OPERATING EXPERIENCE REVIEW TO DEVELOP RISK INSIGHTS RELATED TO REPOSITORY OPERATIONS

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ABSTRACT

This report documents a review of operating experience at nuclear facilities that perform operations similar to those anticipated at a potential repository at Yucca Mountain, Nevada. This review identified activities and operations that may initiate or contribute to event sequences that could result in radiological consequences. The operating experience review was conducted by collecting information on events that have occurred at other facilities that may have operations similar to those anticipated at the potential repository. Information was extracted from published reports by the U.S. Department of Energy, U.S. Nuclear Regulatory Commission, and industry data sources. Additional information was gathered through discussion of fuel handling operations with personnel from the Areva La Hague facility.

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission is preparing to review a license application for a potential high-level waste geologic repository at Yucca Mountain, Nevada. The risk-informed, performance-based regulation that governs licensing a repository is 10 CFR Part 63. The regulation at 10 CFR 63.111(c) requires a preclosure safety analysis of the Geologic Repository Operations Area (GROA) for the period before permanent closure, otherwise known as "preclosure." As defined in 10 CFR 63.102(f), the preclosure safety analysis is "a systematic evaluation of the potential hazards, initiating events, and their resulting event sequences." Furthermore, the initiating events included in the preclosure safety analysis should be "consistent with precedents adopted for nuclear facilities with comparable or higher risks to workers and the public." For this reason, operating experience at existing nuclear facilities was reviewed to identify potential hazards, initiating events, and resulting event sequences for operations that are similar to those expected at the potential repository.

This document describes the results of the operating experience review. The review was conducted by identifying the major waste handling operations expected to be performed at the GROA and then searching data sources for previous events that had occurred at facilities where similar operations are conducted. The resulting events were screened for relevance to the proposed operations at the repository.

The primary objective of this review was to gain risk insights on any activities or operations that have the potential to lead to an event sequence at the GROA that could result in a radiological dose to workers or the public. Risk insights in this context include identification of any hazardous activity or operation that may have contributed in the past to an event or that could reasonably cause or be a contributing factor to an event, with or without a specific link to radiological consequences. The risk insights gained from this review will be used to risk inform the review of the anticipated license application from the U.S. Department of Energy for the potential repository at Yucca Mountain.

The scope of the operating experience review was limited to the following major areas:

- Operations involving spent nuclear fuel assemblies, canisters, or casks
- Systems that may be important to safety such as heating, ventilation, and air conditioning for nuclear facilities, electrical power systems, or instrumentation and controls
- Hazards involving fires and explosions
- Human errors and administrative controls

This review primarily focuses on operations that may release radioactive material to the environment causing exposure to workers and/or the public. In addition, any activity or operation that could reasonably cause or be a contributing factor to an event or event sequence was evaluated—even those without a specific link to a radiological consequence. With the exception of electrical power systems, the time period for these operating experience reviews was limited to 10 years—from 1996 until present—to ensure the results are relevant to current design and operational practices in the nuclear industry.

The majority of the events reviewed involve human error. The most frequently identified potential causes of human error were less than adequate planning or procedures, work-arounds,[1] less than adequate maintenance or inspection, misleading or unreliable instruments, and less than adequate training. None of the events posed any health or safety hazard to the public because of intervention by operator actions or mitigating features. However, two events resulted in worker doses that ranged from low to approaching the dose limit. One event occurred when workers routinely defeated interlocks, and the other event occurred because of a poor interlock design. Therefore, instrumentation and control systems, and in particular, interlocks, may be an important area for technical staff to focus their review efforts. In addition, human activities (both operational and maintenance related) and interactions with equipment may be important areas on which to focus review of the potential license application, because the findings indicate human errors were a primary or contributing factor in many previous events. This report presents risk insights on operational hazards gained from this review and identification of potentially important to safety structures, systems, and components.

On the basis of the insights gained from the operating experience review, the following topics related to prevention of events are identified for potential discussion with DOE: (i) use of interlocks in the facilities and measures to prevent them from being defeated; (ii) means to prevent fires and explosions; (iii) means to detect weld cracks; (iv) the potential for loss of helium atmosphere in a container; (v) acceptance criteria for containers, such as waste packages and transportation, aging, and disposal canisters; (vi) measures to prevent dropping loads into or near the spent fuel pool; (vii) reliability of lifting and moving equipment such as cranes and the spent fuel transfer machine; and (viii) consideration of common human performance factors in the design of facilities and operations and the preclosure safety analysis. Discussions related to mitigation of events may include (i) measures to ensure the availability of equipment, such as HEPA-filtered HVAC exhaust systems and related essential electrical power, and (ii) means to mitigate the loss of multiple safety systems as a result of fires and explosions.

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[1]Alternative method of accomplishing a task when the normal or specified method cannot be used.

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QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT

DATA: No CNWRA-generated original data is contained in this report. Data used in this report are primarily obtained from other publicly available sources. Each data source is cited in this report and should be consulted to determine the level of quality for those cited data.

ANALYSES AND CODES: No codes were used for this report.

ABBREVIATIONS AND ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CNWRA	Center for Nuclear Waste Regulatory Analyses
CR	Condition Report
DOE	(U.S.) Department of Energy
DSC	Dry Shielded Canister
FSAR	Final Safety Analysis Report
GROA	Geological Repository Operations Area
HEPA	High Efficiency Particulate Air (filters)
HLW	High-Level Waste
HSS	(DOE) Office of Health, Safety and Security
HVAC	Heating, Ventilation, and Air Conditioning
IN	Information Notice (published by NRC)
INEEL	Idaho National Engineering and Environmental Laboratory (now INL)
INL	Idaho National Laboratory
ISFSI	Independent Spent Fuel Storage Installation
kV	Kilo Volts
LER	Licensee Event Report
LL	(DOE HSS) Lessons Learned (Database)
MIT	Massachusetts Institute of Technology
NEI	Nuclear Energy Institute
NUREG	NRC Regulatory Guide
OE	(DOE) Operating Experience (Weekly Summary)
RAI	Request for Additional Information
TC	Transfer Cask
VCC	Vertical Concrete Cask
WIPP	Waste Isolation Pilot Project
WSRC	(DOE) Washington Savannah River Company*

*Formerly Westinghouse Savannah River Company

1 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) is preparing to review a license application for a potential high-level waste geologic repository at Yucca Mountain, Nevada. The risk-informed, performance-based regulation that governs licensing a repository is 10 CFR Part 63. As a part of this preparation, staff identified potential waste handling operations that may be performed at the Geologic Repository Operations Area (GROA). Staff then reviewed and summarized information on operating experience at nuclear facilities having operations similar to those expected at the geologic repository at Yucca Mountain. This information will be used to gain risk insights that may be applicable to the proposed operations at the GROA. Risk insights include identification of any hazardous activity or operation that may have contributed to a past event that could reasonably cause or be a contributing factor to an event with or without a specific link to radiological consequences.

Reviewing operating experience from facilities with similar operations aids in identifying operational hazards, initiating events, and subsequent events that could affect operations at the GROA. Staff understanding of potential GROA operations is based on information provided by U.S. Department of Energy (DOE) at technical exchanges (DOE, 2007, 2006), as well as staff experience and understanding of similar operations. In many cases, events that were reviewed did not have an associated consequence (e.g., radiological dose to a worker or the public); however, there could potentially have been a consequence because a safety concern was identified. For example, a safety concern could be an event involving an equipment failure in which equipment that was relied on in a facility safety analysis was found to have failed or be in some way unavailable. This information can be used to determine which operations or activities expected at the GROA are potentially risk significant.

Operations were selected for this review based on engineering judgment and experience, with an emphasis on those operations that involve handling spent nuclear fuel or high-level waste. This review focuses on those events that were captured in a lessons-learned database, published literature, or reports. Because some operations at the GROA may be unique to the repository, risks associated with those operations, if any, are not identified through this review. However, much of the more than 50 years of nuclear industry experience is applicable because the operations are similar to the proposed operations at the GROA. Therefore, the information is useful in risk informing the staff review of a potential license application.

1.1 Background

The NRC collects, categorizes, and distributes information on operating experience at various licensed nuclear facilities. This information is a critical input to NRC programs and efforts aimed at ensuring that licensees are aware of and apply any lessons learned. Hence, operating experience plays a key role in continuously improving the safety record of the nuclear power industry. Although there may be some unique operations at the GROA, it is expected that many operations are similar to those performed at existing nuclear facilities.

1.2 Objectives

The primary objective of this activity was to gain risk insights on operational hazards that could potentially initiate or contribute to event sequences that may result in a radiological consequence at the GROA. After reviewing operating experience, staff identified events related to major operations, their causes, consequences, and any lessons learned. This information is provided in Section 3 and the appendix. Even though the focus and emphasis for review of a potential license application will ultimately depend on the DOE approach to demonstrate compliance with the preclosure performance objectives, the information obtained from operating experience may aid staff in understanding which operations, based on past experience, may be risk significant.

Additional objectives include (i) identifying structures, systems, and components that may play a major role in preventing an event or in mitigating the consequences of an event sequence (see Section 5), and (ii) guiding the staff in preparing for future interactions with DOE.

Staff can use the information from the review of operating experience to review the hazards and initiating events identified in the DOE Preclosure Safety Analysis. In addition, this information can be used to identify potential key safety systems in the repository. NRC staff can use this review to become familiar with design features that may have been associated with previous problems in the nuclear industry.

1.3 Scope

Information available on the proposed design and operations at the GROA was compared with other facilities that perform similar operations. The following areas were selected for review.

- Operations
  - Handling of fuel assemblies inside a spent fuel pool
  - Opening and/or closing canisters or casks
  - Heavy load lifting
- Systems
  - Heating, ventilation, and air conditioning (HVAC) for nuclear facilities, including high efficiency particulate air (HEPA) filtration
  - Electrical power systems
  - Instrumentation and controls
- Hazards involving fires and explosions
- Human errors and administrative controls

This review focused on operations where there was a potential for an inadvertent release, leading to a dose to the public or a safety concern potentially leading to a dose to workers.

To ensure the results of this operating experience review are relevant to current practices, the events collected and summarized were limited to a 10-year period—from 1996 until the present—with the exception of the events related to the electrical power supply. For the electrical power supply review, the span of events was from 2003 until present, because of the large number of events. When significant events that occurred before 1996 are identified or discussed, a caution regarding the age of the event is included. The 10-year period helps focus the review on more recent occurrences.

Section 2 of this report describes the methods for reviewing operating experience. Section 3 summarizes the operating experience information obtained from published literature, reports, and electronic data sources. Section 4 contains the risk insights obtained from the information that was collected. Section 5 identifies structures, systems, and components that may potentially be important to safety.

2 METHODS FOR REVIEWING OPERATING EXPERIENCE

This section describes the methods for reviewing operating experience. Based on engineering judgment and experience, staff identified selected areas for review, which are described in Section 1.3. For these areas, staff searched information sources for operating facility events that could be potentially relevant to GROA operations. Sources of information on operating experience for this review included (i) published literature and reports, (ii) electronic data sources (limited to publicly available information), and (iii) a site briefing.

Based on the information collected from the published literature, reports, and electronic data sources, staff identified risk insights (Section 4) along with a list of potential structures, systems, and components important to safety (Section 5).

2.1 Published Literature and Reports

Risk insights relevant to the proposed operations at the GROA can be derived by reviewing DOE and NRC reports that are potentially relevant to the GROA. Table 2-1 lists the literature and reports that were identified and reviewed for relevance to the operations at the potential repository.

The results of the literature review are discussed in Section 3.1.

2.2 Electronic Data Sources and Links

Tables 2-2, 2-3, and 2-4 list the electronic data sources used during this review. Data sources that were proprietary or contained information regarding international experience were not included in this study. A limited set of keywords was selected for each area of review. The keywords are generally sufficiently broad to capture information that would also appear in combination with other words or would appear if derivatives of these keywords were used instead. In addition, during these keyword searches, events identified that were not anticipated to be relevant to GROA operations were screened out. By using broad keywords to perform the searches, staff has confidence that a reasonable sampling of the most important operations or actions were identified.

The electronic data sources were searched using the following keywords and phrases:

- For operating experience related to handling of fuel assemblies in a spent fuel pool, the keywords were fuel assembly, fuel lifting, and fuel pool.
- For operating experience related to opening and/or closing canisters or casks, the keywords were canister, cask, welding, opening, cutting, and milling.
- For operating experience related to heavy load lifting, the keywords were lifting, heavy load, bridge/gantry cranes, and loads.
- For operating experience related to HVAC, or HEPA filtration, the keywords were ventilation, air flow, filter, HEPA, and HVAC.

- For operating experience related to electrical power systems, the keywords were generator, emergency, and safety.

- For operating experience related to instrumentation and controls, the keywords were instrument, control, software, I&C, program, and interlock.

- For operating experience related to fires and explosions, the keywords were fire, combustibles, detection, suppression, explosion, and burn.

- For operating experience related to administrative controls, the keywords were fuel-handling, fuel, human factors, human performance, procedures, procedure compliance, administrative, and administrative controls.

Table 2-1. Documents Containing Relevant Operating Experience

Reference or Document Number	Title	Date of Issue
NUREG–1275, Vol. 12*	Operating Experience Feedback Report, Assessment of Spent Fuel Cooling	February 1997
INL/EXT–05–00960†	Value Engineering Study for Closing Waste Packages Containing Transportation, Aging, and Disposal Canisters	November 2005
NUREG–1774‡	A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002	July 2003
NUREG–1738§	Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants	February 2001
DOE G 414.1-4‖	Safety Software Guide for Use with 10 CFR 830, Subpart A, Quality Assurance Requirements, and DOE O 414.1C, Quality Assurance	June 17, 2005
INEEL/EXT–99–01318¶	Ventilation Systems Operating Experience Review for Fusion Applications	December 1999

*NRC. NUREG–1275, "Operating Experience Feedback Report." Vol. 12. Washington, DC: NRC. February 1997.
†Allen, S., M. Berry, M. Borland, M. Clark, A. Conner, K. Croft, T. McJunkin, A. Ogurek (Bechtel SAIC Company, LLC), D. Pace, M. Rice (Bechtel SAIC Company, LLC), L. Seward, C. Shelton-Davis, R. Shurtliff, K. Skinner, H. Smartt, D. Wadsworth, and A. Watkins. "Value Engineering Study for Closing Waste Packages Containing TAD Canisters." INL/EXT-05-00960. Idaho Falls, Idaho: Idaho National Laboratory. November 2005.
‡NRC. NUREG–1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002." Washington, DC: NRC. July 2003.
§NRC. NUREG–1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants." Washington, DC: U.S. Nuclear Regulatory Commission. February 2001.
‖DOE. DOE/G 414.1-4, "Safety Software Guide for Use With 10 CFR 830 Subpart A, Quality Assurance Requirements, and DOE O 414.1C, Quality Assurance." Washington, DC: DOE. 2005.
¶Cadwallader, L.C. Ventilation Systems Operating Experience Review for Fusion Applications." INEEL/EXT–99–001318. Idaho Falls, Idaho: Idaho National Engineering and Environmental Laboratory. December 1999.

Table 2-2. NRC Data Sources on Operating Experience		
Reference or Link	**Title**	**Login / Password Required**
https://nrcoe.inel.gov/lersearch	NRC Licensee Event Reports	Yes (from NRC)
http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/	NRC Information Notices	No
http://www.nrc.gov/reactors/operating/ops-experience/human-factors.html	Human Factors Information System	Yes (from NRC)

Table 2-3. DOE Data Sources on Operating Experience		
Reference or Link	**Title**	**Login / Password Required**
http://www.eh.doe.gov/ll/occurrences.html*	Weekly Operating Experience Summaries	No
https://www.hss.energy.gov/csa/analysis/ll/oellproducts.html†	DOE Office of Health, Safety, and Security Lessons Learned Database	Yes (from DOE)
http://www.wipp.energy.gov/orps/orps.htm‡	WIPP Occurrence Reports	No
http://www.hanford.gov/rl/?page=308&parent=0§	Project Hanford Lessons Learned	No

*DOE. "Weekly Operating Experience Summaries." Washington, DC: DOE. <http://www.eh.doe.gov/ll/occurrences.html> June 18, 2007.
†DOE. "DOE Office of Health, Safety, and Security Lessons Learned Database." Washington, DC: DOE. <https://www.hss.energy.gov/csa/analysis/ll/oellproducts.html> June 18, 2007.
‡DOE. "WIPP Occurrence Reports Database." Carlsbad, New Mexico: DOE. <http://www.wipp.energy.gov/orps/orps.htm> June 18, 2007.
§DOE. "Project Hanford Lessons Learned Database." Richland, Washington: DOE. <http://www.hanford.gov/rl/?page=308&parent=0,> June 18, 2007.

Table 2-4. Additional Data Sources on Operating Experience

Reference or Link	Title	Login / Password Required
Collection of references (i.e., Inspection Reports, Licensee Event Reports, Letters)*	Dry Storage Information Forum, New Orleans, May 2–3, 2001, Lessons Learned	No
https://portal.navfac.navy.mil/portal/page?_pageid=181,3457291,181_3457371:181_3457451&_dad=portal&_schema=PORTAL†	Navy Crane Corner 32nd edition, December 2001, through 53rd edition, March 2007	No

*Nuclear Energy Institute. "Nuclear Energy Institute Dry Storage Information Forum." New Orleans, Louisiana: Nuclear Energy Institute. May 2001.
†U.S. Navy. "Navy Crane Corner." 32nd Edition (December 2001) through 53rd Edition (March 2007). Washington, DC: U.S. Navy. <https://portal.navfac.navy.mil/portal/page?_pageid=181,3457291,181_3457371:181_3457451&_dad=portal&_schema=PORTAL> (June 18, 2007).

2.3 Site Briefing

Additional risk insights can be obtained from observing existing facilities with similar operations or discussing operations with personnel from those facilities. Staff determined that the La Hague facility is appropriate to this review because spent fuel handling operations at La Hague are similar to those proposed by DOE at the GROA. A detailed briefing on both spent fuel handling and cask unloading operations in a spent fuel pool was given by personnel from the La Hague facility.

3 OPERATING EXPERIENCE INFORMATION AND DISCUSSION

3.1 Information from Published Literature and Reports

Published literature and reports were important information sources on operating experience. The relevant documents identified during the review are discussed briefly in the sections that follow.

3.1.1 NUREG–1275, Volume 12, Operating Experience Feedback Report, Assessment of Spent Fuel Cooling

This report (NRC, 1997) provides operating experience with regard to loss of spent fuel pool cooling and loss of spent fuel pool coolant (to include coolant leakage) scenarios. Coolant leakage may occur if a fuel assembly or piece of equipment is dropped and damages the spent fuel pool liner. An event occurred more than 10 years ago where the spent fuel pool liner was punctured when a core shroud bolt was dropped. Although core shroud bolts will not be handled at the GROA, components and equipment as heavy or heavier than core shroud bolts such as canisters and casks will be handled in the pool, and if dropped, they could damage the pool liner. This report also identified that “more than 30 situations involved loads heavier than allowable that were moved or could have potentially been moved over the spent-fuel pool.” It indicated that “less than 20 percent of these events involved actual downward motion or drops of objects (usually fuel assemblies) into the spent-fuel pool.”

Potential risk insights from this NUREG are included as part of Section 4.1.1, Handling Fuel Assemblies in a Spent Fuel Pool.

3.1.2 INL/EXT–05–00960, Value Engineering Study for Closing Waste Packages Containing Transportation, Aging, and Disposal Canisters

This document addresses how, or whether, the waste package closure system technology would change based on the use of transportation, aging, and disposal canisters at the repository. It discusses different alternatives for a closure system design and a recommended design in which the system is nonradiation hardened and remotely automated (with personnel intervention if necessary). A table ranking the alternatives is included in which criteria such as industrial safety risk, exposure risk, and throughput are considered. The report indicates that “experience has shown that remote automation is important to ensure high throughput schedules, minimize personnel exposure, and improve quality” (Allen, et al., 2005). Based on experience with the Three Mile Island fuel repackaging project and the Naval Reactors Facility, this report concludes that semiautomated welding with manual inspection is time and labor intensive.

Potential risk insights from this NUREG are included as part of Section 4.1.2, Opening and/or Closing Canisters or Casks.

3.1.3 NUREG–1774, A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002

This NUREG (NRC, 2003) provides operating experience information on (i) very heavy load lifts (i.e., lifts greater than 27 tonnes [30 tons]), (ii) heavy load lifts (i.e., lifts greater than 1 tonne [1.1 ton] but less than 27 tonnes [30 tons]), (iii) below-the-hook crane events [(i.e., "... an event where rigging or handling errors resulted in an event")], and (iv) spent fuel pool crane load drops. It estimates the rate of load drops per demand for very heavy loads to be 5.6×10^{-5}. This estimate was developed from three very heavy load drops during an estimated 54,000 lifts. These very heavy load drops were the result of human error (i.e., rigging failures), not crane failures that could have been prevented by using a single failure-proof crane. In addition to load drops, crane load limits have been exceeded on a few occasions when moving loads over the spent fuel pool. Other operating experience includes gear failure resulting in the crane "locking up" when removing fuel from the core and operator negligence in pretesting a crane interlock during movement of the crane over the spent fuel pool.

Potential risk insights from this NUREG are included as part of the following sections:

- Section 4.1.1, Handling of Fuel Assemblies Inside a Spent Fuel Pool
- Section 4.1.3, Heavy Load Lifting

3.1.4 NUREG–1738, Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants

This report (NRC, 2001) documents a study of spent fuel pool accident risk at decommissioning nuclear power plants. It discusses heavy load drops, loss of cooling scenarios, and loss of coolant inventory.

For the case involving loss of cooling, it identifies two events. One event in December 1998 at Browns Ferry Unit 3 involved a temperature increase of approximately 14 °C [25 °F] over a 2-day period (NRC, 2001). It indicates that, "this event ... was not detected by the control room indicators because of a design flaw in the indicators" (NRC, 2001). In a second event at the Duane Arnold Unit 1 in January 2000, the spent fuel pool temperature increased by 22–28 °C [40–50 °F] due to human error in restoring the cooling system following maintenance activities (NRC, 2001). The event was undetected for approximately 2 ½ days because the plant had no alarm for high fuel pool temperature, although temperature indicators were present in the control room.

In regard to loss of coolant inventory, this report estimates the frequency of a fuel assembly being uncovered because of loss of coolant inventory to be 3×10^{-9} per year. The primary reason for this low frequency is the assumption that coolant can only drain the pool at a certain limited rate and there is sufficient time (estimated to be 40 hours) for corrective action even in the case of a larger leak. In regard to heavy load drops, the assumption was made that only spent fuel casks are heavy enough to catastrophically damage the pool if dropped; however, no events were identified.

Potential risk insights from this NUREG are included as part of Section 4.1.6, Instrumentation and Controls.

3.1.5 DOE G 414.1-4, Safety Software Guide for Use With 10 CFR Part 830, Subpart A, Quality Assurance Requirements, and DOE O 414.1C, Quality Assurance

This document (DOE, 2005) states that "safety software failures or unintended output can lead to unexpected system or equipment failures and undue risks to the DOE/National Nuclear Security Administration mission, the environment, the public, and the workers." This document provides guidance for establishing and implementing effective quality assurance processes with regard to nuclear facility safety software applications. It makes the following specific points:

- "Software process capability models such as the Software Engineering Institute's legacy Software Capability Maturity Model (SW-CMM) and the more integrated model, Capability Maturity Model Integration (CMMI), are proven tools to assist in the selection of practices to perform for achieving a level of assurance that the processes performed will produce the desired level of quality for safety software" (DOE, 2005).

- "Software can experience partial failures that can degrade the capabilities of the overall system that may not be immediately detectable by the system. In these instances, other design techniques, such as building fault detection and self-diagnostics into the software, should be implemented" (DOE, 2005).

Potential risk insights from this guide are included as part of Section 4.1.6, Instrumentation and Controls.

3.1.6 Idaho National Engineering and Environmental Laboratory/EXT–99–01318, Ventilation Systems Operating Experience Review for Fusion Applications

This report (Cadwallader, 1999) reviews system operation and failure experiences for air ventilation systems of magnetic and inertial fusion nuclear facilities. A summary of operating experience with ventilation systems is provided, along with ventilation system components, failure rates, and component repair times. Personnel safety issues related to operating or maintaining ventilation systems are identified and discussed. The data presented in this report can be used for safety or risk analyses of general ventilation systems at nuclear facilities.

A review of 2 years of log book entries at an experimental fission power reactor at the Idaho National Engineering and Environmental Laboratory yielded the following faults with ventilation systems:

- Fan-related problems (e.g., motor grounding, vibration problems)

- Fan belt problems (e.g., loose or needed replacement)

- Smoke dampers

This report describes ventilation system failures due to human error; for example, as infrequent filter replacement or pinning a damper in a fixed position. Several equipment aging problems were also noted. Loss of offsite power caused a system relay to fail, thereby causing the ventilation system not to restart on alternate power. Other aging problems included fan rotor bearing failures and physical separation of air filters from their metal frames, reducing their efficiency and safety.

Several ventilation-related safety concerns and problems were identified. Ventilation ducts have retained particulates, which have exploded when dangerous levels have accumulated. Ventilation system ducts can also become a conduit to bypass confinement if the system isolation valves do not seal the ducts as required. Exhaust air ejectors (i.e., chimneys that allow air leaving the facility to be lofted into the atmosphere) of ventilation systems can also pose safety risks when they are blocked by accumulation of bird nests, rain water accumulation, or ice formation. Finally, tall chimneys are more susceptible to seismic activity.

Potential risk insights from this report are included as part of Section 4.1.4, HVAC and Filtration.

3.2 Information From Electronic Data Sources and Links

The electronic data sources identified in Section 2.2 were searched to identify potentially relevant events. These events are summarized in the sections that follow. For each of these events, more detailed information is included in the appendix.

3.2.1 Handling of Fuel Assemblies Inside a Spent Fuel Pool

Table 3-1 summarizes the events that were identified for spent fuel pools with more detailed information given in appendix Table 1. The Licensee Event Report database yielded four events related to operating experience of spent fuel pools. The DOE lessons learned database provided two additional events. There was one event in an NRC Information Notice.

Human error is the implied root cause for these events. Procedural problems (either operators did not follow procedures or procedural guidance was less than adequate) are the most frequently cited root cause (Events S1, S2, S3, and S4). Less than adequate operator training was the second most frequently cited contributing factor (Events S5, S6, and S7). Most of these events involved failure in planning and less than adequate procedures and training. In many of the events, the failures appear to be related to documents that were not consistent with plant conditions (e.g., inconsistencies with design basis, new reactor core designs).

A variety of operational errors are documented in the events shown in Table 3-1. For example, in one case, the fuel movement over the spent fuel pools exceeded the height limit (Event S1). In Event S2, the load limit could have been exceeded for a fuel assembly movement over the spent fuel pool. In another case (Event S3) during a fuel shuffle, the fuel assembly was dropped 0.13 m [5 in] due to an operator error with the grappling device. While in Event S4, ambiguous guidance resulted in the spent fuel pool water level not being maintained at the required shielding level when a fuel assembly did not fully seat in the storage rack.

Risk insights related to fuel movement over a spent fuel pool are included in Section 4.1.1. Spent fuel transfer machines are identified in Section 5 as structures, systems, and components potentially important to safety because they are the prime handling machines of spent fuel to and from a spent fuel pool.

Table 3-1. Summary Results for Handling of Fuel Assemblies Inside a Spent Fuel Pool		
ID No.	Locator	Description
S1	Yankee Nuclear Power Station E*: 03/15/00 R†: 04/15/00	The maximum fuel assembly travel height over “ungrated” spent fuel pool racks is 0.15 m [6 in] over the top of the rack. The fuel in this incident was lifted 0.33 m [13 in] above the racks, which is outside the design basis. Although there were no consequences from this event, this event clearly shows the result of excluding design basis height limits in plant procedures that specify the steps and pathways necessary in moving fuel assemblies. This event shows the importance of updating plant procedures to mitigate the effect of human error. (LER Database 0292000002)‡
S2	Haddam Neck Plant E: 02/19/97 R: 03/19/97	A preliminary evaluation determined that fuel assembly loads exceeding the 748 kg [1,650 lb] limit could have been moved over the spent fuel assemblies. The root cause for this event was the exclusion of fuel assemblies from the technical specification requirements in 1989 because at that time, the weight of fuel assemblies would never exceed 748 kg [1,650 lb]. Since then, new core designs have resulted in heavier fuel assemblies, which have reduced the safety margin. (LER Database 2131997004)‡
S3	Waterford Steam Electric Station E: 04/28/97 R: 06/27/97	During a fuel shuffle, a new fuel assembly disengaged from the spent fuel handling tool and dropped 0.13m [5 in]. The root cause for this event was human error. There were no consequences. The spent fuel handling tool was approximately 75 percent open and locked. Positive locking is provided between the grappling device and the fuel assembly to prevent inadvertent uncoupling. There were no administrative controls in place to ensure optimum tool orientation. As a result, the locking device was oriented away from the operator of the spent fuel handling machine. (LER Database 3821997018)‡
S4	Harris Nuclear Plant E: 01/16/99 R: 02/05/99	While the reactor operated at 100 percent power, personnel noticed that one of the boiling water reactor assemblies that was being moved did not fully seat in the storage rack. The fuel assembly was hung up on a bent channel fastener. The procedure guidance on the minimum depth of water above the fuel did not consider the possibility that a fuel assembly could get caught on a fastener and not seat fully in the rack. The procedure was revised to require a minimum of 7 m [23 ft] of water above the fuel to address this possibility. The root cause for this event was ambiguous guidance regarding channel fastener tolerances and the fact that the fasteners could bend under specific circumstances. (LER Database 4001999001)‡

Table 3-1. Summary Results for Handling of Fuel Assemblies Inside a Spent Fuel Pool (continued)		
ID No.	Locator	Description
S5	Idaho Nuclear Technology and Engineering Center E: 11/24/04 R: 11/29/04	Two lifting slings were damaged during an operation to lift and move 26,308-kg [29-ton] fuel shipping casks. The operators had noticed that the slings were damaged. The root cause for this event was that the nylon strings were improperly attached to a steel lifting attachment, which cut into the slings. This event could have been avoided if the suggested standard was incorporated into the work practice. (DOE Operating Experiences Summary ORPS Report: ID–BBWI–FUELRCSTR–2004–0006)§
S6	Hanford Plant E: 03/05/04 R: 06/28/04	The design authority noticed that a chain hoist used to move spent nuclear fuel within the K-West Basin did not have a current inspection sticker. The procedure requires that operations personnel perform a preuse check on the hoist that includes ensuring that certifications are current. That inspection had not been performed. The source of this problem was traced to operator negligence. This event shows the importance of following approved procedures and work plans. Approved lift plans and safe rigging guides should be emphasized and described in the work plan in order to mitigate events such as the one described. (DOE Operating Experiences Summary ORPS Report: RL–PHMC–SNF–2004–0017)§
S7	North Anna E: 03/24/01 R: 02/13/02	At the North Anna Power Station of Virginia Electric and Power Station, a certain type of Westinghouse fuel assembly may drop during movement. Similar events had occurred at Prairie Island in 1981 and at several foreign plants in the 1980s. The fuel assembly had separated at the top bulge joint that connects the stainless steel grid sleeves to the Zircaloy guide tube. No fission gas activity was detected afterwards, indicating that none of the fuel rods in the assembly had been fractured by the drop. Westinghouse developed a tool to lift fuel assemblies without putting a load on the bulge joint.

*Note: E—Event Date
†Note: R—Report Date
‡NRC. “Licensee Event Reports Database.” Washington, DC: NRC. <https://nrcoe.inel.gov/lersearch>. (June 18, 2007). Refer to report number indicated in description column.
§DOE. “Weekly Operating Experience Summaries.” Washington, DC: DOE. <http://www.eh.doe.gov/ll/occurrences.html>. (June 18, 2007). Refer to report number indicated in description column.

3.2.2 Opening and/or Closing Canisters or Casks

No events were identified for opening of canisters or casks. However, there were four events related to the closing of canisters or casks. These events are summarized in Table 3-2 and described in more detail in appendix Table 2.

All four events involved human errors. Two of the events (Events C1 and C3) involved "active" failures [e.g., argon gas was used instead of helium (Event C1) and hydrogen gas ignition (Event C3)]. The other two events (Events C2 and C4) involved "latent" failures that were discovered some time after the human errors were committed. The information source descriptions are not adequate to identify any potential causes for these human errors.

In two of the four events (Events C1 and C2), the issue related to the loss of helium atmosphere in the container. The loss of helium atmosphere may result in heat up of the contained fuel, which can potentially lead to fuel cladding degradation. One event (Event C3) involved the ignition of hydrogen gas during the welding of the shield lid on a Ventilated Storage Cask–24 multiassembly sealed basket. The last event (Event C4) involved a failure to flush-grind welds on the sealing surface of shipping containers resulting in the lids not sealing. None of these events resulted in injury to personnel or damage to fuel.

Risk insights involving the loss of helium atmosphere, the ignition of hydrogen gas, and the failure to flush grind welds on sealing surfaces are included in Section 4.1.2. In addition, canisters and casks are identified as structures, systems, and components that are potentially important to safety in Section 5 because their function is to contain radioactive material.

3.2.3 Heavy Load Lifting

This section discusses events obtained from the Licensee Event Reports, NRC Information Notices, the Navy Crane Corner, and DOE Office of Health, Safety, and Security Lessons Learned Database. These events are summarized in Table 3-3 and described in more detail in appendix Table 3(a)–(c).

The risk insights gained from the information in the databases highlight the importance of human error such as not following procedures, not complying with technical specifications, not performing maintenance, and not providing training. These risk insights are enumerated in Section 4.1.3. The cranes, canisters and casks, and spent fuel pool involved in lifting and moving heavy loads (spent fuel) are identified in Section 5 as structures, system, and components important to safety because a heavy load drop has a potential to result in radiological consequences.

Table 3-2. Summary Results for Opening and/or Closing Canisters or Casks		
ID No.	**Locator**	**Description**
C1	Susquehanna Steam Electric Station E*: 07/26/02 R†: 08/26/02	" ... a NUHOMS® dry fuel storage Dry Shielded Canister (DSC) was filled with argon gas instead of helium gas, ..." However, there were no actual adverse consequences to the health and safety of the public as a result of this event. (LER Database 3872002005)‡
C2	Sierra Nuclear Corporation: Palisades, Point Beach, Arkansas Nuclear One E: 03/95 to 03/97 R: 07/30/97	This response documents the issues surrounding weld problems involving the Ventilated Storage Cask multiassembly sealed basket, which has two closure lids, a shield lid, and a structural lid. One of the issues involved delayed cracking (i.e., cracking that appears after some time has elapsed after welding) [Response to CAL 97-7-001 (NEI Dry Storage Lessons Learned)]§
C3	Point Beach Nuclear Plant E: 05/28/96 R: 05/31/96	A hydrogen gas ignition occurred during the welding of the shield lid on a Ventilated Storage Cask–24 multiassembly sealed basket, which contained spent fuel assemblies. The investigation into the possible sources of hydrogen focused on a zinc-based coating applied to the internal surfaces of the multiassembly sealed basket. The consideration was that "zinc may have reacted chemically with the acidic borated water from the spent fuel storage pool to produce hydrogen." (NRC Information Notice 96-34)‖
C4	West Valley Nuclear Services Company E: 08/28/02 R: 02/27/03	Nineteen shipping containers were being prepared for offsite disposal when the sound of air escaping was heard from the lids of three containers. The contents of a number of containers consisted of canisters that had held spent nuclear fuel in a storage pool for approximately 30 years. There was a failure to flush grind welds on the lid sealing surface, which did not allow full gasket compression. (DOE Office of Health, Safety, and Security Lessons Learned Database: 2003–OH–WVNS–001)¶

*Note: E—Event Date
†Note: R—Report Date
‡NRC. "Licensee Event Reports Database." Washington, DC: NRC. <https://nrcoe.inel.gov/lersearch>. June 18, 2007. Refer to report number indicated in description column.
§Nuclear Energy Institute. "Nuclear Energy Institute Dry Storage Information Forum." New Orleans, Louisiana: Nuclear Energy Institute. May 2001.
‖NRC. "Information Notices." Washington, DC: NRC. <http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/> June 18, 2007. Refer to report number indicated in description column.
¶DOE. "DOE Office of Health, Safety, and Security Lessons Learned Database." Washington, DC: DOE. <http://www.eh.doe.gov/ll/occurrences.html>. June 18, 2007. Refer to report number indicated in description column.

3.2.3.1 Licensee Evaluation Reports and Information Notice Databases

A total of 17 events from the Licensee Event Report database and 2 events from the NRC Information Notice database were identified (Table 3-3). Eight reported events (Events HA1, HA2, HA4, HA7, HA9, HA11, HA12, and HA13) were in unanalyzed conditions or beyond the design basis category, wherein the lifting and moving operations were conducted in conditions beyond the design/analysis performed in devising the lifting procedure. Two events (Events HA3 and HA14) were in the unreviewed safety question conditions category, wherein the radiation dose consequences resulting from not following the procedure were not evaluated or reviewed in developing the procedure. Five events (Events HA6, HA8, HA10, HA15, and HA16) reported moving heavy loads when the physical safety system and/or radiation containment safety system was not functioning—a violation of not following approved procedures. Three events (Events HA5, HA10, and HA17) resulted from not checking the settings of safety equipment prior to lifting heavy loads. One event (Event HA19) resulted from not following procedures during construction, and the defective work lead to cracking and failure of the crane rail.

All 19 events appear to involve human errors as contributing causes. Four events involved procedures that were not followed (including a skipped step) as a contributing factor. At least seven events involved less than adequate procedures as a contributing factor.

