Team Cognition and Outage Management:

Improving Nuclear Power Plant Resilience

by

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ABSTRACT

Nuclear Power Plants (NPP) have complex and dynamic work environments. Nuclear safety and organizational management rely largely on human performance and teamwork. Multi-disciplinary teams work interdependently to complete cognitively demanding tasks such as outage control. The outage control period has the highest risk of core damage and radiation exposure. Thus, team coordination and communication are critically important during this period. The purpose of this thesis is to review and synthesize teamwork studies in NPPs, outage management studies, official Licensee Event Reports (LER), and Inspection Reports (IRs) to characterize team brittleness in NPP systems. Focusing on team brittleness can provide critical insights about how to increase NPP robustness and to create a resilient NPP system. For this reason, more than 900 official LERs and IRs reports were analyzed to understand human and team errors in the United States (US) nuclear power plants. The findings were evaluated by subject matter experts to create a better understanding of team cognition in US nuclear power plants. The results of analysis indicated that human errors could be caused by individual human errors, team errors, procedural errors, design errors, or organizational errors. In addition to these, some of the findings showed that number of reactors, operation year and operation mode could affect the number of reported incidents.

Keywords: Teamwork, team cognition, nuclear power plant outage, outage control center, LER, resilient systems, resilience.

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CHAPTER 1: Introduction

The object of the study is to look critically at human and team errors reported in the official reports of the US nuclear power plants. In this way, it aims to determine important human, team and organizational factors that are correlated with team performance in a high-risk working environment. Identifying the problem areas of teamwork and the contributing factors will help to understand the weakness of the NPPs' outage system. Determining and improving these weaknesses and problem areas can boost the resilience of NPPs. Resilience is defined as a ability of a system to absorb disruption and still continue to maintain its normal operations. The research questions are:

- *What are the human, team, and organizational issues associated with events reported in the official reports such as the Licensee Event Reports and the Inspection Reports?*
- *What initiating events in the official reports are related to human, team and organizational errors during the outage control management in the NPPs?*

Previous NPP incidents

On March 28, 1989, a major Nuclear Power Plant (NPP) accident occurred at Three Mile Island (TMI) located in Londonderry Township, Pennsylvania. The reactor core experienced a partial meltdown (45% of the core) due to a malfunction of the cooling system. The chain of events revealed that lack of supervision and oversight by a maintenance employee resulted in the blockage of the emergency feed water system (Gandhi & Kumar, 2011). The investigation showed that inadequate emergency planning further exacerbated the problem. During the accident, hundreds of warning alarms were activated. However, the operation room team was unable to detect the cause of the problem. This event happened during the outage and seriously damaged the reactor core and an amount of nuclear radiation was released into the environment.

The TMI accident increased the attention of safety and human factors in US NPPs. However, there remain a significant number of incidents. For example, maintenance personnel at David-Besse Nuclear Reactor in Ohio discovered in 2002 that nuclear material ate through 150 mm of the carbon steel reactor head (Dien, et al., 2004). The reactor had just 9.5 mm of steel left, due to engineering design redundancy, which prevented a high-pressure explosion. This incident is referred to as a "near-nuclear accident" in the NPP industry (Dien, et al. 2004). Although this problem was discovered in the nick of time, it should have been identified in previous outage controls or maintenance periods.

A magnitude 9.0 earthquake and consequent tsunami initiated a catastrophic nuclear disaster in Northern Japan (Chino, et al. 2011). Immediately after the earthquake, three nuclear reactors were shutdown. However, due to infrastructure damage to the facility, the external power was lost and the other three reactor cores melted down in a short time (Hollnagel & Fujita, 2013). Fifty minutes after the earthquake, the tsunami demolished the usable part of the power plant facility; fuel tanks were washed away and emergency backup generators were wrecked. As a result, electricity in all reactors was lost. Within a couple days, three reactors of the power plants blew up (Hollnagel and Fujita, 2013) and a significant amount of nuclear material was released into the environment. Although, the disaster started by an exogenous uncontrollable factor, design problems of the NPP exacerbated the consequences. Additionally, the investigation showed that the emergency teams' response to the disaster was not efficient (Hollnagel and Fujita, 2013). All of these elements decreased the resilience of the NPP. Lack of team resilience was criticized.

NPP operations are very complex and complicated; multiple departments and expert teams have to work together to complete cognitively demanding tasks in high-risk environments in short periods of time. Any minor mistake could cause delays and may gradually increase the risks of disasters such as Three Mile Island and Fukushima.

Outage control is one of the hardest operations of a nuclear power plant. During this period, the NPP has to manage safety requirements, financial pressure, technical issues and coordination challenges (Schulman, 1993). This makes the plants more vulnerable than during normal operations (Kecklund & Svenson, 1997). There is a higher risk to cause significant core damage than during normal operation (Laakso, Pyy & Reiman 1998). According to a German Risk Study, human activities were reported to contribute 63% of core-melt events (Birkhofer, 1980). The Three Mile Island accident is the archetypal case, which happened during outage period.

Background

Previous incidents show that human activities are one of the significant contributing factors of nuclear power plant incidents. There has been much interest in the concept of human factors errors and their relevance to high-risk working environment incidents. In the following literature review section, the findings of relevant studies will be summarized. The literature review is covered in four sections: (1) Team Cognition, (2) Outage/Maintenance Control, (3) NRC Licensee Event Reports (LER) Analysis and (4) Resilient Systems.

In the literature review, firstly previous team cognition studies and theories are discussed. Eleven team cognition studies in NPPs are summarized and assessed based on four dimensions: (1) Situation Awareness (SA), (2) Coordination and Communication, (3) Leadership, and (4) Non-technical skills. All of the research appears to be centered on regular operation teamwork. Secondly the details of outage control and maintenance management of NPPs are reviewed and related research is examined under three categories: (1) Organizational Structure, (2) Technological Improvements, (3) Comparison of Regular Operation vs. Outage Control. Although, the outage control literature provides some important information about outage operations and human factors, it is limited in its investigation of team cognition. For a better understanding, the official reports are reviewed. Thirdly the official reports were explained. Because there is inadequate knowledge about teamwork in the outage literature, the official reports are analyzed to gather information to understand human errors and teamwork errors during outage control management. Lastly, after the Fukushima Disaster, improving NPP resilience has been discussed. In this section, the resilience studies from different fields are evaluated and principles of resilience are specified. In the following sections, the details will be discussed separately.

CHAPTER 2: Literature Review

Team Cognition

The most comprehensive definition of a team is given by Salas et al. (1992): a "distinguishable set of two or more people who interact dynamically, interdependently, and adaptively toward a common and valued goal/objective/mission, who have each been assigned specific roles or functions to perform, and who have a limited life-span membership" (p. 4). Team cognition is similar to individual cognition; just like individuals, teams can plan, learn, solve a problem, make a decision, design or assess a situation (Cooke et al, 2012). All of these different cognitive processes can be observed and analyzed at a team level. Cooke, et al. (2007) define team cognition as a "cognitive activity that occurs at a team level" such as planning, solving problems, assessing situations, and decision-making (Cooke, Gorman, Myers, & Duran, 2012). Individual cognition is embedded within team cognition (Cooke, Gorman, & Rowe, 2004). Therefore, team cognition studies are strongly influenced by individual cognition and primarily based on two perspectives: Shared Mental Models (SMMs) and Interactive Team Cognition (ITC).

Shared Mental Model (SMM)

The concept of a mental model has been used to refer to a mental representation of a physical system or events (Jonker, Van Riemsdijk, & Vermeulen, B, 2011). Mental models initially were used to understand human interaction with physical systems (Jonker et al., 2011). In the early 1990's, team cognition researchers adapted a mental model theory to explain team functions. Team performance is closely related to having a shared understanding of team and task among team members (Salas et al., 1992). SMM is defined as “knowledge structures held by individual members of a team that enable them to form accurate explanations and expectations for the task and coordinate their actions and adapt their behavior to demands of the task and other team members” (Converse, S., Cannon-Bowers, J. A., & Salas, 1993, p. 228).

The SMM theory is a traditional and well-recognized team cognition theory that uses the information processing approach to investigate how teams operate in cognitive settings. According to this theory, team activities are analyzed by “Input (I)- Process (P)- Output (O)” information flow diagrams, which are very similar to computer information processing (Lachman, Lachman & Butterfield, 1979 & Mathieu et al., 2000). The I-P-O framework suggests that team performance and outcome are influenced directly or indirectly by team processes and interaction (Mathieu et al., 2000).

According to the theory, SMMs indicate shared knowledge about the team, team objectives, team members’ roles, and interaction patterns (Kraiger, & Wenzel, 1997). Kraiger and Wenzel (1997) suggest that SMMs improve the team performance because team members are aware of task expectations and objectives. SMM theory emphasizes the importance of information flow with the Input (I) component. Input (I) is the starting point of team cognition measurement. There is valuable SMM research related to improved team performance and effectively shared information among team members. The research shows that communication frequency (Roberts & O’Reilly, 1979 & Foushee & Manos, 1981), field experiences (Dyer, 1984), and harmonized use of resources (Klimoski & Mohammed, 1994) are positive outcomes of teams with similar mental models. All of this research provides critical information to understand team cognition when shared information and team process are measurable. Usually in these studies, small teams are used to understand the effect of SMM on team cognition. The literature review results showed that more research should be completed to understand how SMM could be applied to larger complex teams in high-risk environments.

Interactive Team Cognition (ITC)

The Interactive Team Cognition (ITC) theory is inspired by ecological psychology (Cooke, Gorman, & Rowe, 2004). According to ITC, perception and thought at the team level are innately dynamic (Cooke et al., 2004). The center of cognitive processing in a team is *between* the individual and their environment, which includes other team members (Cooke et al., 2004; Gorman, 2014). Three main assumptions are used to form ITC theory: "(1) Team cognition is dynamic, (2) Team cognition is tied to context, and (3) The team is the preferred unit of analysis for studying team cognition" (Cooke et al., 2012, p. 256-257).

The shared knowledge of the team is built through team interaction (Gorman, 2014). Interaction is one of the key concepts of the theory and it is accepted as a critical element of team cognition. Therefore, it is claimed that team cognition is in the form of team members' interactions in the context of the task environment (Cooke et al., 2012). According to ITC, a team is the unit of analysis for team cognition measurement. Measuring individual data to analyze team cognition or using static team measurements are less desirable methods. Team cognition is not collective data from team members. It is a dynamic product of team interactions and activities. According to the theory, team performance should not analyzed over time as is dynamic. Team training should not emphasize transferring knowledge; it should emphasize the methods that increase the effective and efficient team interactions and activities (Cooke et al., 2012).

In ITC, Coordinated Awareness of Situation by Teams (CAST) is proposed to measure Team Situation Awareness (TSA; Gorman, Cooke, & Winner, 2006). CAST is a measurement method, which is designed to measure coordination-based situation awareness (Gorman, Cooke, Pederson, & DeJoode 2005). It is stated that measuring TSA cannot be a collection of team members' SA. According to Coordinated Awareness of Situation by Teams (CAST), coordinated perception and coordinated actions are the main focus of TSA. Good TSA is not dependent on all team members being individually aware of the situation. Communicating factual information to the right team member at the right time and coordinating the team action is the critical component of the TSA.

Why Interactive Team Cognition in NPPs?

There are three reasons that ITC perspective is more suitable than SMM in NPPs.

The first reason is the heterogeneous team approach of ITC; in NPPs, expert heterogeneous teams complete complex tasks. According to SMM, effective teams should have a shared mental representation and knowledge-set among team members. Although this principle can be applied to homogenous teams (or groups), it is not possible to create a shared knowledge system in highly skilled heterogeneous teams. According to ITC, in complex environments team members cannot have overlapping shared mental models, instead they have distributed situation awareness.

The second reason is that NPPs require large teams in a complex environment; the SMM mostly focuses on small team structure that can create a common team mental model. It is easy for small teams to create a shared knowledge system. However, in a complex working environment with many of team members, it is not realistic to expect that. According to the ITC, for large teams, team interactions are more important than creating shared knowledge.

The third reason is related to Coordinated Awareness of Situation by Teams (CAST); the ITC brings a different approach to situation awareness at the team level. According to ITC, sharing the right information to the right team member at the right time is more critical than creating a static knowledge system or individual awareness. This approach emphasizes the importance of timely information sharing in team level and encourages effective communication.

The fourth reason is the unique approach to team training that is based on ITC. Nuclear power plants operations heavily rely on procedures, so training is largely procedural. According to this theory, team training should challenge teams to create novel and innovative solutions when communication and coordination disrupted. It is designed to disrupt the habits of standard procedures and allow discovery and practice of new procedures (Gorman, Cooke, and Amazeen, 2010). Perturbation Training is designed to improve the adaptive skills of teams. It can be used to adjust inflexibility and rigidity of nuclear power plant teams who are procedurally trained.

Team Cognition Studies in Nuclear Power Plants

Team interaction and team cognition studies in NPPs gained popularity in the early 1990s. These NPP studies were completed by a diverse set of countries such as the United Kingdom, Finland, Hungary, Sweden, the United States, and South Korea. All participants had experience in control room operations. The participants' activities and interactions were observed during normal operations in the main control room or during simulated scenarios, which were emergency situations such as a bomb explosion in a reactor.

Eleven NPP-teamwork articles were selected for this review section. The selection of the articles was determined by three primary criteria: (1) content (teamwork, team cognition, crew performance in NPPs, control room training centers and research labs), (2) experimental setting (in NPPs or NPP simulation centers) and (3) participants (current or former NPP employees). In Table 1 a short summary of eleven articles is given based on topic, participants, training, team task, data source and findings.

Table 1

Nuclear Power Plant Team studies

Author (Year)	**Topic**	**Participants**	**Training**	**Team Task**	**Data Type**					**Findings**
					Interview	**Observations**	**Surveys**	**Scenarios**	**Team Performance**	
Montgomery, Toquam, and Gaddy (1991)	Team Interaction Skills Evaluation	13 instructor/senior instructors at Diablo Canyon NPP	Day-long training session team interaction skills	Responding 3 plant emergency simulation scenarios	X	X			X	-Training decreased stereotype accuracy error. -Complex judgment skill helps to lower stereotype accuracy error.
Roth, Mumaw, and Levism 1994	An empirical investigation of operator performance in cognitively demanding simulated	Up to 11 crews from 2 NPP, two simulated emergencies total of 38 cases	No special training.	Responding to two operational emergencies as a team.				X	X	-Situation awareness and response planning are key cognitive skills for effective teams. -Some NPP procedures aren’t fully addressed some of the high-cognitive tasks; diagnosis and situation awareness.

	emergencies									
Sebok (2000)	Influences of interface design and staffing levels	26 operators from Loviisa NPP, Finland. 8 different teams with two or four team members	Simulator training is given. Data collection training is provided.	Responding five different emergency scenarios			X		X	In a conventional plant, larger groups performed better. In an advanced plant, both groups performed well, however, in small teams situation awareness was higher.
Crichton and Flin (2004)	Identifying and training non-technical skills of nuclear response teams	18 emergency response team members from two UK nuclear sites	No training	The participants completed the critical decision-making (a cognitive task analysis) interview based on real or simulated emergency trainings.	X			X		Critical skills are identified as decision-making, situation awareness, communication, teamwork and stress management.

Vincente, Mumaw an Roth (2004)	Operator monitoring in a complex dynamic work environment: A qualitative cognitive model based on field observations	38 operators form three NPPs, approximately 288 hours of observation during operation	No training is given.	Team members were observed during normal operation period. All activities were video recorded.	X	X				-Human information processing relies on active knowledge-driven monitoring. -Between operators, there is a strong distributed cognition component in that operators rely on external representations to offload cognitive demands.
Carroll, Hatakenaka, and Rudolph (2006)	Naturalistic decision making and organizational learning in NPP: Negotiating meaning between managers and problem investigation teams	27 problem investigation teams from three different NPPs.	No training is given.	Team members completed a questionnaire about team management, team change, team learning, out-of-the-box thinking and deep causal analysis			X			-Teams with managers or supervisors as team member have better shared understanding. -Teams with more diverse departments produce deeper and more creative analysis. -Teams need more information access for effective learning.

Patrick, James, Ahmed, and Halliday (2006)	Observational assessment of situation awareness (SA), team differences and training	20 participants from the same NPP (5 shift teams, each team has 4 members)	No special training. All data are collected during bi-annual simulator training in NPP	3 different simulation scenarios are used. 1- Normal operation, handover with a leak. 2- Start-up (Outage control) with partially closed valve. 3- Bomb explosion-emergency operation		X		X	X	-There was no consistent pattern of SA between different teams. SA was situation-specific. -Team performance and team coordination were closely related with the shift leader. -Control room training should cover planning, problem solving, team coordination attention, communication and knowledge
Juhasz, and Soos (2007)	Impact of non-technical skills on NPP team performance: task load effects on communication	24 operators from The Paks NPP, Hungary. (4 teams and each team has 6 member	No training	The team members have to solve a simulation scenario as a team. This scenario has different level of task load (low,				X	X	-Higher task load decreases the frequency of communication -Shared mental model plays important role in managing and coping. -During high task load, operator's attention is more focused and narrowed, which leads loss of some relevant information -Poor teams' conversations are

		)		moderate, and high) And all team members have to involve to solving control task.						indicating incomplete information flow.
O'Connor, O'Dea, Flin, and Belton (2008)	Identifying team skills required by NPP operations personnel	38 operations team personnel are interviewed from 3 different NPP, UK	No training	The participants completed the Critical Incident Technique (CIT) interview about challenging situations.	X					According to interview results, nuclear team skills taxonomy is created: shared situation awareness, team focused decision-making, communication, co-ordination and influence with 16 component elements.
Stachowski, Kaplan and Waller (2009)	The benefits of flexible team interaction during crises	14 NPP control room crews from a northeastern US NPP (61 operator)	No training is given, the participants are observed during their regular	14 different teams faced with almost the same emergency scenario		X		X	X	-Higher performance teams exhibited fewer, shorter and less complex interaction pattern. -Flexible teams who have higher adaptability have higher performance.

			emergency training. (Behavioral observation)							
Kim, Park, and Kim (2011)	Characteristics of communications observed from off-normal conditions of NPP	8 teams from the same NPP, Korea. Each team has 4 members	No training, NPP simulation facility is used.	Recognizing off-normal activities and communicate during emergency situation.		X		X	X	-Teams who used standard communication protocol have better view of situation awareness, efficiency and clarity. -Teams who used the Standard communication protocol but the team members who are located in different placed (no face-to-face communication) more likely to miscomprehension or a wrong action.

The details of the Table 1 are discussed in the following sections.

Situation Awareness

Situation Awareness (SA) originated from aviation studies (Salas et al., 1995; Patrick et al. 2008) and has been widely accepted by cognitive scientists. Endsley (1988) defined SA as “the perception of the elements in the environment, the comprehension of their meanings, and the projection of their status in near future” (p. 97). Cooke et al. (2012) described it as “unity of responsibility for maintaining awareness of a dynamic environment” (p. 259). In other words, it is a shared and accepted responsibility of team members to preserve mindfulness of system status. According to these studies, SA is closely related to shared understanding of team members in the same operational process. Team Situation Awareness is closely related to creating shared mental models in team cognition research (Salas, et al 1995). The researchers conclude that there is a positive correlation between team situation awareness and team performance (Bolstad, & Endsley, 2003; Cooke, Gorman and Rowe, 2004).

Similarly, team cognition studies of NPPs support that there is a positive correlation between situation awareness and team performance (Roth, Mumaw, & Levism 1994; Sebok, 2000; Patrick et all 2006; O’Connor et al 2008; Kim, Park & Kim 2011). SA is one of the key aspects of effective teamwork, which is important to developing team understanding, anticipating possible issues, and maintaining an overview of tasks (Roth, Mumaw, & Levism (1994), Kim, Park, & Kim (1994) and O’Connor, O’Dea, Flin & Belton (2008)). Additionally, Roth, Mumaw, and Levism (1994) argued that existing NPP protocols are not fully addressing high-level cognitive tasks (i.e. diagnosis and situation awareness). The protocols usually offer step by step actions without providing rationale. In contrast, Kim, Park, and Kim (2011) claimed that current protocols are effective to create situation awareness clarity and efficiency. They suggest that in order to save time and decrease misinterpretations, the standard protocols should be used. Sebok (2000) investigated the size of teams in control rooms and concludes that the situation awareness of teams is closely related to communication and that small teams have better situation awareness.

Although, Patrick, James, Ahmed, and Halliday (2006) accepted the importance of situation awareness for a team, they took a different perspective of SA and claimed that SA is situation specific; there is "no general SA ability or attribute" (Patrick et al 2006). Similarly, Vincente, Mumaw and Roth (2004) emphasized that NPP operators are selected from different departments, which creates distributed cognition and situation awareness among NPP operators.

This approach is very similar to the heterogeneous team processing perspective of the ITC theory. According to the ITC, homogenous teams typically have more mental model similarities than heterogeneous teams (complex, specialized teams). However, complex tasks are usually completed by highly specialized heterogeneous teams, which are formed by different experts with unique skills. For example, NPP teams are formed from engineers, physicists, chemists, maintenance workers, supervisors, etc. For such a team, each member has a special function and skill set. Therefore, it is difficult to claim that each member will have the same common knowledge because each team member interprets the task according to his or her specialized perspective. For highly heterogeneous teams, situation awareness is the process of getting the right information to the right person at the right time in response to a change in the environment (Cooke, Gorman, & Winner, 2009). For a better evaluation of situation awareness of heterogeneous teams, more research should be completed.

Coordination and communication

Communication is defined as active information flow between interdependent team members (Cooke & Gorman, 2006), whereas coordination is described as "effectiveness of a team organized as a unit to perform a task both in time and space dimensions as well as in terms of division of responsibilities and command and control" (Chang & Mosley, 2006; p. 1020). A team's cognitive products depend on the level of dynamic interaction between team members (Cooke & Gorman, 2006).

Coordination and communication problems are investigated together and the research results illustrate that these team processes have a significant effect on team accidents. For example, the National Aeronautics and Space Administration (NASA) research revealed that coordination, communication and leadership errors are the root cause of 70% of aviation accidents (Cooper et al., 1980). Moreover, it has been found that 43% of surgical errors are caused by communication problems in the health care industry (Gawande et al, 2003).

In NPPs, human information processing relies on active knowledge-driven monitoring (Vincente, Mumaw & Roth, 2004). In order to complete a cognitively complex task in a high-risk environment, dynamic coordination and communication should be prevalent (Roth, & O'Hara, 2002; Zaccaro, Rittman, & Marks, 2002).

The distributed cognition of operators strongly depends on effective information flow between team members (Vincente, Mumaw & Roth, 2004). Thus, they can synchronize team actions without sacrificing safety requirements.

Communication has been described as a critical non-technical skill for a NPP operator (Crichton & Flin 2004; O'Connor, O'Dea, Flin, & Belton 2008). Stachowski, Kaplan and Waller (2009) noticed that the teams with higher performance exhibited a pattern of fewer, shorter and simpler interactions. Juhasz and Soos (2007) discovered that a higher workload is associated with decrease in the frequency of communication between team members. Additionally, they claim that incomplete information flow is one of the reasons for poor team performance in NPP control rooms.

Rognin and Blanquart (2001) conclude that effective communication between operators positively influences team SA, unity, and alertness. These results were consistent with Patrick, James and Ahmed (2006). They called to attention the importance of SA and conclude that: (a) inadequate communication, (b) lack of attention, (c) misperception, and (d) inadequate knowledge can decrease the SA of teams in a control room. However, existing literature is limited to create a conclusion about communication and coordination in the NPPs.

Leadership

Complex dynamic systems such as NPPs are managed by organizational teams, which have to deal with high levels of information flow, dynamic situations, and time constraints. Team leaders play a key role in coordinating team actions and decision-making (Zaccaro, Rittman, & Marks, 2002). Leadership is considered one of the characteristics of effective teams (Zaccaro et al., 2002; Carte, Chidambaram, & Becker, 2006; Yule et al., 2006; Mitropoulos, & Memarian, 2012). Team leadership has a noteworthy role in team processing and outcome (Kozlowski, & Ilgen, 2006). It is an important element to create safety in a working environment (Carroll, Hatakenaka, and Rudolph, 2006), and improve safe participation, as well as collaborative team learning (Martínez-Córcoles et al., 2012; Mitropoulos & Memarian, 2012). In high-risk industries, leadership can change the cognitive and behavioral processes of the team (Mitropoulos, & Memarian, 2012).

Research about the impacts of leadership on NPP teams is sparse. Patrick, James, Ahmed, and Halliday (2006) observed five different NPP teams' interactions and activities during three different emergency scenarios. The results showed that the team performance and team coordination were closely related to the shift leader, who is the most senior team member and has the highest responsibility during the operation (O'Connor et al., 2008). The authors conclude that the control room supervisor has a critical role to coordinate and oversee team actions. Similarly, Carroll, Hatakenaka, and Rudolph (2006) investigated twenty-seven cases from three different NPPs. They conclude that when managers or supervisors work as active team members, the team performance and shared knowledge improves.

Non-technical skills

Various teamwork studies have stressed that team performance is closely related to team members' non-technical skill set. According to the aviation research, non-technical skills are crucial to create effective communication and coordination between team members without risking safety (Flin, & Maran, 2004). Surgical team research shows that some of the surgical team failures are directly related to non-technical aspects of teamwork such as communication, leadership, and decision-making. (Yule, Flin, Paterson-Brown, & Maran, 2006)

Teams should be proactive in solving problems in high-risk environment. Thus, besides having technical skills, team members should bring a diverse skill set to the team. Decision-making is accepted as one of the main non-technical skills in medicine (Reader et al, 2006; Yule et al, 2006). Similarly, critical thinking is highlighted in aviation (Flin et al, 2003). Teams should be able to search for different information resources to manage challenging situations (Roth, & O'Hara 2002).

The workload in NPPs requires high levels of cognitive skills. Previous studies show that challenging tasks can be completed by flexible (Stachowski, Kaplan & Waller, 2009), adaptive (Montgomery, Toquam, & Gaddy, 1991) and diverse teams (Carroll, Hatakenaka, & Rudolph, 2006). The dynamic work environment of nuclear plants requires unique cognitive skills to cope with the demands (Vincente, Mumaw, and Roth, 2004).

Montgomery, Toquam, and Gaddy (1991) conducted the first teamwork investigation in NPPs. They revealed that the control room team members should have complex judgment skills to minimize stereotype accuracy error. This finding is consistent with Chang and Mosley (2006) and Patrick, James, Ahmed, and Halliday (2006). Chang and Mosley (2006) claim that problem solving is an individual's inherent cognitive skill, which can significantly affect the team members' attitude to solving problems and team performance in NPP.

Other researchers conclude that for effective teamwork, team members should have unique cognitive skills. For example, Roth, Mumaw, and Levism (1994) noted that response planning is one of non-technical skills of NPP operators. Furthermore, Vincente, Mumaw, and Roth (2004) claim that response planning is closely related to identifying goals, creating various solutions, analyzing the possible response plans and deciding on the best course of action. Moreover, the authors claim that time management is just as critical as response planning (Vincente et al., 2004).

Crichton and Flin (2004) found that stress management should be one of the important abilities due to the demanding nature of the working environment. Being stressful significantly affects the operator’s information processing and comprehension abilities (Chang & Mosley, 2006). Likewise, Vincente et al. (2004) observed that the successful operators frequently try to decrease their cognitive demands during non-work hours. Worledge (1992) summarizes all these skills as “diagnostic and troubleshooting skills” and defines them as key elements for an operator’s performance. Different empirical research has emphasized that non-technical and cognitive skills are crucial to create effective teamwork. As a result, it is suggested that NPP training should be improved accordingly to support team members (Worldege, 1992; Patrick, James, Ahmed, and Halliday, 2006).

All of these studies have provided important information about the operation room team cognition dynamics. However, they provide very limited insight about outage control teams. Unfortunately, there appears to be no empirical study that has investigated the team dynamics of outage control or management. Thus, there is a gap in the NPP literature. In order to fill this gap, a review of official Licensee Event Reports (LER) was conducted and the results are shared in Section 3. But before the LER analysis, previous outage control studies are discussed. Although, the focus of these studies is not on teamwork, they provide insightful information about the outage control context.

Outage (Maintenance) Control

Outage Control Centers (OCC) are usually referred to as "temporary command centers" (St. Germain, Thomas, and Farris, 2014) or "tactical command centers" (Hurlen, Petkov, Veland & Andresen, 2012) of a nuclear plant during the refueling process. Multidisciplinary teams of experts work together to keep the outage on track by coordinating tasks across the plant and to resolve unexpected issues (Hurlen et al., 2012; St. Germain et al., 2014).

The OCC team has a multidisciplinary structure that consists of outage managers, planners, engineers, experts, and health and safety staff (Hurlen et al., 2012). Figure 1 shows a simple diagram of the multidisciplinary structure of outage/maintenance management. During outage/maintenance, the total number of workers in the nuclear plant can be 2000 or higher. Some of these workers may be temporary contractors who are not very familiar with the reactor. In the field, multiple interdependent teams from different departments work together to complete the complex tasks. The department supervisors work with field teams and OCC to create synchronized coordination and communication.

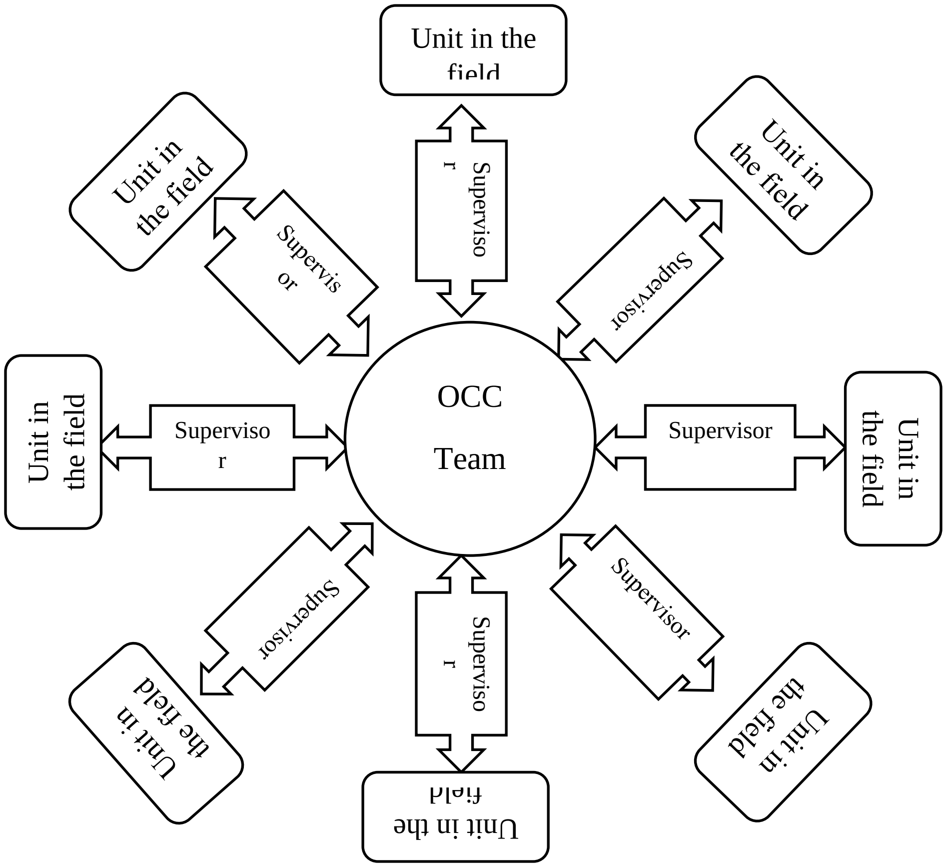

Figure 1. Outage Control Management diagram. In outage/maintenance period, the total number workers in the nuclear plant can be around 2000. All information will be gathered in the OCC.

During the outage period, usually 20-35 days, the OCC teams attempt to complete a variety of tasks including: nuclear waste cleaning of the reactor, refueling the nuclear rods, necessary nuclear safety controls and regular maintenance activities (St. Germain et al., 2014). They are usually referred to as cognitively demanding complex activities (Ghandi & Kumar 2011) and knowledge- intensive work (Reiman & Oedewald, 2006; Barley, 1996). Completing these complex tasks in a short time frame is challenging due to the depletion of intense labor and financial resources. Generally, a nuclear reactor generates around $1 million to $1.5 million per day in revenue (St. Germain, Thomas, and Farris, 2014) when operating at full capacity. During outage control, the nuclear power plant will be on non-productive mode (Hurlen et al., 2012), which creates financial pressure on the organization and OCC teams (Reiman & Oedewald, 2006).

Due to the time restriction, the outage schedules are determined in advance with department experts. Usually, the maintenance activities are interdependent. In order to complete the outage as planned, the maintenance crews and operation room have to employ effective communication and coordination. Real-time information is crucially important to track progress on high-reliability tasks (Bourrier, 1996; Hurlen et al., 2012). Any misunderstanding can also risk lives of NPP employees. Misunderstandings can increase the possibility of a nuclear accident as well as core damage (Laakso et al., 1998).

Unfortunately, outage management has received little attention from researchers. Previous studies generally focus on organizational structure, ergonomic aspects of outage control room or technological improvement. The details of these studies are discussed in the following section.

Organizational Structure

Bourrier (1996) is one of few researchers who focused on US NPP outage management in the 1990s. She investigated methods of effective outage management by comparing two different organizational structures. A1 and A2 are US reactors, which have different outage control perspectives. The A2 NPP has a more flexible and adaptive organizational structure. The outage management team is proactive and responds to any problem or maintenance issue rapidly at all levels. She defines this system as a “self-correcting” system. In a self-correcting system, action is taken immediately or a correction plan is developed. In contrast, the A1 plant has a more “controlling, centralized management” structure and has more dedicated resources such as longer planning time and more employees. The results show that for the same task, A1 management has a longer response time than A2 management. This result is in agreement with Hollnagel and Fujita (2012) who claim that resilience in NPP can be improved by creating a flexible and proactive management system.

Additionally, Bourrier (1996) reported that four real-world problems influence the effectiveness of outage /maintenance: (1) lack of effective coordination between field maintenance workers and outage control room operators, (2) lack of a very detailed and specific procedure, (3) the need to cope with unanticipated situations, and (4) ensuring proper execution and work quality.

Technological Improvements

The University of Chicago built the first nuclear reactor in the world, Chicago Pile 1 (Generation I), in 1942. Since then, control room design has changed and reactor capacities have increased. Most US nuclear reactors were commissioned in the 1970s and 1980s (Thomas, Lawrie, & Niedermuller, 2016). Information technologies have significantly improved since the first reactor; however, the US outage control centers have not been able to cope with these changes. It is hard to organize and monitor complex activities by using obsolete methods. They require additional time and dedicated resources. For example, some NPPs in the US still use simple communication methods, such as runners that deliver papers, face-to-face communication, daily printouts of schedules, and static white boards (St. Germain, 2015). Sometimes when information is received, it is already outdated (Thomas, Lawrie, & Niedermuller, 2016).

St. Germain, Thomas, Farris, and Joe (2014) focused on the improvement of OCC information technologies. The Advanced Outage Control Center (AOCC) research is designed to effectively utilize communication and coordination technologies for outage management and problem solving (St. Germain et al., 2014). The use of collaborative software, mobile worker devices, remote cameras, computer-based work packages, and wireless networks are some of the additional developments for AOCCs.

Hurlen, et al. (2012) also emphasized the importance of integrating collaborative technologies in the OCC. They state that one of the main challenges of the OCC is acquiring reliable real-time information of work progress. They suggested using interactive tabletops: a “scenario composer” touch interface for scheduling and a real-time collaboration platform. They briefly discussed the design problems of existing control centers and highlighted the importance of user-centered designs for an effective work environment.

Thomas, Lawrie, and Niedermuller (2016) support the idea of improving communication technologies and emphasized the financial effect of obsolete control centers. They claim the business benefit of using advanced technology in OCCs would decrease labor and non-labor resource allocation (such as time, technology) and save approximately $7.79 million annually.

All of these studies reflect the importance of current control centers communication and coordination techniques. Team performance research shows that computerized procedures increase team performance in nuclear plant control rooms (Huang & Huang, 2009; Sebok, 2000). In order to create proactive teams in a high-reliability industry, it is crucial to have real-time communication and coordination.

Comparison of Regular NPP Operation vs Outage Control

For a better understanding of the significance of outage control, the following section compares normal NPP operation versus outage management in the literature. Table 2 shows a summary of the literature on operation modes, control centers of operations, and nuclear safety levels (St. Germain, et al., 2014; Hurlen, et al., 2012; Laakso et al.,1998), work related factors (Jacobsson, & Svensson, 1991; Kecklund & Svenson, 1997; Reiman & Oedewald, 2006; Ghandi& Kumar, 2011), team structures (Hurlen et al., 2012), time (Hurlen et al., 2012), task (Jacobsson & Svensson, 1991; Lee, T., & Harrison, K., 2000).

Table 2

Normal Operation vs Outage Control Operation Periods.

