handling. All other conditions will allow optimal burnup.

In combination, the marked cells in the Contextual Activity diagram indicate the underlying constraints of the particular operational mode.

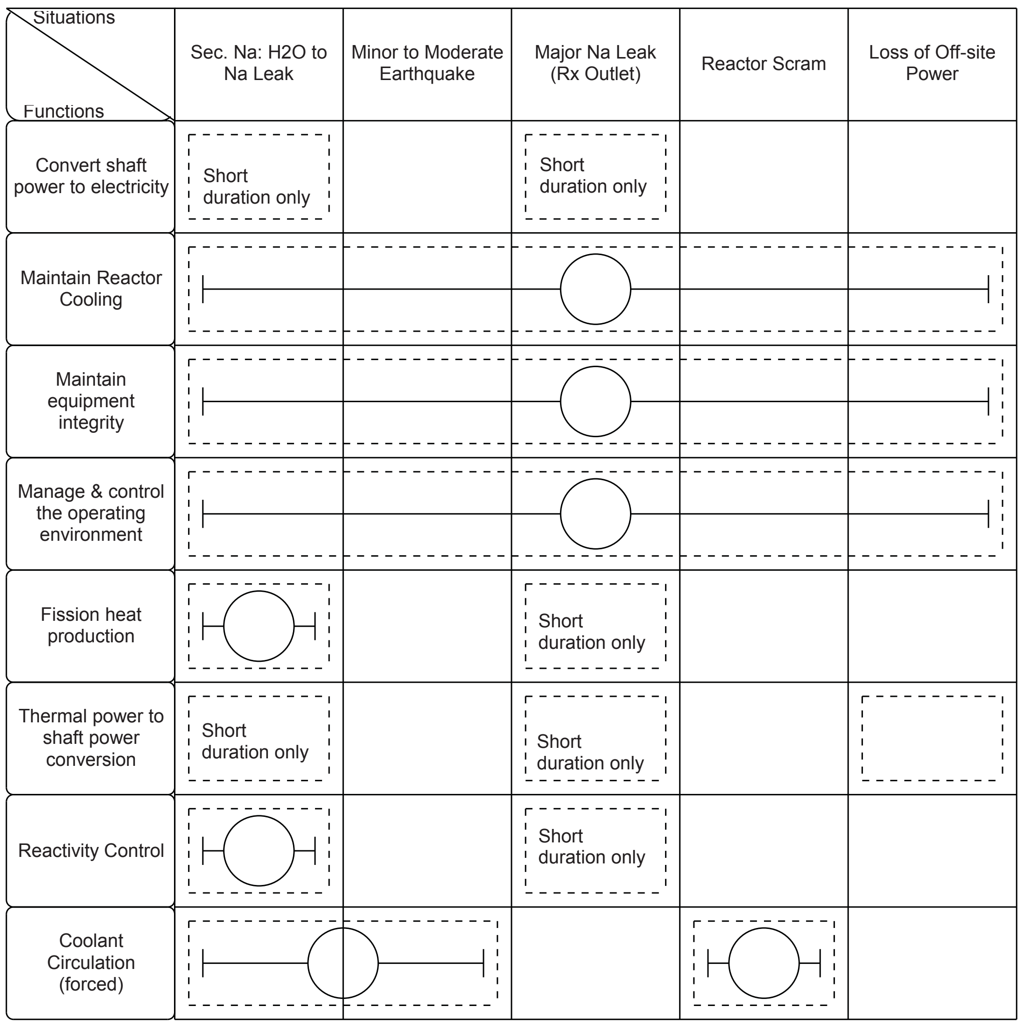

Figure 14: Contextual Activities - Anticipated Operating Occurrences

Figure 14 indicates the functions that apply during five typical Anticipate Operating Occurrences (AOOs). Four of these AOOs were analyzed in the operational scenarios described before and in the diagrams that follow.

Of interest in this diagram is the short duration of some functions in specific conditions. For example, it may be possible to produce power for a short duration with a water-to-sodium leak or with a major sodium leak until the reactor scrams.

Primary Sodium Cooling System analysis

A separate analysis was done for the Primary Sodium Cooling System because it includes the most important subsystems involved in controlling reactor temperature.

The AH for the system is shown in Figure 15, with the Contextual Activities in Figure 16:

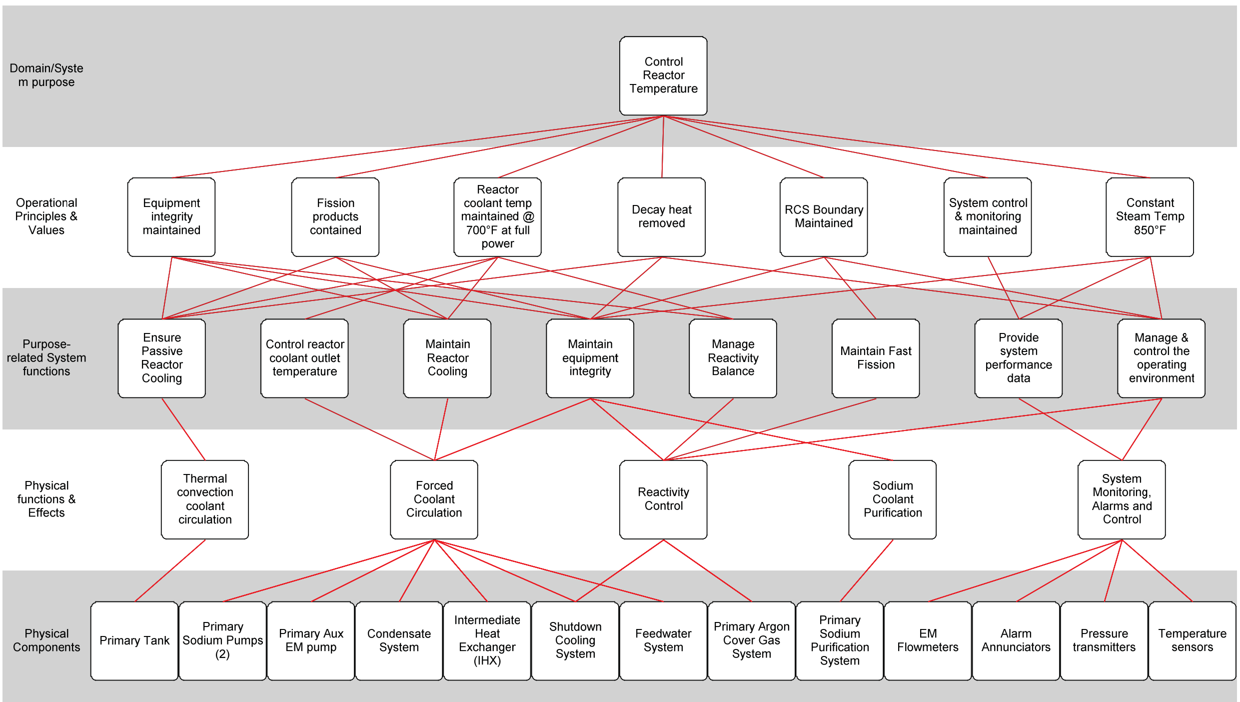

Figure 15: Abstraction Hierarchy - Primary Sodium Cooling System

As seen in Figure 15,the ultimate purpose of the Primary Sodium Cooling Systems is to control the reactor temperature.

The Contextual Activities shown in Figure 6 indicate the functions that are applicable during the various plant modes, including restricted and unrestricted fuel handling:

