The NRC and Nuclear Power Plant Safety in 2012

TOLERATING THE INTOLERABLE

Union of Concerned Scientists
Citizens and Scientists for Environmental Solutions

The NRC and Nuclear Power Plant Safety in 2012: Tolerating the Intolerable

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Union of Concerned Scientists

Citizens and Scientists for Environmental Solutions

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EXECUTIVE SUMMARY

Safety nets are often deployed to protect tightrope walkers and trapeze artists during performances. If a performer falls and slams to the ground through a ripped portion of the net, more needs to be done than mending broken bones and ruptured organs—the rip in the safety net needs to be fixed. It is simply unacceptable to tolerate a rip just because performers seldom need a safety net.

The U.S. Nuclear Regulatory Commission (NRC) is tolerating the intolerable: a ripped nuclear safety net. Granted, nuclear reactors do not fall into the net every day. And so far the United States has been lucky—with limited and notable exceptions, reactors that have fallen have avoided the ripped portion of the safety net. The more often the net is used and the more the net itself is abused, however, the more likely it becomes that someday workers or the public will be harmed by a nuclear reactor accident.

In 2012, the NRC reported 14 "near-misses" at nuclear plants. Just to be clear about the gravity of the situation, a "near-miss" is an event that increases the chance of core meltdown by at least a factor of 10, thus prompting the NRC to dispatch some level of special inspection team to investigate the event. Over the past three years, 40 of the nation's 104 nuclear reactors experienced one or more near-misses. That is a rate greater than one near-miss per month. The NRC must take two steps to reduce the frequency of near-misses before some reactor falls through the ripped section of the net.

First, the NRC already investigates each near-miss to determine what happened and why. The NRC should formally evaluate all safety violations identified during its near-miss inspections to determine whether the agency's baseline inspections could have, and should have, found these safety problems sooner. Such insights from the near-misses may enable the NRC to make adjustments in what its inspectors examine, how they examine it, and how often they examine it, so no violation can go undetected.

Second, the NRC must require that individual plant owners find and fix problems in their testing and inspection procedures. Many of the near-misses last year involved design and operational problems that had already existed for years— sometimes even decades—prior to the incidents in question. The plants' tests and inspections are supposed to find and fix such problems, yet failed to do so. Plant owners must be formally required to evaluate why their testing and inspection failed to find and fix longstanding problems.

Within the NRC itself, rips in the safety net must also be fixed. Regulations *are* the safety net. The simplest repair available is for the NRC to enforce existing regulations, using its ability to impose fines on owners and shut down reactors that violate safety regulations.

Unfortunately, the NRC has repeatedly failed to enforce essential safety regulations. Last year, for example, the NRC approved an additional delay in compliance with fire protection regulations at the Browns Ferry Nuclear Power Plant near Decatur, Alabama. The NRC adopted the fire protection regulations in 1980 after a disastrous 1975 fire at—of all places—the Browns Ferry nuclear plant. If the latest schedule is met, Browns Ferry will have operated for fully 35 years out of compliance with fire protection regulations that its own fire inspired. Another key NRC safety regulation prohibits a reactor from operating longer than six hours if it suffers a leak of cooling water. In 2012, however, the NRC did nothing when the Palisades Nuclear Power Plant in Michigan operated for nearly a month despite cooling water leaks.

These examples of tolerating the intolerable should be case studies for regulatory ineptitude. Failing to enforce existing safety regulations is literally a gamble that places lives at stake. The NRC must enforce its own regulations,

Tolerating the intolerable reflects a poor safety culture. Last November, the NRC met to discuss the results of the latest in a series of triennial surveys conducted by a consultant of its safety culture and climate. The NRC's discussion of the 2012 survey was held behind closed doors—about as plain an indicator of a poor safety culture as the sordid results themselves. A poor safety culture and unwillingness to discuss working conditions openly go hand in hand.

Among other disconnects, the 2012 survey revealed that half of the NRC's work force had heard about co-workers who received negative reactions from supervisors and senior managers after raising a concern. Only 41 percent of the work force felt that the NRC had taken significant steps to address key issues identified in past surveys of the agency's safety culture. Yet, the survey revealed that the NRC's senior managers believe conditions are far better than the rest of the agency believes.

The 2012 survey suggests the underlying reason for the shortcomings in the NRC's safety culture: There is a large perception gap between how NRC senior managers view conditions within the agency and how the work force views them. NRC managers cannot fix problems they do not believe to exist.

The U.S. Congress was instrumental in guiding the NRC into doing more about safety culture problems at nuclear plants a decade ago. Now once again, Congress must compel the NRC to take the same medicine for the same affliction.

The good news is that the NRC already knows how to fix such shortcomings and regain the proper safety focus—it has been working to do so at individual nuclear power plants over the past decade. Now the NRC needs to internalize those lessons and practices to heal itself.

It should not take a disaster at a U.S. nuclear power plant to undertake the necessary reforms at the NRC.

Step one: stop tolerating the intolerable.

CHAPTER 1. THE COP ON THE NUCLEAR BEAT

The Nuclear Regulatory Commission (NRC) is to owners of nuclear reactors what local law enforcement is to a community. Both are tasked with enforcing safety regulations to protect people from harm. A local police force would let a community down if it investigated only murder cases while tolerating burglaries, traffic violations, and vandalism. The NRC must similarly be the cop on the nuclear beat, actively monitoring reactors to ensure that they are operating within regulations, and aggressively engaging owners and workers over even minor violations.

The Union of Concerned Scientists (UCS) has evaluated safety at nuclear power plants for over 40 years. We have repeatedly found that NRC enforcement of safety regulations is not timely, consistent, or effective. Our findings match those of the agency's internal assessments, as well as of independent agents such as the NRC's Office of the Inspector General and the federal Government Accountability Office. Seldom does an internal or external evaluation conclude that a reactor incident or unsafe condition stemmed from a lack of regulations. Like UCS, these evaluators consistently find that NRC enforcement of existing regulations is inadequate.

This report—like its predecessors—chronicles what the agency is doing right as well as what it is doing wrong. Our goal is to help the NRC achieve more of the former and avoid more of the latter.

THE REACTOR OVERSIGHT PROCESS AND NEAR-MISSES

The NRC monitors the health and safety of nuclear plants using its Reactor Oversight Process (ROP). In this process, the NRC's full-time inspectors assess operations and procedures attempting to detect problems before they lead to more serious and events. The ROP features seven cornerstones of reactor safety (Table 1). Using this process, the NRC issued nearly 200 reports on its findings last year alone.

When a safety-related event occurs at a reactor or a degraded condition is discovered, the NRC evaluates the chance of damage to the reactor core. Most incidents discovered at nuclear power plants have low risk. If the event or condition did not affected that risk—or if the risk was increased only by a very small amount—the NRC relies on measures in the ROP to respond.

When an event or condition increases the chance of reactor core damage by a factor of 10, however, the NRC is likely to send out a special inspection team (SIT). When the risk rises by a factor of 100, the agency may dispatch an augmented inspection team (AIT). And when the risk increases by a factor of 1,000 or more, the NRC may send an incident inspection team (IIT) (NRC 2010c).

When an event or discovery at a reactor results in the NRC sending out an inspection team, UCS refers to it a "near-miss."

The teams go to the sites to investigate what happened, why it happened, and whether the incident poses any safety implications for other nuclear plants. The teams take many weeks to conduct an investigation, evaluate the information they gather, and document their findings in a publicly available report.

Both routine ROP inspections and investigations of the special teams identify violations of NRC regulations. The NRC classifies violations in five categories, with Red denoting the most serious, followed by Yellow, White, Green, and Non-Cited Violations.[1] For certain violations that do not lend themselves to classification by their risk significance, the NRC uses four severity levels, with level I being the most serious.

The classifications dictate the extent of the responses the NRC expects from plant owners as well as the extent the NRC's follow-up to the violations. For example, for a Green finding, a plant owner would be expected to fix the non-conforming condition and NRC inspectors might verify proper resolution during their next planned examination of that area, whether that opportunity was scheduled next month or next year. For a Yellow or Red finding, however, the plant owner would be expected to supplement the actions taken for a Green finding with measures designed to determine whether the problem was an isolated case or reflective of a broader, programmatic breakdown. Moreover, the NRC's follow-up inspections are typically more timely for Yellow and Red findings than for Green and White findings.

THE SCOPE OF THIS REPORT

Chapter 2 summarizes all the "near-misses" at nuclear reactors that the NRC reported in 2012, although some actually occurred in 2011. Near-misses are events that prompted the agency to dispatch an SIT, AIT or IIT. In these events, a combination of broken or impaired safety equipment and poor worker training typically led owners of nuclear plants down a pathway toward potentially catastrophic outcomes. After providing an overview of each event, this chapter shows how one problem led to another in more detail for that event, and notes any "tickets" the NRC wrote for safety violations that contributed to the near-miss.

This detailed review of all the near-misses in 2012 provides important insights into trends in nuclear safety, as well as into the effectiveness of the NRC's oversight process. For example, if many near-misses stemmed from

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[1] The NRC also uses a "Greater than Green" classification for security violations that are White, Yellow, and Red to convey to the public some distinction about security problems without also pointing potential saboteurs to plants having serious security vulnerabilities.

failed equipment, such as emergency diesel generators, the NRC could focus its efforts in that arena until it arrests declining performance. Chapter 2 therefore uses the year's safety-related events to suggest how the NRC can prevent plant owners from accumulating problems that will conspire to cause next year's near-misses—or worse.

With the near-misses attesting to why day-to-day enforcement of regulations is vital to the safety of nuclear power, the subsequent three chapters then highlight NRC performance in monitoring safety through the reactor oversight process. Chapter 3 evaluates trends from the near-misses since 2010 when the UCS initiated this series of reports covering NRC performance. Chapter 4 describes occasions in which effective oversight by NRC inspectors took action to prevent safety problems from snowballing into near-misses or even more dangerous situations. Chapter 5 then describes cases where ineffective NRC oversight failed to prevent dangerous situations—or actually set the stage for them.[2]

Chapter 6 summarizes findings from the near-misses in Chapter 2, the trend analysis of Chapter 3, the examples of positive outcomes from the reactor oversight process in Chapter 4, and the examples of negative outcomes from that process in Chapter 5. This concluding chapter notes which oversight and enforcement strategies worked well for the NRC in 2012 and which did not. Furthermore, Chapter 6 recommends steps the NRC should take to reinforce behavior among plant owners leading to commendable outcomes, and steps the NRC should take to alter behavior that produces outcomes that pose risks to employees and the public.

UCS's primary aim in creating the annual reports is to spur the NRC to improve its own performance as well as that of reactor owners.

[2] These examples represent similar situations at other plants. Future reports may include a different number of examples.

Table 1. Seven Cornerstones of the NRC's Reactor Oversight Process	
Initiating events	Conditions that, if not properly controlled, require the plant's emergency equipment to maintain safety. Problems in this cornerstone include improper control over combustible materials or welding activities, causing an elevated risk of fire; degradation of piping, raising the risk that it will rupture; and improper sizing of fuses, raising the risk that the plant will lose electrical power.
Mitigating systems	Emergency equipment designed to limit the impact of initiating events. Problems in this cornerstone include ineffective maintenance of an emergency diesel generator, degrading the ability to provide emergency power to respond to a loss of offsite power; inadequate repair of a problem with a pump in the emergency reactor-core cooling system, reducing the reliability of cooling during an accident; and non-conservative calibration of an automatic temperature set point for an emergency ventilation system, delaying startup longer than safety studies assume.
Barrier integrity	Multiple forms of containment preventing the release of radioactive material into the environment. Problems in this cornerstone include foreign material in the reactor vessel, which can damage fuel assemblies; corrosion of the reactor vessel head from boric acid; and malfunction of valves in piping that passes through containment walls.
Emergency preparedness	Measures intended to protect the public if a reactor releases significant amounts of radioactive material. Problems in this cornerstone include emergency sirens within 10 miles of the plant that fail to work; and underestimation of the severity of plant conditions during a simulated or actual accident, delaying protective measures.
Public radiation safety	Design features and administrative controls that limit public exposure to radiation. Problems in this cornerstone include improper calibration of a radiation detector that monitors a pathway for the release of potentially contaminated air or water to the environment.
Occupational radiation safety	Design features and administrative controls that limit the exposure of plant workers to radiation. Problems in this cornerstone include failure to survey an area properly for sources of radiation, causing workers to receive unplanned exposures; and incomplete accounting of individuals' radiation exposure.
Security	Protection against sabotage that aims to release radioactive material into the environment, which can include gates, guards, and guns. After 9/11, the NRC removed discussion of this cornerstone from the public arena.

CHAPTER 2. NEAR-MISSES AT NUCLEAR POWER PLANTS IN 2012

In 2012, the NRC reported on 14 significant events that prompted the agency to send teams to analyze problems at those reactors (Table 2).[3] Eleven of these events triggered investigations by special inspection teams (SIT) in respond to a 10-fold increase in risk of core damage, and three triggered an augmented inspection team (AIT) inspection (in response to a 100-fold increase in risk of core damage). The events resulting in AIT inspections are colored red in Table 2. No events triggered an incident inspection team (IIT) inspection (in response to a 1,000-fold or greater increase in risk of core damage).

These events are near-misses because they raised the risk of damage to the reactor core—and thus to the safety of workers and the public. As the end of this chapter will show, lessons from these near-misses reveal how the NRC can apply its limited resources to reap the greatest returns for public safety.

Table 2: Nuclear Near-Misses in 2012

Reactor and Location	Owner	Highlights
Brunswick Steam Electric Plant, Unit 2 Southport, NC	Progress Energy	SIT: Excessive leakage of cooling water from the reactor vessel, determined to have been caused by the improper installation of the reactor vessel's head, led to an emergency being declared and the reactor being shut down.

[3] Numbering becomes cumbersome because nuclear plants can have multiple reactors, safety- and security-related events can affect one or all reactors at a plant, and some reactors experienced multiple events. Table 2 here and Table 4 later in the report attempt to clarify who had what near-miss.

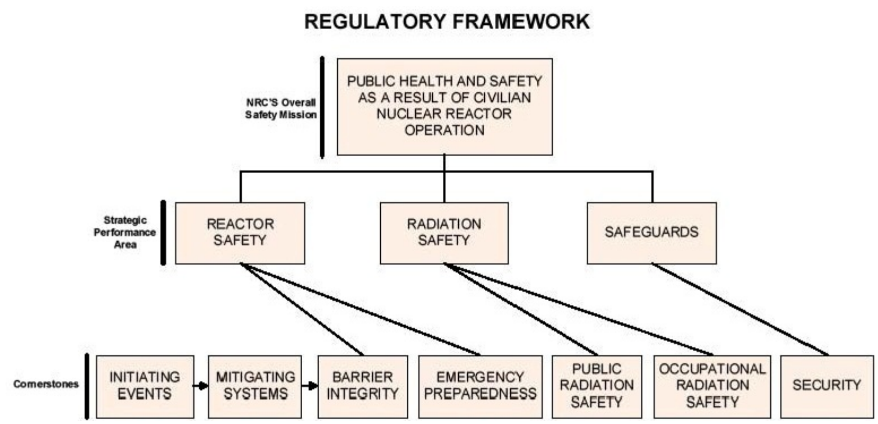
Byron Station, Unit 2 Byron, IL	Exelon Generation Co., LLC	SIT: Equipment failure in the switchyard triggered an automatic shut-down of the reactor. A design deficiency caused emergency equipment to be de-energized until workers took steps to isolate the problem and restore power from the emergency diesel generators.
Catawba Nuclear Station, Unit 1 York, SC	Duke Energy Corp.	SIT: After an age-related problem caused one of four reactor coolant pumps to fail, the Unit 1 reactor and turbine automatically shut down as designed. Due to a design error in a recent modification, the decreasing voltage output by the main generator caused electrical breakers to open that disconnected Units 1 and 2 from the offsite power grid. One of the emergency diesel generators started but failed to supply electricity to safety equipment due to another design error when it was installed in 1984.
Farley Nuclear Plant, Units 1 and 2 Dothan, AL	Southern Nuclear Operating Company, Inc.	SIT: Security problems prompted the NRC to conduct a special inspection. Details of the problems, their causes, and their fixes are not publicly available.
Fort Calhoun Station, Omaha, NE	Omaha Public Power District	SIT: The NRC investigated a fire that disabled half of the 4,160 volt and two-thirds of the 480 volt power supplies for emergency equipment at the plant and triggered the declaration of an Alert—the third most serious of the NRC's four emergency classifications.
Fort Calhoun Station, Omaha, NE	Omaha Public Power District	SIT: Security problems prompted the NRC to conduct a special inspection. Details of the problems, their causes, and their fixes are not publicly available.