	Operation	**Outage Control**
Nuclear Reactor Status	**Mode 1**-Operation Reactor's pressure and temperature are adjusted. Power > 5% and gradually increase to 100%	Based on stage of outage, 5 possible operation modes: **Mode 2**-Startup: Power <5% **Mode 3**-Hot Standby **Mode 4**-Hot Shutdown **Mode 5**-Cold Shutdown **Mode 6**-Refuel: Temperature
Nuclear Safety	-No entering reactor chamber unless there is an emergency. -All monitoring will be done from the control center.	-Reactor is in non-productive mode. Based on work order, the field workers can be in the reactor chamber. -During refueling, workers will be replacing reactor nuclear rods (nuclear waste will be removed, new nuclear fuel will be placed. (St. Germain et al., 2014) -Higher risk of nuclear radiation. The workers have limited time due to nuclear radiation (Hurlen et al., 2012) -Higher risk of nuclear core damage (Laakso, Pyy & Reiman 1998)
Control Center	Main Control Room	Outage Control Center (St. Germain et al., 2014)

Work related factors	-Higher satisfaction with work performance (Kecklund & Svenson, 1997) -Minor mistakes are significantly related with (a) work demand and distribution, (b) sleepiness during afternoon and night shifts, (c) decreased aspiration level and knowledge and experience to manage to work tasks (Kecklund & Svenson, 1997)	-Increased coping strategies involving increased efforts, besides decreased job satisfaction and work performance and increased use of delegation of tasks to others (Kecklund & Svenson, 1997) -More alert however, still higher risk of minor mistakes (Jacobsson, & Svensson, 1991; Kecklund & Svenson, 1997) -Minor errors are significantly related with (a) Increased work demand, (b) work distribution, (c) planning problems and (d) coordination issues (Kecklund & Svenson, 1997) -Large variety of tasks and tasks with high safety significance (Reiman & Oedewald, 2006)
Team members	-Facility employees (usually employee number is stable. If there is an unexpected issue, other departments will support.)	-**Ad-hoc OCC team:** Facility Employees/experts from different departments -Field workers: Field supervisors (facility employees) and non-facility temporary contracted workers who may be unfamiliar with the plant (Hurlen et al., 2012) -Workers in the field/reactor can be around 2000 or more.
Training	Scheduled regular trainings for normal operation and emergency	Scheduled regular trainings for normal operation and emergency
Time	12-18 months unless there is an unusual event	Planning and scheduling: 6-12 months. Execution: 10-35 days. (Hurlen et al., 2012)
Task	-More routine task load (day-workers and shift workers) (Lee, T., & Harrison, K., 2000)	- Increased work strain, shiftwork including night-work and reduced social support (Jacobsson, & Svensson, 1991) -85-90% of the tasks are pre-scheduled. (day-workers and shift workers). However, still during maintenance, 10-15% unplanned tasks should be completed.

During OCC a large variety of tasks are completed in a very short time. Employees report that they feel more stressed than during regular operation.

Additionally, they are more likely to make mistakes (Jacobsson, & Svensson, 1991; Kecklund & Svenson, 1997). They are less satisfied with their jobs and feel higher anxiety and reduced social support (Jacobsson, & Svensson, 1991).

The literature, as shown in Table 2, shows that regular operation and outage management operations have very different dynamics of level of workload, team structure, stress, and working environment. However, previous studies mostly focused on normal operation time. Teamwork during an outage has also been neglected. In order to understand teamwork in NPP, the outage control period should be investigated separately from normal operations. In this way, the system weakness and strength can be revealed. The information extracted from outage control teamwork can be used to improve system resilience. In the following section, the importance of teamwork in a resilient system is discussed.

Resilience Engineering

In high-risk industries, two methods have been used to understand the requirements of safety management: (1) accident analysis, and (2) risk assessments. In accident analysis, accidents are usually defined in terms of cause-effect relationships. A linear propagation of effects is identified that explains the accidents (Hollnagel, Woods, & Leveson, 2006). In risk assessment, statistical predictions are used to estimate possibilities of system failure. Risk assessment uses calculated predictions based on risk indications or identified factors in a system. Although these two methods provide important information about the system, both methods ignore unknown factors and only rely on existing failure indications. In other words, both methods fail to provide information about a system weakness if there has been no previous related incident.

Resilience engineering brings a new approach to safety management in a complex system. It focuses on how to help people handle complex situations (expected or unexpected) under pressure to achieve success (Hollnagel, et al., 2006). The review of the literature shows that the concept "system" is used to discuss organization, regulators, operators and stakeholders. It offers a new safety assessment perspective in complex, interconnected, and ultra-high-reliability systems (Hollnagel & Fujita, 2012). A resilient system considers safety as a core value, not a commodity that can be counted (Hollnagel, et al., 2006). For a better understanding, different resilience definitions are presented in Table 3. It provides an overview of some of the definitions, taxonomies and contributing factors from different studies.

Table 3

Resilience definitions from different fields

Author (year)	**Resilience definition**	**Taxonomy/Principles**	**Contributing Factors to Resilience**
Walker, Holling, Carpenter & Kinzig (2004)	Resilience is the capacity of a system to absorb disturbance and recognize while undergoing change so as to still retain essentially the same function, structure, identity, and feedbacks.	1. System Vulnerability (Robustness) 2. System Wealth (Redundancy) 3. Flexibility (Responsiveness) (Holling, 2001)	Human actions are considered one of the main components of adaptability of a system.
Hollnagel,	Resilience is the ability to adjust its	1. Ability to learn (Learning)	In Hollnagel, and Fujita's (2011) article,

Woods, & Leveson (2006)	action prior to, during, or following changes and disturbance, so that it can continue to perform as required when a disruption or a major mishap occurs or in the presence of continuous stresses.	2. Ability to monitor (Monitoring) 3. Ability to anticipate (Anticipating) 4. Ability to respond (Responding)	it is stated that each principle requires Input (I), Output (O), Resource(R), Control (C), and Time (T) aspects. Each aspect can be closely related with human activities in the system.
Dinh, Pasman, Gao, & Mannan (2011)	Resilience is the ability to recover quickly after an upset, has been recognized as an important characteristics of a complex organization handling hazardous technical operations.	1. Minimization of failure 2. Early detection 3. Flexibility 4. Controllability 5. Limitation of effects 6. Administrative controls and procedures	• Design factor • Detection potential • Emergency response • Human Factor • Safety management system
Jackson, & Ferris (2012)	Resilience is the ability to adapt changing conditions and prepare for, withstand, and rapidly recover from disruptions. (White House, 2010)	(They used Wood's (2006) taxonomy, which was created based on ecosystem observation.) 1. Capacity 2. Flexibility 3. Tolerance 4. Coherence	14 principles are created. Principle no.5 is defined as "Human in the loop". "Human in control" and "Reduce human error" are defined as sub-principles.
Kamanja, and Jonghyun (2014)	Resilience is the ability of a system to recover from a disturbance, so that can sustain required operations under both expected and unexpected conditions.	1. Anticipation 2. Robustness 3. Adaptation 4. Collective Functioning 5. Learning Organization	Effects of human factors are explained under each principle. Crew performance/teamwork is specially emphasized in "Collective Functioning".

Based on these definitions, three main characteristics of a resilient system can be highlighted: (1) A resilient system can absorb disturbances by adapting to new situations and adjusting its operations, (2) While absorbing the disturbance, the system will maintain its operational structure and functions and (3) The system is fully equipped to adapt to anticipated and unanticipated conditions.

Holling (2001) created the first resilience system taxonomy inspired by ecosystem resilience; ecosystems can maintain structures and functions despite critical disturbances. According to him, (a) System Vulnerability (Robustness), (b) System Wealth (Redundancy) and (c) Flexibility (Responsiveness) are the key concepts of resilience engineering. His approach inspired many other scientists. Hollnagel (2009) created four resilience abilities of a resilient system: (1) learning, (2) monitoring, (3) anticipating and (4) responding.

Resilience in Nuclear Power Plants

The Fukushima NPP disaster was one of the most horrific NPP accidents in the nuclear industry. The consequences were global. From March 12 to April 5, 2011 a significant amount of nuclear materials (with a half-life of 30.1 years) were released into the atmosphere (Chino et al., 2011). A sizeable amount of radioactive materials was deposited on the topsoil (Yasunari et al., 2011), which is impossible to be removed. More than 80% of radionuclides were transported offshore and deposited into the Pacific Ocean (Steinhauser, Brandi, & Johnson, 2014). Between April and May, 2011, the water samples from Greece and Russia showed the maximum levels of radioactivity in water since Chernobyl (Bolsunovsky, & Dementyev, 2011). After the Fukushima disaster, the attention on nuclear power plants and safety assessments increased significantly. NPPs were subsequently criticized for being out- of-date and unable to handle dynamic complexities.

Fukushima NPP started operations in 1971, around the same time as US nuclear reactors. According to Probabilistic Risk Analysis (PRA) and Tsunami Assessment Method (prepared by Tsunami Evaluation Subcommittee, Japan), it passed all of the nuclear safety requirements for earthquake and tsunami damages at that time (Hollnagel & Fujita, 2012). In other words, based on existing assessment technics at that time, Fukushima NPP satisfied all the safety requirements based on expected risk conditions. Hollnagel and Fujita (2012) harshly criticized this situation. They claimed that these analysis methods gave overconfidence to the NPP operators until the plant was hit by a severe accident. They argued that the Fukushima nuclear disaster was exacerbated by a systematic failure of organization. Lack of resilience and underestimated past earthquake and tsunami information were the main reasons for the catastrophic disaster.

Hollnagel and Fujita (2012) summarized all failures into two main categories: (1) anticipation problems which has two subcategories: (a) lack of design related anticipation, and (b) lack of operation related anticipation, and (2) responding to the disaster problems. Lack of resilience in human factors is discussed under the title "responding to the disaster". It is stated that the disaster would be more manageable with a flexible organizational structure that can respond under exceptional circumstance (Hollnagel & Fujita, 2012).

Similarly, Kamanja, and Jonghyun (2014) claim that classical probabilistic risk assessment methods do not reflect the real brittleness of current old-fashioned nuclear plants. For safe operation in NPPs, organizations should be able to anticipate expected and unexpected risks caused by hardware, human or organizational failure

(Kamanja & Jonghyun, 2014).

Team Cognition and Resilience Engineering

Resilience engineering is a new concept that has been discussed over the last couple decades. Although there are many articles that discuss the theoretical framework of resilience, there appears to be no empirical quantitative research that describes how to achieve it. The lack of quantitative research may lead to some confusion (Park, Seager, Rao, Convertino, & Linkov, 2013).

Even though it is not labeled as “resilience”, team cognition researchers have been investigating some aspects of resilient systems. Results from quantitative team research shows that an effective team can handle challenging tasks under pressure and stress. In the team cognition literature, these kinds of teams are defined as “self-correcting” (Bourrier, 1996), “proactive” (Gavin & McPhail, 1978), “adaptive” (Montgomery et al., 1991; Gorman, Cooke, & Amazeen, 2010), “flexible” (Stachowski et al., 2009) or “diverse” (Carroll et al., 2006). All of these labels indicate different components of team cognition in a resilient system. Team cognition studies can be useful to cover the empirical research gap of resilience studies.

One of the best applications of resilience engineering and teamwork was demonstrated by NASA, which has a record of achieving some of the most challenging engineering tasks. Apollo 13 was the fifth mission to the moon. It is also known as “the most successful failure of NASA” (Rerup, 2001). During the third of day the mission, a cryogenic tank of the lunar module exploded in space. The explosion caused the loss of breathable oxygen, power in command module and loss of cabin heat. Thus, the crew had very limited time and resources to survive. Despite these insurmountable obstacles, NASA engineers were able to create an incredible team effort to help to Apollo 13 to reenter the earth's atmosphere safely and without injury (Lovell & Kluger, 1994). NASA was able to accomplish such a hard task due to two resilient characteristics of NASA teams: (1) anticipation, which is ability to predict possible failures, and (2) improvisation, which is the ability to adjust actions with a new way based on past experience (Rerup 2001).

Licensee Event Reports (LERs)

Despite the attention given to studying teamwork in main control rooms, there appears to be no empirical studies that analyze team interactions and team cognition during outage management/maintenance. Past studies have investigated the ergonomic aspects of outage control (Hurlen et al., 2012), the technological improvement of the control center (St. Germain et al., 2014), and the organizational structure of outage management (Bourrier 1996).

NPPs are strictly regulated, and their operation details are copiously documented, Licensee Event Reports (LERs) are only one of them. LERs are publicly available narrative reports filed by employees of NPPs that provide critical insight into plant operations and incidents. In some studies, LERs were used for mathematical risk estimations such as estimation of common- cause failure probability calculation (Evans, Parry, & Wreathall, 1983), reliability analysis (Salo, & Svenson, 2002), and human reliability research (Pyy, 2001).

Svenson and Salo (2001) published one of few articles that use Swedish NPP LERs to understand human errors in the nuclear industry. Ninety-three LERs of four different Swedish nuclear reactors in 1999 were investigated to understand detection time, mode of detection, and qualitative differences in the reports. According to the results, 10% of the errors remained undetected for 100 weeks or longer, 40% of the errors were detected through regular testing and 40% through alarms. The rest of the errors were discovered in other ways. The results showed that a greater number of LERs or error reports could be a sign of higher safety standards. In other words, employees with more reports were also more willing to report errors to be able to learn from them (Svenson, & Salo, 2001). Although the article does not cover US NPPs and LERs, still it helps the industry to understand the significance of LERs.

Previous research has indicated that human activities are closely correlated with operational incidents (Minarick & Kukeilka, 1982; Fleming, Silady & Hannaman, 1979; Svenson, & Salo, 2001), though it is likely that more complex system issues are also pertinent. Incidents related to teamwork or team cognition failures are of the utmost importance (Cooke, Gorman, Myers, & Duran, 2013). The initial LER analysis is more exploratory than confirmatory. This analysis can be used as a starting point for team error analysis in NPPs. Further research should examine LER data across different NPPs, especially focusing on errors related team cognition failures. According to the resilience studies, learning from previous mistakes is one of the critical features of a resilient system. The LER analysis can used to identify system weaknesses. By improving these areas, team cognition and system resilience can be strengthened.

A resilient system is able to handle all expected and unexpected challenges without sacrificing safety. In NPPs, transition from traditional safety approaches to resilient systems can be achieved by improving multiple factors such as design, technology, organizational management and proactive teamwork.

Overview

NPP operations are very complex. In particular, outage control is one of the most critical of operations with a high risk of radiation and core damage. In order to complete tasks, multiple-disciplinary teams have to work together interdependently. Time constraints, financial pressure and complexity of tasks significantly influence the team performance. Even though all of these factors make the outage control period more vulnerable than normal operation, most previous research has neglected the outage control period. There seems to be no empirical study on team cognition that has solely investigated team dynamics during the outage period. The need for future research on OCC is also indicated in official LERs. The LER database provides a rich data set for the analysis of unanticipated events in NPP operations, which includes regular operation period and outage period. Therefore, further research should be done to examine outage control teamwork with the goal to improve safety, productivity, and resilience through improvements in team cognition. For this purpose, a comprehensive official document analysis will be conducted and subject matter interviews will be completed.

CHAPTER 3: Method

In this study, an exploratory research method was used to identify potential causes of team errors and to detect associative components that can be influencing the individual human and team errors in NPPs during the outage control period. Two different methods will be used to collect the research data: 1) official document analysis, and 2) expert interviews.

Document Analysis

Document analysis can be a very complicated task because the documents are created without having an intention to be used in research (Bowen, 2009). However, transforming these textual data into critical and beneficial knowledge can provide essential information. In this study, document analysis was applied to some of the publicly available NPP documents to extract some information about teams and team cognition in the NPPs.

A summary of the document analysis is shown in Figure 6. All of the documents are obtained from the database of Nuclear Regulatory Commission. Mainly two publicly available official materials will be used: a) the Licensee Event Reports (LERs) from 2000-2016 and b) the Inspection reports (IRs) from 2000-2017. Each of these documents will be discussed separately in the following sections.

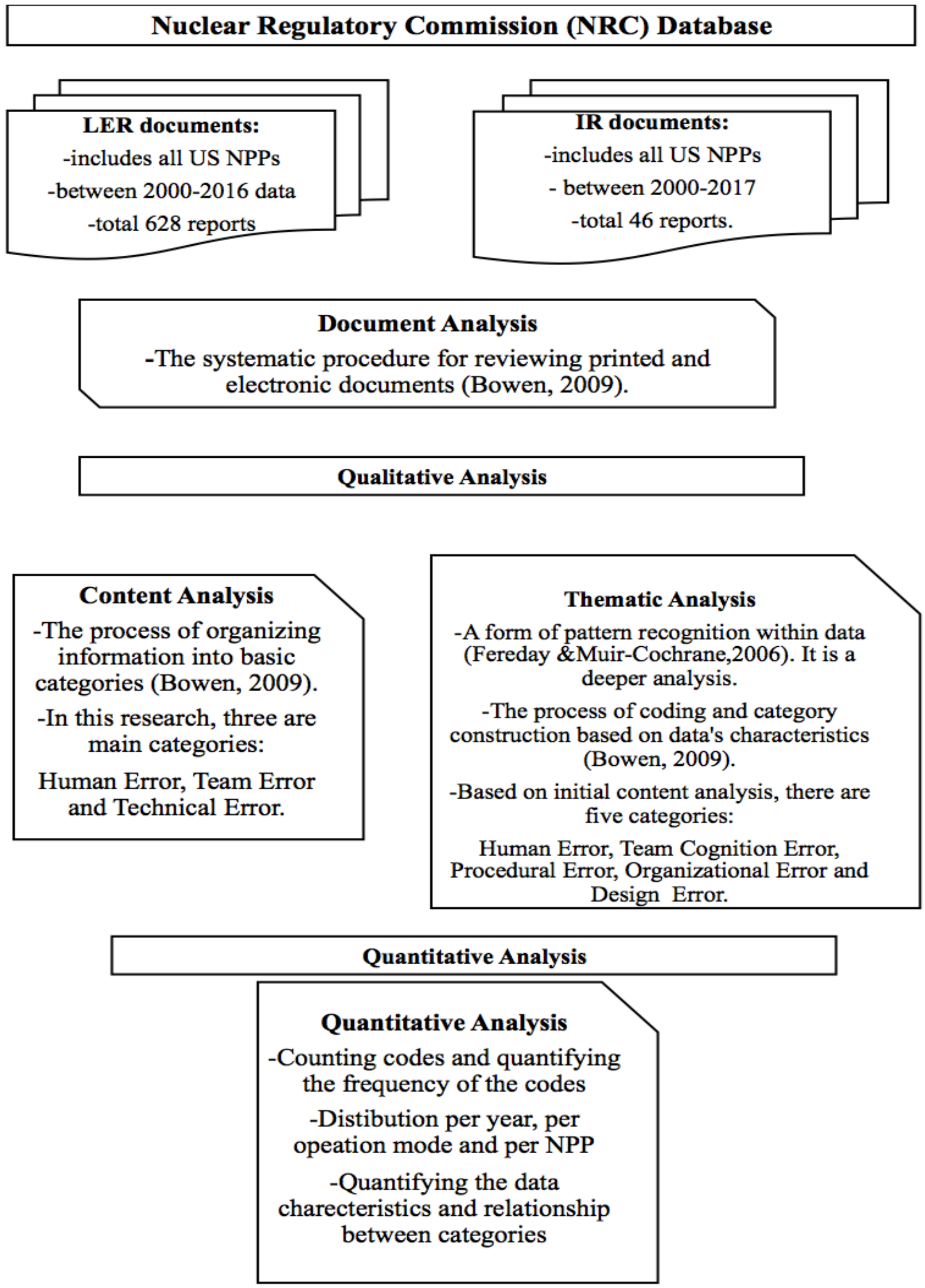

Figure 2. The Summary of data analysis

Data Sources

Three fundamental data sources will be used to gather data for this study: a) The Licensee Event Reports, b) The Inspection Reports, and c) The Subject Matter Expert Interviews.

Licensee Event Reports

A nuclear power plant (Licensee) shall submit a Licensee Event Report (LER) within 60 days after an unusual event or incident discovered in the nuclear power plant ("The Licensee Event Report System", 2017). The LER should summarize the detail of the event and describe essentials information about the reactor condition and the nuclear safety of the facility. In the LERs, it is also required to report any human performance related root cause ("The Licensee Event Report System", 2017). It is the responsibility of a Licensee to prepare the LERs to submit it to the NRC. Consequently, employees of a nuclear power plant create the LERs. The reports are mostly technical reports, each with 4-13 pages text document.

In general, the LERs are divided into two main parts: the abstract and the narrative sections. In the abstract, general information of the event is described with details. In the narrative part, the description of the activity, the analysis of the event, causes of the event, the corrective actions and previous occurrences subcategories are listed. As an example, one of the short LERs from the Oyster Creek Nuclear Reactor is attached at the Appendix A, page 108.

Palo Verde Nuclear Reactor LER Analysis from 1985-2015

Human contribution and group interdependencies that are associated with unsafe behavior are typically categorized as "human error" in the literature (Jacobs & Haber, 1994). Although human error is one of the main causes of incidents, this category is overused as a "catch-all" to cover all micro and macro organizational factors (Jacobs & Haber, 1994). This situation might affect the number of report human and errors in the official reports. Some of the design and organizational issues could be listed under the "human error" category. Initially, 228 licensee event reports between the years of 1985 to 2015 were analyzed; the summary is attached as an Appendix B on page 113-120. All of these reports are collected from the NRC search engine. The reports provide information about the three different reactors' outage management at Palo Verde Nuclear Power Plant.

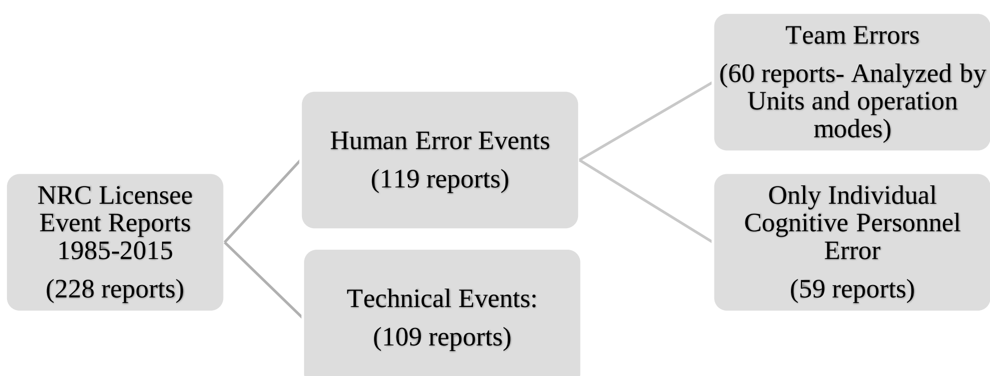

Figure 3. 1985-2015 Palo Verde Nuclear Power Plant LER Analysis.

After a deeper analysis of the LERs, seven keywords were found, which are commonly used for the incidents related with human errors. The keywords are: “Human error, personnel error, cognitive error, inadequate, deficiency, insufficient, lack of ”. In order to filter human error related incidents, these seven key phrases were used. The NRC search engine is designed with advanced searching features. In Table 6, the details of the NRC search engine are given.

Table 4

The NRC search engine setting for the LER analysis

	LER Search Engine Details
Report Years:	January 1, 2000- December 31, 2016
Plants:	All
Dockets:	All
NRC Regions:	I, II, III, IV
Operating Mode:	Startup, Hot Standby, Hot Shutdown, Cold Shutdown, Refuel
Reportability:	All
Power Level:	0%-100%
Keywords:	“ Human error, personnel error, cognitive error, inadequate, deficiency, insufficient, lack of ”.
Keyword Scope:	Full Document
Search Result:	617 LERs

Inspection Reports

The US nuclear power plants operate with strict nuclear regulations. The Nuclear Regulatory Commission (NRC) diligently oversees the operation of the US nuclear industry (“2017 Inspection Reports”, 2018). The NRC performs regular inspections to carefully examine whether the NRC’s requirements are followed or not (“2017 Inspection Reports”, 2018). The Inspection Reports (IRs) are created to report observations and findings of an NRC examination. Dissimilar to the LERs, the IRs are prepared by the NRC’s inspectors based on clearly defined guidelines. The IRs provide very detailed information about quality and suitability of products used in NPPs, human activities and services during regular operation period, outage control management and refueling period (“2017 Inspection Reports”, 2018).

In this research, the IRs are the second main data source of the official document analysis. Similar to the LERs, the IRs are also publically available documents at the NRC website. The IRs are usually around 24-88 pages. As an example, the shortest IR is attached as Appendix C, page 121-132. In the IRs, there are four main sections: a- Summary of Findings, b- Reactor Safety, 3- Radiation Safety, and 4- Other Activities. In this research, the IRs analysis was limited to 2000-2017 operation years in order to examine the most recent data. Table 5 shows the details of the NRC search engine setting.

Table 5

The NRC search engine setting for the IR analysis

	IR Search Engine Details
Report Sent Date:	2000-2017
Site/Plant Selection:	All
NRC Regions:	I, II, III, IV, V
Procedures:	7111120- Refueling and other Outage Activities
Operating Mode:	Startup, Hot Standby, Hot Shutdown, Cold Shutdown, Refuel
Cornerstone:	All
Significance:	All
Item Types:	Finding
Status:	Open and Closed
ROP	PIM
Draft:	No
Keywords:	None
Search In:	Full Inspection Report
Search Result:	46 IRs

The Subject Matter Expert Interviews

Qualitative researchers investigate phenomena with multiple data sources with distinctive research methods to design an objective analysis (Stemler, 2001). By using different data sources, the aim is to validate the data and increase the credibility of research findings (Bowen, 2009 and Stemler, 2001). Examining collected data by using multiple resources and methods is defined as a data triangulation (Bowen, 2009). Shapiro and Markoff (1997) stated that content analysis is only acceptable when the findings are assessed with another research method. In this research, expert interviews will be used as a data triangulation method to evaluate document analysis findings and to help to validate research results.

Participants

For the interviews, 2-4 subject matter experts were recruited from the Idaho National Laboratory (IDL). The participants will be a human factors researcher or nuclear engineer who has experience or knowledge of outage control management in the US NPPs. The details of the expert interviews are discussed separately.

Materials

The subject matter interviews were completed via Skyward or phone call. For a video call, the interviewees and interviewers used a laptop or desktop computer. With the permission of the interviewees, the video calls were recorded for a detailed analysis.

Data Analysis

The data analysis of this study is shaped by two main approaches: qualitative analysis and quantitative analysis. Qualitative analysis is a process of a rigorous investigation of situations, events, and interactions (Labuschagne, 2003) to create empirical knowledge. In this research, the main reason to use qualitative analysis is to create a meticulous examination of the characteristics of verbal data to reveal specific properties of human and team errors in NPPs. For this purpose, the qualitative analysis is divided into two different techniques:

a) Content Analysis: This method will be used to evaluate the official documents and structured expert interviews. It is also used to create the main categories, which are explicitly reported, in the official reports.

b) Thematic Analysis: This process will be used after the initial content analysis. During the content analysis data is divided into basic categories identified ahead of the analysis. The main purpose of thematic analysis is to reveal previously unknown themes of the official records with a deeper inquiry.

The quantitative analysis is concerned with quantifying specific features and patterns of the dataset, calculating the frequency of particular events, and measuring the distribution of the human error data in different situations. Table 8 shows the summary of the data analysis process of the study. The details of each section will be discussed separately in the following sections.

Table 6

The summary of data analysis process

<table>
<tr><th></th><th>Method</th><th>Data Source</th><th>Analysis</th></tr>
<tr><td rowspan="3">Data</td><td rowspan="2">1.Document Analysis</td><td>1. 1.Outage Licensee Event Reports (LERs) 2000-2016.
All US nuclear power plants are included.
The documents are retrieved from the Nuclear Regulatory Commission database.</td><td rowspan="2">a. Qualitative Analysis (Content Analysis, and Thematic Analysis)
b. Quantitative Analysis</td></tr>
<tr><td>1.2.Outage Inspection Reports (IRs) 2000-2017.
All US nuclear power plans are included.
The documents are retrieved from the Nuclear Regulatory Commission database.</td></tr>
<tr><td>2.Expert Interviews</td><td>2.1 Expert Interviews
The participants will be recruited scientists and researchers.</td><td>a. Qualitative Analysis (Content Analysis)</td></tr>
</table>

Qualitative Analysis

Content Analysis

An error is an intentional sequence of cognitive or physical actions that is unsuccessful to achieve its targeted outcome (Reason, 1990). Human errors in high fidelity industries are described as unsafe acts or violation of procedural directions (Reason, 2000). Human errors in a complex system are not always causes of failure. They can be consequences of systematic factors (Reason, 2000). Content analysis is one of the methods to use to investigate the causing factors. Content analysis is described as an organized, replicable research method for evaluating textual data into content categories based on coding directions (Stembler, 2001). It is a commonly used as a qualitative research method (Hsieh & Shannon, 2005). It provides a flexible methodology to evaluate textual data. Hsieh and Shannon (2005) presented three distinct content analysis methods: conventional, directed and summative. Rather than focusing on word counts, the conventional content analysis endeavors to interpret the meaning of textual data (Hsieh & Shannon, 2005). The conventional content analysis is usually used to collect data when existing theory or related literature review is limited (Hsieh & Shannon, 2005). Table 7 displays a summary of the content analysis that will be used.

Table 7

The summary of content analysis

Type of Content Analysis	**Codes**	**Sources of Codes**	**Data Source**	**Reason**
Conventional Content Analysis	Codes are defined based on Palo Verde LERs/ during LERs and IRs analysis.	Originated from the official documents.	LERs, IRs	Due to limited existing literature review or theory.

Conventional Content Analysis: the LERs and IRs

In this research, the conventional content analysis method will be used as the first step of rigorous document analysis. The conventional content analysis helps researchers to discover and describe keywords or key phrases of potential interest from textual data in a standardized fashion. The primary purpose of using this method is to collect and organize human and team error incidents in a systematic format. In this way, it is aimed to decrease the complexity of textual data by creating content categories. Figure 4 shows the tentative divisions of the initial content analysis: human error, team error and non-human error (technical error). The categories were selected in order to eliminate technical errors. The same content categories will be used for the analysis of LERs and IRs.

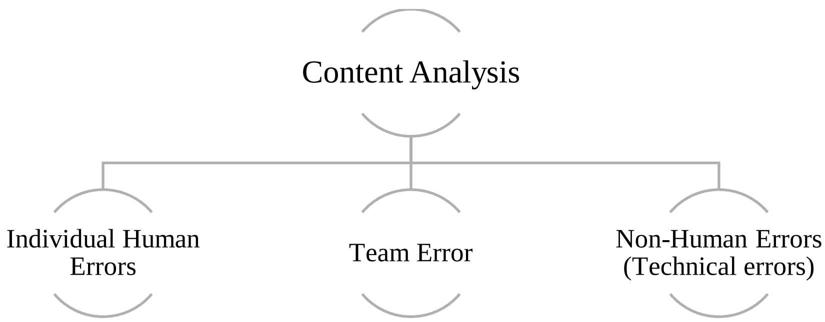

Figure 4. The categories of initial content analysis for the LERs and the IRs

The rules of the content analysis are described in Table 8. At the end of this study, the LERs and IRs will be divided into three categories, and non-human errors will be excluded from the research. In the next step of the document analysis, the team and human error reports will be subcategorized separately. This stage of the study is named thematic analysis.

Table 8

Content Analysis Categories for the LERs and IRs

Categories	Details
Individual Human Errors	*Description*: Incidents caused by individual mistakes or errors. *Inclusion Criteria*: a) In the report, human error should be clearly stated. b) The employee who is involved in the incident clearly stated. *Example:* *a) "*…Human Error resulted in the condition going undetected when the requirement to LRT the valve following manual operation was waived…"(LER 3692007002). b) "…The Shift Manager signed that the valve alignment had been completed following statement to that effect by the Outage Control Center Coordinator (a previously SRO-licensed individual). However, the alignment had not been completed…" (LER 2512008002).
Team Errors	*Description*: Incidents caused by multi employees' involvement. *Inclusion Criteria*: a) In the report, team error should be clearly stated. b) Employees who were involved with the incident should be clearly stated. *Example:* *a) "*... the human performance cause of this event was that the operating crew did not meet expectations for effective teamwork to ensure proper decision making…" (LER 3482015001) b) "…The Shift Manager and Control Room Supervisor did not provide effective oversight as unplanned activities caused a loss of focus…." (LER 4832008003)

Non-human Errors (Technical Errors)	*Description: Incident solely caused by technical issues.* *Example: "...* The direct cause of the malfunctioning rod control in-hold-out switch was attributed to dirt on the push buttons of the switch which resulted in sluggish operation…"

Thematic Analysis

Thematic analysis is defined as a procedure of coding and category construction based on data's characteristics (Bowen, 2009). A thematic analysis can be described as a form of pattern recognition within the dataset (Fereday & Muir-Cochrane, 2006), and creating pattern categories for related patterns into categories (Aronson, 1995). It gives an in-depth examination of specifically identified topics and themes. The process of thematic analysis has three main steps: 1) carefully examining the data, 2) constructing categories and performing coding, and 3) revealing essential themes to a phenomenon (Bowen, 2009).

1. *Closely examining the data:* In this research, the content analysis is used as the first step of thematic analysis. During the content analysis, the data is examined and the incidents reports related to human and team errors are separated for a thematic analysis.
2. *Constructing categories and performing coding:* The initial Palo Verde LER content analysis provided very important information. Based on the patterns of the analysis, main categories were created. The analysis showed that there are five main categories that can be used to create thematic classification: *a)*

Human Cognitive Error, b) Team Cognition Error, c) Procedural Error, d) Organizational Error, and e) Design Errors. Figure 5 displays the thematic analysis categories. Based on these five categories, the LERs and IRs will be categorized for a more in-depth analysis. In the code manual, Table 9, the description of the group, inclusion criteria, one explicit example and one implicit example are shared.

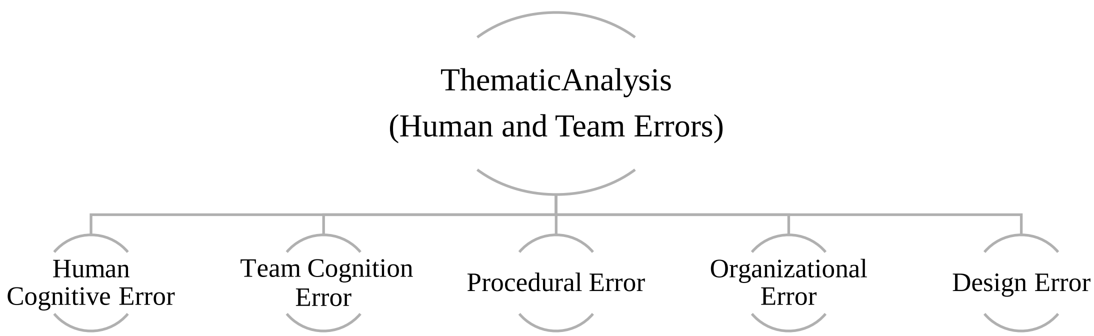

Figure 5. The categories of thematic analysis for the LERs and IRs

3. *Revealing essential themes*: After performing coding and sorting data into categories, each category will be examined thoroughly to determine similarity and relationships between the datasets. By using this method, it is aimed to discover the patterns between the datasets and to identify relevant themes.

Table 9

The code manual of thematic analysis

CATEGORIES	DETAILS
Individual Human Error	*Description:* Incidents caused by individuals during a task completion process due to misunderstanding, memory overlap or information gap or other factors. These errors can be occurred in performance, planning, or decision-making process. *Inclusion Criteria:* In the document, human error should be explicitly or implicitly mentioned in the cause section or in the corrective action section of the report. *Explicit Example: "*...The maintenance technician performing the reassembly did not utilize all applicable human performance tools (i.e., self-checking, questioning attitude, and stop when unsure) during the reassembly process...(LER 4132012003)". *Implicit Example:* "…The cause of the error is attributed to inadequate programmatic controls for maintaining configuration control of the 5x5 array...(LER 2982009001)".
Team Error	*Description:* Incidents caused by two or more than two personnel during a task completion process. These errors can be triggered due to coordination, communication, task performance, supervision or leadership issues. *Inclusion Criteria:* In the document, team error should be explicitly or implicitly mentioned in the cause section or in the corrective action section of the report. *Explicit Example: "*...The cause of this event was inadequate communication between the SM and RE personnel which led to the incorrect conclusion that the CEA position indicators were inoperable…(LER 3682008002)". *Implicit Example: "*...The work instructions for the normal control system calibration performed in Mode 5 was not performed as written …(LER 3682014004)".

Procedural Error	*Description:* Incidents caused by inadequate guideline, unsatisfactory procedural direction or lack of sufficient operational guidance. *Inclusion Criteria:* In the document, procedural issues should be explicitly or implicitly mentioned in the cause or the corrective action section of the report. *Explicit Example: "*...The root cause analysis identified the inadvertent output breaker trip to be a result of inadequate procedural guidance for setup of the emergency diesel generator over speed trip limit switch. BFN had no formal procedural guidance for OTLS setup…(LER 2592011003) *Implicit Example: "*...The cause of this event was inadequate verification techniques used to verify the correct power cable to cut. Contributing causes were: Incorrect labels, a difficult to read label, incorrect drawing, and using an instrument to check for voltage when no voltage should be expected. …(LER 3152010001)".
Organizational Error	*Description:* Unintended errors caused in organizational setting, which can be caused by individual or multiple organizational roles. *Inclusion Criteria:* In the document, organizational issues should be explicitly or implicitly mentioned in the cause section or in the corrective action section of the report. *Explicit Example: "*...The cause of this event was a lack of management reinforcement of standards and expectations associated with procedure adherence…(LER 2442006004)". *Implicit Example: "*...Weaknesses in the schedule development process led to missing or incorrect logic ties to the appropriate plant conditions and modes…(LER 2442006006)".