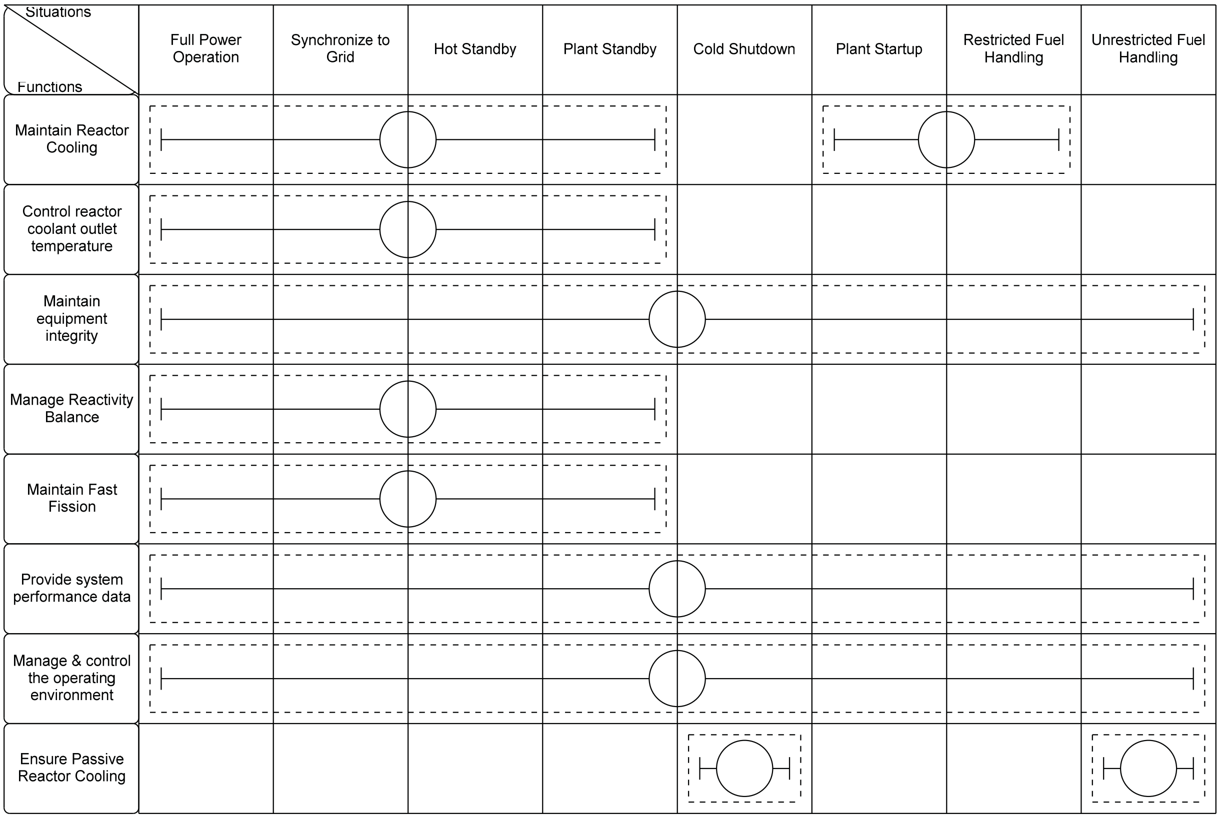

Figure 16: Contextual Activities - Primary Sodium Cooling System

All AH diagrams that follow will apply the same paradigm, except that for abnormal operations, the top level describes the intended recovery condition after necessary mitigation actions were taken, for example, “Establish Safe Plant Condition”.

In addition, the event scenario diagrams include Contextual Activity and Strategies Analysis diagrams that describe the conditions, constraints, and alternative recovery strategies for those events.

9.3.2 Event Scenario 1: Loss of Normal Power

Loss of normal power is defined as the loss of power from the grid or the EBR-II turbine generator to the 13.8 kV bus. Many changes occur automatically: 1) the reactor scrams automatically, 2) the main primary pumps stop, 3) secondary sodium and recirculating pumps stop, 4) all electrical heating of the primary and secondary systems is lost, 5) 400 and 125 kV diesel generators start and load, 6) all motor driven pumps stop, except equipment supplied from UPS, e.g. turning gear oil pumps, seal oil pumps and the emergency feedwater charging pump.

Figure 17 shows the systems and functions that are involved in the event and in achieving a safe plant condition. Figure 18 describes the various possible emergent conditions during the event.

Figure 19 and Figure 20 describe the various alternative strategies to recover from the condition, with or without scram.

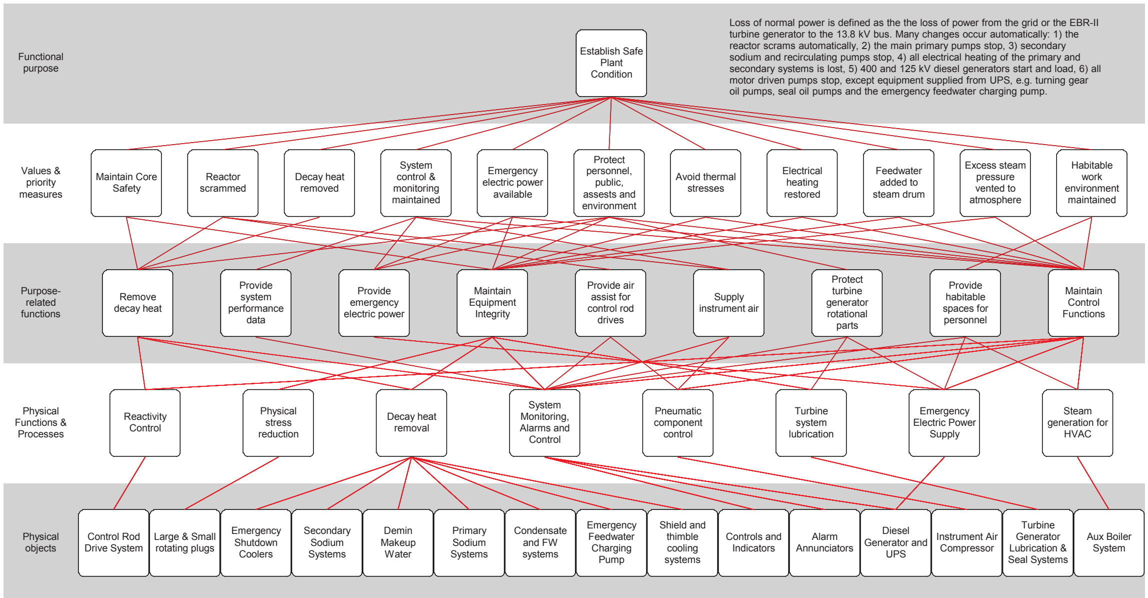

Figure 17: Abstraction Hierarchy - Loss of Normal Power

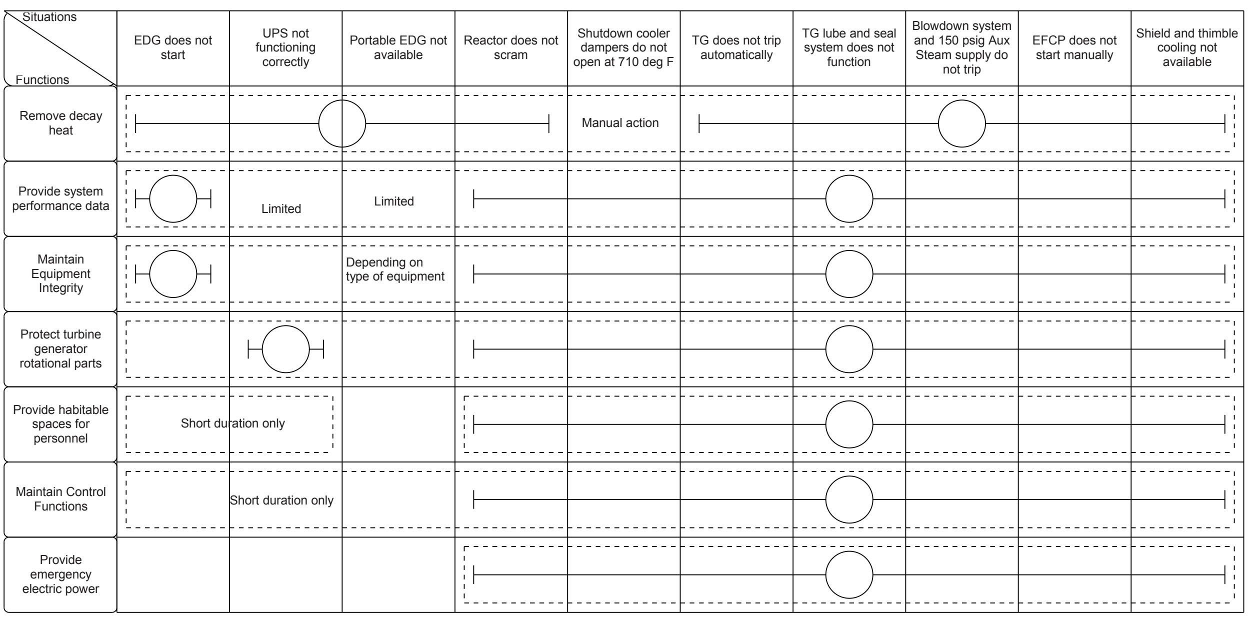

Figure 18: Contextual Activities - Loss of Normal Power

The most critical action after a loss of normal power is to ensure that emergency power is available to the cooling systems. For this purpose both Emergency Diesel Generators (EDGs) should start immediately. If one or both EDGs should fail to start, operators will attempt to identify and correct the fault as soon as possible. If this still fails, they will start the portable EDG to restore power to equipment essential for protecting assets such as the turbine generator.

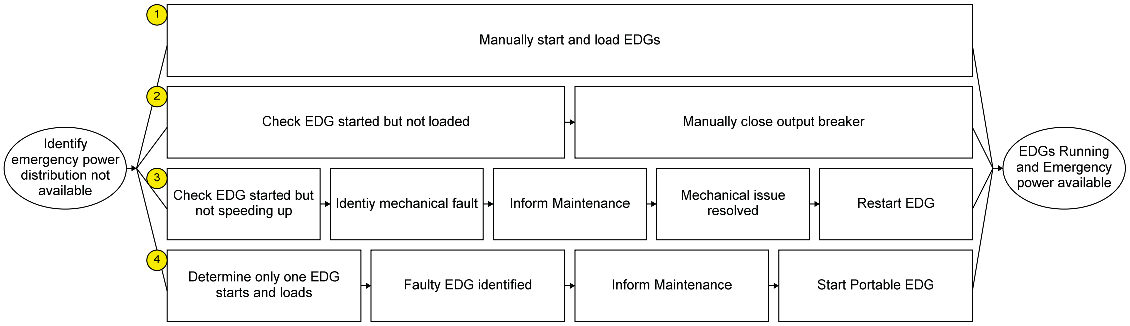

Figure 19: Strategies Analysis - Mitigation of autostart failure of EDGs

The loss of power may affect a number of systems that will require a shutdown. If for some reason automatic scram fails to actuate, there are various strategies to ensure scram, depending of the severity of system failures, such as failed Control Rod Drive Mechanisms.

