The NRC and Nuclear Power Plant Safety in 2011

LIVING ON BORROWED TIME

Union of Concerned Scientists
Citizens and Scientists for Environmental Solutions

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I also thank John A. (Jack) Grobe, who retired from the Nuclear Regulatory Commission (NRC) in early 2012. I first met Jack when he chaired an NRC panel overseeing restart of the two reactors at the DC Cook plant in Michigan after prolonged outages from 1997 to 1999. At public meetings, Jack would greet me and other stakeholders by asking, "How are we doing?" instead of the more common, "How are you doing?" It was not a gimmick: Jack sincerely wanted feedback on the NRC's performance. While he preferred positive comments, he was equally receptive to negative ones.

Jack reprised that role when he chaired an NRC panel overseeing restart of the Davis-Besse reactor in Ohio after a prolonged outage from 2002 to 2004. Jack then relocated from the NRC's Region III office in Lisle, IL, to the agency's headquarters outside Washington, DC. One of his last assignments was to serve on an NRC task force to evaluate lessons learned from last year's Fukushima disaster.

When Jack emailed me last fall about his pending retirement, he did not inquire, "How have I done?" Had he asked, I would have answered, "Extremely well." Jack is among many NRC employees who have earned my respect over the years. They give me reason to believe that the agency is working to fulfill its pledge to protect people and the environment, and that it can become even better at that role.

EXECUTIVE SUMMARY

This report is the second in an annual series on the safety-related performance of owners of U.S. nuclear power plants, and the Nuclear Regulatory Commission (NRC), which regulates them. The NRC's mission is to protect the public from the inherent hazards of nuclear power.

In 2011, the NRC reported on 15 special inspections it launched in response to problems with safety equipment, security shortcomings, and other troubling events at nuclear power plants. "The NRC and Nuclear Power Plant Safety 2011: Living on Borrowed Time" provides an overview of each of these significant events—or near-misses.

This overview shows that many of these events occurred because reactor owners either tolerated known safety problems or took inadequate measures to correct them. For example, the owner of the Oconee nuclear plant in South Carolina installed a backup reactor core cooling system in 1983. However, in 2011—more than a quarter-century later—workers discovered a problem with the system that would have rendered it useless in an accident. (Oconee is a twin to Three Mile Island, which in 1979 experienced a partial meltdown of its Unit 2 reactor core stemming from inadequate cooling.)

Oconee's backup system included electrical breakers located inside the reactor containment building. These breakers are equipped with devices that open if the temperature of the breakers gets too high, protecting them from damage by overheating. However, plant employees had set those devices to activate at temperatures lower than those that would occur inside the containment building during an accident—meaning that the safety system would not protect the reactor core from overheating.

Another significant safety-related event in 2011 occurred at the Braidwood and Byron nuclear plants in Illinois. Workers at those plants had instituted a practice in 1993 of deliberately draining water from the piping to a vital safety system. They did so to reduce corrosion caused by the drawing of untreated lake water into the system. However, their solution would have prevented this vital safety system from functioning properly during an accident.

This report also provides examples where onsite NRC inspectors made outstanding catches of safety problems at the Fort Calhoun, Hatch, and LaSalle nuclear plants—before these impairments led to events that required special inspections, or to major accidents. At the LaSalle plant in Illinois, NRC inspectors challenged operators' practice of leaving a

test tank partially filled with water. Safety studies of what would happen at the plant during an earthquake assumed that this tank would be empty, because workers were supposed to fill it with water only during infrequent testing. NRC inspectors questioned whether the weight of the water in the tank could cause it to collapse or topple during an earthquake. After an analysis confirmed their suspicions, workers at LaSalle drained the tank and revised procedures to keep it empty except during testing. The NRC's catch at LaSalle led workers at the Duane Arnold plant in Iowa to discover and correct the same problem.

However, the NRC did not always serve the public well in 2011. For example, NRC inspectors identified numerous problems with reactor safety components and operating procedures during Component Design Bases Inspections (CDBIs). Inspectors are supposed to use CDBIs to determine whether owners are operating and maintaining their reactors within specifications approved during design and licensing. Some of the problems concerned containment vent valves, battery power sources, and emergency diesel generators—components that affected the severity of the disaster at the Fukushima Dai-Ichi nuclear plant in Japan.