Design Error	*Description:* Errors caused by design related mistakes, which take time to noticed. Usually these errors are identified as “latent human error”. *Inclusion Criteria:* In the document, design issues should be explicitly or implicitly mentioned in the cause section or in the corrective action section of the report. *Explicit Example: “*The root cause of this event is a design vulnerability associated with relaxation of the EDG 3 fuse holder fingers which was not properly mitigated. The existing design lacks circuit continuity indication that is not mitigated by design or testing…(LER 3252016002) *Implicit Example:* “…The root cause of the event was the FCS design process failed to identify the silver plating of bus bar material as a critical interface when specifying replacements for the original circuit breakers…(LER 2852011008)”.

Structured Expert Interview

Interviews have been used as a powerful tool to collect qualitative data. For this research, a structured interview method is selected. This interview method is aimed to keep the interviewees focused on the research topic and data analysis results. The main intention of the expert interview is to have individuals with knowledge about NPP operations evaluate and validate the research findings. The interview questionnaires will be generated after the thematic analysis of the LERs and IRs. The questions will be focused on the results of the document analysis. Open-ended questions were used to obtain detailed answers from the interviewers. Each interviewee had sufficient time to think about and answer the questions. With the interviewees’ permission, each interview session was recorded. The interview transcripts were summarized and shared with the participants. The transcript summaries were included to present the opinions of the participants on the data analysis results.

Quantitative Analysis

Quantitative analysis is concerned with quantifying the similarities, differences, and relationships within the dataset (Labuschagne, 2003). The quantitative analysis is used to create objective measurement and statistical comparison to the qualitative study. It establishes a calculable reflection of research findings. By using quantitative analysis, it is intended to measure the frequency of codes and keywords and to quantify the data characteristics to understand the relationships between the themes.

Reliability

In Stemler 2001 article, it is indicated that reliability of a content analysis or thematic analysis can be discussed in two terms:

a) *Stability:* Obtaining the same results by using the same codes. It can be also defined as consistency of the coding process in the same research.

b) *Reproductivity (inter-rater reliability):* Replicating the same study by using the same coding scheme in the same documents (Stemler, 2001). To create consistency and to measure the reliability of the coding, two different raters, who have no previous LER knowledge, will use the code manually to categorize a random data sample. Based on an agreement between the raters and the researcher, Cohen's Kappa equation will be used (Cohen, 1960) to calculate the reliability of the coding. The Cohen's Kappa equation is:

$$K = (Pa-Pc)/(1-Pc)$$

where:

Pa= the proportion of units that the raters agree upon

Pc= the proportion of units for agreement by chance (Cohen, 1960). If Kappa score approaches “1”, then it is assumed that coding is perfectly reliable. If Kappa score is close the “0”, then it is anticipated that the coding is very poor (Stemler, 2001). Some researchers declared that “0.61 and up” is a reasonable kappa score for an acceptable overall agreement (Wheelock et al., 2000 and Landis &Koch, 1977). In this research, it is aimed to obtain 0.61 or higher for the agreement between the raters.

Validity

In this research, all the research methods are selected to find the best approach to answer the research questions. Choosing the best practices is as vital as validating research findings. It is expected that the data triangulation validation based on three information sources will generate the research validation. By using three different resources, it is aimed to corroborate the research results through cross verification and categorization validation. The data resources are:

a) Comprehensive Licensee Event Reports (LERs) analysis between 2000-2016: Total of 628 reports from the US NPPs,

b) Comprehensive Inspection Reports (IRs) analysis between 2000-2017: Total of 46 reports from the NPPs,

c) The subject matter expert interviews that have experience with human factors issues in the US NPPs.

The LERs and IRs analysis provide critical insight information. The subject expert interviews will help to validate accuracy of data analysis, evaluate the research results and understand the meanings of the findings.

Ethical Consideration

All research ethics rules were followed during data collection and data evaluation process. In the study, informed consent forms, participants' privacy, data confidentiality, the right to withdraw from the research and the right for recording interviews and text of discussions by the participants were observed.

CHAPTER 4: Results

Before the content analysis, the Cohen's Kappa equation (Cohen, 1960) was used to test the reliability of the codes. Two graduate students coded 10 different cases. Both students had no experience with the LERs and IRs. They received a total of 35 min presentation on the coding process The Kappa scores were calculated 0.71, which is higher than the accepted reasonable reliability score of 0.61 (Wheelock et al., 2000 and Landis & Koch, 1977).

Palo Verde Nuclear Power Plant LER Analysis

In NPPs, there are six different operating configurations, which are categorized based on based reactor power level, reactor pressure, water temperature, and other technical details. During an outage period, five different operation modes should be completed before reactivation. For a better understanding, Figure 6 shows the average working hours of teams to complete each mode for a nuclear reactor. This data provides important information about the workload of outage control. Incensement on workload may affect the number of reported cases. According to average working hours, Mode 6-Refueling, Mode 5- Cold Shutdown and Mode 4- Hot Shutdown take the most time, respectively.

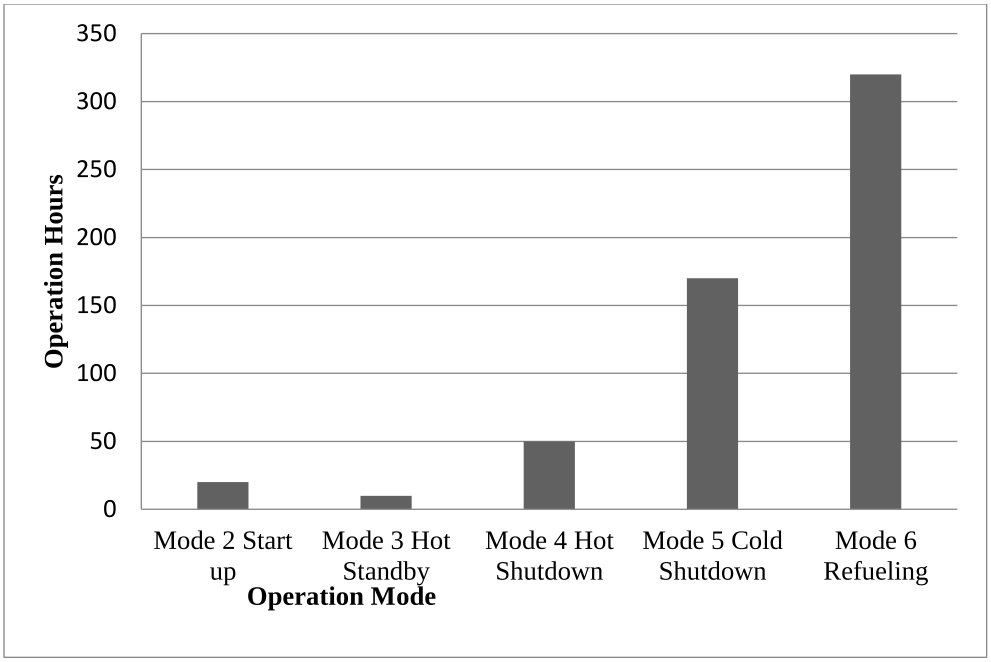

Figure 6. Operation hours for each reactor mode

Using the data from Palo Verde Nuclear Power Plants, the first Licensee Event Reports (LERs) were coded. The lists of the LERs are shared as Appendix B on page 113. The data pertained to the outage control period between 1990-2015. No keywords or phrases were used to filter the data. The analysis was started with simple initial elimination. Based on initial analysis, 54 LERs were excluded because there were caused by technical problems such as malfunctioning pump, aging instrument parts...etc. Table 10 shows the results of content analysis of the LER documents. The highest percentages are bolded on Table 10.

Table 10

1985-2015 Palo Verde Nuclear Power Plant LER Content Analysis by Operation mode

Mode (Palo Verde Unit 1, 2 and 3)	Total incidents	Reported Human Errors out of total events	Percentage of Reported Human Errors
Mode 2- Start Up	13	7	53.8%
Mode 3- Hot Standby	**36**	**24**	**66.7%**
Mode 4- Hot Shutdown	40	19	47.5%
Mode 5- Cold Shutdown	**76**	**42**	**55.3%**
Mode 6-Refueling	63	27	45%

Based on Table 10 data, the relationship between the operation mode and the reported human errors was analyzed. The Chi-Squared test of independence was performed to examine the relationship between the operation mode and the reported human errors. The relationship between these variables was significant. $\chi 2$ (4, N=228), $p < .001$.

Yearly Analysis

Figure 7 shows the error distribution based on operation year between 1985-2015 of the Palo Verde Nuclear Power Plant. The highest number of individual human errors and team errors are reported between 1985-1990. Then, there is a significant change in both categories. Individual human errors were relatively stable across time between 1990-2015. Most team errors and individual human errors were reported between 1985-1989, which was the first four years of the operation.

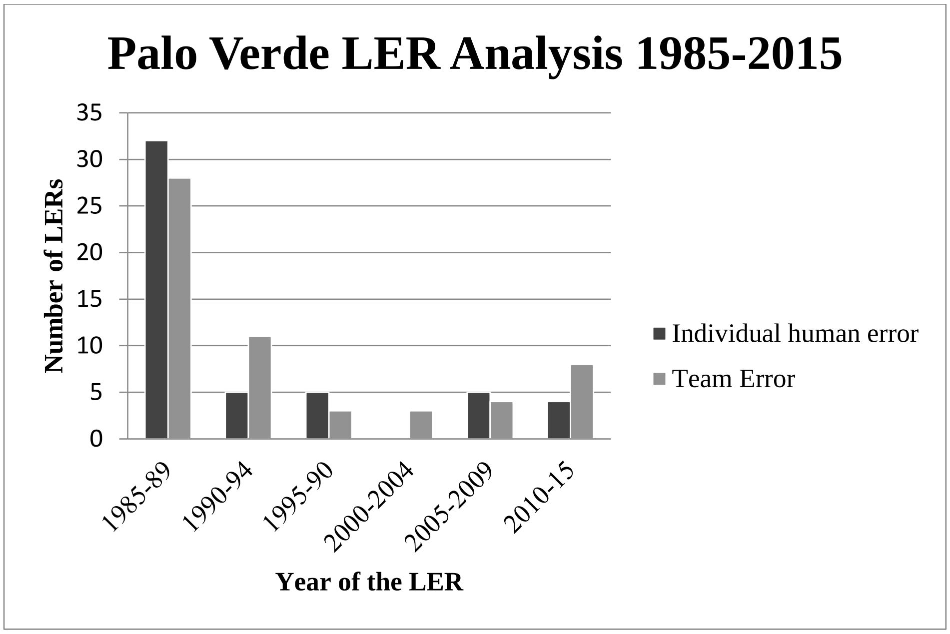

Figure 7. Individual Human Errors vs Team Errors between 1985-2015

Comparison of Different Reactors

Palo Verde Nuclear Power Plant has three identical nuclear reactor units and each reactor has its own outage schedule. The same outage control teams managed all three nuclear units outage controls. In order to understand the impact of the operation modes on human and team errors, the LERs are categorized according to the modes separately for each nuclear reactor units were separated. Figure 8 shows the number of individual human errors of each reactor unit during different operating modes. Similarly, Figure 9 shows the distribution of team errors per operation mode.

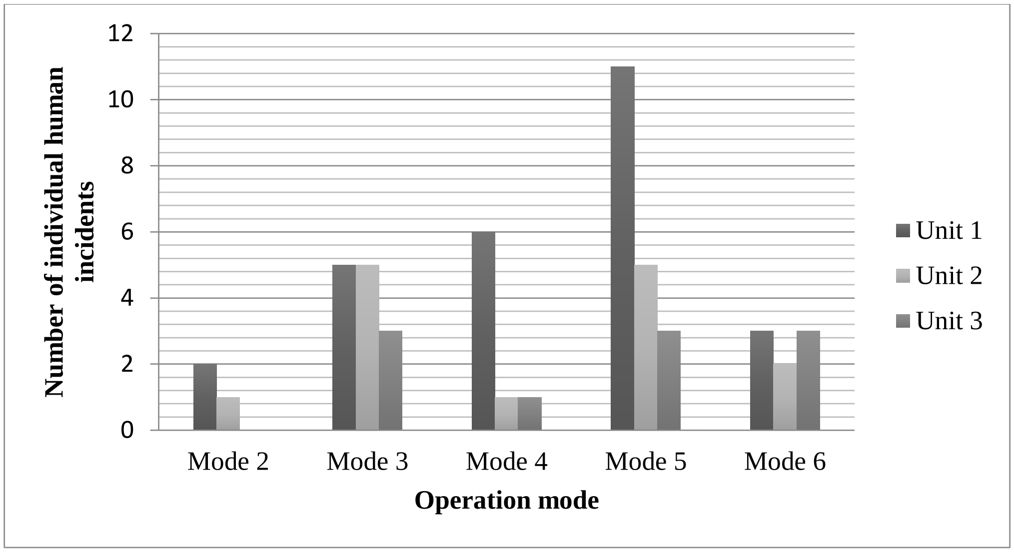

Figure 8. 1985-2015 Individual human errors vs operation modes

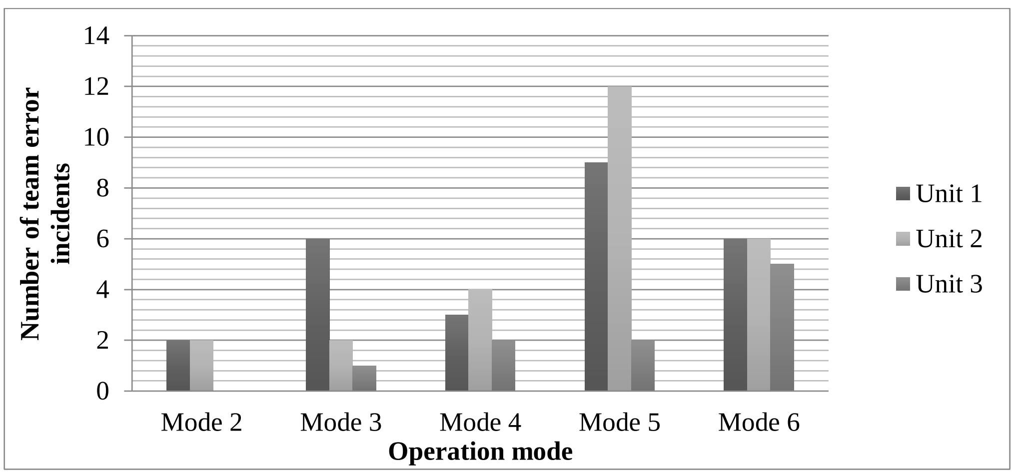

Figure 9. 1985-2015 Team errors vs operation modes

The results show that the highest percentage of human errors was reported during "Mode 5 – Cold Shutdown" and the second highest percent of team errors was reported during "Mode 6- Refueling". The incident reports provide critical data to characterize the fragility of outage management. The results are consistent with Laakso et al.'s (1998) research.

Additionally, the analysis shows that there are different patterns of errors between different reactors. Unit 3 human error and teamwork error ratios are less than Unit 1 and Unit 2. Because it is a complex system, multiple factors can be attributed to this pattern of results. Further research should be conducted to understand the factors underlying this pattern.

Palo Verde NPP 1985-2015 Outage Control Incident Analysis by Job Categories

During an outage control period, temporary contractors who work together to complete the outage control with facility employees. Some of the LERs also reported temporary workers could involve in the reported incidents. In the Hunlen et al. 2012 research, it was stated that during outage control, non-facility temporary workers may not be unfamiliar with the plants. This situation increases the number of human errors and the incidents. In order to investigate that, all reported personnel titles were recorded and the list is given as Table 11. Initial analysis showed that there were a limited number of non-facility contractor incidents recorded, whereas there is a higher number of an incident associated with facility employees. Only 5 out of 87 non-facility contractors were involved an incident. This finding contradicts with the research of Hunlen et al. 2012.

Table 11

Number of reported job categories in the LER

Job categories	Facility Non-Licensed	Facility Licensed	Non-facility Contractor	Total
Personnel	5	22	2	28
Supervisor	2	11		13
Operators	1	12		13
Control room operators		5		5
Control room personnel		5		5
Administrative		4		4
Planner	1			1
Shift Technical Advisor		2		1
Assistant Shift Supervisor		1		1
Maintenance		2		2
Engineer	1	3	1	5
Technician	5	1	2 (non-licensed)	8
Operation Crew		1		1

2000-2017 Thematic Analysis of the US Nuclear Power Plants LER

The list of the nuclear power plants and their LER numbers are shared in Appendix D. All reports were coded based on five categories. LER reports include a section, Root Cause Analysis, on each report to explain possible causing factors. During the analysis, it was observed that some of the incidents occurred because of overlapping issues and multiple factors were listed. For better understanding, a Venn diagram is created, and shared below. The highest error was reported as “team errors” and it is followed by procedural incidents (caused by lack of procedures, insufficient procedures or inadequate work packets), organizational incidents (caused by organizational mistakes of management, supervision, planning, resourcing, training…etc) and design errors (incidents caused by outdated design issues) respectively. Individual Human Error ratio was only 2.9%. There are multiple overlapping areas formed between these categories. The highest percentage was observed between Team Errors and Procedural Errors, which is significantly higher than Team-Organizational and Team-Design Errors. The analysis results showed that 77.3% incidents have a single main reason, 19.8% of the incidents have two main reasons and only 0.4% incidents had three main reasons. There was no report found, which caused by four categories.

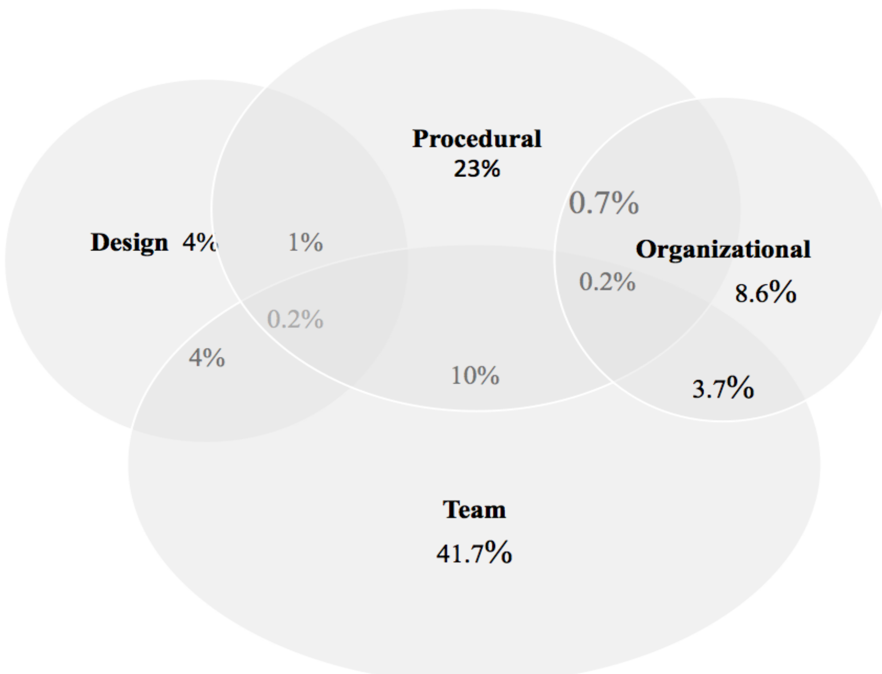

Figure 10. 2000-2017 US Nuclear Power Plants Thematic Analysis

The US nuclear plants have 23 single-reactor, 32 two-reactor, 9 three-reactor and 1 four-reactor nuclear power plants. The organizational management of single reactor is usually different than 3-reactor nuclear power plants. The reactor number is affecting the number of facility employees who work during outage control and the total number of outage control in a year. As it is mentioned previously, usually each reactor will have an outage control every 6 months. For a better comparison, the power plants were grouped into single reactor, two- reactor and three-reactor. Each graph shows the number of LERs that were analyzed and the number of human errors.

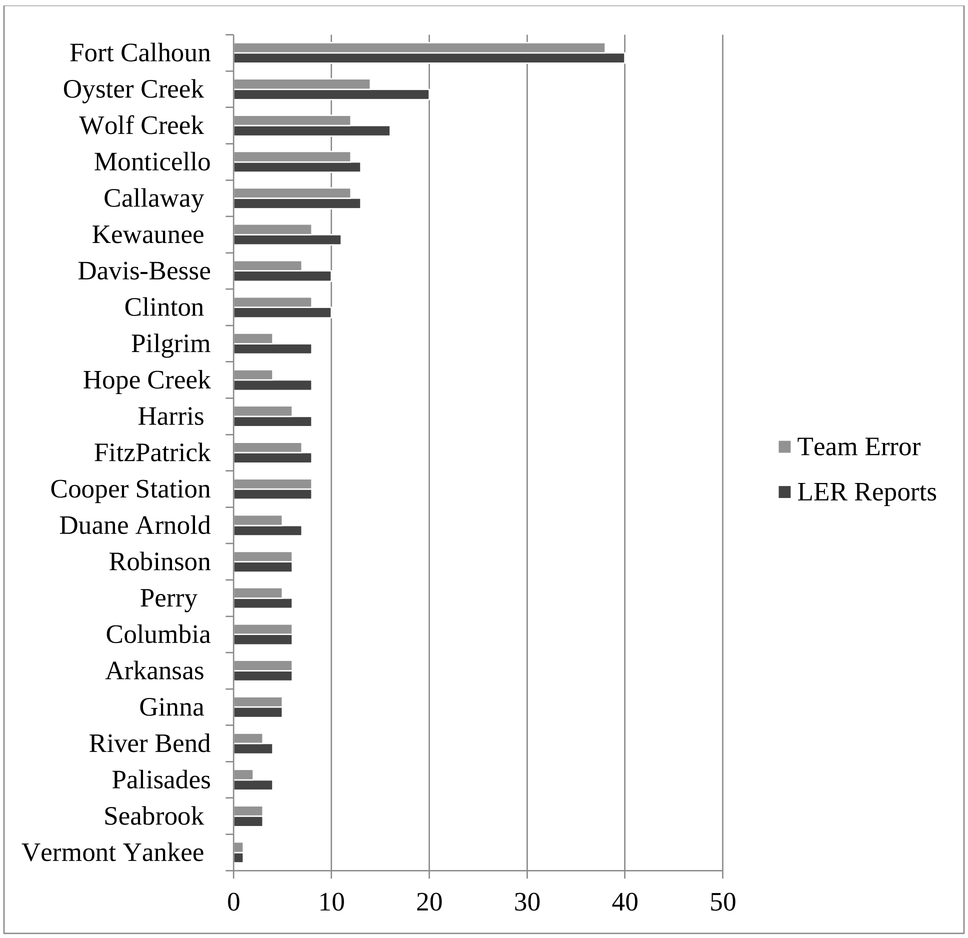

Figure 11. Single reactor nuclear power plants LER and Team Error comparison

The highest number of incidents for single reactors was found as Fort Calhoun that is higher than any other plant. The lowest number of LER and Team Error was reported for Vermont Yankee nuclear power plant.

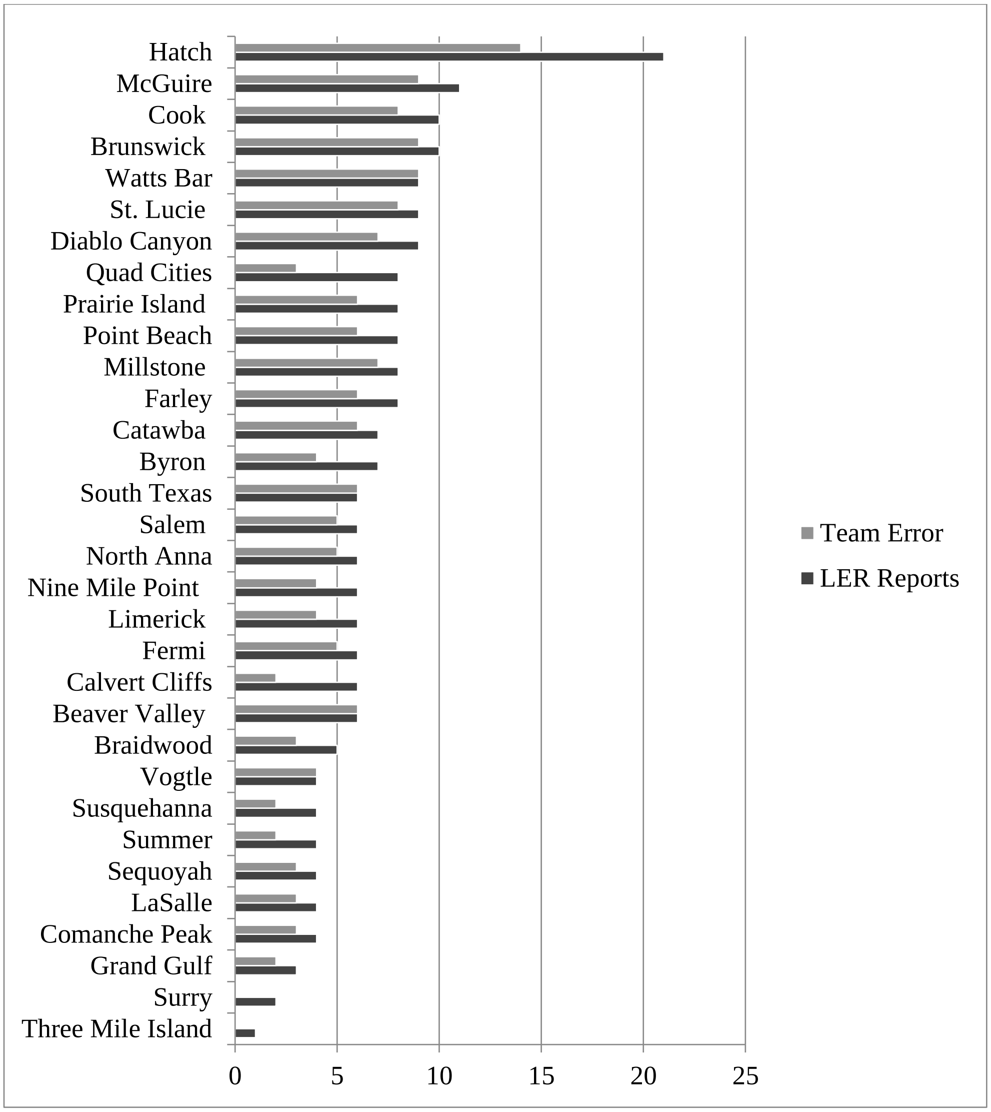

Figure 12. Two- reactor nuclear power plants LER and Team Error comparison

Hatch Nuclear Power Plant has the highest number of the number of LERs and reported Team Errors. Three Mile Island Nuclear power plant has the lowest number of LERs. Surry and Three Mile Island had no reported team errors in this group.

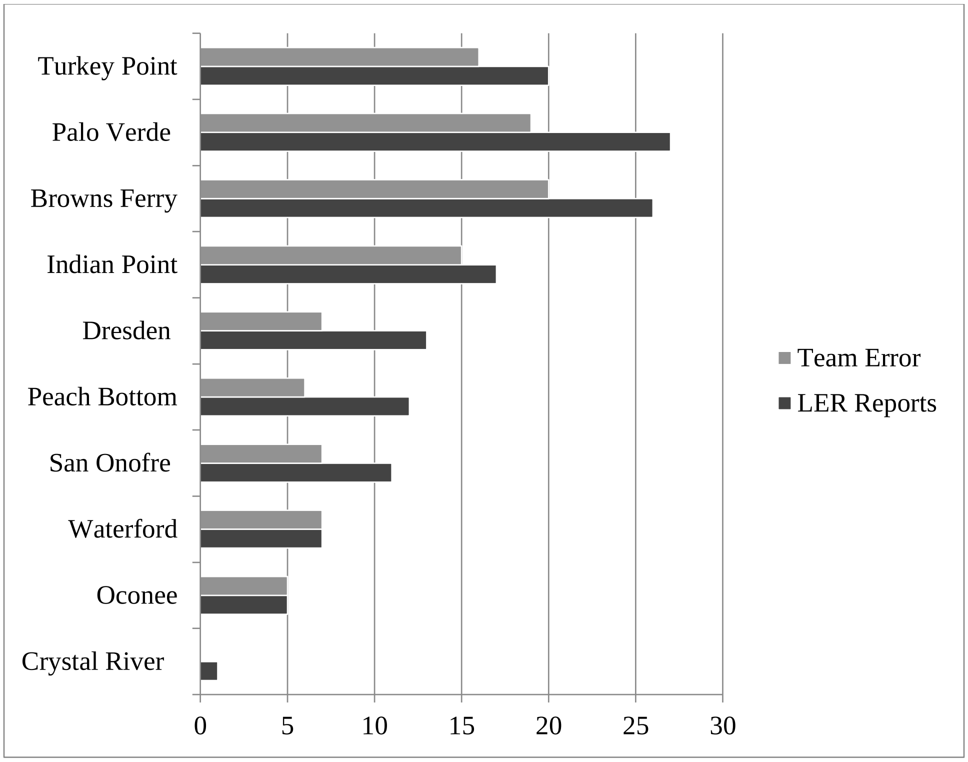

Figure 13. Three-reactor and four-reactor nuclear power plants the number of LER and Team Error comparison

Table 12

Comparison of different reactor numbers vs Human Errors

	LER Reports	Human Error	Percentage
One-Reactor	221	182	82%
Two-Reactor	216	161	74%
Three & Four-Reactor	139	102	73%

Palo Verde Nuclear Power Plants reported the highest number of the LERs among this group. The highest team errors were reported by Browns Ferry Plants. Crystal River Nuclear Power Plants had the lowest LERs and there was no reported team error. Turkey Point nuclear power plant was the only four-reactor plant in this analysis Table 12 shows the number of LERs, reported human errors based on reactor number. In order to investigate the relationship between the number of reactors and team errors, a Chi-Squared Independent analysis was completed. The relationship between these variables was significant. $\chi 2$ (2, N=576), $p < .001$.

Key phrases

All LERs incidents were reviewed and some of keywords were categorized based on their thematic analysis. The team, procedural, organizational and design errors were coded by using the following key phrases shared in Table 13. These key phrases provide insight about the challenges of the teams who work during the outage control.

Table 13

Key phrases used in the LERs to report Team, Procedural, Organizational and Design Issues

Team Errors	**Procedural Errors**	**Organizational Errors**	**Design Errors**
Failure to recognize	Lack of procedure	Work management problem	Design calculation
Decision making	Inadequate procedure	Work control	Design change
Problem identification and resolution	Not adhere to procedural requirements	Resources and work practice	Latent design issues
Conservative decision making	Inadequate work package	Work instruction issues	
Reviewing decision		Not providing reasonable assurance	
Not appropriately incorporate risk insights		Supervisory and management oversight issues	
Not using conservative assumption		Insufficient guidance and training	
Fail to follow the procedure		Not implementing appropriate corrective action	

Inspection Reports (IRs) Analysis

Operation mode

Inspection Reports (IRs) are official reports based on Nuclear Regulatory Commission (NRC) regulation. The inspectors from NRC regularly review the plants and submit reports for improvements. Previous research examined higher workload of the nuclear power plant teams (Juhasz, & Soos, 2007). In order to understand the impact, the reported operational modes were noted during the coding process. A total 41 IR documents were analyzed. Twenty-four out of 41 reports did not mention incidents in the operation modes. Only 17 IRs stated the incident in the operation mode. Six out of 17 IRs reported multiple incidents, which were also counted separately. Based on these limited data, it was found that Mode 5 –Cold shutdown had the highest number of incidents (9/17) and Mode 6- Refueling was the second highest (6/17) and Mode 4-Hot Shutdown followed (6/17), respectively.

Yearly Analysis

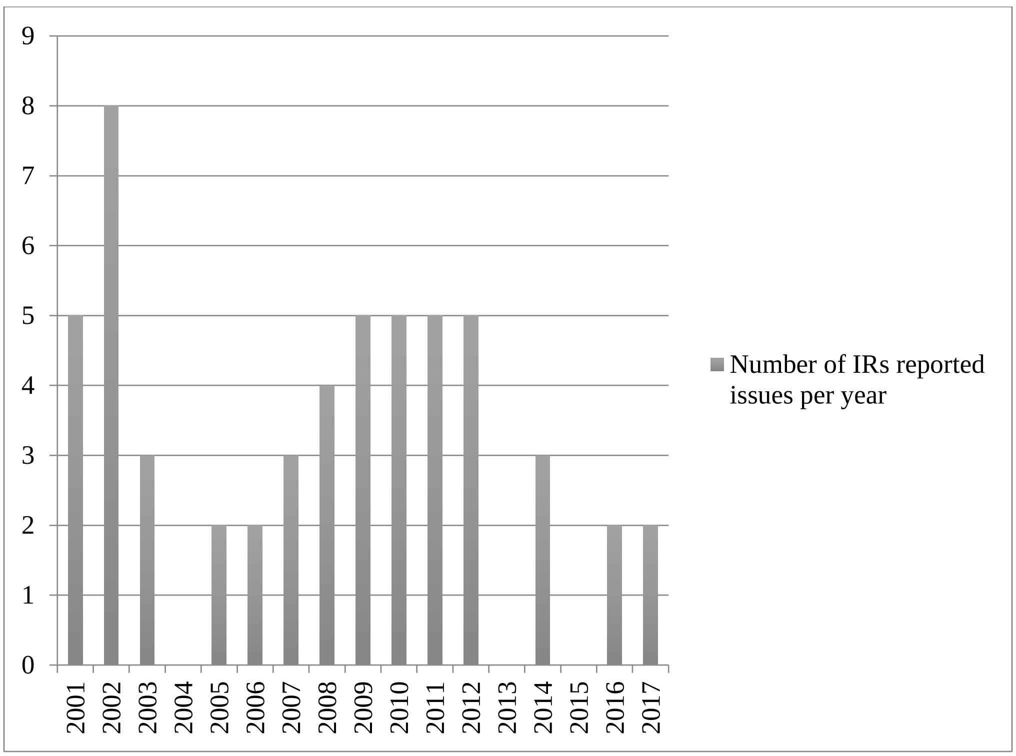

Figure 14. 2000-2017 IRs reported outage control issues related with human errors

The most of Inspection Reports (IR) were reported in 2002. There are some years in which there was no IR report submitted to the NRC. These years are 2004, 2013 and 2015. The average of the IR is around 3 reports per year.

2000-2017 Thematic Analysis of IR

The IRs were prepared based on the NRC's guidelines and taxonomy, which are intended to inspect and report any concern regarding the operation of the US plants. These reports focus on the operation practices of the licensees. This situation reflected on the IRs. Table 14 shows the reported responsible parties. The data analysis revealed that 71% of the reports stated that the licensees are the main responsible party.

The incidents were grouped based on the thematic analysis and explicitly reported reasons. The analysis is shared in Table 13. The highest percentage of the incidents was 17%. Problem identification and resolution was the highest percentage team error that was reported. The incidents related with work control, management, and work practice issues are listed under Team-Organizational Error. These incidents were also calculated as 17%. The second highest percentages were Team-Failure to follow procedure and Organizational-Inadequate procedures.

Table 14

Thematic analysis of Inspection Reports reported reasons

Category	Reported Reason	Number of incidents	Percentage
Team	Problem identification and resolution	8	17%
Team	Failure to follow procedure	5	10%
Team	(Conservative) decision making	4	8.70%
Team	Not appropriately risk insight	1	2.20%
Organizational	Inadequate procedure	5	10%
Organizational	Resources	3	6.50%
Organizational	Training	3	6.50%
Organizational	Adherence procedure	3	6.50%
Organizational	Inadequate work package	2	4.35%
Organizational	Supervisory and Management	1	2.20%
Team and Organizational	Work control/management/practice issues	8	17%
Team and Organizational	Implementing appropriate corrective actions	1	2.20%
Human		1	2.20%
Design	Design/ Design calculation	2	4.35%

Expert Interviews

For the expert interviews total of 26 recruitment emails were sent to researchers and nuclear engineers. A total of three researchers accepted to participate. The longest interview was 1 hour 10 min and the shortest interview was 23 min. Each interview was audio recorded with verbal or written consent. There was no video recorded interview. All recordings were deleted after the interview transcription.

The interview questions are shared in Appendix G, page 183. There were total of five questions are asked. Each question is focused to collect information about:

1. Question: Five main categorization: Individual, Team, Procedural, Organizational and Design Errors
2. Question: Procedural Errors
3. Question: Organizational Errors
4. Question: Design Issues
5. Question: Individual vs Team Error ratio

In Table 14, each participant's answers for each question are briefly summarized. During the interview each participant shared his or her agreement, disagreements, suggestions and unique inside information. Table 15 shows the comparison of interview results based on similarities and differences.

Unique background of the participants gave a chance to get different approaches for the same questions. Even though they all acknowledged four of five questions, they all brought a unique aspect to explain the data. Table 15 shows the summary of each interview based on their similarities and differences.