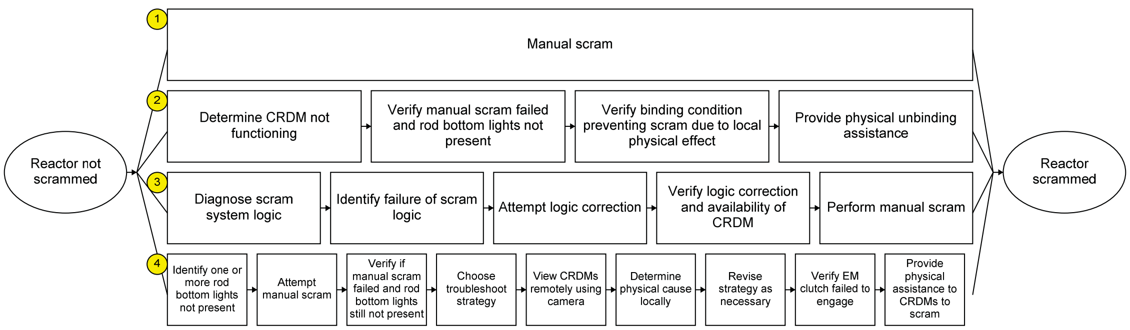

Figure 20: Strategies Analysis - Mitigation of failed automatic scram

A potential emergent condition during or immediately following a loss of normal power is a failure of the reactor to scram automatically. A typical cause would be malfunctioning of the control rod drive mechanisms. Figure 20 shows four possible strategies to scram the reactor.

9.3.3 Event 2: Water-to-sodium leak

A water-to-sodium leak results in a reaction that creates heat and liberates hydrogen, causing rapidly increasing temperature and pressure in the affected evaporator or superheater. The magnitude of the temperature and pressure changes depends on the size of the leak. Localized high temperature and pressure could cause failure of evaporators or superheaters.

The analysis below indicates that the ultimate objective is not only to protect workers, public and the environment, but also to prevent damage to equipment.

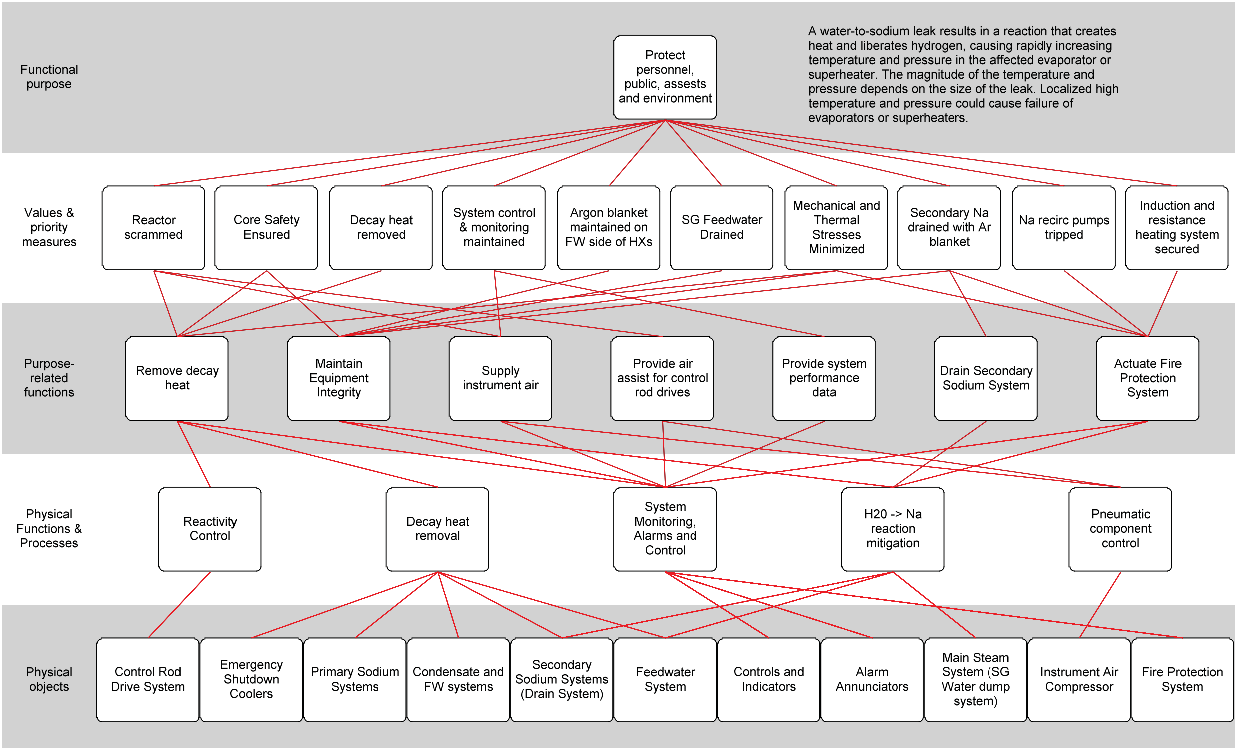

Figure 21: Abstraction Hierarchy - Water-to-Sodium Leak

Figure 21 highlights the typical ultimate purpose of any strategy to mitigate a condition that challenges the safety and integrity of plant systems. Protecting workers, public and the environment is necessary in the unlikely event of a radioactive release, but more typically actions will be taken to protect investment and prevent any emergent condition that may damage plant equipment.

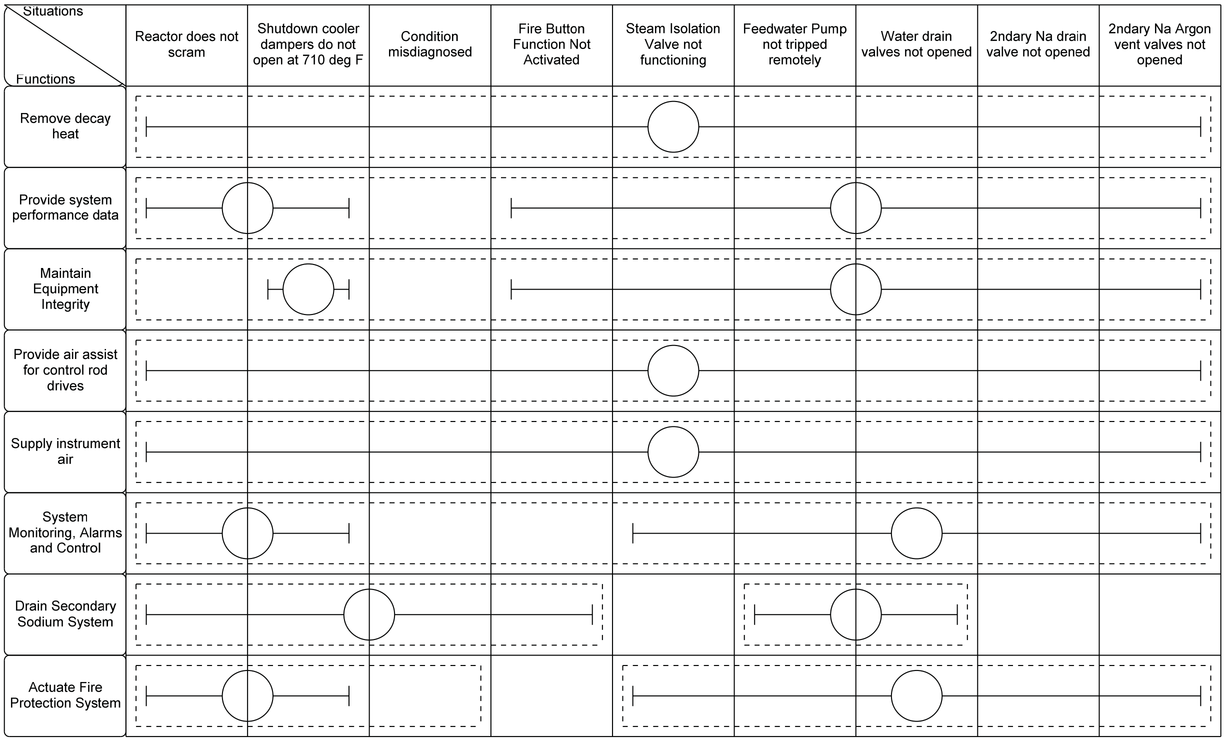

Figure 22: Contextual Activities - Water-to-Sodium Leak

Figure 22 shows the most important functions that are involved in several emergent conditions. Figure 23 below describes five alternative strategies to prevent a water-to-sodium reaction or to recover from the condition.

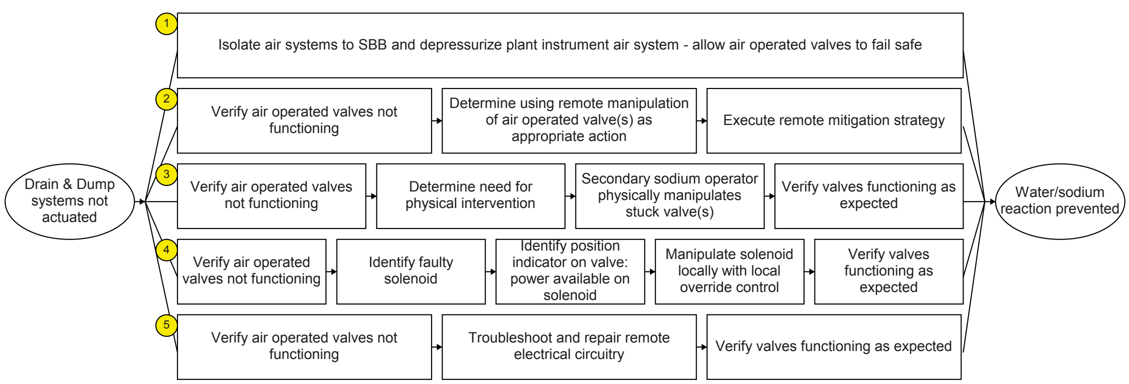

Figure 23: Strategies Analysis - Mitigation of Drain Systems Failure

However, depending on the severity and also the location of the condition, it may be possible to misdiagnose the exact nature of the condition. Figure 24 describes alternative strategies to ensure that the condition is diagnosed as quickly as possible to recover from the condition or to minimize the consequences of a water-to-sodium reaction.