Harris Nuclear Power Plant, Raleigh, NC	Progress Energy	SIT: As the reactor was being shut down for a scheduled refueling outage, workers tested the closing time of the three main steam isolation valves. These valves are designed to close within five seconds during an accident to limit the amount of radioactivity released to the atmosphere. The NRC dispatched an SIT after it took one valve 37 minutes to close and another 4 hours and 7 minutes.
Palisades Nuclear Plant, South Haven , MI	Entergy Nuclear Operations, Inc.	SIT: Workers shut down the reactor about a month after they detected a small cooling water leak. The NRC sent an SIT to the site after the source of the leak was determined to be a location where any leakage required the plant to be shut down within six hours.
Palo Verde Nuclear Generating Station, Units 1, 2, and 3 Wintersburg , AZ	Arizona Public Service Company	SIT: Security problems prompted the NRC to conduct a special inspection. Details of the problems, their causes, and their fixes are not publicly available.
Perry Nuclear Power Plant, Perry, OH	FirstEnergy Nuclear Operating Company	SIT: Security problems involving failures to prevent unauthorized individuals from entering secure areas of the plant prompted the NRC to conduct a special inspection.

River Bend Station, St. Francisville, LA	Entergy Operations, Inc.	AIT: The operators manually shut down the reactor on May 24 after an electrical fault on the motor of a feedwater pump caused it to stop running. A failed relay prevented the electrical breaker for the motor from opening to isolate the electrical fault. The fault propagated through the electrical distribution system, causing the breaker supplying power to the 13,800 volt electrical bus to open. Due to another electrical cable problem on May 21, all of the plant's circulating water pumps and non-emergency cooling water pumps were being powered from this single electrical bus. Its loss caused the plant's normal heat sink to be lost and stopped the supply of cooling water to equipment in the turbine building and to some emergency equipment.
San Onofre Nuclear Generating Station, Units 2 and 3 San Clemente, CA	Southern California Edison Company	AIT: Operators shut down the Unit 3 reactor following a leak inside a steam generator replaced less than a year earlier. The NRC dispatched an AIT after eight steam generators tubes failed pressure testing and inspections identified extensive and unusual degradation in the steam generators of both units.
Wolf Creek Generating Station, Burlington, KS	Wolf Creek Nuclear Operating Corporation	SIT: Erratic performance of an emergency diesel generator during a routine test prompted the NRC's special inspection. The SIT determined that an improper fix to another problem four months earlier impaired the emergency diesel generator's control system.
Wolf Creek Generating Station, Burlington, KS	Wolf Creek Nuclear Operating Corporation	AIT: After one electrical fault in the switchyard caused the main generator to shut down automatically, a second electrical fault disconnected the plant from its offsite electrical grid.

SITs and AIT dispatched by the NRC identified 18 violations of NRC safety regulations. Figure 1 classifies these violations by the seven cornerstones of the reactor oversight process (ROP).[4]

Figure 1: Near-Misses in 2012 by Cornerstones of the Reactor Oversight Process



Red	**3**	**0**	**0**	**0**	**0**	**0**	**0**
Yellow	**0**	**1**	**0**	**0**	**0**	**0**	**0**
White	**0**	**0**	**0**	**0**	**0**	**0**	**0**
>Green*	**n/a**	**n/a**	**n/a**	**n/a**	**n/a**	**n/a**	**4**
Green	**4**	**6**	**0**	**0**	**0**	**0**	**0**
IV	**0**	**0**	**0**	**0**	**0**	**0**	**0**
Total	**7**	**7**	**0**	**0**	**0**	**0**	**4**

*After 9/11, the NRC stopped publicly releasing the color assigned to security violations other than that some have been classified as White, Yellow, or Red—greater than Green.

Source: Top half of figure: NRC; bottom half: UCS.

The NRC special investigative teams classified three safety violations at Fort Calhoun as Red—the most serious—in 2012. The teams classified one safety violation at Catawba as Yellow, the second most serious classification.

Each near-miss reported by the NRC in 2012 is described below in alphabetical order by plant name (matching the order in Table 2).

Brunswick Steam Electric Plant, Unit 2, NC

The Near-Miss

The NRC sent an SIT to the site after an attempted restart of the Unit 2 reactor had to be aborted because reactor cooling water was leaking into the containment building. Workers found that the top of the reactor vessel had not been installed properly (NRC 2012v).

[4] For more information on the cornerstones and related NRC inspections, see Table 1 and *http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/cornerstone.html.*

How the Event Unfolded

On November 15, 2011, operators began restarting the Unit 2 reactor at Brunswick. The reactor had been shut down 11 days earlier to find and replace a damaged fuel assembly in the reactor core.

About 17 hours into the startup, the operators observed indications of unusually high leakage of reactor cooling water into the containment building. Given all the valves and piping—much containing water or steam at high pressure—inside containment, minor leakage collectively amounting to around one gallon per minute is normal. A maintenance crew entered the containment to repair a valve that had been leaking a little amount before the outage.

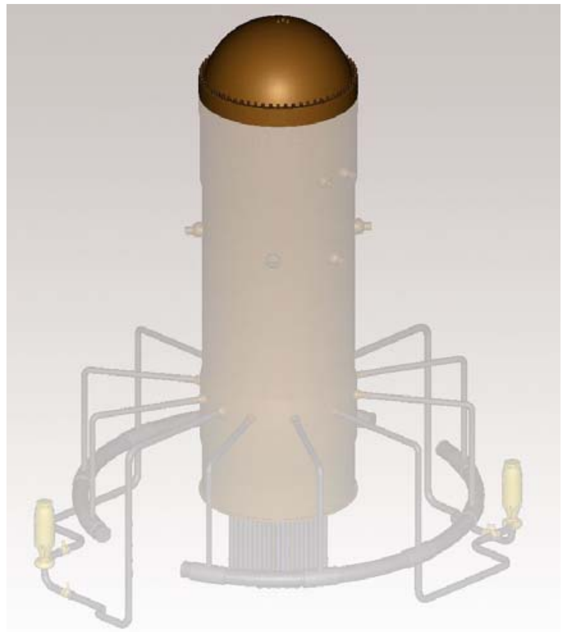
Despite repairing the valve, the leak rate increased. At 2:12 am on November 16, the leak rate was 5.88 gallons per minute, which exceeded the maximum leak rate allowed by the operating license. Workers had a few hours to find and fix the leak or the reactor would have to be shut down.

Workers re-entered the containment building and observed water dripping from equipment and running down the walls. The operators in the control room noted the temperature in the upper part of the containment was 240°F, or 40°F above the normal temperature in this area with the reactor at 100 percent power.

The leak rate continued to increase. When it exceeded 10 gallons per minute at 2:53 am, the control room operators ordered workers to leave the containment building for their safety. An Unusual Event—the least serious of the NRC's four emergency classifications—was declared at 3:01 am based on the high rate of leakage inside the containment. The operators manually scrammed the reactor (i.e., depressed two pushbuttons causing all the control rods to rapidly insert into the reactor core to terminate the nuclear chain reaction) at 3:09 am. As the temperature and pressure inside the reactor vessel decreased following the shutdown, the leak rate also decreased.

The following day, workers turned several of the retaining bolts holding the reactor vessel head on top of the reactor vessel by hand. This should not have been possible. As shown in Figure 2, the reactor vessel head is the dome-shaped part on top of the cylindrical reactor vessel housing the reactor core. The pressure inside the reactor vessel during operation can rise to nearly 1,100 pounds per square inch. The many heavy-duty nuts and bolts are intended to securely fasten the head to the reactor vessel and withstand the internal force.

Figure 2: Reactor vessel and its head



Workers had unbolted and removed the reactor vessel head earlier in November to enable the damaged fuel assembly to be removed and replaced. When installing the head back onto the reactor vessel, workers used two different methods to verify that each nut was properly tightened on its bolt.

First, workers used a stud tensioner that provided a digital read-out of the force applied to the nut. The installation procedure directed workers to tighten each nut to 13,000 pounds per square inch force. The workers tightened each nut until the digital read-out window displayed 1,300. They believed the value in the window was ten times the force applied. Instead, the window displayed the actual force. Thus, the workers tightened each nut to one-tenth of the proper force.

The second verification method relied on the fact that the bolts “stretch” as their nuts are tightened. The installation procedure directed workers to tighten each nut until its bolt elongated by 0.041 to 0.049 inches. As workers tightened the nuts, they recorded values of minus 0.001 to plus 0.004 inches. They assumed these values were not actual elongation amounts but rather the deviation from the target elongation (i.e., 0.045 plus or minus 0.004 inches). Instead, the measurements were the actual bolt elongation. Because the nuts had not been properly tensioned, the bolts had not properly elongated.

The NRC’s special inspection team discovered the plant’s owner discontinued formal training on reactor vessel disassembly and reassembly in 2000. Only 4 of the 13 workers who reassembled the reactor vessel in November 2011 had been formally trained and qualified to do the tasks.

Because the reactor vessel had not been properly reassembled, the rising pressure inside the reactor vessel during the startup lifted the head enough to squirt water out between the flanges. This hot water increased the temperature in the upper region of the containment and drained down into the basement where instruments detected the increasing leakage rate.

NRC Sanctions

The SIT identified no violations of regulatory requirements associated with this incident (NRC 2012v).

On December 10, 2012, the NRC issued Information Notice 2012-21, “Reactor Vessel Closure Head Studs Remain Detensioned During Plant Startup,” to all owners of nuclear plants in the U.S. notifying them of the problem at Brunswick with the expectation that they would take applicable steps to avoid similar problems at their facilities (NRC 2012a).

UCS Perspective

The SIT reported that only 4 of 13 workers were properly trained and qualified for this task. The SIT further determined that the owner used to train workers in this task, but had stopped the training more than a decade earlier.

Criterion II of the NRC’s regulation governing quality assurance for nuclear plants requires owners to have programs that “take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of quality by inspection and test. The program shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained” (NRC 2007). These regulatory requirements were clearly violated and the SIT should have cited it.

Byron Station, Unit 2, IL

The Near-Miss

The NRC sent an SIT to the site after the Unit 2 reactor automatically shut down from full power because of an electrical fault in the plant’s switchyard. A design deficiency prevented the electrical protection system from isolating the fault as intended. Consequently, the fault propagated to cause all the emergency equipment for the unit to be de-energized. The operators took steps to isolate the fault eight minutes later and to restore power to vital equipment from the emergency diesel generators (NRC 2012q).

How the Event Unfolded

Shortly after 10:00 am on January 30, 2012, a portion of the “C” phase power line for the Unit 2 station auxiliary transformer (SAT) in the 345,000-volt switchyard broke and fell to the ground causing an electrical fault.

Per the normal configuration for reactor operation, the switchyard supplied electricity to in-plant equipment such as the non-safety–related 6,900-volt Reactor Coolant Pump (RCP) Buses 258 and 259. Electricity from the main transformer (i.e., power being produced by the unit itself) was powering the rest of the in-plant equipment like the non-safety-–related 6,900-volt RCP Buses 256 and 257.

The “C” phase fault was detected by the electrical protection system for 6,900-volt RCP buses 258 and 259. This protection system compares the phase-to-phase voltages to detect problems. If the voltage between phases differs by more than a specified amount (e.g., one phase goes towards zero volts due to a short or another phase’s voltage rises due to too much current flow), the protection system “trips” to isolate equipment from a potentially flawed power source. In this case, it triggered an automatic shut-down of the reactor. Control rods inserted into the reactor core within seconds to terminate the nuclear chain reaction.

The 4,160-volt safety-related (SR) buses 241 and 242 were also being supplied, as was normal, from the 345,000-volt switchyard. Those two buses supply electricity for all the emergency equipment needed on Unit 2. Each bus had its own emergency diesel generator in standby and ready to power the bus if the supply from the switchyard became unavailable.

The electrical protection system for safety-related buses 241 and 242 had an undetected design problem that surfaced during this event. It compared the "A" phase voltage to the "B" phase voltage and the "B" phase voltage to the "C" phase voltage. As with the protection scheme for the non-safety–related 6,900-volt buses, this protection scheme sensed a problem when the difference between the voltages from the two phases exceeded a specified amount, in this case 2,730 volts. But this scheme required problems to be sensed both in the "A" to "B" phase check and in the "B" to "C" phase check. The "C" phase fault caused the "B" to "C" phase check to sense a problem. But the "A" and "B" phases matched within the limit, so the protection system did not isolate the safety-related buses 241 and 242 from the faulted 345,000-volt switchyard.

Two situations conspired to make things worse. The automatic shut-down of the reactor meant that the Unit 2 main generator was no longer producing electricity. As designed, in-plant equipment that had been powered from the main generator (i.e., Unit Auxiliary Transformers (UAT) 241-1 and 241-2 as well as 4,160-volt non-safety–related buses 243 and 244) automatically transferred to their backup power sources—that is, the faulted 345,000-volt switchyard. The electrical protection system for the 6,900volt non-safety–related buses 256 and 257 sensed the "C" phase fault and automatically opened breakers that isolated the buses from the faulty power supply. All four of the reactor coolant pumps turned off and were no longer circulating cooling water through the reactor core.

The second situation was far more threatening. While the electrical protection system for safety-related buses 241 and 242 did not detect the "C" phase fault, the protection systems for ALL of the emergency equipment supplied from these buses did detectthat too much electrical current was flowing through the cables. (With the "C" phase faulted, the electrical current flow through the "A" and "B" phases increased in compensation.) ALL of the emergency equipment for Unit 2 was automatically disconnected from its power sources.

Among other things, that meant that the cooling water system for the emergency diesel generators and the equipment needed to cool the reactor core and its containment was no longer available. Workers opened valves to cross-tie the cooling system with the cooling system on Unit 1 to recover those vital functions.

Ironically, ALL of the Unit 2 emergency equipment was de-energized even though each component had two separate, independent power supplies available. Both emergency diesel generators were available. Had the electrical protection system for the safety-related buses 241 and 242 functioned as intended, they would have been isolated from the faulty switchyard and de-energized. When they lost power, that would have signaled the emergency diesel generators to start automatically and re-power the buses within seconds. Alternatively, each of the buses was equipped with a connection to an available 4,160-volt safety-related bus on Unit 1, but required operator actions to connect these backups.

By checking instruments in the control room, the operators identified the “C” phase problem. That awareness along with a report from a worker in the switchyard about seeing smoke coming from station auxiliary transformers 242-1 and 242-2 prompted the operators to open electrical breakers about eight minutes after the reactor trip that isolated in-plant equipment from the 345,000-volt switchyard. This action essentially duplicated what would have happened earlier had the design problem not existed. The emergency diesel generators automatically started and re-powered safety-related buses 241 and 242. The operators manually closed electrical breakers to also re-power non-safety–related buses 243 and 244.

Power had been restored to vital electrical buses, but not yet to emergency equipment supplied by those buses. Many electrical components powered from the buses had been automatically disconnected due to too much electrical current flow. Workers had to reconnect the components individually, often from panels out in the plant rather than from the control room, to restart them.

At 8:00 pm on January 31, 2012, the Unusual Event—the least serious of the NRC’s four emergency classifications—was terminated. Workers repaired the “C” phase fault, performed checks to verify that station auxiliary transformers 242-1 and 242-2 were undamaged, and restored the normal supply of electricity to Unit 2 from the 345,000-volt switchyard (NRC 2012q).

NRC Sanctions

The SIT identified no violations of regulatory requirements associated with this incident (NRC 2012q).

UCS Perspective

The operators did a commendable job responding to this event. Had the plant’s designers done as well, the operators would not have been tested.

The SIT reported on an undetected design problem in the electrical protection system for safety-related buses 241 and 242. Criterion III of the NRC’s regulations governing quality assurance at nuclear plants requires that owners have for “verifying and checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program” (NRC 2007). Had the testing program for the electrical protection system fully tested the various phase-to-phase voltage configurations that were possible, it would have found—and fixed—the design error then. Instead, the design problem went undetected until it increased the severity of an event.

By failing to flag this clear violation of its safety regulations, the NRC did not require this owner to remedy deficiencies in its testing program. It is imperative that the NRC cite violations of its safety regulations. The importance is not to sanction “sins of the past” but rather to compel the remedies needed to prevent future sins.

Catawba Nuclear Station, Unit 1, SC

The Near-Miss

The NRC sent an SIT to the site after three unrelated electrical problems caused the Unit 1 reactor to shut down automatically from full power, both reactor units to be disconnected from the offsite power grid, and an emergency diesel generator to fail.