Table 15

A brief summary of expert interviews

Question	**Interview 1**	**Interview 2**	**Interview 3**
1. Categories: Human, Team, Procedural, Design and Organizational Issues	Categorization makes sense at certain level. It is a reasonable way to organize the data from a theoretical standpoint. But there are some actual errors, which may not cleanly fit in such as latent and active errors.	The categorization fits perfectly with the Human Error Taxonomy that is being used in IDL. But we usually do not differentiate between individual error and team errors.	Categorization makes sense, but there should be more detailed subcategories. For example team errors can be caused by communication, coordination…etc.
2. Procedural Issues	Two main issues of procedural errors are identified here; insufficient procedures or good procedures but not followed.	There is difference between procedures in control room and the field. The procedures in control room are constantly rehearsed and revised. But in the field there are some environmental distractions.	There are some procedural errors, and there are a variety of procedural error types.
3. Design Issues	There might be some design issues, but nuclear industry is good at working as a team to fill	NPPs are being operated for a long time and tested in a sense. All the possible issues are	Good experienced operator can correct design issues.

	the gaps. NPPs are operated since 1950s, all the design challenges are already known.	identified. Older plants have more experience and fewer incidents.	
4. Organizational Issues	There are many subcontractor workers and augmented teams of these workers. For any organization, the ability of people to work effectively during outage is hard. So organizational standpoint it can be challenging.	Organizational issues are very broad; it can be anything from management to structure. But there are plants with good management practices and there are plants, which prioritize production over safety.	I think it depends on culture. For instance in Korea, NPPs are public entities. In US they are private organizations. Some researches show Korean supervisor has higher authority to determine and make a decision. But in US, field operators can share their opinions with senior operators.
5. Individual Errors vs Team Errors	I think it is generally correct. Almost every activity in a plant is completed by a team. So the data makes sense.	In some ways it is harder for teams to do an error. They may be reluctant report a worker failure to not to blame anyone.	I guess this is also related with culture. In South Korea, individual errors are higher than team errors. In US plants it might be different.

Table 16

Comparison of interview findings

Question	Similarities	Differences
1. Categories: Human, Team, Procedural, Design and Organizational Issues	All three participants acknowledged th individual, team, design, procedure anc organizational issues are observed in the nuclear power plants. Participant 1 and participant 2 defined categorizations a good fit.	**Participant 1** mentioned cases that might not fit into these categories. It is noted that some of the cases are caused by overlapping issues. **Participant 2** stated that current human error taxonomy does not differentiate between individual and team error, which is a valuable distinction in this research. **Participant 3** recommended to get into details of each incident and create subcategories: training, communication, coordination…etc.
2. Procedural Issues	All three participants acknowledged that there are different types of procedural errors: lack of procedure, insufficient procedure and not following procedures can be a cause of an incident.	**Participant 1** stated that it is an important distinction to make between individual error vs crew error. **Participant 2** emphasized that there should be an alienation between control room procedures and construction field. The control room procedures are defined as well practiced and controlled, where as construction field procedures are open to environmental distractions. **Participant 3** emphasized that there are different types of procedures and variety of

		reasons to have procedural errors.
3. Design Issues	All participants accepted that there are fewer design issues than other factors, they all emphasized that well trained teams and crew members can overcome these design challenges.	**Participant 1:** Experienced nuclear industry teams are able to close the gap. There are significant years of operation and experience, teams already encountered with different design issues. They were able to work around it. **Participant 2:** The NPPs were operated for years and tested many different ways. So these design issues were not dormant for years. Older plants have fewer errors of this nature. There are also rare events where the plant is not put through before. Fukushima was an example of design problem, tough. It was very rare incident. **Participant 3:** Operators could address almost all design issues. It is mentioned that not all design issues create nuclear accidents.
4. Organizational Issues	All participants stated that outage control is a complex and complicated task for any organization.	**Participant 1:** It is not my expertise area to discuss. But organizational structure standpoint it is a challenge to work with augmented teams during outage. **Participant 2:** There is a shift in work force; less nuclear submarine operators are in the system, who are very well trained and authoritarian. They are replaced by high school grads. The proficiency of navy training is hard to fulfill. The change in work force affects the organizational management. **Participant 3**: The organizational structure is

		related with the culture. For example South Korean plant structures are different than the US plants.
5. Individual Errors vs Team Errors	The statement of each participant was focused on a different aspect of team vs individual error.	**Participant 1:** That is generally correct. Teams are responsible to complete tasks. There is usually second checker: a reader and a doer. Better and greater technology can help the teams to make up some deficiencies. **Participant 2:** Teams are less likely to make a mistake. The lower number of individual error might because of reluctant of reporting. In LERs it is hard to point finger to supervisor or another worker. **Participant 3:** Like organizational culture, the US plants are different than Korean plants. In US plants, team members can share their opinion during decision-making process. In Korean plants supervisors have the authority to make a decision and they are more authoritarian.

CHAPTER 5: Discussion

This research was aimed at investigating the team errors and identifying the other factors that are associated with team errors during outage control management. Detecting the teams and associated factors will help the nuclear industry increase the resilience of the system. This study was directed at answering two main research questions.

1. ***Research question:*** *What are the individual human, team, and organizational issues associated with events reported in the official reports such as the Licensee Event Reports and the Inspection Reports?*

   The answer to this question is covered in terms of individual human errors, team errors and organizational issues respectively.

Individual Human Errors

Results from the analysis of Palo Verde LERs showed that most individual human errors occurred during the initial operation years and have decreased over the years. These data may indicate that gaining experience in the working environment, improving based on previous mistakes and taking necessary actions over the years have affected number of the reported individual errors. But still more investigation is needed to support it.

Also the number of reported individual errors was low compared to other errors in the LERs and IRs. This finding might be related to multiple factors:

1. *Reluctance to report a plant employee:* O The expert interviews revealed that the employees of nuclear power plants are less likely to report individual mistakes in the LERs because they did not want to point a

finger at any other worker.

2. *Purpose of official reports:* In the IRs, the investigators mostly focused on the licensee performance, plant management and work practices. So they reported fewer individual errors than any other kinds of errors. In these reports the number of reported procedural errors, management issues, team errors and design issues were higher than individual human errors.

3. *Complex nature of working environment:* The outage control tasks are complex and interconnected with different tasks. Working in a highly complex working environment requires a strong collaboration and effective teamwork between the workers. The expert interviews showed that teams, not individual workers, mostly complete the tasks. Multiple workers are assigned a task to read to procedure of a task, perform the task requirement and review the quality of the work. Individual human errors are more likely to be corrected in the field.

Team Issues

Both LERs and IRs provided information related to team challenges in the US nuclear plants. The analysis of US nuclear power plant LERs showed that more than 40 % of the reported errors were associated with team errors. Based on the analysis, these errors could be related to two factors: Team Cognition problems in teams and not following procedures as a team.

1. *Team Cognition problems:* The analysis of LERs and IRs keywords provide essential details of reported team issues. According to the LERs, the teams that committed errors struggled to recognize failure, to make a decision, to identify problems and resolutions, to review decision, to analyze risk insights

and to use conservative assumptions. According to the IRs, the teams struggled to identify problems and resolution, to make conservative decisions, to review work practices and to implement appropriate corrective actions. Both LER and IR analysis support the results of previous studies, which identified the nuclear power plant team issues such as decision-making (Crichton, & Flin 2004; O'Connor, O'Dea, Flin, & Belton, 2008), identifying or diagnosis problem (Roth, Mumaw, & Lewism, 1991), and risk analysis as a team (Montgomery, Toquam, & Gaddy, 1991). These findings showed that the teams did not struggle because of lack of shared mental model. But they struggled due to team interaction issues. ITC theory emphasizes that teams should be trained to improve team interaction to increase overall team effectiveness.

2. *Not following guidelines or procedures:* Teams who work in a complex system are designed to work as a team with detailed procedures and guidelines. In nuclear plants, work packets and procedures are created by the organization to guide the teams to complete a task. The LERs, IRs and expert interviews revealed that some of the events were caused by not following the provided procedure as a team. One of the experts stated that there is a difference between not having a procedure and not following a procedure for human errors in the nuclear industry. He explained that sometimes due to time pressure or environmental distractions, teams fail to follow the given procedures and guidelines. He explained this situation with an example. The workers in the field have to get some of the tools from workshops in order to complete a task. Sometimes they have to wait in the line to get a tool. During this waiting time, teamwork is disrupted and

team members are distracted unintentionally. These situations increase the possibility of not following the given procedure to complete tasks. Similarly, another expert mentioned that financial pressure and time increase the stress on the team, and some times teams skip necessary steps to save time.

All this information showed that teams in nuclear plants strongly depend on procedures. Although procedural guidelines are a vital component of team performance, the teams should be able to foresee procedural disruption and take corrective actions. In addition to that, teams in nuclear power plants are mostly *ad hoc* teams. Ad hoc teams are not intact teams (Hinski, 2017). Because of the nature of these teams, following procedures and guideline as a team can be a challenge. Hinski (2017) investigated *ad hoc* medical team trainings. In this research, it was found that in order to improve team performance, team training methods should be focused on how the team functions, and performs as a unit effectively (Hinski, 2017). These findings aligned with the principles of ITC and emphasize the importance of training that are aimed to improve team interaction. The adaptive teams, which are trained on team interaction, can handle procedural disruptions.

Organizational Issues

Thematic analysis of LERs and IRs revealed that organizational factors have an important impact on the number of reported incidents. These factors include; providing necessary training, creating sufficient procedures and guidelines, and enhancing safety culture.

1. *Providing necessary trainings:* Both LERs and IRs showed that some of

the human errors occurred because of lack of sufficient training. In this research, training issues were listed under the organizational issues. The nuclear industry regulations keep the nuclear power plants responsible for necessary training. The thematic analysis results showed that some of the events should have been prevented with necessary training. In the reports as corrective actions training requirements were listed both in LERs and IRs. These results aligned with previous research that explained training issues in the US nuclear plants. (Patrick, James, Ahmed, & Halliday, 2006).

2. *Creating sufficient procedures and guidelines:* During outage control management, nuclear power plants increase the number of workers to complete refueling. Some of the workers are hired with a temporary contract to provide the outage control. These contractors work together with facility employees. It is very important to provide sufficient guidelines and procedures for the temporary workers and facility employees who need to collaborate effectively. It also supports the Bourrier (1996), which highlighted the importance of the detailed procedures, proper execution, supervision and work quality in the nuclear plants.
3. *Enhancing safety culture:* The expert interviews and official reports showed that organizational planning, arranging necessary resources and materials, enhancing following safety regulations, work control and supervision are a crucial part of outage control management. This findings support other research which reveals that organizational issues are related to planning and resourcing (Kecklund, & Svenson, 1997), and supervisory and management (Carroll, Hatakena, & Rudolph, 2006).

2. ***Research question:*** *What initiating events in the official reports are related to human, team and organizational errors during the outage control management in the NPPs?*

The research findings revealed three main initiating factors based on thematic and quantitative analysis, which include effect of operation modes, number of reactors that plant has, and operation experience.

Operation Mode

The LERs and IRs were analyzed thoroughly to examine individual and team errors. Initial analysis compared the operation modes and reported human errors. Most team errors were reported during Mode-5 Cold Shutdown and Mode-6 Refueling, which are also the modes of longest duration. In nuclear plants, the number of operational hours increases with the complexity of task. For example, during Mode 6-Refueling the teams in the field place nuclear rods into the reactor. Due to radioactivity safety rules, they have limited time to complete it. On other hand, Mode 2 Start Up has a shorter time to complete. The Refueling phase requires greater workload than Startup phase, which includes nuclear waste removal and placing new nuclear rods. It also creates a greater possibility for team errors. These findings support previous research of Juhasz, and Soos (2007) and Jackson and Svensson, (1991). Juhasz, and Soos (2007), concluded that higher workload decreases the frequency of communication between the team members in nuclear power plants and increases team errors. During the high task load the operator's attention is more focused, but also narrowed (Juhasz, & Soos, 2007) and errors are more likely to occur (Kecklund, & Svenson, 1997). The findings of this research aligned with the previous results. Still, deeper investigation should be conducted to understand how operation modes affect team interaction and human errors.

Number of reactors in plants

Additionally, the number of the nuclear reactors and number of errors were compared. The total number of the reactors provides information about the labor capacity of the plants and their management. The analysis results showed that there is a significant relationship between the reactor numbers and number of LERs that submitted. Smaller nuclear power plants seem to have higher number of incidents and reported human errors than those with three or four reactors. This finding was interesting and unexpected because having more nuclear reactors requires more outage control periods to plan and execute for a plant. It can be thought that it is more likely to observe higher human errors or report LERs and IRs for the bigger plants. During the interviews, this finding was shared with two experts. Both experts shared their opinions about the results. One of the experts said that the number of the incident reports shows the strength of the safety culture in an organization. The proactive plants encourage workers to report any incident, not to blame them, but to improve any existing pitfalls. They encourage documentation of the incidents, so it would be prevented in the future. The other expert said he does not think that there is a linear relationship between the nuclear reactor number and the reported incidents. He also shared that the plants with more reactors (2-reactors and 3-reactors) have more facility employees who work full time in the facility that are familiar with the working environment. However, the smaller plants have only limited employees and hire non-facility contractor workers during the outage control period. This situation may create some communication, coordination and interaction problems in team level. It might be the reason that more incidents in smaller nuclear power plants were observed.

This result could be an indication of strong teams of larger plants, which have to face more problems and also gain more experience in the plants. Planning and managing multiple outage control each year for bigger plants can be a challenge and strength for the teams. The teams gain valuable experience during each reactor refueling process and they can transfer this experience to future outage controls. In smaller plants, the outage period is fewer, and their teams execute fewer outage control. There was no published research to support these findings. A deeper analysis is required to investigate other underlying factors.

Operation Experience

Figure 7, LER analysis of Palo Verde, showed that most of the individual human errors and team errors were reported during the first five-years of the operation. The numbers showed a noticeable change after 2000. Similarly, at the beginning of 2000, the reported incidents in the IRs decreased. During the expert interviews, two participants explained this situation. Both participants emphasized that the teams in NPPs learned the challenges and deficiencies of the system over the years and adapted their solution effectively. Currently the teams are able to handle serious issues such as design problems. The Stachowski, Kaplan and Waller (2009) study showed that the teams who are more adaptive, performance better in nuclear power plants than those who are less adaptive. In this research, the yearly analysis showed that the teams gained the adaptive skills of the challenging working environment over the years. Fewer incidents are reported as experience increased.

Interactive Team Cognition and Team Errors

Interactive Team Cognition (ITC) is influenced by the ecological psychology approach (Cooke, German, & Rowe, 2004) and emphasizes the importance of team coordination, communication and interaction. According to ITC, the focus of the cognitive activity in a team is between individual and their environment (Cooke et al., 2004; Gorman, 2014). The results show that 41.7% of the incidents occurred due to team errors, whereas 58.3% of the errors occurred because of the procedural errors, organizational errors and design errors all together. During the expert interviews, one of the participants shared that there should be a distinction between the teams in the field and control room. He stated that team errors in the field are more likely to occur because there are many temporary workers. An outage control room is mostly full time facility employees who have experience to work as teams in the plants. This suggestion aligns with the ecological approach of the ITC theory. The weakness and strength of the teams cannot be understood and analyzed independently from the environment in which they work. The outage management teams in the field and outage control team in the control room have different dynamics compared to regular operations.

ITC stresses the importance of team interaction, communication and coordination for an effective teamwork. The data analysis revealed that teams could have issues analyzing the problem, deciding on to the best solution and completing a high quality work during the outage control. The outage control is one of the hardest tasks to complete for any nuclear power plant. Having time limitations, working in a high-risk environment and having financial pressures creates serious challenges for the teams. The information in the field and control room is very dynamic. All these factors especially increase the importance of team coordination, communication and coordination.

The ITC theory emphasizes the importance of training aimed to improve the methods of team interaction and activities (Cooke et al., 2012). The Coordinated Awareness of Situation by Teams (CAST) was proposed in accord with ITC theory to measure team situation awareness (Gorman, Cooke, & Winner, 2006). CAST stresses coordinated perception and coordinated actions of team members. In addition, the ITC theory states that for large teams, such as nuclear power plant teams, team interaction is more important than creating a shared understanding.

Working in a complex environment, requires robust training to coordinate and communicate effectively as team. It is also important to have a clear understanding of interaction at the team level, as well as between other teams. These expectations can be set by effective training strategies, which are designed to address limitations and deficiencies of teams (Gorman, Cooke & Amazeen,2010). ITC suggests that team training should be designed to address decision-making, task analysis, effective communication and coordination to create adaptive teams (Gorman, Cooke, & Amazeen, 2010). The reported team errors in nuclear plants happened mostly because of failures recognizing situations as a team, making decisions as a team, reviewing decisions as team, and not appropriately incorporating risk as a team. These findings indicate that some of the incidents occurred because of weakness of team interactions. These problems might be address through ITC-based team training.

ITC theory investigated different training methods to find the most effective way to create adaptive teams. Perturbation Training is a new approach, which was formed based on team needs of adaptive teams. It is designed to disrupt the standard communication patterns to force teams to coordinate in a novel way (Gorman, Cooke, & Amazeen, 2010). This kind of training is aimed to challenge usual routine communication and procedural rigidity. Similarly, McNeese, Demir, Cooke and Myers (2018) investigated team coordination in human-autonomy teaming. The research results revealed that implicitly pulling and pushing information in training accelerates team coordination (McNeese, Demir, Cooke, & Myers 2018; Hinski, 2017). Both approaches in designing team training can address some of the reported team errors in the official reports.

Resilient Systems and Teams

Resilience of a system is described as the ability to learn, ability to monitor, ability to anticipate, and ability to respond (Hollnagel, Woods, & Leveson, 2006). In addition to these, Kamanja and Joghyun (2014) included "Collective functioning" which is focused on the crew performance and teamwork. Adaptive skills of teams are defined as one of the important features of the resilient systems. It shows that experienced teams are one of the most valuable assets of nuclear power plant to create resilient system. During the interviews, one of the experts defined the teams as crutches of the nuclear industry, in that they have to be flexible and bend over backwards to handle design challenges. Similarly, the second expert also shared a very similar opinion and stressed how highly educated, and trained professionals are working in the system. He said that these highly trained team members work under time and financial pressure effectively and still keep safety as a priority. These teams work in the system for a long time and resolved different issues as they face them. These results support that the teams in the industry improved over the years.

Having highly trained teams is important as is creating a safe and productive working environment for the teams. Improving initiating factors such as organizational issues, procedural problem or design issues can considerably impact teamwork in the plant. In order to create strong robust resilient systems in nuclear power plants, it is important to optimize the working environment.

Implications

These research findings can be useful in different areas. First, the research findings may provide critical information about human and team incidents in NPPs. The data can be impactful to identify and to improve some of the problematic areas. It can be used to enhance resilience in the NPPs. These findings revealed that teams could face some unprecedented problems during the outage control period. It is still important to work as a team, make a decision as team, and execute the best solution as a team, which should be practiced during trainings. Perturbation training and coordination coaching, which are grounded in ITC principles, can help to address team-training issues in nuclear power plants. This model of training is not focused on procedural actions as team, it is designed to challenge standard communication and coordination to help teams to create novel solutions.

The analysis of these reports provides a broad example of possible challenges and issues. One of the important tools of the resilient systems is to learn from the mistake and anticipate future problems. The nuclear industry can use these data to create well-designed procedures, highly trained teams and proactive management. In order to extract the most accurate information, the initial review should be completed manually, and some keywords and phrases should be identified. This research can be used as an initial demonstration of this method. At the end of this research, critical keywords and phrases related to human and team errors in the NPPs were derived that these findings can be used in later data mining efforts.

Future Studies

The research methods used here can be extended in a number of ways. Artificial intelligence researches try to find the most efficient and effective way of document analysis. Human-in-the-loop machine learning is a combination of manual human analysis and artificial intelligent data mining process. For a human in the loop machine-learning project, initial data analysis should be completed by a human; such as categorization, classifications, and included/excluded keywords. Based on t the manual human analysis process, deeper machine learning codes can be created to imitate a similar analysis. This research can be a starting point for other researchers who want to use the LERs and IRs to extract insight information without being in the facilities.

Limitations

There are some significant limitations in this research. First, the LERs and IRs are prepared to report unexpected incidents and events to the Nuclear Regulatory Commission (NRC). These reports are not created to be used in a research. They rely on the judgments and opinions of the reporters based on the NRC guideline. Therefore, in this research the official reports are being used outside of their primary purpose. This is one of the limitations of this study. Second, the official reports provide mostly technical details; there is insufficient information about the details of human and team performance issues. It is known that NPPs have a separate human performance evaluation reports, but these reports are confidential. Third, different reporters or inspectors prepare the LERs and the IRs. The employees of nuclear power plants write the LERs. However, the IRs are written by the inspectors of the NRC. This situation creates inconsistency in the reporting terminology and taxonomy.

In this study more than 800 LERs were analyzed. As a regulation rule, the LERs have a different reporting requirement and the employees of the power plant submit these reports. This concern was also shared by one of the experts during the interview. The reports might be affected based on who created them. If an NPP employee writes a report, it is less likely to point a finger at who really is responsible for the incident. Because of that, it is more likely that the report will not include necessary details. Similarly, more than 40 IRs were analyzed in the study. Even though there was clear human factors guideline to report human errors, these reports were mostly focused on the licensees as an organization. Therefore, it is more likely to get a report related to organizational issues than other factors. Fourth, three subject matter experts were willing to participate in the interviews.

All participants have significant experience as researchers in human factors and human reliability. But there was no nuclear power plant employee who was willing to join. The nuclear industry is one of the industries in which it is really hard to get information from the plants. This factor may affect the interview results and the number of participants. Last, the error categorizations were created based on initial analysis in which the focus was on human errors. A different researcher with different research questions could categorize the same incidents differently. For example, the incidents can be grouped into two groups; dormant and active error. Even though the aim is to create and use an objective coding procedure, it can be argued that the categorization is still a subjective element.

CHAPTER 6: Conclusions

A resilient system can handle all expected and unexpected challenges without sacrificing safety. In NPPs, transition from traditional safety approaches to a resilient systems approach can be achieved by improving multiple factors such as design, technology, organizational management and proactive teamwork.

NPP operations are very complex. In particular, outage control is one of the most critical of operations with a high risk of radiation and core damage. In order to complete tasks, multiple-disciplinary teams need to work together interdependently. Time constraints, financial pressure, and complexity of tasks significantly influence the teams' performance. Even though these factors make the outage control period more vulnerable than normal operations, most previous research has neglected the outage control period. There seems to be no empirical study on team cognition that has solely investigated the outage period team dynamics. The LERs and IRs database provides a rich data set for the analysis of unanticipated events in NPP operations, which includes regular operation period and outage period. Therefore, this research was designed to examine outage control teamwork with the goal to improve safety, productivity, and resilience through improvements in team cognition. The results showed that some of team errors not only occurred because of weaknesses of the teams, but they occurred because of the procedural errors, organizational errors, and design issues. Improvement of these areas will help to improve the team performance as well as system resilience of the nuclear plants. Still more research should be conducted to understand these factors.

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APPENDIX A

SAMPLE LISENCEE EVENT REPORT (LER)

AmerGen
An Exelon Company

AmerGen Energy Company
Oyster Creek
US Route 9 South, P.O. Box 388
Forked River, NJ 08731-0388

www.exeloncorp.com

10 CFR 50.73

July 5, 2006
2130-06-20357

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555 - 0001

Oyster Creek Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219

Subject: Licensee Event Report 2006-001-00, Manual Scram Inserted During Planned Reactor Shutdown to Expedite Plant Cool-down

Enclosed is Licensee Event Report 2006-001-00, Manual Scram Inserted During Planned Reactor Shutdown to Expedite Plant Cool-down. This event did not affect the health and safety of the public or plant personnel. This event did not result in a safety system functional failure. There are no new regulatory commitments made in this LER submittal.

If any further information or assistance is needed, please contact Rich Milos, Regulatory Assurance at 609-971-4973 or Jeff Dostal, Operations, at 609-971-4572.

Sincerely,

James J Randich

James Randich
Plant Manager, Oyster Creek Generating Station

Enclosure: NRC Form 366, LER 2006-001-00

cc: Administrator, USNRC Region I
USNRC Project Manager, Oyster Creek
USNRC Senior Resident Inspector, Oyster Creek
File No. 06035

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Oyster Creek, Unit 1	05000 219	1 OF 3

4. TITLE

Unplanned Manual Scram Inserted During Planned Reactor Shutdown to Expedite Plant Cooldown

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
05	06	2006	2006	- 001 -	00	07	05	2006	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: N

10. POWER LEVEL: <1%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

[ ] 20.2201(b)	[ ] 20.2203(a)(3)(i)	[ ] 50.73(a)(2)(i)(C)	[ ] 50.73(a)(2)(vii)
[ ] 20.2201(d)	[ ] 20.2203(a)(3)(ii)	[ ] 50.73(a)(2)(ii)(A)	[ ] 50.73(a)(2)(viii)(A)
[ ] 20.2203(a)(1)	[ ] 20.2203(a)(4)	[ ] 50.73(a)(2)(ii)(B)	[ ] 50.73(a)(2)(viii)(B)
[ ] 20.2203(a)(2)(i)	[ ] 50.36(c)(1)(i)(A)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(ix)(A)
[ ] 20.2203(a)(2)(ii)	[ ] 50.36(c)(1)(ii)(A)	[X] 50.73(a)(2)(iv)(A)	[ ] 50.73(a)(2)(x)
[ ] 20.2203(a)(2)(iii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(v)(A)	[ ] 73.71(a)(4)
[ ] 20.2203(a)(2)(iv)	[ ] 50.46(a)(3)(ii)	[ ] 50.73(a)(2)(v)(B)	[ ] 73.71(a)(5)
[ ] 20.2203(a)(2)(v)	[ ] 50.73(a)(2)(i)(A)	[ ] 50.73(a)(2)(v)(C)	[ ] OTHER
[ ] 20.2203(a)(2)(vi)	[ ] 50.73(a)(2)(i)(B)	[ ] 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Frank Meyer, Operations Support	609-971-4827

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	SE	CDU	F175	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On May 6, 2006, Oyster Creek was shutting down to start a Forced Outage (1F10) to repair a leak from the Steam Packing Exhauster Cooling Condenser (EIIS: SE). This leak was adding excess volume to the radiological liquid waste processing systems (Radwaste). During the shutdown, it was decided to perform a manual reactor scram to improve plant cool down and thereby minimize the volume of component leakage being sent to Radwaste (EIIS: WD).

This manual SCRAM was not a scheduled item during the pre-outage planning. The decision to insert the manual scram was made during the shutdown process. It was not performed to avoid reaching an automatic scram setpoint.

The 1F10 reactor shutdown was planned to be accomplished using full manual control rod insertion and manual scram was not a planned activity. The manual scram was performed after reducing reactor power level to the Source Range and the reactor was subcritical at the time.

All Control Rods fully inserted as a result of the manual scram, and all systems performed as designed.

NRC FORM 366 (6-2004)

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION
(1-2001)

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Oyster Creek, Unit 1	05000219	2006	- 001 -	00	2 OF 3

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

Description of Event

On May 6, 2006, Oyster Creek was shutting down to start a Forced Outage (1F10) to repair a leak from the Steam Packing Exhauster Cooling Condenser (EIIS: SE). This leak was adding excess volume to the radiological liquid waste processing systems (Radwaste). During the shutdown, it was decided to perform a manual reactor scram to improve plant cool down and thereby minimize the volume of component leakage being sent to Radwaste (EIIS: WD).

This manual SCRAM was not a scheduled item during the pre-outage planning. The decision to insert the manual scram was made during the shutdown process. It was not performed to avoid reaching an automatic scram setpoint.

The 1F10 reactor shutdown was planned to be accomplished using full manual control rod insertion and manual scram was not a planned activity. The manual scram was performed after reducing reactor power level to the Source Range and the reactor was subcritical at the time.

All Control Rods fully inserted as a result of the manual scram, and all systems performed as designed.

Analysis of Event:

No Engineered Safety Features (ESF) /Emergency Core Cooling Systems (ECCS) actuations accompanied the manual scram. Therefore the consequences of this event were minimal. It was a valid Reactor Protection System (RPS) (EIIS: JC) actuation. The manual scram is considered event driven since the excessive rate of water volume being transferred to the Radwaste Facility was projected to exceed the liquid processing system's capability prior to reaching cold shutdown and securing the Condensate system. The 1F10 shutdown was planned to be accomplished using full manual control rod insertion – no manual scram was planned.

Since the reactor was already sub-critical at the time of the manual scram, an 8-hour report was required and made on May 6, 2006. Actuation of the Reactor Protection System is reportable under 10 CFR 50.73(a)(2)(iv)(A).

Cause of Event:

The failure of the Steam Packing Exhauster (SPE) Cooling Condenser resulted in a high rate of leakage that exceeded the capability of Radwaste to process and return the sump and drain tank effluents for continued operation. The potential need to manually scram the reactor to improve plant cool down was recognized, but not documented as a planned activity during pre-outage planning. Operations made the decision to improve plant cool down, and isolation of the SPE Cooling Condenser leak, by inserting a manual scram of the reactor to avoid over burdening the Radwaste Liquid processing system.

NRC FORM 366 (6-2004)

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION
(1-2001)

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Oyster Creek, Unit 1	05000219	2006	- 001 -	00	3 OF 3

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

Corrective Actions:

(1) Repaired the leaking tubes in the SPE Cooling Condenser.

(2) Performed a Root Cause Evaluation to prevent recurrence.

Additional Information

A. Failed Components:

The Steam Packing Exhauster Cooling Condenser failed and was the cause for entering the Forced Outage.

B. Previous similar events:

None

C. Identification of components referred to in this Licensee Event Report:

Components	IEEE 805 System ID	IEEE 803A Function
Steam Packing Exhauster Cooling Condenser	EIIS: SE	EIIC: CDU
Radiological Liquid Waste Processing Systems (Radwaste).	EIIS: WD	EIIC: PFR
Reactor Protection System	EIIS: JC	EIIC: XC-RCT

NRC FORM 366 (6-2004)

APPENDIX B

PALO VERDE NUCLEAR POWER PLANTS LISENCEEE EVENT REPORT (LER)

1985-2015

LER Number	Date	Reactor	Mode	Root Cause	Teamwork Error
5281987026	10/27/1987	Unit 1	6-Refueling	Cognitive Personnel Error	
5282010003	2-May-10	Unit 1	6-Refueling	Technical Specification Violation	TEAM- lack of verification
5281989009	8/16/1989	Unit 1	6-Refueling	Cognitive personnel error by the QC inspector (contractor, non-licensed).	TEAM- communication
5281989014	6/1/1989	Unit 1,2 and 3	6-Refueling	Personnel Error	TEAM - review and follow up
5281989013	7/26/1989	Unit 1,2 and 3	6-Refueling	Cognitive personnel	
5281989025	11/3/1989	Unit 1	6-Refueling	Inadequate Instructions	TEAM- coordination and communication
5281989022	11/20/1989	Unit 1	6-Refueling	Cognitive Personnel Error	TEAM- communication
5281989020	9/7/1989	Unit 1	6-Refueling	Cognitive Personnel Error	TEAM-multi team member failed to perform a review (communication/coordination)
5281989018	10/26/1989	Unit 1, 2 and 3	6-Refueling	Personnel Error	TEAM- failed to review, prepare and approval of qualification
5291985006	12/31/1985	Unit 2	6-Refueling	Personnel Error	
5291990003	3/31/1990	Unit 2	6-Refueling	Personnel Error	TEAM- coordination- inadequate identification of loads
5291990005	4/17/1990	Unit 2	6-Refueling	Cognitive Personnel Error	Team-Coordination
5291991006	10/27/1991	Unit 2	6-Refueling	Combination of cognitive personnel errors	TEAM- communication
5291991009	12/31/1991	Unit 2	6-Refueling	Cognitive Personnel Error	TEAM-operator and inspector failed to complete the task
5292011001	4/8/2011	Unit 2	6-Refueling	Personnel Error-control room	TEAM - communication and coordination

5301989005	6/28/1989	Unit 3	6-Refueling	Combination of cognitive personnel errors	TEAM-team follow –up (communication, coordination)
5301989006	6/6/1989	Unit 3	6-Refueling	Cognitive Personnel Error	TEAM- coordination
5301989008	6/30/1989	Unit 3	6-Refueling	Personnel error (contractor, non-licensed)	
5301992004	10/9/1992	Unit 3	6-Refueling	Cognitive Personnel Error	
5301994003	5/6/1994	Unit 3	6-Refueling	Personnel Error	TEAM- coordination and communication
5292003003	11/23/2003	Unit 2	6-Refueling	Personnel Error	Team- no confirmation
5291996002	4/1/1996	Unit 2	6-Refueling	Personnel Error	
5292011001	4/8/2011	Unit 1	6-Refueling		TEAM- coordination and communication
5301989009	7/28/1989	Unit 3	6-Refueling	Cognitive Personnel Error	
5301988004	10/12/1998	Unit 3	6-Refueling	Personnel Error	TEAM- coordination and communication
5302010001	10/7/2010	Unit 3	6-Refueling	Personnel Error	TEAM - review
5281989022	11/20/1989	Unit 1	6-Refueling	Inadequate administrative program	TEAM administrative
5281985006	1/31/1985	Unit 1	5-Cold shut-down	Human error	
5281985009	3/21/1985	Unit 1	5-Cold shut-down	Personnel error associated with a procedural deficiency	
5281985013	2/23/1985	Unit 1	5-Cold shut-down	Human error-procedural inadequacy	
5281985014	3/2/1985	Unit 1	5-Cold shut-down	Human error-procedural inadequacy	
5281985017	4/5/1985	Unit 1	5-Cold shut-down	Cognitive Personnel Error	
5281985023	4/6/1985	Unit 1	5-Cold shut-down	Cognitive Personnel Error	TEAM
5281985025	3/14/1985	Unit 1	5-Cold shut-down	Cognitive Personnel Error	

5281985077	11/21/1985	Unit 1	5-Cold shut-down	Cognitive personnel error	
5281985084	11/10/1985	Unit 1	5-Cold shut-down	Cognitive Personnel Error	TEAM
5281985091	11/22/1985	Unit 1	5-Cold shut-down	Cognitive Personnel Error /misinterpretation	TEAM- documented but not completed (coordination, communication)
5281986019	03/12/198	Unit 1	5-Cold shut-down	Personnel error due to an inadequate procedure	
5281986032	5/15/1986	Unit 1	5-Cold shut-down	Procedural deficiency	
5281988019	7/22/1988	Unit 1	5-Cold shut-down	The physical layout of the work location contributing to a personnel error	
5281989011	4/25/1989	Unit 1	5-Cold shut-down	Cognitive personnel error that was contrary to an approved procedure	
5281989012	5/10/1989	Unit 1,2 and 3	5-Cold shut-down	Human error-procedural inadequacy	TEAM- in adequate cross-disciplinary reviews
5281989015	9/1/1989	Unit 1,2 and 3	5-Cold shut-down	Lead failure	
5281989017	10/23/1989	Unit 1,2 and 3	5-Cold shut-down	Cognitive Personnel Error	TEAM-two engineers failed to identify
5281990003	3/21/1990	Unit 1	5-Cold shut-down	Personnel error in that Technical Specifications were misinterpreted by the unit management	TEAM
5291986004	1/24/1986	Unit 2	5-Cold shut-down	Personnel error	TEAM- lack of verification
5291986007	2/27/1986	Unit 2	5-Cold shut-down	Inadequate procedure to notify control room personnel	TEAM- inadequate procedure created communication problem
5291986045	7/15/1986	Unit 2	5-Cold shut-down	Cognitive personnel error	
5291987001	2/8/1987	Unit 2	5-Cold shut-down	Cognitive personnel error	

5291997007	10/6/1997	Unit 2	5-Cold shut-down	Ineffectively addressing past industry operating experience	TEAM -coordination (the same problem occurred in the past, but not fixed)
5291988005	2/21/1988	Unit 2	5-Cold shut-down	Cognitive personnel error on the part of utility, licensed personnel.	TEAM- lack of proper responding
5292008001	5/21/2008	Unit 2	5-Cold shut-down	Personnel error	
5282010002	5/7/2010	Unit 1, 2, 3	5-Cold shut-down	Personnel error-ineffective reviews	TEAM-ineffective review
5291989009	5/9/1989	Unit 2	5-Cold shut-down	Personnel error on the part of the Instrument and Control Technician	TEAM-lack of review
5291991003	9/27/1994	Unit 2	5-Cold shut-down	Personnel error	TEAM-night shift-communication and coordination
5291997003	9/7/1997	Unit 2	5-Cold shut-down	Procedural deficiency	
5292012002	10/7/2012	Unit 2	5-Cold shut-down	Inadequate guidance	TEAM- communication
5302013001	10/7/2013	Unit 3	5-Cold shut-down	Cognitive personnel error.	
5302006004	5/6/2006	Unit 3	5-Cold shut-down	Insufficient procedure	
5302006003	4/2/2006	Unit 3	5-Cold shut-down	Human error	TEAM- communication and coordination
5301989013	9/26/1989	Unit 3	5-Cold shut-down	Cognitive personnel error	
5292008001	5/21/2008	Unit 2	5-Cold shut-down	Cognitive personnel error	
5291997007	10/6/1997	Unit 2	5-Cold shut-down	Human error	

5291997003	9/7/1997	Unit 2	5-Cold shut-down	Personnel error	
5281988001	1/4/1988	Unit 1	5-Cold shut-down	Cognitive Personnel Error	TEAM - coordination and communication
5291990006	6/18/1990	Unit 2	5-Cold shut-down	Cognitive personnel error	TEAM - coordination and communication
5291988011	5/22/1988	Unit 2	5-Cold shut-down	Multiple human error	TEAM - coordination and communication
5281987028	12/17/1987	Unit 1	5-Cold shut-down	Cognitive Personnel Error	
5281988005	01/23/1988	Unit 1	5-Cold shut-down	Cognitive Personnel Error	TEAM + inadequate procedure
5301994	12/9/1994	Unit 3	5-Cold shut-down	Cognitive Personnel Error	TEAM
5281985050	8/9/1985	Unit 1	4- Hot shutdown	Administrative controls were inadequate	TEAM
5281985061	8/3/1985	Unit 1	4- Hot shutdown	Personnel error	
5291986010	4/15/1986	Unit 2	4- Hot shutdown	Cognitive personnel (utility) errors.	TEAM - coordination and communication
5292005006	5/15/2005	Unit 2	4- Hot shutdown	Technical-and work management	TEAM -Technical-and work management
5302015002	5/1/2015	Unit 3	4- Hot shutdown	Human error	
5302015004	5/1/2015	Unit 3	4- Hot shutdown	lack of adequate guidance	TEAM - multi level
5301989011	12/6/1989	Unit 3	4- Hot shutdown	Personnel error	TEAM -insufficient controls
5291986024	5/14/1986	Unit 2	4- Hot shutdown	Personnel error	TEAM
5291986014	3/25/1986	Unit 2	4- Hot shutdown	Personnel error	TEAM
5291986013	3/24/1986	Unit 2	4- Hot	Cognitive personnel error	

			shutdown		
5282006001	3/20/2006	Unit 1	4- Hot shutdown	Human error in that operational fundamentals	TEAM -operational management
5281988004	2/29/1988	Unit 1	4- Hot shutdown	Procedural inadequacy	
5281987027	7/28/1987	Unit 1	4- Hot shutdown	Personnel error	
5281987017	6/30/1987	Unit 1	4- Hot shutdown	Cognitive personnel error	
5281987007	1/20/1987	Unit 1	4- Hot shutdown	Personnel error	Team
5281986025	4/16/1986	Unit 1	4- Hot shutdown	Cognitive personnel error	
5281986016	1/27/1986	Unit 1	4- Hot shutdown	Personnel error	
5281985060	8/9/1985	Unit 1	3- Hot Standby	Human error-procedural inadequacy	TEAM-inadequate administrative controls
5281985061	8/3/1985	Unit 1	3- Hot Standby	Cognitive personnel error	
5281985064	8/26/1985	Unit 1	3- Hot Standby	Personnel error	
5281985069	8/19/1985	Unit 1	3- Hot Standby	Cognitive personnel error	TEAM-inadequate administrative controls
5281985075	12/13/1985	Unit 1	3- Hot Standby	Personnel error	
5281985084	11/10/1985	Unit 1	3- Hot Standby	Cognitive personnel error	TEAM-Communication and coordination
5281986035	5/13/1986	Unit 1	3- Hot Standby	Cognitive personnel error	
5281989017	10/23/1989	Unit 1,2 and 3	3- Hot Standby	Cognitive personnel error.	TEAM-the failure by 2 engineering personnel
5281991011	10/29/1991	Unit 1,2 and 3	3- Hot Standby	Personnel error	
5281986037	6/26/1986	Unit 2	3- Hot Standby	Cognitive personnel error by a utility- licensed operator.	