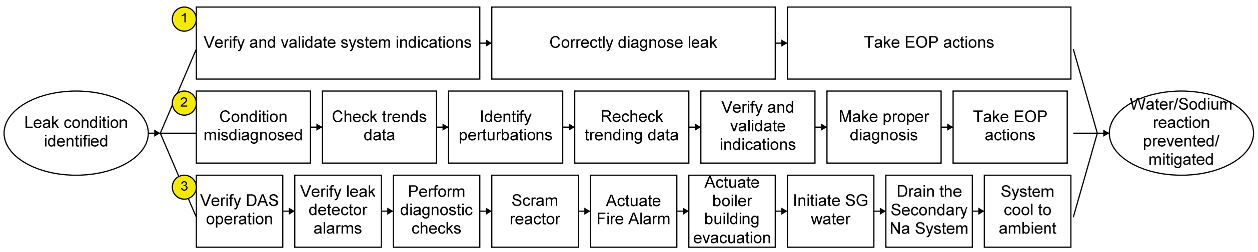

Figure 24: Strategies Analysis - Mitigation of misdiagnosed condition

9.3.4 Event 3: Earthquake (minor to moderate)

An earthquake presents two hazards: 1) reactor shutdown may not be possible if control and safety rods or the drive mechanisms bind, and 2) piping systems may fail, releasing water, sodium or high-pressure steam. The earthquake detection system scrams the reactor (control and safety rods). Operating action involves ensuring the reactor has scrammed and placing plant system in as safe a condition as possible. Depressurization and draining of some systems may be desirable. An earthquake may also cause a loss of electrical power, leading in severe cases to a station black-out.

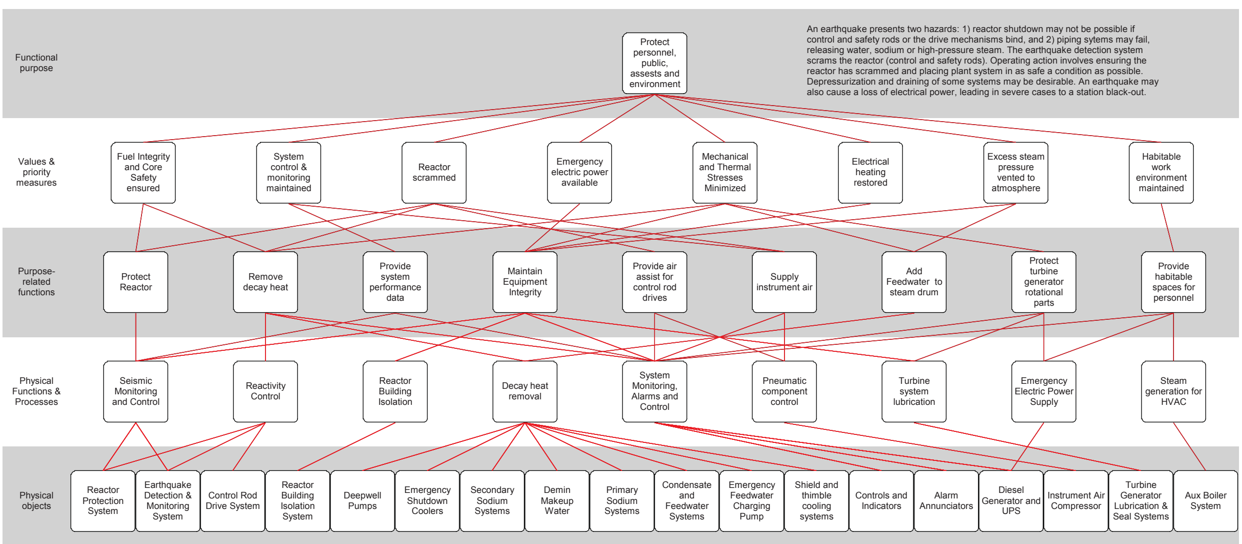

Figure 25: Abstraction Hierarchy - Minor to Moderate Earthquake

The response to an earthquake requires immediate actions to prevent or mitigate the results of equipment damage that might lead to a radioactive release. For this reason, like with loss of power, the ultimate purpose is also to protect workers, public and the environment. In addition, much attention is paid to actions necessary to protect investment and prevent any emergent condition that may damage plant equipment. Figure 24 and Figure 25 show how equipment damage from an earthquake might affect the availability of several functions, for example, the ability to supply emergency electric power to essential equipment.

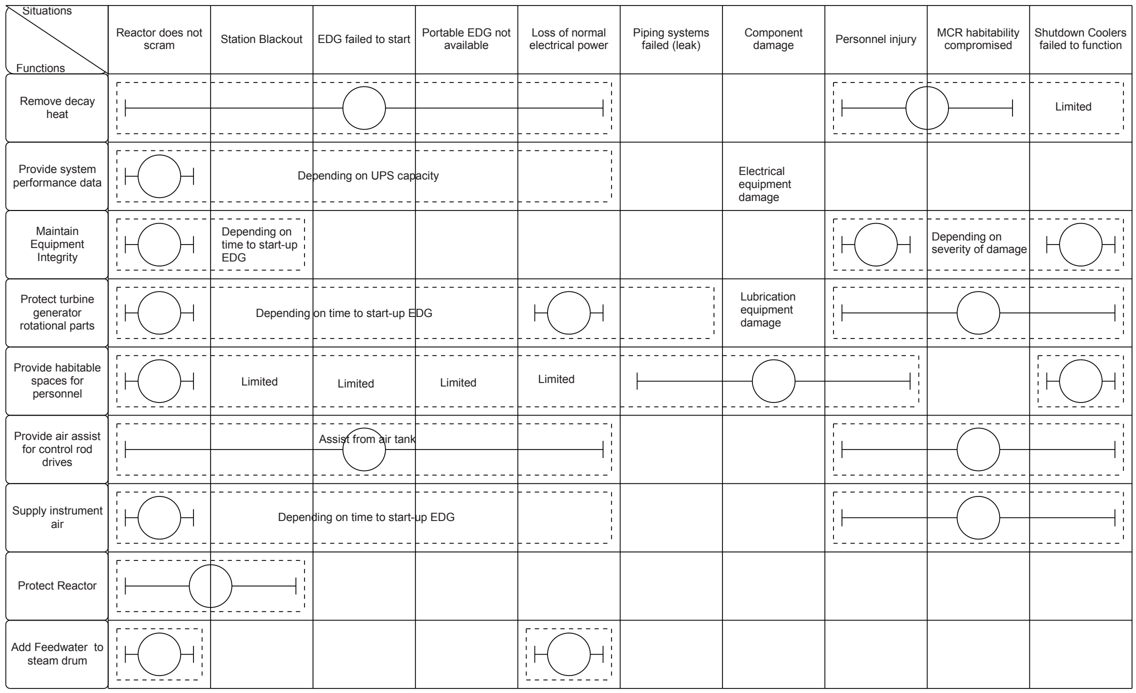

Figure 26: Contextual Activities - Minor to Moderate Earthquake

Three specific emergent conditions are identified:

1. EDGs fail to start automatically (same as Figure 18 above), leading to actions to attempt a manual start, or in worst cases to start the portable diesel generator. As shown in Figure 26equipment damage may severely affect the availability of certain functions. For example, if EDGs fail to start, emergency power from uninterruptible power supplies (UPSs) will be available only for a short duration.
2. Reactor does not scram automatically (Figure 27), leading to four possible strategies to scram manually.
3. Evacuation of certain plant areas, coupled with strategies to identify and recover injured personnel.