Table 3-3. Summary Results for Heavy Load Lifting: Licensee Event Reports and Information Notices

ID No.	Locator	Description
HA1	Millstone 1 E*: 02/26/96 R†: 02/26/96	Heavy loads suspended over irradiated fuel in the spent fuel pool. The plant procedure does not show a safe load path over irradiated fuel, and a heavy load evaluation was not performed for this lift. This event was determined to be an unanalyzed condition, which compromises plant safety. (LER Database 2451996016)‡
HA2	Turkey Point 3 & 4 E: 07/29/96 R: 07/29/96	Failure to reflect heavy load design information in procedural controls. Location of heavy load exclusion areas was not documented correctly in procedures controlling the lift of heavy loads. Changes in the safe load path for heavy loads was not reflected in administrative procedure. As a result, heavy loads have been lifted over restricted area. (LER Database 2501996009)‡
HA3	Robinson 2 E: 04/22/97 R: 04/22/97	Investigation revealed that certain spent fuel shipping cask handling activities had been conducted outside the design and licensing basis of the plant. Lifting the cask with a nonsingle failure-proof crane with the valve box covers removed is not covered by the shipping configuration drop analysis and represents an unreviewed safety question. However, final evaluation concluded that the offsite doses resulting from a postulated cask drop with a less than fully secured cask are a small fraction of the 10 CFR Part 100 limits. (LER Database 2611997005)‡
HA4	Diablo Canyon 1 & 2 E: Not specified R: 04/30/02	Unanalyzed condition due to heavy load movement over a restricted area. With the unit in Mode 5 (Cold Shutdown), a 63,503-kg [70-ton] LP turbine cover was moved over a Unit 1 turbine building in a heavy loads restricted area above the diesel generators and 4kV vital bus ventilation, contrary to Inter-Departmental Administrative Procedure MAI.ID14, “Plant Crane Operating Restrictions.” (LER Database 2752002003)‡

Table 3-3. Summary Results for Heavy Load Lifting: Licensee Event Reports and Information Notices (continued)		
ID No.	**Locator**	**Description**
HA5	Prairie Island 1 E: 05/08/99 R: 05/08/99	Containment Inservice Purge system not isolated during heavy load movement over fuel. The reactor upper internals, a heavy load, was transported over the open fueled reactor vessel with the Containment Inservice Purge system operating. A procedure step in D58.1.6 for closing the Containment Inservice Purge system containment isolation valves was inadvertently missed and was not discovered until after the upper internals had been set in the reactor vessel. (LER Database 2821999005)‡
HA6	Prairie Island 1 & 2 E: 02/03/97 R: 02/07/97	Transporting a heavy load over irradiated fuel or safe shutdown equipment without establishing the required conditions. Reactor coolant pump upper bracket and rotor {heavy load 19,051 kg [21 tons]} were moved over irradiated fuel on February 3, 1997, without a specific load-handling procedure defining the safe load path and without containment isolated. The reactor building safe load path requirements stated in Operations Manual Section D58, Control of Heavy Loads were not followed. Operations Manual Section D58 states that "With the reactor head removed, loads greater than [953 kg] 2,100 lb SHALL NOT [emphasized in the database] be moved within [4.6 m] 15 horizontal feet of the irradiated fuel without specific written procedures per step 5.3.5" and containment isolation requirements satisfied. Neither of these provisions were satisfied. (LER Database 3061997001)‡
HA7	Arkansas 1 E: 03/06/96 R: 03/06/96	Load exceeding technical specifications weight limit moved over fuel stored in the spent fuel pool as a result of conflicting procedural guidance, which resulted from an inadequate review during procedure development. Arkansas Nuclear One personnel were lifting the cask loading pit gate, which weighs approximately 1,814 kg [4,000 lb], in preparation for storing it on the edge of the spent fuel pool. Due to the presence of steel tabs on top of the gate, it had to be rotated 180° to be stored. As the craft personnel were rotating the gate, a Senior Reactor Operator observed that it was partially positioned over the fuel in the pool. The operator immediately halted the work and directed the craft personnel to reposition the gate and move it to its storage location without passing it over any fuel. (LER Database 3131996004)‡

Table 3-3. Summary Results for Heavy Load Lifting: Licensee Event Reports and Information Notices (continued)		
ID No.	**Locator**	**Description**
HA8	Cook 1 & 2 E: 06/04/00 R: 06/28/99	Control of auxiliary building crane main load block over spent fuel pool. During performance of crane interlock testing, the east Auxiliary Building crane was operated over the spent fuel pool without the spent fuel pool ventilation system in the charcoal filter mode of operation as required by the compensatory actions and without the main load block deenergized as required by technical specification. The spent fuel pool ventilation system was declared operable but degraded because the system cannot react quickly enough to a high radiation signal to close the charcoal filter bypass dampers and prevent radioactive gases from a fuel handling accident from being released to atmosphere without passing through the charcoal filters. Compensatory actions were required, placing the spent fuel pool ventilation system in the charcoal filter mode of operation prior to movement of fuel or any load within or over the spent fuel pool. (LER Database 3152000005)‡
HA9	Brunswick 1 & 2 E: 05/06/97 R: 05/06/97	Spent fuel shipping cask handling activities had been conducted outside the design bases. Specifically, the site procedures controlling the lifting and loading of an IF-300 spent fuel shipping cask prescribe the use of rigging that is not single failure proof during transfer from the tilting cradle to the secondary yoke, contrary to existing analyses. During previous spent fuel shipping cask handling activities when the nonsingle failure proof lift condition existed, the safety-related valve box covers were not installed. Current spent fuel analyses bound a 9.1-m [30-ft] cask drop with the safety-related valve box covers installed. There is not an existing analysis for a spent fuel shipping cask drop without the valve covers installed. This event was caused by an incomplete understanding of the scope of the NEDO–10084-4, Vectra IF-300 Shipping Cask Consolidated Safety Analysis Report, and NUREG–0612, Control of Heavy Loads at Nuclear Power Plants. (LER Database 3251997004)‡

Table 3-3. Summary Results for Heavy Load Lifting: Licensee Event Reports and Information Notices (continued)		
ID No.	**Locator**	**Description**
HA10	Beaver Valley 1 & 2 E: 08/25/97 R: 08/25/97	Spent fuel pool crane interlocks and physical stops not tested prior to use in accordance with technical specifications. With Unit 1 in Mode 1 at 100 percent power, the spent fuel pool crane had been moved over the storage pool during relamping operations without first performing a surveillance procedure required by technical specification. Technical Specification Surveillance Requirement 4.9.7 requires that the crane interlocks and physical stops that prevent crane travel with loads exceeding 1,361 kg [3,000 lb] shall be demonstrated operable within 7 days prior to crane use and at least once per 7 days thereafter during crane operation. This is an operation prohibited by technical specification. There was no actual movement of heavy loads over the spent fuel pool. The crane was relocated to a position not over the fuel pool, and the required surveillance testing was performed to satisfy the technical specification surveillance requirement. (LER Database 3341997028)‡
HA11	Millstone 2 E: 10/22/01 R: 10/22/01	Movement of heavy loads not addressed in procedure. No safe load path existed for lifts of new fuel shipping containers and spent resin casks in the area of the cask washdown pit and the associated lifting device is not single failure proof. Safety-related commodities are located both in the pipe trench below the cask pit floor and on the west wall of the railroad access bay. Previously, it was identified that a 45,359-kg [50-ton] reactor coolant pump motor was stored in the cask washdown pit and that the drop of this motor would result in failure of the floor and potential damage to safety-related components in the pipe trench. Remedial corrective actions taken to date include marking the location of the pipe trench on the railroad access bay floor and removal of the reactor coolant pump motor from the cask washdown pit using a NUREG–0612 compliant lift. (LER Database 3362001007)‡

Table 3-3. Summary Results for Heavy Load Lifting: Licensee Event Reports and Information Notices (continued)		
ID No.	**Locator**	**Description**
HA12	Trojan E: 02/26/99 R: 02/26/99	An unanalyzed movement of a nonfuel loaded transfer cask. In the Fuel Building of the permanently defueled Trojan Nuclear Plant, an Independent Spent Fuel Storage Installation rigging crew was performing preoperational testing of the Independent Spent Fuel Storage Installation transfer cask lift system in accordance with an approved procedure. During the test, the test-weight loaded transfer cask was lifted above its analyzed limit while over the dry cask load pit, which should also have contained water to minimize impact loading on the cask load pit. There was no fuel in the transfer cask or in the cask load pit. There was no component failure during this event. This event is reported in accordance with 10 CFR 50.73(a)(2)(i)(B) as an operation or condition prohibited by technical specifications. The procedure was misinterpreted as not limiting transfer cask height directly over the cask load pit. Also, the procedure did not implement a transfer cask drop event calculation assumption that the cask load pit contain water to absorb impact energy, in the event of a drop. (LER Database 3441999001)‡
HA13	Davis-Besse E: 04/16/96 R: 04/16/96	Inadequate control of heavy loads in the containment building. A Potential Condition Adverse to Quality Report documented lifting the reactor vessel head lifting tripod and improperly traversing a portion of the open reactor vessel with fuel in the reactor. The reactor vessel head lifting tripod is considered a heavy load and is procedurally restricted from movement over the open reactor vessel with irradiated fuel in the reactor. The reactor vessel head lifting tripod was moved from the west secondary shield wall, across the northeast portion of the reactor vessel to the incore tank area. Further review determined that this event involved a postulated drop scenario that was not bounded by previous heavy load evaluations (a condition outside the design basis). Immediate corrective action included direction from the Plant Manager to the Outage Directors and training of affected personnel to reemphasize load path restrictions in the containment vessel. Commitments for handling of heavy loads with the Polar Crane were reviewed and additional corrective actions were to be implemented prior to the next refueling outage as determined necessary. (LER Database 3461996005)‡

Table 3-3. Summary Results for Heavy Load Lifting: Licensee Event Reports and Information Notices (continued)		
ID No.	**Locator**	**Description**
HA14	Harris E: 03/04/97 R: 03/04/97	In-plant spent fuel cask handling activities. Investigation determined that spent fuel cask handling activities have been conducted outside of the design and licensing basis of the plant. Investigation revealed that evaluation of a cask drop to a flat surface, documented in Final Safety Analysis Report 15.7.5.2, did not consider the potential consequences of dropping or otherwise damaging a loaded spent fuel cask after it had been prepared for unloading (i.e., with the cask head detensioned and valve box covers removed). Consequently, the existing cask drop evaluation in Final Safety Analysis Report 15.7.5.2 does not address a potential drop of a cask in a less than fully secured condition. This event was caused by an incomplete understanding that the 10 CFR 50.59 evaluations for procedure CM-M0300 failed to identify that the cask drop analysis conducted to confirm that the cask could withstand a 9.1-m [30-ft] free drop without a loss of integrity only applied to a cask in a fully secured, ready-for-shipment (10 CFR 71-compliant) condition. This is an unreviewed safety question. (LER Database 4001997004)‡
HA15	Wolf Creek E: 11/09/97 R: 11/09/97	Heavy loads moved in containment outside of heavy load analysis requirements. During a review of heavy load report, it was identified that during past outages heavy loads were moved in the Containment Building in a manner that was inconsistent with the heavy load analysis assumptions. Specifically, analysis assumes that both trains of residual heat removal will be operable in Mode 5 (Cold Shutdown) and Mode 6 (Refueling), yet Wolf Creek Nuclear Operating Corporation Technical Specifications allow one train to be operable in Mode 5 if the loops are filled and the secondary side water level of at least two Steam Generators is >10 percent of the wide range. Wolf Creek Nuclear Operating Corporation met the technical specification requirements, but did not recognize the analysis assumptions for residual heat removal when moving heavy loads in containment. Corrective action included revision of the controlling procedure to be consistent with the analysis. (LER Database 4821997026)‡

Table 3-3. Summary Results for Heavy Load Lifting: Licensee Event Reports and Information Notices (continued)		
ID No.	**Locator**	**Description**
HA16	Callaway E: 08/14/98 R: 08/14/98	Heavy load movement discrepancy. It was determined that during past plant outages, heavy loads have been moved in the Containment Building in a manner inconsistent with the heavy load analysis assumptions. Specifically, Callaway's analysis assumes both trains of residual heat removal will be operable in Modes 5 and 6. Callaway Technical Specifications only require one operable train of Residual Heat Removal in Mode 5 (Cold Shutdown), with reactor coolant system loops filled or in Mode 6 (Refueling), with greater than 7 m [23 ft] above the reactor vessel flange. Callaway has met the technical specification requirements, but did not recognize the analysis requirements for residual heat removal while moving heavy loads in containment when in Modes 5 and 6. The heavy load program will be reviewed to ensure no future deviations from the program. (LER Database 4831998008)‡
HA17	Summer E: 04/12/99 R: 04/12/99	The technical specification surveillance requirement requires that each auxiliary hoist and associated load indicator be demonstrated operable within 100 hours prior to start of core alterations by performing a load test. This surveillance was performed satisfactorily on the initial configuration of hoist and load indicator prior to the start of core alterations. On lifting the first control rod drive shaft to be unlatched, the crew noted that the load cell did not indicate the correct weight. The crew did not unlatch the drive shaft. The crew was not aware that the load indicator was selected for "Peak Load" instead of "Continuous" readout, giving a misleading indication. The crew suspected that the existing load cell had failed, and a new second load cell was requisitioned and installed. The results were unsatisfactory. Both the original and replacement load cells were taken to the calibration laboratory where it was discovered that both were set for peak load indication. The load cells were changed to continuous readout, both tested satisfactory for accuracy, and crane operation progressed with no impediments. (LER Database 3951999003)‡

Table 3-3. Summary Results for Heavy Load Lifting: Licensee Event Reports and Information Notices (continued)		
ID No.	**Locator**	**Description**
HA18	Prairie Island 04/30/96	Inaccurate calibration of crane overload-sensing system. While lifting a loaded spent fuel storage cask from the spent fuel pool for transfer to the transport bay, the single-failure-proof overhead crane handling system automatically stopped on overload, about 0.13 m [5 in] from the high hook point. Upon investigation, it was determined that the cause was premature actuation of the crane overload-sensing system. The set point of the overload-sensing system was set too low. Upon activation of this system, conventional holding brakes are activated and the load is held in position. The system was bypassed to move the load. Later investigation revealed that the overload-sensing system was inaccurately calibrated during the load cell setting adjustment. This information was sent to all nuclear power plants. (Information Notice 96-26)§
HA19	Trojan 04/30/96	Residual stresses resulting from flame cutting slots in crane rail during construction. Reactor building crane rail failed as indicated by cracking across the top flange. Much of the failure was preexisting because the rails did not have slots for installing bolts, and slots were burnt in the field during construction. Flame cutting the slots, without careful preheating and controlled cooling, left residual stresses. This heat-affected zone in high carbon steel was sensitive to hydrogen cracking and subsequent brittle crack propagation. Bending of rail head resulted in misalignment that in turn caused failure of bearings of bridge truck wheels. (Information Notice 96-26)§
*Note: E—Event Date †Note: R—Report Date ‡NRC. “Licensee Event Reports Database.” Washington, DC: NRC. <https://nrcoe.inel.gov/lersearch>. (June 18, 2007). Refer to report number indicated in description column. §NRC. “Information Notices.” Washington, DC: NRC. <http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/>. (June 18, 2007). Refer to report number indicated in description column.		

3.2.3.2 DOE Lessons Learned Database

Seven events from the DOE Lessons Learned database (Table 3-4) were reviewed. In three of the events (Events HB3, HB6, and HB7), inspections were less than adequate. Planning (or prepared plans) were less than adequate in at least two events (Events HB3 and HB4). Manufacturing and maintenance failures were involved in one event each (Events HB1 and HB2).

Similar to the pattern found in the NRC Licensee Event Report database, human errors accounted for the majority of the events. Equipment-related problems (e.g., old bus bar electrical cable falling from the crane due to aging) ranked second after human errors.

Table 3-4. Summary Results for Heavy Load Lifting: DOE Lessons Learned Database		
ID No.	**Locator**	**Description**
HB1	Savannah River Site 105L 11/01/01	The load block of a 27,216-kg [30-ton] bridge crane was descending to the floor when the operator tried to raise the load after hearing an unusual noise. The crane manufacturer did not install a split ring locking washer. A retaining nut on the outboard side of the holding brake drum backed off due to the missing washer, allowing the brake drum to slide completely off the motor shaft. This was an isolated incident as a result of improper installation by the manufacturer. (DOE HSS LL Database LL–WSRC–2001–0011)*
HB2	Savannah River Site 717-F 07/10/03	A 480-V electric bus bar wire 18.29 m [60 ft] long broke off from a Shaw Box 27,216-kg [30-ton] crane; it did not extend far enough to contact the personnel or equipment at floor level. The wire was used to carry power to the trolleys. The electrical systems were original (1952). No maintenance was performed on them other than a visual look during maintenance. Electrical wiring on older cranes should be inspected and maintained on a periodic schedule. (DOE HSS LL Database 2003–SR–WSRC–0012)*
HB3	No location identified	Weight data marked on legacy equipment can be either inaccurate or difficult to interpret. Prudent and well-planned lift evolutions are still vulnerable if based on faulty load estimates. Independent check of lift plans and procedures should include critical review of input parameters, as well as adequacy and currency of known and estimated weights associated with the lift. (DOE HSS LL Database L–2001–OR–BJCPORTS–0501)*
HB4	Hanford 08/07/01	Facility management determined the crane capacity had been exceeded when a 2,722-kg [3-ton] crane was used to move a rectangular grout container with debris. Operating cranes at or near load limits must be done with caution, detailed planning, and close supervision to prevent exceeding the limits. (DOE HSS LL Database 2001–RL–HNF–0027)*
HB5	Honeywell Federal Manufacturing & Technologies, Kansas City 03/19/02	A 136,078-kg [150-ton] gantry toppled while being used to lift a 6,350-kg [14,000-lb] milling machine. The operator used a forklift to pull the load while the crane lowered the load so the milling machine would stay upright. To counteract the movement when the milling machine tilted toward the forklift, the operator increased the tension of the forklift. The wheels of gantry lifted off the rails. The crane toppled and landed on a lathe nearby. The complex and unusual lift (created by the center of gravity on the spindle column of the milling machine, which was off center) caused the application of the wrong rules and techniques for the situation by the construction crew. The crew failed to identify hazards of the job and control the area as specified in the HFM&T Safety Handbook. (DOE HSS LL Database 2002–KCP–FM&T/KC–0001)*

Table 3-4. Summary Results for Heavy Load Lifting: DOE Lessons Learned Database (continued)		
ID No.	**Locator**	**Description**
HB6	Fluor Hanford 04/30/02	During the removal of a Beneficial Uses Shipping System cask lid, the trolley supporting the chain hoist separated from its I-beam, allowing the 680-kg [1,500-lb] beneficial uses shipping system cask lid to fall 0.76 m [2.5 ft] onto a plastic pallet. The trolley apparently failed because the castle retaining nut that holds the two halves of the device together came loose. No pin or locking device was installed in either castle nut. Facility management should verify that all rigging equipment, including portable gantries and associated trolley assemblies, are currently being inspected by qualified personnel according to manufacturer's written instructions for the specific equipment. (DOE HSS LL Database 2002–RL–HNF–0025)*
HB7	Fluor Hanford 09/04/01	In preparation for performing a critical lift of 3-82-B waste shipping containers, an inspection of the cask lift fixture identified a bent arm on the fixture. The device was inspected in February 2001. The inspection did not identify the arm as a problem, even though the surveillance prescribed looking for such deficiencies. The bent arm was not thought to be a load-bearing component, so the bend was considered acceptable. An engineering analysis showed that the existing design, even without the bend, did not meet buckling criteria as a load-bearing member. Facility operators should look critically at equipment to ensure deficiencies are not accepted because "they have always been that way." (DOE HSS LL Database 2001–RL–HNF–0034)*
*DOE. "DOE Office of Health, Safety, and Security Lessons Learned Database." Washington, DC: DOE. <http://www.eh.doe.gov/ll/occurrences.html>. June 18, 2007. Refer to report number indicated in description column.		

3.2.3.3 Navy Crane Corner

Ten events from the Navy Crane Corner (Table 3-5), a quarterly publication focusing on the Navy crane operations, were reviewed. Consistent with the general trend from both the NRC License Event Report and DOE Lessons Learned database, all 10 Navy crane events can be attributed to human error.

Table 3-5. Summary Results for Heavy Load Lifting: Navy Crane Corner		
ID No.	**Locator**	**Description**
HC1	Unspecified 12/01/01	Two bridge cranes {36,287 kg [80,000 lb] and 13,608 kg [30,000 lb]} each were being used to lift another bridge crane weighing 42,184 kg [93,000 lb] when one crane was overloaded. A complex lift plan used on a previous lift by this crane annotated the wrong location of the center of gravity. The error was not corrected prior to the lifting. Due to the limited lifting clearance, only one load-indicating device was used, which was on the lower capacity crane. The load on the lower capacity crane was 4,527 kg [9,960 lb], which meant the load on the higher capacity crane was 37,745 kg [83,040 lb], exceeding its rated capacity. Complex lifting plans should be reviewed, all information should be verified for accuracy, and there should be sufficient margin in the lifting cranes to allow for errors in the estimate. (Navy Crane Corner, 32nd Edition, 33rd Edition)*

Table 3-5. Summary Results for Heavy Load Lifting: Navy Crane Corner (continued)		
ID No.	**Locator**	**Description**
HC2	Unspecified 03/01/02	While operators were performing a travel test on a monorail system, the monorail beam buckled and the test load dropped to the floor. The monorail beam had been modified, apparently without an adequate engineering evaluation. Alterations to load-bearing components must be properly engineered and approved by Navy Crane Center. (Navy Crane Corner, 33rd Edition)*
HC3	12/03	A mechanic was attempting to lift an engine out of a vehicle using a 4,536-kg [10,000-lb] capacity bridge crane and a 1,814-kg [4,000-lb] capacity load leveler (a triangular, below-the-hook lifting device). The load leveler was incorrectly adjusted so that the sling attached to the front of the engine was supporting the entire load. During the lift, the engine oil pan became wedged against the frame of the vehicle, preventing the engine from being lifted. The mechanic failed to see the clearance problem and continued hoisting, thereby overloading the load leveler and causing the crane hook swivel to break. Thus the load was dropped. During all lifts, it is necessary to ensure that adequate clearance is maintained between loads, rigging gear, and any possible obstructions. (Navy Crane Corner, 40th Edition)*
HC4	06/04	An accident occurred on a bridge crane that utilized a radio control system. The radio control's transmitter malfunctioned and caused the crane to move unexpectedly. Upon investigation, it was found that one of the circuit boards was modified without the original equipment manufacturer's knowledge. The operator did not follow the original equipment manufacturer's diagnostics guide in the owner's manual when equipment malfunctioned. (Navy Crane Corner, 42nd Edition)*
HC5	09/04	Out of all Navy shore crane accidents reported in the last 3 fiscal years, 37 percent (193 accidents) occurred without a load on the hook. Almost all the accidents were attributed to human errors. Some of the more common accidents include (a) collisions with objects in the crane travel path (58 total), (b) two-block accidents (29 total), (c) wire rope damage (25 total), and (d) damage during ODCLs, set-up, and securing operations (32 total). (Navy Crane Corner, 43rd Edition)*
HC6	12/04	A Category 3 bridge crane was two-blocked during its monthly documented preuse check per NAVFAC P–307. The wire rope had been spooling on top of itself and had two-blocked in the previous month. An investigation revealed that the crane was being side loaded, causing misspooling of the wire rope and causing the hoist block to be out of position of the geared limit switch. Loads shall be lifted vertically only. Operators shall not allow side loads to be applied to the hook. (Navy Crane Corner, 44th edition)*
HC7	03/05	During a scheduled weight test of a Category 3 bridge crane, the weights were dropped when the hook separated from the hoist block. Investigation revealed that after a nondestructive evaluation test of the hook, the hook was reassembled incorrectly. A thrust ring that fits around the two washer halves used to retain the hook shank in the hoist block was omitted. When the load was lifted, the washer halves spread, pulling the hook through the hoist block and thereby dropping the load to the ground. Ensure that maintenance and inspection personnel perform properly and applicable technical manuals are available for the proper disassembly and assembly of components. (Navy Crane Corner, 45th edition)*

Table 3-5. Summary Results for Heavy Load Lifting: Navy Crane Corner (continued)		
ID No.	**Locator**	**Description**
HC8	06/05	A fire pump rotor was dropped when the wrong lifting fixture was used. After shop personnel realized they had assembled the rotor incorrectly, they decided to remove the motor with its end caps and bearings attached as a unit to save time. Not knowing the weight as a unit, they got a lifting fixture normally used for larger rotor assemblies. The fixture did not properly grip the assembly and as it was lifted, it slipped out of the fixture and dropped onto a pallet. Do not take shortcuts, and follow proper procedures. (Navy Crane Corner, 45th edition)*
HC9	12/06	An 18.3-m [60-ft] triple-laced column {approximately 25,004 kg [55,125 lb]} was being prepared for installation. A crane was used to upload the column. When the load was vertical, the main hoist wire rope pulled loose from the terminal end wedge socket connection, thereby dropping the load and hoist block to the ground. Investigation revealed that when the crane's hoist block was reeved, the wire rope was not properly seated in the wedge socket. The wedge socket was damaged from misuse; the wedge would not fully engage into the socket. Wedge socket end connections must be inspected for faulty component fit and damage from frequent change outs. (Navy Crane Corner, 52nd edition)*
HC10	12/06	A crane hoist was two-blocked causing the wire rope to part and the hoist block to fall. An activity experienced an electrical storm, which caused electrical damage and power loss to a number of buildings and bridge cranes. When the repairs were completed, the correct electric power phasing was not verified. The repair resulted in a reversal of all motor rotation on the bridge cranes. A bridge crane operator realized the crane functions were reversed but continued to operate the crane. The operator raised the hoist block into the limit switch, which did not work due to the phase reversal condition. The hoist two-blocked; the wire rope parted and the hoist block fell to the floor. Crane operators are responsible for reporting all adverse or off-normal conditions to supervision. (Navy Crane Corner, 52nd edition)*
*U.S. Navy. "Navy Crane Corner." 32nd Edition (December 2001) through 53rd Edition (March 2007). Washington, DC: U.S. Navy. <https://portal.navfac.navy.mil/portal/page?_pageid=181,3457291,181_3457371:181_3457451&_dad=portal&_schema=PORTAL> (June 18, 2007).		

3.2.4 Heating, Ventilation, Air Conditioning, and Filtration

A total of 16 events were identified from the Licensee Event Report database and the Hanford Lessons Learned database that were pertinent to heating, ventilation, air-conditioning, and filtration. These events are summarized in Table 3-6 and described in more detail in appendix Table 4.

Ventilation dampers appear to be most prone to failure (Events V7, V12, and V15). Ventilation damper problems can result in the ventilation becoming inoperable (Events V1, V2, V5, V9, and V10). Inadequate procedures or misinterpretation of existing procedures was the next highest cause of ventilation-related problems (Events V8, V11, and V14). Finally, two events were identified (Events V13 and V16) that were directly related to filter failure.

All but 1 (i.e., fire alarm failure in Event V6) of the 16 events appear to have involved a human error at some point. Five of the events involved system or equipment design problems that could ultimately be attributed to human errors. One event involved a "work-around": a rope installed to restrain the exhaust damper in Event V1. Another event involved a communication

Table 3-6. Summary Results for HVAC* and Filtration		
ID No.	**Locator**	**Description**
V1	Arkansas Nuclear Unit 1 E†: 09/28/99 R‡: 10/27/99	The flow rate of the fuel handling area ventilation system was below the technical specifications requirement while irradiated fuel was being moved in the spent fuel pool. A rope had been installed to restrain the ventilation exhaust damper in the open position during a previous electrical outage. The rope allowed the damper to close enough to reduce flow rate and thereby cause the ventilation system to become nonoperational. The root cause for this was inappropriate work practices and a deficient work plan. (LER 3131999004)§
V2	Prairie Island Unit #2 E: 04/29/97 R: 06/30/97	Both trains of spent fuel pool special ventilation were inoperable when operators opened one of the personnel doors to gain entry into the spent fuel pool enclosure. This is a clear violation of the Technical Specification 3.8.D.3, which states, "suspend all fuel handling operations and crane operations with loads over the spent fuel when both trains are out of service." The cause of this event was associated with a misinterpretation of the Technical Specification 3.8.D.3. (LER 2821997007)§
V3	Arkansas Nuclear Unit 1 E: 05/21/99 R: 06/21/99	When a radioactive spent fuel pool purification filter was moved, a radiation field was created at a detector for the Control Room Emergency Ventilation System that was severe enough to trigger the actuation of the ventilation system. This event did not create a threat to the safe environment for the control room personnel or the plant itself. The root cause for this event is attributed to the sensitive nature of the design of the ventilation system, which results in it being susceptible to spurious actuations. (LER 3131999001)§
V4	Cook Nuclear Plant E: 04/22/98 R: 08/04/99	It was discovered that the response time of the fuel handling area ventilation system for transition from the normal to the emergency filtration mode may not be adequate to prevent an unfiltered release from a refueling accident. As a result, the fuel handling ventilation system was declared inoperable. Because the ventilation system was taken out of operation before any incident, this event had minimal impact on health and safety of the public. This event stresses the importance of periodically testing the response time of key ventilation systems during the lifetime of the ventilation system. (LER 3151998029)§

Table 3-6. Summary Results for HVAC* and Filtration (continued)		
ID No.	**Locator**	**Description**
V5	Cook Nuclear Plant Unit 2 E: 07/20/00 R: 08/21/00	The spent fuel pool exhaust ventilation system was inoperable with fuel inspections in progress. Auxiliary building crane inspections were also in progress which require the ventilation system to be operable. Ventilation systems are not capable of responding to a fuel accident quickly enough to prevent an unfiltered release to the atmosphere. Therefore, a compensatory action was put in place to ensure that the spent fuel pool ventilation system was in the charcoal filter mode of operation during fuel handling operations. This event is a violation of Technical Specification 3.9.12. The cause for this event is inadequate communication between two work groups involved in the Auxiliary building crane inspections and control room personnel. Upon discovery of the malfunctioning ventilation system, control room personnel suspended movement of fuel within the spent fuel pool area. Because the affected areas were not contaminated, the safety significance of this event is minimal. (LER 3162000011)§
V6	Virgil C. Summer Nuclear Station E: 10/16/00 R: 11/15/00	The alarm failed to inform operators that the negative pressure of the spent fuel pool area had fallen below the recommended value. The cause of the alarm failure is unknown, but it is suspected that the alarm function drifted outside the acceptable range. At the time of discovery, no heavy loads were being transported above the spent fuel pool; therefore, the safety impacts of this event are minimal. (LER 3952000009)§
V7	Nine Mile Point Unit 1 E: 05/21/98 R: 06/22/98	It was discovered that the fire dampers would fail closed as a result of loss of offsite power. These dampers are required to be open during fuel transfer to provide a source of filtered air. The root cause for this event is the failure to recognize the various modes of damper operation for different plant conditions. (LER 2201998012)§
V8	Dresden Nuclear Power Station, Unit 2 E: 08/20/98 R: 09/18/98	The unit supervisor reviewed technical specifications to permit movement of fuel in the spent fuel pool area. Permission to move the fuel was given, but the unit supervisor failed to recognize that moving fuel placed the reactor in two technical specification modes that required operation of the HVAC. The Control Room HVAC system had a leak and was scheduled for maintenance. Moving fuel without the HVAC functioning is a violation of license conditions for operation`. The next day, this oversight was discovered and all activity was suspended. The causes of this event were (i) inadequate work planning/implementation process; (ii) knowledge deficiency of technical specification content; and (iii) decline in operator performance specific to management and recognition. (LER 2371998012)§

Table 3-6. Summary Results for HVAC* and Filtration (continued)		
ID No.	**Locator**	**Description**
V9	Millstone Power Station Unit 3 E: 03/1/06 R: 10/04/06	With the plant operating in Mode 1 at 100 percent power, both trains of the Control Room Emergency Ventilation System were made unavailable as a result of the valve air actuator being removed from the air inlet isolation valve. The cause of this event was a failure to recognize and correct an operating practice associated with an allowed mode of operation (isolated filtered recirculation) after the valve actuator was removed from Unit 3 as per Technical Specification 3.7.7. (LER 4232006001)§
V10	Hope Creek Plant E: 5/25/00 R: 6/23/00	The four running Filtration, Recirculation, and Ventilation Systems tripped on low flow as a result of inadvertent closure of a manual damper located in their common supply duct. The cause for this event was inattention to detail during the installation of a manual damper locking device. Contributing to this event was inadequate procedural guidance regarding damper locking device installation. Upon discovery, the damper was opened and the ventilation was restored to full operability. This event took place during surveillance testing and there was no impact on health and public safety. (LER 3542000009)§
V11	St. Lucie Units 1 and 2 E: 01/30/01 R: 08/13/02	It was discovered that inadequate procedural guidance for operation of the control room ventilation system during the emergency recirculation mode could have led to inadequate control room pressurization. This discovery was made after the plant was in Mode 1 operating at 100 percent. This event was caused by procedural inadequacies that could have allowed operation of the control room ventilation system without proper alignment of outside air make-up to the control room envelope. The operators were not clearly directed to open the outside air intake valves to establish air make-up to the control room. Improper operation of the control room ventilation system has the potential for operator doses to exceed harmless levels. However, the dose was a fraction of the 10 CFR Part 100 limits as a result of this event. Therefore, this event did not have any adverse impact on the health and safety of the public. (LER 3352001001)§

Table 3-6. Summary Results for HVAC* and Filtration (continued)		
ID No.	**Locator**	**Description**
V12	Indian Point 3 E: 01/26/05 R: 03/24/05	With the reactor at 100 percent power, it was discovered that the control room damper was operating with linkage in the reverse position, which rendered it inoperable. The apparent cause of this incident was incomplete work instructions (no detail was provided on how to connect the linkage during installation). Contributing to this was the failure of postwork testing to ensure that the damper was working correctly. Corrective action was taken to repair the damper and more explicit repair/installation procedures were issued. There was no significant health effect to the public because the system maintained functional capability. (LER 2862005001)§
V13	Hanford Site 222-S Laboratory E: 03/26/02 R: 06/26/02	Workers smelled a petrol odor within the laboratory. Three laboratory workers experienced nausea and headache symptoms. One worker was transported to the hospital for further evaluation. An investigation concluded that the likely cause of the odor was a gas-operated generator, whose exhaust was placed too close to the ventilation intake. This human error was a result of the lack of procedures for vehicle traffic or maintenance near ventilation intake areas. (Hanford Lessons Learned Database: 2002–RL–HNF–0035)‖
V14	Hanford Site E: 09/05/01 R: 01/09/02	Personnel were testing smoke detectors in the ventilation system air handling units (supply fans). They bypassed the input shutdown devices for the supply fans and introduced simulated smoke into the ventilation duct work. The simulated smoke tripped the output shutdown devices, causing dampers associated with the air handling units to close and the fans to shut down on low flow. A diesel-powered exhaust fan started to maintain negative pressure for containment in the facility. This event was caused by human error because the test procedure in use did not specify bypassing the output device nor did it provide sufficient detail for accessing the appropriate software menu for selecting the output devices. This was the first annual test of the smoke detectors with the shutdown devices in service. To rectify this problem, facilities should validate ventilation test procedures after modifying ventilation control systems and advise testing agencies of any necessary changes to their work control testing procedures. Further, ventilation-system-cognizant engineers should closely monitor testing on their systems to ensure safe and proper testing. This is especially important when testing is taking place for the first time after a system modification. (Hanford Lessons Learned Database 2002–RL–HNF–0001)‖

Table 3-6. Summary Results for HVAC* and Filtration (continued)		
ID No.	Locator	Description
V15	Hanford Site E: not reported R: 09/30/2005	During a system maintenance, the exhaust damper was found to not respond properly. The damper was found to stick in the closed position and sometimes in the 3/4 position. The root cause for this event was the nonexistence of a Failures Modes and Effects Analysis in the development of the Safety Basis, which did not recognize a damper failure. (DOE Office of Health, Safety and Security Lessons Learned Database, 2005–RL–HNF–0033)¶
V16	Hanford Site	Exhausters were found to be operating above their rated flow in violation of ANSI/ASME N509 Section 4.3. A HEPA-filtered portable exhauster was found to be operating at twice the rated flow rate of the HEPA filter. The root cause for this event was a misinterpretation of the ANSI/ASME N509 requirements by the manufacturer. (DOE Office of Health, Safety and Security Lessons Learned Database, 2001–RPP–HNF–IB–01–05)¶

*Note: HVAC—Heating, Ventilation, and Air Conditioning
†Note: E—Event Date
‡Note: R—Report Date
§NRC. “Licensee Event Reports Database.” Washington, DC: NRC. <https://nrcoe.inel.gov/lersearch>. (June 18, 2007). Refer to report number indicated in description column.
‖DOE. “Project Hanford Lessons Learned Database.” Richland, Washington: DOE. <http://www.hanford.gov/rl/?page=308&parent=0,> (June 18, 2007).
¶DOE. “DOE Office of Health, Safety, and Security Lessons Learned Database.” Washington, DC: DOE. <http://www.eh.doe.gov/ll/occurrences.html>. (June 18, 2007). Refer to report number indicated in description column.

failure. At least seven of the events involved less than adequate plans (either formal procedures or other work plans).

Among all the identified ventilation-related events, only one event had consequences sufficiently severe to require special response: Event V13 resulted in a laboratory worker being admitted to the hospital for further evaluations. This was caused by the exhaust fumes of a running generator that was placed too close to the main ventilation intake. Although other events had the potential of exposing workers to harmful levels of radiation, they were corrected in time to prevent any adverse impact to the radiological health and safety of the workers involved.

Risk insights involving HVAC systems in regard to ventilation air flow, filter leakage, and plugging, are included in Section 4.1.4. In Section 5, HVAC systems are characterized as mitigative when considering their importance to safety because they filter airborne radioactive particulates, thereby mitigating the consequences of an event sequence.

3.2.5 Electrical Power Systems

References to electrical power systems including backup power and power distribution were pervasive in experience databases because almost all active structures, systems, and components involve electrical power. Electrical power system experience reports overlap with Instrumentation/Controls, Ventilation/HVAC, and Fire/Explosion experience reports.

To better manage the scope and relevance of review of experience databases with widespread references to electrical power systems, investigations were limited to experiences reported since January 1, 2003 and were focused on off-normal events involving backup components of electrical power systems. Backup components of electrical power systems are typically used to maintain power to structures, systems, and components that are important to safety during offsite power outages or failed normal onsite power distribution systems.

During the review of experience reports and focusing on those reports reflecting experience specific to backup electrical power systems and components, five general categories of event contributors were identified:

(1) Offsite power grid outages and disturbances

(2) Onsite power system component failures due to age, degradation, lack of preventive maintenance, or defects

(3) Inadequate planning, such as the use of work-arounds or less than adequate procedures or work plans

(4) Personnel errors and inadvertent actions, especially during maintenance and testing

(5) Inadequate design—especially relevant to modifications for which the plan fails to fully evaluate the effects on the balance of the system

To illustrate some of these contributing factors, six relevant events were analyzed. They are summarized in Table 3-7 and described in more detail in appendix Table 5. Event E1 reflects Category 4 in the preceding list. Event E2 illustrates Categories 3 and 5; Event E3 reflects Categories 2 and 5; Event E4 reflects Categories 3, 4, and 5; Event E5 illustrates Categories 1 and 3; and Event E6 illustrates Category 4. All but one (Event E5) illustrate the observed relationship between modifications or maintenance activities and the occurrence of loss of power events.

All of the example events listed in Table 3-7 involved human errors. In three of the events (Events E1, E4, and E6), human errors initiated the event or degraded conditions. In the other events, human errors resulted in failures of plant defenses that were not discovered until a later time and degraded plant response. In nearly all cases, the human error involved some type of planning failure (either less than adequate procedures, work plans, design, or preventive maintenance plans).

A number of the events occurred coincidently with maintenance activities or when other related or unrelated structures, systems, and components were returned to service. Often, the electrical power system problems were discovered during or shortly after these activities (Events E4 and E6) and may have been related to changes in the equilibrium of a system of many components working together. This can occur when the sensitivity of the system to a slight change in the adjustment or behavior of a single component may not be well understood, especially as the system ages. In some cases, configurations were not fully or properly restored after maintenance or tests, and the problem did not show up until a later abnormal condition occurred (i.e., a latent human failure).