5281986038	12/27/1986	Unit 2	3- Hot Standby	Cognitive personnel error	
5281990008	7/4/1990	Unit 2	3- Hot Standby	Cognitive personnel error	TEAM -OCC
5281996003	1/22/1996	Unit 2	3- Hot Standby	Procedural error	
5281990007	6/17/1990	Unit 1	3- Hot Standby	Cognitive personnel error	TEAM -OCC
5281990009	10/27/1996	Unit 1	3- Hot Standby	Personnel error	
5282005002	2/17/2005	Unit 1	3- Hot Standby	Cognitive personnel error	TEAM
5282006002	3/21/2006	Unit 1	3- Hot Standby	Human error	
5282003004	12/8/2003	Unit 2	3- Hot Standby	Human performance errors made by Engineering and Operations personnel	TEAM -Engineering and Operations personnel
5282011002	5/2/2011	Unit 2	3- Hot Standby	Procedural inadequacies	
5282012003	11/2/2012	Unit 2	3- Hot Standby	Inadequate work instructions	
5301991007	8/31/1991	Unit 3	3- Hot Standby	Cognitive personnel error	
5301991009	11/18/1991	Unit 3	3- Hot Standby	Cognitive personnel error	
5301991010	11/15/1991	Unit 3	3- Hot Standby	Personnel error.	
5302001002	10/29/2001	Unit 3	3- Hot Standby	Inadequate change management	TEAM - management
5281985038	5/30/1985	Unit 1	2- Start up	Human error	
5281986021	2/6/1986	Unit 1	2- Start up	Cognitive personnel error	
5282005005	8/26/2005	Unit 1	2- Start up	Individual and crew failures to implement expected requirements	TEAM -individual and crew failures to implement expected requirements
5282011005	11/22/2011	Unit 1	2- Start up	Latent organizational weaknesses with the modification and corrective action processes	TEAM- organizational weakness
5291986035	6/11/1986	Unit 2	2- Start up	Cognitive error by the Shift Supervisor (licensed-utility)	
5291987002	3/20/1987	Unit 2	2- Start up	Cognitive personnel error in that the Shift Supervisor, Day Shift Supervisor and the Shift Technical Advisor	TEAM - (Communication and coordination)
5291988012	6/15/1988	Unit 2	2- Start up	Cognitive personnel error	TEAM - (Communication and coordination)

APPENDIX C

SAMPLE INPECTION REPORT (IR)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

July 26, 2002

R. T. Ridenoure
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, Nebraska 68023-0550

SUBJECT: NRC INSPECTION REPORT 50-285/02-02

Dear Mr. Ridenoure:

On June 29, 2002, the NRC completed an inspection at your Fort Calhoun Station. The enclosed report documents the inspection findings which were discussed on July 2, 2002, with you and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified a finding that involved nonconservative processes that could affect the availability of mitigating systems during plant heatup conditions. This issue was evaluated under the risk significance determination process as having very low safety significance (Green).

The NRC has increased security requirements at the Fort Calhoun Station in response to terrorist acts on September 11, 2001. Although the NRC is not aware of any specific threat against nuclear facilities, the NRC issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The NRC continues to monitor overall security controls and will issue temporary instructions in the near future to verify by inspection the licensee's compliance with the Order and current security regulations.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/ by Jeffrey A. Clark acting for

Claude E. Johnson, Chief
Project Branch C
Division of Reactor Projects

Docket: 50-285
License: DPR-40

Enclosure:
NRC Inspection Report
50-285/02-02

cc w/enclosure:
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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-285
License:	DPR-40
Report:	50-285/02-02
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	Fort Calhoun Station FC-2-4 Adm., P.O. Box 399, Hwy. 75 - North of Fort Calhoun Fort Calhoun, Nebraska
Dates:	March 31 through June 29, 2002
Inspectors:	J. Kramer, Senior Resident Inspector L. Willoughby, Resident Inspector W. Walker, Senior Project Engineer B. Baca, Health Physicist, L. Ellershaw, Senior Reactor Inspector
Accompanying Personnel:	I. Barnes, Consultant
Approved By:	Claude E. Johnson, Chief, Project Branch C

SUMMARY OF FINDINGS

Fort Calhoun Station
NRC Inspection Report 50-285/02-02

IR 05000285-02-02; 03/31-06/29/2002; Omaha Public Power District, Fort Calhoun Station, Integrated Resident & Regional Report; Refueling and Outage Activities

The inspection was conducted by resident and regional inspectors. This inspection identified one Green finding. The significance of the issues is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process."

Cornerstone: Mitigating Systems

- Green. The licensee exercised a nonconservative decision making process when controlling foreign materials in containment and eliminating the potential for blocking the emergency core cooling system suction strainers. As a result, approximately 20 55-gallon drums with paper taped onto the lid on the same elevation as the sumps and several hundred other pieces of tape remained in containment during initial plant heatup following a refueling outage.

  This finding was of very low safety significance because the containment emergency sumps remained available (Section 20).

Report Details

Summary of Plant Status

The unit began the inspection period at approximately 98 percent power. On May 4, 2002, the unit was shutdown for Refueling Outage 20 and entered cold shutdown. Upon completion of the outage activities, the unit entered hot shutdown on May 30. On June 1, the reactor was made critical and the unit was placed on-line the following day. On June 7, the unit achieved approximately 100 power and remained at the power level for the remainder of the inspection period.

1. **REACTOR SAFETY**
   Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed partial equipment walkdowns. The inspectors checked portions of the equipment to identify any discrepancies between the existing and proper alignment as determined by system piping and instrumentation drawings or plant procedures:

- Shutdown Cooling Purification System using Procedure OI-SC-5, "Shutdown Cooling Purification," Revision 16, on May 8, 2002.
- Emergency Diesel Generator 2 locally at the engine and switch alignment in the control room on June 12, 2002.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors reviewed selected plant areas to determine if the licensee appropriately implemented the fire protection program. Specifically, the licensee adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capabilities, and maintained passive fire protection features in good material condition. The inspectors checked the following areas:

- Diesel Generator 2 room on June 12, 2002
- Electrical switch gear room on June 17, 2002
- East battery room on June 17, 2002
- Air compressor and auxiliary feedwater pumps room on June 27, 2002

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Inspection Activities Other than Steam Generator Tube Inspections

Performance of Nondestructive Examination (NDE) Activities

The Fort Calhoun Station inservice inspection program is committed to the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1989 Edition, no Addenda for the third 10-year interval. The third 10-year interval will end following the next refueling outage, currently scheduled for the fall of 2003.

a. Inspection Scope

The inspectors observed portions of the following inservice inspection examinations:

System	Component/Weld Identification	Examination Method
Safety Injection	Elbow-to-Pipe, 4-CH-12, Weld 08	Ultrasonic Examination
Component Cooling Water	Pipe-to-Pipe, 10-AC-2004, Weld 25	Ultrasonic Examination
Component Cooling Water	Elbow-to-Pipe, 10-AC-2003, Weld 23	Ultrasonic Examination
Main Steam	Pipe-to-Elbow, 28-MS, Weld 04	Ultrasonic Examination

During the performance of each examination, the inspectors verified that the correct NDE procedure was used, procedural requirements or conditions were as specified in the procedure, and test instrumentation and equipment were properly calibrated. The inspectors reviewed the NDE certification packages of the observed contractor personnel and verified that they had been properly certified in accordance with ASME Code requirements. The inspectors also verified that indications revealed by the examinations were compared against the ASME Code-specified acceptance standards and appropriately dispositioned.

The inspectors reviewed the licensee's NDE records for certain examinations that were performed during the current outage to verify that either required or committed NDE activities were performed in accordance with ASME Code requirements, and indications and defects, if present, were appropriately dispositioned. These included ultrasonic examinations of the weld overlays in reactor vessel head Heated Junction Thermocouple Housings 7 and 11 and liquid penetrant examinations of the three butt welds in peripheral Control Element Drive Mechanism Housing 37.

In addition, the inspectors also observed eddy current examinations of the "J" welds in four of the eight control element drive mechanism upper seal housings that the licensee committed to perform.

The inspectors determined, with respect to reactor vessel head control element drive mechanism nozzles, that the licensee had completed the ASME Code required number of examinations specified for the third 10-year interval.

Finally, the inspectors determined that the licensee, as allowed by the ASME Code, deferred the volumetric examinations required for Category BA full penetration reactor vessel nozzle welds until the next refueling outage.

b. Findings

No findings of significance were identified.

.2 ASME Code Repair and Replacement Activities

a. Inspection Scope

The inspectors reviewed Work Order Packages 76794 and 91590 for ASME Code Section XI repair and replacement of Safety Injection Check Valve SI-194 and Hand Control Valve HCV-331 and reviewed Safety Injection Check Valve SI-197, respectively.

The inspectors observed the fit up and welding of these components and verified that these activities were in accordance with the specified welding procedure specifications. The welding procedure specifications were verified, by review of the procedure qualification records, to be appropriately qualified. All welding material used on these work orders was properly identified and controlled. The inspectors also observed the subsequent fiber optic visual examination of the root passes and radiographic setups and shoots of the completed welds. The inspectors reviewed all of the applicable radiographic film and reader sheets. All indications were appropriately dispositioned, and identified defects were removed, repaired, and reexamined.

The inspectors observed and verified that controls were in place to assure that welding materials were properly stored, identified, certified, and distributed.

b. Findings

No findings of significance were identified

.3 Steam Generator Tube 2R15 Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee's in-situ screening criteria to verify that the criteria were in accordance with industry guidelines. The estimated size and number of tube wear flaws identified up to the date of the inspection were compared to the operational assessment predictions from the previous outage. The inspectors also reviewed the eddy current examination scope and expansion criteria to determine if the Technical Specifications, industry guidelines, and commitments to the NRC were being met.

The inspectors reviewed the areas of potential degradation (based on site-specific and industry experience) to verify that such areas were being inspected. The eddy current probes and equipment were reviewed to ascertain if they were properly qualified for the expected types of tube degradation.

The inspectors observed the collection and analysis of eddy current data by licensee personnel to verify that the eddy current procedure was being followed and that indications were being appropriately dispositioned.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalifications (71111.11)

a. Inspection Scope

The inspectors observed licensed operator requalification training activities, including the licensed operators' performance and the evaluators' critique. The inspectors compared performance in the simulator on June 17, 2002, with performance observed in the control room during this inspection period.

The inspectors placed an emphasis on high-risk licensed operator actions, operator activities associated with the emergency plan, and previous lessons learned items. These items were evaluated to ensure that operator performance was consistent with protection of the reactor core during postulated accidents.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

During the inspection period, the inspectors reviewed licensee implementation of the Maintenance Rule. The inspectors verified structure and component scoping, characterization, safety significance, performance criteria, and the appropriateness of goals and corrective actions. The inspectors compared the licensee's implementation of the Maintenance Rule to the requirements outlined in 10 CFR 50.65. The inspectors reviewed the following components:

- Pressurizer Spray Control Valves PCV-103-1 and PCV-103-2

- Inconel Pressurizer Nozzles
- Fire Suppression System Deluge Valves FP-526, FP-798, FP-749, FP-513, FP-708, FP-210, and FP-211

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors reviewed the licensee's risk assessments for equipment outages as a result of planned and emergent maintenance to evaluate the licensee's effectiveness in assessing risk for planned and emergent activities. The inspectors compared the licensee's risk assessment and risk management activities against requirements of 10 CFR 50.65(a)(4) and the recommendations of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2. The inspectors also discussed the planned and emergent work activities with planning and maintenance personnel. The inspectors reviewed and observed emergent work on the following:

- Heated Junction Thermocouples venting while draining to hot midloop on May 7, 2002
- Safety Injection Tank Level Instrumentation calibration while draining to hot midloop on May 7, 2002
- Electrical Bus Maintenance Activities planned to occur while in hot midloop on May 10, 2002
- Reactor Coolant Pump seal replacement while in cold midloop on May 28, 2002.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the technical adequacy of operability evaluations to verify that they provided adequate justification that the equipment could still meet its Technical Specification, Updated Safety Analysis Report, and design bases requirements. The following evaluations were reviewed:

- Diesel generator operability with fuel oil particulate contamination results being out of specification high (Condition Report 200200792)

- Diesel Generator operability after a pipe used in the sampling process fell into Fuel Oil Tank 10 (Condition Report 200200838)

- Auxiliary Spray Valve HCV-249 operability with the close stroke time greater than the acceptable value (Condition Report 200207735)

- Shutdown Cooling System Piping operability with Snubber SIS-134 being removed for repairs and fuel in the reactor (Condition Report 200201248)

- Containment Penetration M-39 Isolation Valve HCV-425B exceeding administrative leak limits (Condition Report 200201592)

- Containment Penetration M-45 Isolation Valves HCV-2504A and HCV-2504B exceeding administrative leak limits (Condition Report 200201912)

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors observed the installation of a modification to the Auxiliary Feedwater System. Specifically, a 4-inch gate valve was installed in the system to allow full flow testing of Auxiliary Feedwater Pumps AFW-6 and AFW-10 without stationing a dedicated operator at main and auxiliary feedwater crossconnect Valve HCV-1384. The inspectors reviewed Work Order 00117552, Engineering Change 30191, and Quality Control Reports 20021299 and 20021308.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors verified that postmaintenance tests were adequate to verify system operability and functional capabilities. The inspectors verified that testing met design and licensing bases requirements, Technical Specifications, the Updated Safety Analysis Report, inservice testing, and licensee administrative procedures. The inspectors verified test results for the following components:

- Closed indication for Pressurizer Vent Stop Valve HCV-178 repairs on May 29, 2002

- Air actuator NCV-6680B-2-0 repairs on May 29, 2002

- Work Order 00109544, Emergency Diesel Generator anti-freeze replacement on June 12, 2002

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors periodically observed plant conditions to verify that safety systems and support systems, including electrical distribution, were properly aligned with the shutdown operations protection plan. In addition, the inspectors observed the following evolutions: reactor shutdown, midloop operations, core reload from containment and the spent fuel building, and plant heatup. The inspectors performed a containment cleanliness and material readiness tour prior to the unit's return to power.

b. Findings

The licensee exercised a nonconservative decision making process when controlling foreign materials in containment and eliminating the potential for blocking the containment emergency sumps. As a result, approximately 20 55-gallon drums with paper taped onto the lid on the same elevation as the sumps and several hundred other pieces of tape remained in containment during initial plant heatup following a refueling outage. This was determined to be of very low safety significance using the Significance Determination Process.

On May 29, 2002, the inspectors performed a walkdown of the containment building to verify cleanliness and material readiness prior to the unit returning to power. The inspectors observed approximately 20 55-gallon drums with paper taped onto the lid on the same elevation as the sumps and several hundred other pieces of tape throughout containment. The unit, at that time, was at 300 degrees and pressurized to 600 psi with a heatup in progress. Licensee Technical Specification 2.3, "Emergency Core Cooling," states in part that all equipment required to function during accident conditions are to be operable prior to the reactor being made critical. Licensee Standing Order SO-M-10, "Foreign Material Exclusion," specifies that the intent of the procedure is to eliminate the potential for restricting flow to the emergency core cooling system suction strainers. The inspectors reviewed this and the revision of Procedure OI-CO-1, "Containment Closeout." The inspectors noted that neither procedure addressed keeping the emergency core cooling system suction strainers free of debris during plant heatup. The decision by the licensee to only consider the operational condition of the suction strainers, with respect to debris in containment, just prior to criticality and not during plant heatup, was nonconservative. The inspectors informed outage management about the observations. The unit, at that time, was at 300 degrees and pressurized to 600 psi with a heat-up in progress. The licensee dispatched personnel into containment to address the inspectors concerns. The inspectors reviewed Procedures OI-CO-1, "Containment Closeout," Revision 23, and SO-M-10, "Foreign Material Exclusion," Revision 22. The inspectors noted that neither procedure addressed keeping the emergency sumps free of debris during plant heat-up. The licensee initiated Condition Report 200202326 to address the inspectors' concerns.

The inspectors evaluated the significance of the issue. The inspectors determined this issue had a credible impact on safety because of the potential for plugging the containment emergency sumps (Group 1 question answered yes). The inspectors determined that the issue could credibly affect the availability of a mitigating system (Group 2 question answered yes). However, Using the significance determination process Phase 1 worksheet, the inspectors answered "no" to all questions under the mitigating systems cornerstone and therefore the issue screened out as Green (FIN 285/2002002-01).

This finding is in the licensee's corrective action program as Condition Report 200202326.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed/reviewed the following surveillance tests to ensure that the systems tested were capable of performing their safety function and to assess their operational readiness. Specifically, the inspectors verified that the following surveillance test met Technical Specifications, ASME Section XI test requirements, the Updated Safety Analysis Report, and licensee procedural requirements:

- Channel B safety injection, containment spray and recirculation actuation signal test on April 19, 2002
- Shutdown cooling pump refueling leakage test on April 24, 2002
- Personnel access lock O-ring seal test on May 28, 2002
- Control element assemblies drive system interlocks check on May 30, 2002

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Modification EC 30027, "Restrain or Disable Refueling System Fuel Handling Machine (FH-1) in the Retracted Position" to verify that the safety functions of the system were not affected.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS2 ALARA (As Low as Reasonably Achievable) Planning and Controls (71121.02)

a. Inspection Scope

The inspectors interviewed radiation workers and radiation protection personnel to determine if low dose waiting areas were utilized, personnel were maintaining doses ALARA, and radiation workers were receiving appropriate job supervision and radiation protection coverage. The inspectors reviewed a summary of ALARA and radiological worker performance condition reports written since November 2001. The following condition reports were reviewed in detail: 200103694, 200103725, 200200268, 200200379, 200200288, and 200200392.

The inspectors reviewed nine radiation work permit packages. These packages included five pre-outage and four spent resin transfer recovery packages for work activities with the highest estimated personnel collective exposures and selected total effective dose equivalent ALARA evaluations. The following radiation work permits were reviewed: 02-2510, "Steam Generator Support in High Radiation Areas"; 02-3007, "Investigate Cause and Extent of Radiological Conditions Following Dewatering of the Spent Resin Storage Tank"; 02-3008, "Drain Vent Header in Rooms Al-100, 7, 13 and 16"; 02-3009, "Dewater Spent Resin Storage Tank (SRST)"; 02-3011, "Restore Radiological Conditions Pre-SRST Overfill"; 02-3509, "Primary Side Steam Generator Work Requiring Full Jumps"; 02-3510, "Primary Side Steam Generator Support Work"; 02-3529, "Upper Guide Structure Lift Rig Repair"; 02-3534, "Inspection of Nuclear Well Cooling System Insulation."

The following items were reviewed and compared with regulatory requirements to determine whether the licensee had an adequate program to maintain occupational exposures ALARA:

- ALARA program procedures

- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information

- Use of engineering and administrative controls to achieve dose reductions, including four temporary shielding request packages (TSR 02-034, TSR 02-035, TSR 02-037, TSR 02-038) and scheduling of work activities

- Individual exposures of selected work groups (steam fitter mechanics, mechanical maintenance, radiological operations, and radwaste operations)

- Five preoutage ALARA packages for work activities with the highest estimated personnel collective exposures (02-08, "Fuel Movement"; 02-10, "A/B Steam Generator Primary Side Services"; 02-11, "A/B Steam Generator Secondary Side Services"; 02-12, "Reactor Head Removal and Replacement Tasks"; 02-18, "Replacement of RC-3C Rotating Assembly")

- Draft 2002 ALARA Program One Year and Five Year Plan, Nuclear Safety Review Group Observation 02-QUA-018 and selected quality assurance surveillance observations from November 2001 to February 2002 (2001-480, 481, 576, and 592 and 2002-21, 49, 126, 158, 161, and 166)

- Hot spot and point source tracking and reduction program

- Overall facility source term reduction plan

- Radiological work planning and interfaces between various departments

- ALARA Committee Meeting minutes since November 2001

There were no declared pregnant workers since November 2001.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA5 Other

Temporary Instruction 2515/145: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

a. Inspection Scope

The inspectors observed and reviewed licensee activities in response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," issued on August 3, 2001, in response to identified circumferential cracking in control element drive mechanism nozzles at other facilities.

The licensee performed a 100 percent visual inspection of the reactor pressure vessel head nozzle penetrations using a contractor developed procedure approved by the licensee. The licensee used a robotic device to perform a 360 degree inspection around each nozzle penetration. The penetrations near the reactor pressure vessel head outer edge could not be inspected by the robotic device due to the head and insulation configuration. These areas were inspected using a boroscope that was attached to the robotic device.

The licensee detected no evidence of boric acid deposits as described in NRC Bulletin 2001-001. However, the licensee observed boric acid stains in some locations on the reactor pressure vessel head and on some nozzles that were associated with flange leakage. The vessel head contained small debris particles located around nozzles where they penetrate the reactor vessel head and a few small mechanical fasteners. The debris was easily blown away from the nozzle area with air. All deficiencies identified were dispositioned.

The licensee's quality control personnel involved with the inspection were VT-2 qualified. They received additional training from the contractor on how the robotic inspection would take place. The training consisted of how the robot was operated, the limitations, and practical experience on a reactor pressure vessel head mockup.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meeting Summary

.1 The inspectors presented the inspection results to Mr. D. Bannister, Plant Manager, and other members of licensee management at the conclusion of the ALARA planning and controls inspection on April 12, 2002. The licensee acknowledged the findings presented. No proprietary information was identified.

.2 The inspectors presented the inspection results to Mr. R. Ridenoure, Division Manager, Nuclear Operations, at the conclusion of the inservice inspection program inspection on May 22, 2002. The licensee acknowledged the findings presented.

The inspectors noted that, while proprietary information was reviewed, none would be included in this report.

.3 The inspectors presented the inspection results to Mr. R. Ridenoure, Division Manager, Nuclear Operations, at the conclusion of the resident inspectors' inspection on July 2, 2002. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

KEY POINTS OF CONTACT

Licensee

D. Bannister, Plant Manager
C. Bloyd, Component Testing Supervisor
G. Cavanaugh, Acting Manager, Nuclear Licensing
J. Chase, Division Manager, Alliance
S. Coufal, ALARA Technician, Radiation Protection
D. Dryden, Engineer, Station Licensing
M. Core, Manager, System Engineering
J. Goodell, Manager, Operations
P. Hamer, Component Testing, Inservice Inspection Program Coordinator
B. Lisowyj, Design Engineer
J. McBride, ALARA Technician, Radiation Protection
J. McKinley, Acting Manager, Maintenance
E. Matzke, Licensing Engineer
R. Phelps, Division Manager, Nuclear Engineering Division
M. Puckett, Manager, Radiation Protection
R. Reno, ALARA and Radiological Equipment Supervisor, Radiation Protection
R. Ridenoure, Division Manager, Nuclear Operations
K. Steele, Operations Supervisor, Radiation Protection
C. Williams, ALARA Technician, Radiation Protection
K. Woods, Senior Nuclear Design Engineer

Others:

R. Maurer, Westinghouse Corporate NDE Level III
K. Rajan, Westinghouse Steam Generator Programs

ITEMS OPENED AND CLOSED

50/285/02-01 FIN Nonconservative processes for Controlling Containment Cleanliness (Section 1R20)

DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

APPENDIX D

LISENCEEE EVENT REPORT (LER) ANALYSIS of US NUCLEAR POWER PLANTS 2000-2015

LER Number	Plant Name	Mode	Event Date	Details	Category	Subcategory
3682008002	Arkansas 2	3	04/07/2008	The cause of this event was inadequate communication between the SM and RE personnel which led to the incorrect conclusion that the CEA position indicators were inoperable.	team	communication
3682009002	Arkansas 2	6	09/07/2009	It was determined that the failure to apply proper containment closure controls was caused by inadequate procedure instructions for restoration of the affected system following local leak rate testing.	team	procedure
3682009003	Arkansas 2	6	09/08/2009	Investigation revealed that the error in plugging was caused by a failure to use proper independent verification.	team	technical
3682014004	Arkansas 2	3	06/09/2014	The work instructions for the normal control system calibration performed in Mode 5 was not performed as written and did not require a second verification after critical adjustments were performed.	team	performance+ work permit
3682015001	Arkansas 2	6	10/26/2015	A human performance error resulted in this condition due to using the wrong sample flow indication to verify flow to the Containment Purge and Exhaust Isolation Process Monitor.	team	
3342006005	Beaver Valley 1	3	08/25/2006	Human performance errors occurred during the implementation of procedure changes associated with the Extended Power Uprate License Amendment Request	team	performance+ check
3342009003	Beaver Valley 1	6	04/23/2009	The wood was left behind as a result of inadequate housekeeping practices during the original construction of the containment wall.	team	technical+performance
3342010002	Beaver Valley 1	5	10/02/2010	The most probable root cause is less than adequate (LTA) consideration of vibration induced fatigue in the design process.	design	
3342013002	Beaver Valley 1	5	10/04/2013	The foreign material was introduced as a result of inadequate worker practices and quality control during the original construction of the containment wall.	team	technical+performance
4122011001	Beaver Valley 2	6	03/25/2011	The root cause is the vendor (Fairbanks Morse Engine) design change control process was inadequate in that the design change made to the fuel injection pump supply lines incorporated the use of an unsuitable ferrule.	design	team
4122012003	Beaver Valley 2, Beaver Valley 1	3	10/30/2012	The root cause was determined to be an inaccurate perception of the applicability of the turbine trip TS in combination with less than adequate information in plant documents.	team	procedure
4562007003	Braidwood 1	3	10/24/2007	To gain margin to temperature limits, a walkdown of the MSIV rooms identified several sections of hot pipe with the insulation missing, including the MSSVs. During the process of determining the proper insulation of the MSSVs, a number of opportunities to determine whether installing insulation would affect MSSV operability existed but failed to recognize the impact	team	performance
4562009001	Braidwood 1	6	04/08/2009	The cause of this event was determined to be a historic human performance issue related to the amount of technical rigor applied during the review of the distorted eddy current as identified by the computer screening system in the fall 007 outage.	team	historical

4562010002	Braidwood 1	3	08/16/2010	. The Braidwood procedure for reactor trip response does not provide adequate guidance to alert operators to potential LCO 3.3.9 entry during plant transients. • Emergency operating procedures do not reference Technical Specifications. Per the Westinghouse Owners Group writer's guide, Technical Specification entries are expected during plant transients.	team	procedure
4562013002	Braidwood 1	6	09/14/2013		technical	
4562015002	Braidwood 1	6	04/03/2015		technical	
2592007001	Browns Ferry 1	2	05/21/2007		team	coordination &communication
2592007002	Browns Ferry 1	2	05/24/2007		team	implemetation +follow-up
2592008002	Browns Ferry 1	4	11/23/2008		technical	
2592010003	Browns Ferry 1	3	10/23/2010		team	coordination &communication+cognition
2592010004	Browns Ferry 1	5	10/27/2010		team	follow-up+supervision
2592011005	Browns Ferry 1	4	04/27/2011	Reactor water level lowered below +2 inches.	team	leadership
2592012010	Browns Ferry 1	5	11/22/2012	. Due to the design of Marotta instrument line check valves, it is possible to install the check valve in a reverse orientation. The work instruction does not require the technician to ensure the valve is installed with the correct orientation, nor does it require the installation to be independently verified. This was determined to be the root cause of the event.	team	procedure
2592014004	Browns Ferry 1	5	10/04/2014		technical	
2592011002	Browns Ferry 1, Browns Ferry 2, Browns Ferry 3	4	04/28/2011		design	
2592011003	Browns Ferry 1, Browns Ferry 2, Browns Ferry 3	5	05/02/2011	Inadequate procedural guidance for setup of the EDG OTLS resulting in a failed safety system Inadequate procedural guidance for other safety related (SR) components resulting in a failed EDG Inadequate procedural guidance for other SR components resulting in a failed safety system	team	procedure

2592012009	Browns Ferry 1, Browns Ferry 2, Browns Ferry 3	5	10/31/2012		team	cognition
2602009006	Browns Ferry 2	5	05/24/2009	The root cause of this event was incomplete restoration of the SDV clearance and lack of awareness of the SDV system configuration. The control room operators were not cognizant of the fact that the SDV was isolated and full of water in combination with the associated SDV high level scram signal being bypassed.	team	cognition+coordination&communication
2602009008	Browns Ferry 2	3	09/30/2009	**A. Immediate Cause** The immediate cause of the event was a failed Woodward EG-M control box. Woodward personnel identified an age related failure of a timing capacitor within the EG-M.	team	cognition+tec hnnical
2962007003	Browns Ferry 3	3	09/22/2007		team	design
2962008001	Browns Ferry 3	5	05/05/2008	The initial investigation found that when the operator closed the normal supply breaker to 4KV Unit Board 3B, there was indication of disagreement between the demand breaker position and the actual breaker position. The Unit Supervisor directed the operator to transfer the board back to the alternate supply. However, the alternate supply breaker did not close resulting in the 4KV Unit Board being de-energized.	human	
2962009003	Browns Ferry 3	2	03/22/2006	**Immediate Cause** The immediate cause for the inoperable RCIC pump was the EG-R actuator nonconformance and the resulting reduced stability of the RCIC governor control system during RPV injection. The EG-R was absent critical parts that would keep the RCIC pump from oscillating during RPV injection.	team	technical
2962010003	Browns Ferry 3	6	03/26/2010		team	guidance
2962011001	Browns Ferry 3	5	05/12/2011	Upon discussion with Operations, the electricians were directed to lift the leads while energized. The logic prints reviewed did not show the neutral for the relays. The direct cause of this event was the lifting of the wiring that was landed on the common side of a relay coil.	team	procedure+training+cognition
2962011002	Browns Ferry 3		05/22/2011			
2962012002	Browns Ferry 3	5	04/07/2012	The immediate cause of the event was inadequate packing on MSIV	team	procedure
2962012004	Browns Ferry 3	2	05/24/2012		technical	
2962012006	Browns Ferry 3	5	05/25/2012		technical	
2962013002	Browns Ferry 3	4	02/11/2013		team	cognition+design
2962016002	Browns Ferry 3	5	02/22/2016	The cause of each component or system failure or personnel error, if known: Troubleshooting determined switch failure was caused by a failure of the 6-6C contacts on	team	cognition+procedure

				the 52STA switch, from and a binding of the 52STA Cam Linkage. This binding was caused by a misalignment of the switch to linkage interface, due to improper installation.		
2962016003	Browns Ferry 3		02/23/2016			
3252010001	Brunswick 1	3	02/27/2010	A contributing cause to this event was that licensed operator training did not provide sufficient detail for the interrelations between the condenser hotwells and the components that input to them on a shutdown unit.	human	training
3252010002	Brunswick 1	2	04/25/2010	The select cause of this event was the failure to effectively use the concurrent verification during the performance of procedure	team	verification
3252014002	Brunswick 1	6	03/09/2014	This event resulted from a deficiency in procedure OMMM-054, "Temporary Power Feed Documentation.	human	procedure
3252014003	Brunswick 1	6	03/13/2014	The root cause of these events is that the design of the secondary containment airlock door interlocks is not robust enough to prevent inoperability of secondary containment.	design	team
3252016002	Brunswick 1, Brunswick 2	6	3/4/2016	The root cause of this event is a design vulnerability associated with relaxation of the EDG 3 fuse holder fingers which was not properly mitigated. The existing design lacks circuit continuity indication that is not mitigated by design or testing.	team	design
3252012003	Brunswick 1, Brunswick 2	6	04/09/2012	The root cause of this event is inadequate use of human performance tools when connecting recorders in preparation for performing OMST-DG11R	team	performance
3242006002	Brunswick 2	6	11/11/2006	The root cause of this event is the failure to have procedural guidance to inspect the condenser water boxes for missing tube plugs following a LOOP event.	team	procedure
3242007001	Brunswick 2	6	03/26/2007	The root cause of this event was inadequate procedures. Existing operating procedures did not provide adequate guidance identifying prerequisites required to be met prior to performing core alterations, due to control rod movement.	team	procedure
3242007002	Brunswick 2		04/17/2007		technical	
4542008001	Byron 1	6	03/28/2008	Due to the time frame the installation error was made (i.e., 1976), the investigation was unable to determine the cause for this installation error. The most probable cause was inadequate quality control oversight of the installation process.	team	planing(timing)
4542014003	Byron 1	6	03/15/2014		technical	
4542015005	Byron 1	6	09/18/2015		technical	
4542005003	Byron 1, Byron 2	6	09/20/2002	The most probable cause is the authors and reviewers of the original TS and TS Bases wording inadvertently used imprecise language. The outage schedule was reconfigured to not allow the performance of LLRTs that provide direct access from containment to the auxiliary building during core alterations or movement of irradiated fuel within containment.	team	procedure(wording)
4552007001	Byron 2	6	04/09/2007		technical	