While it was good that the NRC identified these problems, each CDBI audits only a very small sample of possible trouble spots. For example, the CDBI at the Harris nuclear plant in North Carolina examined just 31 safety-related items among literally thousands of candidates. That audit found 10 problems. Beyond ensuring that the plant's owner corrected those 10 problems, the NRC should have insisted that it identify and correct inadequacies in the plant's testing and inspection regimes that allowed these problems to exist undetected in the first place. The true value of the CDBIs stems from the weaknesses they reveal in the owners' testing and inspection regimes. But that value is realized only when the NRC forces owners to remedy those weaknesses.

We found that the NRC is allowing 47 reactors to operate despite known violations of fire-protection regulations dating back to 1980. The NRC is also allowing 27 reactors to operate even though their safety systems are not designed to protect them from earthquake-related hazards identified in 1996. Eight reactors suffer from both afflictions. The NRC established safety regulations to protect Americans from the inherent hazards of nuclear power plants. However, it is simply not fulfilling its mandate when it allows numerous plant owners to violate safety regulations for long periods of time.

These and other positive and negative examples of NRC actions do not represent the agency's best and worst performances in 2011: we did not review all the efforts by inspectors last year. Instead, the examples highlight patterns of NRC behavior in which inspectors uncovered serious safety problems and forced owners to correct them—while at other times allowing dangerous situations to persist.

The positive examples clearly show that the NRC can be an effective regulator. The negative examples attest that the agency still has work to do to become the regulator that the public deserves.

Overall, our analysis of NRC oversight of safety-related events and practices at U.S. nuclear power plants in 2011 suggests these conclusions:

- Nuclear power plants continue to experience problems with safety-related equipment and worker errors that increase the risk of damage to the reactor core—and thus serious harm to employees and the public.

- Recognized but misdiagnosed or unresolved safety problems often cause significant safety-related events at nuclear power plants, or increase their severity.

- When NRC inspectors discover a broken device, an erroneous test result, or a maintenance activity that does not adhere to established procedure, they all too often focus just on that problem, not its underlying cause. Every such finding should trigger an evaluation of why the owner failed to find and fix the problem *before* NRC inspectors discovered it.

- The NRC can better serve the U.S. public, and plant owners, by consistently enforcing its own safety regulations.

- Four of the special inspections occurred at plants owned by Entergy. While the company may simply have had an unlucky year, corporate-wide approaches to safety may have contributed to this poor performance. When conditions trigger special inspections at more than one plant with the same owner, the NRC should formally evaluate whether corporate policies and practices contributed to the shortcomings.

The chances of a disaster at a nuclear power plant are low. When the NRC finds safety problems and ensures that owners address them—as onsite inspectors did last year at Fort Calhoun, Hatch, and LaSalle—it keeps risks to workers and the public as low as practical. But when the NRC tolerates unresolved safety problems—as it still does at dozens of reactors violating fire-protection and earthquake-related regulations—this lax oversight allows that risk to rise. The more owners ignore such safety problems, the higher the risk climbs.

While none of the safety problems in 2011 caused harm to plant employees or the public, their frequency—more than one per month—is high for a mature industry. The severe accidents at Fermi (a plant in Michigan that suffered a partial core meltdown) in 1966, Three Mile Island in 1979, Chernobyl in 1986, and Fukushima Dai-Ichi in 2011 occurred when a handful of known but uncorrected problems led to catastrophes. That plant owners could have avoided nearly all the near-misses in 2011 had they corrected known deficiencies in a timely manner suggests that neither the owners nor the NRC has completely internalized the lessons from those accidents.

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, parking 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 for even minor violations.

The Union of Concerned Scientists (UCS) has evaluated safety at nuclear power plants for nearly 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.

With study after study showing that the NRC has the regulations it needs but fails to enforce them, UCS decided that an annual report chronicling only the latest examples of lax enforcement would be futile. Instead, this report—like its predecessor last year—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

When a safety-related event occurs at a reactor, or workers or NRC inspectors discover a degraded condition, the NRC evaluates whether the chance of damage to the reactor core has risen (NRC 2001). If the event or condition has not affected that risk—or if the risk has increased only incrementally—the NRC relies on its reactor oversight process (ROP) to respond.

The ROP features seven cornerstones of reactor safety (Table 1). In this process, the NRC's fulltime inspectors monitor operations and procedures at nuclear plants, attempting to detect problems before they lead to more serious violations and events. The NRC issued nearly 200 reports on such problems last year alone.

Most safety-related incidents and discoveries at nuclear power plants are low risk. However, when an event or condition increases the chance of reactor core damage by a factor of 10, 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).