Table 3-7. Summary Results for Electrical Power Systems		
ID #	**Locator**	**Description**
E1	Fort Calhoun Station E*: 07/21/04 R†: 04/15/05	During a monthly surveillance test of an emergency diesel generator, a fuse failed at the end of the test, rendering the emergency diesel generator degraded and inoperable. The abnormal output readings and functions after the test sequence were noted but corrective actions were not pursued and the emergency diesel generator was declared "operable." The failed emergency diesel generator condition was noted at the beginning of the next monthly surveillance test of the emergency diesel generator. The emergency diesel generator was inoperable and unavailable to perform its mitigating function under Loss of Offsite Power and other hazard conditions for 29 plant operating days, which is outside the technical specification for mitigation of postulated core damage. (NRC Inspection Report 05000285/2005010)‡
E2	Nine Mile Point E: 02/12/02 R: 01/31/03	An emergency diesel generator was out of service for maintenance while the plant power level was at approximately 100 percent. During this time, an uninterruptible power supply, which was needed to ensure power to an alternate (redundant) safety-related bus was lost, leaving the plant with degraded safety systems and causing a reactor shutdown per technical specification for the plant. The root cause for this event was an inadequate design, which allowed a single component failure to disable the uninterruptible power supply bus. The failure was due to a degraded power supply in the uninterruptible power supply system resulting from age-related degraded condition and the lack of preventive maintenance. Several such components were replaced, a design modification was developed to correct the single-point-failure deficiency, and procedures modified to correct preventive maintenance practices. (LER 2202002003)§
E3	Brunswick E: 05/12/05 R: 07/11/05	An emergency diesel generator was inoperable for planned maintenance. During the interval, electrical power was lost to one 4160 VAC Emergency Bus. As a result of the power loss, the Reactor Coolant System Leakage Detection Instrumentation became inoperable. Also, a redundant control building air compressor did not start, which resulted in inoperable control room air conditioning and control room emergency ventilation. The root cause for the loss of electrical power was a design feature in the control logic that, under some conditions, caused a higher probability of failure for the emergency buses to be properly powered by offsite power. Other contributory faults (broken wire termination) either resulted from or were exposed by this abnormal configuration. Faulty components were replaced, a design modification was developed to address the conditional reduced reliability problem, and preventive maintenance procedures were improved. (LER 3252005004)§

Table 3-7. Summary Results for Electrical Power Systems (continued)		
ID #	**Locator**	**Description**
E4	Turkey Point E: 03/08/06 R: 07/28/06	An unexpected loss of power to a 4 kV safety-related bus occurred during a maintenance operation. An emergency diesel generator automatically started and restored power as planned; however, it began to fail. The emergency diesel generator was able to provide needed power for the duration of the event, but required manual supervision and adjustments to stay in service. After normal power was restored, it was discovered that both redundant emergency diesel generators had been left in an inappropriate configuration upon completion of an earlier modification to the system. As a result of the emergency diesel generator problems, supported equipment performance was degraded. The cause was the use of an incorrect plant procedure. Corrective actions included a design modification to help minimize the probability of a reoccurrence and establishment of procedures to ensure that correct component-specific procedures are used. (LER 2502006005)§
E5	Peach Bottom E: 09/15/03 R: 11/07/03	A lightning strike at an offsite power station {56.3 km [35 mi]} interrupted offsite power to the generating station resulting in an automatic reactor scram. All emergency diesel generators started properly, but one failed within about an hour, affecting the operation of some important components. An “Unusual Event” was declared as it was determined that the level of safety at the plant was potentially degraded. The Emergency Operations Center and the Technical Support Center were activated. The cause of loss of offsite power was due to poor protective relaying maintenance by the offsite power provider. The failure of the emergency diesel generator was determined to be due to improper construction or maintenance. Maintenance procedure revisions by both the commercial power provider and plant operations have been implemented, and all emergency diesel generators have undergone extensive testing. (LER 2772003004)§
E6	Palo Verde E: 04/02/06 R: 05/26/06	A 4 kV safety bus lost power during surveillance testing. An emergency diesel generator was running during testing and supplying the bus, but became inoperable causing the 4 kV safety bus to lose power. A redundant emergency diesel generator started and restored power after an expected startup delay. This caused the plant to enter an “Abnormal Operation Procedure” due to a degraded electrical system. During the loss of safety bus power, the Control Room Essential Filtration System and Control Room Emergency Air Temperature Control System were rendered inoperable. The cause of the failure of the backup power system was incorrectly installed test control relay jumpers as required by surveillance test procedures. Corrective actions included added procedural steps to require peer checks and prejob guidance and definition of responsibilities for relevant personnel. (LER 5302006003)§

*Note: E—Event Date
†Note: R—Report Date
‡NRC. “Final Significance Determination for a White Finding and Notice of Violation—Fort Calhoun Station—NRC Inspection Report 05000285/2005010.” Arlington, Texas: NRC Region IV. April 2005.
§NRC. “Licensee Event Reports Database.” Washington, DC: NRC. <https://nrcoe.inel.gov/lersearch>. (June 18, 2007). Refer to report number indicated in description column.

Risk insights involving off-normal experiences for backup power components of electrical power systems are included in Section 4.1.5. Major parts of normal electrical power systems and practically all backup power systems provide power for systems that are identified as important to safety in Section 5. These systems would be important to safety if they function to prevent or mitigate the consequences of an accidental release of radioactive material.

3.2.6 Instrumentation and Controls

There were eight events related to instrumentation and controls—four of which related to interlocks (Events I2, I3, I4, and I5). The events are summarized in Table 3-8 and described in more detail in appendix Table 6. The events ranged from personnel not maintaining instrumentation and control equipment in good working order (Event I1) to making unauthorized modifications to interlock switches (Event I2) or routinely defeating interlocks (Event I3). There were errors in maintaining equipment that resulted in equipment (Events I1 and I8) and interlock (Event I5) failures. One event involved conflicting indications (Event I4), one in which an alarm setpoint was improperly set (Event I6), and one in which controllers on the plant network became unresponsive due to excessive network traffic (Event I7).

All eight events involved human errors. Human errors may be categorized as "active" (possibly involving an erroneous action on the part of an operator) or "latent" (possibly caused by an incorrect maintenance action that is not discovered until some later time). Six of the events (Events I1, I2, I3, I4, I6, and I8) involved active failures (although the failure in Event I8 was a hardware, not a human failure). In turn, two of the events (Events I1 and I8) were dependent on prior maintenance errors. The remaining two events (Events I5 and I7) involved latent failures—one of which was due to maintenance (Event I5) and another due to a design failure (Event I7). Instrumentation or indication problems contributed to five events (Events I1, I3, I4, I6 and I7). In two of the five events (Events I1 and I3) involving instrument problems, operators were conditioned to expect spurious or unreliable indications, so correct indications were ignored. Less than adequate human–machine interface was a factor in another of these events (Event I4). Finally, two of the eight instrumentation and controls events involved work-arounds (Events I2 and I3).

Risk insights involving instrumentation and controls are included in Section 4.1.6. In addition, Section 5 identifies instrumentation and control systems as structures, systems, and components that are potentially important to safety. The reason these systems may be important to safety is that they may prevent operators from performing unsafe actions, particularly in the case of interlocks.

Table 3-8. Summary Results for Instrumentation and Controls		
ID No.	**Locator**	**Description**
I1	Oak Ridge Y-12 Site E*: 07/21/03 (related events: 06/03 and 08/98) R†: 02/09/04	Valve lineup problems highlighted instrumentation and control problems as well as human errors that have lead to equipment failure and personnel injury. Instrumentation and control equipment (i.e., pump discharge pressure gauge) was not kept in good working order, and operators ignored the status board for equipment because it was known to be unreliable. (OE Summary 2004-03)‡
I2	Thomas Jefferson National Accelerator Facility E: 05/05/04 R: 06/14/04	An individual made an unauthorized modification to interlock switches on a safety system that was designed to protect personnel from exposure to ionizing radiation. The modification was made so that the equipment could be used to conduct tests. In this event, personnel injury did not result. (OE Summary 2004-12)‡
I3	NRC-licensed irradiator facility that sterilizes medical supplies E: 04/21/04 R: 03/07/05	"Defeating the safety interlocks to enter the irradiator had been a common practice at this facility for years." In this case, when the safety interlocks were defeated, two workers entered the irradiator and received doses of 4.4 and 2.8 rem in a matter of seconds. Prior to this, "the operator and alternate radiation safety officer assumed the control panel indicating the still-exposed source rack was spurious." (OE Summary 2005-05)‡
I4	Oak Ridge National Laboratory Central Facility E: 11/15/01 R: Not provided	" ... a researcher received an estimated dose of 12 millirem to the eyes as a result of accidental exposure to x-rays from an x-ray machine." In this case, a researcher was checking a problem with the experimental setup while the interlock enclosure doors were open and did not notice the "shutter open" indicator on the machine as the view of the indicator was partially obscured. Instead the researcher relied upon the console indicator, which was readily observable but was actuated by the console switch—not the shutter mechanism itself. (OE Summary 2002-01)‡
I5	Savannah River Site E: 11/2004 R: 12/18/06	A site-programmable alarm module was replaced with another module considered to be a like-for-like replacement. This like-for-like determination was incorrect because the new module would not fail-safe when the module sensed an "open" in the input sensor signal or loss of power. In this case, a steam isolation valve would not have closed, because a safety-significant interlock would not have failed safe on loss of input sensor signal. (DOE Office of Health, Safety, and Security Lessons Learned Database: 2006–SR–WSRC–0052)§

Table 3-8. Summary Results for Instrumentation and Controls (continued)		
ID No.	**Locator**	**Description**
I6	Yankee Nuclear Power Station E: 06/26/01 R: 08/22/01	During a Nuclear Safety (Quality Assurance) Audit, the spent fuel pit area radiation monitor alarm setpoints were found set above the allowed limit. The alarm setpoints are required to be set less than 5 mR/hr [5 × 10^{-5} Gy/hr] or two times the background radiation level, whichever is greater, while moving irradiated fuel, control rods, or sources. The background radiation level was 2 mR/hr [2 × 10^{-5} Gy/hr], while the alarm setpoints were at 7 mR/hr [7 × 10^{-5} Gy/hr]. (LER Database 0292001001)‖
I7	Browns Ferry Unit 3 E: 08/19/06 R: 04/17/07	Nonsafety related controllers that were on an ethernet network became unresponsive due to excessive integrated computer system network traffic. Both safety-related and nonsafety-related equipment may be on the plant network. Therefore, it is important to protect devices on the plant network to ensure safe operation. (NRC IN 2007-15)¶
I8	No location identified E: Date not identified R: 03/2005	A hoist contactor on a crane failed to deenergize. "The operator was pressing the up button, and when the button was released, the hoist block continued to rise." The component failure was caused by a maintenance error. (Navy Crane Corner, 45th Edition, March 2005. Equipment Deficiency Memorandum–074)#

*Note: E—Event Date
†Note: R—Report Date
‡DOE. "Weekly Operating Experience Summaries." Washington, DC: DOE. <http://www.eh.doe.gov/ll/occurrences.html> (June 18, 2007). Refer to report number indicated in description column.
§DOE. "DOE Office of Health, Safety, and Security Lessons Learned Database." Washington, DC: DOE. <http://www.eh.doe.gov/ll/occurrences.html>. (June 18, 2007). Refer to report number indicated in description column.
‖NRC. "Licensee Event Reports Database." Washington, DC: NRC. <https://nrcoe.inel.gov/lersearch>. (June 18, 2007). Refer to report number indicated in description column.
¶NRC. "Information Notices." Washington, DC: NRC. <http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/> June 18, 2007. Refer to report number indicated in description column.
#U.S. Navy. "Navy Crane Corner." 32nd Edition (December 2001) through 53rd Edition (March 2007). Washington, DC: U.S. Navy. <https://portal.navfac.navy.mil/portal/page?_pageid=181,3457291,181_3457371:181_3457451&_dad=portal&_schema=PORTAL> (June 18, 2007).

3.2.7 Fires and Explosions

Fire and explosion events are closely related accidents. The fire-related events were grouped into 11 categories and tabulated in Table 3-9. Twenty-six events from different DOE and NRC sources (e.g., NRC Information Notices and DOE Lessons Learned database) were reviewed (Table 3-10). Sprinklers and other fire suppression-related events ranked first in terms of sheer number of occurrences. Electrical system-related fires ranked second in terms of number of events. All the fires were single-source occurrences. None of the fires actually threatened the reactor core. The events summarized in Table 3-10 are described in more detail in appendix Table 7.

Table 3-9. Fires and Explosions Database Search Results		
Category	Number of Events (NRC)	Number of Events (DOE)
Sprinkler and fire water piping systems	4	5
Electrical system-related incidents	6	None Reported
Fire extinguishing (Halon) systems	1	None Reported
Failure of a fire extinguisher	None Reported	1
Ineffectiveness of fire barrier material	1	
Emergency diesel generator fire	1	None Reported
Welding fire	1	None Reported
Fire near a hydrogen tank—explosion potential	None Reported	1
Configuration control	None Reported	3
Hot work (cutting) fire	None Reported	1
Incorrect design calculation	None Reported	1

Table 3-10. Summary Results for Fires and Explosions: Fires		
ID No.	Locator	Description
F1	Beaver Valley Unit 1 E*: 08/18/06 R†: 05/03/07	On August 18, 2006, a fire began during a welding evolution for a plant mod to install a ventilation duct through a 3-hour fire barrier. The wall separates a shop area from the safety-related West cable vault. After an opening was made in the wall, workers inserted a metal sleeve through the opening, stuffed combustible material into the annulus adjacent to the West cable vault steel plate, and sealed it with duct tape. A fire started when angle clips on a ventilation sleeve box were being welded. The fire was put out manually after 6 minutes. Although there was no actual consequence, this event shows that a 3-hour fire barrier could be defeated when it is breached and combustible material is introduced. The fire resulted from the use of improper material being in the improper places and an ignition source being directly applied. The fire prevention, design modification review, procedural training, and the compensation measures all failed to prevent the fire. (NRC IN 2007-17)‡
F2	Peach Bottom E: 08/15/06 R: 05/03/07	On August 15, 2006, combustible roofing material on the E-3 emergency diesel generator building caught on fire near the diesel exhaust pipe penetration (through roof) area. The fire lasted 35 minutes before it was put out by the fire brigade. Prior to the fire, the emergency diesel generator had been running for 21 hours as a part of the 24-hour endurance surveillance test. During the extended emergency diesel generator run, the steel penetration sleeve heated to the point that caused the adjacent roofing material to ignite. The air gap between the roof and the stack was below the minimum design gap. The fire resulted from construction/repair work that was not installed properly around an ignition source. (NRC IN 2007-17)‡

Table 3-10. Summary Results for Fires and Explosions: Fires (continued)		
ID No.	**Locator**	**Description**
F3	San Onofre Nuclear Generating Station Unit 3 E: 02/03/01 R: 09/20/02	On February 3, 2001, a 4.16 KV breaker faulted and initiated a fire. The firefighters used water to put out the deep-seated fire after unsuccessful attempts to extinguish the fire with dry chemicals. Consequences included loss of power to Unit 3 nonsafety-related systems and a reactor trip. This event highlighted the fact that equipment rated at 4.16 kV or higher is vulnerable to particularly energetic faults. (NRC IN 2002-27)‡
F4	Prairie Island Unit 1 E: 08/03/01 R: 09/20/02	On August 3, 2001, a fire occurred in the 12-4 cubicle along the left side of the breaker. The fire brigade used water to put out the fire after unsuccessful attempts to extinguish the fire with CO_2 and Halon. Fire was attributed to a poor electrical connection between the breaker 12-4 C-phase primary disconnect assembly and the 1MY bus stab. This event pointed out that use of a small quantity of water was effective in putting out energized electrical equipment fires. (NRC IN 2002-27)‡
F5	Ft. Calhoun E: 08/03/01 R: 09/20/02	On December 19, 2001, the underrated cord overheated and ignited the plastic and a rubber air hose nearby. The fire generated heavy smoke, which activated a deluge sprinkler system in a different fire area, spraying water on safety-related motor control centers. The use of an improperly modified plug led to the cord being underrated, thereby causing the fire due to overheating of the extension cord. This event shows that procedural requirements should be followed prior to engaging a temporary modification. (NRC IN 2002-27)‡
F6	Maanshan (Taiwan) Unit 1 E: 03/08/01	On March 8, 2001, a fault started in the safety-related 4.16kV switchgear supply circuit breaker. It caused explosions, arcing, smoke, and ionized gases, which propagated to adjacent safety-related 4.16 kV switchgear and damaged 6 switchgear compartments. The damage resulted in complete loss of the faulted safety bus and its emergency diesel generator and loss of offsite power to the undamaged safety bus. Ferromagnetic resonance was the cause of the event. (Raughley and Lanik, 2002)‖
F7	Diablo Canyon Unit 1 E: 05/15/00 R: 02/2002	On May 15, 2000, a fault occurred on the 12 kV bus duct between the auxiliary transformer and 2-12 kV buses. The sustained fault resulted in arcing in the 12 kV bus duct that jumped to and damaged the 4.16kV bus duct from startup transformer 1-2. Startup transformer 1-2 tripped, causing the loss of 4.16 kV to the 3 vital buses. It resulted in fire and loss of offsite power. (Raughley and Lanik, 2002)‖ NRC IN 2000-14‡

Table 3-10. Summary Results for Fires and Explosions: Fires (continued)		
ID No.	**Locator**	**Description**
F8	Palo Verde Unit 2 E: 04/04/96 R: 04/04/96	On April 4, 1996, an operator discovered smoke and fire in the Train B Direct Current equipment room on the 30.48-m [100-ft] level of the auxiliary building. Smoke was noticed at the Train B emergency lighting uninterruptible power supply panel in the control room. The fire was located in the 480/120 V essential lighting isolation transformer. It led to loss of power to Train B control room emergency lighting circuits, some general plant essential lighting, and plant fire detection and alarm system panels. The fire was related to and caused by a design error in the electrical grounding, which dated back to plant construction. (NRC IN 1997-01)‡
F9	Location: unspecified E: 04/24/01 R: 06/28/01	On April 24, 2001, Underwriter Laboratory issued a news release re: the failures of certain Model GB sprinkler heads made by the Central Sprinkler Co. (Lansdale, PA), (NRC IN 2001-10)‡
F10	Farley Unit 1 E: 03/04/96 R: 03/22/99	On March 4, 1996, 5 of 11 sprinkler system automatic control valves (Grinnell Model A4 deluge valves) failed to trip open during surveillance testing. Sprinkler system automatic control valves are used in fire protection systems that protect areas housing both safety- and nonsafety-related equipment for fire safe shutdown. (NRC IN 1999-07)‡
F11	Locations: affected nuclear power plants E: not specified R: 04/10/06	A 2005 NRC testing showed that both Hemyc and MT, two commonly used fire barrier materials for circuits and instrumentation protection at nuclear power plants, failed to provide the protective function (1-hour and 3-hour) intended for compliance with existing regulations. (GL 2006-03)§
F12	Locations: affected nuclear power plants E: not specified R: 07/19/02	This Information Notice alerts licensees to potential concerns with using heat collectors on sprinklers and fire detectors installed to satisfy NRC fire protection requirements. (NRC IN 2002-24)‡
F13	Calloway and Wolf Creek E: 01/12/05 R: 02/04/05	The piping to the manual-pneumatic actuators in the Halon systems protecting safety-related equipment was found to be reversed, resulting in a 2-second delay in delivering Halon. (NRC IN 2005-01)‡
F14	Washington Nuclear Project 2 E: 06/17/98 R: 08/18/98	On June 17, 1998, a water hammer occurred and caused the rupture of a 0.3 m [12-in] fire protection isolation valve in the fire protection system in the reactor building, dumping 617,020 L [163,000 gal] of fire water. (NRC IN 1998-31)‡
F15	East Tennessee Tech Park E: not specified R: 02/17/98	Fire suppression hardware was being installed on UF6 cylinders without the knowledge of the staff responsible for maintaining the cylinders. The event highlights the need to include appropriate maintenance personnel during planning activities for equipment additions. (DOE LL Database, L–1998–OR–LMESETTP)¶
F16	Pacific Western Technology E: not specified R: 10/03/00	A fire started from an arc originating from an overhead electric radiant heater in a metal building with a concrete floor. The building was originally used for decontamination purposes. Lack of sufficient administrative control of the facility allowed combustible material to be stored below the heater. (DOE LL Database, Y–2000–OR–BJCPAD–1001)¶

Table 3-10. Summary Results for Fires and Explosions: Fires (continued)		
ID No.	**Locator**	**Description**
F17	Location: unspecified E: not specified R: 02/05/07	A fire protection valve in Building 6-609 failed during the weekend, resulting in draining of two fire water tanks. Water 0.5 m [19 in] was found in the equipment room of the building when staff returned to work the next week. Consequences included water damage to the building and equipment and the fire protection system was out of service due to insufficient water pressure. (DOE LL Database USER-3 2007–NV–NTS–003)¶
F18	Hanford E: 08/02 R: 12/30/02	During a routine internal pipe inspection in August 2002, a maintenance crew discovered an inordinate accumulation of debris in the crossmain of a dry pipe fire protection sprinkler system. If left unremoved, the debris could have negatively affected the sprinkler performance. (DOE LL Database, 2002–RL–HNF–0069)¶
F19	Argonne National Laboratory-East E: not specified R: 07/8/98	Eleven of the 12 pendant sprinklers failed to operate when the links were fused during a operability test. Failure of sprinklers to open could prevent the initiation of water flow signal, thereby delaying the emergency response. (DOE LL Database, CH–AA–ANLE–ANLEESN–1998–001)¶
F20	Savannah River Site E: not specified R: 12/15/06	An incorrect number {227 m^3 [8,000 ft^3]} of air per pound of wood was used in an accident analysis. The correct number should be 2.3 m^3 [80 ft^3] of air per pound of wood. Because of the error, the calculations concluded that the fire would be air limited because the structure's design restricted the air flow. (DOE LL Database, 2006–SR–WSRC–0051)¶
F21	Los Alamos National Laboratory E: not specified R: 09/21/06	Corrosion products and nonmanufacturer paint were found on up to 60 percent of the sprinkler heads (depending on room location) in the facility. The automatic sprinkler system is a safety significant structure, system, or component and is required to be continuously operable. No actual consequences were identified. (DOE Database, LANL–ESHQ–2006–0001)¶
F22	Hanford E: not specified R: 11/18/03	A 0.9-m [36-in] minimum egress width was not maintained in the T-Plant tunnel. A hose reel on the wall projected into the 0.9-m [36-in] egress space when the Large Diameter Container trailer was in place. No actual consequences were identified. (DOE LL Database, 2003–RL–HNF–0033)¶
F23	Oak Ridge National Laboratory E: not specified R: 06/12/00	Twenty-three tons of quick lime was delivered to Oak Ridge National Laboratory to be stored at an outdoor berm constructed with straw bales. Rain was pouring down one day after delivery. Rain water reacted with the quicklime exothermically, melted the plastic covers, and ignited the straw bales. (DOE LL Database, Y–2000–OR–BJCX10–0601)¶
F24	Port of Rotterdam E: 08/25/00 R: 10/04/00	On August 25, 2000, a gas-operated dry chemical fire extinguisher at the Port of Rotterdam, Netherlands, exploded when activated, killing an employee with shrapnel. (DOE LL Database, HQ–EH–2000–02)¶

Table 3-10. Summary Results for Fires and Explosions: Fires (continued)		
ID No.	**Locator**	**Description**
F25	Location: unspecified E: 11/03/97 R: 11/17/97	On November 3, 1997, a flexible exhaust duct caught on fire when a piece of hot slag from a nearby cutting operation fell on the duct. The fire watch extinguished the fire. The fire watch received medical treatment for smoke inhalation. (DOE LL Database, L–1997–OEWS–45–02)¶
F26	James A. FitzPatrick Nuclear Power Plant E: 01/99 R: 09/09/01	An operator noticed a fire starting after he had aligned valves at the H_2 storage facility in preparation for putting the H_2 injection system into service. Fire potentially endangered the nearby H_2 storage tanks. The overhead 115kV reserved power lines were deenergized to protect firefighters. (DOE LL Database, AAN-U-01-112A)¶

*Note: E—Event Date
†Note: R—Report Date
‡NRC. "Information Notices." Washington, DC: NRC. <http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/> (June 18, 2007).
§Note: GL—Generic Letters.
‖Raughley, W.S. and G.F. Lanik. "Operating Experience Assessment Energetic Faults in 4.16 kV to 13.8 kV Switchgear and Bus Ducts that Caused Fires in Nuclear Power Plants, 1986-2001." Washington, DC: NRC. February 2002.
¶DOE. "DOE Office of Health, Safety, and Security Lessons Learned Database." Washington, DC: DOE. <https://www.hss.energy.gov/csa/analysis/ll/oellproducts.html> (June 18, 2007).

Risk insights involving fires and explosion are included in Section 4.1.7. Additionally, Section 5 identifies fire protection systems as structures, systems, and components that are potentially important to safety for the following reason—fires have the potential to compromise multiple safety systems. For instance, fires originating in electrical cabinet(s) may cut off electricity supply to structures, systems, and components important to safety. Smoke from a fire has the potential to interfere with the HEPA filtration capability in removing radioactive particles.

3.2.7.1 Sprinklers and Other Fire Suppression Systems

The most noteworthy fire suppression event occurred at the Washington Nuclear Project Unit 2, where a fire water piping system design weakness led to a water hammer when the fire pumps were commanded to come online to maintain system pressure, rupturing a 0.3 m [12 in] fire water isolation valve and dumping more than 605,664 L [160,000 gal] of water. As a consequence, certain portions of the reactor building were flooded.

Issues related to sprinkler systems and fire water piping systems spanned from sprinkler failures to large amounts of debris accumulated in the sprinkler piping systems. In one incidence, sprinkler heads were painted over with nonmanufacturer paint, thereby raising the question of sprinkler effectiveness. In another incident, pendent sprinklers failed to operate during an operability test. In general, failure of the fire water piping system invariably led to building flooding. Conversely, failure/compromise of the sprinklers did not lead to any actual consequences, because they were detected in time.

3.2.7.2 Electrical System-Related Fires

This is the single largest category of fire events in terms of number of fires reported in the database. Within this category, fire occurrences at medium range voltage (< 13.8 kV) systems (e.g., buses, breakers) appear to be a concern. Electrical cabinet fires of that voltage range often create arcing smoke and explosion. This type of fire could lead to more severe damage to equipment because the heat release rate could be much higher than that which may be predicted in a facility fire hazard analysis. The heat release rate in some of the fire hazard analyses appeared to only include combustibles from the cables and other obvious fuel sources. They did not include the contribution of electrical energy in the heat release rate calculations.

The temperature of electrical fires tended to be high. In several events, fire returned after it was initially extinguished with dry agents. Firefighters had to use water as a last resort to cool down the affected area before the deep-seated fire inside the cabinets could be extinguished. If not contained to its origin in a timely manner, the fire could easily spread and damage adjacent (safety as well as nonsafety related) electrical equipment, thereby cascading the impact to other parts of the plant.

3.2.7.3 Configuration Control

Configuration control ranged from failure to properly maintain an egress route to not following warnings on incompatible materials. Lack of or insufficient configuration control led to actual fires in two of the three events.

3.2.7.4 Welding and Other Hot-Work Fires

Hot work has been widely known as a potential source of fires. In one incidence, a 3-hour fire barrier was defeated when a metal sleeve was inserted through an opening connecting two sides of the wall. A hot-work fire started on one side of a wall and spread to the other side where safety-related cables were located because combustible materials were incorrectly used to fill the gap between the sleeve and the opening.

Half (13 out of 26) of the identified example events involve human errors. Two of the events were actual fires that were directly caused by human errors. For the remaining events that involve human errors, maintenance or design errors either contributed to the initiation of a fire or led to a deficiency (e.g., presence of combustible materials, safety equipment degradation).

3.2.7.5 Human Errors and Administrative Controls

There were numerous instances where the operating experience review identified issues related to human errors and administrative controls. In the events reviewed, the human errors in following the administrative controls did not lead to any actual radiological consequences. However, because administrative controls (i) are in place to ensure safety (e.g., if credited or relied upon in a safety analysis report or design basis evaluation); (ii) provide defense in depth; and (iii) are prevalent in nuclear materials-handling operations, it is worthwhile to review what kinds of administrative controls have been defeated in the past and why.

The following are examples of common failures in administrative controls that are captured in nuclear fuel- and cask-handling operating experience:

- Violations of technical specifications criteria

- Available procedures less than adequate

- Work plans less than adequate
- Failure to follow procedures (including work-arounds)
- Failure to follow instructions
- Verification and checks less than adequate
- Inspections less than adequate
- Failure to obtain regulatory approval for facility changes (for example, that meet 10 CFR 50.59 “changes, tests, and experiments” criteria for nuclear power licensees)
- Quality control less than adequate
- Problem identification and resolution and implementation of corrective actions less than adequate
- Training less than adequate

Causes for the defeat of these administrative controls were found to include:

- Communication less than adequate
- Procedure development less than adequate
- Work planning and coordination less than adequate
- Attention to detail less than adequate
- Work practices or craft skills less than adequate
- Awareness of plant conditions by personnel less than adequate
- Technical knowledge of personnel less than adequate
- Mistaken calculations in safety evaluations
- Work package preparation less than adequate
- Corrective action program less than adequate

Overall, these potential causes can be associated with less than adequate planning or preparation of plans (e.g., formal procedures or requirements, work plans, or packages). This conclusion is not surprising, because the operations or activities represented in the events reviewed are all field activities that are conducted under a broad range of plant or facility conditions. In contrast, prepared plans usually represent a subset of the full range of conditions, either because the full range of possible conditions or the implications of these conditions have not been considered or anticipated. Unfortunately, the tendency to make procedures more restrictive can lead to more work-arounds because such procedural restrictions may only apply to certain plant or facility conditions [e.g., the discussion of writing another procedure as noted in Reason (1997)]. According to Reason, there is often a conflict between what work practices ensure efficiency and formal rules that " ... often re[duce] the range of permitted action to far less than those necessary to get the job done under anything but optimal conditions." Because many operations and activities are not performed under "optimal conditions," Reason concludes it is important to make good decisions on *when* it is acceptable to use good work practices (for efficiency) versus *when* more prescriptive and restrictive procedural steps should be followed and stop-work practices employed when uncertainty or unexpected circumstances are encountered (NRC, 2006). Optimally, making such "good decisions" is part of a training, job experience, and an activity and plant condition-specific planning process.

This review does not capture all relevant experience with respect to human errors and failure of administrative controls at nuclear facilities since 1996.[1] Rather, the following sources were searched, along with keywords and years used for each.

- Licensee Event Reports. Reviewed all entries captured under activity of "fuel-handling" in the Human Factors Information System database. 1996–2006.

- NRC Inspection Reports for Nuclear Power Plants. Reviewed all entries captured under activity of "fuel-handling" in the Human Factors Information System database. 2000–2006.

- NRC Generic Communications–Bulletins, Information Notices, and Regulatory Issue Summaries. Searched for "fuel" in title. 1996–2006.

- DOE Lessons Learned Database: Searched red, yellow, and blue entries using available keywords related to administrative controls, which were "human factors," "human performance," "procedures," "procedure compliance," "administrative," and "administrative controls." 1996–present (early 2007).

- Hanford Lessons Learned Database. Searched red, yellow, and blue entries using available keywords related to administrative controls, which were "human factors," "human performance," "procedures," "procedure compliance," "administrative," and "administrative controls." 2006–present (early 2007) (the search capability years).

Some events were screened out from this report because of relevance (e.g., if the event related to the reactor structures, systems, and components unlikely to be used at the GROA).

Table 3-11 lists a few examples of events related to human errors in following administrative controls. See appendix Table 8 for details on all individual events and Section 4.2 for a discussion of overall insights on human performance. (The ID numbers in Table 3-11 correspond to the ID numbers in the appendix Table 8).

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The number of events that included problems related to administrative controls is too numerous to capture each individually. In addition, many of the events did not involve equipment and/or activities expected at the GROA. Therefore, the goal was to narrow the search with keywords to capture a large cross-section of relevant events without capturing large numbers of irrelevant or less relevant events.

Table 3-11. Summary Results for Administrative Controls		
ID No.	**Locator**	**Description**
A1	Various Nuclear Power Stations R*: 07/11/1997	NRC Information Notice 97-51: Problems Experienced with Loading and Unloading Spent Nuclear Fuel Storage and Transportation Casks. This information notice describes an NRC staff request of some licensees to provide information related to the movement of spent fuel storage or transportation casks without the lids on those casks being secured in place. As a result of the information request, one licensee realized that the existing practice of moving transportation casks with the lids only partially secured and with vent and drain lines exposed as a result of the removal of protective covers did not meet the assumptions in the cask drop analysis documented in the facility's updated final safety analysis report. (NRC Information Notice 97-51)†
A21	Watts Bar 1 Nuclear Power Station R: 05/10/05	Fuel movement began in the spent fuel pool for inspection of fuel assemblies, while the containment hatch was opened and the containment purge system activated for refueling outage support, which in turn made both trains of the Auxiliary Building Gas Treatment System inoperable. The Auxiliary Building Gas Treatment System is required to be operable during movement of irradiated fuel assemblies. Event causes included inadequate systems operation instruction and inadequate fuel handling instruction/technical knowledge. (LER 50-390/2005-001)‡
A46	McGuire Nuclear Power Station R: 01/26/06	During loading of spent fuel into a canister, a fuel assembly with a decay heat calculated to be approximately 1.437 kW was misloaded into the cask, exceeding the $\leq$0.958kW criterion. The cause was operational personnel inadvertently retrieving the assembly from spent fuel pool location RR–34, rather than the assembly from location PP–34. (NRC Inspection Report 05000370/2005005; NRC Event Notification Report 42203).

Table 3-11. Summary Results for Administrative Controls (continued)		
ID No.	Locator	Description
A56	Palo Verde Nuclear Power Station R: 08/09/2004	The improper positioning of a fuel pool cleanup suction valve and inadequate level monitoring resulted in three losses of spent fuel pool inventory events. These events were attributed to weaknesses in problem identification and resolution and human performance issues including lack of awareness of plant conditions by operations personnel. (NRC Inspection Report 05000528/2004003, 05000529/2004003, 05000530/2004003)
*Note: R–Report Date †NRC. "Information Notices." Washington, DC: NRC. <http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/> (June 18, 2007). ‡DOE. "DOE Office of Health, Safety, and Security Lessons Learned Database." Washington, DC: DOE. <https://www.hss.energy.gov/csa/analysis/ll/oellproducts.html> (June 18, 2007).		

The risk insights related to human errors and failure of administrative controls are summarized in Section 4.2.

3.3 Information From Site Briefing

A briefing on the La Hague site was conducted on May 4, 2007 (Areva, 2007).[2] This briefing provided background information to staff that may be useful during the review of a potential license application. For example, the receipt of spent fuel was discussed, allowing staff to consider operational characteristics such as cask staging time and fuel characteristics such as its storage time in a pool prior to shipment. Both of these characteristics may be important to reviewers when considering events that may occur during the receipt and handling of fuel at the GROA. The methods of unloading fuel were discussed along with the advantages and disadvantages of each. Staff may consider La Hague's operating experience involving their methods of unloading fuel when reviewing the proposed cask unloading operations at the GROA. In addition, crane hoist designs were discussed. These designs may be useful to reviewers who may need to evaluate the reliability of cranes proposed for the GROA. Based on this briefing, the following items may be useful to staff in reviewing proposed GROA operations.

- At the cask receipt and transfer building, the staging time for casks is typically between 1 day and 3 weeks.

- Prior to shipping fuel to the facility, the used fuel is stored in a pool next to the reactor for 1–3 years. For the GROA, fuel is expected to be cooled at least 5 years prior to receipt.

- The facility has two methods for unloading used fuel transport casks. Casks may be unloaded in a wet unloading area or a dry unloading area. The dry unloading area minimizes cask contamination and produces very little low level waste; however, if damaged fuel is received, it would be unloaded in the wet unloading area. The dry

[2]Areva. "La Hague-NRC Briefing." Washington, DC: Areva. (Unpublished). May 2007.

unloading area processes from 230 to 240 casks per year. The wet unloading area processes approximately 120 casks per year.

- The La Haque facility has experience with three types of hoist drive systems used on the cranes in the facility (Areva, 2007):

  - P3: Single hoist with a motor brake; calculated hoist failure rate = 1.8×10^{-6} per hour of use

  - P2: Single hoist with a motor brake and a brake drum; calculated hoist failure rate = 1.0×10^{-8} per hour of use

  - P1: Dual hoist drive unit with 4 brakes; calculated hoist failure rate = 1.7×10^{-10} per hour of use

4 RISK INSIGHTS

Using the information collected in the operating experience reviews as a basis, a number of potential risk insights were identified for the proposed operations at the potential geologic repository. These risk insights generally relate to the types of operations that are anticipated to be performed at the repository. The design of the repository may incorporate features to minimize these risks or to mitigate the circumstances from such occurrences; if so, the staff will adjust its insights accordingly.

The risk insights gained from the operating experience review will be relevant to the license review process at two different stages. The first stage relates to risk insights on the design and construction of the GROA, which are of primary importance when evaluating the license application for construction authorization. The second stage relates to risk insights on the operation of the GROA facilities, which will be of greater significance to the technical reviewers at the time of the final reviews for a license to receive and possess radioactive waste. In addition, risk insights on facility operations include areas that may be the focus of inspections during or after construction, as well as preparation for initial operations (e.g., review of operating procedures).

4.1 General Risk Insights

The information in the following subsections includes risk insights from published literature and reports (Section 3.1) and risk insights from electronic data sources (Section 3.2).

4.1.1 Handling Fuel Assemblies in a Spent Fuel Pool

The risk insights related to handling fuel assemblies in a pool are summarized as (i) dropping the fuel assemblies during a lift, (ii) dropping very heavy loads into or near the spent fuel pool, (iii) potentially damaging either the fuel stored in racks or the structural integrity of the pool liner, (iv) maintaining the spent fuel pool water level, (v) damaging fuel assemblies or racks when inserting or removing fuel, and (vi) inadequately ventilating or creating insufficient negative pressure during fuel handling.

The movement of heavier than allowed loads over the spent fuel pool, discussed in Section 3.1.1, continues to be a problem, although there are procedures that address this concern (NRC, 1997). An unforseen drop of a load into the spent fuel pool could cause a liner rupture and spent fuel pool leakage resulting in environmental problems and loss of coolant in the spent fuel pool. Several loss of pool water level events have taken place while level instrumentation was either inoperable or already actuated for other reasons. The loss of pool water level events did not result in radiation consequences, because the leaks were captured between the fuel pool liner and the concrete spent fuel pool structure. Ventilation events (damper problems) could affect the capability of the HVAC system to mitigate potential radiation release.