How the Event Unfolded

On April 4, 2012, the Unit 1 reactor at the Catawba nuclear plant was operating at full power and the Unit 2 reactor was shut down for refueling. Electrical power to vital equipment on both reactors was being supplied from Unit 1.

Four motor-driven pumps circulated cooling water through the Unit 1 reactor core. Age-related degradation of the insulation for a power cable to one reactor coolant pump caused an electrical fault that stopped the pump. Sensors detected the drop in flow from that pump and initiated the automatic and rapid shut-down of the reactor and the turbine/generator as designed.

The automatic shut-down of the turbine/generator opened two electrical breakers that disconnected it from the offsite power grid. As the electrical output from the tripped turbine/generator decreased, sensors caused other electrical breakers to open, which entirely disconnected the plant from the offsite power grid.

This power grid disconnection was supposed to occur only when the generator was online; moreover, disconnection was supposed to be automatically bypassed when the generator tripped. The frequency of alternating current electricity from the generator must match the frequency of electricity on the offsite grid. Sensors monitor both frequencies and start automatic protective measures when imbalances are detected.

Shut-down of the generator is a perfectly valid reason for its output to drop below that on the offsite grid. In the original design at Catawba, this protection circuit was automatically bypassed whenever the generator output breakers opened. The sensors would still detect a mismatch between the generator's frequency and the grid's frequency, but would no longer trigger any protective reactions.

The company had recently replaced the relays in this protection circuit on Unit 1. But they failed to tell the vendor about this bypass provision and the replacement relays did not have this feature. Additionally, the procedure used by workers at Catawba to test the replacement relays following their installation had been developed based on the incorrect information given to the vendor rather than from the original design requirements for the system. Consequently, the replacement relays successfully passed the deficient test procedure.

These same relays were being replaced on Unit 2 during its refueling outage. The new replacement relays had the same design deficiency as those already replaced on Unit 1. The disconnection event exposed the problem and led to relays on both units being replaced with properly designed and tested relays.

Both emergency diesel generators for each reactor unit automatically started and supplied electricity to vital in-plant equipment until offsite power connections were restored more than five hours later.

About three hours after offsite power had been lost, workers started a fifth emergency diesel generator. The batteries used by the plant's security system were becoming exhausted and the fifth emergency diesel generator would replenish them and sustain power to the security system equipment. Even though the fifth emergency diesel generator started, it would not supply power to the batteries.

Examination revealed that a design flaw dating back to original installation prevented the fifth emergency diesel generator from functioning properly. This fifth emergency diesel generator had been installed around 1983 specifically in the event of a station blackout. While it also supplied power to security equipment, its primary purpose was to power equipment needed to cool the reactor core.

For nearly 30 years, workers periodically tested this fifth emergency diesel generator. Normally in standby (idle) mode, the routine tests verified that the unit would start up and provide the needed amount of electricity within the specified time limit. During the tests, however, all the vital equipment was not physically connected to the emergency diesel generator. Instead, a test circuit simulated all the equipment loads being supplied from the emergency diesel generator.

After the generator's failure to supply power, workers found a wiring error that dated back to original installation in 1983. When the emergency diesel generator was started for real, the circuit simulating all the loads being connected during testing was not bypassed. The voltage regulator for the emergency diesel generator thought it had to power all the simulated loads as well as all the real loads. To do so required dropping the voltage to about 400 volts, far below that needed to operate the safety equipment. Thus, even though the emergency diesel generator was running, the design error prevented it from supplying electricity to equipment.

NRC Sanctions

The SIT identified two violations of regulatory requirements associated with the ROP's *mitigating systems* cornerstone (NRC 2012r):

- Failure to follow procurement procedures when the replacement relays for the Unit 1 electrical protection circuit were installed and tested.
- Failure to follow procurement procedures when the replacement relays for the Unit 2 electrical protection circuit were purchased.

The NRC classified the first violation as Yellow and the second as Green (NRC 2012i).

The NRC also identified a violation of the plant's operating license for having operated for nearly 30 years with a wiring error that impaired the emergency diesel generators. In this case, the NRC exercised "enforcement discretion"—its term of art for not sanctioning a plant owner for a safety violation (NRC 2012c).

UCS Perspective

The SIT's classification of the Yellow and Green violations were appropriate.

The NRC's failure to enforce its regulations in the case of the wiring error is entirely inappropriate. One could argue that the 30-year old wiring error was made by a different generation of workers and managers making it unfair to penalize current management. But there were many opportunities over the past three decades to have identified this wiring error. Several times each year, the emergency diesel generators are tested and evaluated by the system engineer. By giving the current owner a free pass on a serious safety violation, the NRC is aiding and abetting deficient testing and inspection regimes that allowed this error, and could allow countless other errors, to remain undetected for years. The NRC must take safety seriously and enforce its safety regulations.

Farley Nuclear Plant, Units 1 and 2, AL

The Near-Miss

The NRC sent an SIT to the plant in response to security-related problems. Reflecting the NRC's post–9/11 procedures, the SIT report on the problems and their remedies is not publicly available. However, the cover letter sent to the plant owner with the SIT report is publicly available, and indicates that the agency identified no violations or significant findings (NRC 2012x).

UCS Perspective

The scant information publicly available about this security near-miss prevents any meaningful commentary.

Fort Calhoun Station, NE (first incident)

The Near-Miss

The NRC sent an SIT to the plant following a fire (NRC 2012r).

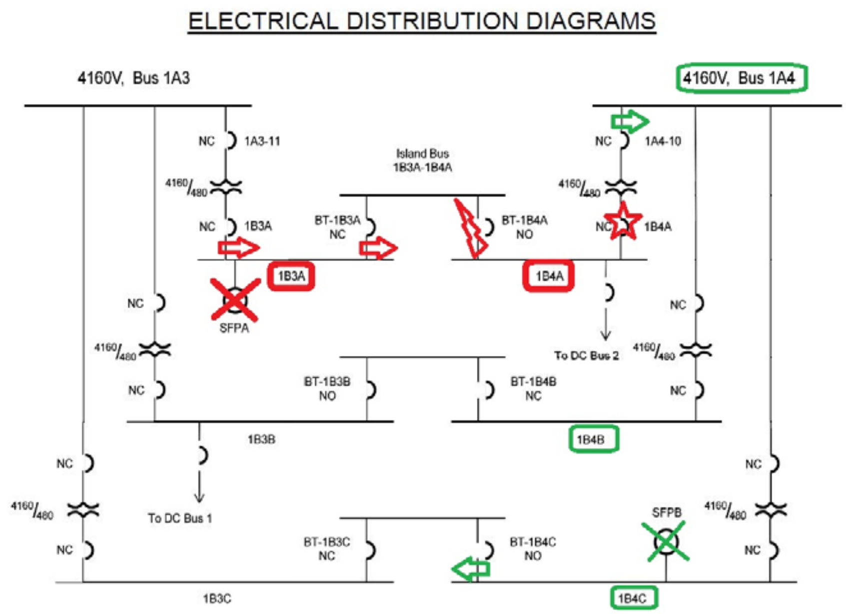
How the Event Unfolded

At 9:27 am on June 7, 2011, the 480-volt electrical breaker 1B4A (identified with a red star in Figure 3) catastrophically failed. Its failure started a fire. Smoke and soot from the fire allowed electricity to arc across open electrical breaker BT-1B4A to cause voltage fluctuations on 480-volt electrical buses 1B3A-1B4A and 1B3A. These electrical transients caused breakers 1B3A and BT-1B3A to automatically open to isolate the electrical problem. The opening of these two breakers de-energized 480-volt electrical bus 1B3A, which in turn de-energized spent fuel pool cooling pump A (labeled SFPA in the figure). Spent fuel pool cooling pump A had been running at the time to cool the irradiated fuel stored in the spent fuel pool.

The failure of breaker 1B4A prevented it from opening, either automatically or in response to efforts by the operators in the control room. Instead, the operators manually opened breaker 1A4-10 to stop electricity flowing from 4,160-volt bus 1A4 to and through the damaged breaker. Per procedure, the operators then de-energized 4,160-volt bus 1A4. As a result, 480-volt buses 1B4B and 1B4C were de-energized. Spent fuel pool cooling pump B (SFPB) was supplied from bus 1B4C. Both spent fuel pool cooling pumps were now unavailable and would remain so for nearly 90 minutes.

The operators closed breaker BT-1B4C. This enabled electricity from 4,160-volt bus 1A3 to flow through 480-volt bus 1B3C and re-energized 480-volt bus 1B4C. The operators then started spent fuel pool cooling pump B. The spent fuel pool's water temperature rose 3°F during the 90 minutes that cooling was lost.

Figure 3: Electrical distribution system at Fort Calhoun



The NRC team discovered that workers noticed an acrid odor in the small room housing electrical breaker 1B4A and 480-volt bus 1B4A three days before the fire. The day before the fire, workers again noticed an acrid odor in the room and entered a report in the plant's corrective action program about it. But even though the plant had instruments onsite to perform thermographic scans of the electrical equipment to identify hot spots that might be sources of the acrid odor, it was not used. The NRC team also determined that workers did not employ low-tech techniques such as opening electrical panels to look for smoke or abnormally high temperatures.

The failed electrical breaker was among 12 breakers replaced in November 2009. The replacement breakers were of a different size and material than the original breakers. These design differences created the potential for the breakers to experience higher temperatures during operation due to increased resistance to electrical current flow. The higher temperature exacerbated the situation by increasing the oxidation rate of internal parts, adding even more resistance to current flow.

During installation, the replacement breakers did not align properly in the breaker compartments so workers made unapproved on-the-spot changes to make them fit. Following installation, workers used a hand-held mirror to visually determine if the pieces seemed to fit together properly. They did not measure the incoming and outgoing electricity to confirm consistency with characteristics of the original breakers.

Following replacement of the breakers, workers periodically cleaned the breakers. But they cleaned only part of the connection between the breakers and their buses. They did not remove hardened grease that was present on other parts. Over time, the grease increased the electrical resistance which increased the temperature of the breakers during operation.

The NRC team identified a deficiency in the plant's fire protection design. Specifically, the plant's design was supposed to prevent a fire or electrical transient from propagating from one safety train to another safety train. The 4,160-volt buses 1A3 and 1A4 are in different safety trains, yet a fire affecting 480-volt bus 1B4A crossed isolation boundaries to impair 480-volt bus 1B3A in the other safety train. Consequently, a single fire de-energized both spent fuel pool cooling pumps. The two pumps had been powered from different safety trains specifically to prevent such an outcome.

The NRC team also identified several problems with workers' response to the fire including:

- After breaker 1B3A unexpectedly opened and de-energized the only running spent fuel pool cooling pump, the operators made repeated attempts to close it from the control room. But the design of this electrical breaker required that it be reset locally after opening locally, a fact that the operators did not understand from their training and operating procedures.
- Contrary to procedures, the onsite fire brigade leader relinquished command to the offsite fire department when it arrived. Onsite command is required because offsite responders are not trained on the plant's systems and may not understand the adverse consequences from de-energized or disabling equipment during their fire-fighting efforts.
- No one searched the fire area for potential victims nor was an accountability check performed to verify that no workers were missing.

NRC Sanctions

The SIT identified three violations of regulatory requirements associated with the ROP's *initiating events* cornerstone (NRC 2012r):

- Failure to control changes to the plant's design in a proper manner when 12 electrical breakers were replaced in November 2009 with ones of different size and material.
- Failure to implement proper, timely, and effective corrective actions to ensure proper cleaning of electrical breaker connections and evaluate abnormally high connection temperatures.
- Failure to ensure that the 480-volt electrical power distribution system was adequately protected from a single problem adversely affecting redundant safety components.

The NRC classified the violations as Red (NRC 2012n).

UCS Perspective

The Red classifications in this case correlate more to the NRC's prior impression of this plant owner than to the safety severity of the violations.

NRC was already irked about conditions at Fort Calhoun and therefore came down hard on this problem. For comparison, the event described below at Harris, where key valves that need to close within five seconds to protect workers and the public took as long as 4 hours and 7 minutes to do so, and the event described above at Catawba where a wiring error impaired emergency diesel generators for nearly 30 years did not receive even a Green finding from the NRC although these conditions were arguably more dangerous. But the NRC was not already irked at those owners.

The ROP cannot be the NRC's mood ring signaling the agency's fondness level for various nuclear plant owners. The NRC must consistently enforce compliance with its safety regulations by all owners and impartially mete out appropriate sanctions when violations are identified.

Fort Calhoun Station, NE (second incident)

The Near-Miss

The NRC sent an SIT to the plant in response to security-related problems. Reflecting the NRC's post–9/11 procedures, the SIT report on the problems and their remedies is not publicly available. However, the cover letter sent to the plant owner with the SIT report is publicly available, and indicates that the agency identified one or more violations having greater-than-Green significance (NRC 2012u).

UCS Perspective

The scant information publicly available about this security near-miss prevents any meaningful commentary.

Harris Nuclear Power Plant, NC

The Near-Miss

The NRC sent an SIT to the plant after one of the three main steam isolation valves took 37 minutes to close, and a second took 4 hours and 7 minutes to close during testing. All three valves are supposed to close within 5 seconds during an accident to limit how much radioactivity is released to the atmosphere (NRC 2012k).

How the Event Unfolded

The Shearon Harris nuclear plant features a three-loop Westinghouse pressurized water reactor. Water flowing through the reactor core when the plant operates is heated to over 500°F but kept from boiling due to high pressure. The hot, pressurized water is carried from the reactor vessel via large pipes to three steam generators. The pressurized water flows through thousands of tubes in the steam generators. Heat conducted through the thin metal walls of the tubes boils water in the three steam generators to generate steam, which flow to the turbine to make electricity. Meanwhile, the pressurized water is pumped back to the reactor vessel to be reheated.

In the pipes between the three steam generators and turbine are main steam isolation valves, which are normally open during operation. In case of accident, these valves are designed to close automatically within five seconds to limit the amount of radioactivity released to the environment. If one or more tubes within the steam generators broke, hot, pressurized water from the reactor core along with any radioactivity it contained would leak out. The main steam isolation valves protect workers and the public by closing when necessary to limit how much radioactivity escapes from the containment building.

On April 12, 2012, workers were shutting down the Harris reactor to enter a refueling outage. The process included a routine test of how long it took each of the three main steam isolation valves to travel from the fully opened to the fully closed position. An operator in the control room turned a switch signaling main steam isolation valve “A” to close. It closed in 4.51 seconds, safely within the 5-second limit. When the switch for main steam isolation valves “B” and “C” were turned, the valves did not close.

The main steam isolation valves feature large springs that close them. The springs for each valve are designed to provide 63,988 pounds (nearly 32 tons) of closing force. To open a valve, compressed air is supplied to push against the spring force and open the valve. This is a fail-safe feature: on loss of power or loss of compressed air pressure, the springs automatically close the valves.

Main steam isolation valve “B” closed 37 minutes after its compressed air supply was removed. Main steam isolation valve “C” took 4 hours and 7 minutes to close. Had Harris experienced an accident in which these valves needed to close, their much delayed closure could have resulted in considerably more radioactivity being released to the environment.

Workers disassembled all three main steam isolation valves and evaluated nearly 30 potential causes for the failures to close. They found that corrosion had caused some of the internal parts of the valves to swell in size nearly 20 percent. This expansion effectively locked the valves in place against the spring force—as great as it was— even after the compressed air had been removed.

The valves had been installed during the plant’s construction more than a quarter of a century earlier. The valves’ manufacturer subsequently developed valves using material more resistant to corrosion, but had not also recommended that customers with older valves upgrade to the improved design. After this event, workers replaced all three main steam isolation valves with the new models. All three valves then tested successfully within the five second closure time requirement.

From the plant’s initial startup in 1987 through 2000, workers tested the main steam isolation valves quarterly per the manufacturer’s recommendation. That testing involved closing each valve ten percent to verify proper functioning of the valve, actuator, and control circuit. In 2000, the plant’s owner discontinued this quarterly testing.

The NRC’s SIT discovered that the air-operated main steam isolation valves had never been tested under the air-operated valve testing program. The valves had been classified as Category 2 valves which do not require testing. But the NRC’s SIT determined that the main steam isolation valves met the Category 1 definition as having an active safety-related function with high safety significance.

NRC Sanctions

The SIT identified two unresolved issues related to the testing protocols for the main steam isolation valves. As of December 31, 2012, the NRC had not announced its decision on what it would do, if anything, about these issues.

UCS Perspective

This event, along with the Brunswick event, are troubling. Both involved measures (valve testing here, worker training at Brunswick) that the same owner of the two plants discontinued in the 2000/2001 timeframe to save money. It took ten years for the consequences of these improper cost-cutting measures to reveal themselves. What other consequences are lurking to initiate or exacerbate the next near-miss, or worse?