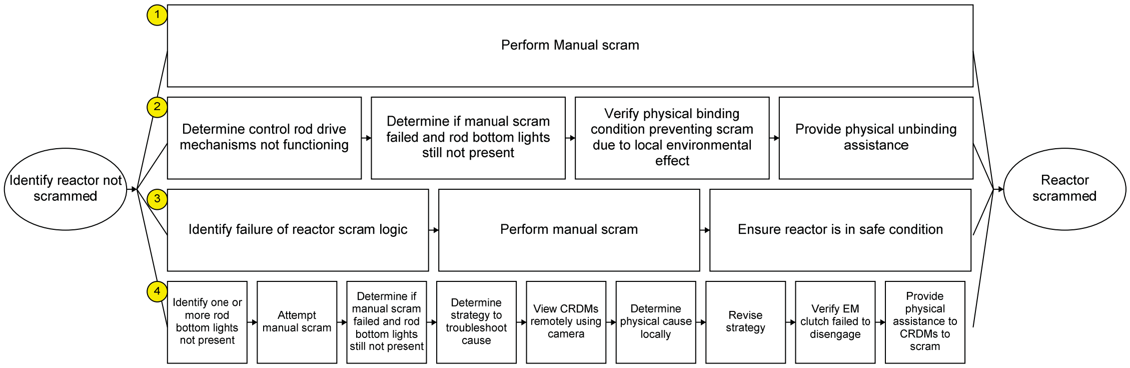

Figure 27: Strategies Analysis - Earthquake - Mitigation of Failed Automatic Scram

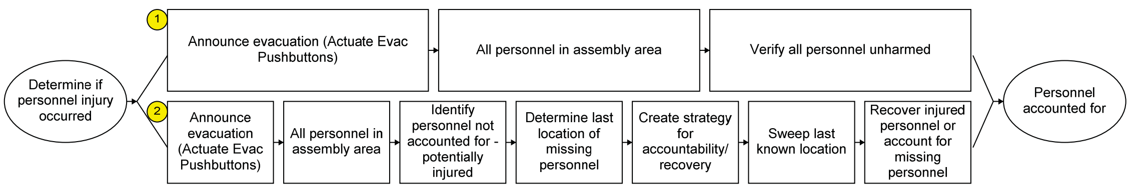

Figure 28: Strategies Analysis – Earthquake - Mitigation of Personnel Injury

9.4 Appendix D: EBR-II Performance Shaping Factors

This appendix describes the analysis of the Performance Shaping Factors (PSFs) identified for the following EBR-II operating scenarios:

1. Reactor Scram (Manual or Automatic)
2. Secondary Sodium Systems – Water-to-Sodium Leak
3. Minor to Moderate Earthquake
4. Loss of Normal Power

9.4.1 PSF Analysis Method

A human reliability analysis (HRA) and human factors analyst worked with a former EBR-II operator and subject matter expert (SME) to review the PSFs associated with the execution of operator actions. The SME had experience in conducting and designing evacuation exercises at the plant which provided valuable insights for at least two of the four scenarios; loss of normal power and response to earthquake conditions. In addition, he was able to identify instances where equipment access and information was critical in terms of accident mitigation and response. Additionally, the original EBR-II technical specifications, PRA, and procedure sources were available to aid in the analysis. The use of PSF analysis as an adjunct to the CWA was found to be a valuable approach in identifying areas of strength and weakness in the EBR-II design and operational concepts. However, the findings also need to be interpreted in light of the technology available to operating crews at the time and what was the norm for the nuclear industry as well.

9.4.2 Identification of Performance Shaping Factors Used in the Analysis

Four publically available and widely used HRA sources were consulted for use in the present WDA report; NUREG-1792 (HRA best practices), NUREG-6883, SPAR-H Human reliability analysis, NUREG 1784 (Fire PRA) because of the potential water sodium interaction described in Scenario 3, and NUREG-1278 (THERP). A mapping exercise was performed by human factors practitioners with the result that 11 PSFs used to characterize emergency accident conditions at EBR-II were selected as follows:

- *Procedures* - includes quality, availability, accuracy, relevance to context, and use of formal procedures in plant response
- *Training* – includes experience, expertise, and suitability of training to the scenario under review
- *Teamwork* - includes communications requirements and provisions for: lock-out/tag-out expertise, coordinated activities both within and outside of procedures
- *Fitness for Duty* – including aspects of fatigue, injury, exposure to heat or cold for long periods of time, fatigue factors
- *Cues and Feedback* – prominence of cues, i.e., the salience, quality, timeliness, availability of trend information, time from initiating event or other events until first alarm
- *Complexity* – presence of multiple faults, parallel tasking, difficult sequencing, masked symptoms

- *I&C and Ergonomics* – operator interface, need for plant protective equipment (PPE), special tools, ingress and egress
- *Operator workload* – includes stress, mental physical aspects or both including instance where crew has knowledge of impending time constrains
- *Timing* – includes time available and time required, travel time, drain down time, importance of timing in event progression and event response
- *Staffing and Resources* – including expectations that CR staff is to be increased or decreased, or staff capabilities are likely to be stretched
- *Work Environment* – includes fire, flood, seismic, high temperatures, inert gas environments

Since this was the first documented attempt at a PSF analysis for EBR-II, the analysts sought to determine whether the PSF analysis was sensitive to differences between the scenarios. As it turns out, the semi-qualitative analysis method employed demonstrated this sensitivity fairly well. In terms of measurement, a 7-point Likert scale was employed as follows:

Rating	Meaning
1	Strongly hinders performance
2	Moderately hinders performance
3	Weakly hinders performance
4	Neutral or negligible influence
5	Weakly helps performance
6	Moderately helps performance
7	Strongly helps performance

The scores represent consensus among the experts. Since this study was limited in the number of experts participating in the analysis, it was thought that the ordinal data collected on the PSFs would be sufficient, in fact given the extreme nature of these scenarios and the inherent uncertainty in modeling the expected cognitive burden and stress upon the crew, a Likert scale is probably appropriate. Given this, we took this qualitative shaping factors analysis and attempted to place it within the contextual physical and cognitive environment in which the operators would find themselves. This context was defined in terms of what had to be done to protect the plant, what was required by procedures, and the expected availability of systems, information and communication channels.

The four scenarios analyzed were:

- General reactor SCRAM
- Loss of Normal Power
- Sodium to water leak
- Minor to Moderate Earthquake

9.4.3 Scenario 1: Reactor Scram

The most highly rated PSFs for operator response during reactor SCRAM were Procedures, cues and feedback. EBR-II operators were supported by clear well formatted procedures on Reactor SCRAM; it was a basic prerequisite in both licensing and training. Additionally, the conditions for initiating SCRAM including times when automatic SCRAM may fail and an operator response was required were well documented. The procedure was also detailed in terms of effectively considering multiple personnel. Training was positive, but no simulator was available to support operator training on SCRAM; this will be rectified with any new licensed design. Training was effective because of the yearly written exams and oral boards that was part of EBR-II operator training and qualification program.

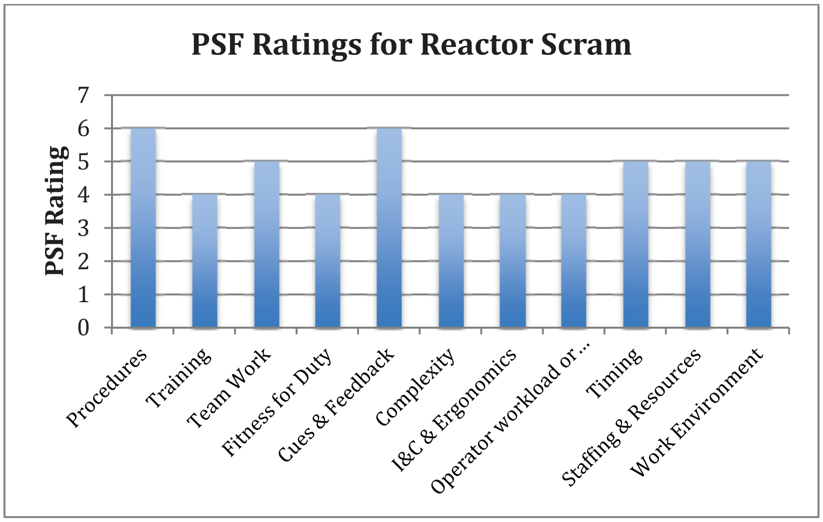

Figure 29: EBR-II PSF Analysis: General Reactor Scram

Team response for SCRAM potentially involved a large complement of personnel from the field - - SSO (secondary sodium operator), FCO (field coolant operator), PCO (panel coolant operator) and the PPO (power plant operator), EPO (electric plant operator) as well shift foreman, shift manager and the reactor operator who takes action in the control room. Although the crews did very well during normal operation, and communicated effectively, there was little documentation on how they performed in terms of communication for real or simulated events. The designation of roles and responsibilities and communications during response to reactor SCRAM were well defined; the shift manager (SM) had the responsibility to monitor plant response and initiate outside communications. The shift foreman (SF) monitored and directed communications between crewmembers including those outside of the control room and acted as the on-scene-commander for most events. Fitness for duty was not a factor in terms of response to plant SCRAM and therefor a score of “4” neutral or negligible is assigned. There was a chance that crew shift length could be extended due to Idaho weather, however, the crews typically were in good physical health including partaking in many outdoor activities year round, and thus, snow that could potentially extend a shift was not an overwhelming concern. Further, SCRAM response was taken in the control room and unlike scenarios involving earthquake or flooding, out of control room actions were not anticipated for immediate actions.

Operator cues were moderately helpful in influencing operator response and rated “6”. The cues presented to the operators were multiple; rod bottom lights, scram alarms, and other cascading alarms. Multiple sensors were present, and rapid feedback on plant conditions was present. The independent cues included: rod bottom lights, negative reactor period, nuclear instrument power at zero, secondary pump trip, and turbine generator trip. This was rated as “6” and not higher because it represents the industry standard at the time and while informative, the indication is good but not exceptional. Complexity, is neutral during this type of event and is assigned a “4”, there are no multiple faults, masked symptoms or sequence difficulty to consider. Even though some tasking was sequential and others were concurrent among crewmembers, there was no appreciable impact expected to act toward helping or hindering operator response.

Instrumentation and control and ergonomics was only weakly positive (4) because the multiple indications available to operators at EBR-II were not co-located with one another, i.e., on the same panel. The same was true for controls - the ergonomics were adequate.