4.1.2 Opening and/or Closing Canisters or Casks

The operating experience review indicated that potential event initiators or event contributors related to opening and/or closing canisters and casks were (i) the use of an incorrect purge gas, (ii) the presence of cracks in the closure weld of the cask lid, (iii) the presence of hydrogen gas prior to welding, and (iv) inadequate sealing surfaces on casks. In addition, for weld closure operations, INL/EXT–05–00960 (Allen, 2005) indicates that remote automation of weld closure equipment is important to ensure high throughput schedules, minimize personnel exposure, and improve quality; whereas semiautomated welding with manual inspection has proven to be time and labor intensive. Although events were not identified where injury to personnel or damage to fuel occurred, events were identified where there was the potential for injury to personnel and potential for damage to fuel. These events are described in the appendix Table 4, and related insights are described in the following paragraphs.

- When either a canister is filled with the incorrect purging gas or cracks form during welding (Events C1 and C2 in Table 3-2 and appendix Table 4), the concern is a loss of the helium atmosphere. The loss of the helium atmosphere can lead to excessive heat up of the fuel, with the potential to result in fuel cladding damage. Due to human error, a canister was filled with argon instead of helium. The potential importance of this event to the operating experience review is that this case highlights latent organizational weaknesses that contributed to a human error event. Errors of this sort may potentially result in fuel cladding damage for fuel shipped to the GROA in any canisters other than transportation, aging, and disposal canisters.

- In addition, potential ignition sources need to be considered for welding operations (Event C3 in Table 3-2 and appendix Table 4). If hydrogen gas can build up in a cask prior to welding, then ignition of this gas may occur when welding operations commence. If this occurs, the fuel can potentially be damaged.

- The consideration of acceptance criteria may be an important aspect in equipment performance as it was when welds were not flush ground on the sealing surfaces of casks (Event C4 in Table 3-2 and appendix Table 4). If errors are made when equipment is accepted, then equipment such as a cask may be used even though it would not meet specifications and, consequently, not function properly. If a sealing surface is relied upon but a sufficient seal does not exist, then radioactive particulates could potentially be released.

4.1.3 Heavy Load Lifting

The databases list events at nuclear facilities that involve lifting heavy loads. Loads heavier than 27 tonnes [30 tons] are generally categorized as heavy loads. A few mishaps of loads lighter than 27 tonnes [30 tons] are also listed in the database and were considered in this study. A majority of the events with potential radiological consequences involved operator errors. However, all events were mitigated by preventive actions, and none resulted in serious radiological consequences. All the facilities were designed to comply with deterministic regulations, and most of the equipment performed as designed. Because there was no requirement for an integrated safety analysis with a systematic consideration of initiating events, event sequences, and consequences during the design of these facilities, similar to the preclosure safety analysis requirements in 10 CFR Part 63, there was no consideration of operating errors or human factors in the design. The insights from these events (i.e., risk insights) are operational hazards that could be initiating events, which could result in event sequences with a potential for radiological consequences. The lessons learned or root causes of these events can provide useful insights for evaluating operational hazards in the preclosure safety analysis of the facilities at GROA.

A majority of the events point to human errors such as less than adequate planning, less than adequate maintenance or inspection, less than adequate training, and not following procedures. Examples of specific human errors in heavy load lifting events based on review of operating experience information in Section 3.2.3 include the following.

- Procedures for movement of heavy loads and safe load paths were not followed. Occasionally, procedures were not clear or provided conflicting guidance.

- Loads were moved in situations that were not analyzed for safe handling of loads. Loads heavier than those used in the safety analysis calculations were moved/lifted, resulting in a "outside the design basis condition."

- Deficient equipment was not recognized, because the function of the equipment was not fully understood; equipment was used outside its intended functions. Lifting equipment was improperly used (e.g., side loading), causing equipment failure (two-blocking). In one instance, failure to know the weight of the load led to incorrect selection of a lifting fixture (insufficient capacity of rigging).

- Crane interlocks and physical stops were not tested before using the crane, as required by the technical specifications. In one instance, the load indicator of the crane was improperly set.

- The locking mechanism was improperly installed by the crane manufacturer, causing the brake drum to slide off the motor shaft, thus leading to dropping a load block. In one instance, the lack of a locking mechanism to hold a retainer nut together contributed to the separation of a chain hoist from its I-beam.

- Hoisting equipment was modified without proper engineering analysis, leading to the buckling of a monorail beam, thus causing the load drop during a test run. In one instance, a thrust ring to retain the hook shank of a bridge crane was omitted during the reassembly of the hook. The hook was separated from the hoist block during a weight test, thereby dropping the load to the floor.

- Improper modification of a circuit board on a radio control transmitter without the knowledge of the equipment manufacturer contributed to the malfunction of a bridge crane radio control system. In one instance, improper connection (phase reversal) to a hoist motor resulted in the malfunction of a crane hoist, leading to equipment damage.

- Cracking of the rail flange due to a misaligned crane rail resulted in failure of the bearings of the bridge truck wheels. The crack was initiated by inappropriate use of a cutting torch to enlarge slots in the web of the rail during construction.

These scenarios may be considered potential initiating events for operational hazards that could lead to event sequences with a potential for radiological consequences. As such, these risk insights should be considered during the review of the preclosure safety analysis for GROA operations.

Proper design of equipment and systems is expected to result in a safe facility. A fundamental assumption, however, is that the operators and maintenance personnel will follow safe operating procedures, comply with technical specifications, correctly follow recommended maintenance and administrative procedures, and take required training. The scope of the operational hazards review is influenced by the degree of completeness and level of details of these administrative controls (e.g., training availability of well-written procedures) provided in the license application for the proposed repository

4.1.4 HVAC and Filtration

Ventilation systems play an important role in mitigating accidental radioactive release in nuclear facilities. Based on operating experiences (Section 3.2.4), the following risk insights have been gained (Cadwallader, 1999).

- Partial loss of ventilation air flow caused by a fan fault or a fan under maintenance
- Complete loss of ventilation air flow caused by loss of power or a faulty damper
- Ventilation air flow reversal caused by plugging of the filter
- Loss of filtration caused by filter rupture
- Loss of negative pressure caused by damper failure or filter plugging
- Ventilation duct leakage due to duct wall cracking or aging
- Filter fire caused by accumulation of explosive dust in the filter
- Maintenance and repair procedures for ventilation systems

These risk-relevant events can be mitigated by regular maintenance, air flow measurement sensors that may detect and warn the operator of a potential problem, regular filter inspection and replacement, regular inspection of the filter ducts for early capture of cracks within the filter duct walls, and above all, proper operator training and proper work practices to reduce the effect of human operator error or misinterpretation of work procedures.

4.1.5 Electrical Power Systems

Electrical power systems provide normal operating power to almost all active structures, systems, and components at a facility such as the potential repository. Emergency backup power systems additionally provide backup power to a subset of active structures, systems, and components that are important to safety. If backup emergency electrical power systems are compromised or degraded, then the ability of the facility to operate safely in the event of unplanned losses of normal power is degraded. In the case of the repository, sustaining planned operation of the HVAC system in the fuel handling areas during abnormal events is anticipated to be a priority.

System designs are scrutinized to anticipate component failure probabilities and the effects of such faults on the operation of a system and to design robust systems that can continue to operate safely while faults are present. These practices can increase the complexity of systems to achieve these goals. The need for constant upgrade, modification, and surveillance testing of structures, systems, and components presents special problems in maintaining the integrity and intended fault tolerance of such complex systems.

The documentation of system designs and included nuances relied upon for reliability and fault tolerance is typically robust in the nuclear power industry. This information is also typically reflected in operating and maintenance procedures intended to ensure that the systems are properly used and that the deigned integrity of the system is retained.

The operating experiences reviewed illustrate some instances where these efforts have been less than successful. None of the events reviewed resulted in compromised public or worker safety or a radiological release. Nevertheless, the events reflect problems that could lead to such results if these or similar events occur coincidently with other sensitive normal or abnormal plant operations.

In the case of the undetected failed fuse that disabled an emergency diesel generator for 29 days, a followup study indicated a clear relationship between this failure and a degraded ability of the plant systems to adequately compensate and prevent core damage if other postulated events had occurred during that time. The failure of a number of personnel to recognize that nonfunctioning components or out-of-range readings could indicate a compromised system is a key element of this event.

In several of the events analyzed, previously completed modifications or previously completed maintenance activities left the emergency backup power systems degraded or unavailable to function as intended. In some cases, the inadequacies were discovered during other unrelated planned maintenance or routine surveillance testing. In other cases, the undetected problems were discovered only when the emergency backup power systems were needed to compensate for unexpected failures in the normal power distribution systems, leading to changes in operations to better deal with degraded operational safety until the problems were rectified. In these cases, the inadequacy of procedures and practices allowed less than robust or faulty conditions to remain in place, contributing to problems at a later time. Better modification design and analysis and better planning for maintenance and test activities were required to overcome these problems.

Deficiencies in the design and analysis of systems that failed to meet the robust fault-tolerant goals were also reflected in the experience databases. In some cases, the analysis of the system did not account for all of the potential failure mechanisms or results of a range of interactions between components or even all subsystems that were important in providing reliable backup power systems. Furthermore, the designer or analyzer in some cases may not have considered the effects of incorrect human operations or interpretations that can produce abnormal configurations or faults that may not result from a robust technical analysis of the mechanization design of a component. In most cases, designs and/or procedures were improved after an event investigation to eliminate or reduce the likelihood of repeating such experiences.

Risk insights developed during these reviews include the following.

- Comprehensive plans, including procedures and processes, must be in place and be followed. Using the correct and specific plans and practices (especially for different plant configurations and conditions) and maintaining a questioning attitude (e.g., recognizing off-normal readings or results) is important.

- System modifications can compromise the integrity of a system, especially component modifications that are limited and localized in scope. The risk may be mitigated by comprehensively reviewing design modifications at a system level and reviewing relevant operating and test procedures and practices.

- System and component design issues are typically addressed during system design, analysis, test, and rollout, but the effects of aging and continuing human interaction during testing and maintenance over the life of the system is sometimes not adequately considered. Associated risks may be reduced by considering potential component fault mechanisms, service-life drifts in the behavior of components, and the potential for misinterpretation or misapplication of component features during the life of the system.

- There was an observed coincident relationship between reportable electrical power and backup power systems incidents and recent or current modifications or testing for all events in Table 3-7. These experiences may have been improved by planning surveillance or other testing only when other sensitive operations are suspended. The reportable experiences for Events E1, E5, and E6 may have been mitigated by the conduct of “acceptance” testing of the entire normal and backup power systems in both normal and abnormal conditions after implementing design modifications or routine maintenance and testing and before declaring systems operational and ready for service.

4.1.6 Instrumentation and Controls

Instrumentation and controls involve safety-related equipment such as interlocks and alarms as well as equipment that is relied upon for normal operations such as crane hoist controls. The operating experience review summarized in Section 3.2.6 indicated that interlocks can fail, have a faulty design, or be defeated by personnel. Other equipment relied upon for normal operations such as crane hoist controls can fail as a result of maintenance errors. In addition, as indicated in NUREG–1738 (NRC, 2001), operational failures (e.g., failure to maintain spent fuel pool cooling) may occur due to faulty design of alarms and indicators. Software can be an important aspect of instrumentation and control systems, and as indicated in DOE G 414.1-4, “software can experience partial failures that can degrade the capabilities of the overall system that may not be immediately detectable by the system” (DOE, 2005). If this were to occur, instrumentation and control systems that rely on software may become unavailable and the failure may not be immediately detectable.

Staff identified cases in which personnel injury occurred and cases in which personnel exposure occurred as a result of instrumentation and controls events. These events are described in Table 6 of the appendix, and insights from these events as well as others found during the operating experience review are summarized in the following paragraphs.

- For the case of valve lineup problems (Event I1 in Table 3-8 and appendix Table 6), instrumentation and control equipment (as well as other equipment) was not maintained in good working order. Workers were not using a status board; they were not verifying conditions with a pressure gauge, because the gauge was inoperable; and their procedures were inadequate because they did not indicate normal pressure and temperature ranges. This event highlights the problems that may occur when operators are set up for failure by specific plant or facility conditions (e.g., inoperable or unreliable instrumentation) combined with less than adequate planning. Such situations represent the broad range of conditions and problems associated with routine field operations that can lead to events and event sequences.

- The events involving interlocks (Events I2, I3, I4, and I5 in Table 3-8 and appendix Table 6) indicate that interlocks by themselves will not always ensure worker safety. If workers defeat interlocks or make unauthorized or erroneous modifications to them, then worker safety may be compromised as a result. Conflicting indicators and poor interlock design may also contribute to (even set up) workers operating equipment unsafely.

- For the event involving the spent fuel pit area radiation monitor (Event I6 in Table 3-8 and appendix Table 6), the improper alarm setpoints had existed for a period of time before being identified. For this improper setting, the time interval for the alarms to sound was negligibly longer than it would have been at the required setting. However, this event shows that inadequate operating procedures (i.e., an operating procedure that did not identify the technical specification requirements) and training failures (i.e., failure to train technicians on a revised alarm setpoint methodology) can affect the operation of safety equipment so that it does not function correctly.

- When both safety-related and nonsafety-related equipment are connected to the same plant ethernet network (Event I7 in Table 3-8 and appendix Table 6), nonsafety-related equipment can potentially affect the performance of other equipment (including safety-related equipment) that is on the network. Therefore, the design and configuration of the control system is important when considering safe facility operation.

- For the event in which a crane hoist continued to move upwards uncontrollably (Event I8 in Table 3-8 and appendix Table 6), a maintenance person replaced a component with the wrong component. This event highlights the importance of considering human error in maintenance activities when evaluating the performance of equipment that may be used in normal operations because maintenance activities may lead to initiating events and may play a role in event sequences.

4.1.7 Fires and Explosions

A total of 26 fire/explosion-related events from various operating nuclear facilities were surveyed. The insights from reviewing these events are summarized as follows.

- Electrical cabinet fires (particularly in the range of 4.16 kV to 13.8 kV) appear to warrant special attention at nuclear facilities. Energized circuits carried more energy than that predicted by the heat release rate calculations assumed in some of the safety analyses. Electrical systems that have been carefully designed and laid out are less vulnerable to electrical fires that threaten both safety and nonsafety equipment. Sometimes water is an effective means to extinguish electrical fires if used judiciously.

- Configuration control is an important element of an overall facility fire safety program. Configuration control encompasses multiple aspects, from minimizing the combustibles to supplementing additional fire control measures when existing facility layout is altered, thus diminishing the effectiveness of fire control assumed in the facility safety analysis.

- As evidenced by the Washington Nuclear Project No. 2 event, a-priori system weaknesses may be dormant for a long time until the system is called upon to perform a safety function. Reviewers should pay appropriate attention to such system weakness(es) when reviewing the fire protection systems.

- Regular maintenance of fire protection equipment helps ensure the soundness of its operation. Maintenance activities should not interfere with the intended safety function(s) of fire protection equipment.

4.2 Human Performance Issues

As operating experience shows, human performance is an important part of operations at nuclear facilities. Human performance contributed to the majority of the events reviewed under all of the activities considered for this report. In general, human actions contribute to initiating events and can play an important role in mitigating potential accidents and controlling the evolution of events after initiation of a potential accident sequence. Human actions are also important in preinitiator situations because of the potential for latent failures in hardware (e.g., safety-system components), such as errors in maintenance and inspection. In addition, administrative controls for safety influence human performance and in turn may rely on human performance for successful execution.

The ability to develop specific human performance insights from existing event records, such as those reviewed in this effort, is limited by the nature and level of detail available in the data sources. Most database systems (including those reviewed here) are not designed to effectively identify the causes of human errors. As a consequence, many human-caused events are attributed to procedures, training, work practices, or attention to detail, with associated fixes, such as formal reminders to personnel to follow procedures. Research[1] has shown that such fixes may temporarily solve the immediate problem but usually do not solve the problem in the long term or address the underlying cause for the problem.

The following general insights are offered from this effort. First, the distribution or percentage of events that were attributed to human causes is consistent with the nuclear power industry [i.e., approximately 70 percent (Gertman, et al., 2001)] and human errors, generally, across industries. Second, the majority of potential causes of operator errors were some kind of planning failure. Finally, a noticeable number of the human errors identified involved work-arounds or shortcuts. The “fixes” for such problems were changes to procedures or training.

While the event sources used in this study could not provide supporting details, event analyses in other studies have identified a variety of causes that may be applicable to fuel handling activities. Examples of such potential reasons for work-around (all of which are natural and expected human responses) are

- The desire to find a more efficient and faster approach to perform required activities

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[1] For example, see Reason, J., *Managing the Risks of Organizational Accidents*. Burlington, Vermont: Ashgate Publishing Limited. 1997.

- The need to balance competing, and sometimes even conflicting, priorities or demands (e.g., meeting schedules for completing activities versus performing activities "by the book")

- Conflicts between formal procedures and standard practice (e.g., the end result of one of the noted reasons and/or a poorly designed procedure that doesn't take into account operator boredom or complacency due to the repetitiveness of tasks)

- Complacency

- Opaqueness of procedure steps or training information (especially with respect to seemingly needless or seemingly over-conservative steps).

4.3 Summary of Risk Insights

The majority of the events identified in this operating experience review involved human error; the most frequently identified potential causes were less than adequate planning or procedures, work-arounds,[3] less than adequate maintenance or inspection, misleading or unreliable instruments, and less than adequate training. None of the events identified posed any health or safety hazards to the public; however, safety systems were compromised due to loss of electrical power (e.g., emergency diesel generators being unavailable due to improper maintenance), and two events occurred where workers experienced a dose (both involving interlocks). These two events occurred when workers routinely defeated interlocks and experienced conflicting indications in combination with a poor interlock design. Therefore, instrumentation and control systems, and in particular interlocks, may be an important area for technical staff to focus their review efforts. In addition, human activities (both operations and maintenance) and human interactions with equipment may be important areas to focus reviews. Although several events did not result in a consequence to workers or the public, this operating experience review highlights the extent to which human error has been involved in events at facilities having operations similar to those of the GROA.

NRC may want to discuss with DOE ways in which the design of the GROA facilities could prevent or mitigate the types of events identified in this operating experience review. Discussions related to prevention of events may include (i) identification of interlocks used in the facilities and measures to prevent them from being defeated; (ii) means to prevent fires and explosions; (iii) means to detect weld cracks; (iv) the potential for loss of helium atmosphere in a container; (v) acceptance criteria for containers, such as waste packages and transportation, aging, and disposal canisters; (vi) measures in place to prevent dropping of loads into or near the spent fuel pool; (vii) reliability of lifting and moving equipment such as cranes and the spent fuel transfer machine; and (viii) consideration of common human performance issues in the design of facilities and operations and the preclosure safety analysis. Discussions related to mitigation of events may include (i) measures to ensure the availability of equipment, such as HEPA-filtered HVAC exhaust systems and related essential electrical power, and (ii) means to mitigate the loss of multiple safety systems as a result of fires and explosions.

Table 4-1 summarizes the risk insights related to design flaws or operational hazards resulting from this review.

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[3]Alternative method of accomplishing a task when the normal or specified method cannot be used.

Table 4-1. Summary of Risk Insights	
Hazard	**Description**
Dropping a fuel assembly in the spent fuel pool	Human error is the implied root cause of these events. Procedural problems (either operators did not follow procedures or procedural guidance was less than adequate) are the most frequently cited root cause.
Fuel cladding degradation	Human error resulting in the loss of helium atmosphere in a canister or cask may result in heat up of the contained fuel, which can potentially lead to fuel cladding degradation.
Heavy load drop	A majority of the events with potential radiological consequences involved operator errors. However, all events were mitigated by preventive actions, and none resulted in serious radiological consequences.
Loss of filtration	Ventilation dampers appeared to be the most prone to failure, which can result in the ventilation system becoming inoperable. Inadequate procedures or misinterpretation of existing procedures was the next highest cause of ventilation-related problems.
Loss of backup electrical power	A number of the events occurred coincidently with maintenance activities or when other related or unrelated structures, systems, and components were returned to service. In some cases, configurations were not fully or properly restored after maintenance or tests, and the problem did not show up until a later abnormal condition occurred.
Loss of interlock function	The events involving interlocks indicate that interlocks by themselves will not always ensure worker safety because they can fail, be inadequately designed, or be defeated by operators. Conflicting indicators and poor interlock design may also contribute to workers operating equipment unsafely.
Loss of safety systems due to fire	Fires have the potential to compromise multiple safety systems. For example, fires originating in electrical cabinet(s) may cut off the electricity supply to structures, systems, and components important to safety.
Human actions resulting in preinitiators	Human actions are significant contributors to preinitiator events because of the potential for latent failures in hardware (e.g., safety-system components), such as errors in maintenance and inspection.
Defeat of administrative controls	Administrative controls for safety influence human performance, and in turn, may rely on human performance for successful execution.

5 POTENTIAL IMPORTANT TO SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS

Based on this review of operating experience, several structures, systems, and components were identified as likely to play a role in preventing an event sequence, or in mitigating the consequences of an event sequence. These are identified in Table 5-1 based on the definition of important to safety in 10 CFR 63.2. This definition uses both the terms “preventive” and “mitigative” to describe structures, systems, and components important to safety. Based on this review and associated staff judgments, cranes, spent fuel pools, instrumentation and control systems (to include interlocks), and canisters and casks are associated with preventing event sequence occurrence. HVAC systems (to include HEPA filtration) and electrical systems supplying power to HVAC systems are associated with mitigating the consequences of an event sequence once it has started by maintaining the capability to filter airborne radioactive particulates. Similarly, fire protection systems have the capability of mitigating the consequence of an event sequence involving fire and or explosion.

The NRC regulation at 10 CFR 63.112(e) specifies that the analysis of important to safety structures, systems, and components also identifies measures to ensure the availability of safety systems. Operating experience shows that some of these measures are likely to be administrative controls.

Table 5-1. Potential Important to Safety Structures, Systems, and Components

Potential ITS SSC	Preventive	Mitigative
Cranes to include overhead bridge cranes used to transfer casks and the spent fuel transfer machine used in the spent fuel pool	✔	
Spent fuel pool	✔	
Heating, ventilation, and air conditioning systems to include HEPA filtration		✔
Electrical systems to supply power to the heating, ventilation, and air conditioning systems		✔
Fire protection systems (both detection and suppression)		✔
Instrumentation and control systems to include interlocks	✔	
Canisters and casks	✔	

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APPENDIX
EXPANDED TABLES OF OPERATING EXPERIENCE SEARCH RESULTS

Table 1. Detailed Incidents Related to Handling Fuel Assemblies in a Spent Fuel Pool								
ID No.	**Source**	**Reference No.**	**Report or Event Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
1	LER*	0292000002	E†: 03/15/00 R‡: 04/15/00	The maximum fuel assembly travel height over "ungrated" spent fuel pool racks is 0.15 m [6 in] over the top of the rack. The fuel in this incident was lifted 0.33 m [13 in] above the racks, which is outside the design basis.	No consequences	Yankee Nuclear Power Station	Failure to adequately incorporate design basis information into plant procedures	This event shows the importance of enforcing the design basis in the plant procedures.
2	LER	2131997004	E: 02/19/97 R: 03/19/97	On February 19, 1997, with the plant defueled, a preliminary evaluation determined that fuel assembly loads exceeding the 748-kg [1,650-lb] limit could have been potentially moved over the spent fuel assemblies.	None	Haddam Neck Plant	The root cause was removing the exclusion for fuel assemblies from the technical specification requirements in 1989, because at that time, the weight of fuel assemblies would never exceed 748 kg [1,650 lb]. Since then, new core designs have resulted in heavier fuel assemblies, which have reduced the safety margin.	Corrective action was taken to prohibit any movement over the fuel pool. The long-term corrective action will be to amend the technical specifications in accordance with the guidance for improved Standard Technical Specifications to restore the exclusion for the fuel assemblies.
3	LER	3821997018	E: 04/28/97 R: 06/27/97	On April 28, 1997, during a fuel shuffle, the new fuel assembly disengaged from the Spent Fuel Handling Tool and dropped 0.13 m [5 in].	This accident did not pose any health or safety hazards to the public.	Waterford Steam Electric Station	Human error. The spent fuel handling tool was found approximately 75 percent open and locked.	1. There were no administrative controls in place to ensure optimum tool orientation. As a result, the locking device was oriented away from the operator of the Spent Fuel Handling Machine. 2. Improve the training given to operators—increase the detail on mechanics of operating grapple on Spent Fuel Handling Machine
4	LER	4001999001	E: 01/16/99 R: 02/05/99	Spent fuel pool water level not maintained above 7.01 m [23 ft]. With the unit at 100 percent power, personnel noticed that one of the boiling water reactor assemblies that was being moved did not fully seat in the storage rack.	None	Harris Nuclear Plant	The root cause for this event was the ambiguous guidance regarding channel fastener tolerances and the fact the fasteners could bend under specific circumstances.	

Table 1. Detailed Incidents Related to Handling Fuel Assemblies in a Spent Fuel Pool (Continued)								
ID No.	**Source**	**Reference No.**	**Report or Event Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
5	OE Summary§	ORPS Report ID—BBWI-FUELRCSTR-2004-0006 (http://www.hss.energy.gov/csa/analysis/oesummary/oesummary2004/oe 2004-23.pdf)	E: 11/24/04 R: 11/29/04	On November 24, 2004, at Idaho Nuclear Technology and Engineering Center, two lifting slings were damaged during an operation to lift and move 26,308-kg [29-ton] fuel shipping casks. The operators had noticed that the slings were damaged.	None (this was an exercise)	Idaho Nuclear Technology and Engineering Center	Nylon strings were improperly attached to a steel lifting attachment, which cut into the slings.	The lessons learned are: 1. Follow approved lift plan and ensure the rigging selection and configuration are correct 2. Wire Rope and Slings of DOE-STD-1090-2004, “Hoisting and Rigging Standard” should be reviewed during training of operators
6	OE Summary	ORPS Report RL-PHMC-SNF-2004-0017 (http://www.hss.energy.gov/csa/analysis/oesummary/oesummary2004/oe2004-13.pdf)	E: 03/05/04 R: 06/28/04	On May 3, 2004, at Hanford, the design authority noticed that a chain hoist used to move Spent Nuclear Fuel within the K-West Basin did not have a current inspection sticker. The procedure requires operations personnel to perform a preuse check on the hoist that includes ensuring that certifications are current. That inspection had not been performed when the hoist was used on April 30.	None	Hanford Plant/K-West Basin	Operator negligence.	A special report on Hoisting and rigging is available at www.eh.doe.gov/HR_INPO_Style_FinalDraft_01-20-04.pdf Retrain operators — Make them aware of the dangers of not following standard procedures
7	NRC	2002-009	E: 03/24/01 R: 02/13/02	Potential for Top Nozzle Separation and Dropping of a Certain Type of Westinghouse Fuel Assembly—A certain type of Westinghouse fuel assembly may drop during movement. Similar events had occurred at Prairie Island in 1981 and several events at foreign plants in the 1980s. The fuel assembly had separated at the top bulge joint that connects the stainless steel grid sleeves to the Zircaloy guide tube.	Because the assembly bottom nozzle was already in the cell, the falling assembly did not contact any other fuel assemblies or the rack structure. No fission gas activity was detected afterwards, indicating that none of the fuel rods in the assembly had been fractured by the drop.	North Anna Power Station of Virginia Electric and Power Company	Hot cell metallography after the earlier events indicated that the likely root cause was intergranular stress corrosion cracking accelerated by the presence of chlorides, fluorides, and sulfates.	

*Note: LER–NRC. “Licensee Event Reports Database.” Washington, DC: NRC. <https://nrcoe.inel.gov/lersearch> Refer to report number indicated in description column.
†Note: E–Event Date
‡Note: R–Report Date
§Note: OE Summary–Operating Experience Summary
‖NRC. “Information Notices.” Washington, DC: NRC. <http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/>

Table 2. Detailed Incidents Related to Opening and/or Closing of Canisters or Casks								
ID No.	**Source**	**Reference No.**	**Report or Event Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
1	LER*	387200200500	E†: 07/26/02 R‡: 08/26/02	Dry Fuel Storage Canister Filled With Incorrect Gas Due To Human Error—Due to human error, argon was used instead of helium to fill a Dry Shielded Canister. This event was caused by human error. Argon is used as a welding shield gas, and helium is used as a heat transfer media in the canister.	The potential for heat up of the fuel and cladding damage was the issue for this event because argon has approximately 1/10 the thermal conductivity of helium. However, an analysis was conducted, and it was determined that there was no fuel damage or radiological releases for this particular event. The analysis determined that the fuel in this Dry Shielded Canister could reach 381 °C [717 °F], which was below the design basis short-term fuel cladding temperature limit of 570 °C [1,058 °F] and the long-term fuel cladding temperature limit of 421 °C [790 °F].	Susquehanna Steam Electric Station—Unit 1	Human error. Previously, helium cylinders were supplied in cylinders that were dark blue, and argon was supplied in teal-colored cylinders. A change in the compressed gas vendor resulted in a change in the color of the helium cylinder to teal as well. For this event, both argon and helium cylinders were on the same cart, and personnel did not read the labels to verify that they were using helium instead of argon.	None
2	NEI Dry Storage Lessons Learned‖	Response to CAL 97-7-001	E: 03/1995 to 03/1997 R: 07/30/97	Ventilated Storage Cask Weld Cracking Evaluation in Response to CAL 97-7-001—Cracks formed on the lid welds of the Ventilated Storage Cask Multi-Assembly Sealed Basket. In addition, there was an issue of delayed cracking in which cracks could appear some time after a weld was complete and could therefore potentially go unnoticed during the weld examinations.	If cracks form, then the helium gas inside the cask can escape. The loss of the helium atmosphere inside the cask could result in cladding degradation.	Sierra Nuclear Corporation: Palisades, Point Beach, Arkansas Nuclear One	Root causes of the weld cracking were: 1. "...an existing condition in the shell material...," 2) "...fit-up problems with the structural lid and backing ring and by the presence of moisture near the shield lid-to structural lid seal weld, and 3) "...hydrogen-induced [cracking], resulting from a combination of high residual stresses due to joint restraint, presence of hydrogen in the weld wire, and a shell material that is susceptible to heat affected zone (HAZ) hydrogen cracking."	For the instance of a delayed crack, it was determined that the expected maximum delay times associated with each weld were less than the time interval from completion of welding to inspection of the weld, and therefore, a delayed crack would appear in time to be detected during the welding inspection. To correct the problems associated with weld cracking, the following changes were identified to the process: 1) use of a 93 °C [200 °F] preheat for all of the lid welds, which would be maintained for a minimum of 1 hour after completion of the final pass of the weld and 2) "... the use of welding consumables with low hydrogen levels to provide reasonable assurance against hydrogen-induced cracking."

Table 2. Detailed Incidents Related to Opening and/or Closing of Canisters or Casks (Continued)								
ID No.	Source	Reference No.	Report or Event Date	Description	Consequences	Plant or Site	Root Causes	Notes
3	NRC	96-34	E: 05/28/96 R: 05/03/96	Hydrogen Gas Ignition during Closure Welding of a VSC-24 Multi-Assembly Sealed Basket—A hydrogen gas ignition occurred during the welding of the shield lid on a ventilated storage cask multiassembly sealed basket which contained spent fuel assemblies.	"The gas ignition displaced the shield lid (weighing about 2,898 kilograms [6,390 pounds]), leaving it in place but tipped at a slight angle, with one edge about 7.6 centimeters [3 inches] higher than normal." There was no evidence of damage to the spent fuel assemblies.	Point Beach Nuclear Plant	The investigation into the possible sources of hydrogen focused on a zinc-based coating applied to the internal surfaces of the multiassembly sealed basket. The consideration was that "zinc may have reacted chemically with the acidic borated water from the spent fuel storage pool to produce hydrogen."	Following the ignition, the multiassembly sealed basket was continuously purged with nitrogen, fully flooded, and then returned to the spent fuel pool.
4	DOE Office of Health, Safety and Security Lessons Learned Database	2003-OH-WVNS-001	E: 08/28/02 R: 02/27/03	Shipping Containers Release Internal Pressure and Radiation Dose Changes—Nineteen shipping containers were being prepared for offsite disposal when the sound of air escaping was heard from the lids of three containers. "The contents of a number of the containers consisted of canisters which had held spent nuclear fuel in a storage pool for approximately 30 years. The canisters were encrusted in 'barnacles' . . . "	The result of barnacles shifting, breaking, and flaking off caused the contact dose rate to increase.	West Valley Nuclear Services Company	"During the root cause analysis, it was discovered that the container supplier had failed (as required by acceptance criteria), to flush grind welds on the lid sealing surface which did not allow full gasket compression.	Lessons identified: 1) "Minimize length of outdoor storage to prevent deterioration or excessive environmental exposure." 2) "Ensure that work documents include closure instructions and acceptance criteria provided by the container supplier or engineering when container size is different and could affect what is normally a proven and acceptable work method (skill of the craft)." 3) "Foam containers with high dose contents that may shift or have flaking radioactive deposits can become movable debris in the container."

*Note: LER—NRC. "Licensee Event Reports Database." Washington, DC: NRC. <https://nrcoe.inel.gov/lersearch>
†Note: E—Event Date
‡Note: R—Report Date
§DOE. "DOE Office of Health, Safety, and Security Lessons Learned Database." Washington, DC: DOE. <https://www.hss.energy.gov/csa/analysis/ll/oellproducts.html>
‖Nuclear Energy Institute. Nuclear Energy Institute Dry Storage Information Forum. New Orleans, Louisiana: Nuclear Energy Institute. May 2001.

Table 3(a). Detailed Incidents Related to Heavy Load Lifting: Licensee Event Reports and Information Notices								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
1	LER*	2451996016	02/26/96	Heavy Loads Suspended over Irradiated Fuel in the Spent Fuel Pool. With the plant shut down and the reactor in the cold shutdown condition, the spent fuel pool gates were suspended over irradiated fuel in the spent fuel pool while they were being moved from their storage hangers. The Millstone station procedure for control of heavy loads, MP 790.4, indicates a safe load path for the fuel pool gates from their storage location to their operating position or their floor lay-down area. The procedure does not show a safe load path over irradiated fuel. When plant maintenance personnel questioned the safe load path, engineering personnel advised them that a heavy load evaluation was not performed for this lift. The control room shift manager was contacted, and the reactor building overhead crane was tagged out of service until the issue was reconciled. This event was determined to be an unanalyzed condition that significantly compromises plant safety.	There were no safety consequences as a result of this event.	Millstone 1	A unanalyzed condition that significantly compromises plant safety.	An engineering analysis was performed to provide a single failure-proof handling and safe load path for fuel pool gate movement. Special procedure was developed to implement the single failure-proof handling and safe load requirements.
2	LER	2501996009	7/29/96	Failure to Reflect Heavy Load Design Information in Procedural Controls. On July 29, 1996, Florida Power & Light Company discovered that the location of Heavy Load exclusion areas was not documented correctly in procedures controlling the lift of heavy loads. Documents sent to the NRC in 1982 as part of the review of a Technical Evaluation Report changed the location of the safe load path for heavy loads from that described in the original response in 1981. This change in the safe load path description was not reflected in Administrative Procedure 0–ADM–717. As a result, heavy loads have been lifted over restricted areas without the procedurally required evaluation and approval. A discrepancy was also found in the procedure definition of the size of a heavy load.		Turkey Point 3 & 4	The cause of the event was the lack of a satisfactory process in 1982 for the capture of revisions to design documentation unrelated to physical modifications.	Incorporation of design information into operating procedures is governed by a commitment tracking process. This process was started in 1984.

Table 3(a). Detailed Incidents related to Heavy Load Lifting: Licensee Event Reports and Information Notices (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
3	LER	2611997005	04/22/97	Condition Outside Design Basis Spent Fuel Shipping Cask Unreviewed Safety Question. On April 22, 1997, with H. B. Robinson Steam Electric Plant 3 Unit No. 2 operating at 100 percent power, the results of an investigation revealed that certain spent fuel shipping cask handling activities had been conducted outside the design and licensing basis of the plant. Specifically, an IF–300 spent fuel shipping cask configured for fuel loading by removing the cask valve box covers. The loaded cask is then lifted with a nonsingle failure-proof crane from a decontamination facility to the cask rail car, where the cask valve box covers are then installed. Lifting the cask with the nonsingle failure-proof crane with the valve box covers removed is not covered by the shipping configuration drop analysis. An evaluation was completed that concludes this condition represents an unreviewed safety question outside the design basis of the plant.	There have been no significant adverse safety consequences associated with this condition.	Robinson 2	This condition was caused by inadequate evaluations for cask handling procedures. Personnel conducting these evaluations failed to identify the limitations of the drop analysis as it applied to cask handling operations, versus shipping configuration accident conditions. Procedures for spent fuel cask handling operations have been administratively placed on hold, pending resolution of this condition.	A postulated spent fuel shipping cask drop with the valve box covers removed could lead to an offsite release that exceeds the “no release” result of a cask drop specified in the licensing basis. However, results of the final evaluation concluded that the offsite doses resulting from a postulated cask drop with a less than fully secured cask are a small fraction of the 10 CFR 100 limits and the acceptance criteria in the Standard Review Plan.
4	LER	2752002003	04/30/02	Unanalyzed Condition Due to Heavy Load Movement Over a Restricted Area. With the unit in Mode 5 (cold shutdown), a main turbine low pressure, 63,503-kg [70-ton] low pressure turbine cover was moved over a Unit 1 turbine building heavy loads restricted area above the diesel generators and 4kV vital bus ventilation, contrary to Inter-Departmental Administrative Procedure MAI .ID14, “Plant Crane Operating Restrictions.”	Violation of procedures. No consequences.	Diablo Canyon 1 & 2	The primary cause of the event was the failure of utility and outage contract turbine maintenance personnel to comply with procedures for movement of loads within a heavy loads restricted area.	Corrective actions to prevent recurrence include a review of this event for all crane operator and rigging qualifications; having a person knowledgeable in the heavy loads rigging program present at the prejob tail boards; and modification of plant procedures, requiring engineering to ensure, during their evaluations, that the movement of heavy loads will not violate restrictions for load handling.
5	LER	2821999005	5/8/99	Containment Inservice Purge System Not Isolated During Heavy Load Movement Over Fuel. On May 8, 1999, with Prairie Island Unit 1 in a refueling shutdown (Mode 6) and Unit 2 operating at 100 percent power, the plant staff identified that during the just completed performance of D58.1.6, Reactor Upper Internals Replacement, the reactor upper internals, a heavy load, was transported over the open fueled reactor vessel with the Containment Inservice Purge system operating. A procedure step in D58.1.6 for closing the containment inservice purge system containment isolation valves was inadvertently missed and was not discovered until after the upper internals had been set in the reactor vessel.	No adverse impact on the health and safety of the public and was of negligible safety significance.	Prairie Island 1	Failure to follow the procedure.	Analysis of the event concluded that this event has had no adverse impact on the health and safety of the public and was of negligible safety significance.