The ROP's report cards are issued quarterly based on information collected during the prior three months. The ROP's quarterly grades determine the extent and focus of NRC's oversight effort. The Harris event happened in April 2012. The SIT issued its report in July. The NRC's final answer on the two unresolved issues identified by the SIT should have been reached no later than October 2012—three months after the SIT's report and six months after the actual event—but in early 2013 has still not been announced. "Justice delayed is justice denied" should be a cliché, not the NRC's decision-making process. Swift resolution of safety problems identified by SITs and AITs is essential to applying timely and appropriate NRC oversight resources.

Palisades Nuclear Plant, MI

The Near-Miss

The NRC sent a SIT to the plant after workers shut down the reactor because of a leak of about 18 gallons per hour of cooling water determined to be through the reactor coolant pressure boundary. The plant's operating license does not permit the reactor to operate for more than six hours with such leakage; however, the reactor operated for nearly a month under those conditions.

How the Event Unfolded

Workers began shutting down the reactor at 11:07 pm on August 11, 2012. Since the reactor had restarted on July 10 from another unplanned maintenance outage, operators had detected that reactor cooling water was leaking into the containment building. On July 16, the leak rate exceeded NRC's threshold limits. Workers entered the containment several times trying to find the source of the leak, but were unable to locate it.

The leakage rate increased until it reached 0.3 gallons per minute (about 18 gallons per hour) late on August 11. Management directed the operators to shut down the reactor to allow workers to access all of the containment building to search for the leak. On August 12 after the reactor had been shut down, workers identified the leak to be from a crack in one of the control rod drive mechanism (CRDM) housings.

Control rods contain material that absorbs neutral subatomic particles called neutrons. Neutrons are released when atoms fission (split apart) to power the nuclear reactor. Control rods are withdrawn vertically upward from the reactor core to permit more neutrons to be available to cause atoms to split faster, increasing the power level. Control rods are inserted downward into the reactor core to make fewer neutrons available and reduce the reactor core's power level or shut it down entirely.

An electric motor for each control rod enables its insertion and withdrawal. The electric motors are outside of the metal reactor vessel that houses the reactor core, and are mounted on top of the reactor vessel's head. Metal poles connect the motors with the control rods. Each metal pole passes through a roughly 4-inch diameter penetration through the 6-inch thick reactor vessel head. The CRDM housing encloses the electric motor and accommodates the upward and downward travel of the metal pole and the control rod connected to the pole's lower end. Part of the metal CRDM housing cracked and was leaking reactor cooling water. The owner reported it to the NRC as being reactor coolant pressure boundary leakage (Entergy 2012).

Like other U.S. pressurized water reactors, Palisades has four limits in its operating license for reactor coolant leakage:

1) 150 gallons per day leakage through the tubes within any single steam generator,
2) 10 gallons per minute leakage through identified pathways,
3) 1 gallon per minute leakage through unidentified pathways, and
4) No leakage through the reactor coolant pressure boundary.

Whenever any of the first three limits is exceeded, the reactor can operate for up to four hours while workers attempt to reduce the leakage rate back within the specified limits. Whenever any of the first three limits is exceeded for four hours or whenever the fourth limit is exceeded, the reactor must be shut down within six hours.

Thus, while the NRC-issued operating license allowed Palisades to operate for only up to six hours with this reactor coolant pressure boundary leakage, the reactor operated for nearly a month.

The problem stems from having four limits and only three monitoring methods. Steam generator tube leakage is monitored by detectors that only react to leaking tubes. Those detectors reliably determine whether the 150 gallon per day limit is being met.

Identified leakage is also effectively monitored. For example, the large pumps that circulate water through the reactor vessels experience small amounts of leakage along their pump shafts. The reactor cooling water leaking from the pump shafts is collected. The detector trending that collection reliably identifies the source of that leakage.

Unidentified leakage, as its name suggests, could be coming from anywhere. Almost anywhere, as it's known not to be leaking through steam generator tubes or from identified sources like the coolant pumps. Basically, these leak detectors track the water that ends up in the containment's basement. It could have come from leaking valves, cracked piping, or literally thousands of other places inside containment—including the reactor coolant pressure boundary.

When unidentified leakage is detected, it's commonly assumed to be someplace other than the reactor coolant pressure boundary. Statistically, this assumption is valid the majority of the time. A minority of the time, as in this case at Palisades, the assumption is bogus. This flawed assumption enabled

Palisades to operate for over 30 days with safety degradation that its operating license only permitted to exist for up to six hours.

NRC Sanctions

None. The NRC did not sanction the owner for its bad safety guess (NRC 2012d).

UCS Perspective

The NRC's inaction in this matter is inexcusable. Safety requirements warrant that this reactor shut down within six hours if reactor coolant pressure boundary leakage occurs. Yet the reactor was operated for nearly a month. The NRC must enforce safety requirements.

Palo Verde Nuclear Generating Station, Units 1, 2, and 3, AZ

The Near-Miss

The NRC sent a SIT to the plant in response to security-related problems. Reflecting the NRC's post–9/11 procedures, the SIT report on the problems and their remedies is not publicly available. However, the cover letter sent to the plant owner with the SIT report is publicly available, and indicates that the agency identified one violation having greater-than-Green significance (NRC 2012t).

UCS Perspective

The scant information publicly available about this security near-miss prevents any meaningful commentary.

Perry Nuclear Power Plant, OH

The Near-Miss

The NRC sent a SIT to the plant after its owner reported failures to prevent unauthorized individuals from entering secure areas. The NRC identified one violation having greater-than-Green significance (NRC 2012m).

How the Event Unfolded

On January 25, 2012, the owner informed the NRC that its security plan failed to control access to protected areas of the facility properly, creating the potential for unauthorized individuals to enter.

Nuclear plants feature three primary security zones: (1) the owner-controlled area, (2) protected areas, and (3) vital areas. The owner-controlled area is essentially the entire acreage that the nuclear plant is built on. This zone includes parking lots, warehouses, office buildings, training buildings, etc. which require no security from a nuclear sabotage perspective. Security fences surround protected areas on the owner-controlled area; for example, the reactor building and the intake structure. Security fences accompanied by armed guards, motion sensors, and infrared detectors prevent unauthorized individuals and vehicles from entering protected areas. Within protected areas, security doors that can only be opened with a physical key or a computer access card further limit access to vital areas like the control room and emergency diesel generator building.

Perry's owner informed the NRC that its security program for monitoring underground pathways and other unattended openings "were insufficient to detect and prevent unauthorized access to the protected area." In the past, other owners have found similar problems. For example, the very large diameter pipe carrying cooling water from the nearby lake, river, or ocean to the plant becomes a very large tunnel when the plant is shut down and the cooling system is turned off. And underground concrete trenches carrying electrical cables into the plant have sometimes been found to have sufficient space to allow individuals to crawl through (FirstEnergy 2012).

NRC Sanctions

The SIT identified one violation classified as being greater-than-green (NRC 2012m).

UCS Perspective

When I worked at the Browns Ferry nuclear plant in the early 1980s, similar problems were identified and reported. That was three decades ago. This "discovery" at Perry came more than a decade after 9/11. Why did NRC's security inspections and the owner's own efforts miss such problems for so many years?

While the security veil hides some information about this problem and its resolution, the available information strongly suggests that the NRC is requiring that the owner fix the identified problems of uncontrolled access into the plant's protected area. That's good, but there is no evidence that the NRC is also requiring that the owner fix its testing and inspection regime shortcomings that allowed this security problem to remain undetected for so many years. Both fixes are equally important and yet the NRC seems willing to accept half measures. That is fully wrong.

River Bend Station, LA

The Near-Miss

The NRC sent an AIT after a fairly routine event—trip of the reactor due to loss of a single feedwater pump—cascaded into loss of the normal heat sink and loss of cooling water to emergency and non-emergency equipment, because of problems in the in-plant electrical distribution system (NRC 2012g).

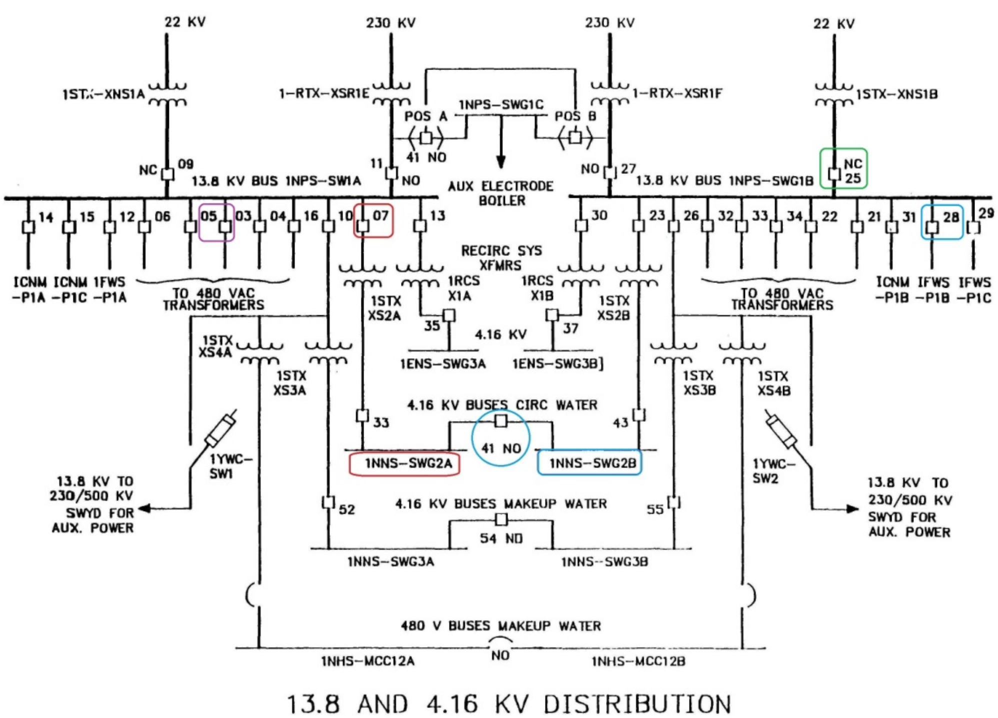
How the Event Unfolded

At 2:52 pm on May 21, 2012, the River Bend reactor automatically scrammed from 100 percent power. Rising pressure within the main condenser (which condenses steam back to water) had caused the turbine to shut down automatically, which in turn sent a signal to shut down the reactor automatically. The control rods entered the reactor core within seconds to terminate the nuclear chain reaction.

The rising pressure in the condenser occurred when two of the four circulating water pumps suddenly stopped running. The reduction in circulating water flowing through the main condenser resulted in less effective cooling of the steam. The pressure inside the main condenser, normally maintained far below atmospheric pressure to help “pull” steam through the turbine, began rising. When instruments detected that the main condenser was unable to handle the amount of steam entering it from the turbine, they signaled the automatic shut-down of the turbine. In turn, when instruments detected that the turbine was no longer able to handle the amount of steam being produced by the reactor, they signaled the automatic shut-down of the reactor.

Five minutes after the turbine and reactor shut-downs, operators in the control room were notified of a fire in an underground electrical cable vault. They dispatched the fire brigade to the scene. The fire brigade reported a fire in the highest tray carrying electrical cables within the vault; they used portable extinguishers to put it out.

Figure 4 Electrical distribution system at River Bend



One of the electrical cables supplying power from 13,800-volt electrical bus 1NPS-SW1A to 4,160-volt electrical bus 1NNS-SWG2A had failed. Its failure started the fire and caused an electrical breaker (number 07 within the red rectangle in Figure 4) to open. The two circulating water pumps powered from 1NNS-SWG2A stopped running when the bus was de-energized. The other two circulating water pumps, powered from unaffected 4,160-volt electrical bus 1NNS-SWG2B, continued running, but were unable to supply enough flow through the main condenser to prevent the turbine from tripping.

To allow the reactor to be restarted before the damaged cables were repaired or replaced, workers closed a normally open electrical breaker (number 41 within the blue circle in Figure 4) that cross-tied electrical buses 1NNS-SWG2A and 1NNS-SWG2B. This meant that all four circulating water pumps were now being powered from 13,800-volt electrical bus 1NPS-SWG1B through the connected 4,160-volt buses.

Two days later, on May 23, workers began restarting the reactor. By midday on May 24, they had increased the reactor's power level to 33 percent. One of three feedwater pumps was running and two of four circulating pumps were running. At 1:48 pm, the operators started a second feedwater pump (1FWS-P1B in Figure 4) needed for higher power operation. An electrical cable supplying power to the motor of the feedwater pump failed. A protective relay that was supposed to open an electrical breaker (number 28 within the blue rectangle in Figure 4) to stop the flow of electricity to the motor and prevent the fault from propagating failed to function. This electrical problem forced another electrical breaker (number 25 within the green rectangle in Figure 4) to open. Its opening de-energized 13,800 volt bus 1NPS-SWG1B and in turn de-energized both of the cross-connected 4,160-volt buses 1NNS-SWG2A and 1NNS-SWG2B.

The loss of these two buses turned off all feedwater pumps, all circulating water pumps, and all service water pumps at the plant. The operators manually scrammed the reactor. The lowering water level inside the reactor vessel caused by the loss of the feedwater pumps would have resulted in an automatic reactor shut-down signal within seconds anyway.

The loss of all circulating water pumps meant that the reactor lacked its normal heat sink (the main condenser). At 2:05 pm, the operators closed the main steam isolation valves in the four pipes between the reactor vessel and the turbine to stop steam from flowing into the main condenser. This meant that steam being produced by the reactor core's decay heat had no place to go. Pressure inside the bottled up reactor vessel rose. Per procedures, the operators used the safety relief valves to discharge steam to the suppression pool (the backup heat sink) and control pressure inside the reactor vessel.

The loss of all feedwater pumps meant that the reactor lost its normal source of makeup water (the feedwater system). Per procedures, the operators used the reactor core isolation cooling system to provide makeup water.

The loss of all service water pumps meant that the plant lost its normal cooling water supply for emergency and non-emergency equipment. The standby service water system started and resumed the flow of cooling water to emergency equipment.

Beginning at 3:01 pm, workers began restoring power to in-plant equipment by closing electrical breakers to cross-connect buses (i.e., the extension of the remedy obtained by closing breaker number 41).

At 3:34 pm, the operators opened the safety relief valves to reduce pressure inside the reactor vessel. The pressure drop caused the indicated water level inside the reactor vessel to rise above the setpoint that automatically shuts down the reactor core isolation cooling system. Just as cracking the cap of a shaken soda bottle allows bubbles to form and the beverage level to rise, a sudden reactor vessel pressure reduction allows steam bubbles to form and the indicated water level to rise.

After closing the safety relief valves, the operators restarted the reactor core isolation cooling system at 3:45 pm.

By 2:00 am on May 25, the operators had cooled the reactor water temperature down below 212°F. This milestone reflected that conditions at the plant had been stabilized.

The NRC's AIT identified several shortcomings at the plant that factored into the event. The main problem areas were:

- Operator conduct: The NRC identified numerous variances between how control room operators at River Bend behaved and practices that have been standard in the industry for years. Inconsistent usage of three-way communications was among the examples cited by the NRC. Standard industry practice is for oral communications to be stated by the sender, repeated back by the receiver, and then verified by the sender to ensure the proper message had been understood. That practice was not followed at River Bend. The NRC also noted poor practices regarding control room annunciators. Annunciators are audible and visual alarms about off-normal plant conditions. When an instrument detects that a plant parameter (e.g., reactor vessel pressure or spent fuel pool water temperature) is outside its normal range, an annunciator in the control room will flash on and off and sound a loud alarm. The NRC observed operators at River Bend silencing annunciators without scanning the panels to see what condition had caused the alarm. And the NRC reported that the control room operators failed to provide periodic updates to other members of the crew. It is standard industry practice for control room operators to update their colleagues about changes in key parameters so as to maintain proper situational awareness among the entire crew.
- RCIC Design Flaw: The NRC found that the reactor core isolation cooling (RCIC) system had failed to start automatically following the May 21 event. The pressure transient caused by the rising main condenser pressure caused a false indication of high steam flow in the pipe going to the RCIC turbine.[5] Workers modified the plant in 2007 to protect against such a false indication, but that modification was not successful. On May 31, 2012, workers modified the plant again to install a fix that the rest of the industry implemented years ago.
- Corrective Actions for Prior Relay Failure: The NRC's AIT discovered that an electrical fault on February 12, 2011, caused a fire when an electrical breaker (number 05 within the magenta rectangle in Figure 4) failed to open as designed. The vendor manual for these types of breakers recommended periodic testing. Further, a vendor's service advisory published in 1981 cautioned against mechanical binding as a failure mode for this type of breaker and recommended periodic exercising of breakers to protect against it. Following the February 2011 failure, maintenance procedures at River Bend were revised to incorporate this recommendation, but the AIT found no evidence that it had ever been done.
- Cable Reliability Program: Workers determined that the electrical cable failure that initiated the May 21 event was caused by moisture

[5] The RCIC system uses a turbine supplied by steam produced by the reactor core's decay heat connected to a pump to supply makeup water to the reactor vessel. Excessive steam flow in the pipe to the RCIC turbine could indicate that downstream portion of piping has ruptured. When excessive steam flow is detected, valves automatically close to stop the loss of inventory from the reactor vessel and the release of radioactivity into secondary containment.

that entered through a cable splice and caused the cable's insulation to degrade at an accelerated rate. Although the NRC had cautioned that moisture intrusion could degrade underground cables at faster rates, only 14 of 57 cables identified as potentially susceptible at River Bend had been tested. The NRC AIT determined that the cable reliability program had shortcomings.