The teams go to the sites to investigate what happened, why it happened, and any safety implications for other nuclear plants. These 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 those of the special teams identify violations of NRC regulations. The NRC classifies these violations in five categories, with Red denoting the most serious, followed by Yellow, White, Green, and Non-Cited Violations. For certain violations that do not lend themselves to classification by their risk significance, the NRC uses four severity levels, with level I the most serious.

THE FOCUS OF THIS REPORT

Chapter 2 summarizes "near-misses" at nuclear reactors that the NRC reported on in 2011: 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 operators 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 review of all near-misses in 2011 provides important insights into trends in nuclear safety, as well as the effectiveness of the NRC's oversight process. For example, if many near-misses stemmed from 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 these near-misses attesting to why day-to-day enforcement of regulations is vital to the safety of nuclear power, the next two chapters highlight NRC performance in monitoring safety through the reactor oversight process. Chapter 3 describes occasions in which effective oversight by NRC inspectors produced positive outcomes. That is, these inspectors took action to prevent safety problems from snowballing into even more dangerous situations, such as the near-misses noted in Chapter 2. Chapter 4 then describes cases where ineffective NRC oversight failed to prevent dangerous situations—or actually set the stage for them.[1]

Chapter 5 summarizes findings from the near-misses in Chapter 2, the examples of positive outcomes from the reactor oversight process in Chapter 3, and the examples of negative outcomes from that process in Chapter 4.

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

This chapter notes which oversight and enforcement strategies worked well for the NRC in 2011 and which did not. This chapter also recommends steps the NRC should take to reinforce behavior among plant operators 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 this and ensuing annual reports is to spur the NRC to improve its own performance as well as that of reactor owners and operators. Future reports will highlight steps the agency took to reinforce effective oversight and eliminate lax enforcement, and to ensure that plant owners comply with NRC safety regulations. We did not include such a discussion in this report for two primary reasons: two of our findings do not support trend analysis, and the NRC properly focused on responding to the Fukushima disaster rather than assessing the recommendations in our 2010 report for potential implementation.

Table 1. Seven Cornerstones of the 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 respond to a loss of offsite power; inadequate repair of a problem with a pump in the emergency core cooling system, reducing the reliability of cooling during an accident; and non-conservative calibration of an automatic 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

Table 1. Seven Cornerstones of the Reactor Oversight Process	
	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 properly survey an area 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 2011

In 2011, the NRC reported on 15 significant safety- and security-related events at nuclear reactors that prompted the agency to send special investigative teams to analyze problems at those reactors (Table 2). Fourteen of these events triggered an SIT, one triggered an AIT, and none triggered an IIT. (The Wolf Creek event actually occurred in 2010, but the NRC issued its report on it in 2011.)

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 the 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.

In 2011 the NRC also conducted two sets of special inspections in response to the Fukushima disaster in Japan. The NRC inspected all operating nuclear power plants for their readiness to withstand severe acts of nature (NRC 2011x). The NRC also inspected all plants to obtain information on procedures voluntarily developed by their owners to respond to severe accidents (NRC 2011u). Because both sets of inspections were (hopefully) one-time examinations linked to the Fukushima accident, we excluded them from this report.

Table 2: Nuclear Near-Misses in 2011

Reactor and Location	Owner	Highlights
Braidwood Joliet, IL	Exelon	SIT: After NRC inspectors questioned the practice of draining water from portions of the essential service water piping to the auxiliary feedwater pumps (to avoid corrosion damage from untreated water leaking past isolation valves), analysis revealed that this key emergency system might not function during an accident. The NRC team also discovered that workers failed to declare an emergency in response

Reactor and Location	Owner	Highlights
		to the recurring failure of all control room alarms.
Byron Rockford, IL	Exelon	SIT: After NRC inspectors questioned the practice of draining water from portions of the essential service water piping to the auxiliary feedwater pumps (to avoid corrosion damage from untreated water leaking past isolation valves), analysis revealed this key emergency system might not function during an accident.
Callaway Jefferson City, MO	Union Electric Co.	SIT: Routine testing of an emergency pump intended to prove that it was capable of performing its safety functions during an accident actually degraded the pump. The pump's manufacturer recommended against running the pump at low speeds, but this recommendation was ignored during the tests.
Cooper Nebraska City, NE	Nebraska Public Power District	SIT: Workers replacing detectors used to monitor the reactor core during low-power conditions were exposed to high levels of radiation when they deviated from the prescribed procedure.
Millstone Unit 2 Waterford, CT	Dominion	SIT: Despite a dry run of an infrequently performed test on the control room simulator and other precautionary measures, errors during the actual test produced an unexpected and uncontrolled increase in the reactor's power level.
Monticello Minneapolis, MN	Nuclear Management Co.	SIT: During a periodic test of the fire sprinkler system, workers found that rust particles inside the system's piping blocked the flow of water past a valve. The NRC determined that the plant owner had not properly evaluated numerous warnings about corrosion inside fire protection