Stress is present in any plant event, but in this case the internal operator response would likely have been one of concern regarding ensuring that the core was adequately protected. In terms of mental workload, the amount of workload was not a factor. There was moderate visual search required to validate the scram and the key event indicators such as control and safety rods in full down position, and reactor power at 0%, were obvious to crewmembers. Alarms were present on annunciator panels and the plant computer, so it was not difficult for the control room crew to determine the critical information. There were no potentially physically demanding tasks to perform. In this instance, anticipated crew stress was determined to be “4”, negligible in terms of impact upon verifying the automatic scram or in hitting the single manual scram button prominently in the main control room.

The time for the crew to respond was adequate and even if the automatics should fail, the actions to be taken were straightforward. Time is rated “5” and was weakly helpful for this event; there was adequate time. Multiple well-trained staff were needed for response and all were expected to be in the control room or close by. Unlike other scenarios, no additional staff or equipment were required to respond to the scram conditions. Staffing was assigned a “5” as it was helpful factor but not overly influential.

9.4.4 Scenario 2: Loss of Normal Power

In this scenario the analysts considered the Loss of Normal Power that is provided by the INL grid loop. A further assumption was that critical systems would continue to operate as emergency electric, uninterruptable power supplies (UPS) units, and DC battery power are assumed to be available. For example, auxiliary pump, instrument air, emergency lighting, demineralized water and the secondary sodium pump were available, and the secondary sodium pump would be on alternate power when it was switched over. The operators were aware from their training that they were to conserve steam generation system water and make sure that emergency power was functioning properly. The most highly rated PSFs for operator response during a Loss of Normal Power were procedures and teamwork. The most negative PSFs were complexity and workload. Complications could arise if the automatic scram function failed and the operators delayed in taking action to perform a manual reactor scram. Loss of Normal Power was also associated with an earthquake but there were so many other factors associated with earthquake conditions that this was considered in a separate scenario.

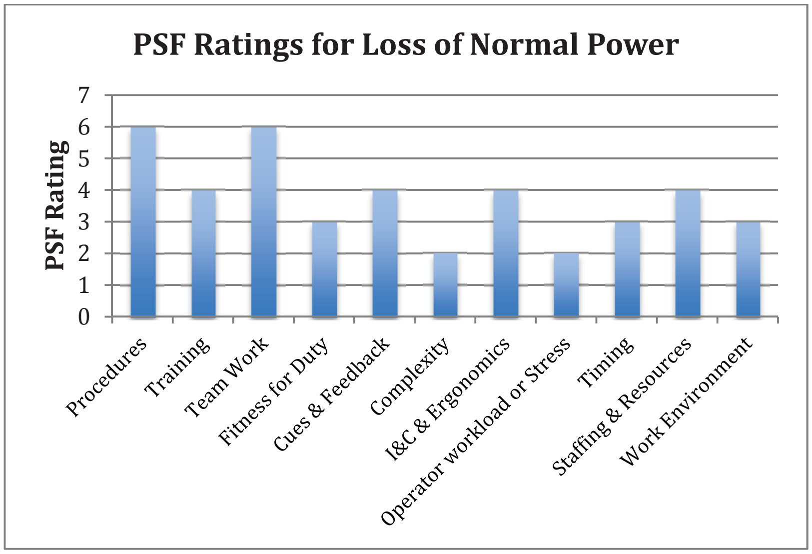

Figure 30: EBR-II PSF Analysis: Loss of Normal Power

In general, the PSF ratings when compared to the Reactor scram scenario were more negative for Loss of Normal Power. This is consistent with the operator uncertainty regarding what systems would function, when power would be restored, sheer amount of work to be performed and just the general severity associated with loss of normal power.

Procedures received a rating of "6" because they were clearly written, did a good job of considering multiple personnel and the requirements, transition to site-wide procedures for emergency action management, were well laid out, and straightforward. Training was rated as neutral, there was no hands-on training available for Loss of Normal Power and no simulator was available for EBR-II. However, training in general was well administered and combined yearly exams, oral boards, and memorization. The shift foreman participated in site-wide emergency training and had experience in assuming the responsibilities of the on-scene commander. Emergency response personnel also drilled on a regular basis.

Teamwork figured prominently in crew response and was at least as important as other shaping factor influences and for normal operations, EBR-II had a history of good team work and assignment and execution of roles and responsibilities. The full crew complement was involved in the Loss of Normal Power response. The SM and SF monitors plant response and directs the communications among crewmembers. As a historical note, the crew response during the 1990's Western States rolling blackout was executed without complication. However, there was no operating experience with actual or simulated events for the analysts to review so training was rated "6", helpful. With more information or actual exercises, it could have been rated "7".

Fitness for duty was not a factor in terms of response to plant SCRAM and therefore a score of "3", hindering was assigned. There was a chance that crew shift length could be extended due to Idaho weather conditions and there was high potential for outside actions in these conditions based upon the length of the outage and the cause of the event. Multiple alarms associated with the event, and credit was given for the main control room (MCR) staying on during the course of the event. Rod bottom lights and scram alarms were assumed to be present in the MCR. Trending information is a different story: scram trends would be available on the plant computer and in multiple locations within the MCR, however, power distribution fault indication was independent of that emergency source and it is expected that fault trend information will not be available to the supervisor and crew. Cues are strong and it is assumed that operators will perceive, comprehend their meaning and predict where the plant is headed. These independent sources included: rod bottom lights, negative reactor period, zero power readings on the nuclear instrumentation, secondary pump trip, and turbine generator trip. No mental calculation by the crew was required. Cues and feedback were helpful, but offset by some of the loss of trend information, the corresponding PSF for cues and information was assigned a rating of “4”.

Although the event is universal within the nuclear industry, considered an emergency condition, and how to respond is at the forefront of every trainer’s mind, the potential for the event to be relatively complex in terms of information uncertainty was always present. Complexity has been rated “2” because it was judged to be moderately hindering in terms of expected crew response. The list of what can make many Loss of Normal Power events is relatively long: simultaneous loss of power and scram, masked symptoms, multiple personnel required for response, multiple procedures in effect, infrequently used Site Wide emergency procedures in effect, and requirements for parallel and sequential task execution, among other factors.

I&C and Ergonomics were rated neutral, i.e., “4”, mainly because the ergonomics of the control room were generally good. However, for this scenario, although multiple indications from redundant instrumentation were available. Futhermore, individual elements were not co-located, i.e., information and control are located across various panels. As stated in the introduction to the event, electrical trending information and fault data were not available, however, what was available was sufficient for meeting performance requirements for this event, given it was no more complicated than described in our scenario description. The operator accommodation to the layout, although not ideal, was not thought to be related to a predictable decrement in performance.

Operator workload and stress during Loss of Normal Power was present in a variety of ways with the potential to hinder operator response and was rated “2”. Mental workload may be much less than physical workload; there were low visual search requirements to validate loss of off-site power and reactor scram, important indications were obvious to crewmembers. However, many immediate actions to validate the status of plant systems had to be taken outside of the control room and out of doors. Additionally, the shift foreman’s workload would increase as they prepared to conduct oversight for crew response and fulfill the responsibilities of the on-scene commander. It is the analysts’ opinion that during this event there would have been a strong negative and stressful component present, even if there were an automatic reactor scram due to loss of normal off-site power. The crew would be concerned with validating that the core was adequately protected because the consequence of such a situation was of great concern. A smaller stress response would occur when the MCR switched over from normal lighting to emergency lighting. This is based on the experience of our EBR-II SME. Finally, increase in Shift Foreman's mental stress level was expected due to on-scene commander responsibilities and uncertainty regarding site-wide conditions.

Time available was thought to be slightly hindering and is rated “3”. Although the self-protecting nature of the reactor ensured adequate time available to respond, there was likely to be uncertainty in the operator's mind regarding time available. This is because there could be so many variations in what systems, subsystems and processes may be available. The rating was not lower because the actions required were straightforward. Staffing is rated “4”. With the exception of times when rounds were performed, all necessary personnel were expected to be available to help from the MCR. Outside staff and/or equipment may be required for response and was considered because it is known that these services were in close proximity to EBR-II. EBR-II training exercises indicate that the emergency response team would have been available within 10 minutes or less. Drills conducted every year indicate that the mobile EDG could be available within 20 minutes from time of request.

The work environment was expected to negatively hinder operator performance, but to a slight or weak degree. Work environment was rated “3”. It is expected that numerous actions would be required outside the MCR under a wide range of environmental conditions, including the possibility of actions needed for limiting shutdown cooler decay heat removal, mobilizing the mobile EDG and associated cable connections (these actions were not routine in nature)

9.4.5 Scenario 3: Water-to-Sodium Leak

When a sodium to water leak occurred, hydrogen was liberated as a byproduct. Control room personnel worked to protect equipment while ensuring personnel safety. At EBR-II the secondary sodium system was drained in order to mitigate for future interactions, reduce severity, and to allow access to repair the leak. The major indication for this event in the MCR was the Hydrogen Meter Leak Detector (HMLD) and/or Secondary Cover Gas Hydrogen Leak Detector (CHMLD). The operators had to confirm that there were no evolutions that would account for the rise in the trend data for hydrogen. If the leak was in the area of the superheater or evaporator, then there would also be an increase in temperature. With the current EBR-II design all actions needed to be taken by the operators, no automatic actions could be taken to substitute crew responses.