- Procedure Adequacy: The NRC AIT observed the attempted restart of the reactor on June 1. During the startup, indications were received in the control room that two safety relief valves (SRVs) were leaking with the reactor pressure at 600 pounds per square inch (normal pressure at full power is approximately 1,000 pounds per square inch). The vendor recommended opening and closing the valves to see if they would fully close and stop the leakage. Because this condition is seldom encountered, management invoked a procedure titled "Infrequently Performed Tests of Evolutions." This process is frequently invoked within the industry to provide additional pre-planning steps and controls to be applied when conducting non-routine evolutions. The NRC AIT noted that the process employed at River Bend departed significantly from standard industry practice. The standard process includes a formal pre-job briefing to be reviewed by the operating crew shortly before conducting the evolution. The pre-job briefing covers potential problems and their associated contingencies. At River Bend, the potential problems were limited to "SRV sticks open" and no contingencies were provided. In addition, the actual procedure used by the operators during the evolution was written assuming the reactor was in a different operating configuration. It included many provisions not applicable to the actual conditions on June 1 and lacked many measures that were applicable to those conditions. The NRC AIT reported that while the "Infrequently Performed Tests of Evolutions" procedure "was appropriately referenced, it did not appear to have been effectively implemented for its intended purpose."
- Onsite Safety Review Committee: The NRC AIT observed the meeting conducted by the Onsite Safety Review Committee on May 31. The committee comprises senior managers at the site and is intended to provide an independent review "to assure the plant is operated and maintained in accordance with the operating license and applicable regulations." During the May 31 meeting, issues were reviewed related to the pending restart of the reactor. The NRC AIT noted that the Onsite Safety Review Committee had not been provided the latest information on the feedwater pump motor failure that initiated the May 24 event, had not been informed about the modification to the RCIC system intended to remedy the failure it experienced during the May 21 event, and had not been provided current information regarding the electrical breaker problems that factored into the February 2011 and May 24, 2012, events. And while the Onsite Safety Review Committee did examine issues that had been categorized as involving degraded or non-conforming conditions, the list they had been provided was incomplete as it had

not been updated since the May 21 event to include issues arising from the May 24 event.

- Fire Brigade: The NRC AIT learned that the fire brigade's response to the May 21 fire was slowed because the battery for the fire brigade vehicle was depleted. The overhead light inside the fire brigade van had been left on, exhausting the battery. Fire brigade members transferred equipment to a security vehicle and headed off to the fire. At the time of the May 24 fire, the fire brigade leader was in the medical center waiting to take a random fitness-for-duty drug and alcohol test. The fire brigade leader was told that responding to the fire before taking the screening test would be considered failure to take the test, meaning the individual's access badge would be pulled. The individual waited to take the screening test and reported to the fire location after the fire had been extinguished. The NRC AIT observed that difficulties had been encountered assembling the full fire brigade complement (a total of five persons) during both the May 21 and May 24 fires. Both fires occurred on day shift when many other workers were available. The NRC AIT questioned whether the fire brigade function could be fulfilled during nights and weekends when station staffing levels were minimal.

NRC Sanctions

The AIT identified four violations of regulatory requirements associated with the ROP's *initiating events* cornerstone:

- Failure to develop adequate controls for low-power opening and closing of safety relief valves.
- Failure to identify and correct a condition adverse to quality in that after an electrical breaker failed in February 2011, steps were not taken to identify and correct other susceptible components.
- Failure to establish adequate preventative maintenance procedures for electrical breakers because vendor recommendations for periodic testing were initially omitted from maintenance procedures and then not performed after the procedures had been revised to include them.
- Failure to establish an effective cable reliability program.

The AIT also identified three violations of regulatory requirements associated with the ROP's mitigating systems cornerstone:

- Failure to identify and correct a condition adverse to quality in that spurious trips of the reactor core isolation system were not remedied until after multiple failures.
- Failure to declare the reactor core isolation system inoperable after it failed to start automatically following a turbine trip.
- Failure to assure adequate fire brigade staffing levels.

The AIT also identified one violation of regulatory requirements not specifically addressed within the ROP's cornerstones:

- Failure to follow the Onsite Safety Review Committee procedure in that the committee failed to accomplish the independent review of station activities.

The NRC classified all eight violations as Green (NRC 2012b).

UCS Perspective

This near-miss best illustrates the need for the recommendations made in the Observations section at the end of this chapter. The AIT documented many significant departures between standard practices long adopted by the rest of the nuclear industry and practices at River Bend.

- Why did it take until this near-miss for these many gaps to be identified?
- Should NRC's routine inspections have found at least some of these factors sooner?
- Should the plant owner's tests and inspections have found at least some of these factors sooner?

Violations such as the eight identified by the AIT are opportunities to adjust the NRC's inspection efforts for greater effectiveness and to find and fix shortcomings in the owner's testing and inspection regimes.

In this case as in most cases, the NRC allowed the owner to treat only the symptoms of the problem that created the violations. The owner will revise a few procedures and alter its training program a little to satisfy the NRC. But the real problems that allowed these violations to form and fester until this near-miss exposed them will continue to undermine safety. The NRC must cease its safety charade and get serious about enforcing all its safety regulations.

The NRC's regulations governing quality assurance at nuclear plants (NRC 2007) require that owners find and fix safety problems in a timely and effective manner. When evidence such as these eight violations clearly demonstrate that owners did not comply with the quality assurance regulations, the NRC must enforce these regulations to compel the reforms that prevent future safety shortfalls.

San Onofre Nuclear Generating Station, Units 2 and 3, CA

The Near-Miss

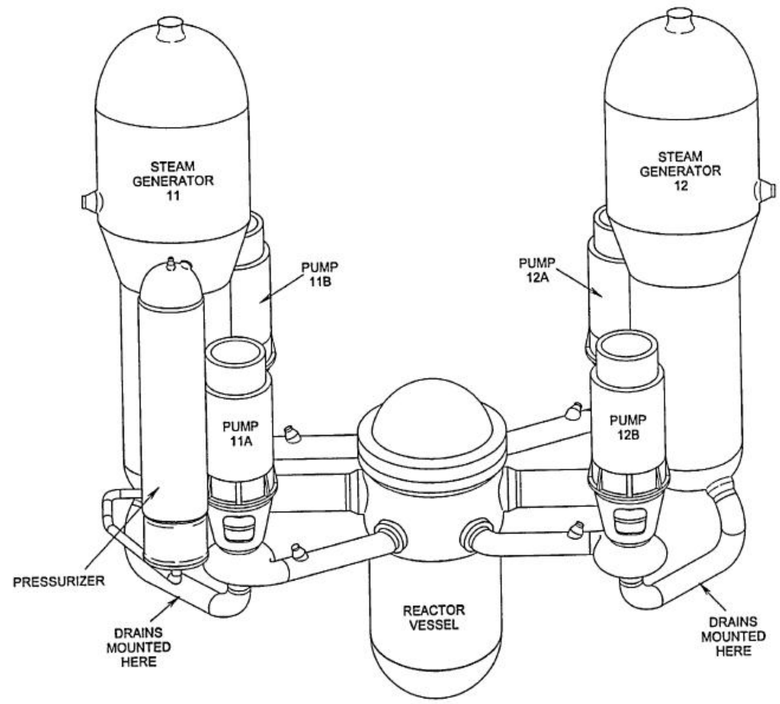
The NRC sent a AIT to the plant after unexpected degradation was identified for the tubes within the recently replaced steam generators on Units 2 and 3 (NRC 2012j).

How the Event Unfolded

On January 31, 2012, operators received indications that a small amount of reactor coolant water was leaking through one or more of the thousands of tubes inside the two steam generators for the Unit 3 reactor. In accordance with their procedures, the operators manually shut down the reactor.

The primary system for each of the two operating reactors at San Onofre consists of a reactor vessel housing the reactor core, two steam generators, four reactor coolant pumps, one pressurizer, and associated piping and valves. Water heated while flowing past the reactor core is pumped through thousands of tubes inside the steam generators. Heat is conducted through the thin metal walls of the tubes to boil water surrounding the tubes within the steam generator. The cooled water is pumped back to the reactor vessel to be reheated. Pipes carry steam from the steam generators to the turbine where it is used to make electricity.

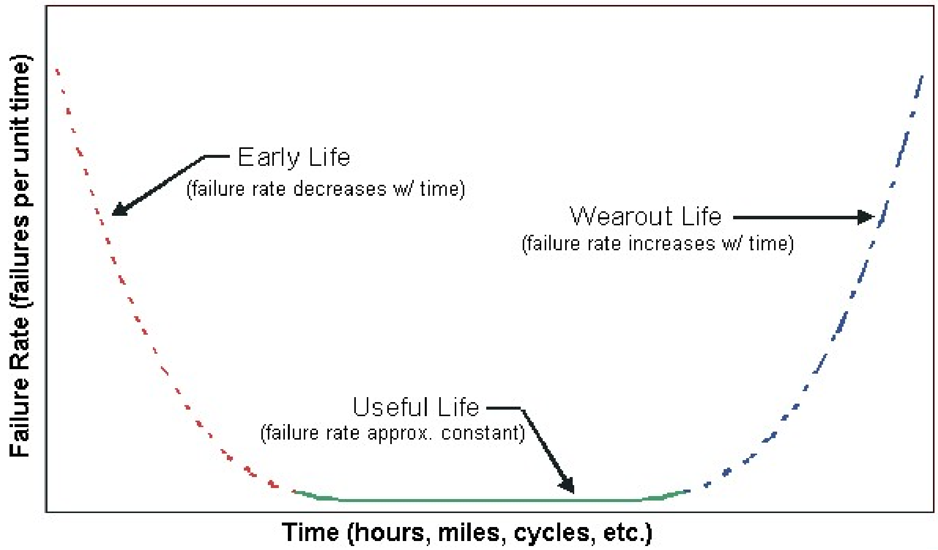
Figure 5: Primary system components at San Onofre



By 2009, the original steam generators on both Unit 2 and Unit 3 had been in service for nearly 30 years. Between September 2009 and April 2010, workers replaced the steam generators on Unit 2 during a refueling outage. Between October 2010 and February 2011, workers replaced the steam generators on Unit 3 during another refueling outage. . Operators restarted Unit 3 on February 18, 2011, and it ran less than a year before experiencing its tube leak in January 2012. At that time, Unit 2 was shut down for another refueling outage.

A failure trend, commonly called the “bathtub curve” due to its shape, explains what happened to San Onofre’s steam generators. The curve shows the chance of failure over a product’s lifetime. The failure rate is initially high during the “break-in” phase due to material defects, assembly errors, and related problems.. With time, the failure rate declines until it is lowest during the useful middle of its operational life. The failure rate increases on the right-hand side of the curve during the “wear-out” phase as aging mechanisms such as rusting and embrittlement increase the likelihood of failure. With the original steam generators at San Onofre approaching, if not already in, this wear-out phase, its owners elected to replace the steam generators on both operating reactors.

Figure 6: The Bathtub Curve



But the replacement steam generators began service not in the middle part of the bathtub curve where failure rates are lowest, but on the left-hand portion of the curve during the break-in phase.

The company sought to upgrade performance with the replacement steam generators. The original steam generators had tubes made from a type of steel called Alloy 600. This metal had very good heat transfer properties. But Alloy 600 was found to be vulnerable to stress corrosion cracking, a degradation mechanism that caused tubes to be removed from service by plugging them when cracks grew to 40 percent or more through their thin walls. The replacement steam generators featured tubes made from a different materials called Alloy 690 which had been shown to be significantly more resistant to stress corrosion cracking.

While the replacement steam generators were equipped with better materials, they had different designs. Each of the original steam generators had 9,350 tubes approximately ¾-inches in diameter with walls about 0.043 inches thick. Each replacement steam generator had 9,727 tubes approximately ¾-inches in diameter with walls about 0.048 inches thick.

Inspections of the tubes within the Unit 2 and 3 replacement steam generators identified more than 7½ percent of the tubes in each having indications of wear—some wear quite significant—even though they were brand new.

Table 3: Replacement Steam Generator Tubes with Wear Indications				
	Wear <20%	Wear >20%	Total*	Percent
Unit 2-88	1,006	78	734	7.5
Unit 2-89	1,264	69	861	8.9
Unit 3-88	1,371	390	919	9.4
Unit 3-89	1,288	363	887	9.1

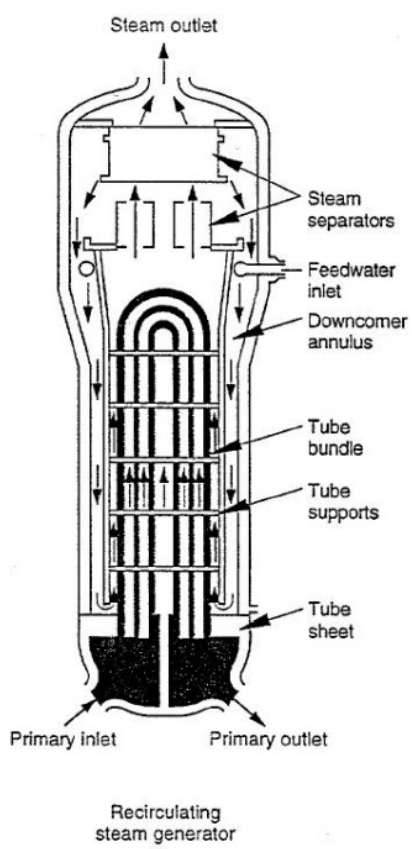
Columns 2 and 3 are the number of wear spots found, which includes multiple wear spots on individual tubes. Here “20%” refers to wear that has eroded 20% of the original thickness of the tube’s wall. Column 4 gives the total number of tubes that show at least one detectable wear spot. Column 5 refers to the percentage of the 9.350 tubes in each steam generator having detectable signs of wear.

✶Total exceeds the sum of the tubes with wear less than 20 percent and the tubes with wear greater than 20 percent because tubes can have multiple wear indications, some less than 20 percent and others greater than 20 percent.

Most of the tube damage within the Unit 3 replacement steam generators has been attributed to tube-to-tube wear. Hot water flowing inside the tubes and steam flowing outside them caused tubes to vibrate back and forth, bumping into adjacent tubes. These were not violent collisions like the *Titanic* ramming an iceberg but thousands upon thousands of small bumps.

The flow patterns inside the steam generators had been extensively analyzed using computer programs modeling the temperature, pressure, and flow conditions. In hindsight, it appears those computer simulations were not realistic.

Figure 7: Steam generator profile



As implied by its name, the function of a steam generator is to produce steam. Hot water from the reactor vessel enters the tubes via the primary inlet and exits through the primary outlet shown in Figure 7. Water is supplied to the area outside the tubes through the feedwater inlet. Baffles force this feedwater flow to first encounter the tubes at their lower regions. The feedwater is heated to the boiling point as it flows upward along the tubes.

The computer models consider the geometry of the steam generators and the properties (e.g., temperature, pressure, and velocity) of the water inside and outside the tubes to calculate things like where along the length of the tubes the feedwater reaches boiling and forms steam bubbles.

The tube-to-tube wear considered to have
ide the Unit 3 replacement steam generators
enon termed “fluid-elastic instability.” This
es forming earlier than predicted in the
ubbles, less dense than water, applied more
the portions in the U-shaped bend region.