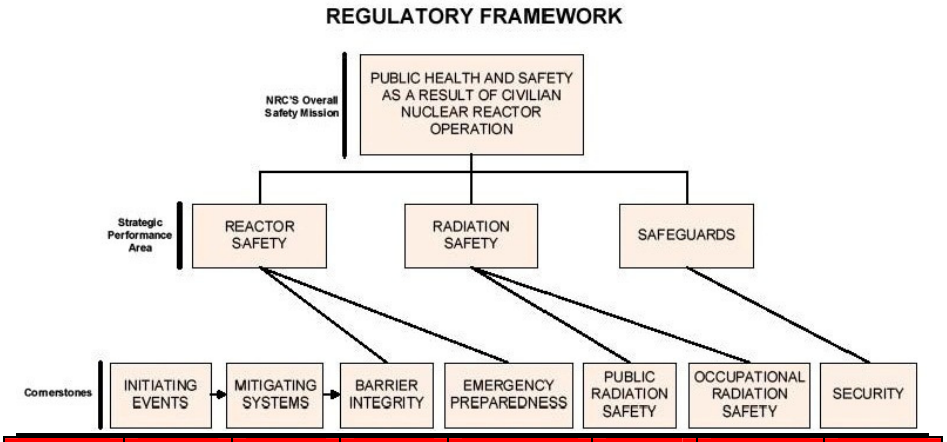
Reactor and Location	Owner	Highlights
		piping at other nuclear plants, and had not incorporated the information into maintenance practices.
North Anna Richmond, VA	Dominion	AIT: An earthquake of greater magnitude than the plant was designed to withstand caused both reactors to automatically shut down from full power.
Oconee Greenville, SC	Duke Energy	SIT: Workers discovered that an emergency system installed in 1983 to protect the reactor core from overheating in the event of a station blackout, pipe break, fire, or flood would be disabled by the high temperature inside the containment during such an accident. The high temperature would cause electrical components within the emergency system to fail.
Palisades South Haven, MI	Entergy	SIT: When a pump used to provide cooling water to emergency equipment failed in September 2009 because of stress corrosion cracking of recently installed parts, workers replaced the parts with identical parts. The replacement parts failed again in 2011, disabling one of three pumps.
Palisades South Haven, MI	Entergy	SIT: Workers troubleshooting faulty indicator lights showing the position of the emergency airlock door inadvertently shut off power to roughly half the instruments and controls in the main control room. The loss of control power triggered the automatic shutdown of the reactor and complicated operators' response.
Perry Cleveland, OH	FirstEnergy	SIT: Problems during the replacement of a detector used to monitor the reactor core during low-power conditions exposed workers to potentially high levels of radiation.

Reactor and Location	Owner	Highlights
Pilgrim Plymouth, MA	Entergy	SIT: Security problems prompted the NRC to conduct a special inspection. Details of the problems, their causes, and their fixes are not publicly available.
Pilgrim Plymouth, MA	Entergy	SIT: When restarting the reactor after a refueling outage, workers overreacted to indications that the water inside the reactor was heating up too rapidly, and lost control of the reactor. The plant's safety systems automatically kicked in to shut down the reactor.
Turkey Point Unit 3 Miami, FL	Florida Power and Light Co.	SIT: A valve failure stopped the flow of cooling water to equipment, including the reactor coolant pump motors and the cooling system for the spent fuel pool.
Wolf Creek Burlington, KS	Wolf Creek Nuclear Operating Co.	SIT: Workers overlooked numerous signs that gas had leaked into the piping of safety systems, impairing the performance of pumps and flow-control valves.