Table 3(a). Detailed Incidents related to Heavy Load Lifting: Licensee Event Reports and Information Notices (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
6	LER	3061997001	02/07/97	Transporting a Heavy Load over Irradiated Fuel or Safe Shutdown Equipment without Establishing the Required Conditions. On February 4, 1997, it was determined that 22 Reactor Coolant Pump upper bracket and rotor (heavy load 19,051 kg [21 tons]} was moved over irradiated fuel on February 3, 1997, without a specific load handling procedure defining the safe load path and without containment isolated. The movement of this heavy load in containment did not follow the reactor building safe load path requirements stated in Operations Manual Section D58, Control of Heavy Loads. D58 states that “With the reactor head removed, loads greater than [953 Kg] 2,100 lb shall not be moved within [4.6 m] 15 horizontal feet of the irradiated fuel without specific written procedures per step 5.3.5 and containment isolation requirements satisfied. Neither of these provisions were satisfied. Following this event, two subsequent heavy load lifts were recognized as not meeting the intent to control heavy load lifts where safe shutdown equipment could be affected.		Prairie Island 1 & 2	Noncompliance with procedures and human factor considerations of the implementing procedures.	Corrective actions, taken and planned, included procedure compliance, adequacy of the heavy loads program, human factor considerations of the implementing procedures, and training and qualifications of the individual(s) with responsibilities for implementation of the program features.
7	LER	3131996004	03/06/96	Load In Excess Of Technical Specifications Weight Limit Moved Over Fuel Stored In The Spent Fuel Pool As A Result Of Conflicting Procedural Guidance Which Resulted From An Inadequate Review During Procedure Development. On March 6, 1996, Arkansas Nuclear One personnel were lifting the cask loading pit gate, which weighs approximately 1,818 kg [4,000 lb], in preparation for storing it on the edge of the spent fuel pool. Due to the presence of steel tabs on top of the gate, it had to be rotated 180° to be stored. As the craft personnel were rotating the gate, a Senior Reactor Operator observed that it was positioned partially over the fuel in the pool. The operator immediately halted the work and directed the craft personnel to reposition the gate and move it to its storage location without passing it over any fuel.		Arkansas 1	The cause of this event was inadequate procedure, which contained conflicting information regarding the movement of spent fuel pool gates. The root cause of the event was an inadequate review during development of the procedure, which failed to identify that the special instruction contained in the attachment conflicted with the Technical Specifications.	Crew briefings were conducted regarding this event, and interim controls were established to ensure that an operator is present when the cask loading pit gate is moved pending procedure revision to remove the conflicting information.

Table 3(a). Detailed Incidents related to Heavy Load Lifting: Licensee Event Reports and Information Notices (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
8	LER	3152000005	06/04/00	Control of Auxiliary Building Crane Main Load Block Over Spent Fuel Pool. On June 28, 1999, the spent fuel pool ventilation system was declared operable, but degraded because the system could not react quickly enough to a high radiation signal to close the charcoal filter bypass dampers and prevent radioactive gases from a fuel handling accident from being released to atmosphere without passing through the charcoal filters. To maintain the system in an operable status, compensatory actions were required. These actions included placing the spent fuel pool ventilation system in the charcoal filter mode of operation prior to movement of fuel or any load within or over the spent fuel pool and procedure changes. On June 4, 2000, during performance of crane interlock testing, the east Auxiliary Building crane was operated over the spent fuel pool without the spent fuel pool ventilation system in the charcoal filter mode of operation as required by the compensatory actions and without the main load block deenergized as required by Technical Specification 3.9.12 footnote. On June 5, 2000, this condition was determined to be reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by technical specifications.		Cook 1 & 2	The apparent cause for this event is the failure to establish adequate administrative controls for the degraded condition, with a contributing cause of an operator knowledge weakness.	The crane interlock verification procedure will be revised; training will be provided to Operations personnel relative to the requirements of Technical Specification 3.9.12.
9	LER	3251997004	05/06/97	Spent Fuel Shipping Cask Handling Activities. On May 6, 1997, Units 1 and 2 were operating at rated power when it was determined that spent fuel shipping cask handling activities had been conducted outside the design bases. Specifically, the site procedures controlling the lifting and loading of an IF-300 spent fuel shipping cask prescribe the use of rigging that is not single failure proof during transfer from the tilting cradle to the secondary yoke, contrary to existing analyses. This transfer occurs on the 6.1 m [20 ft] elevation of the reactor building at a lift height of approximately 2.1 m [7 ft]. In addition, during previous spent fuel shipping cask handling activities when the nonsingle failure-proof lift condition existed, the safety-related valve box covers were not installed. Current spent fuel analyses bound a 9.1 m [30 ft] cask drop with the safety-related valve box covers installed. There is not an existing analysis for a spent fuel shipping cask drop without the valve covers installed. This event was caused by an incomplete understanding of the scope of the NEDO-10084-4, Vectra IF-300 Shipping Cask Consolidated Safety Analysis Report and NUREG–0612, “Control of Heavy Loads at Nuclear Power Plants.”		Brunswick 1 & 2	The discrepancy between the design bases and existing site procedures related to IF–300 spent fuel shipping cask handling is attributed to a misunderstanding of vendor procedures, vendor safety analysis, and elements of NUREG–0612.	Site procedures for spent fuel shipping, handling, and receiving have been placed on administrative hold pending resolution of this issue and NRC review and approval of a related change to the licensing/design bases. A load drop analysis for the postulated spent fuel shipping cask drop accident will be performed in accordance with the applicable guidance of NUREG–0612. In addition, a review of the applicable regulatory requirements and site procedures related to spent fuel shipping cask handling will be performed to ensure consistency.

Table 3(a). Detailed Incidents related to Heavy Load Lifting: Licensee Event Reports and Information Notices (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
10	LER	3341997028	08/25/97	Spent Fuel Pool Crane Interlocks and Physical Stops Not Tested Prior to Use in Accordance with Technical Specifications. On August 25, 1997, at 1,400 hours with Unit 1 in Mode 1 at 100 percent power, the Control Room identified that the spent fuel pool crane had been moved over the storage pool during relamping operations without first performing a surveillance procedure required by technical specifications. Technical Specification Surveillance Requirement 4.9.7 requires that the crane interlocks and physical stops that prevent crane travel with loads exceeding 1,364 kg [3,000 lb] shall be demonstrated operable within 7 days prior to crane use and at least once per 7 days thereafter during crane operation. Contrary to technical specification, the spent fuel pool crane was used without verifying that the crane interlocks and physical stops were operable. This is an operation prohibited by technical specifications and is reportable pursuant to the requirements of 10 CFR 50.73(a)(2)(i).		Beaver Valley 1 & 2	The apparent causes of this event were 1) Procedures—Procedures that which control the operation and checkout of the spent fuel pool crane during non-refueling evolutions are inadequate. 2) Program/Process—Work Planning was unaware of the requirement to perform surveillance testing prior to crane use, and the requirement was not included in the work package. 3) Equipment Labeling—Large metal placards mounted on the crane pertaining to movement of heavy loads were misleading, and the crane safety switch/key was not labeled to identify safety prerequisites. 4) Training/Qualification—The Mechanical Maintenance Technician operating the crane lacked knowledge of the requirement to perform surveillance testing prior to crane use.	There was no actual movement of heavy loads over the spent fuel pool. The crane was relocated to a position not over the fuel pool, and the required surveillance testing was performed to satisfy the Technical Specification Surveillance Requirement.
11	LER	3362001007	10/22/01	Movement of Heavy Loads Not Addressed in Procedure. It has been identified that no safe load path exists for lifts of new fuel shipping containers and spent resin casks at Millstone Unit No. 2 in the area of the cask washdown pit and the associated lifting device is not single failure proof. Safety-related commodities are located both in the pipe trench below the cask pit floor and on the west wall of the railroad access bay. Load lifts on the order of 7.3 m [24 ft] are required to bring material into and out of the spent fuel pool area via this load path. Previously, it was identified that a 45,359-kg [50-ton] reactor coolant pump motor was stored in the cask washdown pit and that the drop of this motor would result in failure of the floor and potential damage to safety-related components in the pipe trench. Remedial corrective actions taken to date include marking the location of the pipe trench on the railroad access bay floor and removal of the reactor coolant pump motor from the cask washdown pit using a NUREG–0612 compliant lift.		Millstone 2	The root cause for the failure to identify heavy load paths is inadequate engineering work practices in the Millstone engineering department in the area of programs.	

Table 3(a). Detailed Incidents related to Heavy Load Lifting: Licensee Event Reports and Information Notices (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
12	LER	3441999001	2/26/99	An Unanalyzed Movement of a Non-fuel Loaded Transfer Cask. On February 26, 1999, in the Fuel Building of the permanently defueled Trojan Nuclear Plant, an Independent Spent Fuel Storage Installation rigging crew was performing preoperational testing of the independent spent fuel storage installation transfer cask lift system in accordance with an approved procedure. During the test, the test-weight loaded transfer cask was lifted above its analyzed limit while over the dry cask load pit, which should also have contained water to minimize impact loading on the cask load pit. There was no fuel in the transfer cask or in the cask load pit. The gate between the spent fuel pool and the cask load pit was closed. There was no component failure during this event. This event is reported in accordance with 10 CFR 50.73(a)(2)(i)(B) as an operation or condition prohibited by Technical Specifications. A procedure limited transfer cask height along a safe load path leading to and from the cask load pit, but was misinterpreted as not limiting transfer cask height directly over the cask load pit. Also, the procedure did not implement a transfer cask drop event calculation assumption that the cask load pit contain a level of water.		Trojan	The event root causes were personnel error, inadequate training, and an inadequate procedure.	A procedure change was made to clarify that transfer cask lift height restrictions apply directly over the cask load pit and to require a specified cask load pit water depth for certain moves over the cask load pit.
13	LER	3461996005	04/16/96	Inadequate Control of Heavy Loads in the Containment Building. On April 16, 1996, a potential condition adverse to quality report documented lifting the reactor vessel head lifting tripod and improperly traversing a portion of the open reactor vessel with fuel in the reactor. The reactor vessel head lifting tripod is considered a heavy load and is procedurally restricted from movement over the open reactor vessel with irradiated fuel in the reactor. The reactor vessel head lifting tripod was moved from the west secondary shield wall, across the northeast portion of the reactor vessel to the incore tank area. Further review determined that this event involved a postulated drop scenario that was not bounded by previous heavy load evaluations—a condition outside the design basis.		Davis-Besse	Lack of knowledge on what constituted a heavy load and a safe lift load path over fuel caused the inadequate evaluation of this reactor vessel head lifting tripod lift.	Immediate corrective action included direction from the Plant Manager to the Outage Directors and training of affected personnel to reemphasize load path restrictions in the containment vessel. During the investigation, it was discovered that other lifts occurred that encroached upon the heavy load exclusion zone at the containment vessel periphery. Commitments for handling of heavy loads with the polar crane will be reviewed, and additional corrective actions will be implemented prior to the next refueling outage as determined necessary.

Table 3(a). Detailed Incidents related to Heavy Load Lifting: Licensee Event Reports and Information Notices (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
14	LER	4001997004	03/04/97	In-Plant Spent Fuel Cask Handling Activities. On March 4, 1997, with the plant operating in Mode 1 at 100 percent power, an investigation determined that spent fuel cask handling activities were conducted outside of the design and licensing basis of the plant. Investigation revealed that the Harris Nuclear Plant evaluation of a cask drop to a flat surface, documented in Field Safety Analysis Report 15.7.5.2, did not consider the potential consequences of dropping or otherwise damaging a loaded spent fuel cask after it has been prepared for unloading; that is, with the cask head detensioned and valve box covers removed. Consequently, the existing cask drop evaluation in Field Safety Analysis Report 15.7.5.2 does not address a potential drop of a cask in a less than fully secured condition.		Harris	Unreviewed safety question. Carolina Power and Light Company will review the subject event with personnel responsible for making changes to CM–M0300 and other procedures to be used for loading and unloading NRC-approved transportation packages.	This event was caused by an incomplete understanding of the purpose and scope of the IF–300 Cask Safety Analysis Report and a misconception that following the requirements of the Cask Safety Analysis Report would maintain the cask in a condition that was analyzed for a free drop of 9.1 m [30 ft], through air, onto a flat, essentially unyielding horizontal surface. A contributing cause is inadequate 10 CFR 50.59 evaluations for procedure CM–M0300 that failed to identify that the cask drop analysis conducted to confirm that the cask could withstand a 9.1 m [30 ft] free drop without a loss of integrity only applied to a cask in a fully secured, ready-for- shipment (10 CFR 71-compliant) condition. This is an unreviewed safety question pending NRC review and approval of the unreviewed safety question submittal.
15	LER	4821997026	11/09/97	Heavy Loads Moved in Containment Outside of Heavy Load Analysis Requirements. During a review of Wolf Creek Nuclear Operating Corporation's heavy load report, Wolf Creek Nuclear Operating Corporation personnel identified that during past outages, heavy loads were moved in the Containment Building in a manner that was inconsistent with the heavy load analysis assumptions. Specifically, Wolf Creek Nuclear Operating Corporation's analysis assumes that both trains of residual heat removal will be operable in Modes 5 and 6, yet Wolf Creek Nuclear Operating Corporation Technical Specifications allow one train to be operable in Mode 5 if the loops are filled and the secondary side water level of at least two steam generators is >10 percent of the wide range. Wolf Creek Nuclear Operating Corporation met the technical specification requirements, but did not recognize the analysis assumptions for residual heat removal when moving heavy loads in containment. A review of the control room logs from Refuel IX, Fall 1997, confirmed that with one train of residual heat removal inoperable in Mode 6, heavy loads were moved over the one operable train of residual heat removal. Mode 5 is Cold Shutdown and Mode 6 is Refueling.		Wolf Creek	The root cause of this event is inadequate procedural guidance. Procedure AP 14–001 lacked adequate restrictions in that it did not reflect the analyses' assumption that both trains of residual heat removal would be available. The discrepancies between AP 14-001 and Wolf Creek Nuclear Operating Corporation-4 were not identified by any of the parties involved in the development, review, or handling of these documents. The cause of these discrepancies are historical in nature and could not be determined.	Corrective action included revision of the controlling procedure to be consistent with the analysis.

Table 3(a). Detailed Incidents related to Heavy Load Lifting: Licensee Event Reports and Information Notices (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
16	LER	4831998008	8/14/98	Heavy Load Movement Discrepancy. During review of the Callaway Plant heavy load report, Callaway Engineers determined that during past plant outages, heavy loads have been moved in the Containment Building in a manner inconsistent with the heavy load analysis assumptions. Specifically, Callaway's analysis assumes both trains of residual heat removal will be operable in Modes 5 and 6. Callaway technical specifications only require one operable train of residual heat removal in Mode 5, with reactor coolant system loops filled, or in Mode 6, with greater than 7.01 m [23 ft] above the reactor vessel flange. Callaway has met the technical specification requirements, but did not recognize the analysis requirements for residual heat removal while moving heavy loads in containment when in Mode 5 (Cold Shutdown) and Mode 6 (Refueling).		Callaway	Conditions described were caused by failure to identify the discrepancy between the generic letters 81-07 response and plant conditions allowed by technical specifications. The Callaway Heavy Loads Program will be reviewed and revised as required to ensure that the movement of heavy loads in containment does not affect the capability to remove decay heat from the core.	Callaway will submit an amended response addressing the discrepancy between the heavy load analysis and technical specification requirements. The heavy load program will be reviewed to ensure no future deviations from the program.
17	LER	3951999003	04/12/99	The technical specification surveillance requirement requires that each auxiliary hoist and associated load indicator be demonstrated operable within 100 hours prior to start of core alterations by performing a load test. This surveillance was performed satisfactorily on the initial configuration of hoist and load indicator prior to the start of core alterations. On lifting the first control rod drive shaft to be unlatched, the crew noted that the load cell did not indicate the correct weight. The crew did not unlatch the drive shaft. The crew was not aware that the load indicator was selected for "Peak Load" instead of "Continuous" readout, giving a misleading indication. The crew suspected that the existing load cell had failed and a new (second) load cell was requisitioned and installed. The results were unsatisfactory. Both the original and replacement load cells were taken to the calibration lab where it was discovered that both were set for peak load indication. The load cells were changed to continuous readout, both tested satisfactory for accuracy, and crane operation progressed with no impediments.	No significant impact on the plant safety.	Summer	The apparent cause of this situation is a lack of familiarity with the technical specification surveillance requirements and operational procedures for the auxiliary hoist in regards to the load cell indicator. Misunderstanding the feature of the load cell led to the decision that the load cell was faulty and needed replacement.	Procedures associated with core alteration (reactor engineering, fuel handling, surveillance testing, etc.) will be revised, as appropriate, to either provide adequate caution or initial conditions to assure cognizance of the measuring and testing equipment characteristics used for these.

Table 3(a). Detailed Incidents related to Heavy Load Lifting: Licensee Event Reports and Information Notices (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
18	NRC†	96-26	04/30/96	While lifting a loaded spent fuel storage cask from the spent fuel pool for transfer to the transport bay, the single-failure-proof overhead crane handling system automatically stopped on overload, about 0.13 m [5 in] from the high hook point. Upon investigation, it was determined that the cause was premature actuation of the crane overload-sensing system. The set point of the overload-sensing system was set too low. Upon activation of this system, conventional holding brakes are activated and the load is held in position. The system was bypassed to move the load. Later investigation revealed that the overload-sensing system was inaccurately calibrated during the load cell setting adjustment. This information was sent to all nuclear power plants.	Loaded cask was hanging above the pool until the overload-sensing system could be bypassed and load moved.	Prairie Island Nuclear Generating Plant	The overload-sensing system of the single-failure-proof overhead crane was inaccurately calibrated during load cell setting adjustment.	Information notice to all nuclear power plants on this potential problem.
19	NRC	96-26	04/30/96	Reactor building crane rail failed as indicated by cracking across the top flange. Much of the failure was preexisting because the rails did not have slots for installing bolts and slots were burnt in the field during construction. Flame cutting the slots, without careful preheating and controlled cooling, left residual stresses. This heat-affected zone in high carbon steel was sensitive to hydrogen cracking and subsequent brittle crack propagation. Bending of rail head resulted in misalignment, that in turn caused failure of bearings of bridge truck wheels.	Failure of crane bridge rail and bearings of bridge truck wheels.	Trojan	Inappropriate use of cutting torch to enlarge drilled holes to slots in the web of the rail.	Information notice to all nuclear power plants on this potential problem.

*Note: LER—NRC. “Licensee Event Reports Database.” Washington, DC: NRC. <https://nrcoe.inel.gov/lersearch>
†NRC. “Information Notices.” Washington, DC: NRC. <http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/>

Table 3(b). Detailed Incidents Related to Heavy Load Lifting: DOE Lessons Learned Database								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Issues**	**Plant or Site**	**Root Causes**	**Notes**
1	DOE HSS LL Database*	LL–WSRC–2001–0011	11/1/01	The load block of a 27,216-kg [30-ton] bridge crane was descending to the floor when the operator tried to raise the load after hearing an unusual noise from the crane.	A potential load drop	Savannah River Site 105L	The crane manufacturer did not install a split ring locking washer. A retaining nut on the outboard side of the holding brake drum backed off due to the missing washer, allowing the brake drum to slide off completely off the motor shaft.	This was an isolated incident as a result of improper installation by the manufacturer. The crane was manufactured by Whiting Corporation. The Whiting Service Group installed the brake assembly. Corrective actions: All Savannah River Site overhead and gantry (bridge) cranes shall immediately be evaluated on proper design and installation of holding brake applications.
2	DOE HSS LL Database	2003–SR–WSRC–0012	07/10/03	A 480-V electric bus bar wire 18.3 m [60 ft] long broke off from a Shaw Box 27,216-kg [30 ton] crane; it did not extend far enough to contact the personnel or equipment at floor level. The wire was used to carry power to the trolleys.	A potential shock to people	Savannah River Site 717-F	Metal fatigue. The crane was put in since 1951/1952. The electrical systems were original. No maintenance was performed on them other than a visual look during maintenance.	The mechanical systems of the crane underwent a complete overall and was rebuilt in 2001. Lessons learned†: Electrical wiring on older cranes should be inspected and maintained on a periodic schedule. Actions taken: Ensure electrical wiring on older cranes receive inspections and periodic maintenance.
3	DOE HSS LL Database	L–2001–OR–BJCPORT S-0501	05/16/01	Weight data marked on legacy equipment can be either inaccurate or difficult to interpret. Prudent and well-planned lift evolutions are still vulnerable if based on faulty load estimates.	Overload equipment damage, personnel injury	Not specified in the database; it's a Bechtel-Jacobs report‡	Report did not specify: 1. Marking on the equipment fades as equipment ages. 2. Incorrect marking on the lifting equipment.	Lessons Learned: 1. Independent, 2nd check of lift plans and procedures should include critical review of input parameters, as well as adequacy and currency of known and estimated weights associated with the lift. 2. Dynamometers and other innovative mechanisms may need to be employed for early load verification. Such measurements may be prudent if load weight imprecision is large and/or available over-capacity of lifting equipment is

Table 3(b). Detailed Incidents Related to Heavy Load Lifting: DOE Lessons Learned Database (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Issues**	**Plant or Site**	**Root Causes**	**Notes**
4	DOE HSS LL Database	2001–RL–HNF–0027	08/07/01	Facility management determined the crane capacity had been exceeded when a 2,722- kg [3-ton] crane was used to move a rectangular grout container with debris.	Equipment damage and/or personnel injury.	Hanford	Report did not specify. Fluctuation of dynamometer reading when lifting the load near the top end of rated crane capacity	Lessons Learned: Operating cranes at or near load limits must be done with caution, detailed planning, and close supervision to prevent exceeding the limits.
5	DOE HSS LL Database	2002–KCP–FM&T/KC–0001	03/19/02	A 136,078-kg [150-ton] gantry toppled while being used to lift a 6,350-kg [14,000-lb] milling machine. The operator used a forklift to pull the load while the crane lowered the load so the milling machine would stay upright. In an effort to counteract the movement when the milling machine tilted toward the forklift, the operator increased the tension of the forklift. The wheels of gantry lifted off the rails. The crane toppled and landed on a lathe nearby.	Personnel injury and/or equipment damage	Honeywell Federal Manufacturing & Technologies, Kansas City	The complex and unusual lift (created by the center of gravity on the spindle column of the milling machine - it was off center) caused the application of the wrong rules and techniques for the situation by the construction crew. The crew failed to identify hazards of the job and control the area as specified in the Honeywell Federal Manufacturing & Technologies Safety Handbook.	No personnel injury occurred in this case. The forklift driver and the lathe machinist escaped unharmed. Corrective actions: 1. Manufacturer's lifting recommendations and techniques should be reviewed prior to attempting lifts. 2. Lifting processes should be stopped and re-evaluated whenever the load does not respond in the anticipated manner. 3. Employee empowerment, accountability and the expectation to intervene when unsafe actions or conditions exist must be clearly established, understood, and implemented. 4. Any employee and subcontractor employee is expected to question the adequacy of identified work zones.

Table 3(b). Detailed Incidents Related to Heavy Load Lifting: DOE Lessons Learned Database (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Issues**	**Plant or Site**	**Root Causes**	**Notes**
6	DOE HSS LL Database	2002–RL–HNF–0025	04/30/02	During the removal of a beneficial uses shipping system cask lid, the trolley supporting the chain hoist separated from its I-beam, allowing the 681.8 kg [1,500 lb] beneficial uses shipping system cask lid to fall 0.76 m [2.5 ft] onto a plastic pallet.	Personnel injury and/or equipment damage	Fluor Hanford	The trolley apparently failed because the castle retaining nut that holds the two halves of the device together came loose. No pin or locking device was installed in either castle nut.	No personnel injury or equipment damage occurred in this case. Corrective actions: 1. Facility management should verify that all rigging equipment, including portable gantries and associated trolley assemblies are currently being inspected by qualified personnel according to manufacturer's written instructions for the specific equipment. The inspections should include proper adjustments and a check that all retention devices are in place. 2. Trolley wheels must be matched to the rails on which they ride. Trolleys with tapered wheels should be used only on tapered rails and trolleys with straight wheels should be used only on flat rails.
7	DOE HSS LL Database	2001–RL–HNF–0034	09/04/01	In preparation for performing a critical lift of 3-82-B waste shipping containers, an inspection of the cask lift fixture identified a bent arm on the fixture. An engineering analysis showed that the existing design, even without the bend, did not meet buckling criteria as a load-bearing member.	Load drop, personnel injury and/or equipment damage	Fluor Hanford	In 1996, a lifting fixture was designed and fabricated to handle the cask. One arm of the lifting fixture was bent shortly after it was placed into service. The device was inspected in February 2001. The inspection did not identify the arm as a problem even though the surveillance prescribed looking for such deficiencies. The bent arm was not thought to be a load bearing component so the bend was considered acceptable.	LL: This event was initiated by an observation from a member of staff operations. A questioning attitude and an environment of open communications for feedback of potential safety concerns supported this discovery and led to the actions stated in this LL. Corrective actions: Facility operators should look critically at equipment to ensure deficiencies are not accepted because “they have always been that way.”

*DOE. “DOE Office of Health, Safety, and Security Lessons Learned Database.” Washington, DC: DOE. <https://www.hss.energy.gov/csa/analysis/ll/oellproducts.html>
†Note: LL—Lessons Learned
‡Bechtel-Jacobs report

Table 3(c). Detailed Incidents Related to Heavy Load Lifting: Navy Crane Corner								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
1	Navy Crane Corner*	32nd Edition	12/2001	Two bridge cranes, 36,364 kg [80,000 lbs] and 13,636 kg [30,000 lbs] each were used to lift another bridge crane weighing 42,273 kg [93,000 lbs] when one crane was overloaded. A complex lift plan used on a previous lift by this crane annotated the wrong location of the center of gravity. The error was not corrected prior to the lifting. Due to the limited lifting clearance, only one load-indicating device was used, which was on the lower capacity crane. The load on the lower capacity crane was 4,527.3 kg [9,960 lbs], which meant the load on the higher capacity crane was 37,745.5 kg [83,040 lbs], exceeding its rated capacity.	Crane overload	Unspecified	Unspecified	Complex lifting plans should be reviewed, and all information should be verified for accuracy. When the center of gravity is large and complex shapes must be estimated, there should be a sufficient margin in the lifting cranes to allow for errors in the estimate.
2	Navy Crane Corner	33rd Edition	03/2002	While operators were performing a travel test on a monorail system, the monorail beam buckled and the test load dropped to the floor. The monorail beam had been modified, apparently without an adequate engineering evaluation.	Load drop	Unspecified	Unspecified	Alterations to load-bearing components must be properly engineered and approved by the Navy Crane Center. Test directors must ensure all required tests are performed and that loads are lifted only high enough to perform the required tests and stand clear of test loads to the maximum extent possible.
3	Navy Crane Corner	40th Edition	12/2003	A mechanic was attempting to lift an engine out of a vehicle using a 4,546 kg [10,000 lb] capacity bridge crane and a 1,818 kg [4,000 lb] capacity load leveler (a triangular shape below-the-hook lifting device). The load leveler was attached only to one of two lifting eyes at the rear corners of the engine and a lifting eye in the front of engine. The load leveler was incorrectly adjusted so that the sling attached to the front of the engine was supporting the entire load. During the lift, the engine oil pan became wedged against the frame of the vehicle, preventing the engine from being lifted. The mechanic failed to see the clearance problem and continued hoisting, thereby overloading the load leveler and causing the crane hook swivel to break. Thus the load was dropped.	Load drop	Unspecified		When using below-the-hook lifting devices, it is critical to install and use them in accordance with manufacturer's instructions. In addition, prior to and during all lifts, ensure that adequate clearance is maintained between loads, rigging gear, and any possible obstructions.
4	Navy Crane Corner	42nd Edition	06/2004	An accident occurred on a bridge crane that utilized a radio control system. The radio control's transmitter malfunctioned and caused the crane to move unexpectedly. Upon investigation, it was found that one of the circuit boards was modified without the original equipment manufacturer's knowledge.	Unauthorized repair of crane radio controls			Follow original equipment manufacturer's diagnostics guide in the owner's manual when equipment malfunctions. The original equipment manufacturer is not responsible for repairs performed by the user or an unauthorized repair facility. Radio transmitters or receivers repaired by an unauthorized person or repair facility may result in unintended operation that may cause

Table 3(c). Detailed Incidents Related to Heavy Load Lifting: Navy Crane Corner (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
5	Navy Crane Corner	43rd Edition	09/2004	Out of all Navy shore crane accidents reported in the last 3 fiscal years, 37 percent (193 accidents) occurred without a load on the hook. Almost all the accidents were attributed to human errors. Some of the more common accidents include. 1. Collisions with objects in the crane travel path (58 total), 2. Two-block accidents (29 total), 3. Wire rope damage (25 total) 4. Damage during ODCLs, set up and securing operations (32 total).				Same level of attention would be paid to operating cranes with or without a load when conducting lift operations.
6	Navy Crane Corner	44th Edition	12/2004	A Category 3 bridge crane was two-blocked during its monthly documented preuse check per NAVFAC P-307. The wire rope had been spooling on top of itself and had two-blocked in the previous month. An investigation revealed that the crane was being side-loaded causing misspooling of the wire rope and the hoist block to be out of position of the geared limit switch.	Two-blocking			Loads shall only be lifted vertically. Operators shall not allow side-loads to be applied to the hook. Following an accident or suspected accident, NAVFAC P-301 requires operators to stop operations and notify supervision. Management must ensure all applicable personnel are trained to these requirements.
7	Navy Crane Corner	45th Edition	03/2005	During a scheduled weight test of a Category 3 bridge crane, the weights were dropped when the hook separated from the hoist block. Investigation revealed that after an non-destructive evaluation test of the hook, it was reassembled incorrectly. A thrust ring that fits around the two washer halves used to retain the hook shank in the hoist block was omitted. When the load was lifted, the washer halves spread, pulling the hook through the hoist block, and thereby dropping load to the ground.	Load drop			Management must ensure that maintenance and inspection personnel perform properly and applicable technical manuals are available for the proper disassembly and assembly of components.

Table 3(c). Detailed Incidents Related to Heavy Load Lifting: Navy Crane Corner (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
8	Navy Crane Corner	45th Edition	06/2005	A fire pump rotor was dropped when the wrong lifting fixture was used. After shop personnel realized they had assembled the rotor incorrectly, they decided to remove the motor with its end caps and bearings attached as a unit to save time. Not knowing the weight as a unit, they got a lifting fixture normally used for larger rotor assemblies. The fixture did not properly grip the assembly and as it was lifted, it slipped out of the fixture and dropped onto a pallet.	Load drop			Personnel should not take shortcuts or make guesses as to the appropriate equipment to use. Management must ensure personnel perform "as-trained" to follow proper procedures.
9	Navy Crane Corner	52nd Edition	12/2006	A 18.3 m [60 ft] triple-laced column 25,004 kg [≈ 55,125 lbs] was being prepared for installation. A crane was used to upload the column. When the load was vertical, the main hoist wire rope pulled loose from the terminal end wedge socket connection, thereby dropping the load and hoist block to the ground. Investigation revealed that when the cranes hoist block was reeved, the wire rope was not properly seated in the wedge socket. The wedge socket was damaged from misuse, and the wedge would not fully engage into the socket.	Load drop			Wedge socket end connections are subject to wear/tear and must be inspected for faulty component fit and damage from frequent change outs. NAVFAC P-307 provides special precautions pertaining to the use of wedge socket connections.
10	Navy Crane Corner	52nd Edition	12/2006	A crane hoist was two-blocked causing the wire rope to part and the hoist block to fall. An activity experienced an electrical storm, which caused electrical damage and power loss to a number of buildings and bridge cranes. When the repairs were completed, the correct electric power phasing was not verified. The repair resulted in a reversal of all motor rotation on the bridge cranes. A bridge crane operator realized the crane functions were reversed but continued to operate the crane. The operator raised the hoist block into the limit switch, which did not work due to the phase reversal condition. The hoist two-blocked; the wire rope parted and the hoist block fell to the floor.	Two-blocking			Crane operators are responsible for reporting all adverse or off-normal conditions to supervision. Operation of a malfunctioning crane is an unsafe act that can cause equipment damage and/or personnel injury.

*U.S. Navy. "Navy Crane Corner," 32nd Edition (December 2001) through 53rd Edition (March 2007), <https://portal.navfac.navy.mil/portal/page?pageid=181,3457291,181_3457371:181_3457451&_dad=portal&_schema=PORTAL.

Table 4. Detailed Incidents Related to HVAC*, Ventilation, or Filtration

ID No.	Source	Reference No.	Report or Event Date	Description	Consequences	Plant or Site	Root Causes	Notes
1	LER†	3131999004	E‡: 09/28/99 R§: 10/27/99	On September 28, 1999, the flow rate of the Fuel Handling Area Ventilation System was below the Technical Specifications requirement while irradiated fuel was being moved in the Spent Fuel Pool. Ventilation flow rates were reduced below requirements at approximately 1810, and fuel handling was stopped at 1983, when the condition was discovered. A rope had been installed to keep the damper in the open position during an electrical outage. Installation of the rope was neither called for in the work plan nor was it authorized.	No consequences	Arkansas Nuclear One, Unit #1	Inappropriate work practices and deficient work plan.	The deficient work plan will be corrected, and other work plans for electrical outage will be reviewed prior to their next use.
2	LER	2821997007	E: 04/29/97 R: 06/30/97	On April 29, 1997, at Prairie Island 1 and Prairie Island 2 plants, both trains of spent fuel pool special ventilation were inoperable when operators opened one of the personnel doors to gain entry into the spent fuel pool enclosure. This is a clear violation of the Technical Specification 3.8.D.3, which states, "suspend all fuel handling operations and crane operations with loads over the spent fuel when both trains are out of service." The cause of this event was associated a misinterpretation of the Technical Specification 3.8.D.3. This event has occurred historically throughout the plant operation.	No other equipment faulted during the event and no operator action was taken as personnel were not aware the action constituted a violation	Prairie Island 1 and Prairie Island 2 plants	The root cause for this event was the lack of a procedure that should have been in place that requires all doors to be closed when handling fuel over crane operation with loads over the spent fuel pool.	Although there may be a technical specification (in this case Technical Specification 3.8.D.3), the interpretation associated with this technical specification may not be clear enough for working personnel. Technical Specification Interpretation 3.8-3 has been deleted. A procedure changed that required all doors to be closed when handling fuel over the spent fuel pool. Additional training will take place before the next movement of fuel.
3	LER	3131999001	E: 05/21/99 R: 06/21/99	On May 21, 1999, when a radioactive spent fuel pool purification filter was moved, a radiation field was created at a detector for the Control Room Emergency Ventilation System that was severe enough to trigger the actuation of the ventilation system.	This event did not threaten the safe environment for the control room personnel or the plant itself.	Arkansas Nuclear One Unit 1	The root cause for this event is attributed to the sensitive nature of the design of the ventilation system, which results in it being susceptible to spurious actuations.	Consideration was given to installing shielding around the detectors or increasing the actuation set point. It was determined that neither option was desirable. Permanent shielding would reduce detector sensitivity. Additional training has been provided.

Table 4. Detailed Incidents Related to HVAC*, Ventilation, or Filtration (Continued)								
ID No.	Source	Reference No.	Report or Event Date	Description	Consequences	Plant or Site	Root Causes	Notes
4	LER	3151998029	E: 04/22/98 R: 08/04/99	On April 22, 1998, at the Cook Nuclear Plant Unit #1, it was discovered that the response time of the fuel handling area ventilation system for transition from the normal to the emergency filtration mode may not be adequate to prevent an unfiltered release from a refueling accident. As a result, the fuel handling ventilation system was declared inoperable.	Because the ventilation system was taken out of operation before any incident, this event had minimal impact on health and safety of the public.	Cook Nuclear Plant Unit #1	Deficiency in the original design. The system was not designed for the proper response time.	This event stresses the importance of periodically testing the response time of key ventilation systems during the lifetime of the ventilation system. Calculations indicated that in the event of a failed response, the dose to workers would be minimal. Therefore, it was decided that all subsequent operations involving fuel movement within the storage pool would be conducted in the emergency filtration mode.
5	LER	3162000011	08/21/20	On July 20, 2000, at the Cook Nuclear Plant Unit #2, the spent fuel pool exhaust ventilation system was inoperable with fuel inspections in progress. Auxiliary building crane inspections were also in progress, which require the ventilation system to be operable.	Because there was no contamination of the affected areas, the safety significance of this event is minimal.	Cook Nuclear Plant Unit #2	The cause for this event is inadequate communication between two work groups involved in the Auxiliary building crane inspections and control room personnel. Upon discovery of the malfunctioning ventilation system, control room personnel suspended movement of fuel within the spent fuel pool area.	Ventilation systems are not capable of responding to a fuel accident quickly enough to prevent an unfiltered release to the atmosphere. Therefore, a compensatory action was put in place to ensure that the spent fuel pool ventilation system is in the charcoal filter mode of operation during fuel handling operations. This event is a violation of Technical Specification 3.9.12. The lesson learned from this event is the need for a procedure to provide clear and concise instructions to ensure that all required systems and equipment are operable when fuel is moved inside the spent fuel pool area or moved over the spent fuel pool. A review of existing fuel handling procedures indicated that there is adequate instruction regarding the requirement to operate the spent fuel pool exhaust ventilation system in the charcoal filter mode when fuel is being moved. The new operations procedure will be enhanced. Therefore, no preventive actions were deemed necessary.

Table 4. Detailed Incidents Related to HVAC*, Ventilation, or Filtration (Continued)								
ID No.	**Source**	**Reference No.**	**Report or Event Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
6	LER	3952000009	R: 10/16/00 R: 11/15/00	The alarm failed to inform operators that the negative pressure of the spent fuel pool area had fallen below the recommended value.	At the time of discovery, no heavy loads were being transported above the spent fuel pool; therefore, the safety impacts of this event are minimal.	Virgil C. Summer Nuclear Station	The cause of the alarm failure is unknown, but it is suspected that the alarm function drifted outside the acceptable range.	The corrective action taken was to add the verification of the spent fuel pool area differential pressure to the logging that is performed every 12 hours. The requirement to record the building pressure differential will be for all modes.
7	LER	2201998012	E: 05/21/98 R: 06/22/98	On May 21, 1998, at Nine Mile Point Unit #1, it was discovered that the fire dampers would fail closed as a result of loss of offsite power. These dampers are required to be open during fuel transfer to provide a source of filtered air.		Nine Mile Point Unit #1	The root cause for this event is inadequate evaluation of the dampers.	Corrective actions included modifications to the fire dampers to reopen following power restoration from a loss of offsite power.
8	LER	2371998012	E: 08/20/98 R: 09/18/98	The unit supervisor reviewed technical specifications to permit movement of fuel in the spent fuel pool area with the ventilation not operational. Permission to move the fuel was given, but the unit supervisor failed to recognize that moving fuel placed the reactor in two technical specification modes: Mode 1 and Mode 2. The next day, this oversight on the part of the supervisor was discovered and all activity was suspended.	Detailed calculations and analysis of the potential risks concluded that this event had a minimal impact. Had the load bundle dropped into the spent fuel pool, there could have been a release of radioactive krypton.	Dresden Nuclear Power Station Unit #2	The causes of this event were 1. inadequate work planning/implementation process; 2. knowledge deficiency of technical specification content; 3. decline in operator performance specific to management and recognition.	Operations management has developed and is currently implementing a Departmental Improvement Initiative to address global performance weaknesses.
9	LER	4232006001	E: 03/1/06 R: 10/04/06	On March 1, 2006, at the Millstone Power Station Unit #3, with the plant operating in Mode 1 at 100 percent power, both trains of the Control Room Emergency Ventilation System were made unavailable as a result of the valve air actuator being removed from the air inlet isolation valve.	No safety related concerns identified.	Millstone Power Station Unit #3	This event was caused by a failure to recognize and correct an operating practice associated with an allowed mode of operation (isolated filtered recirculation) after the valve actuator was removed from Unit 3 as per Technical Specification 3.7.7.	Address the correct “filtration pressurization” in Technical Specification 3.7.7—Revise technical specification regarding the different modes of operation of the ventilation system. Provide additional training for Unit 3 licensed operators.
10	LER	3542000009	E: 05/25/00 R: 06/23/00	On May 25, 2000, at the Hope Creek Plant, the four running filtration, recirculation, and ventilation system tripped on low flow as a result of inadvertent closure of a manual damper located in their common supply duct.	This event took place during surveillance testing, and there was no impact on health and public safety.	Hope Creek Plant	This event was caused by inattention to detail during the installation of a locking device of the manual damper. Contributing to this event was inadequate procedural guidance regarding damper locking device installation. Upon discovery, the damper was opened and the ventilation was restored to full operability.	To prevent reoccurrence, the ventilation system operation will be revised to require independent verification of the ventilation/damper system to operate the spent fuel pool exhaust ventilation system in the charcoal filter mode when fuel is being moved. The new operations procedure will be enhanced. Therefore, no preventive actions were deemed necessary.