Figure 8: Looking down at steam generator tubes showing problem area on primary inlet (hot) side

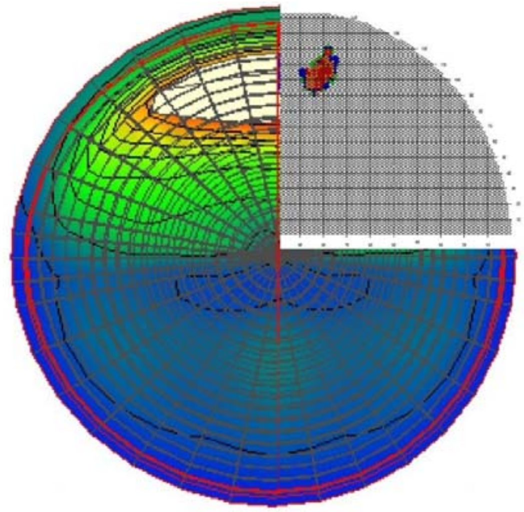


Figure 8 looks down at a cross-section of the steam generator and shows the location in the upper right quadrant where most of the tube-to-tube damage was found superimposed on the NRC's computer calculation of steam flow velocities. The bottom half has relatively low velocities because these tubes carry cooler water flowing towards the primary outlet. The upper half has the tubes filled with hotter water that produces more steam with higher velocities. The steam velocities towards the periphery are higher than the velocities in the center because the tubes in the center bend first and are therefore shorter than the tubes near the outside. The outer tubes have a longer heating length over which to produce steam.

Because the computer models originally used to analyze the replacement steam generator design failed to accurately predict steam and water flow conditions inside the replacement steam generators, they failed to accurately determine the forces causing tubes to vibrate and rub against each other. The role and placement of anti-vibration bars and retainers within the steam generator to handle the forces and dampen any resulting vibrations was consequently misunderstood. Once the damage was ascertained, the computer models were modified to generate output like Figure 8.

NRC Sanctions

None (yet).

UCS Perspective

As shown in Table 2, the Unit 3 replacement steam generators experienced more extensively damaged tubes than did the Unit 2 replacement steam generators. Yet, puzzlingly, the Unit 2 steam generators operated for a longer period (nearly a year longer) under virtually identical conditions. The theories on what caused the damage will not be valid until they are able to explain this disparity fully and honestly.

The NRC must not allow these reactors to restart until the cause of the steam generator degradation has a robust explanation rather than the flimsy, unsubstantiated one offered by the plant's owner to date.

Wolf Creek Generating Station, KS (first incident)

The Near-Miss

The NRC initiated a SIT investigation after one emergency diesel generator experienced load swings of up to 500 kilowatts when loaded to 5,800 kilowatts during a test run on September 1, 2011. The cause was determined to be an improper adjustment to the control system in May 2011 during an attempt to resolve another problem (NRC 2012s).

How the Event Unfolded

During a component design basis inspection conducted by the NRC in July 2007, the NRC team identified that the calculation for loads on the emergency diesel generators under accident conditions failed to account adequately for some design equipment loads. More than a year later (November 2009), the calculation was revised to account for all the design bases loads on the emergency diesel generators. But the test procedure for the emergency diesel generators was not revised and continued to test them at the lower (and thus non-conservative) loadings.

In August 2011, workers revised the emergency diesel generator test procedure to require loading to between 5,800 and 6,201 kilowatts, the maximum loading established by the revised design bases accident calculation.

Workers started Emergency Diesel Generator A on September 1, 2011, for its first test using the revised procedure. As the operator increased its loading to 5,800 kilowatts, an operator in the control room observed the emergency diesel generator's output to be oscillating up to 500 kilowatts. Operators declared Emergency Diesel Generator A inoperable.

When troubleshooting discovered that the load oscillations had been occurring since May 2011, the operators started Emergency Diesel Generator B on September 2, 2011, to see if it had the same problem. Its test was completed successfully without incident.

Workers found that during maintenance on Emergency Diesel Generator A on May 23, 2011, one of the gain settings in the control circuit was changed from 2.0 to 1.0. The vendor manual cautioned against reducing this gain setting below 1.5 to avoid load swings under high loading conditions. The gain setting was returned to 2.0 and Emergency Diesel Generator A successfully retested. It was returned to service on September 4, 2011.

NRC Sanctions

The SIT identified two violations of regulatory requirements associated with the ROP's *mitigating system* cornerstone:

- Failure to include essential information needed to adjust the emergency diesel generator control circuit following correct maintenance procedures.
- Failure to test the emergency diesel generators adequately; first by not revising the test procedure to account for increased design bases loadings, and second by not properly testing the emergency diesel generators following modifications to their control circuits.

The NRC classified both violations as Green (NRC 2012s).

UCS Perspective

The SIT's findings and classifications were appropriate.

Wolf Creek Generating Station, KS (second incident)

The Near-Miss

The NRC initiated an AIT investigation after two separate, unrelated electrical faults resulted in loss of the plant's normal sources of electricity. While both emergency diesel generators automatically started and supplied power to essential equipment, other equipment problems complicated the operators' response to the event (NRC 2012p).

How the Event Unfolded

The Wolf Creek nuclear plant was operating at 100 percent power on January 13, 2012. Electricity produced by the main generator went to the switchyard where transmission lines carried it out to the offsite electrical power grid.

At 2:02 pm, breaker 345-60 in the switchyard experienced an electrical fault. Protective systems reacted by automatically opening that breaker along with breakers 345-90, 345-120, 13-48, and 69-16 to isolate the 345,000-volt East Bus from the rest of the switchyard and in-plant equipment to prevent the fault from propagating.

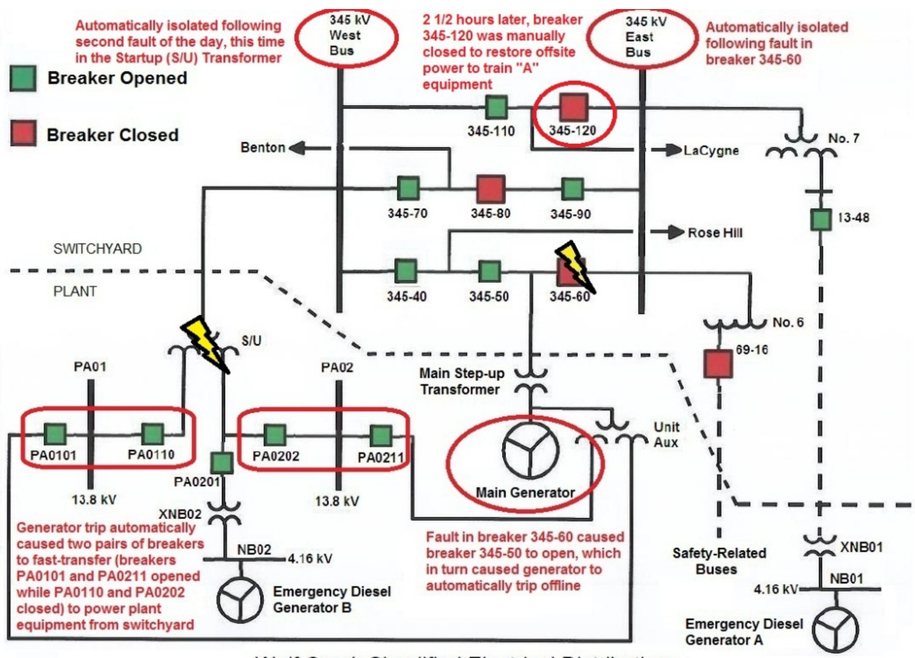
A second protective system reacted in parallel to close breaker 345-60 automatically, and also automatically shut down the main generator, which in turn triggered the automatic shut-down of the reactor.

The main generator's trip interrupted the normal supply of electricity to in-plant equipment. As a result, two pairs of electrical breakers automatically performed what is called a "fast transfer"—PA0101 opened as PA0110 closed while PA0211 opened as PA0202 closed. These transfers reconnected in-plant equipment to power supplies, now from the offsite grid through the 345,000-volt West Bus portion of the switchyard.

A second electrical fault intervened at this point. An electrical fault on the start-up transformer automatically caused breakers 345-100, 345-70, 345-40, PA0110, PA0201, and PA0202 to open to isolate 345,000 volt West Bus. With the main generator tripped and all connections to the offsite grid isolated, Wolf Creek experienced a total loss of offsite power.

Emergency diesel generators A and B automatically started and re-supplied power to essential plant equipment in about 8 seconds. Two and a half hours later, the operators manually closed breaker 345-120 to reconnect half of the plant's essential equipment with an offsite power source. Workers fully restored power to the plant four days later, on January 17.

Figure 9: Electrical distribution system at Wolf Creek



Wolf Creek Simplified Electrical Distribution

Numerous equipment issues complicated the operators' response to the event. The two major complications involved pumps for the fire-protection system and the power-operated relief valve (PORV). Before the event, the normal diesel-driven fire-protection water pump had been taken out of service due to a prior failure. Its backup, a temporary diesel-driven fire-protection water pump, lacked automatic start capability and had been drained of water to prevent it from freezing. Operators experienced difficulty priming the pump to prepare it for service. As a result, the plant lacked a working fire water supply system for about nine hours after the event. The AIT determined that management did not properly implement compensatory measures for the impairment of the fire protection system.

The loss of the normal power supply to in-plant equipment de-energized the instrument air system. Per design, air-operated valves in essential safety systems went to their fail-safe positions. This action stopped water flow through the reactor coolant system's letdown system and boosted flow through the reactor coolant system's charging system to maximum. Maximum makeup to the reactor vessel without any drainage flow pathways increased the pressure inside the vessel. The increasing pressure caused the power-operated relief valve (PORV) to open 23 times to discharge some cooling water from the vessel and protect it from over-pressurization damage. The concern here is that PORVs have a long history of opening and then failing to reclose, providing a pathway to drain cooling water from the reactor vessel. A stuck-open PORV was a major contributor to the March 1979 partial meltdown of the Three Mile Island Unit 2 reactor in Pennsylvania.

NRC Sanctions

The AIT identified no violations of regulatory requirements associated with this incident (NRC 2012p).

UCS Perspective

The AIT's conclusion seems appropriate. There's no reason to suspect that the two electrical faults should have been detected by the owner's testing and inspection regimes but were not.

Observations on the Near-Misses in 2012

Nuclear power plants are designed, built, and operated using a single-failure criterion—the single failure of a component or the single failure of a worker is not supposed to result in reactor core damage. The safety philosophy is intended to place as many highly reliable barriers as possible on the pathway to reactor core damage, to make it as unlikely as possible to reach that undesirable destination.

The most distressing aspect of the near-misses in 2012 was how many of them violated the single-failure criterion. At Catawba and Wolf Creek, electrical faults disconnected the plants from their offsite sources of electricity. In another near-miss at Wolf Creek, the onsite backup source of electricity was found to have been incapacitated by a "fix." At Brunswick, workers attempted to restart a reactor without properly fastening the head atop the reactor vessel. The loosely fastened head allowed cooling water to escape from the reactor vessel. At Harris, valves that were supposed to close within five seconds to terminate the release of radioactivity to the environment took as long as 4 hours and 7 minutes to close.

Safety equipment is tested and inspected often. Why did all these tests and inspections not find these problems sooner? At Harris, the answer is simple—workers had never tested the valves. It's impossible to flunk a test never taken. In most cases, however, the equipment was in fact being tested and inspected periodically. But the tests and inspections utterly failed to find problems. Consequently, more and more problems accumulated at the plants, waiting for an opportunity to conspire to overwhelm the single-failure criterion.

The NRC must make owners figure out why their testing and inspection regimes failed to find existing problems. Testing and inspection tasks are not performed just to keep workers occupied until it's time to go home. The essential tasks are supposed to ensure that safety equipment will function as intended to protect workers and the public. Too many of these near-misses revealed serious deficiencies in the testing and inspection regimes. Those deficiencies must be remedied.

CHAPTER 3. TRENDS FROM NEAR-MISSES 2010-2012

This chapter describes our analysis of the data from the nuclear reactor near-misses reported in our 2010, 2011, and 2012 reports.

As presented in Table 4, 56 near-misses were reported at 40 different reactors over this three year period. The number of reactors experiencing near-misses remained fairly constant year to year: 18 in 2010, 17 in 2011, and 16 in 2012.[6] Over this three-year period, nearly 40 percent of U.S. reactors experienced a near-miss.

That 56 near-misses occurred at 40 reactors means some reactors are repeat offenders. Table 4 shows that Wolf Creek tops the frequent offender list with four near-misses over three years. In fact, Wolf Creek experienced at least one near-miss each year.

The Palisades and Fort Calhoun reactors tied for second with three near-misses in three years.

From the glass half-full perspective, 64 of the nation's 104 reactors did not experience a near-miss between 2010 and 2012. If performance during this three-year period is representative of overall industry performance, however, then it may only be a matter of time before near-misses occur at those reactors as well.

The 2010-2012 data indicate the "average" reactor has a roughly one-in-six chance each year that it will experience a near-miss. With reactors originally licensed for 40 years and most being relicensed for an additional 20 years, that rate—if sustained—means the typical reactor could experience 7 near-misses over its 40-year lifetime and about 10 near-misses over 60 years.

While none of the 56 near-misses over the past three years caused harm to workers or the public, the "safety pyramid" provides ample reason to

[6] Numbering becomes cumbersome because nuclear plants can have multiple reactors, safety- and security-related events can affect one or all reactors at a plant, and some reactors experienced multiple events. Table 2 here and Table 4 later in the report attempt to clarify who had what near-miss.

reduce their occurrence. Introduced by H. W. Heinrich in his 1931 book *Industrial Accident Prevention*, the safety pyramid explains the relationship between the numbers of accidents and their severity levels.[7] As suggested by its name, the larger the base of minor accidents, the more often major accidents accidents will occur. By reducing the situations and behaviors that lead to near-misses, one reduces the number of minor accidents and serious accidents, too.

To reduce the number of near-misses, the NRC should include in its special inspection team (SIT) and augmented inspection team (AIT) processes a formal evaluation of the agency's baseline inspection effort. The baseline inspection effort covers the array of inspections conducted by the NRC at every nuclear plant in the country. When SITs and AITs report safety violations, the NRC should determine whether its baseline inspection effort could have, and should have, found the safety violations before they contributed to near-misses. The insights from the near-miss violations may enable the NRC to make adjustments in what its inspectors examine, how they examine it, and how often they examine it so as to become more likely to find violations, if they exist.

More than two decades ago, the NRC and the nuclear industry undertook parallel efforts aimed at reducing the number of scrams, or unplanned reactor shut-downs, that were occurring. Those efforts were very successful. In 1988, the average reactor experienced about 2.5 unplanned shut-downs annually (NRC 1993). By 2011, the last year data were reported, the typical reactor experienced 0.4 unplanned shut-downs annually (NRC 2012o). In other words, the typical reactor went more than two years between unplanned shut-downs.

With comparable attention to reducing the number of near-misses that are occurring, the NRC and the industry would likely achieve similar reductions. Or they can continue the status quo, hoping the plants reach the end of their operating licenses before their luck runs out.

[7] See http://emeetingplace.com/safetyblog/2008/07/22/the-accident-pyramid/ for additional details.

Table 4: Near-Misses 2010 to 2012				
	Total Number of Near Misses	Near Misses in 2010	Near Misses in 2011	Near Misses in 2012
Number of Reactors with Near Misses	56	19	19	18
Number of Unique Reactors	40	18	14	8
Arkansas Nuclear One Unit 1	1	1		
Arkansas Nuclear One Unit 2	1	1		
Braidwood Unit 1	2	1	1	
Braidwood Unit 2	2	1	1	
Brunswick Unit 1	1	1		
Brunswick Unit 2	2	1		1
Byron Unit 1	1		1	
Byron Unit 2	2		1	1
Callaway	1		1	
Calvert Cliffs Unit 1	1	1		
Calvert Cliffs Unit 2	1	1		
Catawba Unit 1	2	1		1
Catawba Unit 2	1	1		
Cooper	1		1	
Crystal River Unit 3	1	1		
Davis-Besse	1	1		
Diablo Canyon Unit 2	1	1		
Farley Unit 1	1			1
Farley Unit 2	2	1		1
Fort Calhoun	3	1		2
Harris	1			1
HB Robinson	2	2		
Millstone Unit 2	1		1	
North Anna Unit 1	1		1	
North Anna Unit 2	1		1	
Oconee Unit 1	1		1	
Oconee Unit 2	1		1	
Oconee Unit 3	1		1	
Palisades	3		2	1
Palo Verde Unit 1	1			1
Palo Verde Unit 2	1			1
Palo Verde Unit 3	1			1
Perry	2		1	1
Pilgrim	2		2	
River Bend	1			1
San Onofre Unit 2	1			1
San Onofre Unit 3	1			1
Surry Unit 1	1	1		
Turkey Point Unit 3	1		1	
Wolf Creek	4	1	1	2

"Unique Reactors" tracks the number of reactors experiencing near-misses. For example, Brunswick Unit 2 had a near-miss in 2010 and was counted among the unique reactors that year. When it experienced another near-miss in 2012, it was not counted as a unique reactor that year.