4552010001	Byron 2	4	04/19/2010	The causes of the inadvertent actuation signals include Operations supervisory oversight of the FWIV testing activity was less that adequate, and the FWIV surveillance procedure was inadequate in that it did not provide for the specific recovery of SG level during the conduct of the procedure if levels were approaching the low actuation setpoint.	team	supervision+procedure
4832006001	Callaway	4	11/14/2005	Evaluations performed by Callaway engineering personnel have determined that PORV stroke times measured during surveillance testing did not account for all of the delay times credited in the Design Bases CONS Analyses of Record. Further reviews determined the allowed delay times could not be met by the control loop.	team	calculation
4832008003	Callaway	4	10/11/2008	• OTG-ZZ-00006, Plant Cooldown Hot Standby to Cold Shutdown, is the only procedure utilized for the cooldown of the plant from NOP/NOT. This procedure did not contain a provision to ensure that another heat sink • The BOP operator did not inform the crew prior to opening the MSIV. • The Shift Manager and Control Room Supervisor did not provide effective oversight as unplanned activities caused a loss of focus.	team	procedure+coordination+communication
4832008004	Callaway	6	10/17/2008	The investigation found that the root cause of the failure to have the containment purge and exhaust system in service during core alterations with the equipment hatch open was a failure to adequately or completely implement Callaway Operating License Amendment 152 in procedures.	team	performance
4832008007	Callaway	3	12/12/2008	Maintenance personnel did not understand that a relay was energized to enable the IR HI Flux Reactor Trip signal bypass and that 118-VAC control power feeds this relay in SSPS. This was the root cause of the event.	team	cognition
4832009003	Callaway	3	11/04/2008	he Root Cause for this event is that the procedure did not meet the requirements of the Callaway Procedure Writing Manual in regards to acceptance criteria.	team	procedure
4832011005	Callaway	4?	10/24/2011	First, no Preventative Maintenance task existed to replace the grease in the valve actuator. Grease had been added to the ASD manual isolation valve ABV0040 in 2007, but the existing grease was not removed before the fresh grease was added. The site failed to develop a maintenance/testing program for the ASD isolation valves to ensure that they could be stroked closed to meet the time requirement in the Safety Analysis.	team	performance+check
4832011006	Callaway	6	11/09/2011	The root cause of this condition was attributed to human performance error during development of a modification to replace sections of steel piping with HDPE piping in the ESW system. Specifically, personnel developing this modification did not effectively evaluate the failure modes of the HDPE piping installed in Room 3101.	team	performance+check
4832011007	Callaway	6	11/13/2011	In light of the need to transition the plant from "No Mode" to Mode 6 during a refueling outage, a cause evaluation determined that procedure OSP-BL-00001 did not include adequate instructions to control the status of valve BGV0601 in "No Mode."	team	procedure+communication
4832013004	Callaway	6	04/18/2013	• Personnel responsible for developing plans to prop open the CBE door (to provide temporary power) did not review TS 3.7.10 for applicability. • Procedures and written instructions did not direct the user to check for the TS Mode of Applicability, leading to the incorrect TS condition being referenced in the FPIP. • Operations personnel did not promptly review TS 3.7.10 when it was identified that the CBE door would be propped open.	team	planing+procedure+performance
4832013006	Callaway	6	05/08/2013		technical	

4832013007	Callaway	5	05/19/2013	The cause of this event was a human performance error committed during maintenance activities on the Startup Transformer which involved the following two critical errors:	team	performance
4832014005	Callaway	3	11/18/2014	The root cause of the event was an incorrect decision by Operations personnel to deviate from the sequence of steps in the approved test procedure,	team	cognition
4832015004	Callaway	3	08/11/2015	The root cause of the card failures was determined to be a vendor design deficiency. The defective positioner cards have been replaced and measures have been taken to remove defective spares from future plant use.	design	
4832016001	Callaway		04/20/2016	The root cause of the event is that the original Essential Service Water (ESW) system design did not appropriately account for water column separation and collapse pressure transients inherent during operation.	design	
3172014003	Calvert Cliffs 1		02/28/2014		technical	
3172014005	Calvert Cliffs 1	3	03/14/2014	The apparent cause was that Technicians failed to apply human performance tools to maintain proper configuration control.	team	review
3172006001	Calvert Cliffs 1, Calvert Cliffs 2	6	03/24/2006	An engineering evaluation determined that the original amptector design setpoint, established by a vendor, was not adequate because the setpoint did not consider all potential loads	human error	
3182010002	Calvert Cliffs 2	5	02/23/2010	the apparent cause of the pinhole was a latent weld defect created during the original valve manufacturing process.	technical	
3182013001	Calvert Cliffs 2	3	02/17/2013		technical	
3182013002	Calvert Cliffs 2		03/12/2013		technical	
4132006002	Catawba 1	3	05/22/2006	The identified flood protection deficiencies were attributed to inadequate design and configuration control of features to protect against flooding. The flood protection deficiencies were corrected by the installation of new flood protection seal barriers.	design	weather
4132010002	Catawba 1	4	02/18/2010	The cause of the failed seal weld was inadequate weld control when the weld was fabricated during initial construction. The weld failure resulted from the presence of a discontinuity involving a metal removal process.	team	control+design
4132011002	Catawba 1	4	04/23/2011	This event was caused by the placement of a tagout prior to its scheduled execution time without fully recognizing its effect on plant operation.	team	org+supervision+cognition
4132012003	Catawba 1	3	12/22/2012	The cause of this event was determined to be human performance error by the maintenance technician during the assembly of the turbine and pump. In addition, the governing maintenance procedure was deficient in that it lacked detail concerning the required orientation of a turbine component during the assembly process.	human	cognition
4142007002	Catawba 2	5	11/05/2007		unclear	
4142010001	Catawba 2	5	01/11/2010	The cause of the event was that unclear TS Bases did not address inoperability of the NS system when the valves in question were de-energized.The root cause for entering Mode 4 with an inoperable NS train on April 16, 2009, was due to a failure to recognize the tie between ECCS TS 3.5.3 and NS TS 3.6.6.	team	procedure

4142010002	Catawba 2	4	10/17/2010	The cause of the failed seal weld was inadequate weld control when the weld was fabricated during initial construction.	team	control
4612008003	Clinton 1	6	01/21/2008	The actuator had not been re-lubricated since initial installation during a refueling outage that ended in May 1999. No preventive maintenance activities existed to lubricate or overhaul the actuator.	team	performance
4612010002	Clinton 1	4	02/03/2010	The cause of the 1B21-F032B.check valve to fail its leak rate test was age-related degradation of the lubrication causing increased friction in the actuator.	team	performance
4612010004	Clinton 1	6	01/17/2010	The station did not use conservative decision-making before proceeding with implementation of the OPDRV procedure.	team	cognition
4612011005	Clinton 1	6	12/01/2011	A subsequent evaluation concluded that the pre-test stroking was unacceptable preconditioning; therefore, the as-found LLRT was invalid and considered to be a missed surveillance and a condition prohibited by Technical Specification 3.6.1.3. The test procedure incorrectly required the preconditioning and caused this event.	team	procedure
4612011006	Clinton 1	6	12/07/2011		technical	
4612011007	Clinton 1	6	12/08/2011		technical	
4612011008	Clinton 1	5	12/18/2011	The cause evaluation for this event identified two root causes, one cause was the lack of rigorous process controls while removing and installing the permanent shutdown and upset level instruments reference leg pipe, specifically, instructions on how to fill the shutdown and upset level instruments reference leg pipe were inadequate and there was insufficient guidance on how to perform a check of the restored instrument	team	control+guidance
4612011009	Clinton 1	6	12/01/2011	Test procedure CPS 9843.01V001 was inadequate in that the draining operation from the previous LLRT test on a motor-operated valve (MOV) in this line moved the check valve disc from its seat	team	procedure
4612013007	Clinton 1	2	10/28/2013	The SM did not review the surveillance requirements contained within the procedure being performed nor did he reference TS or TS Bases documents as required by process. Additionally, no peer check was sought by an independent SRO as expected by management	team	check
4612016007	Clinton 1	5	05/17/2016	The root cause evaluation for this event determined that the corrective actions to prevent recurrence of the condition identified June 18, 2007 (LER 2007-003) failed to eliminate or significantly reduce below threshold any of the three factors required for IGSCC to exist (susceptible material, tensile stresses, and aggressive environment).	team	check+historical
3972006002	Columbia	4	11/03/2006	The cause of this event was an inadequate procedure step derived from inaccurate technical information in procedure SOP-RHR-SDC-BYPASS.	team	procedure
3972011001	Columbia	5	06/29/2011	The Control Room Supervisor and Shift Manager did not verify the required action statements specified in the TS and Bases as required. This was determined to be the apparent cause.	team	communication+coordination+procedure
3972013003	Columbia	5	06/03/2013	The cause of this event was inadequate procedure guidance for actions to take when unexpected OPDRV conditions are encountered	team	procedure
3972013004	Columbia	5	06/04/2013	The preliminary apparent causes have been identified as: (1) a lack of a standard for Work Order instructions involving the removal and installation of Jumpers, resulting in personnel having to rely on experience and skill of the craft for the proper way of executing	team	procedure+cognition

				anddocumenting required modifications; and (2)inadequate decision making resulting in the use of a post maintenance testing procedure after the installation of the Spectrum bucket in May 2011 that did not adequately prove operability.		
3972015004	Columbia	5	05/22/2015	The temporary loss of E-SM-7 and resulting emergency diesel actuation was due to a human performance error that occurred when the electrician reconnected the meter test lead incorrectly. When discovering the detached test lead, the electrician proceeded to re-attach it to the meter and connected it incorrectly.	team	performance+ supervision+training
3972015005	Columbia	4	06/25/2015	The root cause of the event is that the procedure for calibrating the level indicating switches is not in alignment with the vendor manual with respect to setting the mechanical stops. Contributing causes were that there was no established preventativemaintenance to ensure the stops were set correctly and there was no verification that the level switch indication was on scale prior to entering Mode 2.	team	procedure
4452010002	Comanche Peak 1, 2	3	01/20/2010	The cause analysis of this event determined that the plant design and original operating philosophy was not compatible with the NRC's clarification of the intent of TS 3.3.2, Function 6.g.	team	design+org
4452011003	Comanche Peak 1, 2	5	10/18/2011	The cause of this event was an inadequate design change for the airlocks.	team	design
4462011001	Comanche Peak 2	2	04/26/2011		technical	
4462014002	Comanche Peak 2	3	04/25/2014	The cause of the event has been determined to be due to a failure to transmit pertinent information to support a decision to change plant operating modes.	team	procedure
3152006003	Cook 1	6	10/24/2006	The causes of this event were: Inadequate surveillance procedure: The governing surveillance procedure allowed steps establishing initial conditions to be performed in any sequence; thus it failed to establish adequate controls to ensure the CPS was removed from service prior to placement of the output mode selector switch in the test position.	team	supervision
3152008004	Cook 1		04/25/2008		technical	
3152010001	Cook 1	3	04/09/2010	An In-Depth Apparent Cause Evaluation analysis determined that the Apparent Cause of cutting the wrong power supply cable was that verification actions performed by electricians failed to verify the correct cable to be cut.	team	procedure
3152013001	Cook 1		03/31/2013		technical	
3162006003	Cook 2	5	04/27/2006	The root cause of the inadvertent SI actuation was the failure of instrument maintenance personnel to implement procedure use and adherence requirements when using the bypass function to clear the standing reactor trip signals.	team	work practice
3162006004	Cook 2	5	04/21/2006		team	performance
3162007001	Cook 2	5	10/28/2007	event can be attributed to human error due to inadequate validation of assumptions. The emergent work order tasks were created in an expeditious manner to ensure completion of the work within the known window of opportunity. The job planner understood that as-found LLRTs are required for these valves, but incorrectly assumed that the maintenance was	team	cognition

				required because the valves failed their as-found LLRTs.		
3162010002	Cook 2	5	10/15/2010	This procedure format can lead to errors due to not providing a method or process to track completed portions of the surveillance and document discrepancies as the inspection progresses.	team	procedure
3162010003	Cook 2	4	11/30/2010	Contributing causes were lack of adequate oversight of the field work such that exceeding work scope was not identified, and a failure of the modification and work control processes to implement the TS Surveillance requirement as a post-modification test/inspection to verify the divider barrier was operable prior to entering Mode 4.	team	work practice+surv alience
3162012001	Cook 2	5	03/22/2012	An evaluation of the deficiencies concluded that the divider barrier seal was inoperable, as it did not meet the requirements of the seal inspection surveillance. It is not known when the deficiencies occurred. It is conservatively concluded that the deficiencies existed during plant operation in Modes 1 through 4. Because the deficiencies had not been recognized, actions required by TS for an inoperable divider barrier seal in Modes 1 through 4 were not taken.	team	
2982006002	Cooper Station	5	04/06/2003	The cause was inadequate procedures. After adoption of Improved Technical Specifications, CNS failed to include in applicable procedures a caution to alert operators that continuing scram time testing above 212 F after hydrostatic test was completed could violate LCO 3.10.1. The procedural inadequacy was caused by a human error of not considering that perforining other evolutions as part of hydrostatic testing	team	procedure
2982009001	Cooper Station	5	11/01/2009	The cause of the error is attributed to inadequate programmatic controls for maintaining configuration control of the 5x5 array.	human	
2982009003	Cooper Station	5	11/07/2009	The operating crew did not demonstrate sufficient control of reactor pressure while Shutdown Cooling was in operation. Reactor pressure decreased to a negative value resulting in flashing of water to steam while in Shutdown Cooling.	team	procedure
2982011002	Cooper Station		04/17/2011	The root cause of the event is a lack of inspection protocol for large electric motors which includes a check for loose bolts.	team	procedure
2982012004	Cooper Station	4	10/14/2012	The root cause was the station procedure provides insufficient guidance to avoid automatic closure of the isolation valves during shutdown cooling heatup and flush when the reactor temperature is higher than 212 degrees Fahrenheit.	team	procedure
2982012005	Cooper Station	6	10/17/2012	The root cause was corrective actions put in place to preclude the purchase of SWBPs with high pressure volute area flushing ports were not effectively implemented.	team	
2982012006	Cooper Station		11/07/2012	The root cause was written instructions requiring the vent cap be replaced to ensure secondary containment integrity when the air sampling was complete and the Z sump was returned to service, did not exist.	team error	procedure
3022005005	Crystal River 3		11/14/2005			
3462006002	Davis-Besse		03/18/2006		technical	
3462007002	Davis-Besse	5	12/30/2007	Neither the system operating procedure section for filling and venting Decay Heat Removal Train 1 following maintenance in Modes 1 to 3 nor the Operations Evolution Order contained instructions to vent from DH73, Decay Heat Pump 1 Discharge Line Leak Test Connection Valve, which was the high point of piping drained during the train outage.	team	procedure

3462008001	Davis-Besse		01/04/2008		technical	
3462010001	Davis-Besse	5	03/02/2010	The root cause of the event is less than adequate time allowed for development of the modification that re-designed the SFRCS in that the schedule for implementing the modification was accelerated.	team	management
3462010002	Davis-Besse	6	03/12/2010	The direct cause of this event is primary water stress corrosion cracking (PWSCC) of the CRDM nozzles.	team	technical+cognition
3462012001	Davis-Besse	6	05/19/2012	The cause of this event was determined to be less than adequate administrative controls for maintaining the DC System power source operability with the system cross-tied during shutdown conditions. Subsequent testing showed the equipment was operable, and procedures will be revised to add a prerequisite for ensuring operability of the motor control center being transferred to, or to ensure both EDGs are operable.	team	administration
3462012002	Davis-Besse	3	06/06/2012		design	
3462014002	Davis-Besse	3	05/05/2014	The cause of this event was not completely identifying and assessing all risks and consequences before conducting the tube replacement.	technical	
3462016002	Davis-Besse	3	01/30/2016	The cause of this event was inadequate procedural guidance contained in the Trip Recovery Procedure with a corrective action to revise the procedure.	team	procedure
3462016004	Davis-Besse	5	04/05/2016	he cause was determined to be less than adequate installation instructions. Corrective actions are to revise the Maintenance procedure and implement further training.	team	procedure
2752007002	Diablo Canyon 1	3	05/27/2007	immediate cause:the column adapters and inadequate verification of the washer installation.	team	cognition+ coordination &communication
2752009001	Diablo Canyon 1	4	03/22/2009		technical	
2752011001	Diablo Canyon 1		11/06/2010		team	documentation
2752012003	Diablo Canyon 1	5	06/07/2012		team	cognition
2752015001	Diablo Canyon 1		12/31/2014		technical	
2752015002	Diablo Canyon 1	5	10/05/2015		team	performance+ procedure
3232009003	Diablo Canyon 2	6	10/23/2009	1. A legacy issue from 1991 resulted in MP E-53.10V1, not including adequate guidance for rotor coordination if a limit switch is reset. 2. Maintenance Procedure MP E-53.10V1 and MP E-53.10S did not identify that	team	calculation+procedure

				performance of specific steps requires implementation of MMD M-000073-1. That resulted in MMD M-000073-1 not being properly implemented. 3. Engineering Calculation V-07 did not provide adequate precautions and limitations regarding the potential uncertainty in the final value when calculating stroke time.		
3232013002	Diablo Canyon 2		03/12/2013		team	technical
3232013003	Diablo Canyon 2	4	03/18/2013		team error	supervision
2372009004	Dresden 2	5	11/02/2009		technical	
2372009005	Dresden 2	5	11/03/2009		technical	
2372009008	Dresden 2	5	11/03/2009	plant personnel discovered that a small bore pipe was not connected to two of its supports and the piping appeared to be bent downward away from the supports. Additionally, there was one pipe support that was determined to be missing.	technical	
2372011003	Dresden 2	5	10/17/2011		team	procedure
2372011005	Dresden 2	5	10/28/2011		technical	
2372013008	Dresden 2	5	11/25/2013		technical	
2372015002	Dresden 2	4	02/07/2015		team	procedure
2372009006	Dresden 2,3	5	11/12/2009			
2372009007	Dresden 2, 3	5	11/27/2009		design error	
2372011004	Dresden 2, 3	5	10/24/2011		team	performance
2492008003	Dresden 3	5	11/03/2008		team	procedure
2492010002	Dresden 3	4	11/01/2010		team	design
2492014001	Dresden 3	5	11/06/2014		technical	
3312007002	Duane Arnold	6	02/12/2007	The work order steps that opened the 3 eight inch penetrations between the control room floor and the cable spreading room were not followed properly, thereby creating a condition that the control room operators were not aware of. DAEC staff also did not understand that opening of penetrations within the control building envelope compartments could render the control building boundary inoperable.	team	coordiantion+ communicatio n

3312007004	Duane Arnold		02/24/2007	An investigation was completed under Apparent Cause Evaluation (ACE) 1697. The cause of this event was a severe winter storm that brought snow, ice accumulation and high winds to the area. This storm caused extensive damage to the area including damage to the electrical grid.	technical	weather
3312007005	Duane Arnold	6	03/02/2007	The apparent cause of this event was determined to be inadequate guidance provided in the Surveillance Test Procedure NS550002. Specifically, the procedure did not provide guidance for bypassing the SDV High Level Scram signal prior to resetting the scram even though the normal scram recovery procedure provides such guidance.	team	procedure+questioning
3312009002	Duane Arnold	unclear	02/03/2009	The ACE determined that cause of the event was a lack of understanding of plant conditions during the period following the plant scram on the part of the radwaste operator. Specifically, theradwaste operator did not know that the LPCI Full Flow Test was in process, and therefore, sending unfiltered, highly contaminated reactor water	team	cognition
3312010005	Duane Arnold	5	11/10/2010	The cause of this event was due to an inadequate procedure revision process that introduced a latent procedure deficiency associated with isolating motive power to the RHR Shutdown Cooling isolation valves when removing the Reactor Protections System (RPS) from service.	team	procedure
3312012005	Duane Arnold		11/24/2012	The work order that completed the maintenance performed on June 21, 2012, did not direct which direction (clockwise or counterclockwise), to turn the containment isolation damper stop rod adjustment nut. The stop rod was adjusted in the wrong direction thus preventing the damper from fully closing.	team	procedure
3312014006	Duane Arnold	6	10/12/2014		technical	
3482006002	Farley 1		04/08/2006		technical	
3482010004	Farley 1	unclear	10/29/2010	A latent procedure error existed in the site procedure governing refueling integrity when a design change for each RHR pump on each unit installed two additional seal cooler vent valves. Although plant drawings were appropriately updated, the impact to the refueling integrity procedure was not recognized.	team	historical
3482012003	Farley 1	6	04/05/2012	The direct cause of the event was the failure to correctly perform a procedural step to parallel the 1-2A EDG with off-site power and return it to automatic standby operation	team	procedure
3482012004	Farley 1	6	04/06/2012	The direct cause of the event was an inadequate test procedure that was missing a step to isolate one of the fault protection relay schemes for PCB 820.	team	procedure
3482013002	Farley 1	6	10/04/2013	The direct cause of this event was determined by examination and testing to be inadequate lubrication of the MOC switch. Causal analysis of the inadequate lubrication determined the root cause of this event to be an inadequate procedure review process in 2002 that resulted in MOC switch preventive maintenance procedures having no associated task directing the performance of the procedure.	team	procedure
3482015001	Farley 1	2	05/05/2015	The human performance cause of this event was that the operating crew did not meet expectations for effective teamwork to ensure proper decision making.	team	coordination+communication
3642014001	Farley 2	3	01/11/2014	The root cause of this event was determined to be that station leadership did not appropriately manage the risk associated with past indeterminate SSPS failures.	team	leadership
3642014003	Farley 2	2	11/15/2014		technical	

3412006001	Fermi 2	6	04/01/2006		technical	
3412007001	Fermi 2	6	10/07/2007	The valve failures were primarily attributed to soft seat erosion by feedwater flow to the point that the seats were not providing an effective seal. It was also determined that the soft seat replacement frequency of two operating cycles was less than adequate to reliably ensure LLRT requirements are met.	team	performance
3412012001	Fermi 2	6	04/11/2012	The STR was not properly cleared before authorizing the performance of the SOP to energize Bus 65E from the maintenance tie breaker.	human	
3412012002	Fermi 2	5	04/26/2012	When the EFCV testing was released, the control room operators failed to recognize that one of the IPCS points being monitored would become invalid as part of the EFCV test. Inadequate impact evaluation for EFCV testing led to a failure to properly monitor a critical parameter.	team	cogniton
3412015007	Fermi 2	6	10/04/2015	Investigation determined that the tag out (STR) configuration established for the maintenance activity did not include closing the RR pump suction or discharge valves which resulted in the RR loop being in direct communication with the reactor coolant system.	team	control
3412015011	Fermi 2	4	9/14/2015	The cause of the event was that an Operator did not inject RCIC in a timely manner to maintain RWL above the Level 3 setpoint.	human	performance
3332006002	FitzPatrick	2	11/04/2006	A HPCI minor maintenance inspection was performed in October 2006 during refuel outage 17.The cause of not recognizing (and thus not reporting) the Mode change with HPCI inoperable is human error. The apparent cause of this error is that HPCI had passed the low-pressure operability test, and thus, was considered to be OPERABLE prior to the Mode change.	team	performance+ check
3332008001	FitzPatrick	6	09/16/2008	The cause of the event was ineffective implementation of the outage risk assessment procedure. There were no safety system functional failures. There were no nuclear, radiological or industrial safety consequences associated with this event. All systems performed as designed and there were no component or system failures.	team	performance
3332008002	FitzPatrick	6	09/23/2008		unclear	
3332008003	FitzPatrick	5	10/07/2008	The cause of the event was the re-scheduling of a trip and lockout relay functional test outside of the bus outage work window without performing a risk assessment review. Though the outage schedulers were recently counseled on the importance of performing a risk assessment for emergent work, they were not as cognizant of the requirement to perform a risk assessment on outage work activities re-scheduled outside an approved system outage work window.	team	work package+communciation
3332008004	FitzPatrick	6	09/23/2008	The apparent cause of the relay failure is installation of the wrong component for the application. While this type of relay has been widely used in this type of application, the cause evaluation determined that these relays are only marginally acceptable in this application due to instrument drift. Therefore, the cause is an original design deficiency.	team	design
3332012005	FitzPatrick	6	10/05/2012	The root cause of this event was an individual not performing the work order steps as written. The contributing causes were conflicting information between the drawing and the work order instructions; inadequate verbal communication between the project manager and the responsible engineer; and inadequate review of the engineering change in accordance with new requirements specified in revision	team	communciation+check

3332012006	FitzPatrick	5	10/16/2012	Operators monitored redundant channels for maintaining actual level. The automatic isolation function for Shut Down Cooling on low reactor water level was lost for a short period of time.	team	performance
3332012009	FitzPatrick	2	11/24/2012	…the shift notes correctly indicated that the procedure was still in progress. However, while completion of the vent and purge evolution was not formally entered into the control room log, a note that the containment had been "de-inerted" was entered, the shift turnover checklist incorrectly stated the venting and purging had been secured.	team	cognition
2852006004	Fort Calhoun	NA	09/12/2006	The analysis concluded the cause of this event was inadequate work practice to physically verify the position of SI-173. Contributing Causes 1. Lack of independent verification of the shutdown cooling system line-up represents failure to provide a barrier of defense to ensure quality verification. 2. Lack of rigorous guidance within the infrequently performed procedure indicates there is a heavy reliance on operator experience.	team	implementation+coordination&communication
2852006005	Fort Calhoun	NA	10/10/2006		team	implementation+follow-up
2852008003	Fort Calhoun	4	03/21/2008		team	cognition
2852009004	Fort Calhoun	3	11/26/2006		team	implementation
2852009005	Fort Calhoun	5	11/06/2009	This adjustment was not covered by a written instruction. The latent root cause of this event was the adjustment would have been an action not covered by any existing retrievable written instructions. The fundamental flawed assumptions and weak work practice reflected the cultural norms and values that existed in the Instrumentation and Control maintenance organization in 2006.	team	implementation
2852010001	Fort Calhoun	4	11/01/2009		team	training
2852011004	Fort Calhoun	3	02/05/2011		team	follow-up+control
2852011008	Fort Calhoun	5	06/07/2011		team	design
2852011009	Fort Calhoun	5	06/26/2011	FCS was also in a Notification of Unusual Event (NOUE), since June 6, 2011, due to high Missouri river level. River level at the time of this event was 1006 feet 6 inches mean sea level.	team	cognition
2852011010	Fort Calhoun	5	06/07/2011	The direct cause of the circuit breaker trip was improperly configured zone selective interlock jumpers on circuit breaker 1B3A.	team	procedure
2852012001	Fort Calhoun		02/10/2012	The causal analysis determined that station senior management, at the time the condition was identified, did not effectively lead recovery efforts to address the NRC component design basis inspection (CDBI) and FCS self-identified flooding issues in AOP-01. This resulted in important flooding related corrective actions not being effectively planned, prioritized or resourced to ensure a success path for AOP-01 within the established timeline.	team	
2852012002	Fort	5	03/02/2012	The Fundamental Performance Deficiencies are addressing the managerial and technical	team	performance+

	Calhoun			oversight causes.		check
2852012003	Fort Calhoun	5	03/12/2012	The apparent cause was identified to be inadequate use of vendor oversight when design information was transmitted to the vendor. The analysis also identified a contributing cause of inadequate review of the calculation provided by the vendor during the owner acceptance process.	team	design
2852012004	Fort Calhoun	5	03/29/2012	The Apparent Cause was determined to be poor vendor documentation which led to Engineering personnel to improperly interpret and apply the information contained in the Static "0" Ring vendor manual.	team	documentation
2852012005	Fort Calhoun	5	02/21/2012	The apparent cause of this event is a lack of technical rigor in the procedure change process employed in 1990s.	team	historical
2852012008	Fort Calhoun	5	09/28/2011	A cause analysis determined that a lack of management oversight and the failure of Engineering to take a proactive approach in the prevention of future test failures lead to this event.	team	management
2852012009	Fort Calhoun	5	12/13/2011	The condition identified on December 13, 2011, was initially reported via Event Notification (EN) No. 47900 as an unanalyzed condition (10 CFR 50.72(b)(3)(ii)(B)) on May 04, 2012 The initial LER submittal was made on July 23, 2012. These notifications were determined to have been made late.	team	reporting(documentation)
2852012013	Fort Calhoun	5	12/07/2011	The analysis concluded that there was inadequate/incomplete procedural guidance for developing Administrative Limits used to protect TS Limits. This includes guidance for understanding how to evaluate and apply uncertainties when developing TS Administrative Limits.	team	administration
2852012014	Fort Calhoun	5	07/11/2012	The Root Cause Analysis completed December 21, 2012, and determined the condition described in this report was due to inadequate ownership review by Omaha Public Power District of plant construction architect/engineer produced calculations.	team error	coordination &communication-ownership
2852012015	Fort Calhoun	5	09/16/2011	The causal analysis identified a number of components located in auxiliary building rooms 4, 13, 21, 22, and 81 that should have been included in the EEQ program. This omission was determined to be the result of insufficient engineering rigor by the preparer and reviewer of the EEQ Program Basis Document.	team error	cognition
2852012016	Fort Calhoun		07/17/2012		technical	
2852012017	Fort Calhoun	unclear	07/26/2012	A causal analysis was conducted and found that the station did not fully implement and or maintain the electrical equipment qualification program. This resulted in a lack of qualification documentation and equipment not qualified for expected design basis accident conditions.	team error	documentation+check
2852012018	Fort Calhoun	unclear	07/27/2012	On July 27, 2012, while performing NRC Inspection Manual Chapter 0350 checklist reviews, the Recovery Engineering Team identified that the containment air cooling and filtering system (CACFS) was not properly tested during cycle 26. It was discovered that surveillance test (ST) IC-ST-VA-0013, as written and performed, did not maintain train separation of the system components during single	team error	performance

				train surveillance testing as required by the USAR		
2852012019	Fort Calhoun	5	08/14/2012	The Apparent Cause is the process for closing the Sluice gates within OI CW-2 did not adequately account for river debris obstructions.	team error	procedure
2852012020	Fort Calhoun	5	12/02/2012	The cause has been determined to be FCS Engineering personnel failing to validate the actual plant configuration and the use of uncorroborated drawing information in completion of design basis calculations.	team error	
2852012021	Fort Calhoun	6	01/29/2012	A causal analysis is in progress and the preliminarily root cause identified failure of the station to compare FlowScan data with approved calculations and a lack of corrective actions. Repacking and testing of the valve has been planned.	team error	
2852013003	Fort Calhoun	6	01/30/2013	A preliminary causal analysis identified that the station failed to obtain vendor technical information on HPSI pump performance in a 10 CFR 50, Appendix B, QA validated format.	human error	
2852013005	Fort Calhoun	5	02/27/2013	In 1995, organizational work practices lacked technical rigor, resulting in FCS personnel incorrectly concluding that an NRC approved probability methodology for tornado missile protection could be applied to an FCS plant modification via the 50.59 process without obtaining a license amendment.	old team error	
2852013006	Fort Calhoun	5	03/02/2013	A causal analysis determined that Omaha Public Power District and its consulting engineering firm failed to specify a compatible material for the pump seals in the original construction specifications.	team error	
2852013010	Fort Calhoun		05/03/2013	A causal analysis determined that Omaha Public Power District and its consulting engineering firm failed to specify a compatible material for the pump seals in the original construction specifications.	team error	
2852013011	Fort Calhoun	5	06/13/2013	The Root Cause Analysis resulted in two causes. Fort Calhoun Station's responses to IE Bulletin 79-01B made inaccurate and simplifying assumptions, without supporting documentation, that ….CFR 50.49. Additionally, the EEQ Program has unique processes that are not integrated into the Engineering Change Process and impacts the sustainability of the EEQ Program.	team error	
2852013013	Fort Calhoun	5	10/18/2012	The current review determined that the components in question, although procured as CQE, had not been maintained as CQE. Additionally, the control loop is classified as non-CQE; therefore, the associated cables were not routed in safety related cable trays. The station was shutdown in MODE 5 when discovered.	team error	
2852013015	Fort Calhoun	5	09/23/2013	A cause evaluation was completed and determined that corrective actions in CR 2009-0687 root cause analysis (RCA) did not resolve water intrusion into Auxiliary Building rooms containing safety related equipment due to lack of technical rigor and flawed decision making.	team error	
2852013016	Fort Calhoun	unclear	11/05/2013	It was previously determined and rmotted in OPPD LERs 2012.017,end 2013-011 that FCS did not fully implement and/or maintain the electrical equipment qualification.		

2852013017	Fort Calhoun	5	10/31/2013	However, review of current design basis documents indicates weakness in following areas: • No formal evaluation of the pump motor performance at the extended flow rates and increased break horsepower (BHP) requirements was performed. • The pump curves were not revised to show the extended flow region of the pump. • The design basis calculation did not contain an updated case for the single failure of a pump …	team error	
2852013018	Fort Calhoun	5	10/28/2013	The short vulnerabilities described in this report have existed since the original design and installation of the DC ammeter circuitry at FCS.	design	
2852015001	Fort Calhoun	5	09/30/2013	The Shift Manager that approved the operability evaluation believed that the reportability aspect of the penetration had been previously reported to the NRC and that no further report was required. The Shift Manager did not confirm that the reportability had been completed under another LER. These issues were discovered during the Electrical Environmental Qualification Program Reconstitution Project. The deficiencies were discovered during extent of condition reviews.	team error	
2852015003	Fort Calhoun	unclear	04/16/2015	A cause analysis was performed and determined that thermal expansion was never considered for the containment riser supports.	human error	
2852015005	Fort Calhoun	4	07/21/2015	A design weakness resulted in the vibrations from RC-3A combined with the cantilevered pipe load causing cyclical stresses on the toe of a weld on the seal inlet pressure pipe tap.	design	
2442006004	Ginna	3	04/09/2005		team error	management
2442006005	Ginna	4	10/09/2006	• Configuration management practices, including the development and administration of the Minimal Essential Equipment List (MEEL), did not meet industry standards.	team error	scheduling
2442006006	Ginna	6	10/14/2006	Configuration management practices, including the development and administration of the Minimal Essential Equipment List (MEEL), did not meet industry standards.	team error	scheduling
2442008001	Ginna	4	05/08/2008	The Operations day shift considered the requirements of Technical Specification Limiting Condition of Operation (LCO) 3.4.6 to be met by the "B" RCS loop and the two available RHR loops. The Control Room Supervisor, however, after performance of a pre-job brief for part of the heatup procedure, mistakenly signed off on a procedural sub-step which verified two Reactor Coolant System loops available and one Reactor Coolant System loop in operation.	human error	cognition
2442009001	Ginna	5	09/16/2009	• One cause is attributed to a re-strike condition that can cause currents to be three to five times greater than locked rotor current. The re-strike condition could continue through multiple cycles. The excessive currents would have been high enough to cause the magnetic breaker to clear the high current condition. • The other cause is related to the design of the control system with interactions between AC and DC control power that resulted in a latent failure mechanism.	design error	
4162012004	Grand Gulf	4	04/28/2012		technical	

4162016001	Grand Gulf	5	03/17/2016	The apparent cause was determined to be that the Baxter Wilson to Port Gibson line does not have pilot scheme protection. The phase-to-phase fault would have cleared sooner with protective relaying.	design	
4002007004	Harris	5	10/19/2007		technical	
4002008003	Harris		08/19/2008	The root cause of this event is that the existing preventative maintenance program does not prevent age related failures of bus duct fuses. The program inspects these fuses, but does not replace them unless faults are identified.	human	procedure
4002010004	Harris	5	11/05/2010	The root cause of the event is the lack of guidance in plant procedure PLP-400, Post Maintenance Testing, for establishing PMT instructions for complex relay replacements.	team	guidance
4002010005	Harris	4	11/09/2010	The root cause is that Operating Procedure OP-111, Section 7.2.2 had incorrect steps that required the breakers for 1RH-25 and 1RH-63 be opened.	team	procedure
4002012001	Harris	5	04/21/2012		design	
4002013004	Harris	5	11/11/2013	The root cause was determined to be an inadequate understanding of the risks associated with degassing the waste gas system with inoperable analyzers.	team	cogniton+procedure+supervision
4002015005	Harris	4	10/8/2016		technical	
4002015004	Harris	5	05/04/2015		team error	
3212005003	Hatch 1	5	11/10/2005		team	procedure
3212009003	Hatch 1	3-hot shot	05/08/2009	Inadequate information on the Mark VI logic concerning the fact that a turbine trip reset occurs when the S1 core processor is rebooted caused this event.	human	procedure
3212009005	Hatch 1	cold shotd	05/15/2009	This event was caused by a personnel error resulting in the development of an inadequate procedure.	team	procedure
3212010001	Hatch 1		03/08/2010		technical	
3212012001	Hatch 1	6	02/28/2012	These concurrent deficiencies resulted in the failure of the on-shift management to account for the prior installation of the modified probe buffer card on Control Rod 22-27 RPIS when considering the removal of the tag-out for the CRD 22-27.	team	self-check+procedure
3212012002	Hatch 1		03/10/2012		technical	
3212012003	Hatch 1		03/13/2012		technical	
3212015002	Hatch 1	7% P	01/20/2015		team error	procedural
3212016001	Hatch 1	6	02/11/2016		team	cognition+procedure
3212006001	Hatch 1, Hatch 2	6	02/17/2006	This event was caused by an inadequate acceptance criteria in the surveillance test procedure. The surveillance test procedure did not account for the 12 second diesel generator start time during a Loss of Offsite Power event (LOSP).	human	cognition