Table 4. Detailed Incidents Related to HVAC*, Ventilation, or Filtration (Continued)								
ID No.	Source	Reference No.	Report or Event Date	Description	Consequences	Plant or Site	Root Causes	Notes
11	LER	3352001001	E: 01/30/01 R: 08/13/02	On January 30, 2001, at the St. Lucie Units 1 & 2, it was discovered that inadequate procedural guidance for operation of the control room ventilation system during the emergency recirculation mode could have led to inadequate control room pressurization.	This discovery was made after the plant was in Mode 1 operating at 100 percent. However, the dose was a fraction of the 10 CFR Part 100 limits as a result of this event. Therefore, this event did not have any adverse impact on the health and safety of the public.	St. Lucie Units 1 & 2,	This event was caused by procedural inadequacies that could have allowed operation of the control room ventilation system without proper alignment of outside air makeup to the control room envelope. The operators were not clearly directed to open the outside air intake valves to establish makeup air to the control room. Improper operation of the control room ventilation system can potentially cause operator doses to exceed harmless levels.	Procedural changes have been made to correct this condition: control room differential pressure indicators were calibrated; the pressure probes were relocated to provide more stable readings.
12	LER	2862005001	E: 01/26/05 R: 03/24/05	On January 26. 2005, at Indian Point 3, with the reactor at 100 percent power, it was discovered that the damper of the control room was operating with linkage in the reverse position, which rendered it inoperable.	There was no significant health effect to the public because the system maintained functional capability.	Indian Point 3	The apparent cause of this incident was incomplete work instructions in that no detail was provided on how to connect the linkage during installation. Contributing to this was the failure of a postwork test to ensure that the damper was working correctly.	Corrective action was taken to repair the damper and more explicit repair/installation procedures were to be issued.
13	Hanford Database	2002–RL–HNF–0035	E: 03/26/02 R: 06/26/02	On March 26, 2002, workers smelled a gasoline odor within the laboratory.	Three laboratory workers experienced nausea and headache symptoms. One worker was transported to the hospital for further evaluation.	222-S Laboratory	An investigation concluded that the likely cause of the odor was a gas-operated generator whose exhaust was placed too close to the ventilation intake. This human error was a result of the lack of procedures for vehicle traffic or maintenance near ventilation intake areas.	Explicit instructions to turn off the engine of idling vehicles will be inserted in the new work/training procedures.
14	Hanford Database	2002–RL–HNF–0001	E: 09/05/01 R: 01/09/02	On September 5, 2001, personnel were testing smoke detectors in the ventilation system air handling units (supply fans). They bypassed the input shutdown devices for the supply fans and introduced simulated smoke into the ventilation duct work. The simulated smoke tripped the output shutdown devices, causing dampers associated with the air handling units to close and the fans to shutdown on low flow. A diesel-powered exhaust fan started to maintain negative pressure for containment in the facility.	None reported	222-S Laboratory	This event was caused by human error because the test procedure in use did not specify bypassing the output device nor did it provide sufficient detail for accessing the appropriate software menu for selecting the output devices. This was the first annual test of the smoke detectors with the shutdown devices in service.	To rectify this problem, facilities should validate ventilation test procedures after modifying ventilation control systems and advise testing agencies of any necessary changes to their work control testing procedures. Further, ventilation system-cognizant engineers should closely monitor testing on their systems to ensure safe and proper testing. This is especially important when testing is taking place for the first time after a system modification.

Table 4. Detailed Incidents Related to HVAC*, Ventilation, or Filtration (Continued)								
ID No.	Source	Reference No.	Report or Event Date	Description	Consequences	Plant or Site	Root Causes	Notes
15	Hanford Database	2005–RL–HNF–0033	E: Not specified R: 09/29/05	During ventilation system maintenance, it was found that the exhaust damper does not always function properly. The damper was found to stick in the closed position and sometimes in the 3/4 open position. (Hanford Lessons Learned Database 2005–RL–HNF–0033)	None reported	Building 2736 ZB	The importance of this event is that the Safety Basis did not consider the consequence of the failed damper. The safety basis relies on the damper being open in the event of an air monitor alarm. Therefore, an unanalyzed condition exists if the damper does not open properly during a spread of contamination. The Safety Basis clearly describes the modes of ventilation system operation, but damper failure was not recognized. The Safety Basis assumed that the damper would fail in the open position.	An Engineering Document Change was initiated to correct this deficiency in the Safety Basis. In addition, the damper was modified so that it is held open by a damper actuator.
16	DOE Office of Health, Safety and Security Lessons Learned Database	2001–RPP–HNF–IB–01–05	E: 12/00 R: 03/22/01	Exhausters were operating above their rated flows in violation of ANSI/ASME N509 Section 4.3. A HEPA-filtered portable exhauster was found to be operating at twice the rated flow of the HEPA filter on the unit. The exhauster was rated at 1,000 cfm, but came equipped with a 500-cfm HEPA filter.	None	CH2M Hill Hanford Group, Inc.	The root cause for this event was a misinterpretation of the ANSI/ASME N509 requirements by the manufacturer.	Similar exhausters were checked for flow and filter ratings. A purchase specification was written for the procurement of a restrictor plate to ensure that the air flow cannot exceed the filter's rated capacity.

*Note: HVAC—Heating, Ventilation and Air Conditioning
†Note: LER—NRC. "Licensee Event Reports Database." Washington, DC: NRC. <https://nrcoe.inel.gov/lersearch>
‡Note: E—Event Date
§Note: R—Report Date
‖Hanford Database

Table 5. Incidents Related to Electrical Power Systems								
ID No.	Source	Reference No.	Report or Event Date	Description	Consequences	Plant or Site	Root Causes	Notes
1	NRC IR*	05000285/2005010	E†: 07/21/04 R‡: 04/15/05	One of two redundant emergency diesel generators that are relied upon to power safety buses during loss of power conditions was failed and inoperable for 29 days. The unavailability of the backup power system was unknown to plant operating staff during that time.	The backup power system failed due to a blown fuse after completion of a routine surveillance test. The failure was indicated by a number of off-normal readings and inoperability of related indicating systems. The unavailability of the backup power source was not recognized or reported till the same indications were observed during the next scheduled surveillance test. The failure degraded the plant's ability to control operations and prevent postulated core damage due to hazards that could have occurred with greater than acceptable probability during that period.	Fort Calhoun Station	Numerous personnel failed to "assign significance" to the abnormal readings and operations of the generator while it was running but after the test sequence was completed. The abnormal readings were noted, but no corrective actions were pursued.	The backup power source failed due to a fuse that blew when the connecting breaker was opened after completion of the monthly surveillance test. The generator passed the test but provided significant indications of failure immediately after the test, before the unit was powered down. No worker or public health or safety issues or radiological releases resulted from this event.
2	LER§	2202002003	E: 12/02/02 R: 01/31/03	An uninterruptible power bus was lost, deenergizing one of two redundant safety buses. This event occurred while the plant was operating near 100 percent power and while an emergency diesel generator (connected to supply the second of the two redundant safety buses) was out of service for maintenance.	All emergency and backup power equipment for at least one of two redundant safety-related power trains is required to be operational while the plant is operating at power levels. The loss of power caused by failure of the uninterruptible power system during this interval degraded the operations and safety of the plant and caused abnormal operations and a shutdown.	Nine Mile Point	The root cause of the deenergized condition of the power bus was that a single point of failure condition was present in the design of the system. The root cause of the failure of the uninterruptible power system was degraded operation of a power supply component with the system. The degraded power supply condition was due to excessive component age and a lack of preventative maintenance.	One of the two safety buses was temporarily connected to a nonrelated safety bus to restore operation, and a shutdown was initiated. Several similar components in the subject system were replaced, and a design modification was developed to correct the single-point-failure deficiency. Procedures were modified and implemented to improve preventative maintenance practices. No worker or public health or safety issues or radiological releases resulted from this event.

Table 5. Incidents Related to Electrical Power Systems (Continued)								
ID No.	**Source**	**Reference No.**	**Report or Event Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
3	LER	3252005004	E: 05/12/05 R: 07/11/05	Electrical power was lost to a 4 kV emergency bus that was energized by normal offsite power. This event occurred while the plant was operating near full power and while an emergency diesel generator was out of service for planned maintenance.	All emergency and backup power equipment for at least one of two redundant safety-related power trains is required to be operational while the plant is operating at power levels. The loss of power to the 4 kV emergency bus, without backup power available "could have prevented the fulfillment of the safety function systems that are needed to mitigate the consequences of an accident" in either of the reactors at the plant. The loss of power event propagated through a number of systems, creating a complex chain of events. Losses of operability of safety-related systems caused automatic reactor containment and isolation actions to be initiated. The control room air conditioning and control room emergency ventilation systems for both reactors also failed.	Brunswick Steam Electric Plant	The root cause of the failure of normal offsite power to energize the emergency bus was attributed to a design fault in the emergency diesel generator control logic. The design allowed reduced reliability in providing offsite power to the emergency buses in the event of a particular alignment. The failure of the control room air conditioning and ventilation systems was attributed to a broken wire termination lug. This fault may have existed prior to the event or occurred during the unusual stresses associated with the event. Corrective actions included initiation of design changes to correct the noted design deficiencies. Immediate reactions included special maintenance, procedures, and training to prevent a reoccurrence while design modifications were in process. Noted wiring problems were corrected using heavier wire termination products.	The loss of power event affected a number of operational safety systems. A reactor coolant system leakage detection instrumentation function became inoperable, initiating primary containment isolation actions. The plant began shutdown operations immediately and completed these operations safely. No worker or public health or safety issues or radiological releases resulted from this event.

Table 5. Incidents Related to Electrical Power Systems (Continued)								
ID No.	Source	Reference No.	Report or Event Date	Description	Consequences	Plant or Site	Root Causes	Notes
4	LER	2502006005	E: 03/06/06 R: 07/28/06	Maintenance operations were being conducted on electrical power system components. The affected reactor was shut down during this time, and other reactors at the plant were operating at near full power. Near the completion and return-to-service of the maintenance activity, the component being maintained did not operate correctly, causing a loss of normal power to a safety-related 4 kV bus. Upon loss of normal power to the 4 kV bus, the emergency diesel generator automatically started and loaded after expected startup and loading delays. The generator began to operate erratically, however, as operations were restored, and the generator required manual supervision and adjustment to keep it online for approximately 8 hours until normal offsite power was restored to the 4 kV bus.	Required grounding provisions for the emergency diesel generators were not provided during (much earlier) previous maintenance operations. This condition affected both emergency generators in the system.	Turkey Point	The cause of this event was determined to be the use of incorrect plant procedures for grounding of the generator startup transformer during an unrelated earlier maintenance activity.	Subsequent to this event, a design modification was implemented that eliminated the need to provide the special grounding provisions. Procedural improvements were implemented to help ensure the use of correct procedures in maintenance activities. No worker or public health or safety issues or radiological releases resulted from this event.
5	LER	2772003004	E: 09/15/03 R: 11/07/03	A brief interruption of power to the reactor safety systems for two reactors occurred as a lightning storm affected a commercial power provider substation 56.3 km [35 mi] away from the plant. This event deenergized safety-related buses and initiated automatic reactor shutdowns. All emergency diesel generators started and loaded after expected startup delays. After approximately an hour of operation using power supplied by the emergency generators, one of the generators spontaneously disconnected due to an indication of low jacket coolant pressure within the generator.	The loss of an emergency generator during an interruption of normal power and resulting reactor shutdown operations can leave a plant potentially unable to fulfill all required safety and mitigation functions in the event of further degradations or accidents. A discretionary “Unusual Event” was declared, and the Emergency Operations Facility and Technical Support Center were staffed to deal with the event.	Peach Bottom	The cause of the offsite power interruption was poor operation of protective relaying at the commercial power provider substation during a lightning event. This condition was judged to be due to lack of preventative maintenance by the commercial power provider. The cause of the emergency generator failure was attributed to leaking copper gaskets in the cylinder liner seals within in the generator engine.	The failures at both the commercial provider substation and within the plant emergency generator have resulted in improvements in preventative maintenance procedures. The cause of the generator engine failure is suspected to be related to manufacturing or preventative maintenance problems, or both, and a formal root cause investigation was initiated. No worker or public health or safety issues or radiological releases resulted from this event.

Table 5. Incidents Related to Electrical Power Systems (Continued)								
ID No.	Source	Reference No.	Report or Event Date	Description	Consequences	Plant or Site	Root Causes	Notes
6	LER	5302006003	E: 04/02/06 R: 05/26/06	During Integrated Safeguard Testing operations, a loss of electrical power to a safety-related 4 kV bus occurred. The plant was in cold shutdown for refueling. Normal offsite power to the safety bus was manually removed during the testing and the emergency diesel generator started in "Emergency Mode." During surveillance testing for the backup generator, the generator failed and the 4 kV safety-related bus was deenergized. The alternate (redundant) emergency generator automatically started and reenergized the bus after expected startup delays.	Operations staff entered "Abnormal Operating Procedures for a Degraded Electrical System." The loss of the 4 kV bus resulted in inoperability of the related (to the affected bus) Control Room Essential Filtration System and Control Room Emergency Air Temperature Control System.	Palo Verde	During testing, temporary jumpers were incorrectly installed in the control system to simulate emergency conditions. The jumpers were inadvertently installed on the wrong control relay and not as required by test procedures.	The plant entered abnormal operations for a degraded electrical system for a short period due to potential degraded safety capabilities during the event. The event resulted in renewed emphasis on peer checking and formal prejob briefing for personnel performing test procedures. No worker or public health or safety issues or radiological releases resulted from this event.

*NRC IR—
†Note: E—Event Date
‡Note: R—Report Date
§LER—NRC. "Licensee Event Reports Database." Washington, DC: NRC. <https://nrcoe.inel.gov/lersearch>

Table 6. Detailed Incidents Related to Instrumentation and Controls								
ID No.	**Source**	**Reference No.**	**Report or Event Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
1	DOE*	2004-03	E†: 07/21/03 06/03 08/98 R‡: 02/09/04	Valve Lineup Errors Cause Near Misses—These three events illustrate multiple human errors that occurred, which resulted in equipment damage and personnel error. Instrumentation and control equipment was inoperable, and this contributed to the event in July 2003.	In one case, a pump casing failed catastrophically resulting in damage to a control room window and injury to an operator.	Oak Ridge Y-12 Site	1. Pump discharge pressure gauge was inoperable. 2. Status board for the equipment was ignored because it was known to be unreliable. 3. Planned maintenance was not completed in a timely manner and was inadequate. 4. Operating procedures did not clearly delineate normal pressure and temperature ranges or were cancelled because they were inaccurate, and then were not replaced.	"These occurrences illustrate the importance of attentiveness to the task at hand, even if the task is routine and has been performed many times before."
2	DOE	2004-12	E: 05/05/04 R: 06/14/04	Unauthorized Safety System Modifications—An individual made an unauthorized modification to interlock switches on a safety system, which were designed to protect personnel from exposure to ionizing radiation. The modification was made so that the equipment could be used to conduct tests.	The interlock switches were part of the lid closure system of a test vessel. If the lid of the test vessel was unlatched during a test, then personnel in the immediate area could have been at risk. Several other detectors and interlocks were also available to shut down the test.	Thomas Jefferson National Accelerator Facility	The unauthorized modifications were made to the interlock switches several months earlier.	Such " . . . events underscore the importance of controlling changes to safety-related structures, systems, and components (SSCs). Unauthorized changes to SSCs can degrade systems relied upon for safety and can create hazards for workers and the public."
3	DOE	2005-05	E: 04/21/04 R: 03/07/05	Defeating Safety Interlocks can be Hazardous—"Defeating the safety interlocks to enter the irradiator had been a common practice at this facility for years." In this case, when the safety interlocks were defeated, two workers entered the irradiator and received doses of 4.4 and 2.8 rem in a matter of seconds. Prior to this, "the operator and alternate radiation safety officer assumed that the control panel indicating the still-exposed source rack was spurious."	"The irradiator is a storage pool that contains two source racks operating simultaneously, each containing approximately 2 million curies of cobalt-60." In this case, one wall of concrete separated the workers from direct exposure to the exposed rack. "In a matter of seconds, two workers that had entered the irradiator received 4.4 and 2.8 rem."	NRC-Licensed Irradiator Facility that Sterilizes Medical Supplies	1. "Defeating safety interlocks and entering the irradiation room had become common practice at this facility." 2. "The licensee did not investigate recurring switch problems to determine their underlying causes or perform preventive maintenance."	"Under no circumstances is it acceptable to defeat or work around interlocks."

Table 6. Detailed Incidents Related to Instrumentation and Controls (Continued)								
ID No.	**Source**	**Reference No.**	**Report or Event Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
4	DOE	2002-01	E: 11/15/01 R: Not Identified	Researcher Receives X-Ray Exposure—A researcher did not notice the "shutter open" indicator on the x-ray machine as the view of the indicator was partially obscured. Instead the researcher relied upon the console indicator, which was readily observable but was actuated by the console switch, not the shutter mechanism itself.	" . . . a researcher received an estimated dose of 12 millirem to the eyes as a result of accidental exposure to x-rays from an x-ray machine."	Oak Ridge National Laboratory Central Facility	"This occurrence highlights the deficiency associated with shutter construction and interlock design philosophies of some x-ray machines."	The Oak Ridge National Laboratory review board investigating the occurrence concluded that "continued reliance on a large number of users with varying backgrounds to notice shutter position indicators would, because of the normal variability of human performance, allow periodic recurrence of this type of event if reliance on a single automatic shutter to control x-ray emissions is an allowed configuration." Some corrective actions included "...adding a set of enclosure door switches to interrupt x-ray generator power and adding a more visible set of shutter-position indicator lamps."
5	DOE HSS LL Database§	2006–SR–WRC–0052	E: 11/2004 R: 12/18/05	Inoperable Fail-Safe Feature on the Safety Significant 9.1E High Temperature/Steam Flow Interlock—A site programmable alarm module was replaced with another module considered to be a like-for-like replacement. This like-for-like determination was incorrect because the new module would not fail-safe when the module sensed an open in the input sensor signal or loss of power.	A steam isolation valve would not have closed, because a safety-significant interlock would not have failed safe on loss of input sensor signal.	Washington Savannah River Company	The like-for-like determination was in error. All relay contacts on the available module would not open (fail-safe) upon loss of an input sensor signal or loss of power to the module.	"A lesson to be learned is that when a section or stated requirement in the facility safety basis is encountered which is not clear or not fully understood, the Technical Support Engineers must be contacted for an official clarification rather than making a judgment call based on past practices and approved functional test procedures."

Table 6. Detailed Incidents Related to Instrumentation and Controls (Continued)								
ID No.	**Source**	**Reference No.**	**Report or Event Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
6	LER	0292001001	E: 06/26/01 R: 08/22/01	Spent Fuel Pit Area Radiation Monitor Alarm Setpoint Set Above Limit Allowed by Technical Specification 3.3—During a Nuclear Safety (Quality Assurance) Audit, the spent fuel pit area radiation monitor alarm setpoints were found set above the allowed limit. The alarm setpoints are required to be set less than 5 mR/hr [5×10^{-5} Gy/hr] or two times the background radiation level, whichever is greater, while moving irradiated fuel, control rods, or sources. The background radiation level was 2mR/hr [2×10^{-5} Gy/hr] while the alarm setpoints were at 7 mR/hr [7×10^{-5} Gy/hr].	"The spent fuel pit area radiation monitor is an instrument required by Technical Specification 3.3 to ensure early detection of inadvertent criticality during fuel handling activities." Although the alarm setpoints were out of tolerance, the area radiation monitor was operable and the time interval to reach the alarm condition compared to the time interval to reach the required setpoint was negligible. Therefore, "an increase in radiation levels would have been detected early enough to preclude any additional impact to the workers or public safety."	Yankee Nuclear Power Station	1. Failure to incorporate guidance from Technical Basis Document 99-75, 'Basis for the Spent Fuel Pool Area Radiation Monitor Background Determination' into plant procedures." 2. "Lack of detail and poor design in procedure OP-4816, 'Functional Test and Alarm Setting of the Area Radiation Monitoring System.' " 3. "Failure to train on Technical Basis Document 99-75, 'Basis for the Spent Fuel Pool Area Radiation Monitor Background Determination.' "	Although no fuel handling evolutions were in progress at the time of discovery of this issue, " . . . fuel handling evolutions in August of 1999 and April/May of 2000 were conducted under similar circumstances where the [spent fuel pit] SFP [area radiation monitor] ARM alarm setpoints were improperly set."
7	IN‖	2007-15	E: 08/19/06 R: 04/17/07	Nonsafety-related controllers that were on an ethernet network became unresponsive due to excessive integrated computer system network traffic. Both safety-related and nonsafety-related equipment may be on the plant network. Therefore, it is important to protect devices on the plant network to ensure safe operation. In this case, two variable frequency drive pump controllers became unresponsive. Operators manually scrammed the unit following loss of recirculation flow.	The plant was placed in a potentially unstable high-power, low-flow condition.	Browns Ferry, Unit 3	"The licensee determined that the root cause of the event was the malfunction of the [variable frequency drive] VFD controller because of excessive traffic on the plant [integrated computer system] ICS network."	Corrective actions included 1. "Developing a network firewall device that limits the connections and traffic to any potentially susceptible devices . . . ." 2. Installing network firewall devices for the controllers.

Table 6. Detailed Incidents Related to Instrumentation and Controls (Continued)								
ID No.	Source	Reference No.	Report or Event Date	Description	Consequences	Plant or Site	Root Causes	Notes
8	Navy Crane Corner¶	Equipment Deficiency Memorandum - 074	E: None given R: 03/05	Electrical Control Failures—A hoist contractor failed to de-energize. "The operator was pressing the up button and when the button was released, the hoist block continued to rise."	Uncontrolled upward movement of hoist block.	None given	"A contactor was undersized for the hoist motor, which caused the contacts to weld closed. Maintenance personnel installed the incorrect contactor."	" . . . personnel should verify the replacement parts are the same size and rating as the existing equipment or as specified in engineering drawings."

*DOE. "Weekly Operating Experience Summaries." Washington, DC: DOE. <http://www.eh.doe.gov/ll/occurrences.html>
†Note: E—Event Date
‡Note: R—Report Date
§DOE. "DOE Office of Health, Safety, and Security Lessons Learned Database." Washington, DC: DOE. <https://www.hss.energy.gov/csa/analysis/ll/oellproducts.html>
‖NRC. "Information Notices." Washington, DC: NRC. <http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/>
¶U.S. Navy. "Navy Crane Corner," 32nd Edition (December 2001) through 53rd Edition (March 2007), <https://portal.navfac.navy.mil/portal/page?_pageid=181,3457291,181_3457371: 181_3457451&_dad=portal&_schema=PORTAL>, U.S. Navy.

Predecisional Draft - Not for Public Findings Related to XXXXXX

Table 7. Detailed Findings Related to Fires and Explosions

ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
1	NRC IN*	IN 2007-17	05/03/07	On August 18, 2006, a fire began during a welding evolution for a plant modification to install ventilation duct through a concrete wall (a 3-hour fire barrier). The wall separates a shop area from the safety-related west cable vault. After an opening was made in the wall, workers inserted metal sleeve through the opening and installed a steel plate that covered the end of the box on the west cable vault side. An annulus existed because the concrete wall opening was larger than the metal sleeve box. A worker stuffed combustible material (i.e., cotton rags) into the annulus adjacent to the west cable vault steel plate and sealed it with duct tape to limit dust and air flow into the cable vault from the shop. Plastic sheeting material used to catch the debris during the boring was also left in place. Fire started when angle clips were being welded on a ventilation sleeve box from the shop side of the fire barrier. The heat transfer through the metal sleeve box ignited the duct tape and rags. The plastic sheeting was ignited by the burning rags. Hot burning plastic fell onto the conduit-protected cables. Smoke from the burning plastics set off a nearby smoke detector. The fire was put out manually after 6 minutes.	No actual consequence specified; potential consequences could include equipment damage and/or personnel injury.	Beaver Valley Unit 1	The licensees root cause analysis determined. 1. Inadequate level of the detail and implementation of the specific fire prevention administrative control and compensation measures. 2. Contrary to National Fire Protection Association 51 B Standards for Fire Prevention During Welding, Cutting and Other Hot Work, combustible material and nonfire-retarding plastic sheeting were within 35 ft of hot work. 3. No combustible material should be used in the annulus in a fire-rated barrier. 4. There was inadequate coordination of compensation measures (fire watch patrols) for breaching fire barriers. 5. Lack of sensitivity to hot work affected risk-significant areas. 6. Plastic sheeting material did not meet National Fire Protection Association 701 Flame Retardant	According to the Information Notice, the Beaver Valley fire resulted from the improper types of material being utilized in the improper places and an ignition source being directly applied. The fire prevention, design modification review, procedural training, and the compensation measures all failed to prevent the fire.

Table 7. Detailed Incidents Related to Fires and Explosions (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
2	NRC IN	IN 2007-17	05/03/07	On August 15, 2006, combustible roofing material on the E-3 emergency diesel generator building caught on fire near the diesel exhaust pipe penetration (through roof) area. The fire lasted 35 minutes before it was put out by the fire brigade. Prior to the fire, the emergency diesel generator had been running for 21 hours as a part of the 24-hour endurance surveillance test. During the extended emergency diesel generator run, the steel penetration sleeve heated to the point that caused the adjacent roofing material to ignite. To prevent excessive heat buildup of the steel penetration sleeve, the design drawing calls for a 1 ½ in air gap below the rain hood, which allows the heated air in the penetration to escape. However, when the emergency diesel generator roof was replaced in 1997/1998, the roofing material and flashing were installed, leaving only a ½ in or less air gap below the rain hood. Furthermore, some of the original nonfire-rated (combustible) buildup roofing or new vapor membrane repair material (combustible) were installed incorrectly and remained, abutting the steel penetration sleeve. The exhaust stack operating temperature is approximately 482 ° (900 °F). The asphalt roofing paper burns at approximately 204 °C (400 °F).	No actual consequence specified; potential consequences could include equipment damage and/or personnel injury.	Peach Bottom	Licensee's root cause analysis determined that 1. There was no oversight and no hold point for a site inspection prior to closing the penetration area and no final inspection upon completion. 2. There was no verification of design requirements (i.e., the air gap). 3. The lack of controls over the roofing contractor allowed combustible materials to come into contact with a surface that normally exceeds the materials ignition temperature.	According to the Information Notice, the Peach Bottom fire resulted from construction/repair work that was not installed properly around an ignition source. The installation was not inspected by the plant staff prior to closing the penetration around the exhaust stack and/or after the work was completed to verify the quality of work. At Peach Bottom, the events involved a lack of basic oversight of design details that should be inherently inspected in the field to ensure quality work products regardless of the area of the plant and perceived risk significance.
3	NRC IN	IN 2002-27	09/20/02	On February 3, 2001, a 4.16 kV breaker faulted and initiated a fire. It took the firefighters 3 hours to extinguish the deep-seated fire. The firefighters had to use water to put out the fire after unsuccessful attempts to extinguish the fire with dry chemical.	Loss of power to Unit 3 nonsafety-related system, a reactor trip, a turbine/generator trip, and an auto start of both Unit 3 emergency diesel generator.	San Onofre Nuclear Generating Station Unit 3	No root cause was given in the Information Notice.	Equipment rated at 4.16 kV or higher is vulnerable to particularly energetic faults. The heat release rate from electrical cabinet fires may be >> (by a factor of 1,000) than assumed in nuclear power plant fire hazard analyses, which typically only include cable insulation as combustibles.
4	NRC IN	IN 2002-27	09/20/02	On August 3, 2001, a fire occurred in the 12-4 cubicle along the left side of the breaker. Fire brigade used water to put out the fire after unsuccessful attempts to extinguish the fire with CO_2 and Halon.	Two fire brigade members were treated for heat exhaustion. The breaker compartment was heavily oxidized. Holes were burnt through the cubicle on either side of the breaker.	Prairie Island Unit 1	Poor electrical connection between the breaker 12-4 C-phase primary disconnect assembly and the 1MY bus stab.	1. Maintenance practice could have contributed to the failure of the primary disconnect assembly by creating poor connection. 2. Use of a small quantity of water was effective in putting out energized electrical equipment fires.

Table 7. Detailed Incidents Related to Fires and Explosions (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
5	NRC IN	IN 2002-27	09/20/02	A 20-A male connector was improperly modified to be connected to a 15-A plug. The cord actually carried a current of 17.39 A. On December 19, 2001, the underrated cord overheated and ignited the plastic and a rubber air hose nearby. The fire generated heavy smoke, which activated a deluge sprinkler system in a different fire area, spraying water on safety-related motor control centers.	In this case, the water on the safety-related motor control centers caused actuation of the 480V bus ground alarms in the main control room. However, if the moisture could have potentially affected (e.g., shorted out) other safety-related equipment, leading to more serious consequences.	Ft. Calhoun	Licensee failed to follow procedural requirements for a temporary modifications to modify the plug. The use of an improperly modified plug led to the cord being underrated, thereby causing the fire due to overheating of the extension cord.	1. Procedural requirements should be following prior to engaging a temperature modification. 2. GDC 3—Requires Structures, systems, and components important to safety shall be designed and located to minimize the probability and effect of fires and explosion, consistent with other safety requirements.
6	NRC	RES Report	02/2002	On February 8, 2001, a fault started in the safety-related 4.16kV switchgear supply circuit breaker. It caused explosions, arcing, smoke, and ionized gases, which propagated to adjacent safety-related 4.16 kV switchgear and damaged 6 switchgear compartments.	The damage resulted in complete loss of the faulted safety bus and its emergency diesel generator and LOOP to the undamaged safety bus due to faulting of its offsite electrical feeder circuit. An independent failure of the redundant emergency diesel generator resulted in loss of all alternating current power.	Maanshan Unit 1, Taiwan	TaiPower recently concluded that ferromagnetic resonance was the cause of the event. It also concluded that there was insufficient electrical separation between safety bus A and B and subsequently added an interlock between bus A and B. TaiPower also added emergency startup of the emergency diesel generators to its regular test schedule.	1. Energetic electrical faults can result in explosions, arcing, fire, ionized gases, smoke, spurious actuation of circuit breakers, other circuit failures, collateral damage to adjacent equipment, and latent equipment failures independent of fires. 2. RG 1.189 (4.1.3.6) “Electrical cabinets (e.g., 4.16kV to 13.8 kV switchgear) present an ignition source for fires and a potential explosive electrical fault that can result in damage not only to the cabinet or origin but also to equipment . . . in the vicinity of the cabinet of origin.” 3. NUREG–1742 states that electrical panel fires were found by most licensees to be one of the most significant potential contributors to fire risk and the methods of analysis applied to panel fires remains an area of quantification uncertainty and debate.*

Table 7. Detailed Incidents Related to Fires and Explosions (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
7	NRC	RES Report	02/2002	On May 15, 2000, a fault occurred on the 12-kV bus duct between the AT and two 12-kV buses. The sustained fault resulted in arcing in the 12-kV bus duct that jumped to and damaged the 4.16-kV bus duct from ST 1-2. ST 1-2 tripped, causing the loss of 4.16 kV to the three vital buses, and the start and loading of all three emergency diesel generators.	Fire and loss of offsite power	Diablo Canyon Unit 1	1. Licensee concluded that the cause of the fault was the thermal failure of the bolted connection of the center conductor of the 12-kV bus. A PVC boot over the connection overheated and created smoke. 2. NRC found that the licensee did not perform PM on the bus duct. Licensee responded that the vendor did not recommend any PM.	Also see Information Notice 2000-14.
8	NRC	IN 1997-01	01/08/97	On April 4, 1996, an operator discovered smoke and fire in the Train B DC equipment room on the 100 ft level of the auxiliary building. Smoke was noticed at the Train B emergency lighting uninterruptible power supply panel in the control room. The fire was located in the 480/120-V essential lighting isolation transformer. The trouble alarms resulting from the lost power supply masked the actual fire alarm in the Train B direct current equipment room; fire department responded and put out the fire within a short period of time.	Loss of power to Train B control room emergency lighting circuits, some general plant essential lighting, and plant fire detection and alarm system panels.	Palo Verde NGS Unit #2	The circuit breaker supplying power to the emergency lighting uninterruptible power supply panel tripped open when wiring insulation in the conduit supplying the power supply panel melted and caused various conductors to short circuit. The circuit breaker trip also deenergized power to the fire detection and alarm panels in the auxiliary building.	1. The fire was related to and caused by a design error in the electrical grounding, which dated back to plant construction. The licensee found similar grounding arrangements in the other two Palo Verde units. 2. The event was of concern because a single electrical fault caused simultaneous fires in the control room and the Train B direct current equipment room which supports postfire safe shut-down capability in the event of a control room fire. This electrical design error is important because it created a fire vulnerability in two separate areas of the plant. The fire could have resulted in operational challenges outside the plant design basis. The vulnerability was caused by the inadequate design of the grounding circuitry from the electrical power supplies, which have been in service since the original construction. The licensee corrective actions included grounding the neutral leg of the isolation transformer and fusing the output of the transformer to limit fault propagation. The licensee also removed ground from the control room emergency lighting uninterruptible power supply panel. These modifications did not affect the isolation function of the transformer.

Table 7. Detailed Incidents Related to Fires and Explosions (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
9	NRC	IN 2001-10	06/28/01	On April 24, 2001, Underwriters Laboratory issued a news release regarding the failures of certain Model GB sprinkler heads made by the Central Sprinkler Co. (Lansdale, Pennsylvania)				The following models are of concern: GB, GB-ALPHA, GB-J, GB-QR, GB-EC, GB-EC, GB-RS, GB-20, GB-20QR, GBR, GB-R1, GB-R2, GBR-LF, GB4, GB4-EC, GB4-FR, GB4-QREC, BB1, BB2, BB3, SD1, SD2, SD3, HIP, ROC, LF, and WS. The sprinkler heads are equipped with either the O-ring seals or disc spring water seals. Only the sprinkler heads with the O-ring seal are of concern.
10	NRC	IN 1999-07	03/22/99	On March 4, 1996, 5 of 11 sprinkler system automatic control valves (Grinnell model A4 deluge valves) failed to trip open during a surveillance testing.	Poor design, deficient maintenance, or inadequate testing of sprinkler system automatic control valves and associated solenoid valves can lead to common-mode failure of the valves to perform their design function.	Farley Unit 1	Licensee's root cause team did not conclusively determine a root cause. However, they noted the plant personnel had used an abrasive cleaning pad to clean the chrome-plated push rod and the push rod guide in the diaphragm retainers. The activity may have created rust particles that caused the diaphragm to stick.	Sprinkler system automatic control valves are used in FP system that protect areas housing both safety- and nonsafety-related equipment for fire safe shutdown. Many of the systems are used to provide primary FP and to meet the requirements of 10 CFR Part 50, Appendix R, Section III.G, *Fire Protection of Safe Shutdown Capability.*
11	NRC	GL 2006-03	04/10/06	A 2005 NRC test showed that both Hemyc and MT, two commonly used fire barrier materials at nuclear power plants, failed to provide the protective function (1-hour and 3-hour) intended for compliance with existing regulations.	Regulations are not being met if the materials are used.	Not specified		1. Hemyc and MT fire barrier systems were installed at nuclear power plants to protect circuits and other electrical and instrumentation features to meet regulatory requirements and plant-specific commitments. 2. All addressees of the GL are required to certain actions in accordance with the request of the GL.
12	NRC	IN 2002-24	07/19/02	This Information Notice alerts licensees to potential issues with using heat collectors on sprinklers and fire detectors installed to satisfy NRC FP requirements.	Incorrectly installed heat collectors could impair sprinkler system response.	Not specified		1. The use of sprinklers with heat detectors installed far below the ceiling has not been demonstrated to be effective and may impair sprinkler system response. 2. Fire areas with large amounts of combustibles (e.g., electrical cables in the cable trays) above the sprinkler may not be adequately protected in accordance with GDC 3 *Fire Protection*.
13	NRC	IN 2005-01	02/04/05	On January 12, 2005, licensees notified NRC of incorrectly connected piping in the Halon systems at both facilities. The piping to the manual-pneumatic actuators in the Halon systems protecting safety-related equipment was found to be reversed.	Vendor testing showed that there could be a 2-second delay in the delivery of Halon to the affected equipment.	Callaway and Wolf Creek nuclear power plants		Improperly configured carbon dioxide and Halon systems have the potential to affect the extinguishing capability of the system.

Table 7. Detailed Incidents Related to Fires and Explosions (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
14	NRC	IN 1998-31	08/18/88	On June 17, 1998, a water hammer occurred and caused the rupture of a 0.305 m (12 in) fire protection isolation valve in the fire protection system in the reactor building, dumping 617,020 L (163,000 gal) of fire water.	Internal flooding of the building.	Washington Nuclear Project Unit 2	Inadequate fire protection system design that leads to a destructive water hammer during anticipated transients when the system is in a normal lineup.	The event started when fire detectors detected smoke from cutting/grinding activities in the diesel generator building. Upon receiving the signal, the fire protection system commanded filling of a normally dry sprinkler header, which led to depressurization of the fire water system and creation of a void in the upper portions of the reactor building vertical fire main risers. An auto-start signal was sent to the fire water pumps on low system pressure. Three fire water pumps immediately started and rapidly reflooded the risers and collapsed the void, thus generating a water hammer.
15	DOE LL Database	L–1998–OR–LMESE TTP	02/17/98	Fire suppression hardware was being installed on UF6 cylinders without the knowledge of the staff responsible for maintaining the cylinders.	None specified in the report.	East Tennessee Tech Park	None specified in the report, insufficient work planning	The need to include appropriate maintenance personnel during planning activities for equipment additions was made clear in this case. Corrective actions included 1. Notify PORTS and Paducah of this event. 2. Provide garage personnel with information they need to include in their files to ensure maintenance would be performed on the new equipment. 3. Contact Industrial prior to performing work on the newly added equipment to ensure the use of appropriate PPE during future maintenance activities.