CHAPTER 4. POSITIVE OUTCOMES FROM NRC OVERSIGHT

This chapter describes situations in 2012 where the NRC acted to bolster the safety of nuclear plants. These positive outcomes are not necessarily the best the NRC achieved last year—we would have had to review and rate all NRC safety-related efforts to make that claim. Nor are these outcomes the only positive ones the NRC achieved last year—far from it. Instead, we chose situations with good outcomes that show that the NRC can be an effective regulator and provide insights into how the agency can emulate these outcomes more broadly and consistently.

Pro-active Efforts on Counterfeit, Fraudulent and Suspect Items

Nuclear power plants contain pipes, motors, valves, cables, relays, gauges, dampers, breakers, indicators and many other widgets and gadgets. Plant owners spend considerable effort each year refurbishing and replacing parts to prevent aging degradation from reducing reliability and safety levels. Their efforts are undermined when counterfeit, fraudulent and suspect items (CFSI) get used. CFSI may fail or malfunction and cause an accident. CFSI failures may also cause an accident worse than would otherwise occur.

The U.S. Department of Commerce published a study in January 2010 (Crawford *et al.*, 2010) on the electronics supply chain used by the U.S. Department of Defense. The study reported a 120 percent increase in electronic counterfeiting since 2005 and noted similar trends in the petroleum, automotive, and transportation industries.

The NRC's Office of the Inspector General (OIG) examined the CFSI issue during its 2010 audit of the NRC's inspection program of vendors who supply safety-related parts and services to the nuclear industry. The OIG's report (NRC 2010a) reinforced the Department of Commerce's findings. The OIG recommended that the NRC strengthen its approach to CFSI.

The NRC developed an action plan for CFSI (NRC 2011). Among other tasks, formal working groups representing all appropriate NRC offices were established on supply chain oversight, communication of best practices within government and industry, and protocols guiding responses to notifications of CFSI. The NRC actively participates in broader federal government efforts such as the Government-Industry Data Exchange

Program (GIDEP). GIDEP disseminates information about CFSI detections as widely and swiftly as possible to contain CFSI usage.

The NRC conducted several public meetings during 2012 as it implemented its CFSI action plan. For example, the NRC met with the Institute of Nuclear Power Operations (INPO) in August regarding the issue. The NRC explained the applicable regulatory requirements for CFSI and reviewed recent discoveries of CFSI in other industries (NRC 2012f).

Figure 10: Fraudulent fire extinguishers

The U.S. Department of Commerce and NRC's OIG identified problems with CFSI in non-nuclear industries. The NRC responded by initiating an array of measures intended to prevent the emerging CFSI threat from reducing safety levels at U.S. nuclear power plants even though CFSI had not been implicated in an accident or near-miss.

This positive outcome also reflects commendable communication and cooperation within both the federal government and the NRC itself. It is a good example of many people across many organizations rowing together towards a common goal.

Sustained Focus on Nuclear Security

In December 2012, the NRC hosted the inaugural International Regulators Conference on Nuclear Security. The conference featured keynote speeches by Yukiya Amano, Director General of the International Atomic Energy Agency, John O. Brennan, Assistant to the President of the United States for Homeland Security and Counterterrorism, and John Perren, Assistant Director of the Weapons of Mass Destruction Directorate at the Federal Bureau of Investigation; sessions covered several security topics with panels consisting of regulators from around the world.

It would be tempting to forego such effort more than a decade after the 9/11 tragedy and following the demise of Osama bin Laden.

It also would be tempting to stay focused on the safety lessons from the more recent Fukushima tragedy and let security concerns languish on the back burner. It is commendable that the NRC took this opportunity at this time to maintain its focus on nuclear security.

It is also commendable that the NRC conducted the conference in public. The NRC complemented the sessions with static displays, such as one showing how protective schemes for nuclear plants are evaluated in tabletop exercises. The speakers and panelists demonstrated that a sensitive topic like security could be discussed responsibly in public.[8]

Observations on Effective NRC Oversight

These examples could be labeled pro-active actions by the NRC. Or one could label them reactive because counterfeit parts plagued the nuclear industry in the late 1980s and 9/11 happened over a decade ago. Regardless of which label one chooses to apply, the NRC should be applauded for undertaking these forward-looking efforts without waiting for prompting from a problem in the U.S. nuclear industry.

Both efforts benefitted from addressing both aspects of NRC's regulatory process—establishing and enforcing regulations.

[8] Evidence of this commendable transparency can be found at http://www.nrcsecurityconference.org/ where security topics are described and associated presentations are posted.

CHAPTER 5. NEGATIVE OUTCOMES FROM NRC OVERSIGHT

This chapter describes situations where lack of effective oversight by the NRC led to negative outcomes. These outcomes are not necessarily the worst the NRC achieved last year. Rather, they shed light on practices and patterns that prevent the NRC from achieving the return it should from its oversight investment.

Safety Culture

In 2011, the NRC issued a policy statement on safety culture that stated "The Commission expects the members of the regulated community to take the necessary steps to promote a positive safety culture by fostering the nine traits outlined in the policy statement as those traits apply to their specific activities" (NRC 2012w). The NRC stated:

> *"Safety culture" refers to the core values and behaviors resulting from a collective commitment, by leaders and individuals, to emphasize safety over competing goals to ensure protection of people and the environment.*

The NRC identified nine traits—a trait being "a pattern of thinking, feeling, and behaving"—associated with a positive nuclear safety culture:

- **Leadership Safety Values and Actions**—Leaders demonstrate a commitment to safety in their decisions and behaviors.
- **Problem Identification and Resolution**—Issues potentially impacting safety are promptly identified, fully evaluated, and promptly addressed and corrected commensurate with their significance.

- **Personal Accountability**—All individuals take personal responsibility for safety.
- **Work Processes—**The process of planning and controlling work activities is implemented so that safety is maintained.
- **Continuous Learning**—Opportunities to learn about ways to ensure safety are sought out and implemented.
- **Environment for Raising Concerns**—A safety-conscious work environment is maintained where personnel feel free to raise safety concerns without fear of retaliation, intimidation, harassment, or discrimination.
- **Effective Safety Communication**—Communications maintain a focus on safety.
- **Respectful Work Environment—**Trust and respect permeate the organization.
- **Questioning Attitude—**Individuals avoid complacency and continuously challenge existing conditions and activities in order to identify discrepancies that might result in error or inappropriate action.

On November 8, 2012, the NRC Chairman and Commissioners heard the results of a safety culture survey of the NRC staff conducted by a consultant. This report was delivered during a Commission briefing closed to the public (NRC 2012e). That fact alone is revealing: organizations with strong, positive safety cultures do not need to discuss work force survey results secretly behind closed doors.

The survey results (Towers Watson 2012) reveal the depths of the NRC's safety culture problems. The NRC's Office of the Inspector General has retained consultants to conduct safety culture and climate surveys of the NRC work force every three years for over a decade. The surveys have reported generally consistent results with no dramatic improvement or degradation between surveys. In 2012, only 41 percent of the NRC's workforce responded that they believed that "significant actions have been taken as a result of the previous Safety Culture and Climate survey." Less than half of the NRC's work force believes the agency is taking the safety culture surveys seriously.

The consultant concluded that only 50 percent of the NRC's work force is fully engaged, defined by "Employees who have high engagement, energy and are enabled." Of the remaining half, 28 percent are de-energized, 13 percent are unsupported, and 9 percent are fully disengaged. The consultant reported that the "NRC is well below benchmarks on recognizing and respecting value of human differences."[9]

Most troubling are the results regarding NRC workers offering different viewpoints about safety within the agency. About half of the respondents indicated they had heard about co-workers who received negative reactions from their supervisors and senior managers after raising differing viewpoints. This is Safety Culture 101, and also touches on the traits expressed in bullets six, eight, and nine of the NRC's own nine points of a positive safety culture. A safety-conscious organization must establish and maintain an environment

[9] The consultant also surveys the work forces at other federal agencies and at high performance private companies to develop these benchmarks.

where all workers feel free to raise concerns without fear of harassment and retaliation. The surveys reveal that NRC lacks that environment. And the majority of the NRC work force believes the agency is not likely to take steps to rectify this sorry situation.

Figure 11: Results from Towers Watson survey of NRC work force

Category Breakdown Matrix

NRC OVERALL (N=2,981)

By Grade Level

A. NRC OVERALL (N=2,981)
B. GG-1 TO GG-10 (N=245)
C. GG-11 TO GG-12 (N=185)
D. GG-13 (N=686)
E. GG-14 (N=864)
F. GG-15 (N=764)
G. SENIOR LEVEL/ADMIN LAW JUDGE (N=25)
H. SES/SLS/EXECUTIVE LEVEL (N=185)

Values displayed are based on Total Favorable — Colored Cells indicate a statistically significant difference

#	Category	A	B	C	D	E	F	G	H
1	Clarity of Responsibilities	85	5	-3	0	-4	2	13	9
2	Communication	75	7	2	-2	-4	1	-2	10
3	Continuous Improvement Commitment	70	5	4	-2	-4	1	-3	13
4	Development	62	8	2	-3	-6	3	-2	19
5	DPO/Non-Concurrence	59	-6	-6	-4	-3	6	-9	23
6	Elevating Concerns	72	3	-2	-2	-5	3	-3	18
7	Empowerment	68	0	0	-3	-5	4	2	20
8	Engagement	78	3	2	0	-3	0	3	7
9	Management	74	8	2	-3	-6	4	-2	17
10	NRC Image	80	2	-1	0	-3	1	6	11
11	NRC Mission & Strategic Plan	83	3	2	-1	-3	1	-5	10
12	Office/Region Management	66	10	1	-4	-6	2	5	22
13	Open, Collaborative Working Environment	71	-1	-2	-3	-4	5	-3	19
14	Performance Management	66	8	2	-5	-4	3	-15	14
15	Quality Focus	63	1	-3	-3	-3	1	5	20
16	Senior Management	67	8	-2	-4	-5	3	-9	19
17	Supervision	77	4	3	-3	-4	2	8	15
18	Training	67	2	4	-1	-3	0	3	8
19	Working Relationships	80	2	0	-1	-3	2	-11	11
20	Workload and Support	73	4	1	0	-3	-1	14	13

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Column A shows the percentage of total NRC responses that were favorable with respect to the category. Columns B through H provide the relative responses by NRC pay-grade classification from lowest level in Column B through executives in Column H. For example, the average response from the 686 NRC respondents at pay-grade GG13 (Column D) on Category 3, Continuous Improvement Commitment, was 68. This response was 2 points below the overall NRC response of 70 (Column A). Red and green colored cells contain results with statistically significant differences from Column A. What is particularly striking is the fact that NRC executives (Column H) give significantly higher ratings than the average in all categories.

This slide from the consultant's presentation to the Commission likely explains why the problems are sustained across the three-year surveys—NRC senior managers do not believe that the problems exist. Where's the largest perception gap? NRC's senior managers rate the differing professional opinion (DPO)/non-concurrence processes 23 points higher than the overall rating. The DPO process[10] is process used by staffers to register disagreement with management decisions. The non-concurrence process is used by NRC staffers to register disagreement with statements and conclusions expressed

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[10] Within private industry, the DPO and non-concurrence processes are commonly handled through the Employee Concerns Program or the Ombudsman Program.

in NRC documents (e.g., safety evaluation reports and research studies). Perhaps the rating depends on which side of the negative reaction one is on.

Fire Non-Protection

On March 22, 1975, a worker using a lit candle to check for air leaks around penetrations through the walls of the room directly below the Units 1 and 2 control room at the Browns Ferry Nuclear Plant (in Athens, AL) ignited material used to seal those penetrations. The ensuing fire burned for nearly seven hours and damaged electrical cables for all the emergency core cooling systems on Unit 1 and most of those systems on Unit 2. Only heroic worker actions averted a double meltdown that day.

In 1980, the NRC adopted fire protection regulations intended to prevent another Browns Ferry fire, or worse.

In 2004, the NRC revised its fire protection regulations to provide an alternate means of managing the fire hazard risk. Owners had the option of complying with either the 1980 or the 2004 set of fire protection regulations.

On March 4, 2009, the Tennessee Valley Authority (TVA, Browns Ferry's owner) informed the NRC of its intent to transition to the 2004 regulations and committed to submitting its plan by March 4, 2012. If Browns Ferry met the 1980 regulations, there was no need for TVA to incur the cost of transitioning to the 2004 regulations. This letter was an implicit concession by TVA that Browns Ferry did not meet the 1980 fire protection regulations and it was more practical for TVA to strive to comply with the alternative fire protection regulations adopted in 2004.

On January 13, 2012, TVA requested an extension of over a year for submitting its compliance plan to the NRC. TVA requested a submittal date of March 29, 2013. The submittal date does not correspond to complying with fire protection regulations—the NRC's review and approval of the submitted plan can take over a year and the NRC then typically allows owners two more years to implement the approved plan.

On May 18, 2012, the NRC granted the request from Browns Ferry's owner for more time to submit its plan for achieving compliance with the 2004 fire protection regulations.

Thus, even if there are no further schedule slippages, the Browns Ferry nuclear reactors will not comply with fire protection regulations—adopted as a direct result of a very near-miss right there at Browns Ferry in March 1975—until March 2016 at the earliest.

For the NRC to allow this plant to operate for at least 35 years out of compliance with fire protection regulations is a clear and irrefutable sign of an ineffective regulator with a negative nuclear safety culture.

Temporary Storage of Spent Fuel

In 2010, the NRC updated its Waste Confidence Decision. The NRC originally issued the Waste Confidence Decision in 1984 in response to a 1979 decision by the United States Court of Appeals in a lawsuit initiated by the State of Minnesota. In that case, the NRC had approved the expansion of onsite spent fuel storage at nuclear plants in Minnesota. The State of Minnesota was concerned that the federal government's failure to make progress towards a permanent repository for spent fuel essentially made each nuclear plant site a de facto repository. The court ruled that the NRC must consider "whether there is reasonable assurance that an off-site storage solution will be available by … the expiration of the plants' operating licenses, and if not, whether there is reasonable assurance that the fuel can be stored safely at the sites beyond those dates" (DC Circuit 2012). The NRC's 1984 Waste Confidence Decision concluded that a federal repository would be available by 2009 and that spent fuel could be safety stored at plant sites for at least 30 years after the reactors permanently shut down.

The NRC revisited the Waste Confidence Decision in 1990, 1999, and 2008. The 2008 review led to the NRC updating the Waste Confidence Decision in 2010. In that update, the NRC concluded that a federal repository would be available "when necessary" instead of by a certain date and that spent fuel could be safely stored onsite for at least 60 years after the reactors permanently shut down.

The 2010 Waste Confidence Decision was legally challenged by several entities: the States of New York, New Jersey, Vermont, and Connecticut; the Prairie Island Indian Community in Minnesota, and a number of environmental organizations including the Natural Resources Defense Council. The petitioners contended that the NRC violated the National Environmental Policy Act of 1969 (NEPA) by failing to prepare an Environmental Impact Statement for its 1984 Waste Confidence Decision.

The NRC contended that NEPA was not applicable to its Waste Confidence Decisions because the decision did not license or relicense any nuclear facilities and therefore were not major federal actions.

The court agreed with the petitioners that the NRC had not fulfilled its obligations under NEPA for its 2010 Waste Confidence Decision. The court vacated the 2010 Waste Confidence Decision and remanded the matter back to the NRC.

The NRC suspended licensing and relicensing of nuclear power reactors until its resolves the Waste Confidence Decision issue (NRC 2012h).

The NRC has a blind spot when it comes to onsite spent fuel storage and NEPA. The United States Court of Appeals has twice found that the agency failed to comply with NEPA with its decisions on safe storage of spent fuel at nuclear plant sites. The NRC has dozens of lawyers on staff within its Office of General Counsel. Yet somehow all this in-house legal expertise repeatedly fails to have the NRC comply with federal statutes.

In addition to correcting shortcomings in its decisions for onsite spent fuel storage once again, the NRC needs to identify and correct the problems with its legal evaluations in this area. It would be prudent for the NRC to review other agency decisions to ascertain if these two errors are isolated or reflective of a programmatic failure contributing to other errors, too. The NRC requires such extent-of-condition assessments by plant owners; it should take the same medicine.

Recurring Reactor Cooling Water Leaks

Suppose you drive a car through a school or hospital zone with a posted 20 miles per hour speed limit and get pulled over by a law enforcement officer. That officer informs you that detection equipment indicated your speed had been 45 miles per hour. You cannot reasonably expect to avoid a speeding ticket by pointing out to the officer that your vehicle lacks a speedometer and you honestly had no clue that you were exceeding the posted speed limit. Unless it's an NRC cop who accepts ignorance about the condition of your car as a valid excuse.