3662007001	Hatch 2	6	02/12/2007	The most likely direct causes of the MSIV failures were from out-of-specification internal valve tolerances and dimensions. The problems found with valve seats that were too wide or cut at the wrong angle were considered to be procedural compliance issues.	team	performance+ procedure
3662007002	Hatch 2	6	02/18/2007		technical	
3662007003	Hatch 2		03/01/2007		technical	
3662007005	Hatch 2	2	03/15/2007	The root cause associated with the tagout event is that the drafter and the reviewer of the tagout did not adequately address the system or functional impact associated with the components that were tagged or removed from service.	team	technical
3662009001	Hatch 2	6	03/12/2009		technical	
3662009002	Hatch 2	6	03/13/2009	The cause of the feedwater outboard valve test failure (only the "A" line was affected) was misalignment caused by internal wear, and missing bearing cover lock pins (this is valve 2B21-F077A). The cause of the feedwater inboard valve test failures (the "A" and "B" lines were affected) was misalignment caused by excessive clearance between the hinge pin and the disc and, the hinge pin adjustment had changed over the operating cycle.	team	performance
3662011001	Hatch 2		04/16/2011	Maintenance personnel attributed the cause of the valve leakage for 2T48-F309 to the fact that the valve disc was not staying centered within the valve body, resulting in a gap between the disc and the seat at the top of the valve. This was caused by workmanship issues in 2009 when neither the valve vendor nor plant Maintenance personnel ensured the valve	team	performance
3662011003	Hatch 2		10/24/2011		technical	
3662013001	Hatch 2	5	02/16/2013	The unplanned RPS actuation was caused by the presence of a water level in the SDV above the trip setpoint and by a less than adequate procedure. The absence of procedure prerequisites to confirm the SDV Hi Level Rod Block and RPS trip signals were not present prior to performing the refueling interlocks surveillance procedure resulted in positioning the reactor mode switch to the "STARTUP" position with the SDV trip signal present.	team	procedure
3662013003	Hatch 2	2	03/18/2013	he vendor workers performing the evolution had marked the tubing prior to removal with permanent marker, but the markings rubbed off and became illegible. The workers mistakenly believed that all the tubing was the same, so they tried to overcome their loss of markings by ensuring that the tubing was placed between the corresponding ports on the hydraulic actuator and remote servo.	team	cognition+per formance
3662016002	Hatch 2	4	05/23/2016	The cause of the actuation of the Group I isolation signal during Turbine Testing was due to inadequate procedure usage. Poor communications between maintenance and operations personnel resulted in false assumptions that the Group I isolation trips were bypassed. Causal factors included procedure weaknesses which resulted in improper interpretation of critical steps.	team	procedure+co mmunication
3542006003	Hope Creek		04/21/2006		technical	
3542006005	Hope Creek	2	05/02/2006	The root cause has been determined to be a latent status control database (SAP) error and a procedure revision error to "Preparation for Plant Startup".	team	historical+pro cedure
3542009002	Hope Creek	unc lear	04/18/2009	These are (1) corrosion bonding of the pilot disk and seat; (2) excessive wear of internal parts; and (3) misalignment of internal parts. Disassembly of the SRVs was performed by the manufacturer at the offsite test facility (NWS Technologies) and witnessed by Hope Creek	team	technical+perf ormance

				(HC) Programs Engineering personnel.		
3542010002	Hope Creek		10/25/2010		technical	
3542010003	Hope Creek	unclear	11/01/2010	A review of available plant documents, including all applicable procedures and the original program data base, and discussions with the IST Engineer identified the apparent cause of the error in grouping as a technical rigor application deficiency that occurred at the inception of the Hope Creek relief valve program development in 1985	team	historical
3542012004	Hope Creek		05/10/2012		technical	
3542013003	Hope Creek	3	06/13/2013	The cause of the leak was determined to be a human performance deficiency in completion of work in the drywell.	team	technical+performance
3542013005	Hope Creek	5	10/18/2013	The cause of the failure of solenoid valve (S/N 481) was determined to be a manufacturer's assembly error.	manufacturer	
2472005004	Indian Point 2		12/22/2005		human	
2472006004	Indian Point 2	2	08/24/2006		team	procedure
2472006006	Indian Point 2	4	11/30/2006	The root cause was proceeding with the work and not stopping work activities after the CR reset failed and continuing the reset locally.	team	cognition
2472008002	Indian Point 2	5	03/28/2008		team	design
2472010003	Indian Point 2	5	03/13/2010		team	coordination &communication
2472012003	Indian Point 2		03/12/2012		technical	
2472014003	Indian Point 2	3	03/18/2014	Cause of the event was scheduling error.	team	scheduling
2472016002	Indian Point 2	5	03/07/2016		team	procedure
2472016003	Indian Point 2	3	03/07/2016		technical	
2472016004	Indian Point 2	not clear	03/29/2016	The direct cause was human error for failure to ensure testing was established to meet new ITS„SRs. The apparent cause of the error is indeterminate due to the time passed since TS conversion by Amendment 238 on November 21, 2003.	team	cognition
2472016005	Indian Point 2	4	03/26/2016		team	cognition+supervision
2472016006	Indian Point 2	4	03/28/2016	The most probable apparent cause was maintenance activities by either IPEC or supplemental personnel that left the SAT states links W105 and W106 open during the 2014 spring outage (2R21).	team	cognition
2862013002	Indian Point 3	4	03/04/2013	The apparent causes were an inadequate pre-job brief and inadequate procedure for Containment Entry and Egress	team	procedure

				(OAP-007, 0-RP-RWP-405) due to poor change management. The pre-job brief failed to cover the requirement to use the dual sump barrier gate access point when in Modes 1-4, nor did it address the type of fencing allowed. The brief did not specify that only steel RP fencing could be used for the RCDT.		
2862013004	Indian Point 3	6	03/14/2013	The apparent cause of the defect was OD initiated stress corrosion cracking of the stainless steel guide tube base material under the fillet weld.	team	technical
2862013005	Indian Point 3		03/27/2013	The root causes were 1) The test procedure did not contain steps to preclude a single point failure, 2) Manufacturing defect of a test equipment lead, 3) Insufficient evaluation of the risk of when the test was being performed.	team	procedure
2862015003	Indian Point 3	5	04/09/2015	The apparent cause was improper implementation of improved TS requirements.	team	
3052005004	Kewaunee	unclear	03/15/2005	Design basis documentation regarding flooding, HELB, seismic, and tornado protection lacked detail and was difficult to retrieve. This made it difficult for the plant staff to identify the actual flooding design basis requirements and determine what actions were required to maintain compliance with them. 2) Some processes related to maintaining the design basis were weak and were inconsistent with industry standards.	human	design+process
3052005012	Kewaunee	unclear	06/10/2005	The cause of this condition is inadequate consideration of component failure modes for the accident scenario of RHR supplying ICS while in containment sump recirculation. This resulted in a failure to recognize the consequences of the RHR-8A(B) valve failing open while ICS is being supplied from RHR during containment sump recirculation.	team	cognition
3052006003	Kewaunee		05/05/2006	The RHR pumps are not protected from non-seismically qualified pipe breaks in the auxiliary building. The specific design criteria is stated in the Updated Safety Analysis Report Section B.5 'Protection of Class I Items'		design
3052006004	Kewaunee		05/19/2006	The cause of the event was a failure to properly apply the EDG derating curves after they were received from the vendor.	team	follow-up+design
3052007005	Kewaunee	3	03/03/2007	This condition has existed for at least the last fourteen years and possibly since initial construction. No definitive cause has been determined.	technical	
3052007006	Kewaunee		11/24/2004		technical	
3052009008	Kewaunee		10/10/2009	The primary cause of this event was that operators did not appropriately consider RCS flow path dynamics and transient boron concentrations when applying the calculated final estimated dilution concentration to performance of the dilution activity.	team	cognition
3052009009	Kewaunee	5	10/15/2009	The tripping and lockout of the Tertiary Auxiliary Transformer, which supplies Safeguards Bus 5, was causedby an incorrectly set transformer relay input parameter. The primary cause of the incorrectly set transformer relay input parameter was the lack of clear written requirements in the design control documentation for adding relay input parameters to programmable digital devices (Basler relays).	team	
3052011002	Kewaunee	6	03/10/2011	The cause of the loss of station backfeed through the MAT was a human performance error in selecting a component from a list of similarly labeled components while using an unapproved method of testing	team	cognition
3052011003	Kewaun	3	03/24/2011	This event was due to inadequate guidance in procedure SP-33-297A, Safety Injection to	team	procedure

	ee			Loop A Cold Leg Check Valve Leakage Measurement.		
3052012004	Kewaunee	5	04/27/2012	The cause of the defect in the original socket weld is not known. Due to the repair activities performed, no definitive cause analysis could be completed.	team	technical
3732009002	LaSalle 1	5-cold	07/20/2009		technical	
3732013004	LaSalle 1	2	04/22/2013	The cause of this event was a weakness in Normal Unit Startup procedure LGP-1-1 in that there was no specific procedure step to verify RCIC operability prior to exceeding 150 psig. LGP-1-1 does contain a limitation stating that RCIC operability is required prior to exceeding 150 psig but this would have been more effective if it had been a specific procedural step.	team	procedure
3732010001	LaSalle 1, LaSalle 2	5-cold	03/03/2010	The cause of the event was that the authorized entrant did not follow the process for entering a vital door or turnstile. The individual did not enter or challenge turnstile #5. A contributing cause was that the second individual did not properly use human performance tools to validate access approval on indication at the security card reader.	team	cognition+communication
3732014001	LaSalle 1, LaSalle 2	5	02/18/2014		human	design
3522012001	Limerick 1	4	02/20/2012	The apparent cause of this event was that valve stroke times were not optimized following test failures. A contributing cause was inherent inaccuracies in valve stroke timing practices.	technical	
3522012006	Limerick 1	5-cs	07/19/2012	The cause of the event was a personnel error caused by a test weakness. The "Pre-control Rod Withdrawal Check and CRD Exercise OPCONs 3,4 With No Core Alterations" surveillance test does not have verification steps to ensure that the STC input is accurate.	team	procedure
3532007001	Limerick 2	unclear	03/10/2007	The event was caused by a failure to anticipate the system response that occurs when the backup scram valve fuses are removed.	team	cognition
3532007002	Limerick 2	5-cs	03/12/2007	This event was caused by a procedure performance error that resulted in only two of four low vacuum isolation channels being bypassed when four channels were to be bypassed by procedure.	team	procedure+communication
3532013001	Limerick 2	5-cs	04/16/2013	The automatic RPS' system actuation was caused by a failure to follow the existing procedure change processes	team	procedure+cognition
3532015002	Limerick 2		04/13/2015		technical	
3692004002	McGuire 1	3-hs	10/07/2002	1SM-3 was assembled incorrectly and accepted due to deficiencies in the procedure used to maintain, re-assemble, and test Main Steam Isolation Valves.	team	procedure
3692005002	McGuire 1	4	04/10/2004		team	communication
3692007002	McGuire 1	2	03/22/2007	Human Error resulted in the condition going undetected when the requirement to LRT the valve following manual operation was waived.	human	technical
3692007003	McGuire 1	6	04/15/2007	The causes of this event were attributed to an inadequate Operator Aid Computer (OAC) alarm response procedure, and the common alarm circuitry of the high flux at shutdown	human	technical

				alarm.		
3692008003	McGuire 1	2	10/31/2008		technical	
3692014002	McGuire 1		09/27/2014		technical	
3702009001	McGuire 2	6	09/29/2009	The Root Cause for the LTOP Technical Specification violation was how an R&R Lift for Test and a procedure interfaced such that the sequence of events was not controlled.	human	
3702011002	McGuire 2		03/30/2011	A cause evaluation determined that the control rod drive mechanism head cable for the L13 control rod, which was replaced during the refueling outage, was incorrectly wired by the vendor, and subsequent testing failed to detect the issue	team	
3702012001	McGuire 2	4	11/02/2012	The cause of this event was incorrect design basis operability and procedure guidance which allowed Operations to wrongly interpret that they could open a manual containment isolation valve in a Mode prohibited by Technical	team	design+procedure+cognition
3702014001	McGuire 2	unclear	04/03/2014	A skill-based human performance error was identified as the probable reason that the NDE performed in 2012 missed the flaw. The size and age of the flaw, as determined by a metallurgy lab analysis, indicate that it has been present for several years. The examination of this line performed in 2012 (during the previous refueling outage) was ineffective and missed the significant flaw.	team	
3702015001	McGuire 2	4	10/07/2015	The cause of this event was a procedure weakness that did not clearly define the CF pump restoration conditions after testing was complete.	Team	procedure
3362006006	Millstone 2	3	10/07/2006	The root cause investigation for this event determined that supplementary instructions provided for the construction of scaffolding in the vicinity of the MSIV lacked sufficient specific information to ensure MSIV operability was not impacted.	team	procedure
3362008002	Millstone 2	6	04/13/2008	The cause of this event was determined to be inadequate configuration control because the tag for the charging pump removed from service to comply with the Boron Dilution TS 3.1.1.3.b. did not provide adequate guidance.	team	control+guidance
3362011001	Millstone 2	5	04/03/2011	The apparent cause of this event was determined to be a design/application deficiency in the use of MSSV exhaust piping sliding bushings as an Enclosure Building boundary.	design	
3362012003	Millstone 2	5	10/15/2012	These deficiencies were historical in nature and appear to be original construction deficiencies. Upon discovery the identified deficiencies were repaired to restore the design basis for flood protection.	design	
3362006006	Millstone 2	3	10/7/2006	The root cause investigation for this event determined that supplementary instructions provided for the construction of scaffolding in the vicinity of the MSIV lacked sufficient specific information to ensure MSIV operability was not impacted.	team	procedure
4232008004	Millstone 3	5	10/20/2008	The cause of this condition was determined to be a latent design error. The original design of the plant did not include a vent path for this section of piping. Subsequent to the discovery of the gas void, the plant was modified to install a vent valve on this line to provide a venting location.	team	design+historical
4232008005	Millstone 3		11/05/2008		technical	
4232013005	Millstone	4	05/15/2013	The most likely cause of the leaking equalizing valve on the MPS3 outer containment door	team	performance

	e 3			was personnel error in that the equalizing valve was most likely inadvertently bumped by personnel in transit causing it to be slightly open.		
4232016002	Millstone 3	3	01/25/2016	This was a human performance error that occurred during refueling outage 3R16 as part of a plant modification (completed in November 2014) which replaced all four Unit 3 FWIV actuators. Site personnel failed to follow the requirements of the work control process and station procedures when installing and removing the temporary jumper in the feedwater control circuit for 3FWS*CTV41C.	team	performance
2632007002	Monticello	6	03/17/2007	On March 17, 2007 at 2346, with the unit in MODE 5 (Refueling) and on Division I of shutdown cooling, Operators were implementing a clearance order isolation in support of a modification to control room metering for Bus 16.	team error	coordination &communication+ownership
2632007003	Monticello	5	04/20/2007		team error	cognition
2632008006	Monticello		09/17/2008		technical	
2632008007	Monticello	5	09/20/2008	The cause of the event was the Operating procedure for starting the CRD pumps, did not include steps to ensure the Reference Leg Backfill System was isolated prior to starting the system and pump. A Contributing Cause of the event was inadequate communication between crew leads and Shift Supervision when difficulties were encountered while performing the Shutdown procedure, and associated Shutdown Checklist.	team error	coordination+communication
2632009001	Monticello	6	04/02/2009		team error	cognition+follow-up
2632009002	Monticello	6	04/02/2009		team error	cognition
2632010006	Monticello	5	11/22/2010		team error	cognition+follow-up
2632011010	Monticello	5	11/27/2011		multi-team error	cognition+briefing
2632011011	Monticello	5	12/01/2011		team	performance+procedure
2632013002	Monticello	6	05/24/2013		human	safety
2632013004	Monticello	5	06/13/2013		team	cognition+documentation
2632015002	Monticello	6	05/02/2015		team	procedure
2632015003	Monticello	6	05/14/2015		team	procedural
2202006002	Nine Mile Point 1	2	06/12/2006		team	performance

2202011002	Nine Mile Point 1		05/06/2011		technical	
2202012007	Nine Mile Point 1	5	11/06/2012		team error	cognition
2202013001	Nine Mile Point 1	2	05/14/2013		technical	
4102012002	Nine Mile Point 2	5	05/25/2012		team	
4102014007	Nine Mile Point 2	5	04/02/2014	This event was caused by the simultaneous opening of Airlock Doors R261-1 and R261-2 by workers as they passed through the doors.	team	performance
3382006001	North Anna 1	3	04/07/2006	The manual reactor trip was the result of having a mismatch in group step counters. This was the result of the Shutdown Bank "A" Group 2 step counter stopping at 215 steps while Group 1 and individual rod position indicators showed continued motion to 225 steps. The failed digital step counter for Shutdown Bank "A" Group 2 was examined upon removal. Internal connections were noted to be inadequate resulting in intermittent operation.	team	peformance
3382010004	North Anna 1	2	10/22/2010	The direct cause of the malfunctioning rod control in-hold-out switch was attributed to dirt on the push buttons of the switch which resulted in sluggish operation.	team	technical
3382013001	North Anna 1	6	09/26/2013	The direct cause for the complete loss of electrical load of the 1H EDG during 24-hour testing was the momentary loss of electrical control power to the DRU. The apparent cause for the momentary loss of electrical control power to the DRU was a loose fuse holder due to numerous removal and re-installation evolutions during design change implementation, testing, and troubleshooting. This lead to the spreading of the fuse clips.	human	technical
3392008001	North Anna 2	3	02/08/2008	No definitive root cause has been identified for the Urgent Failure that caused the Rod Control Group Step Counter deviation.	not identified	
3392008002	North Anna 2	3	10/18/2008	The cause of this event was determined to be inadequate management of emerging equipment issues. It was not noted that, with 2J EDG tagged out, redundant train equipment was required to be maintained operable, or a four (4) hour action would need to be entered per TS 3.8.1.	team	management
3392011001	North Anna 2	3	09/28/2011	The direct cause of manually tripping the 2H EDG was a failed gasket causing a coolant leak. The cause of the coolant leak was insufficient procedural guidance for gasket installation. Maintenance procedures did not provide adequate level of detailed instructions on proper installation of the gasket between the exhaust belt and the coolant inlet bypass fitting.	team	procedure
2692006005	Oconee 1	3	12/09/2006		team	design+procedural
2692006003	Oconee 1, 2, 3	?	06/01/2006		team	cognition+procedure
2692011004	Oconee	5	04/04/2011	One cause is that ONS personnel had an incorrect	team	cognition+im

	1, 2, 3			interpretation of TS 3.4.15 requirements and Regulatory Guide 1.45 recommendations.	error	plementation
2692014002	Oconee 1, 2, 3	6	11/25/2014		team	calcualtion/m odel
2872006001	Oconee 3	6	05/15/2006		team	design
2192006001	Oyster Creek	4	05/06/2006	Oyster Creek was shutting down to start a Forced Outage (1F1 0) to repair a leak from the Steam Packing Exhauster Cooling Condenser	human error	
2192006002	Oyster Creek	6	06/20/2006	The cause of Oyster Creek Generating Station starting up from an outage and running with an MSIV inoperable is a transcription error during the performance of the PPC timing section of the surveillance test in February 2006.	team error	
2192006003	Oyster Creek	5	10/16/2006		technical	
2192007002	Oyster Creek	2	07/20/2007		technical	
2192009002	Oyster Creek	4	02/02/2009	Operations did not recognize the significance of the safety-related and non-safety related shared logic component interface of V-16-14.	team error	cognition
2192009004	Oyster Creek	6	10/25/2008	Temporary Modification was made to the secondary containment so that the trunnion room door could be left open to support maintenance activities.	team	cognition
2192010002	Oyster Creek	2	12/23/2010	On December 23, 2010, a reactor startup was in progress in accordance with Procedure 201, Plant Startup. The reactor was critical with the mechanical vacuum pump operating to draw vacuum on the main condenser.	human	procedure
2192011001	Oyster Creek	5	05/04/2011	The cause of the incorrect Input coefficients was a database error for 10x10 fuel bundles.	technical	
2192012002	Oyster Creek	5	10/29/2012	A detailed investigation was performed and found that the root cause of this event was a wall falling in the switchyard causing a ground fault on the 34.5KV system	team	documentatio n
2192012003	Oyster Creek	5	11/07/2012		technical	
2192012004	Oyster Creek	5	10/29/2012	The SM did not adhere to procedural allidanoe and lacked attention to dotal when he did not require the Mellon to enter tie required TS action statement of 3.17.8. The SM allowed himself to be caught up in mule-tasking *ming the IR review process. The SM did not obtain a peer check for TS operability deimmination of (I) MCA HVAC being inoperable.	team	cognition
2192013001	Oyster Creek	2	10/03/2013		technical	
2192013003	Oyster Creek	5	11/17/2013		human	cognition
2192013005	Oyster Creek	5	12/17/2013	The jumpers required to prevent a full SCRAM for this Mode Switch change were not installed as required by procedure.	team	cognition
2192014004	Oyster Creek	5	09/18/2014		technical	

2192014005	Oyster Creek	5	09/19/2014	The Apparent Cause of this event was inadequate signage on the Trunnion Room door and drain covers.	team	cognition
2192016003	Oyster Creek	2	04/30/2016		team	cognition
2552006007	Palisades	2	11/03/2006		team	cognition+procedure
2552007007	Palisades	not clear	10/01/2007		team error	coordination &communication
2552012001	Palisades	3	08/12/2012		technical	
2552014002	Palisades	6	01/29/2014		technical	
5282005007	Palo Verde 1	3	10/04/2002	The direct cause of this incident was a failure on the part of assigned technicians to remove the screwdriver following the performance of 39MT-9ZZ02, "	team	performance
5282005009	Palo Verde 1	3	12/21/2005	The apparent cause of the vibration is a latent design deficiency which allowed the shutdown cooling suction line to form an acoustic resonator that produced the high vibration.	design	technical
5282006001	Palo Verde 1	4	03/20/2006	The root cause of both events was attributed to human error in that operational fundamentals were not consistently applied for controlling and monitoring plant parameters to ensure compliance with license conditions.	team	communication+coordination+cognition
5282006002	Palo Verde 1	3	03/21/2006	The root cause of the event was attributed to human error in that operational fundamentals were not consistently applied for controlling and monitoring plant parameters to ensure compliance with license conditions.	team	design
5282006001	Palo Verde 1	4	3/20/2006	The root cause of both events was attributed to human error in that operational fundamentals were not consistently applied for controlling and monitoring plant parameters to ensure compliance with license conditions.	team	performance+procedure
5282006005	Palo Verde 1		10/05/2006		technical	
5282007001	Palo Verde 1		07/05/2007		technical	
5282010003	Palo Verde 1		05/02/2010	The investigation found that post-maintenance test procedures were not followed to ensure proper hatch motion after maintenance on both the east and west hatch hoist upper limit switches.	team	performance+equiment
5282011005	Palo Verde 1	2	11/22/2011	The root cause was determined to be latent organizational weaknesses with the modification and corrective action processes that delayed installation of automatic CEDM timer modules (ACTMs) which would minimize the occurrence of dropped or slipped CEAs.	team	admin
5282016001	Palo Verde 1		04/11/2016		technical	
5282007002	Palo Verde 1, 2, 3		07/10/2007	The root cause of failure investigation is not complete	unclear	
5282007003	Palo Verde 1,	3	07/13/2007	The direct cause of the event was an inadequate STP in that the verification of the position of the solenoid valves was not included in the STP. Before the implementation of the ITS,	human	procedure

	2, 3			the requirement for surveillance in 40ST-9AF07, Rev. 1, was to "Ensure that each valve (manual, power operated, or automatic) in the flow path from the CST		
5282010002	Palo Verde 1, 2, 3	5	05/07/2010	The investigation found that administrative barriers were unsuccessful in preventing the calculation error due to ineffective reviews and the lack of a questioning attitude.	team	admin+cognition+performance
5292003003	Palo Verde 2	6	11/23/2003	The root cause associated with the violation of Technical Specification 3.9.2 was that personnel involved made an incorrect operability decision. The initial limited evaluation performed by maintenance personnel only confirmed the instrument drawer was working properly, but provided no confirmation that the pre-amp or the detector was also functioning properly.	team	cognition+communication
5292005006	Palo Verde 2		05/15/2005	A contributing cause was the ineffective implementation of equipment trending and management of work priorities such that the combination of these failures jeopardized the redundancy of the EDG fuel oil filters and strainers.	team	technical+performance
5292006004	Palo Verde 2		07/17/2006		technical	
5292006005	Palo Verde 2		10/07/2006		technical	
5292011001	Palo Verde 2	6	04/08/2011	• A latent organizational weakness existed in the reinforcement of Operations expectations for Technical Specification Decision Making, which allowed the Technical Specification decision to be made without consulting the Technical Specification Bases • Inadequate guidance to facilitate meeting the requirements of LCO 3.3.9, Required Action C.1 in that there is inconsistent terminology relative to Control Room ventilation modes of operation among the LCO, LCO bases, and procedures. • An operator knowledge deficiency exists in the area of the Control Room ventilation system and related Technical Specifications.	team	management+guidance+cognition
5292012002	Palo Verde 2	5	10/07/2012	The cause was determined to be inadequate guidance to ensure temporary fittings on safety-related fluid systems were removed prior to placing the system in service.	team	procedure
5292012003	Palo Verde 2	3	11/02/2012	The cause of this event was determined to be inadequate work instructions. The work instructions did not provide detailed guidance for installing an angled bonnet.	team	guidance+procedure
5302006003	Palo Verde 3	5	04/02/2006	The LER reports an actuation of the B train emergency diesel generator due to a loss of power to one class bus (B train 4.16 KV) caused by human error during testing.	human	
5302006004	Palo Verde 3	5	05/06/2006	The root cause for the event was the test procedure was insufficient and the area operator didn't have the knowledge base to compensate for the procedure inadequacies.	human	procedure
5302007001	Palo Verde 3	5	04/21/2007	The cause of the Unit 3 CS system nozzles blockage was from inadequate consideration of the consequences of overfilling the CS system with borated water.	human	procedure
5302010001	Palo Verde 3	6	10/07/2010	The cause of the Pzr Aux Spray valve failure to operate from the local position at the RSP was a missing wire in the control circuit in the associated auxiliary relay cabinet. The investigation revealed inconsistencies between the as-built condition, and control wiring diagrams (CWD) and supplier documents (SDOC).	team	performance
5302012001	Palo Verde 3	2	04/15/2012	The root cause of the manual reactor trip was the LPPT procedure did not provide contingency direction to insert other CEA groups to compensate for the RCS dilution. A contributing	team	procedure

				cause was the ACTM modification impacts on the operation of CEDMCS were not identified during the design phase.		
5302013001	Palo Verde 3		10/07/2013		technical	
5302015002	Palo Verde 3	4	05/01/2015	The event resulted from improperly installed sealing rings on the valve internals which created excessive friction between internal valve components when the valve was stroked with steam pressure applied.	team	performance
5302015004	Palo Verde 3	4	05/01/2015	The cause of the failure was lack of adequate guidance to perform a walk down during the DEC process. The lack of a local inspection of actual plant conditions resulted in the latent condition (i.e., vibratory displacement of the affected air-line combined with inadequate tubing support) remaining unnoticed prior to the component failure.	team	admin+guidance
2772006001	Peach Bottom 2	5	09/22/2006		team	technical+control
2772006002	Peach Bottom 2		09/28/2006		technical	
2772010003	Peach Bottom 2	5	09/27/2010		technical	
2772014003	Peach Bottom 2		10/29/2014		technical	
2772010002	Peach Bottom 2, Peach Bottom 3	5	09/18/2010	The underlying cause was determined to be due to an inadequate design drawing.	design	
2772010004	Peach Bottom 2, Peach Bottom 3		09/29/2010	This was based on the incorrect assumption that this credit had been found acceptable as part of the licensing associated with Technical Specification	human	cognition
2782009001	Peach Bottom 3		01/21/2009		tehnical	
2782009002	Peach Bottom 3	2	01/26/2009		human error	
2782009007	Peach Bottom 3	3	09/14/2009		team	procedural
2782011001	Peach		09/21/2011		tehnical	

	Bottom 3					
2782011003	Peach Bottom 3		09/25/2011		tehnical	
4402007002	Perry	5	07/11/2007		technical	
4402009002	Perry	unc lear	06/22/2009	The root cause was determined to be an inadequate post modification test which failed to identify the miswiring of two output wires from the diesel generator CO2 Fire Suppression System control panel.	team	cognition+lea dership
4402010001	Perry	6	04/27/2009	The individual human performance deficiencies were addressed in accordance with the company's performance management process. Procedure and program changes were made to address weaknesses in risk perception, use of jumpers and lifted leads, and work on protected equipment.	team	management+ performance(organizational)
4402011003	Perry	2	10/18/2011	This is a latent organizational weakness that had not previously manifested itself due to the working relationship between the Perry switchyard coordinator and operations.	team	management (org)
4402013002	Perry	6	03/25/2013	The cause of the event was a failure to follow procedure in step-by-step sequence.	team	cognition
4402016003	Perry	5	02/11/2016	The failure analysis revealed that the fuse internals were not soldered correctly during the manufacturing process. One of the fuse elements to fuse ferrule connections had flux applied but no solder.	team	technical
2932007003	Pilgrim		04/26/2007		technical	
2932008007	Pilgrim	4	12/20/2008	An eyewitness account by personnel performing an infrared thermographic survey of the ACB-102 disconnect switch in the switchyard observed a flashover at the ACB-102 Line 355 phase B arc horn. This was caused by accumulated snow falling from the overhead bus and bridging the gap to the arc horn, initiating the flashover.	human	
2932011002	Pilgrim	2	02/20/2011	The Root Cause of the event was a failed opportunity to capture and up-date the reactor cooldown procedure with relevant historical pilgrim Operating Experience regarding previously attempted cooldown evolutions using the MPR.	team	
2932011003	Pilgrim	2	05/10/2011	The root cause of this event was determined to be the failure to adhere to established standards and expectations due to a lack of consistent supervisory andmanagement enforcement. Investigation into the event revealed several examples of inconsistent enforcement of administrative procedure requirements and management expectations for command and control, roles and responsibilities, reactivity manipulations, clear communications, proper briefings, and proper turnovers. Three contributing causes were identified. Two contributors to this event were related to distinct aspects of operator fundamentals and the other was related to latent procedure weaknesses. Contributing causes to this event were that operators became overly focused an a single indication and parameter versus reviewing diverse indications, did not display proficiency during the reactor startup in the control of plant parameters and having the correct picture of a successful startup. Additionally, operating procedures lacked detailed guidance for operations from criticality through the IRM heat up range.	team	
2932013005	Pilgrim	4	05/23/2013	The apparent cause of the leak in the HPCI turbine exhaust line was failure to adequately	technical	

				tighten all of the flange bolting due to unique bolting and flange configuration associated with the new check valve and butterfly valve installation during the refueling outage (RFO)-19.		
2932013010	Pilgrim		10/19/2013	The direct cause of MSIV isolation was automatic actuation of the Group I containment isolation signal due to high reactor high water level with the mode switch in START UP. The cause of the high reactor water level was due to unexpected rapid opening of three turbine steam bypass valves.	technical	
2932015004	Pilgrim		04/23/2015	The root cause of this event was not determined because the relay was disposed of prior to any failure analysis being performed.	unknown	
2932015005	Pilgrim		05/22/2015	The cause of the event is that the design limitations of the Offgas System were exceeded by the unusual combination of plant conditions resulting in degrading main condenser vacuum.	design	
2662013002	Point Beach 1	4	04/14/2013		team	procedure
2662016002	Point Beach 1		04/01/2016		human error	
2662005006	Point Beach 1, 2	?	11/08/2005		team	cognition+calculation
3012010001	Point Beach 2	2	06/19/2010		team	calculation
3012010004	Point Beach 2		12/15/2010		technical	
3012011004	Point Beach 2	5	06/13/2011		team error	
3012015005	Point Beach 2	3	10/29/2015		procedural	
3012011003	Point Beach 2, Point Beach 1	2	04/08/2011		technical	
2822007004	Prairie Island 1	3	12/21/2007	Age related degradation of the input/output cards of the load sequencer (Equipment Root Cause), and • No preventive maintenance strategy established for the load sequencers and their subcomponents (Organizational Root Cause).	team	org.
2822008003	Prairie Island 1	4	07/31/2008	the root cause was a failure of the site to adequately control components	team	control
2822009007	Prairie Island 1		09/13/2009			
2822009006	Prairie Island 1, Prairie Island 2	3	10/20/2009		team	technical

2822009007	Prairie Island 2	5	9/13/2009		team	procedure
3062008002	Prairie Island 2	4	10/30/2008	• The mispositioned valve was not labeled, bypassing barriers to make identification more likely. • The procedure for safeguards hold cards and component blocking or locking contains a definition of what components should not be controlled, but does not contain a definition of which components should be controlled per the safeguards hold card program. • The mispositioned valve was not locked in the required position, making mispositioning more likely.	human error	
3062010003	Prairie Island 2	3	05/21/2010		design	
3062014002	Prairie Island 2	5	05/19/2014	The Root Cause evaluation determined that the maintenance procedures and work plans for installation of Containment FCU Head Face Flange gaskets and inlet/outlet piping spacers/gaskets do not meet station planning standards for critical component maintenance and repair to prevent leakage through all ranges of operation.	team	procedural+ design
3062015001	Prairie Island 2		03/07/2015		technical	
2542006003	Quad Cities 1	5	05/14/2006	The Operations crew believed that the previous crew had opened these switches. The previous crew had opened two of the switches, but had not realized that the third switch was required to be opened.	team error	cognition+coordination&communication
2542009003	Quad Cities 1	2	05/30/2009		technical	
2542013002	Quad Cities 1	4	03/11/2013		technical	
2542013003	Quad Cities 1	4	03/26/2013	The most probable cause of the leakage was determined to be a defect in the socket weld attributed to porosity and/or slag, originating from initial construction in 1970, prior to the initial start-up of Unit 1.	technical	design
2542015003	Quad Cities 1	5	03/02/2015		technical	
2542015004	Quad Cities 1	5	03/21/2015		team	procedure+control
2652012001	Quad Cities 2	5	03/19/2012		technical	design
2652016001	Quad Cities 2	5	03/21/2016		design	
4582008001	River Bend	unclear	01/08/2008	The investigation of this event found that relevant technical information had been inadvertently omitted from the work package.	team	org.
4582015002	River Bend	unclear	03/07/2015	The apparent cause of this event was inadequate work practices on the part of the electricians, in that they did not take all available precautions prior to performing the voltage check. The workers recognized the adverse conditions, but did not recognize the need to put into place any robust barriers. The electricians' successful past performance of this type of task likely led to overconfidence.	team	org.
4582016005	River Bend	5	02/24/2016	The first direct cause of the event is that station personnel failed to recognize the breaker vulnerability to this failure mode. As a result, an additional population of breakers that had a	team	cognition+performance+pro

				time-delayed de-energization of the closing relay was identified in February 2016.		cedure
4582016004	River Bend	5	02/24/2016		technical	
2612010003	Robinson 2	6	04/29/2010	The cause of this condition has been determined to be deficiencies in the original plant design and system operating guidance provided by Westinghouse.	design	team+procedure
2612010004	Robinson 2	6	04/26/2010		team error	procedure
2612010005	Robinson 2	5	06/24/2010	The cause of this event has been determined to be the age related failure of an inverter component resulting from the inappropriate closure of a partially completed Preventive Maintenance associated work order.	team error	follow-up
2612012002	Robinson 2	4	03/16/2012		team error	implementation
2612013002	Robinson 2	2	11/05/2013		human error	cognition
2722007001	Salem 1	3	03/27/2007		team	perfromance+ procedure
2722008001	Salem 1	5	11/05/2008	The engineer who developed the test plan and the engineer who performed the independent review of the test plan failed to identify that depressing the B1…	team error	cognition
3112006005	Salem 2	5	10/29/2006		team error	
3112009002	Salem 2	6	10/17/2009		human error	
3112011002	Salem 2	5	04/11/2011		technical	
3112012004	Salem 2	6	11/04/2012		human error	
3612006001	San Onofre 2	5	01/11/2006		technical	
3612009003	San Onofre 2		09/29/2009		technical	
3612009005	San Onofre 2	5C old	02/03/2009	The wiring error described above was introduced during the installation of an upgraded electronic governor and Digital Reference Units (DRU) for the Unit 2 Train A EDG in September 2004.	team	historical+procedure
3612010001	San Onofre 2		01/26/2010		technical	
3612006004	San Onofre 2, San Onofre	4	03/24/2006	This event occurred due to an inadequate assessment of program impacts when moving from outage to on-line surveillance testing. The implementation plan did not have the formality and rigor that would have led to the inclusion of a review by personnel cognizant of surveillance scheduling prior to implementation.	team	control