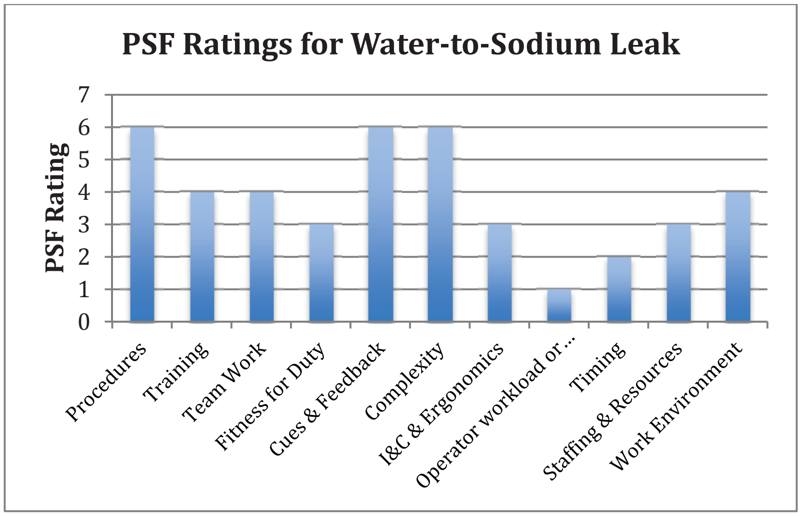

Figure 31: EBR-II PSF Analysis: Water to Sodium Leak

The existing procedures for this event were clearly written and considered multiple personnel effectively. As such, procedures were rated "6". Training was rated "4". There was no simulator training available for sodium response and there was no formal training on water-sodium interaction. We believe this situation would be rectified in any systematic approach to AdvSMR training for sodium fast reactors. Classroom training at EBR-II was good and consisted of a combination of oral boards and written exams. Operators could describe the interaction and potential mitigating actions from memory. Teamwork received a "4" rating. As an aspect of plant response, teamwork was very important. The MCR crew would be in contact and collaborate with field operators. As in other scenarios, the shift manager would be the one to initiate outside communications and the shift foreman would have the responsibility to monitor and direct crew communications inside the MCR. Without more operating experience, a more positive rating than neutral cannot be assigned.

A combination of weather conditions, the potential for longer shifts beyond the 12 hours routinely scheduled, and being subject to outside conditions resulted in fitness for duty being rated "3". For example, weather would not impact the MCR staff, but the tasks for the secondary sodium operator (SSO) could be made difficult through the fatigue and cold endured by weather conditions and possibly, depending upon leak location, time of day, and/or poor visibility. Operator information, feedback and cues were rated "6". Both of the key alarms, hydrogen and secondary sodium cover gas pressure, would actuate within 1-2 seconds of the event. Multiple sensors were present, and trending data were available in multiple locations in the MCR. The alarms that would be present were the HMLD, CHMLD, high secondary sodium pressure, leak probe detector, secondary sodium relief header flow, and/or secondary sodium system rupture disk alarms.

Because the event was easily identified, complexity was a positive influence on crew performance in response to the sodium-water interaction event and was rated "6". There were no multiple faults envisioned for this sequence, no masked symptoms, and all procedural executions were serial in nature. Multiple personnel were involved. Instrumentation, controls and ergonomics was weakly negative and rated "3". The instrumentation of relevant system status information was available, but was distributed among different panels in the control room. Although there were different plant computer screens the relevant information was not adjacent or retrievable through the same screen. On the positive side, the overall ergonomics in the control room was not problematic and evacuation of the SBB was straightforward. However, a number of actions were required outside of the MCR and could be slightly more difficult to access or achieve.

Operator workload and stress was assigned “1” the lowest PSF rating associated with the water-sodium event scenario. The negative performance influence was mental as opposed to physical. There were high visual search requirements to validate the leak. Additionally, there was significant residual cost associated with drain down of the secondary sodium system. The crew knew that if they were in error regarding a leak being present and taking an action that the financial repercussions were extensive, thus, their stress would be very high.

Timing was a negative PSF for this event and was rated “2” for three reasons. There was limited time to respond to the event so that the time pressure felt by the crew was real. Further, the timing of the event related to the location and leak size can lead to severe degradation of plant components and systems. Lastly, reluctance on the part of the crew to make a rapid decision once they have made their assessment can lead to increased damage and cost. Because of the seriousness of draining down the system, we anticipate that they could be caught in a loop of verification and validation before taking action, which would lead to additional complications including further damage.

Staffing and resources was a factor that would slightly hinder response and was rated “3”. Although all necessary MCR personnel were expected to be present, and the SBB evacuation alarm was likely to be activated per procedures, outside resources would be required as part of event response and they did not have training exercises for this scenario. The work environment was neutral and was rated “4”. Control room actions posed no immediate danger to workers inside or outside the MCR and most of the outside actions were pushbutton, workers did not come in contact with the sodium or any conditions that were excessive in heat, cold, radiation, or threat to their well-being.

9.4.6 Scenario 4: Minor-to-Moderate Earthquake

In terms of operator performance during an EBR-II earthquake, three PSFs would hinder performance: complexity, workload, and the actual work environment that was expected to contain debris, potential loss of lighting and other complicating conditions. These three PSFs were rated “2”. Additional significant concerns for the operators were that reactor shutdown may be compromised if control rod drive mechanisms should bind, piping systems may fail releasing their contents, and there might be physical danger for operators from the release of these materials. In extreme instances, water for fire protection may be lost and reduced to injection from mobile sources that would have to be driven to the site.

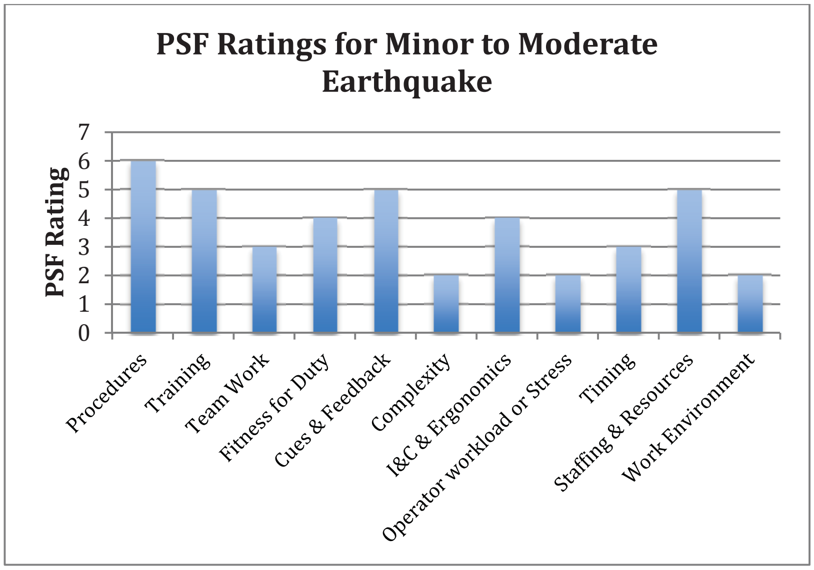

Figure 32: EBR-II PSF Analysis: Minor to Moderate Earthquake

Procedures were a positive influencing factor on performance and were rated "6". They were well written, they considered the roles of multiple personnel involved in the response, and were clear regarding the hand-offs to site-wide procedures for emergency action management. Training beyond classroom training was more difficult. For example, response for earthquake did not lend itself to simulator training and none was provided. For safety reasons, the crew did not participate in hands-on training where piping was sheared and debris was present. With the exception of the shift foreman who received site-wide emergency training as the on-scene commander and drills for emergency personnel, earthquake conditions were not trained for. Much of training that was provided relied upon memorization and exams. For these reasons training was rated "3", it was felt that this rating would be similar across many NPPs with little or no earthquake experience.

Teamwork was a positive influence and rated "5". Although response to the event required full crew involvement and coordination and communication between the MCR and field operators, EBR-II personnel demonstrated these characteristics to a high degree during normal operations. The SM would monitor plant response and initiated outside communications. The SF would monitor and direct communications among the MCR crew. There was no experience in dealing with the earthquake, however; there was an earthquake in Idaho during the 1980's where the Advanced Test Reactor (ATR) reactor crew at INL exhibited good teamwork and this was a consideration in assigning this value. The fitness for duty PSF was rated "4", although personnel were not expected to suffer routinely from general fatigue, winter weather can contribute to extended shift durations that could negatively influence crew performance. If outside conditions associated with the earthquake were such that field operators could be injured leading to inability to perform certain roles and responsibilities this PSF could easily be rated a "3". It was beyond the scope of this PSF study to postulate different classes of injury and its affect upon performance.

Operator cues for this event were kinesthetic (vibratory), auditory, and visual including earthquake alarms, physical effects, scram, secondary pump trip, negative reactivity and rod bottom lights. There were multiple sensors for acceleration and information was present in multiple MCR locations. Since these cues were both obvious and compelling, and no mental calculation on the part of operators was required, I&C and cues were rated "5". They were not rated higher because for follow-on actions it was not clear what indication and cues would be present for operators to rely on.

Complexity would be a negative influence upon operator and crew performance and was rated "2". The earthquake and subsequent automatic scram would happen sequentially and multiple simultaneous faults were expected during the earthquake. There was the potential for symptoms to be masked and multiple procedures would simultaneously be in effect including earthquake, scram, and site-wide emergency. Task execution was serial, parallel, and expected to be impeded by plant and environmental conditions. Additional workers were expected to be on-site, and ingress and egress to locations outside of the control room had the potential to be difficult.