As detailed above in Chapter 2, the operators shut down the reactor at the Palisades Nuclear Plant in Michigan on August 12, 2012, when the unidentified leak rate inside the reactor containment building increased to 0.3 gallons per minute.

After the reactor's shut down, workers determined the source of the leak to be a reactor coolant pressure boundary leak from one of the control rod drive mechanisms.

The plant's operating license requires the reactor to be shut down within six hours in such a situation. The plant's owner indicated the reactor had operated for nearly a month despite reactor coolant pressure boundary leakage (Entergy 2012).

Like other U.S. pressurized water reactors, Palisades has four limits in its operating license for reactor coolant leakage:

1) 150 gallons per day leakage through the tubes within any single steam generator,
2) 10 gallons per minute leakage through identified pathways,
3) 1 gallon per minute leakage through unidentified pathways, and
4) No leakage through the reactor coolant pressure boundary.

Of these leak types, reactor coolant pressure boundary leaks—the ones covered by item 4—pose the greatest threat to safety; hence, operating licenses permit no such leakage. Yet Palisades operated for nearly a month with leakage that safety regulations dictate only be tolerated for six hours. And Palisades is but the latest reactor in a long string of reactors operating far longer than six hours with reactor coolant pressure boundary leaks.

The problem is that the NRC acts as if the fourth limit does not exist. Instead, the NRC enforces only the first three limits. And while the NRC really does enforce those three, 75 percent is not a passing grade when the 25 percent missed is the most significant from a safety standpoint.

The reason the NRC ignores the fourth limit is because nuclear power plants lack instrumentation to monitor and detect reactor coolant pressure boundary leakage; in other words, nuclear plants are like automobiles without speedometers.

When leakage is detected by the available instrumentation, everyone assumes it must be coming from places other than the reactor coolant pressure boundary. Even when later inspection reveals that the leak was actually through the reactor coolant pressure boundary, the fourth limit is never invoked as long as the other three limits are met.

In March 2002, reactor coolant pressure boundary leakage over an estimated six-year period resulted in significant degradation of the reactor vessel head at the Davis-Besse nuclear plant in Oak Harbor, Ohio. A study conducted by the Oak Ridge National Laboratory for the NRC concluded that the damaged head would have failed in another 2 to 11 months of operation. The NRC's risk analysts concluded that this Davis-Besse incident was the closest to reactor core damage that any U.S. reactor came since the 1979 partial meltdown of the Unit 2 reactor at Three Mile Island.

And yet the NRC still refuses to enforce its regulatory requirements on reactor coolant pressure boundary leakage. And by not doing so, the NRC is enabling poor decision-making by plant owners.

The NRC has the authority to impose civil penalties of up to $140,000 per day per violation. In the Palisades case, the owner confessed to operating the reactor with increasing leakage from July 14, 2012, to August 12, 2012—28 full days. When the owner reported it was reactor coolant pressure boundary leakage, the NRC could have and should have imposed a $3.92 million fine for violating the operating license.

By taking safety regulations seriously instead of scoffing at safety violations, the NRC would send a clear message to plant owners and the public. Owners might just decide to install speedometers on their nuclear vehicles so as to know if leakage is through the reactor coolant pressure boundary. Or, they might opt to assume conservatively that leakage is from the worst area—the reactor coolant pressure boundary—and shut down to find and fix the leak instead of tempting fate, and assuredly incurring million-dollar fines, by continuing to operate.

The NRC's nonchalance on this safety requirement is indefensible. Left unchecked, it may someday contribute to America's next nuclear nightmare.

Nuclear Plant Flooding Hazards

On March 11, 2011, Fukushima Daiichi in Japan experienced three reactor meltdowns after tsunami waters flooded equipment used to distribute electrical power at the plant. On March 15, 2012, the NRC Chairman and Commissioners appeared at a hearing conducted by the Subcommittee on Clean Air and Nuclear Safety of the United States Senate Committee on Environment and Public Works. When asked by Chair Barbara Boxer if Fukushima could happen here, four of the five answered "no" (the NRC Chairman was not asked to respond and did not volunteer an answer.)

Yet the NRC issued a confirmatory action letter on June 22, 2010—nearly nine months before the Fukushima disaster—to the owner of the three reactors operating at the Oconee Nuclear Station in Seneca, South Carolina regarding flooding concerns (NRC 2010b). The NRC required the owner to implement measures intended to lessen the likelihood that flood water would not imperil the three reactors should the upstream Jocassee Dam fail. The NRC's risk analysts estimated a nearly 100 percent chance that all three reactors would be damaged if the dam failed, prompting the need for the upgrades. Whether its sea water from a tsunami or fresh water from a failed dam, the flooding threatens nuclear plant safety when essential equipment gets disabled.

The good news is that the NRC had identified the vulnerability of the three Oconee reactors—and nearly 30 other U.S. reactors—to external flooding risks prior to Fukushima and had initiated steps to manage those risks. Oconee's owner used generic dam failure rates in its assessment of the risk to Oconee from the Jocassee Dam. When the NRC's analysts used failures rates from dams of similar construction to Jocassee, the resulting risk was significantly higher—nearly 25 times greater.

The bad news is that the NRC hid this situation from the American public and misled the United States Senate about it. The NRC's 2010 letter to Oconee's owner was not made publicly available until late 2012. Many other

NRC documents about the hazard at Oconee and other nuclear plants were also inappropriately withheld from the public. And Senator Boxer certainly did not receive honest answers to her question during that Senate hearing on March 15, 2012, with the triple meltdown risk at Oconee known by the NRC but not yet resolved.

The NRC's creditability is jeopardized when it improperly withholds information[11] from the public and Congressional oversight committees. If explicit details about the Jocassee Dam's failure modes and associated vulnerabilities at Oconee warrant being withheld for national security considerations (i.e., not providing those who wish us harm the blueprints for conducting successful attacks), by all means do so. But a detail-lite version of the hazard could be made public to balance the public's right to know with the need to guard some information.

The NRC demonstrated achieving this balance in one area after 9/11. The NRC did not withhold all security information. Instead, it informed the public that it was taking steps to improve controls over access to nuclear plants and better protect against insiders and outsiders seeking to sabotage the plants. It quite properly withheld explicit information such as the height of security fences, locations of security cameras, and number of security force personnel at individual plants. But it publicly discussed the security threat and the general steps being taken to protect against it. And as described in Chapter 4, the NRC conducted an international conference on security that was open to the public, clearly demonstrating that it can discuss sensitive topics publicly while maintaining the proper balance of confidentiality.

Americans deserve comparable notification about flooding risks facing the nuclear plants.

Incomplete and Inaccurate Statements

An NRC regulation, specifically §50.9 in Title 10 of the Code of Federal Regulations, requires that information submitted to the NRC by plant owners "be complete and accurate in all material respects" (NRC 1987).

When the NRC staff reviewing applications for licensing action (e.g., permission to operate reactors at higher power levels and requests to reduce the frequency and scope of safety tests) by plant owners identifies additional information it needs to complete its evaluations, the NRC sends a request for additional information (RAI). Each RAI contains one or more questions that the NRC staff needs answered.

A search of the NRC's record-keeping system (called ADAMS for Agencywide Documents Access and Management System) for documents containing the phrase "request for additional information" authored by the NRC and sent to nuclear plant owners returned over 1,000 records just in 2012 alone.

The huge volume of RAIs during 2012—a number typical of prior years— clearly shows that the NRC staff has a questioning attitude. They literally asked thousands of questions of plant owners last year.

[11] The NRC classified the Jocassee Dam materials as Official Use Only, a classification with no legal basis and employed only to keep documents from the public.

But they are apparently not asking one key question—did the owner violate §50.9 by failing in the first place to submit information that was complete and accurate in all material respects?

The large number of RAIs submitted by the NRC staff constitutes *prima facie* evidence that violations may have occurred. But the NRC's RAI process does not include even a screening to evaluate formally whether a §50.9 violation is the reason for (or contributed to) the incomplete and/or inaccurate submittal prompting the need for the RAI.

Not every RAI represents absolute evidence of a §50.9 violation. Yet it is foolhardy to assume that no RAI could ever be the result of a §50.9 violation. But that seems to be the basic assumption behind the NRC's RAI process.

I know from personal experience that assumption is flawed. I worked as a consultant in the licensing departments at the Grand Gulf Nuclear Station (in Port Gibson, MS) and Hope Creek Generating Station (Hancocks Bridge, NJ). At Grand Gulf, the process for preparing documents being submitted to the NRC included speculating about any questions the NRC's reviewers might raise. This exercise was conducted so as to revise the draft to answer those potential questions. The objective was to submit material to the NRC that yielded no, or very few, questions from the agency.

The process at Hope Creek was fundamentally different. There, the process was not to volunteer any information in material being submitted to the NRC. "Make them ask," was the phrase I heard over and over from licensing supervisors in explaining why they had lined through statements and paragraphs in draft documents.

Consequently, an RAI to Grand Gulf was less likely to be a §50.9 violation and more likely to a question that honestly was not anticipated. Conversely, an RAI to Hope Creek might very well address material information that the owner had anticipated would be required but forced the agency to request.[12]

The NRC must take §50.9 seriously. When it issues RAIs to plant owners, the NRC must formally determine whether the reason for the RAIs might be §50.9 violations. That over 1,000 sets of RAIs were sent to plant owners during 2012 strongly suggests that some §50.9 violations were overlooked.

The NRC sanction plant owners that deliberately seek to avoid compliance. Such behavior is part and parcel of a regulator's job.

Observations on Ineffective NRC Oversight

It is laudable that the NRC wants plant owners to establish and maintain positive safety cultures at their nuclear plants. It is laughable that the NRC's own safety culture is so wanting.

The U.S. Congress played a key role in compelling the NRC to improve safety cultures at nuclear power plants. The 2002 discovery of severe reactor vessel head degradation at Davis-Besse was attributed to its owner placing production ahead of safety. The NRC appeared before an oversight subcommittee of the Senate's Environment and Public Works Committee

[12] I hasten to point out that I worked at Grand Gulf and Hope Creek years ago. Policies and practices could easily have changed at these plants since then. However, my more recent communications with colleagues working in licensing departments at U.S. reactors suggests that the "make them ask" approach is not yet extinct.

outlining the many steps it was taking in response to the Davis-Besse debacle. It did not propose doing anything directly about the stated root cause—namely, the owner having lost the proper safety focus. Senator George Voinovich, chair of the subcommittee and representing Ohio where Davis-Besse is located, gave the NRC an option: either address safety culture issues itself or the Senate would do so by legislation. It was an option having only one choice and the NRC made the right choice. The NRC revised its reactor oversight process to include safety culture elements.

It is imperative that the U.S. Congress compel the NRC to take steps to correct its safety culture problems and show marked improvement during the next work force survey in 2015.

The common thread among the remaining negative outcomes involves inadequate enforcement of federal regulations. In the Waste Confidence Decision example, the court vacated the NRC's 2010 Waste Confidence Decision after determining that the NRC failed to comply with provisions of the National Environmental Protection Act. The court's action provides assurance that the agency will comply. In the future, the NRC should comply on its own.

The NRC should emulate the court by making nuclear plant owners comply with federal regulations, too. Safety requirements prohibit reactors from operating for more than six hours with reactor coolant pressure boundary leaks; yet they do so again and again with NRC's tolerance. Federal regulations require plant owners to provide information to the NRC that is complete and accurate in all material respects. The NRC asked more than 1,000 sets of questions to plant owners just last year, strongly suggesting that the NRC is not getting complete and accurate information. Yet the NRC does not formally evaluate whether owners violated this federal regulation—and by not doing so, tolerates inadequate performance by plant owners.

The NRC's job is more than just establishing safety standards at appropriate levels. It also involves consistently enforcing them. From a public health perspective, the only thing worsethan having safety standards set improperly is having them set properly but not followed. Setting safety standards properly means one knows what it takes to protect public health. Failing to enforce them means one really doesn't care if the public is protected or not. That is unacceptable.

CHAPTER 6. SUMMARY AND RECOMMENDATIONS

Chapter 2 summarizes near-misses that the NRC reported at U.S. nuclear plants last year. As Chapter 3 shows, such near-misses have been occurring at a rate of over one per month over the past three years. Given enough chances, it is only a matter of time before near-misses become an actual hit. Public safety would be better served by reducing the frequency of near-misses. The NRC should take two steps to protect the public better:

1) Each SIT, AIT, and IIT should include a formal evaluation of the NRC's baseline inspection effort. The baseline inspection effort covers the array of routine inspections conducted by the NRC at every nuclear plant. When a SIT, AIT, or IIT identifies safety violations that contributed to the near-miss, the NRC's evaluation should determine whether the baseline inspection effort could have, and should have, found the safety violations sooner. Such insights from the near-misses may enable the NRC to make adjustments in what its inspectors examine, how they examine it, and how often they examine it to become more likely to find violations, if they exist.
2) Plant owners must be required to formally evaluate why their testing and inspection regimes failed to find longstanding problems. Many of the near-misses in Chapter 2 involved design and operational problems that existed for years, sometimes decades. The testing and inspection regimes are intended to find and fix such problems, but clearly failed to do so. Plants' programmatic weaknesses must be remedied to offer better protection against future near-misses.

Chapter 5 describes cases where the NRC has failed to enforce safety regulations. The quintessential example involves fire protection, or lack thereof, at Browns Ferry. Last year, the NRC granted its owner another year to submit its plan for complying with fire protection regulations—regulations adopted by the NRC in 1980 as a direct result of a very serious fire at that very plant.

Another enforcement failure involves a regulation against leaks of cooling water from reactors. This regulation prohibits reactors from operating longer than six hours with such leaks, yet the NRC did nothing about the Palisades reactor operating for nearly a month in violation of this key safety regulation. Sadly, this is far from an isolated case. The NRC has looked the other way many times in the past when many reactors violated that essential safety regulation.

And while the NRC has a regulation requiring that owners submit accurate and complete information to the agency, the NRC does not ascertain whether this regulation was violated when it must ask literally hundreds, if not thousands, of questions each year of owners about their submittals. This regulation carries far more than a "cross the t, dot the i" mandate. It requires owners to provide accurate and complete information to the NRC so the agency can make informed regulatory decisions regarding safety. When the NRC must ask owners questions about deficient information, one of the answers must address whether this federal regulation was violated.

In all cases, the NRC must enforce its regulations, using its ability to impose fines on owners and shut down reactors that violate safety regulations.

Chapter 5 describes a fundamental problem at the NRC with its own culture and climate about safety. As is painfully evident in Figure 11, there is a gaping difference between the views of NRC senior managers and the views of the NRC's work force on virtually everything. The NRC's senior managers believe conditions are far better than the rest of the agency believes. The first step in any 12-step program is to admit a problem exists. NRC's senior managers cannot fix problems they do not believe to exist. That is a serious problem. UCS recommends this solution:

- The NRC's safety culture problems, particularly the disparate views between management and work force, have to be rectified. Surveys of the NRC's safety culture are performed every three years. The next one is scheduled for 2015. Steps must be taken in order to reflect substantial improvement by the time of the 2015 survey.

If nothing substantive is done and the 2015 results resemble those in 2012, 2009, and 2006l—shame on lots of people. Shame on the NRC for advocating positive safety culture at the nation's nuclear plants but accepting safety culture shortcomings in its own house. Shame on the Congressional oversight committees for allowing the NRC to limp along with identified impairments. And shame on the President of the United States for not appointing persons to the Nuclear Regulatory Commission who would provide more than lip service to all the agency's safety culture posters and proclamations.

Half of the NRC work force has heard about co-workers who experienced negative reactions from their supervisor and senior managers after offering different views on safety issues. Similarly poor working environments put the Millstone (Millstone Nuclear Power Station, Waterford, CT) and Davis-Besse nuclear plants on the sidelines until proper working environments were restored. NRC needs a dose of that same medicine. If the nation's nuclear safety inspectors cannot do their jobs properly out of fear that their management will retaliate against them, Americans should be afraid. Very afraid.

When the NRC's safety culture shortcomings are fixed, we could expect better performance from this agency. Chapter 4 describes positive outcomes achieved by the NRC last year. With a proper safety culture, the NRC is more likely to achieve more positive outcomes like these and fewer negative outcomes described in Chapter 5.

It should not take a disaster at a U.S. nuclear power plant to undertake the necessary reforms at NRC. Undertaking those reforms now reduces the likelihood of a U.S. reactor becoming the next Fukushima.

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