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3612008005	San Onofre 2, San Onofre 3	3	06/09/2008	This event was caused by (1) a lack of detail in the applicable procedure and (2) lack of oversight by the Control Room Supervisor (Utility, Licensed).	team	procedure+supervision
3622004002	San Onofre 3	3-hot standby	06/06/2004	This event resulted from a miscommunication between Maintenance personnel during a shift turnover. During the day shift, Maintenance personnel found they could not complete the channel calibration procedure (required to return the log channel to Operable status) because the available test equipment did not meet all the acceptance criteria. Because the calibration procedure could not be completed,	team	communication
3622006003	San Onofre 3	4-hot shutdown	03/29/2006	This could have resulted from (1) the threaded holes not being machined deep enough, and/or (2) insufficient thread lubrication causing the bolts to indicate the required torque values prior to fully compressing the seal. SCE confirmed the MNSA was installed in accordance with the vendor's procedure; however, the procedure did not require verification of seal compression by direct measurement when this MNSA was installed.	team	performance+procedure
3622008003	San Onofre 3	4	11/30/2008	SCE cause evaluation concluded that this event occurred due to the lack of a formal process to track and ensure that battery related 7-day and 31-day actions are completed on time. Instead, SCE was relying on the Maintenance personnel to track these required actions. This was not a robust tracking process and was vulnerable to a single point error.	team	supervision
3622011001	San Onofre 3	5	01/25/2011	The administrative controls for tracking entry to the LCO were not activated as required.	team	administration
3622007004	San Onofre 3, San Onofre 2	5	10/18/2007		technical	
4432009001	Seabrook	4	10/01/2009	The root cause of the event was attributed to liberal expectations and standards for implementing procedures that direct major plant evolutions, which led to the operating crew's failure to assess potential risks associated with performing a plant cooldown without placing the residual heat removal system in service.	team	control
4432014002	Seabrook	6	04/06/2014	The root cause was determined to be the revisions of the Design Control Manual in 1991 and earlier did not require a failure modes and effects analysis as part of the design change packages that installed the E7022PA relays.	team	procedure
4432016002	Seabrook	3	03/02/2016	An analysis of this event found the cause to be the on-shift operating crew was processing multiple procedure sections in parallel resulting in steps being performed out of sequence.	team	performance
3272013004	Sequoyah 1	6	10/21/2013	A. The cause of each component, system failure or personnel error, if known: The failure to maintain the penetration closed during movement of irradiated fuel was the lack of positive control of the valves for the lines passing through the	team	control+procedure

				penetration. B. The cause(s) and circumstances for each human performance related root cause: The root cause of this event was determined to be ineffective procedures for controlling penetration breaches during Modes 5 and 6.		
3272016001	Sequoyah 1	3	02/09/2016	A. The cause of each component or system failure or personnel error,	team	cognition
3282008002	Sequoyah 2	3	11/09/2008		technical	
3282011001	Sequoyah 2	2	06/22/2011	The root cause for this event has determined that no robust physical barriers were used to prevent the inadvertent adjustment of the power range neutron low range bistable. The direct cause of this event was a failure of Maintenance personnel to perform the maintenance procedure as written.	team	procedure
4992008001	South Texas 2	6	10/16/2008	The root cause of the error was that both the fuel transfer form preparer and verifier performed inadequate self-checking and review. One contributing factor was a lack of robust guidelines for planning and verification of the SFP configuration and its compliance with Technical Specification requirements which contributed to the human performance error.	team	check+performance+org
4992008002	South Texas 2	5	10/25/2008	The causes of the event were: (1) the outage process did not have specific requirements for processing emergent work or evaluating risk impact on plant conditions; (2) the work planning guidance lacked sufficient rigor; and (3) the work package was complex.	team	procedureplaning/guidance+work package
4992008003	South Texas 2	6	10/14/2008	The root cause of the event was an inadequate calibration procedure. The procedure did not require a functionality check of the internal switch contacts after switch calibration restoration.	team	procedure
4992010006	South Texas 2	3	11/10/2010	The root cause was determined to be a lack procedural guidance. The applicable procedure is being revised to ensure monitoring for indications of excessive leakage prior to entry into Mode 4.	team	procedure
4992011003	South Texas 2	3	11/30/2011	The Cause of the event was determined to involve the revision of the associated maintenance work activity's Preventive Maintenance Instruction (PMI). Specifically, the MODE requirement prerequisites in the PMI were revised without full considerationof the Operational restrictions associated with changing plant conditions during procedure performance.	team	procedure
4992011001	South Texas 2, 1	5	4/30/2011	In 2003, Technical Specifications 3.4.1.4.2 and 3.9.1 were revised to remove references to specific valves that were required to be secured to protect against dilution and replaced with the more generic language discussed in Section I.D above. However, the potential consequences of using the RHT as a fill source was not adequately addressed with respect to compliance with the revised TS requirements, and thus the surveillance procedure used to ensure compliance did not address all potential unborated water sources.	team	procedure
3352010007	St. Lucie 1	3	06/25/2010		technical	
3352012001	St. Lucie 1	5	02/10/2012	A root cause evaluation (RCE) concluded that plant personnel did not fully understand or question the risk significance associated with unprotected relays in the immediate work environment.	team	cognition
3352012002	St. Lucie 1	2	03/18/2012	The root cause of this event was determined to be inadequate design control of a modification which installed new automatic control timing modules (ACTM) with deficient 10 volt regulators that introduced signal noise that resulted in system malfunction.	design	

3352012007	St. Lucie 1	3	04/02/2012	The failure was attributed to procedural inadequacies during the fill and vent of the EDG following maintenance performed during the SL1-24 refueling outage. A contributing factor included inadequate chemistry procedures.	team	procedure+cognition
3352016002	St. Lucie 1	5	8/5/2016	The internal leakage past V3217 was caused by inadequate contractor maintenance practices and procedures that did not ensure V3217 was within acceptable tolerances and correctly assembled. It was identified that maintenance performed in 2013 incorrectly assembled internal valve bushings, and bushing spacers were not installed per the design.	team	performance+procedure
3892006003	St. Lucie 2	6	05/06/2006	The cause for the incomplete inspections is that the data acquisition equipment operator did not recognize that the tube end was incorrectly located and failed to correct the situation. The error occurred only during special interest rotating probe inspection of bobbin indications, and resulted in faulty input to the axial encoder, which determines axial position within the tube. The error appears to be a human performance error.	team	check+cognition
3892012003	St. Lucie 2	unclear	10/07/2012	An apparent cause evaluation concluded that plant personnel did not adequately assess the risk significance associated with exposed differential relay terminals in the immediate work environment.	team	cognition
3892015002	St. Lucie 2	5	09/17/2015	The root cause of the electrical fault was that the protective boots for a bus bar bolted connection, at a vertical riser section, were not installed properly from initial plant construction (legacy human performance error).	team	
3892015003	St. Lucie 2	5	10/19/2015	The limit switch miss-wiring is considered a legacy maintenance human performance event. FPL is investigating potential weaknesses in the post-maintenance test procedure as a potential contributor to this latent failure.	team	performance
3952005002	Summer	3	05/28/2005	This event is attributed to procedural weaknesses in conducting multiple tasks concurrently for plant startup influenced by insufficient administrative controls.	team	procedural+administration
3952008003	Summer	5	05/21/2008	The results of this analysis indicate that the root causes of this event are due to inadequate procedural guidance and training on operation of the voltage regulator in manual control.	team	procedural+training
3952013002	Summer	6	10/31/2012		technical	
3952014002	Summer		04/18/2014		technical	
2812009001	Surry 2		11/29/2009		technical	
2812011003	Surry 2		05/30/2011		technical	
3882009002	Susquehanna 2	5	04/28/2009	The cause of the event was human performance error because station personnel did not follow procedure NDAP-QA-0302, System Status and Equipment Control.	human	performance
3882011002	Susquehanna 2	2	06/27/2011	• Procedure content was less than adequate • The System Monitoring and Health Reporting Program failed to maintain equipment reliability -	team	procedure+control
3882013002	Susquehanna 2	6	05/06/2013		technical	
3882015004	Susquehanna 2		04/11/2015		technical	
2892013001	Three Mile Island 1		11/07/2013		technical	
2502006001	Turkey	6	03/12/2006	The apparent cause of the event was that the shift manager failed to verify communications	team	coordination

	Point 3			were reestablished or have it performed. Additionally, the procedure used to lift the internals by the containment crew did not require verification of communications prior to the lift.	error	&communication
2502006004	Turkey Point 3	5	03/08/2006	The cause of the event was a vendor human error in the configuration of the auxiliary switch contacts on the 3C 480V LC feeder breaker (30302) which went undetected by the vendor test and inspection program and Turkey Point pre-installation checks.	human error	
2502006005	Turkey Point 3	5	03/08/2006	An incorrect plant procedure was used for grounding the startup transformers.	human error	
2502008002	Turkey Point 3	3	03/01/2008	Subsequent investigation determined corrosion in the actuator bearing surfaces may have created friction that prevented the actuator from delivering the seating torque needed to provide an adequate seal.	technical	
2502009001	Turkey Point 3	6	04/01/2009		team	procedure
2502009002	Turkey Point 3	2	05/04/2009		team error	coordination &communication
2502012002	Turkey Point 3	5	06/25/2012		team error	coordination &communication
2502012003	Turkey Point 3	2	08/25/2012		team error	follow-up
2502012004	Turkey Point 3	2	09/06/2012	The post maintenance test (PMT) did not provide for a positive method of tubing orientation verification after replacement.	team error	coordination &communication
2502013004	Turkey Point 3	5	02/27/2013	1) Reach rod universal joint connection failed as a result of failure to complete final installation steps at that location, and 2) The 3-990 valve was not verified to be closed locally.	team error	procedure+coordination&communication
2502014003	Turkey Point 3	3	04/23/2014		team error	procedure
2502010004	Turkey Point 3, Turkey Point 4	6	10/01/2010	The latent design deficiency associated with the sample transport system was determined to be inadequate design change provided by the vendor and inadequate design verification and functional testing performed by Florida Power & Light Company (FPL), and the vendor.	team error	design
2512005006	Turkey Point 4	3	11/11/2005		technical	
2512006001	Turkey Point 4	5	11/28/2006		team error	follow-up
2512006002	Turkey Point 4	3	12/04/2006		team error	management
2512008002	Turkey Point 4	3	05/05/2008	The Shift Manager signed that the valve alignment had been completed following statement to that effect by the Outage Control Center Coordinator (a previously SRO-licensed individual). However, the alignment had not been completed.	team error	follow-up
2512010001	Turkey Point 4	3	11/26/2009		technical	

2512013001	Turkey Point 4	5	03/23/2013		team error	procedure
2512015001	Turkey Point 4	3	11/30/2014		team error	training
2712013001	Vermont Yankee	5	03/19/2013		team error	implementation+follow-up
4242006004	Vogtle 1	3	10/28/2006	The primary cause of this event was inadequate identification of inoperable equipment on the main control board which led the operator to use inaccurate level indication to fill the accumulator.	team	cognition
4242008001	Vogtle 1	3	04/21/2008	The primary cause of this event was inadequate work instructions to perform the necessary pressurizer heater work. The functional testing requirement to complete heater resistance measurements was not properly planned in the work order to ensure that the equipment was properly tested before being placed in service. In addition, human performance tools were not used by the individuals involved in the manipulation of the breakers when it was determined the instructions were not adequate.	team	work package+planning+cognition
4252014003	Vogtle 2	3	10/12/2014	The cause of the event was incorrect setup of the Rod Bank Overlap unit, which resulted in the control rods inserting out of sequence. Human error in conjunction with procedural weakness was the cause of the incorrect setup of the Rod Bank Overlap unit.	team	procedure
4252015002	Vogtle 2	3	03/14/2015	The AFW actuation occurred as a result of human error. The apparent cause of this event is deviation from standards by the operating crew to review the precautions and limitations of the test procedure, as required. Since the unit was in a post-trip recovery condition instead of the expected conditions associated with a normal unit shutdown, the initial conditions of the test procedure were not established and not recognized by the operating crew and performance of the test procedure resulted in the B-train AFW actuation.	team	cognition
3822005002	Waterford 3	3	06/09/2005	The cause of this event is a historical error in the original configuration of TS 3.4.5.1. The CFC condensate flow switches were incorrectly included in TS 3.4.5.1. A contributing cause of this event is that plant personnel did not have an adequate understanding of R.G. 1.45 requirements associated with CFC condensate flow leakage detection monitoring.	team	historical+cognition
3822008002	Waterford 3	6	05/23/2008	Apparent Cause Evaluation established the condition occurred due to procedural inadequacy as the procedures did not provide sufficient detail to assure proper manipulations of valves and venting of sensing lines after installation of the SG narrow range transmitters.	team	procedure
3822008003	Waterford 3	6	05/18/2008	The Apparent Cause established that personnel failed to comply with procedure RF-005-002.	team	communication+procedure
3822008004	Waterford 3	4	09/02/2008	The root cause analysis identified the cause of the condition as a failure to maintain plant equipment status control due to a lack of specific work instructions and a lack of work order documentation of intercell connectors that were loosened or removed.	team	work package
3822009001	Waterford 3	3	05/30/2008	The apparent cause for the plant entering Mode 3 when the EFW AB Pp was inoperable is that there were ineffective procedural barriers in place to ensure the EFW AB Pp was OPERABLE prior to entry into Mode 3.	team	procedure
3822011003	Waterford 3	5	04/30/2011	The wiring error was corrected on May 1, 2011 and EDG A was restored to service on May 2, 2011. Planned corrective actions include procedure changes that will require the use of plant design	team	communication+procedure

				documents to verify as left conditions and strengthening post maintenance testing.		
3822013004	Waterford 3	5	04/29/2013	However, licensed operators both on shift and in outage management positions incorrectly interpreted the meaning of the note.	team	cognition
3902008001	Watts Bar 1	3	03/20/2008	e cause assessment for the event identified an inadequate General Operating (GO) Instruction and an inadequate Instrument Maintenance Instruction (IMI).	team	procedure
3902011001	Watts Bar 1	5	05/09/2011	Not realizing that the crosstie valve was open, the licensed operators resumed filling and venting the CLAs in accordance with SOI-63.01.	team	cognition
3902012003	Watts Bar 1	4	09/21/2009	The cause of this event was that personnel were ineffective at the use of human performance tools. Specifically, personnel were ineffective at self checking to ensure all required procedures were identified during an impact review for a License Amendment request.	team	performance+check+historical
3902012005	Watts Bar 1	5	10/16/2012	The cause of this event was that plant operators did not ensure the alternate feeder breaker hand-switch was held firmly in the "closed" position while initiating the fast board transfer.	team	perfromance
3902015006	Watts Bar 1	3	10/19/2015	equired that the operators take immediate actions to open the RTBs. This action was not performed because the operators were not aware that the SR level trip channels were bypassed at the time the RTBs were closed.	team	cognition+procedure
3912016001	Watts Bar 2	3	04/14/2016	Chemistry procedure 2-CM-6.60, Revision 5, specified the wrong terminal points for bypassing the AFW pump auto/start signals. This event was attributed to a lack of procedural compliance for safety related systems, along with a lack of peer checking, during the procedure revision process.	team	procedure
3912016002	Watts Bar 2	3	05/11/2016	The inability of the TDAFWP to achieve rated speed acceptance criterial was the result of maintenance practices that incorrectly calibrated the speed controller.	team	performance
3912016003	Watts Bar 2	3	05/28/2016	he result of maintenance practices that incorrectly set the TDAFWP governor valve stem spring. A review of the governor valve work order history determined that the stem spring had likely been incorrectly set in November 2015.	team	performance+procedure
4822006003	Wolf Creek	5	10/11/2006		technical	
4822006004	Wolf Creek		10/24/2006	The root cause for this event was a human performance error resulting from instructions that were less than adequate, provided vague guidance, and introduced an opportunity for interpretation of requirements.	team	procedure
4822008004	Wolf Creek	unclear	04/07/2008	The cause of the LOOP was initiated by a human performance error. A Westar Senior Relay Technician inadvertently closed the wrong set of failure trip links	human	performance
4822008005	Wolf Creek	6	04/26/2008	The 'B' train Class 1 E electrical equipment room cooler, power and control cabinets, were replaced in the fall of 2003.	historical	
4822008006	Wolf Creek	4	05/07/2008	The Sub Work Order was miscoded as a Mode 7 restraint instead of a Mode 4 restraint. The miscoding led to the missed operability determination when the 'B' train containment spray system was returned to service.	team	work order
4822009004	Wolf Creek	3	08/21/2009	The cause of the turbine trip and FWIS was a human performance error due to inadequate monitoring of critical operating parameters. The inadequate monitoring resulted from a failure to ensure a dedicated individual maintained the responsibility to monitor steam generator water levels and disabling of the OSA.	team	supervision+communication
4822009009	Wolf	4	08/22/2009	The apparent cause of this event is that during procedure development and again during	team	procedure+co

	Creek			corrective action program evaluations, personnel considered the feedwater isolation on low Tavg coincident with P-4 to be a control function that was not part of the P-4 interlock, but rather used the P-4 interlock as input.		gnition
4822009010	Wolf Creek	4	11/18/2009	The Shift Manager failed to ensure the protected train signs were placed as annotated in the Operational Risk Assessment. No formal process existed to ensure the risk assessment performed for LCO 3.0.4b. was compatible with the applicable Shutdown Risk Assessment or Operational Risk Assessment required for normal mode transition.	team	supervision+procedure
4822011004	Wolf Creek	3	03/19/2011	The root cause of this event was the failure of the operating crew to follow procedure SYS AB-120. The operating crew did not follow the guidance in procedure AP 15C-002, "Procedure Use and Adherence," when a SYS AB-120 step that was tied to a precaution and limitation step was marked "not applicable."	team	performance+procedure+supervision
4822011005	Wolf Creek	6	04/30/2011	The cause of the event was a procedural weakness in procedure STS CR-002, "Shift Log for Modes 4, 5, and 6," which did not require plant operators to check the operability of the source range detectors using the plant computer prior to loading fuel.	team	procedure
4822011006	Wolf Creek	4	05/24/2011	The cause of the event was inadequate operator control of steam generator levels.	team	perfromance
4822012002	Wolf Creek	3	03/19/2012	nadequate procedural guidance existed to ensure that required signals are available in the Modes of applicability, and prior to Mode changes.	team	procedure
4822013002	Wolf Creek		02/04/2013		technical	
4822013003	Wolf Creek	6	02/16/2013	The night shift Instrumentation and Control Supervisor made a non-conservative decision to perform work on nuclear instrumentation cabinet SE054A and source range monitor SENI0031 when fuel offload had not been completed. The night shift Instrumentation and Control Supervisor misunderstood the schedule logic ties in WO 11-347036-000 and did not verify that the WO predecessors were complete.	team	performance+communication
4822014003	Wolf Creek		04/20/2014		technical	
4822015002	Wolf Creek	5	04/16/2015	The apparent cause of this event is the information in Operability Evaluation OE GK-12-017, which addressed a separate issue on the same equipment, enabled control room operators and engineering personnel to rationalize the assumption that the change to the acceptance criteria was bounded and did not impact the ability to meet SR 3.7.11.1.	team	performance

APPENDIX E

INSPECTION REPORT ANALYSIS (IRs) of US NUCLEAR POWER PLANTS 2000-2015

IR	Year	Name	Mode	Details
ANO_2006005	2006	Arkansas Nuclear One, Units 1 and 2	mode 5	The failure of station personnel to manipulate Valve 2DCH-11 in accordance with station procedure was determined to be a performance deficiency….The cause of the finding is related to the crosscutting aspect of human performance associated with work practices because the operator failed to use error prevention techniques like self checking and peer checking which would have prevented the event.
ANO_2010003	2010	Arkansas Nuclear One, Units 1 and 2	mode 5	The inspectors concluded that not all of the identified examples of station personnel's failure to follow Procedure EN-M-118 directly resulted in the introduction of foreign material into a critical system.
ANO_2011005	2011	Arkansas Nuclear One, Units 1 and 3		The failure of operations personnel to implement the requirements of procedure OP-1104.006, "Spent Fuel Pool Cooling System," Revision 51, and close valve SF-10 is a performance deficiency….The finding was determined to have a cross-cutting aspect in the area of human performance, associated with the work control component in that the licensee failed to ensure that work activities to support long term equipment reliability limited operator work-arounds when a torque amplifying device was required to shut valve SF-10 [H.3(b)].
BRAI2002006	2002	Braidwood Station, Units 1 and 2		The inspectors determined that the operator's action in performing activities on the wrong unit was a performance deficiency warranting a significance evaluation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," issued on April 29, 200... The inspectors determined that the error by the operator also affected the cross-cutting area of human performance because, despite several unit-specific visual indications that were available, such as color coding of procedures and components, the operator did not perform adequate self-checking to ensure that he was performing the activity on the correct unit.
BV__2006002	2006	Beaver Valley Power Station, Units 1 and 2		The inspectors determined that the licensee's failure to account for the effects of external conditions, e.g., high winds, that were predicted (National Weather Service) to occur, and its ultimate impact on crane operations (which ultimately impacted completion of head removal preps prior to draining the reactor vessel) to be a performance deficiency.....The inspectors determined that the licensee's decision to drain-down in parallel with the movement of the required equipment into containment was ultimately unsuccessful, since the drain-down was not completed until hours later when the equipment was finally transferred inside containment. However, the inspectors determined that a quantitative analysis was not required because the finding 1) did not increase the likelihood of a loss of RCS inventory, 2) did not increase the likelihood of a loss of residual heat removal (RHR) flow, and 3) did not degrade the licensee's ability to recover RHR once it was lost. Therefore, this finding is of very low safety significance (Green). A contributing cause to this finding is related to the organization subcategory of the humA contributing cause to this finding is related to the organization subcategory of the human performance cross-cutting area.

BV__2008003	2008	Beaver Valley Power Station, Units 1 and 3	mode 6	The inspectors determined that maintenance activities and reduced RCS level operations were not properly coordinated to ensure reactor vessel level remained protected and that changes were understood by the operating crew.The inspectors determined that station personnel's failure to properly coordinate maintenance and operations activities while in a reduced RCS level was a performance deficiency. Operations, Outage, and Clearance Desk personnel authorized maintenance personnel to post a clearance affecting RCS make-up without properly identifying the impact on a critical operational parameter (RCS level)
BYRO200200 7	2002	Byron Station, Units 1 and 2		..The report stated that the bolts failed because the cover did not stay within the machined edge of the coupling flange face. The licensee concluded that critical parameters for installation of the cover were not identified nor placed in written work instructions....The inspectors determined that the failure to place critical parameters in written work instructions for the installation of the turbine coupling windage shield covers was a performance deficiency warranting a significance evaluation in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B "Issue Disposition Screening" issued on April 29, 2002.The inspectors determined that this finding is more than minor because: (1) it impacts the procedure quality attribute of the Initiating Events cornerstone; and (2) it affects the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.
BYRO201400 5	2014	Byron Station, Units 1 and 3	mode 5	The inspectors determined that the licensee's intentional entry into LCO 3.6.3.0 was for operational convenience and constituted a performance deficiency….The inspectors determined that this finding had an associated cross-cutting aspect of Work Management in the Human Performance area because the shutdown and outage work schedules did not contain the rigor required to ensure the isolation valves were maintained operable as required by TS.
CNS_2009005	2009	COOPER	mode 4 and 5	The licensee's failure to identify a condition adverse to quality during the core verification process was a performance deficiency...This finding has a crosscutting aspect in the area of <u>human performance associated with resources</u> because the licensee's procedure for the core verification process is silent on potential identification of foreign material in the core [H.2(c)]..
COOK201200 3	2012	D. C. COOK		The inspectors concluded that this finding was associated with a cross-cutting aspect in the resources component of the human performance cross-cutting area.
CR__2001004	2001	Crystal River		NA
DAVI2012003	2012	DAVIS-BESSE		The inspectors determined that the licensee's practice of obtaining RCP motor bearing oil samples with the pumps running was contrary to established industry standards and the manufacturer's recommendations, and as such constituted a performance deficiency that was reasonably within the licensee's ability to foresee and correct and should have been prevented...This finding had a cross-cutting aspect in the area of

				Human Performance, Work Control component, because the licensee did not appropriately plan the work activity for the collection of RCP oil samples to incorporate risk insights that were available.
DIAB2001006	2001	DABLO CANYON I		This was identified as a cross-cutting issue based on the finding involved elements of operator training, control room alarms and procedures that contributed to distractions to the operators during midloop operations.
DUAN2010002	2010	DANE ARNOLD E		he inspectors determined that the lack of field instructions or procedures restricting the lift heights was contrary to the assumptions used in the drop load analyses and was a performance deficiency…. The inspectors did not identify a cross-cutting aspect associated with this finding because, based on the age of the performance deficiencies, it was not reflective of the current licensee performance.
FCS_2002002	2002	Fort Calhoun Station		The inspectors noted that neither procedure addressed keeping the emergency sumps free of debris during plant heat-up. The licensee initiated Condition Report 200202326 to address the inspectors' concerns.
GG2007003	2007	GRAND GULF		The inspectors determined that the failure to correct a crack in a safety-related structure was a performance deficiency.
GINN2009004	2009	GINNA NUCLEAR	mode 6 and 5	The performance deficiency associated with this finding was a failure of the AO to correctly implement S-7M. This finding is more than minor because it is associated with the Initiating Events Cornerstone and affects the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations...This finding has a cross-cutting aspect in the area of HU because operators did not adhere to the procedural requirements outlined in S-7M and close valve V-8661 prior to initiating the water transfer (H.4.b per IMC 0305).
GINN2012005	2012	GINNA NUCLEAR	mode 6	The performance deficiency associated with this finding was that Ginna personnel did not follow a standard in the conduct of operations procedure; this was within Ginna's ability to foresee and correct and should have been prevented. Specifically, operations personnel did not understand that the SFP level was very close to the 'B' SFP cooling pump trip set point prior to starting the pump, and therefore took no action to prevent the pump from tripping... The finding is more than minor because it is associated with the human performance attribute of the Barrier Integrity cornerstone and adversely impacted the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events..
Hope2003003	2003	HOPE CREEK		No findings of significance were identified.

LASA2007002	2007	LASALLE COUNTY	mode 3	The inspectors determined that there was a performance deficiency associated with the licensee’s drywell head closure bolt de-tensioning activities with the reactor in Mode 3. Specifically, licensee personnel failed to recognize the impact on the Technical Specifications from this activity until questioned by the inspectors...In addition, the inspectors determined that the finding was related primarily to the cross-cutting area of Human Performance as defined in NRC IMC 0305, “Operating Reactor Assessment Program,” since licensee personnel did not use conservative assumptions in decision-making, did not conduct any effectiveness reviews of their decision to partially de-tension the drywell head in Mode 3, and did not adequately review the decision for unintended consequences.
MILL2009005	2009	MILLSTONE Power	mode 3 5	The inspectors determined that Dominion's failure to adequately manage the risk to plant stability associated with the installation of the DCN was a performance deficiency. ...This finding had a cross-cutting aspect in the Human Performance cross-cutting area, Work Control component, because Dominion did not appropriately incorporate risk insights and work scheduling of activities consistent with nuclear safety. [H.3(a)]
MILL2010003	2010	MILLSTONE Power	mode 3 6	The inspectors determined that this finding had a cross-cutting aspect in the Human Performance cross-cutting area, Work Control component, because Dominion relied on the work control process to assure that the RPCCW cooling water was in service to the seal water heat exchanger at the time that the RCS vacuum fill was scheduled...The inspectors determined that the finding had a cross-cutting aspect in the Problem Identificationand Resolution cross-cutting area, Corrective Action Program component, because Dominion did not take appropriate corrective action to address the longstanding adverse conditions associated with control of the FRBVs [P.1(d)].
MONT201100 3	2011	MONTICELLO N		The inspectors determined that the failure to adhere to their work hour control procedures was a performance deficiency warranting further evaluation, because it was the result of the failure to meet a requirement; the cause was reasonably within the licensee’s ability to foresee and correct, and should have been prevented.The inspectors concluded that the finding was not more than minor because it did not impact the cornerstone objectives
NMP_2008003	2008	NINE MILE POINT	mode 4	The finding had a cross-cutting aspect in the area of problem identification and resolution because NMPNS did not take appropriate corrective actions to address corrosion products in the IA system in a timely manner (P.1.d per IMC 0305).
OCO_2001004	2001	Oconee	mode 2	...in part, that written procedures shall be established, implemented, and maintained covering activities outlined...his inadequate procedure issue is being treated as a NCV, consistent with Section VI.A.1 of the enforcement policy and is identified as NCV 50-269,270,287/01-04-02: Inadequate Procedure for Stroke Time Testing of the Emergency Feedwater Control Valves. This violation is in the licensee's corrective action progr

PILG2017002	2017	PILGRIM		An NRC-identified Green finding was identified because Entergy personnel did not follow Procedure 1.3.135, "Control of Doors," to adequately control a condenser bay flood protection door. Specifically, on May 22, 2017, Entergy personnel failed to control door 25A, which is designed to mitigate condenser bay flooding to preclude adversely impacting the important to safety instrument air system...This finding has a cross-cutting aspect of Human Performance - Procedure Adherence, because Entergy personnel did not follow processes, procedures, and work instructions
PALI2005006	2005	PALISADES	mode 3	the work order did not include provisions to cut out and re-weld the supports for fan housings. Subsequently, the licensee added 300 person- hours and 3.206 person-rem to the previously revised RWP. As such, the final approved estimates for the RWP were 772 person-hours and 5.878 person-rem.The licensee's Post-Job Review documented that the failures associated with the work activity included: (1) no welding activities were specified in the initial work order; (2) contractor personnel involved in motor removal were not brought onsite prior to the beginning of the outage to validate person-hour estimates; (3) the ALARA group was not informed when the decision was made to move the motor rebuild work to a higher dose rate work location; and (4) working area dose rates were about four times those assumed in the initial planning.The failure to maintain collective doses ALARA was a performance deficiency which warranted a significance determination
PB__2002005	2002	PEACH BOTTOM	mode 5	Exelon's inadequate maintenance procedure for lifting the 'B' recirculation pump motor during the 2R14 outage, is a performance deficiency
POIN2016002	2016	POINT BEACH	mode 4 to 5	The inspectors determined that the finding was more than minor, because, it was associated with the Barrier Integrity cornerstone attribute of Human Performance and affected the cornerstone objective of providing reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. ...This finding has a cross-cutting aspect of Challenge the Unknown (H.11), in the area of Human Performance, for failing to stop when faced with uncertain conditions.
QUAD2002008	2002	QUAD CITIES		The inspectors also determined that the error by the chemistry individual affected the cross-cutting area of Human Performance because, despite the inadequate procedure, the valve was properly labeled and self-checking should have been used to ensure that the proper air supply was identified.
QUAD2014002	2014	QUAD CITIES		Appendix B, the inspectors determined the performance deficiency was more than minor, and a finding, because the performance deficiency was associated with the Initiating Events Cornerstone attribute of design control and adversely affected the cornerstone objective to limit the likelihood of those events that upset the plant stability and challenge critical safety functions during shutdown, as well as power operations. (The inspectors did not identify a cross-cutting aspect associated with this finding because the concern was related to a design calculation from 2005, and thus was not necessarily indicative of current licensee performance.???)

RBS_2002002	2002	RIVER BEND		During the operations department human performance review and root cause analysis of the event, the licensee determined that time/schedule pressure was not a contributor to the high level trip of the reactor feed pump. Based on observation of the crew for the2 hours leading up to the manual scram, the inspectors determined that there was schedule pressure on the crew prior to the scram. This observation was based on the following: (1) the control room supervisor participated directly in evolutions performed by the reactor operators, such as peer checking and system procedure place keeping; (2) the control room supervisor's stated purpose for the reduction in reactor pressure control setpoint was to cut one half hour from the time required for the reactor cooldown; (3) the control room supervisor had, and used as a guide, a shutdown sequence document generated specifically for this plant shutdown with significant plant evolutions to be performed at various plant conditions compared to expected time of completion and a graph of reactor power verses expected time; and (4) two phone calls came into the control room from the outage control center with the message that they were one half hour behind the expected time line for the shutdown.The inspectors determined that the operator's failure to operate the feedwater level control system promptly in accordance with station procedures resulted in the high reactor water level trip of the running reactor feed pump and was of very low safety significance (Green) because the pump was immediately available for restart when level was reduced and all other reactor makeup systems remained functional. This human performance error was entered into the licensee's corrective action program as CR-RBS-2002-0688.
RBS_2010004	2010	RIVER BEND		Failing to replace a power supply circuit card that had a known material condition restriction in a timely manner and using vague, incomplete, and inaccurate calibration and testing instructions resulted in a plant transient that caused the reactor to exceed 100 percent thermal power was a performance deficiency.

RBS_2011002	2011	RIVER BEND		The failure to develop adequate procedure instructions to inspect Westinghouse CR 82M control rods is a performance deficiency. The finding is more than minor because it is associated with the equipment performance attribute of the reactor safety Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences...This finding has a crosscutting aspect in the area of problem identification and resolution associated with the operating experience component because the licensee failed to appropriately apply all of the CR 82M control rod inspection requirements provided by the control rod vendor [P.2(b)]. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to adequately review the results of the work order to ensure that the cause and extent of condition of the defective push-button was resolved in a timely manner [P.1(c)].
ROB_2012002	2012	Robinson Steam Electric	mode 3	The inspectors determined that the licensee’s entry in ITS LCO 3.4.12 G. was for operational convenience and constituted a performance deficiency...No cross-cutting aspect is associated with this finding as the performance deficiency does not reflect current licensee performance as the licensee has utilized this process for years.
SALM2008005	2008	SALEM		Specifically, not correcting the calculation error resulted in the inaccurate calibration procedure for the 11 RC loop narrow range level indication. This unnecessarily extended the time that the plant was operated in a reduced reactor coolant inventory condition which increased shutdown.The inspectors determined that this finding had a cross cutting aspect in the area of problem identification and resolution because PSEG did not identify the calculation error issue completely, accurately, and in a timely manner commensurate with the safety significance
SALM2011004	2011	SALEM		The inspectors determined that the failure of PSEG to complete the corrective actions for the 23 RCP motor cables in accordance with LS-AA-125, "Corrective Action Program," was a performance deficiency.

STP_2002006	2002	SOUTH TEXAS		A Green noncited violation (NCV) was identified for an inadequate procedure that permitted maintaining hot standby plant conditions with the main steam lines isolated without establishing precautions to drain accumulated condensate. This contributed to an inadvertent safety injection actuation while initiating decay heat removal from an idle steam line. A Green human performance finding was also identified because operators failed to control reactor coolant system (RCS) pressure, causing the lifting of a pressurizer PORV. This event affected Initiating Events and Barrier Integrity Cornerstone objectives...This human performance issue was determined to have very low safety significance using a Phase 2 Significance Determination Process evaluation. The inspectors assumed that all mitigation equipment remained available, but the initiating events that could challenge a pressurizer PORV had the frequency of occurrence increased by a factor of 10, in accordance with Manual Chapter 0609 guidance.
TP__2003002	2003	TURKEY POINT		The inspectors determined that this practice would be in violation of the TS. NRC Generic Letter 82-12, Nuclear Power Plant Staff Working Hours, specified limits on overtime, and stated that deviations from the limits were to be for “very unusual circumstances”. Inappropriate deviations for exceeding the overtime limits can be a significant contributor to worker fatigue and potential for human errors which, if left uncorrected, could become a more significant safety concern. The inspectors concluded that blanket authorization for the entire 18 day refueling outage was not a “very unusual circumstance”.
TP__2009003	2009	TURKEY POINT	mode 2 and 5	A Self-revealing Finding was identified when the licensee did not manage maintenance activities adequately to identify and repair a damaged rod control drive assembly prior to setting the reactor vessel closure head on the reactor vessel flange. As a result, the subsequently filled reactor coolant system had to be again drained to 2 feet below the reactor vessel flange (a high risk activity) placing the unit in the licensee’s risk condition Yellow for repairs... The performance deficiency occurred when the licensee did not adequately plan the RVCH setting activity such that contact with an RCCA was resolved prior to setting the head. Not stopping and investigating the inadvertent bumping and deformation of a RCCA extension shaft resulted in an additional RCS draining evolution and additional time in the reduced inventory condition, thus increasing overall plant risk exposure time... Additionally, the finding affected the cross cutting aspect of Human Performance, Work Practices, Supervisory & Management Oversight (H.4(c)) component because the licensee did not appropriately provide oversight of work activities, including contractors, such that nuclear safety is supported.

WNP_2001004	2001	COLUMBIA GENERATING		The inspectors identified a noncited violation of Technical Specification 5.4.1.a for failure to follow procedures when approving work, which resulted in the temporary loss of shutdown cooling during the outage. In addition, the inspectors noted two other human performance issues (determined not to be violations of NRC requirements) occurred that related to failure to adhere to procedures. The two other issues were: (1) mechanics failed to properly pack a valve that resulted in a forced shutdown to make repairs; and, (2) technicians set the over-frequency relay setpoint too low that resulted in a trip of Reactor Recirculation Pump B, while at power. The findings were determined to be of very low safety significance.

APPENDIX F

EXPERT INTERVIEW RECRUITMENT EMAIL

Participants are needed in a Research Study:

Human Errors in Nuclear Power Plants: Subject Matter Interviews

I am seeking Human Factors Engineers and Nuclear Engineers who has work experience in nuclear power plants. I am a Doctoral Human Factors Engineer student at Arizona State University conducting a study to understand human/team errors in nuclear power plants during outage control management. Participation will attend an online interview to answer 5 questions. Participants will not receive any compensation. Please contact Sally Akca for more information or email sakcahob@asu.edu .

This study has been reviewed and approved by the Arizona State University Institutional Review Board. If you have any questions about your rights as a subject/participant in this research, or if you feel you have been placed at risk, you can contact the Institutional Review Board, through the ASU Office of Research Integrity and Integrity, at (4800) 965-6788.

APPENDIX G

EXPERT INTERVIEW QUESTIONS

1- In this research two main publically available data are used: Licensee Event Reports (LERs) and Inspection Reports (IRs). Based on initial analysis the incidents are categorized under the following categories. Categories: Human, Team, Procedural, Organizational, and Design. What do you think about the categorization of errors?

2- Procedural errors are explicitly stated in both reports. The details show that there are also different levels of procedural issues.; lack of procedure, insufficient procedure, not adhering procedure, not following procedure. Can you please share your opinion about the procedural errors?

3- There are few design related incidents reported. However, overall distribution of the solely design errors is around 4%. Can you please share your opinion about design issues and teamwork in nuclear power plants?

4- NNPs are private institutions and might have different organizational structures. Can you please share your opinion about the impact of an organizational structure on outage control management?

5- One of the findings of the public data analysis shows that human errors, which are caused by single employees, are significantly lower than team errors. What do you think about this data?