Table 7. Detailed Incidents Related to Fires and Explosions (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
16	DOE LL Database	Y–2000–OR–BJCPA D–1001	10/31/00	Fire damaged a portion of a metal building with concrete floor. The building was originally used for decontamination purposes. Later it was used as a storage facility. When the building was returned to its original purpose, material continued to be stored there.	Material/equipment damage, possibly personnel injury.	Pacific Western Tech	Fire was caused by an arcing originating from an overhead electric radiant heater. Lack of sufficient administrative (configuration control) control of the facility allowed combustible material to be stored below the heater.	1. If a facility is used for storage of material other than that originally intended for of the facility, reviews need to be made by fire protection personnel. 2. When the facility operations responsibility for use of a facility is turned over to another party, the facility owner needs to continue oversight on the condition/use of the facility. 3. Changes in material stored in any facility, even temporarily, should be reviewed by fire protection personnel for potential changes in fire protection requirements.
17	DOE LL Database	USER-3 2007–NV–NTS–003	02/05/07	A fire protection valve in Building 6-609 failed during the weekend, resulting in draining of two fire water tanks. Water 0.48 m [19 in] was found in the equipment room of the building when staff returned to work the next week.	Water damage (actual), potential loss of building/equipment/lives if a fire broke out while the fire protection system was out of service due to insufficient water pressure.	Not specified	The fire water tanks were equipped with auto-fill features, but no level alarms. The pumps were left on automatic during weekdays. They were shut off on weekends. Building 6-609 was unoccupied at night and on weekends. No routine surveillance was performed in that building.	The report did not specify how the valve failed. Corrective actions included 1. Recommend a continuous monitoring system be installed for the FW tanks to warn loss of water in the tanks. 2. An alarm system for buildings with sprinkler systems to indicate the loss of pressure in the sprinkler system may be warranted.
18	DOE LL Database	2002–RL–HNF–0069	12/30/02	During a routine internal pipe inspection in August 2002, a maintenance crew discovered an inordinate accumulation of debris in the cross-main of a dry pipe fire protection sprinkler system.	If unremoved, the debris could have negatively affected the performance of the sprinkler system when called upon to put out a fire, thereby causing the loss of property/lives depending on the magnitude of fire.	Fluor Hanford	Inadequate oversight of contract work, especially system flushing, was determined to be the direct cause of this vital fire protection system degradation.	Project records contained signed and witnessed documentation from the contractors attesting that all required water distribution and sprinkler system flushes, in accordance with required National Fire Protection Association Standard (e.g., National Fire Protection Association 13) Proper flushing according to contract requirements, should have removed all debris from the system. Lessons learned: Management oversight of contract work is essential to ensure that work is properly performed according to

Table 7. Detailed Incidents Related to Fires and Explosions (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
19	DOE LL Database	CH–AA–ANLE–ANL EESN–1998–001	07/08/98	Eleven of the 12 pendant sprinklers failed to operate when the links were fused during a operability test. The 12 sprinklers had been replaced to resolve a problem in one of the occurrence reports.	Failure of sprinklers to open prevents the initiation of water flow signal, thereby delaying the emergency response.	Argonne National Laboratory East	None specified in the report; none of the sprinklers were obstructed with foreign material; all exhibited signs of corrosion. However, it's unclear whether corrosion played a role in the failure of the sprinklers.	DOE fire protection community was tasked to develop a corrective action.
20	DOE LL Database	2006–SR–WSRC–0051	12/05/06	An incorrect number 226.5 m^3 [8,000 ft^3] of air per lb of wood) was used in an accident analysis. The correct number should have been 2.3 m^3 [80 ft^3] of air per lb of wood. Because of the error, the calculations concluded that the fire would be air limited because the structure's design restricted the air flow.	None identified; (my words) the fire could either be fuel limited or there could be sufficient air to consume the entire fuel, thus producing a fire with larger heat release rate. (pure conjecture on my part).	Savannah River Site H-Tank Farm	1. The originator of the calculations did not show the calculation of how the input for amount of air required to sustain combustion for a pound of wood was derived. 2. The verified calculation failed to ensure accuracy of the calculation. 3. The National Fire Protection Association Handbook from which the equation was referenced required that inputs obtained from a table be converted to different units before the values could be used in the formula. The handbook did not warn users about the need to convert the values prior to their use in the formula.	Corrective actions: 1. Submitted an error notification form to National Fire Protection Association. 2. Submitted revision to correct the error in the calculations. Performed broadness review of other fire analysis calculations. 4. Shared this lesson learned with other personnel (e.g., management, engineering) as appropriate.

Table 7. Detailed Incidents Related to Fires and Explosions (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
21	DOE LL Database	LANL–ESHQ–2006–0001	09/21/06	Corrosion and nonmanufacturer's paint were identified on approximately 5 to 60 percent of the sprinkler heads (depending on the room) within the facility. The automatic sprinkler system is a safety-significant SSC and is required to be continuously operable.	None identified; the sprinkler heads might not be inoperable when activated, thereby prolonging the duration of a fire. In the worst case, a fire could get out of control depending on its magnitude, origin, and other factors.	Los Alamos National Laboratory TA-55	1. Incomplete automatic sprinkler system data info entry within the Los Alamos National Laboratory master equipment list and computerized maintenance management system databases. 2. Assumption by facility management that performance of the National Fire Protection Association 25-required semiannual automatic sprinkler system inspection, testing, and maintenance evolution twice annually satisfies the National Fire Protection Association 25-required annual inspection, testing, and maintenance evolution. In reality, the scopes of these two evolutions are different.	Improper masking/protection of sprinkler heads during painting of sprinkler piping or renovation/painting of rooms (especially spray painting) is a recurring cause of painted sprinkler heads. Ongoing fire protection and life safety-related facility inspections by the Los Alamos National Laboratory fire protection inconsistently discovered occurrences of painted or corroded sprinkler heads during annual/biennial/triennial facility walkdowns. Corrective actions: 1. TA-55 replaced all suspect sprinklers throughout the facility. 2. Other facilities (e.g., TA-3-29 CMR, TA-3-1076 BSL-3) either replaced suspect heads or removed the painted decorative cover plates and sealing gaskets on the affected sprinkler heads. 3. Facilities prepared additional procedures to incorporate National Fire Protection Association 25 ITM expectations for automatic sprinkler system. 4. Additional training of fire protection inspection personnel was provided to emphasize the National Fire Protection Association 25 visual inspection expectations. 5. There are other actions taken, TNTC.

Table 7. Detailed Incidents Related to Fires and Explosions (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
22	DOE LL Database	2003–RL–HNF–0033	11/18/03	Contrary to the T-Plant Occupancy Permit, the 0.9 m [36-in] minimum egress width was not maintained in the T-Plant tunnel. A hose reel on the wall projected into the 0.9 m [36-in] egress space when the large diameter container trailer was in place.	None specified; personnel egress during emergency may be impeded if the minimum width is not maintained.	Hanford	Administrative controls involving the fire protection engineer in the hazard planning and procedure validation process were weak. No procedural requirement for the fire protection engineer to initially or periodically walk down the requirements of fire permits. Other administrative mechanisms, such as the unreviewed safety question process, would not consistently verify that the physical configuration implemented the fire permit requirements.	The Automated Job Hazard Analysis process did not require the fire protection engineer to be involved unless a hot work permit was listed. Lessons Learned: The fire protection engineer's involvement in walking down the fire hazard analysis requirements can be vital in complying with documented safety analysis controls. Corrective actions: 1. The Automated Job Hazard Analysis was revised to include more specific questions that would require the involvement of the fire protection engineer for fire hazard analysis-related issues. 2. HNF-RD-8589 (will be revised to require the fire protection engineer walk down new and revised fire permits. The next two were listed for consideration 1. Provide training on the fire hazard analysis requirements to unreviewed safety question evaluators, procedure technical authorities, and work control planners whenever facility configuration or processes change 2. Walk down all long-term permits whenever facility configuration or processes change.
23	DOE LL Database	Y–2000–OR–BJCX10–0601	06/12/00	Quicklime (anhydrous calcium oxide/magnesium oxide) reacts with water exothermally. Twenty-three tons of quick lime were delivered to Oak Ridge National Laboratory to be stored at an outdoor berm constructed with straw bales. Rain was pouring down one day after delivery. Water came into contact with the quicklime. Heat of reaction melted the plastic covers and ignited the straw bales.	None specified; air monitor registered normal radiation level; no personnel injury or equipment damage was reported.	Oak Ridge National Laboratory	Configuration control was not adhered to. During the project, straw bales were added to a berm that contained the plastic-covered lime. The modification was implemented in the field without a comprehensive review of the hazards identified in the Material Safety Data Sheets or evaluation of the current industry standards. The field change resulted in placing combustibles near the quick lime, which resulted in the fire.	Quick lime was used to stabilize contaminated sediments so the resulting sludge could be removed for remediation. Lessons Learned: Combustible material close to stored quick lime can result in a fire if the quick lime is exposed to water. Corrective Actions: Ensure a complete hazards analysis, which includes reviewing hazards identified in Material Safety Data Sheets or current industry standards, is performed any time modifications are made to a work plan.

Table 7. Detailed Incidents Related to Fires and Explosions (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
24	DOE LL Database	HQ–EH–2000-02	10/4/00	On August 25, 2000, a gas-operated dry chemical fire extinguisher at the Port of Rotterdam, Netherlands, exploded when activated, killing an employee with shrapnel.	Personnel death and (possibly) property damage.	Rotterdam, Netherlands	Corrosion occurred under a rubber/plastic base protecting the bottom of the extinguisher. The base had trapped moisture next to the shell of the extinguisher, thereby accelerating corrosion. The extinguisher was manufactured in 1987 by Ansul, Belgium.	Also see NRC IN 2001-04. Lessons Learned: Fire extinguishers require periodic maintenance to ensure their readiness for emergency use. Some maintenance actions are vital to ensuring the life and safety of the user. Corrective Actions: According to DOE, facilities should check their fire extinguisher maintenance program to ensure all units are properly inspected and tested (from IN 2001-04). NRC endorses the use of National Fire Protection Association 10 Standard for Portable Fire Extinguishers. National Fire Protection Association 10 provides guidance for selection, installation, design, inspection, and maintenance of portable fire extinguishers.
25	DOE LL Database	L–997–OEWS–45-0 2	11/17/97	On November 3, 1997, a flexible exhaust duct caught on fire when a piece of hot slag from a nearby cutting operation fell on the duct. The fire watch extinguished the fire. The fire watch received medical treatment for smoke inhalation.	No property damage or radiation releases.	An NRC-regulated commercial nuclear hot-cell facility that was undergoing decontamination and decommissioning.	Open flames, electric arcs, hot metals, sparks, and spatter are ready sources of ignition.	Lessons Learned: This incident illustrates the potential dangers involving in welding, cutting, and grinding operations. It also underscores the important role the fire watch plays in safeguarding these activities. Corrective Actions: 1. Prior to fighting a fire with a portable extinguisher, the fire director should be notified. If the fire can't be contained, the area should be evacuated and left to the firefighters. 2. Managers at DOE facilities undergoing decontamination and decommissioning need to ensure that vendors and subcontractors understand the local work control practices and the importance of following safety requirements.
26	DOE LL Database	AAN-U-01-112A	37142	Operator noticed fire start after he had aligned valves at the H2 storage facility in preparation for putting the H2 injection system into service. The fire occurred in 1999.	Fire potentially endangered the nearby H2 storage tanks. The overhead 115kV reserved power lines were deenergied to protect fire fighters. Facility entered corresponding TS LCO for loss of offsite power.	JAF NPP Plant	Organizational and programmatic deficiencies resulted in multiple component failures. The H2 equipment was vendor supplied and maintained. Vendor preventive maintenance program and JAF oversight of the program was deemed inadequate.	Lessons Learned: Property maintaining, monitoring and overseeing of H2 storage facility equipment can minimize the risk of fie or explosion. NRC Special Report 50-333/99-02, ADAMS Accession # 9904010078. Also see IN 2001-12.

**Operating Experience Assessment Energetic Faults in 4.16 kV to 13.8 kV Switchgear and Bus Ducts that Caused Fires in NPPs 1986 - 2001,* ML021290358

Table 8. Operating Experience Related to Administrative Controls								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
1	NRC Information Notice*	97-51	07/11/97	As a result of NRC staff inquiries of some licensees to provide information related to the movement of spent fuel storage or transportation casks without the lids on those casks being secured, one licensee determined that an unreviewed safety question was introduced by the existing practice of moving transportation casks with the lids only partially secured. The practice involved an unreviewed safety question because the actual cask configuration differed from the configuration assumed in the cask drop analysis in the affected facility's safety analysis report.	No radiological consequences. Unreviewed safety question due to facility operations outside basis of facility safety analysis report.	Various	Plant practices evolved over time resulting in an unreviewed safety question because practices no longer matched facility safety analysis report.	Title: NRC Information Notice 97-51: problems experienced with loading and unloading spent nuclear fuel storage and transportation casks
1 (cont)	NRC Information Notice	97-51	07/11/97	Report on licensees' problems during the movement of casks as a result of crane interlocks, errors in the accounting for the weights of cask components, and human error. "In moving cask components at the Davis-Besse Nuclear Power Station, both during the dry run exercises and the actual loading of a cask, actuation of electrical thermal overloads interrupted crane operations during slow speed operation. The licensee subsequently learned that a creeper motor installed on the crane was intended to be used during sustained slow speed movements instead of the main hoist motor."	No radiological consequences. Facility operations outside of assumed conditions.	Davis Besse	Causes include less than adequate reference documents, less than adequate communications	Title: NRC Information Notice 97-51: problems experienced with loading and unloading spent nuclear fuel storage and transportation casks
1 (cont)	NRC Information Notice	97-51	07/11/97	"At Prairie Island . . . a cask remained in the hoisted position above the spent fuel pool for approximately 16 hours while the licensee developed and implemented corrective actions to address an overload-sensing system that was inaccurately calibrated for lifting of a loaded dry storage cask. Changes in the lifting procedure were required at Prairie Island when it was discovered that a dry storage cask weighed more than expected. The weight difference was found to be the result of acceptable variations in manufacturing tolerances that had not been accounted for in previous weight calculations."	No radiological consequences	Prairie Island	Causes include less than adequate calculations from not accounting for manufacturing variations.	
Note that what is captured in the consequences and causes columns is limited by the source documents; the amount of information available for the entries varied by source and individual events. Note also that the term "root cause" is used loosely here. Most of the causes listed in the root cause column are actually contributing causes or causal factors rather than what would be captured as root causes in a formal root cause analysis. For licensee event report and inspection finding entries, what is listed under "root causes" includes the "causal factors" captured in the HFIS database.								

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
2	NRC Information Notice	99-21	06/25/99	Annunciator alarmed in the control room for "Spent Fuel Pool Level/Temperature." Spent fuel pool temperature had reached 126 °F. Nuclear operator subsequently found that spent fuel pool pump 1-2 was not operating as expected. Licensee's investigation revealed that operator logs prepared earlier had verified that the spent fuel pool pump 1-2 was operating as required and that spent fuel pool temperature was 100 °F. "Further investigation revealed that during the day, relay CIAX-H was replaced . . . The control circuit associated with the CIAX relay trips the spent fuel pool cooling pumps during an accident scenario to prevent overloading of the emergency diesel generators."	No radiological consequences. "The relay had been replaced at approximately 1 p.m., and as a result, spent fuel pool cooling had been lost for approximately 4 hours before the high level/temperature alarm was received in the control room. Licensee engineers determined that the spent fuel pool heatup rate was approximately 8 °F per hour and would have resulted in spent fuel pool boiling after approximately 16 hours."	Diablo Canyon	Work order (for relay replacement) less than adequate because it did not contain any precautions or limitations to notify operators of the trip of the spent fuel pool cooling pump as a result of removal of the relay; prejob briefing less than adequate because it did not identify the condition; communications less than adequate to operators or electricians who performed the relay replacement; lack of controls or indications in the control room of the status of the spent fuel pool cooling pumps, the temperature of the spent fuel pool, or the level of the spent fuel pool, other than the aforementioned level/temperature alarm.	NRC Information Notice 99-21: recent plant events caused by human performance errors.
3	NRC Information Notice	97-68	09/03/97	"Diver entered the spent fuel pool...to commence work on an upender limit switch at the south end of the fuel transfer area, the only surveyed and authorized work area...No wall or shield (other than the pool water) separates the area from the fuel storage racks on the east side...[many dose monitors employed]... Unlike previous dives into the refueling cavity which employed underwater closed-circuit television (video) to visually monitor the diver, a technician at the pool surface was assigned to observe the diver through a floating window box during the fourth dive . . ." Diver noticed anomalies in pool, wanted to investigate, while his observers were distracted or couldn't see where he was because of air bubbles from venting dive suit; resulted in diver entering high radiation area, instructors instructing him to survey area to find source instead of evacuate because they didn't understand where he was; dive was only suspended after survey meter reading reached 3 rem/hr.	"Following TLD processing, the licensee calculated a maximum dose to the extremities (right knuckles) of 8.85 mSv (885 mrem) based on a wrist TLD badge shallow dose equivalent result of 4.24 mSv (424 mrem). The licensee also calculated a dose of 2.7 mSv (270 mrem) to the highest exposed portion of the whole body (arm above the elbow) as compared to a maximum TLD reading on the head of 1.37 mSv (137 mrem). The maximum dose to the lower extremity (ankle) was 0.021 mSv (21 mrem) shallow dose equivalent."	Calvert Cliffs	"1. The scope of work was not clearly understood by all parties involved. 2. The diver was given inadequate instructions about the location and magnitude of the radiation sources accessible to him. 3. Positive control over the diver in the pool was inadequate. 4. Licensee failed to adequately evaluate the diver's exposure status before authorizing additional work in the RCA."	NRC Information Notice 97-68: loss of control of diver in a spent fuel storage pool.

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
4	NRC Information Notice	97-39	06/26/97	Information Notice issued “to alert addressees to inadequate safety evaluations performed under Section 72.48 of Title 10 of the Code of Federal Regulations (10 CFR 72.48). Section 72.48, ‘Changes, tests, and experiments,’ states that a holder of an ISFSI license may make changes in the ISFSI described in the safety analysis report (SAR), may make changes in the procedures described in the SAR, or may conduct tests or experiments not described in the SAR, without prior NRC approval, unless the proposed change, test, or experiment involves a change in the license conditions incorporated in the license, an unreviewed safety question, a significant increase in occupational exposure, or a significant unreviewed environmental impact. A proposed change is deemed to involve an unreviewed safety question if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR may be created, or if the margin of safety as defined in the basis for any technical specification is reduced. The licensee is required to maintain records of changes made to the ISFSI, which include a written safety evaluation that provides the bases for the determination that each change, test, or experiment does not involve an unreviewed safety question.”	No radiological consequences. Unreviewed safety questions, inadequate safety evaluations discovered.	Examples from Arkansas Nuclear One, Point Beach, Prairie Island	Potential causes include procedures and work practices for potential licensing-basis change evaluations less than adequate	NRC Information Notices 97-39: Inadequate 10 CFR 72.48 Safety Evaluations of Independent Spent Fuel Storage Installations

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
4 (cont)	NRC Information Notice	97-39	06/26/97	An example: "Another violation was issued to PBNP, in part, because the licensee failed to perform adequate safety evaluations for two procedures. The violation was based on an augmented inspection team inspection at PBNP (Enforcement Action [EA] 96-273 and Inspection Report No. 50-266/301-96005). The licensee did not perform a 10 CFR 72.48 safety evaluation for a lifting evolution that created a potential for dropping the VSC-24 multi-assembly sealed basket (MSB) transfer cask off the top of the ventilated concrete cask, an accident not described in the SAR. The licensee also did not provide sufficient technical justification to support conclusions in a safety evaluation for an MSB weighing procedure. The safety evaluation did not include supporting information that ensured that the shield lid would not be inadvertently removed from the MSB and expose the workers to spent fuel. Before lifting the lid, the licensee developed a new weighing procedure that did not create the potential to inadvertently remove the lid."	No radiological consequences	Point Beach	Inadequate evaluations and procedure checks	
5	NRC Information Notice	99-29	10/28/99	"The NRC has identified two instances where licensees loaded storage casks with spent fuel containing BPRAs and TPDs, which were not authorized in either the site-specific license or the cask [Certificate of Compliance]. Therefore, the licensees violated the terms of their NRC license, which specifies the materials that are permitted for storage in dry casks... In the two known instances where BPRAs and/or TPDs were loaded in dry casks, without NRC authorization, five violations were identified."	"Although the safety significance associated with these specific instances is low, in each case the licensee was required to take immediate action to either correct the cask loading or to justify the continued safety of the as-loaded cask conditions."	Various	Inadequate evaluations; procedures and reference documents or work practices less than adequate.	NRC Information Notice 99-29: Authorized Contents of Spent Fuel Casks
6	NRC Information Notice	02-09	02/13/02	Information Notice "to alert addressees to the recent nozzle separation and dropping of a Westinghouse fuel assembly during movement. Even though the nozzle separation affects only fuel of a type last manufactured almost 20 years ago, the fuel is perhaps being moved to dry storage or high-density racks and could drop during movement...On March 24, 2001, operators at the North Anna Power Station of Virginia Electric and Power Company were inspecting older spent fuel assemblies in advance of transferring them to dry cask storage. As assembly G45 was being returned to its spent fuel rack, the top nozzle separated from the assembly and the assembly dropped about 12 feet into its storage cell. The top nozzle, with the burnable poison rod assembly still attached, remained on the handling tool."	"Since the assembly bottom nozzle was already in the cell, the falling assembly did not contact any other fuel assemblies or the rack structure. There was no collateral damage. An initial visual inspection of the top of the assembly within the cell using a TV camera revealed that the bulge joints connecting the stainless steel sleeves to the Zircaloy 4 guide tubes had failed. No fission gas activity was detected afterwards, indicating that none of the fuel rods in the assembly had been fractured by the drop."	North Anna	Fabrication/quality control: "The method of fabrication of the top grid assembly is believed to have been among major factors in these failures...According to Westinghouse, North Anna visually inspected 208 fuel assemblies; 54 had indications of corrosion at the bulge joint and 10 had indications of cracking."	NRC Information Notice No. 2002-09 Potential for Top Nozzle Separation and Dropping of a Certain Type of Westinghouse Fuel Assembly

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
7	NRC Information Notice	99-15	05/27/99	"The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to a potential problem with use of the vendor's consolidated safety analysis report for the IF-300 spent fuel shipping cask, which could place plants outside their design basis during the loading or unloading of spent fuel."	Facility outside design basis; unanalyzed condition, namely no consequence evaluation available before issue resolved	Example from Harris	Communication between plant and vendor less than adequate. Plant assumed vendor had looked at planned operations and knew about bolts being removed prior to movement. But this placed the cask in a configuration different than transportation-ready (what Part 71 tests analyze); hence. It was unanalyzed.	NRC Information Notice 99-15: Misapplication of 10 CFR Part 71 Transportation Shipping Cask Licensing Basis to 10 CFR Part 50 Design Basis
8	NRC Information Notice	00-11	08/07/00	Information Notice "to remind general and site specific licensees of their responsibilities to assure that the quality assurance requirements of Part 72, Subpart G, to Title 10 of the U.S. Code of Federal Regulations (CFR) have been met before a dry cask storage system is placed in service at their nuclear power plants. The regulations require that nuclear power plant licensees assume full responsibility for the overall safety and operational use of the dry cask storage system at their sites. The nuclear power plant licensee is also responsible for assuring that the fabrication and preparation for use of the dry cask storage system, and the contractor's activities associated with the dry cask storage system, conform with NRC regulations, the Certificate of Compliance (CoC), and the license conditions for the nuclear power plant. This IN discusses a number of examples of inadequate implementation of quality assurance (QA) programs identified in recent NRC inspections."	Inadequate quality assurance	Various	Inadequate oversight of contractor's activities	NRC Information Notice 2000-11: Licensee Responsibility for Quality Assurance Oversight of Contractor Activities Regarding Fabrication and Use of Spent Fuel Storage Cask Systems

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
9	NRC Information Notice	02-03	01/10/02	"During the movement of the ACS [advanced crusher and shearer], the refueling floor local area radiation monitor began to alarm. The cause was a previously unidentified highly radioactive particle which had fallen from the ACS. The particle was later determined to be a 2.78 gigabecquerel (Gbq) [75 millicuries (mCi)] Co-60 particle, reading approximately 8 sievert/h (Sv/h) (800 rem/h) at contact...During the cleanup activities, more than 30 radioactive particles were found on the refueling floor."	"Two high activity radioactive particles found on September 9 and December 6, 2000, had resulted in shallow-dose equivalent (SDE) exposures of 0.12 and 0.17 Sv (12 and 17 rem), which is below the annual SDE limit of 50 rem. The licensee discovered two more high activity particles, a 0.78 Gbq (21 mCi) particle on November 28, and a 0.7 Gbq (19 mCi) particle on December 4, 2000; these particles did not result in significant exposure to personnel. No actual exposures in excess of any annual dose limits occurred during the cleanup activities...Had the particles been directly on the workers' PCs, the TEDE annual limit of 0.05 Sv (5 rem) could have been exceeded in 25 seconds to 2 minutes, and the SDE limit exceeded in 6 to 21 seconds, depending on the activity of the individual particle."	Susquehanna	"The licensee's evaluation had failed to consider properly and account for the potential for substantial dose to personnel from the high-activity particles. Specifically, the 15-minute worker stay time was not adequate to prevent potential overexposures from the particles known to be present in and around the refueling floor."	NRC Information Notice No. 2002-03: Highly Radioactive Particle Control Problems During Spent Fuel Pool Cleanout
10	LER database	50-368/ 2000-003-00	12/14/00	With Unit 2 core reload in progress during a scheduled refueling outage, refueling machine underload indications were received and core reload was suspended. Investigation revealed that the weight of the dummy fuel assembly used to calibrate the refueling machine was approximately 104 lb heavier than the value used for calibration.	No radiological consequences	Arkansas Nuclear 2	Underdeveloped refueling machine calibration procedures; personnel incorrectly calibrated refueling machine	Title: Overload Cut Off Limits For The Refueling Machine Were Not Set As Required By Technical Specifications Due To An Incorrect Dummy Fuel Assembly Weight Being Used for Calibration
11	LER database	50-316/ 2001-005-01	01/11/02	The Rod Control Cluster Assembly (RCCA) tool was mistakenly moved over the spent fuel pool fuel racks. The surveillance requirement to determine the potential impact energy as within this limit before moving each load over the fuel racks also was not performed. The SF crane operator mistakenly moved the load over the spent fuel pool racks, failing to reinstate the hoist height interlock before moving the crane.	The potential impact energy of the RCCA tool is greater than the limit of 24,240 in-lbs detailed by TS.	D.C. Cook 1	Failure to performance peer check prior to RCCA tool movement	Title: RCCA Tool Over Spent Fuel Pool Racks Technical Specification Violation.

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
12	LER database	50-316/ 2000-011-00	08/21/00	The spent fuel pool exhaust ventilation system was inoperable with fuel inspections in progress. Auxiliary building crane inspections were also in progress which require the ventilation system to be operable. Ventilation systems are not capable of responding to a fuel accident quickly enough to prevent an unfiltered release to the atmosphere. Therefore, a compensatory action was put in place to ensure that the spent fuel pool ventilation system is in the charcoal filter mode of operation during fuel handling operations. This event is a violation of Technical Specification 3.9.12.	No radiological consequences	D.C. Cook 2	Inadequate/miscommunication between control room and inspections personnel; no prejob brief conducted for auxiliary building crane inspections; control room personnel not involved in prejob brief for the fuel top nozzle inspections	Title: Spent Fuel Pool Exhaust Ventilation System Inoperable During Fuel Movement. Same event as ID# 5 in Section 3.2.6 HVAC, Ventilation or Filtration
13	LER database	50-237/ 1998-012-00	09/18/98	In violation of TS, fuel was moved while the refrigeration condensing unit of a HVAC was inoperable.	No radiological consequences	Dresden 2	Failure to thoroughly review changes in work schedule; failure to consider contingency measures in LCO review; failure of unit supervisor to recognize violations of TS; work planning process placing too much reliance on operations department to manage TS adherence; failure of senior operator to recognize LCO's effect on refueling activities.	Title: Fuel bundle movement permitted during control room ventilation outage due to programmatic failures within the work planning and execution process. Same event as ID# 8 in Section 3.2.6 HVAC, Ventilation or Filtration
14	LER database	50-348/ 2000-003-00	04/13/00	During fuel shipping activities, a valid radiation alarm occurred on an spent fuel pool ventilation radiation monitor, resulting in an automatic start of the B-train penetration room filtration (PRF) system. This resulted in automatic shutdown of the normal spent fuel pool ventilation system, causing an automatic start of the A-train penetration room filtration system. Although the release of radioactive gases into the spent fuel pool area was expected and the potential for radiation monitors alarming was communicated to the control room, the potential for the automatic start of the penetration room filtration system (which functioned as designed) was not recognized.	No radiological consequences	Farley 1	Procedures for leak detection inadequate; failure to consider effect of gas release when performing fuel shipping activity during work package development, QA, and use.	Title: Penetration Room Filtration Automatic Start During Fuel Sipping
15	LER database	50-348/ 2000-004-00	04/20/00	Three spent fuel assemblies had been loaded in configurations contrary to TS. Manual verification, as well as the review of the verification process, of the acceptability of proposed offload configuration failed to identify that the proposed configuration did not meet the acceptable configurations.	No radiological consequences; TS violation	Farley 1	Personnel responsible for developing, performing, and verifying spent fuel pool configuration did not recognize configuration as unacceptable; personnel responsible for developing spent fuel pool configuration lacked sufficient knowledge to determine an acceptable configuration; lack of detail in core offload procedure; insufficient independent review in the verification process.	Title: Three Spent Fuel Assemblies in Spent Fuel Pool Locations Not Allowed by Technical Specification 3.7.15

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
16	LER database	50-255/ 1999-005-00	12/06/99	During refueling plant shutdown, charcoal filter for fuel storage building ventilation system was not in operation during fuel handling activities, in violation of TS. TS 3.8.4 requires the ventilation system and charcoal filter to be in operation whenever irradiated fuel that has decayed less than 30 days is being handled in the fuel storage building. A licensed operator prematurely signed off on a checklist for the ventilation system; the operator intended to properly align the charcoal filter upon notification that fuel handling activities were to commence, but was not notified when fuel handling activities were authorized by the control room supervisor.	No radiological consequences; TS violation	Palisades	Inadequate maintenance practices, lack of communication by control room supervisor when fuel moves were authorized.	Title: Charcoal Filter Not in Service During Movement of Irradiated Fuel Assemblies
17	LER database	50-362/ 2001-005-00	06/15/01	Fuel movement occurred in the spent fuel pool while the control room emergency air conditioning system was available, but not operable. Control room emergency air conditioning system was available through the DC bus powered by the battery charger, but it was not operable due to the B 125 DC battery disconnect being open to support maintenance on the battery.	No radiological consequences; TS violation	Salem 1	Lack of knowledge, by all individuals involved (licensed operators, outage control center) recent installation of battery disconnect switches and accompanying procedure changes without clarification on control area ventilation system requirements.	Title: Control Room Emergency Air Intake Dampers Inoperable During Spent Fuel Pool Moves
18	LER database	50-362/ 2001-002-00	03/27/02	Both new and irradiated fuel was moved while train B of the Post Accident Cleanup Unit (PACU) was inoperable and PACU train A was not placed in service. TS requires two PACU trains to be operable during movement of irradiated fuel assemblies in fuel handling building.	No radiological consequences; TS violation	San Onofre 3	TS requirements not correctly implemented in plant procedures.	Title: Starting the Movement of Irradiated Fuel with One Train of PACU Inoperable causes TS Violation
19	LER database	50-395/ 1999-003-00	05/06/99	Refueling crew started control rod unlatching evolution during core alteration, when the weight indicated by the load cell was noted to be incorrect. The crew assumed the load indicator had failed and did not notice that “peak load” had been selected for the load cell switch position instead of “continuous.” The crew installed a new load cell, for which the TS Prior-to-use surveillance test was note performed, and unlatched the first control rod drive shaft without requesting permission.	No radiological consequences; TS violation	V.C. Summer	Lack of familiarity with surveillance test requirement and operational procedures—weakness in work package development, QA, and use; inadequate maintenance practices; lack of familiarity with load cell features—inadequate technical knowledge.	Title: Missed Surveillance on Manipulator Crane Load Cell
20	LER database	50-387/ 2002-005-00	08/26/02	Dry Fuel Storage Canister Filled With Incorrect Gas Due To Human Error—Due to human error, argon was used instead of helium to fill a dry shielded canister. This event was caused by human error. Argon is used as a welding shield gas, and helium is used as a heat transfer media in the canister.	The potential for heatup of the fuel and cladding damage was the issue for this event because argon has approximately 1/10 the thermal conductivity of helium. However, an analysis was conducted and determined that there was no fuel damage or radiological releases for this particular event.	Susquehanna 1	Argon and helium canisters same color, stored together (due to change in gas supply vendor)— latent error; mechanic tested a few canisters in car and erroneously assumed all were helium; inspector only verified pressure, verifying the correct gas is used to backfill the DSC was not identified as a “critical” procedure step; no peer check process.	Title: Dry Fuel Storage Canister Filled with Incorrect Gas Due to Human Error. Same event as ID# 1 in Section 3.2.4 Opening and/or Closing Canisters or Casks

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
21	LER database	50-390/ 2005-001-00	05/10/05	Fuel movement began in the spent fuel pool for inspection of fuel assemblies, while containment hatch was opened and containment purge system activated for refueling outage support, which in turn makes both trains of the Auxiliary Building Gas Treatment System inoperable. Auxiliary Building Gas Treatment System is required operable during movement of irradiated fuel assemblies.	No radiological consequences; TS violation	Watts Bar 1	Inadequate systems operation instruction; inadequate fuel handling instruction/technical knowledge	Title: Two Trains of ABGTS Inoperable.
22	NRC Inspection Report	2005003	8/05/05	Administrative controls for foreign material control/exclusion not being followed for the spent fuel pool.	No radiological consequences	Arkansas Nuclear One	Causal factors include: housekeeping less than adequate, work practices less than adequate, individual corrective action less than adequate	PI & R semiannual trend
23	NRC Inspection Report	2004008	01/26/05	Inadvertent bumping of two fuel assemblies due to not following procedures during “unusual” evolution.	One fuel assembly (once burned) suffered damage; no fission product releases	Braidwood 1 & 2	Causal factors include oral communications, work practices, or craft skills less than adequate	Integrated inspection; noncited violation, green finding
24	NRC Inspection Report	2006004	10/30/06	“An NRC-identified non-cited violation of Technical Specification 5.4.1, Administrative Controls (Procedures), was identified for the failure to adhere to procedure requirements when operators injected service air into the steam jet air ejectors and the offgas flowpath. The initial condition that the service air injection was needed for continued hydrogen water chemistry operation was not met. As a result of this procedure adherence deficiency, the licensee had reduced the ability to monitor for actual fuel cladding damage.” (p. 7)	This finding is more than minor because it involved adherence to procedures associated with fuel cladding integrity and affected the Barrier Integrity Cornerstone to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The finding was determined to be of very low safety significance because it was only associated with the ability to monitor fuel barrier integrity. (p. 8)	Byron 1 & 2	This finding was related to the cross-cutting area of Human Performance because the cause was due to failure to adhere to procedures (p. 8).	Integrated inspection; noncited violation (TS), green finding
25	NRC Inspection Report	2005009	11/02/05	Lateral stresses on “stuck” (failure to unlatch) new assembly due to failure to follow procedures; plan to inspect for damage.	Potential FA damage	Byron 1 & 2	Causal factors include work practices or craft skills less than adequate.	Integrated inspection; noncited violation (TS), green finding
26	NRC Inspection Report	2003008	10/30/03	“On September 26, 2003, at 7:46 p.m., while transferring a fuel assembly from its core location to the containment upender/downender, the refueling machine mast contacted the rod cluster control assembly change fixture basket in the fuel transfer cavity. At the time of the incident, the refueling machine was being operated with travel interlocks bypassed continuously due to an obstruction (bent ladder) in the fuel transfer path.”	No radiological consequences. A condition adverse to quality was not promptly (1) identified or (2) corrected (2 findings)	Byron 1	Causal factors include oral communications less than adequate, no oral communication when needed, work practices or craft skills less than adequate, problem resolution inadequate, work practices—failure to stop work/nonconservative decisionmaking, work planning—inadequate staffing for task.	Special inspection; non-cited violation, green finding

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
27	NRC Inspection Report	1999020	02/07/00	Three fuel handlers incorrectly identified/verified the position of the spent fuel pool (SFP) bridge crane over a designated fuel assembly storage location, which resulted in the mispositioning of a fuel assembly within the spent fuel pool during fuel movement. "A fuel handler who operated the SFP bridge crane incorrectly positioned the crane over SFP storage location R-J12 instead of Q-J12, which was adjacent storage. A second fuel handler and the fuel handling supervisor then incorrectly verified the crane's position to be over SFP storage location Q-J12."	"No adverse safety consequences. The fuel storage requirements defined by the SFP criticality analysis were met at all times and the design basis for fuel assembly storage in the SFP was bounded for the mispositioning of a single fuel assembly."	Byron 1 & 2	Causal factors include work practices—inadequate procedural adherence, work practices— inadequate independent verification, awareness/attention— worker distracted.	NCV—violation of 10 CFR Part 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings"
28	NRC Inspection Report	2005005	02/14/06	LTA risk management controls of spent fuel pool water inventory following core off-load; human error common thread (IN–05–16).	Potential partial loss of spent fuel pool water inventory	Callaway	Causal factors include work practice or craft skills less than adequate, procedure or reference documents less than adequate	Integrated inspection; green finding
29	NRC Inspection Report	2002006	07/30/02	"A Non-Cited Violation of Technical Specifications 5.4.1 was identified for workers failing to follow a procedure which contributed to the inadvertent lifting of a double blade guide during fuel movement operations on April 9, 2002."	Affected fuel barrier integrity cornerstone.	Clinton	Causal factors include: awareness/attention—self-check less than adequate, work practices—independent verification less than adequate	Baseline inspection; noncited violation, green finding
30	NRC Inspection Report	2005009	11/29/05	10 CFR 50.59, "Changes, Tests, and Experiments" Violation—Failure to obtain a license amendment prior to implementing a new methodology for determining spent fuel pool heat loading.	No radiological consequences	Columbia	Causal factors include licensing documents—procedure/referenc e documents less than adequate	Baseline inspection; noncited violation (50.59), green finding
31	NRC Inspection Report	2001002	04/18/01	"Technical Specification 5.4.1 states, in part, that written procedures shall be established, implemented, and maintained. Step 5.9.2 of procedure SOP-506, 'Spent Fuel Pool Cooling and Cleanup System,' states to close Valves XSF–0220, XSF–0067 and XSF–0068 following completion of spent fuel pool transfer canal draining operations. Contrary to this requirement, Valve XSF-0220 was found open on February 1, 2001, following completion of transfer canal draining operations which established a gravity drain path from Spent Fuel Pools X–01 and X–02 to the recycle holdup tank." (p. 10)	TS violation	Comanche Peak 1 & 2	Causal factors include work practices or craft skills less than adequate.	Normal Resident Inspectors' inspection; noncited violation, "low safety significance"