Instrumentation, control and ergonomics was expected to be neutral and was rated "4". Controls and displays were located on various panels and otherwise, MCR ergonomics was acceptable. The ergonomics for field operations would depend on conditions and it was difficult to postulate with any degree of certainty what they might be like. Not surprisingly, operator workload and stress was expected to be a negative shaping factor. Workload PSF was rated "2", although there were three earthquake alarms available in the MCR, there was a moderate amount of visual search that was necessary to confirm conditions. The physical workload could be quite high, since immediate actions outside of the control room were required to assess the extent of physical damage to the plant. The shift foreman may end up outside assisting the field operators while performing the role of the on-scene commander and coordinating efforts with emergency response personnel. It was probable that field operators would be concerned for their physical safety and the extent to which additional data would occur associated with aftershock. The MCR crew would have a certain degree of mental stress associated with verifying reactor safety and assessing the impact of unavailable systems upon the ability of the plant to withstand any subsequent events such as fires and unsafe work conditions in general.

Time and timing would be somewhat negative and were rated "3". Although the reactor had a number of safeguards and protective features, the secondary and tertiary effects of the earthquake upon a large number of systems made the time available difficult for them to calculate. The time required for manual scram was not consequential unless the control rod drive mechanisms should bind. However the presence of a sodium leak and potential difficulty associated with verifying outside conditions and damage present was very open ended. Even steam leaks could have a pronounced effect upon return to stable conditions. Staffing was a positive influence on crew performance during earthquake and was rated "5". In terms of the MCR, all required personnel were expected to be present. An exception was when rounds were being conducted and for this analysis we assume that the operator was able to return to the MCR. Outside personnel were required as part of site-wide response, however, the emergency response team was located within 10 minutes of the reactor. The shift foreman would be in contact with personnel outside of the MCR and was trained as the on-scene commander. The present analysis did not consider activation of a Technical Support Center (TSC) or Emergency Operations Center (EOC) and the additional personnel outside of the emergency response team that could be brought in to help the MCR staff.

The work environment was expected to be negative and this PSF was rated “2”. Numerous and potentially difficult actions were required outside of the control room. The environmental conditions for performing these tasks could be difficult as well; hazardous such as fire, flood, sodium reactions or steam leaks as a result of the earthquake could be present.

9.4.7 Summary PSF Ratings and Conclusions

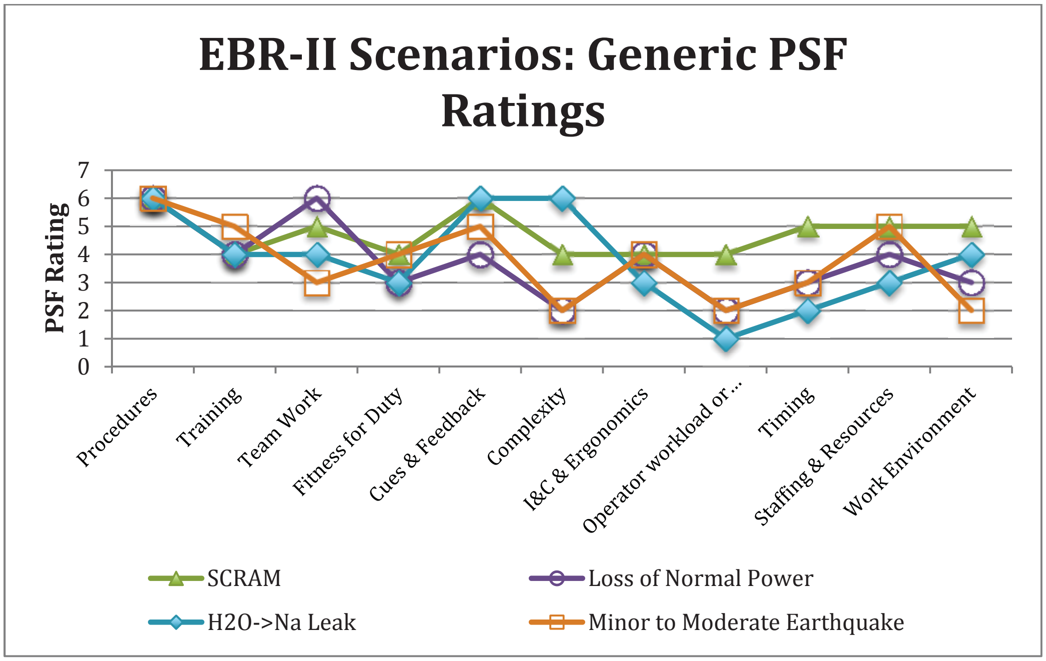

Figure 33: EBR-II Summary PSF Analysis for 4 Scenarios

Figure 33 above presents the PSF assignments for the 4 scenarios: general reactor SCRAM, Loss of Normal Power, water to sodium leak, and minor-to-moderate earthquake. The ratings assigned ranged from 1 to 6 indicating that a good portion of the scale was used. This plus the ease of application of the assessment method suggests that this PSF analysis method may have promise as both an evaluation and design tool. The method is sensitive to differences among the four scenarios as follows:

1. EBR-II had strong procedures and this was a positive influence for meeting all human performance requirements during these events. All scenarios were rated “4”.
2. For three of the scenarios the Training PSF is neutral to moderate due to the lack of an on-site training simulator and experience with the events. The exception is an earthquake where field training in emergency response is part of Shift Foreman training.
3. Teamwork ratings range from neutral to very positive. Water to sodium leak and earthquake are rated neutral. Although coordinated response from the field in support of the associated procedures is not practiced, the crew has a good history of coordination during normal operations. Additionally, troubleshooting and verification and validation of leak location could require additional coordination that is not practiced. Without a good

teamwork history, team work PSF would be rated "3". Teamwork for reactor SCRAM conditions is rated slightly helpful, the coordination and communication exhibited by personnel is believed to transfer to this relatively straightforward event. During Loss of Normal Power events teamwork is predicted to be moderately positive. This is based upon crew response at other INL facilities during the 1970's Western blackout where the teamwork nuclear operators exhibited was above average.

4. Fitness for duty ranged from slightly negative to neutral. Water to sodium events and Loss of Normal Power have the potential to have field operators take a number of actions outside where winter conditions could be present with the result that the operators could become tired and cold. Earthquake is also included in this general category. Reactor scram was rated neutral because most if not all actions are to be conducted from the MCR and operator cold and fatigue are not expected to be issues compromising crew performance. This is not absolute; events occurring at 3:00 AM may impose a negative influence on fitness for duty.
5. The influence of the cues, information, and feedback PSF for the 4 scenarios ranged from neutral to positive. The cues associated with Loss of Normal Power were neutral; the accelerometer-linked alarms and displays in the control and vibration cues provided a more positive influence on crew performance than they did for Loss of Normal Power. The best or most compelling cues and feedback were found in water to sodium leak because of the rapid response time and location for leak, and for reactor scram where multiple indications are present and there is no need for confirmation from the field.
6. The scenarios also varied in terms of the extent of the influence of complexity on operator performance. Not surprisingly, earthquake and loss of off-site power had a great deal of complexity; simultaneous events are expected, equipment may not be available or may be partially damaged, coordination outside of the control room is required, and the progression of these events is not easily determined beforehand. General reactor scram is not associated with a great deal of complexity.
7. The performance influence of I&C and ergonomics ranged from slightly negative to neutral in part, due to the age of the control room and tendency for controls and information to be in various control room locations. The ergonomics for water to sodium leak were slightly negative because of the conditions that could be facing field operators outside of the control room where access to verify and validate leak location could be difficult. The other three scenarios received a neutral ergonomics and I&C PSF ratings.
8. Operator workload and stress PSF ratings varied from strongly negative to neutral. Stress was highest for the water to sodium leak because of the financial consequences associated with draining the secondary sodium system and the potential for field operators to be injured by potential releases including steam release. Earthquake and Loss of Normal Power were stressful because of the gravity of the event and potential uncertainty regarding the extent of system damage outside of the control room that can interfere with plant recovery. General reactor scram was neutral, it was the least stressful of the three events, and actions were taken from the MCR by a well trained staff, with the crew complement needed to perform the actions associated with reactor scram was almost always present.

9. The influence of time and timing on crew performance is negative for three out of four events. In the case of water to sodium leak, reluctance to make decisions to drain the secondary sodium system may result in additional damage. This scenario received "2", the most negative PSF score assigned. Time and timing were negative but less so for Loss of Normal Power and earthquake scenarios. Negative influence comes from the length of time the event is in progress, time to verify conditions, and timing related to requirements for coordinated response among emergency response, the MCR, and field operators.
10. Staffing and resources were neutral or positive for three of the scenarios and negative for one. The operator response to the water to sodium scenario was negatively influenced by lack of personnel with experience in responding to water to sodium interaction and determining the source of the water leak may not be easy. Although a large number of personnel could be involved to response to Loss of Normal Power, all were believed to be present or within a radius of 10-15 minutes. From the main control room perspective, all necessary personnel for response to a general reactor scram were either in the MCR or are within 5 min of the control room. Staffing and resources were rated slightly positive as well, based on personnel and resources being available and their activities being well coordinated during the western power outage that affected INL site operations.
11. In terms of the work environment, each scenario received a different rating. The most negative was the work environment outside of the MCR during earthquake conditions. The potential exists for high heat, steam, extreme cold, toxic gases, flooding, and electrical conditions that could impede work outside of the control room.