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
32	NRC Inspection Report	2003005	07/25/03	"The failure to implement corrective actions to prevent dropping items in the spent fuel storage pool was a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI. During preparations for the refueling outage, the licensee dropped a control rod blade in the pool. This was similar to an event in 1999 when a shroud head bolt was dropped in the pool. The root causes of these two events were similar; however, the corrective actions for the 1999 event failed to preclude the most recent event." (p. 10) "The licensee performed a root cause analysis of this event and concluded that this was an infrequently performed task, some of the personnel involved were unfamiliar with the task, and there was no procedure for this activity. This was despite the fact that Administrative Procedure 0.24, 'Working Over or in Reactor Vessel or Fuel Pool Requirements,' Revision 18, required that 'a SORC [Station Operations Review Committee]/IQA [Independent Qualified Approver] approved document is required for all loads moved over or near irradiated fuel.' Furthermore, on November 1, 1999, the licensee dropped a shroud head bolt while moving it in the SFSP. The lack of a procedure for that activity was cited as a root cause for that event in 1999." (p. 11)	"This finding was more than minor since dropping a control rod blade in the spent fuel pool could be viewed as a precursor to a significant event and was of very low safety significance since it did not represent an actual degradation of any fission product barriers. This finding also had crosscutting aspects associated with problem identification and resolution." (p. 10)	Cooper	Causal factors include on-the-job training less than adequate leading to individual knowledge less than adequate, awareness/attention—work distracted or interrupted, problem resolution—individual corrective action less than adequate, no procedure or reference document where needed.	Integrated inspection; NCV 10 CFR Part 50, Appendix B Criterion XVI (SCAQ), green finding
33	NRC Inspection Report	1999007	12/02/99	"Two non-cited violations were identified for operator errors involving poor procedure adherence that resulted in inadvertent water level decreases in the spent fuel pool and reactor coolant system. Operators responded promptly to the events and terminated the draindowns prior to any impact on reactor coolant or spent fuel cooling systems." (p. 2) "A total of 9,980 gallons was transferred from the SF pool to the BWST and the SF pool temperature rose one degree Fahrenheit...The licensee's investigation determined that the operator had a preconceived valve location in mind and had failed to reference the procedure or use proper self-checking tools." (p. 10)	"Failure to properly implement procedures was the primary cause of these two events, but contributing causes included deficiencies in communications, poor self-checking techniques, and an outage schedule change which moved up some draining activities. Licensee investigations were thorough and corrective actions were prompt and appropriate." (p. 2)	Crystal River	Causal factors include oversight — inadequate supervision, awareness/attention—worker distracted or interrupted, work practices—independent verification less than adequate.	NCV

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
34	NRC Inspection Report	2003010	03/05/04	" . . . repetitive damage to fuel assembly grid straps . . . " (p. 98) "The team noted that ten fuel assemblies were discovered to be damaged in September through December 2002. This was in addition to the seven fuel assemblies discovered to be damaged in March 2002...On February 24, 2003, during the final reload of the cycle 14 core, another new fuel assembly was damaged." (p. 99) "The fuel handlers had spent approximately two hours unsuccessfully trying to load another fuel assembly into place before deciding to change the loading sequence to load another assembly in a potential corner to corner interaction pattern. There was no indication that anyone suggested stopping the process and evaluating the condition, before agreeing to the change in the loading sequence. Over the next three hours, multiple problems were experienced as the licensee attempted to load the fuel assembly, including multiple overload conditions and cable oscillations. The licensee reset the overload setpoints to the least limiting condition at least twice, and even this setpoint was reached. Again, when problems were encountered, the decision was to keep on trying to insert the assembly, rather than stopping and evaluating what was happening."	"The barrier integrity cornerstone was affected as failure of the grid straps has led to fuel leaks. No other cornerstones were affected… the licensee had failed to take corrective actions which prevented recurrence of grid strap damage, a significant condition adverse to quality." (p. 100)	Davis Besse	Causal factors include problem evaluation—causal development less than adequate, problem resolution—individual corrective action less than adequate	Special Team Inspection—CAP implementation; Violation 10 CFR Part 50, Appendix B, Criterion XVI (SCAQ), green finding
35	NRC Inspection Report	2006003	07/27/06	10 CFR 50, Appendix B, Criterion VII "Control of Purchased Material, Equipment, and Services" violated for "failure to control adequately contractors during the Unit 1 refueling outage that resulted in damage to the fuel transfer system. This was a self-revealing violation when a pillar block weld broke resulting in damage to the transfer cart, rails, basket, and dummy fuel assembly."	"This finding is more than minor because it could be reasonably viewed as a precursor to a significant event involving damage to a fuel assembly."	Farley 1	Causal factors include oversight—inadequate supervision, problem identification and resolution incomplete	Integrated inspection; NCV, green finding
36	NRC Inspection Report	2005012	08/10/05	"During the Dry Run activities, the team observed that the licensee used the 15-Ton auxiliary hook to move the MPC lid into the cask welding pit through the Spent Fuel Room hatch. Although the licensee procedures did not prohibit this lift and the lift was consistent with the guidance of NUREG–0612, the team questioned the use of a hoist that is not single failure proof."	"This use of the auxiliary hoist did not meet licensee expectation for heavy load lifting around the spent fuel room area." Licensee later revised procedures "to prohibit use of the auxiliary hook for lifting loads greater than 3,000 lbs. over the Auxiliary Building roof."	Farley 1 & 2	Causal factors include work practices or craft skills less than adequate, procedures and reference documents—no procedure reference document where needed, work practices — independent verification less than adequate, procedures and reference documents less than adequate.	ISFSI Dry Run—Inspection/Observation; Observation

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
37	NRC Inspection Report	1999008	01/05/00	"The licensee lifted the Unit 2 lower internals with the polar crane from the lower refueling cavity to the reactor vessel using procedure FNP-2-MP-1.2, Reactor Vessel Lower Internals Removal and Installation, Revision 7. During the lift, the primary height measuring system malfunctioned, and the highly irradiated portion of the lower internals was exposed. When maintenance personnel recognized this error, the lower internals were placed in the reactor vessel. Corrective actions included procedure revisions, training, pre-job briefing enhancements, a reemphasis of communications and stop work authority, oversight enhancements, Radiation Work Permit (RWP) changes, and a review of related outage activities." (p. 4)	Violation of Tech Spec 6.8.1a	Farley 2	The root cause team concluded that personnel error, combined with an inadequate maintenance procedure and failure to follow administrative control procedure FNP-O-ACP-15.0, Pre-Job Briefing, Rev. 2, and poor communications and oversight by maintenance, health physics, and operations, resulted in the event. Causal factors include oversight— inadequate supervision, work practices—less than adequate team interactions, work planning—pre-job activities less than adequate.	Integrated inspection; NCV
38	NRC Inspection Report	2004008	02/04/05	"Licensee personnel failed to implement the procedural guidance for the proper installation of the refueling shield bridge (cattle chute) which caused a fuel bundle to contact the shield bridge while the bundle was being transported from the reactor core to the spent fuel pool." (Violation of TS 5.4.1.a)	"Impacted the Barrier Integrity cornerstone and if left uncorrected and a fuel bundle struck the refueling shield bridge again, it could lead to the failure of the fuel bundle cladding and the potential release of fission products."	Fermi 2	Causal factors include: work practices or craft skills less than adequate.	Integrated inspection; NCV, green finding
39	NRC Inspection Report	2003006	02/04/04	"A noncited violation was identified as a result of the failure of the spent fuel handling machine operator to follow the procedure for transferring fuel in the spent fuel pool as required by Technical Specification 5.8.1.a. This failure resulted in the dropping of a fuel assembly in the spent fuel pool." (p. 8) "The assembly dropped approximately 2 feet and came to rest against the spent fuel pool wall. The bottom of the assembly straddled four fuel storage cells and the top was leaning against the spent fuel pool wall." (p. 35) "Operator did not perform the procedure steps correctly, resulting in the long tool grapple not being latched in the desired position." (pp. 35–36)	"This finding was more than minor since it is associated with the fuel cladding human performance attribute of the cornerstone. The finding was characterized as having very low safety significance because there was no damage to fuel pins or breach of the spent fuel storage pool liner. This finding also had crosscutting aspects associated with human performance." (p. 8)	Fort Calhoun	Causal factors include work practices or craft skills less than adequate.	Integrated inspection; NCV, green finding
40	NRC Inspection Report	2005005	01/26/06	"Operators were inspecting three fuel bundles from previous cores that were suspected of having fuel rod leaks to identify and correct the causes of this problem." One of the bundles moved was the wrong one. "The evaluation identified [HP] issues regarding improper verification techniques and inadequate procedure implementation. Additional causal factors related to acceptance of lighting limitations in the area were also identified."	No radiological consequences; no loss of fuel integrity	Hope Creek	Causal factors include work practices or craft skills less than adequate	IP 71111.14 "Operator Performance During Non-Routine Evolutions and Events"

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
41	NRC Inspection Report	2002005	09/23/02	"Entergy did not appropriately evaluate and implement short-term actions associated with Condition Report (CR) IP2-2002-07253. The consequence of the finding was the relocation of spent fuel assembly G-28 without the appropriate handling tools and precautions. The finding is more than minor since it could be reasonably viewed as a precursor to a significant event (dropped spent fuel assembly in the spent fuel pool)." (p. 7) "Special handling tools were designed to address recent industry experience involving top nozzle separation on susceptible spent fuel assemblies. Spent fuel assembly G-28 is a susceptible fuel assembly associated with top nozzle separation... Condition report CR-IP2-2002-07253 documented that fuel assembly G28 was not relocated in the location recorded on the fuel move sheets on July 8, 2002. Compounding this record-keeping error, contract personnel did not use the special anchor and tooling device for movement of fuel assembly G28 when it was discovered in the wrong location on July 23, 2002." (p. 16)	"The Significance Determination Process is not modeled for a finding of this type. However, in accordance with NRC Manual Chapter 0612, this finding was reviewed by NRC risk analysts and management and has been determined to be of very low safety significance because no actual consequence existed and there was no unintended radiation worker exposure. The finding was determined to be a violation of 10 CFR 50, Appendix B, Criterion V, and is being treated as a non-cited violation." (p. 7)	Indian Point 2	Causal factors include work practices or craft skills less than adequate, work practices — logkeeping or log review less than adequate	Integrated inspection; NCV, green finding
42	NRC Inspection Report	1999022	12/29/99	"Due to a lack of attention-to-detail and a failure to adequately conduct self-checking and Independent verifications, four fuel positioning errors occurred." (p. 2) 1. "On September 24, 1999, during new fuel receipt activities, a new fuel assembly was oriented in the wrong direction . . . During a fuel pool audit, it was discovered that the assembly was actually oriented in the southwest direction . A prompt investigation identified the root cause as a human performance error due to a lack of attention-to-detail." (p. 8) 2. "On November 1, 1999, following fuel shipping operations in the Unit I spent fuel pool, in the process of returning a fuel bundle to its original spent fuel pool location (L-47), refueling personnel identified that another bundle was already in that location, but an adjacent cell (K-47) was unexpectedly vacant . . . " (p. 9)	"The orientation of the fuel assembly in the fuel pool had no reactivity significance. The purpose of specifying the orientation was to have the fuel assembly in the final core load orientation to minimize the potential for misorienting it in the core." (p. 8)	LaSalle 1	"Root cause of this event was multiple personnel errors which occurred during the fuel movement. In particular, three individuals fulfilling the fuel handler, second verifier, and supervisor roles failed to properly execute the verifications required to ensure that fuel movements occurred in the sequence prescribed by the NCTL and in accordance with LFP-1 00-6. Contributing factors that may have led to the event included use of the Unit 2 refueling bridge in the Unit I fuel pool, a thin crud layer on the Index system, and elevated temperatures on the refuel floor." (p. 9)	NCV

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
43	NRC Inspection Report	1999022	12/29/99	(3) “On November 12, 1999, a fuel assembly was found to be oriented in the wrong direction. Step 691 of the NCTL required the assembly to be oriented in the northwest position. During performance of NCTL Step 701, fuel handling personnel identified that the assembly had been mis-oriented in the southeast position. A prompt iunvestigation determined that due to a lack of attention-to-detail, the refueling bridge operator failed to position the fuel assembly In the proper orientation communicated to him. Also, the independent verifier and SRO in charge of fuel handling operations failed to properly execute a verification of the final fuel assembly position." (p. 9) (4) “On November 14, 1999, during a Unit I final core verification audit, the licensee identified that the fuel assembly located in position 15-22 was oriented in the wrong direction...A prompt investigation determined that the root cause was human performance error due to a lack of attention-to-detail. Specifically, the refueling bridge operator failed to position the bundle in the proper direction per the NCTL. Also, the independent verifier and SRO in charge of fuel handling operations failed to properly execute a verification of the final fuel assembly position.” (p. 9)			Causal factors include awareness/attention—worker distracted or interrupted, work practices—independent verification less than adequate	NCV
44	NRC Inspection Report	2003002	04/23/03	“The inspectors identified a finding of very low safety significance that is also a non-cited violation of Technical Specification 6.8.1, ‘Procedures, because maintenance technicians did not follow procedures while performing an inspection of new fuel bundles. On January 29, 2003, two new fuel bundles fell out of a shipping container as maintenance technicians were raising them to a vertical position...This event occurred as technicians were performing inspections of new fuel. The technicians did not follow two separate steps in the inspection procedure that require them to install restraining bars and a strap. These steps also require a second individual to verify (‘peer check’) satisfactory completion to prevent this event from occurring...” (p. 19)	“The bundles were damaged as they struck the refueling floor; however, there was no breach of the cladding and no contamination or other radiological consequences. Exelon shipped the bundles back to the fuel vendor.” (p. 19) “The inspectors identified that this finding involved a human performance error because technicians did not follow a maintenance procedure. Additionally, ineffective supervisory oversight, another human performance factor, contributed to this event.”	Limerick 1 & 2	“[Exelon's investigation] revealed that: • Technicians did not use or refer to the fuel inspection procedure • Technicians were not aware of a recent change to the procedure that required peer checks • Technicians had received a pre-job brief that covered the critical steps of installing the restraining bars and strap, and • Supervisors did not provide effective oversight of the evolution” (p. 19) Causal factors include work practices or craft skill less than adequate, awareness/attention—self-check less than adequate	Integrated inspection; NCV, green finding

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
45	NRC Inspection Report	2006001	10/30/06	"10 CFR 72.212 written evaluation had not been revised to assess having greater than 12 NAC-UMS casks in the ISFSI when 13 had been placed there." (p. 24)	"The failure to recognize that 10 CFR 72.212(b)(2)(ii), and licensee procedure NSD 211, required that 10 CFR 72.48(c) evaluations be performed for changes to 72.212(b)(2) written evaluations is important because the 72.48(c) evaluation determines whether prior NRC approval is needed before a change can be implemented to the facility or spent fuel storage cask design. This issue is greater than minor because the failure to perform 72.48(c) evaluations on any changes to 72.212 written evaluations had a reasonable likelihood that the changes could require NRC review and approval."	McGuire 1 & 2		Integrated inspection; NCV
46	NRC Inspection Report	2005005	01/26/06	Misload: "a fuel assembly, with a decay heat calculated to be approximately 1.437 kW, was inadvertently retrieved from the wrong location and inserted into the cask," exceeding ≤0.958kW criterion.	"This finding is of very low safety significance because the cask was open to the spent fuel pool, which was borated to approximately 2773 ppm, and the assembly was not unlatched in the cask." Large safety margins left.	McGuire 1 & 2	Causal factors include work practices or craft skills less than adequate	Integrated inspection; NCV, green finding; see also NRC Event Notification Report 42203
47	NRC Inspection Report	1999008	01/01/00	"Two minor errors associated with the review and performance of the "Procedure for Inspection of New Fuel" were identified and demonstrated a weakness relative to procedural use and attention to detail. The errors involved a failure to specifically identify a reference document used to perform the work and a discontinuity between the authorizations on the working and official copies of the procedure."		Monticello	Causal factors include work practices or craft skills less than adequate, awareness/attention—worker distracted or interrupted	Observation
48	NRC Inspection Report	2005005	01/30/06	"Failure to assess the increase in risk for work associated with spent fueling pool (SFP) cooling support systems during a defueled plant condition," violating licensee procedures and requirements of 10 CFR 50.65 (a)(4).	No radiological consequences. "Maintaining decay heat removal (DHR) capability is a key safety function during shutdown conditions, whether the fuel remains in the reactor vessel or is off-loaded to the [SFP]."	North Anna 1 & 2	Causal factors include problem identification and resolution—audit/self-assessment less than adequate	Integrated inspection; NCV (50.65), green finding

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
49	NRC Inspection Report	2000007	10/20/00	"Several examples of poor procedural adherence and inadequate supervision culminated in a personnel error during new fuel receipt and processing. The specific errors that led to dropped fuel assemblies were the failure to install restraining devices on a new fuel assembly container and a lack of supervisory presence to verify proper rigging of the container." (p. 5) "On August 26, 2000, while receiving and processing new fuel, two non-irradiated fuel assemblies fell from their metal container onto the refuel floor of the reactor building... inspectors identified that the licensee failed to follow procedure 205.1 'Receiving and Processing New Fuel,' when railers with new fuel for refueling outage 18R were not properly posted. Additionally, radiation protection personnel and licensee management demonstrated weak communications as demonstrated by the lack of timeliness in initiating a corrective action report (2000-1032) to document this failure to follow procedures."	"Because the procedural errors related specifically to the new fuel receipt inspection and processing procedure, greater potential existed to install fuel that did not meet the requirements of Procedure 205.1; 'Receiving and Processing . . . New Fuel.' " (p. 5)	Oyster Creek	Causal factors includ oral communication less than adequate, oversigh— inadequate supervision, work practices or craft skills less than adequate	Integrated inspection; NCV, green finding
50	NRC Inspection Report	2006008	07/21/06	"A cask liner containing highly radioactive incore detectors became buoyant and floated to the surface of the reactor cavity pool, then filled with water and sank back down to the bottom of the pool."	Worker dose rates spiked, but for only ~12 seconds; no serious doses incurred, workers properly evacuated the area	Palisades	Causal factors include procedures and reference documents less-than adequate, problem identification—less than adequate use of operating experience, work practices — failure to stop work/nonconservative decision-making	Special inspection; NCV, green finding; not sure how relevant this is for the GROA
51	NRC Inspection Report	2006004	07/28/06	Breach cladding of a fuel rod in core; repeat event (previous one in 1993, ineffective CA); poor workmanship and inadequate troubleshooting main causes.	Affects both initiating events and barrier integrity cornerstones; cladding is an important barrier	Palisades	Causal factors include: work practices or craft skills less than adequate, work practices — independent verification less than adequate, problem identification and resolution—individual corrective action less than adequate	Integrated inspection; not sure how relevant this is for the GROA; NCV, green finding
52	NRC Inspection Report	2005012	02/2/06	"While raising a dry fuel storage (DFS) cask from the spent fuel pool following loading of the cask, the emergency brake on the crane engaged. The engaged emergency brake stopped movement of the load resulting in suspension of the load partially out of the pool. During troubleshooting activities, the workers exceeded the bounds of the approved work package by manipulating the brake release. This finding represented a violation of the license by performing work contrary to requirements specified by NUREG-0612."	Not sure; personnel didn't follow procedures	Palisades	Causal factors include work practices or craft skills less than adequate	IP 71111.14 "Operator Performance During Non-Routine Evolutions and Events"; NCV, green finding

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
53	NRC Inspection Report	2006003	07/24/06	"A self-revealing noncited violation of Technical Specification 5.4.1.a was identified for the failure of operations personnel to follow procedures. Specifically, between April 7 and April 12, 2006, operations personnel did not follow Procedure 4OOP-9PCO6, 'Fuel Pool Clean Up and Transfer,' Revision 37, Appendix AU, resulting in Valve PCN-VI 19, 'Cleanup Header Return to the Fuel Canal,' being improperly aligned."	"This resulted in an inadvertent transfer of approximately 1200 gallons of spent fuel pool water to the transfer canal and a spill of contaminated water onto the 120 foot and 100 foot elevations of the fuel building."	Palo Verde 1	Causal factors include work practices—procedural adherence less than adequate, individual corrective action less than adequate, problem identification and resolution—use of operating experience less than adequate, work practices—recognition of adverse conditions less than adequate	Integrated inspection; looks like repeat event, see next entry below; NCV, green finding
54	NRC Inspection Report	2005003	08/02/05	"A self-revealing noncited violation of Technical Specification 5.4.1.a was identified for the failure to follow procedures which resulted in an inadvertent reduction of spent fuel pool water level. Specifically, approximately 1,800 gallons of water was unknowingly directed to the transfer canal when operations personnel failed to follow Procedure 40OP-9PC06, 'Fuel Pool Clean Up and Transfer.' The initial auxiliary operator opened a valve when the step required the valve to be closed and did not open another valve as required by the procedure. A second auxiliary operator performed an inadequate independent verification of the position of the valves." occurred 4/23/05. "This [first] auxiliary operator had the procedure 'in hand' but opened Valve PCN-V119 when the step required the valve to be closed."	"This issue involved human performance crosscutting aspects associated with procedure implementation and operator attention to detail."	Palo Verde 1, 2, & 3	Causal factors include work practices or craft skills less than adequate, work practices—independent verification less than adequate, work practices—failure to stop work/non-conservative decisionmaking	Integrated inspection; NCV, green finding
55	NRC Inspection Report	2005003	08/02/05	" . . . failing to ensure maintenance on safety-related fuel handling equipment was performed by qualified personnel." Inspectors found nonqualified contractors worked on fuel handling during outage.	Administrative controls issue. "The finding is determined to be greater than minor because if left uncorrected it could become a more significant safety concern in that improperly performed maintenance on fuel handling equipment could impact the safe movement of nuclear fuel and increase the probability of a fuel handling accident."			Integrated inspection; NCV, green finding
56	NRC Inspection Report	2004003	08/09/04	"A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion V, 'Instructions, Procedures, and Drawings,' was identified for the failure of the licensee to have written instructions for testing a remotely controlled submersible vehicle in the Unit 1 spent fuel pool. The vehicle became entrained in the common suction line for the spent fuel pool cooling system. At the time of the event, the unit was in refueling operations with 164 of the 241 spent fuel assemblies unloaded into the spent fuel pool."	"The finding is greater than minor because it affected the configuration control and human performance attributes of the initiating events cornerstone objective. This finding cannot be evaluated by the significance determination process." (p. 9)	Palo Verde 1	Causal factors include corrective actions not timely, work practices — failure to stop work/nonconservative decisionmaking	Integrated inspection; NCV, green finding

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
57	NRC Inspection Report	2004003	08/09/04	"A self-revealing noncited violation of Technical Specification 5.4.1.a was identified when personnel failed to follow a maintenance procedure preceding a 12- to 24-inch heavy load drop of a 7,000 pound steam generator snubber level plate inside the Unit 2 containment. The drop was due to a series of errors between the engineering contractor and rigging crews. The snubber plate was dropped in the vicinity of reactor coolant and shutdown cooling piping. (pp. 9–10)	"The finding was greater than minor because it affects the equipment performance and human performance attributes of the initiating events cornerstone objective to limit the likelihood of events that challenge safety functions during shutdown conditions." (p. 10)	Palo Verde	Causal factors include problem identification and resolution—problem not resolved in timely manner, work practices—failure to stop work/nonconservative decisionmaking	Integrated inspection; NCV, green finding
58	NRC Inspection Report	2004003	08/09/04	"A noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, 'Corrective Action,' was identified for the failure to identify the root cause of spent fuel pool inventory loss events and implement corrective actions to preclude recurrence. Specifically, the improper positioning of a fuel pool cleanup suction valve and inadequate level monitoring resulted in three losses of spent fuel pool inventory events. This finding involves problem identification and resolution crosscutting aspects associated with the failure to identify root causes and implement corrective actions. The issue also involved human performance crosscutting aspects associated with mispositioned valves and awareness of plant conditions by operations personnel." (p. 10)	"The finding is greater than minor because it affected the configuration control and human performance attributes of the initiating events cornerstone objective. This finding cannot be evaluated by the significance determination process." (p. 10)	Palo Verde		Integrated inspection; NCV, green finding
59	NRC Inspection Report	2005002	05/05/05	"On February 28, 2005 . . . .while removing a jet pump plug assembly from the reactor vessel, the plug broke loose from the handling pole and roped L-hook while being lifted over the refuel floor auxiliary platform. As a result, the plug dropped approximately 60 feet, primarily through water, and landed on top of several fuel bundles in the reactor core."	No damage to FA. "Failure to use an independent backup method to the handling pole and rope when attempting to lift the plug over the handrail...pre-job briefing did not identify the need for such additional FME controls."	Perry	Causal factors include work practices - team interactions less than adequate, work practices—nonconservative actions taken, oral communications less than adequate, work practices or craft skills less than adequate, work planning—prejob activities less than adequate, work planning - inadequate staffing	Integrated inspection; NCV (TS), green finding
60	NRC Inspection Report	2005003	07/29/05	" . . . overload of an electrical bus during the Unit 1 refueling outage and the loss of one half of the fuel pool cooling system… ineffective corrective actions resulted in the procedures used to monitor loading on cross connected electrical buses being inadequate."	Violation of 10 CFR Part 50, Appendix B, Criterion XVI. "The fuel pool cooling loss did not result in a significant increase in temperatures."	Quad Cities 1 & 2	Causal factors include work practices—housekeeping less than adequate, problem resolution—individual corrective action less than adequate	Integrated inspection; NCV, green finding

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
61	NRC Inspection Report	2000001	04/17/00	"The inspectors identified that refueling personnel installed an operator aid to assist in locating the correct bridge coordinates above the spent fuel pool which did not meet the requirements of Procedure OSP-0001, 'Control of Operator Aids.' This issue was treated as an additional example of a violation of Technical Specification 5.4.1.a which was described in NRC Inspection Report 50-458/99-13." (p. 6) "On March 22, 2000, during refueling activities, the inspectors observed refueling personnel position the refueling bridge above the spent fuel pool by aligning a laser light with coordinates that had been handwritten on duct tape on the wall of the spent fuel building. The inspectors questioned refueling personnel to determine if the operator aid (laser light, duct tape, and handwritten coordinates) had been installed in accordance with an approved procedure or maintenance document. In response, the licensee stated that the laser light, duct tape, and handwritten coordinates were installed before refueling commenced, that it would be removed following refueling activities, and that it had not been installed in accordance with an approved procedure or maintenance document." (p. 10)	TS violation, failure to follow procedure	River Bend	Causal factors include human-system interface and environment - physical work environment/conditions less than adequate	NCV
62	NRC Inspection Report	2002004	01/29/03	"Inadequate door seal evaluation during maintenance activities resulted in both trains of Unit 2 control room emergency air cleanup system (CREACS) inoperable for a time longer than 24 hours." (p. 6) "The licensee erroneously concluded that door RA-114 did not affect the control room pressure boundary and therefore, no TS LCO was entered... The licensee determined that this event was caused by personnel errors because the degraded seal conditions of the doors RA-108 and RA-114 were not adequately evaluated during maintenance activities." (p. 20)	"A self-revealing non-cited violation of Technical Specification 3.7.7 Action b was identified. This finding is greater than minor because it affected the barrier integrity cornerstone objective of providing reasonable assurance that physical design barriers provide protection from radionuclide releases caused by accidents or events. The finding is of very low safety significance because CREACS was able to maintain a positive pressure during the affected period and the control room envelope remained operable with respect to its design bases function of maintaining operator dose within general design criterion (GDC) 19." (p. 6)	Saint Lucie	Causal factors include fitness-for-duty - fatigue/work hour control less than adequate, oversight - inadequate supervision	Integrated inspection; NCV, green finding

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
63	NRC Inspection Report	2005004	10/20/05	" . . . failure to follow the instructions in a maintenance order for the movement of equipment in the Unit 2 spent fuel pool. A four finger control element assembly was dropped in the cask area of the spent fuel pool because it had not been properly grappled . . . violation of 10 CFR Part 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings."	" . . . determined to be greater than minor because if left uncorrected it could become a more significant safety concern in that failing to follow instructions could impact the safe movement of components in the [SFP], and increase the probability of a fuel handling accident."	San Onofre 2	Causal factors include on-the-job training less than adequate leading to individual knowledge less than adequate, work practices or craft skills less than adequate, work planning—work package quality less than adequate	Integrated inspection; NCV, green finding
64	NRC Inspection Report	2001006	06/19/01	"Technical Specification 6.7.1.a requires that written procedures shall be implemented covering the activities described in Appendix A of Regulatory Guide 1.33, Rev. 2. Regulatory Guide 1.33 requires procedures for refueling and core alternations. Contrary to the above, on November 28, 2000, the refueling senior reactor operator failed to properly verify the component type (thimble plug) in fuel assembly J53 and therefore failed to implement Procedure RS0721, Refueling Administrative Control, Rev. 2, Chg. 4. This failure resulted in a thimble plug not installed in the reactor core as required during the last refueling outage." (p. 15)	No radiological consequences	Seabrook	Causal factors include work practices—independent verification less than adequate	Licensee-identified violation
65	NRC Inspection Report	2005003	07/28/05	"The inspectors identified a non-cited violation of Technical Specification 6.8.1 for a self-revealing failure to follow plant procedures prior to and during draining of the fuel transfer canal. Leakage past the spent fuel pit gate seal resulted in inadvertently transferring approximately 10,000 gallons of spent fuel pit inventory to the refueling water storage tank."	" . . . more than minor because it affected the Barrier Integrity cornerstone, in that operators failed to adhere to procedures while changing plant configurations resulting in a loss of spent fuel pit inventory."	Sequoyah 1 & 2	Causal factors include procedure or reference documents less than adequate	Integrated inspection; not sure how relevant this is for the GROA; NCV, green finding
66	NRC Inspection Report	2005005	02/11/06	" . . . violation of Technical Specification 6.8.1.a and Regulatory Guide 1.33, Appendix A, was identified for failure to adhere to Plant Operating Procedure 0POP08-FH-0003, 'Fuel Transfer System,' Revision 26. The failure to follow procedure resulted in fuel movers challenging the interlocks in the fuel transfer system. Specifically, a fuel mover attempted to lower a fuel assembly in the upender while the upender was still rising. The interlock prevented the upender from making contact with the fuel assembly."	" . . . greater than minor, because it involved the potential damage to fuel assemblies."	South Texas 1 & 2	Causal factors include work practices or craft skills less than adequate, awareness/attention - worker distracted or interrupted	Integrated inspection; NCV, green finding

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
67	NRC Inspection Report	2001004	01/28/02	"Technical Specification 6.4.A.8 requires detailed written procedures be provided for Refueling Operations. Technical Specification 6.4.D requires that procedures described in Specification 6.4.A shall be followed. On November 11, 2001, the licensee failed to follow Procedure 0-OP-4.8, in that the transfer of a spent fuel assembly was initiated prior to clearing the top of its storage position. This issue has been documented in the licensee's corrective action program as Plant Issue S-2001-3275." (p. 21)	No radiological consequences	Surry 1	Causal factors include work practices or craft skills less than adequate	Integrated inspection; NCV
68	NRC Inspection Report	2005009	08/14/06	Training issues uncovered during Dry Fuel Storage (DFS) campaign. "The inspector noted that the licensee had self-identified several training-related issues leading up to the fuel campaign...The identified training issues included such items as suggestions to enhance classroom training sessions, the need to ensure that the appropriate level of training is provided based on an individual's experience and knowledge level, and ensuring that training materials were maintained current." (pp. 11-13)		Susquehanna	Causal factors include on-the-job training less than adequate.	ISFSI inspection; Observation
69	NRC Inspection Report	2000005	01/29/01	"TS 6.8.1.a requires that written procedures shall be established, implemented, and maintained covering the activities referenced in Appendix A of Regulatory Guide 1.33. Refueling and Core Alterations are included in that Appendix. Two examples were identified where a fuel assembly was placed in the wrong spent fuel pool location. On October 2, 2000 and again on October 3, 2000, during de-fueling of the core, personnel incorrectly verified the Spent Fuel Pool location of a fuel assembly and placed the fuel assembly in the wrong location. Both examples were contrary to procedural requirements in 4-OP-040.2, Refueling Core Shuffle." (p. 21)	No radiological consequences. "One of the assemblies did not meet the TS 3.9-1 burnup requirements for storage in the location in which it was initially placed." (p. 21)	Turkey Point 2	Causal factors include awareness/attention—self-check less than adequate, work practices—independent verification less than adequate	NCV
70	NRC inspection Report	1999009	01/18/00	"On November 5, the refueling senior reactor operator (SRO) identified that a spent fuel assembly had been improperly transferred from the core into spent fuel pool (SFP) location H41 rather than the intended location at J41. This error was identified at Step 75 of the Core Shuffle 1 procedure when the operators attempted to transfer the assembly from SFP location J41 (which was empty) back into the reactor." (p. 13)		Vermont Yankee	Causal factors include awareness/attention—worker distracted/interrupted	Integrated inspection

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	Source	Reference No.	Report Date	Description	Consequences	Plant or Site	Root Causes	Notes
71	Hanford Information Bulletin	2006-RL-HNF-0014	05/08/06	"An evaluation of HEPA filter aerosol testing at (Fluor Hanford) FH-managed facilities revealed multiple weaknesses in the implementation and control of Contractor provided (CH2M-Hill) In-Place Leak Test procedures. To accommodate a wide variety of system configurations, the generic Vent and Balance (V&B) test procedures are intentionally vague and lack facility and system specifics. As the test procedure clearly states, it is the facility's responsibility to provide supplemental information to support the test...Assigned facility engineers must assume full responsibility for assuring the adequacy of in-place leak tests performed on their systems, regardless of the procedure source."		Hanford	"Apparently, the availability of 'approved' generic aerosol test procedures and the willingness of V&B staff to accept less than adequate or informal test supplements lead some to conclude a hands-off approach to testing is acceptable: it is not."	"Blue" - for information. Title: Control of Externally Prepared Technical Procedures
72	Hanford Information Bulletin	2006-RL-HNF-0035	08/24/06	"In July 2005 and February 2006 Limiting Conditions for Operations (LCO) at a nuclear facility were violated resulting in Technical Safety Requirement (TSR) violations... In one event (documented in Occurrence Report RL-PHMC-PFP-2005-0019), violating a LCO resulted in the movement of fissile material during a declared fissile outage. The LCO required all fissile material movement to be terminated when the Criticality Alarm Horns were declared inoperable. A combination of communication errors and inadequate resources led to the LCO not being adequately controlled and complied with during work activities. During another event (documented in Occurrence Report RL-PHMC-PFP-2006-0006, specific ventilation systems were required to be secured to prevent a release of unfiltered air to the environment. Again, a series of communication errors and inadequate resources led to violating the LCO."	Maintaining nuclear facilities in a safe configuration that is compliant with the facility safety basis is a significant responsibility assigned to facility management personnel.	A nuclear facility—site not specified	"During both of these examples, as well as other TSR violations similar in scope, the individuals responsible to establish the proper conditions for success were engaged in multiple tasks and distracted by other events."	"Blue" - for information. Title: Distractions and Increased Workloads Decrease Effectiveness of Supervisory Control
73	Hanford Information Bulletin	2006-RL-HNF-0011	04/06/06	"This report summarizes seven events related to safety analyses...In cases 1, 4, 5, and 7 unknown hazards or failure modes were not analyzed. In cases 2, 3, and 7, conditions changed over time or were different from those analyzed but had not been recognized as being outside the documented safety analysis (DSA)."	"Operating within an approved safety basis is a primary tenet of safe facility operation. Inaccuracies in that key document detract from our ability to safely operate a nuclear facility, potentially putting our workers and the public at risk."	A nuclear facility—site not specified		"Blue" - for information. Title: Safety Basis Inadequacies Detract from Nuclear Facility Safety

Table 8. Operating Experience Related to Administrative Controls (Continued)								
ID No.	**Source**	**Reference No.**	**Report Date**	**Description**	**Consequences**	**Plant or Site**	**Root Causes**	**Notes**
74	DOE ES & H Bulletin ‖	2005-09	07/25/05	"This Bulletin provides information about several serious events caused by a lack of vigilance and attentiveness on the part of those involved in conducting first-time or infrequently performed high-hazard activities." Bulletin talks about BP refinery accident and four instances of DOE events 1999–2005 that had similar causal factors (although the events involved operations not expected at the GROA).	Failure to identify the hazards, develop appropriate actions, and remain alert to the possible dangers involved in such activities could lead to potentially catastrophic outcomes at DOE sites.	Various in DOE Complex	"Causal factors for these events reveal similar inadequacies in work performed. Procedures • omitting steps • using an incorrect or unapproved procedure • allowing operators' process knowledge to override procedural compliance Hazards analysis • not recognizing the potential for multiple failure modes • failing to comply with existing safety requirements • ineffective emergency management planning Operational oversight • less than adequate command and control during an unfamiliar operation and during upset conditions • insufficient communication of process activities"	Title: Vigilance in New or Infrequent High-Hazard Operations
75	DOE Lessons Learned¶	Y-2000-OR-BJCETTP-0102	01/31/00	While reviewing work planning and scheduling for annual trip testing of dry pipe sprinkler systems in Building K-25, it was discovered that 30 systems had not been tested within their scheduled date for annual testing.	"Facility Managers need to provide direction on information needed for maintaining the status of safety systems and safety significant systems. Communication of reporting and information needs to service providers and facility occupants/users is essential to the maintenance of the status of safety systems and safety significant systems."	Not specified	Not enough information to identify causal factors. Likely candidates: testing program and practices less than adequate, communications less than adequate.	"Yellow." Title: Safety System Status and Management

*NRC. "Information Notices." Washington, DC: NRC. <http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/>
†NRC. "Licensee Event Reports Database." Washington, DC: U.S. Nuclear Regulatory Commission. <https://nrcoe.inel.gov/lersearch>
‡NRC Inspection Report - NEED REFERENCE INFORMATION (ID No. 22)
§Hanford Information Bulletin - WHAT REFERENCE? (ID No. 71)
‖DOE ES&H Bulletin - WHAT REFERENCE? (ID No. 74)
¶DOE. "DOE Office of Health, Safety, and Security Lessons Learned Database." Washington, DC: DOE. <https://www.hss.energy.gov/csa/analysis/ll/oellproducts.html>