In 2011, reports from the SITs and AIT dispatched by the NRC identified 39 violations of NRC safety regulations. Figure 1 classifies these violations by the seven cornerstones of the reactor oversight process (ROP).[2]

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

Figure 1: Near-Misses in 2011 by Cornerstones of the Reactor Oversight Process



Red	0	0	0	0	0	0	0
Yellow	1	1	0	0	0	0	0
White	2	0	0	0	0	1	1
Green	2	23	0	1	0	6	0
IV	0	1	0	0	0	0	0
Total	5	25	0	1	0	7	1

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

The NRC special investigative teams did not classify any safety violations as Red—the most serious—in 2011. The teams did classify two safety violations as Yellow. One occurred at the Oconee plant in South Carolina, while the other occurred at the Palisades plant in Michigan.

BRAIDWOOD AND BYRON, IL

The Near-Miss

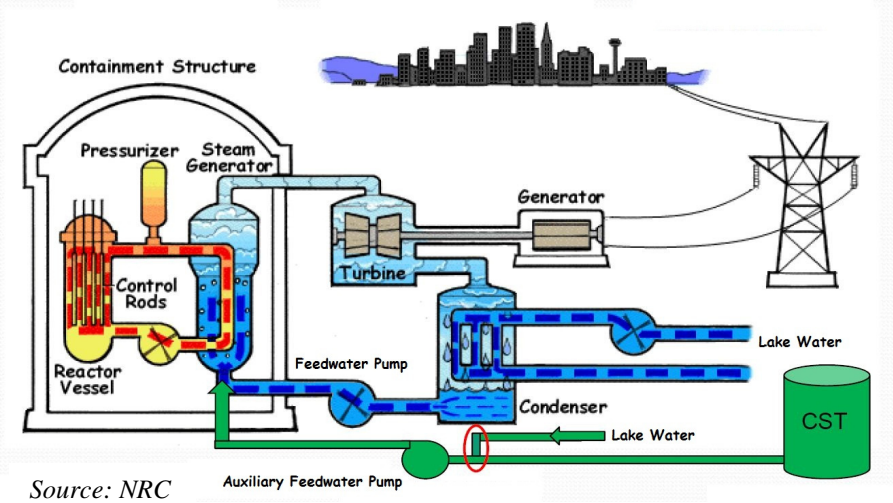
The NRC sent an SIT to these sites after NRC inspectors discovered that workers were intentionally draining water from piping for an emergency cooling system (NRC 2010b).

The SIT determined that an evaluation performed by the company in 1993 to justify the practice was inadequate. After an outside consultant concluded that the practice could prevent the emergency system from functioning during an accident, the owner modified the plants and procedures to keep the piping filled with water. The NRC sanctioned the owner for inadequate evaluation of plant configurations. The NRC also sanctioned the owner for failing to properly install the control room alarm system at Braidwood Unit 2, and for failing to declare an emergency on March 24, 2011, when that alarm system failed.

How the Event Unfolded

On January 31, 2011, NRC inspectors at Byron questioned the longstanding practice of maintaining portions of piping for the essential service water supply to auxiliary feedwater pumps empty of water. The NRC raised the same issue at the Braidwood plant because it had a similar design and operating procedures (Figure 2).

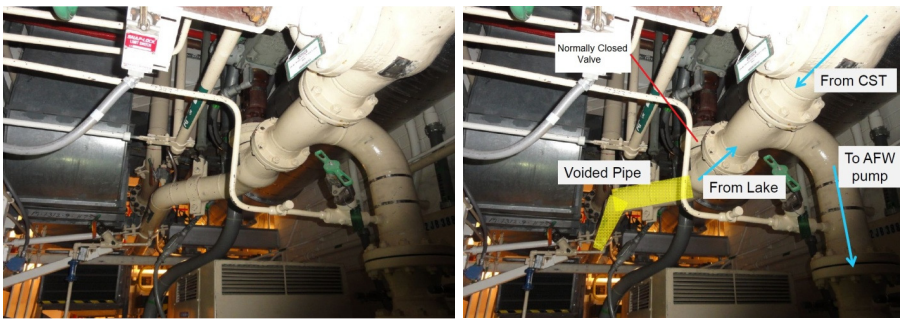
Figure 2. Outline of a Pressurized Water Reactor Such as Byron and Braidwood



Source: NRC

At Byron and Braidwood, the auxiliary feedwater pumps are normally in standby mode, ready to provide makeup water to the steam generators when the feedwater pumps cannot do so. Makeup water absorbs heat from water circulating between the reactor vessel and steam generators. This vital process removes heat produced by the reactor core, preventing damage to the core caused by overheating.

The auxiliary feedwater pumps usually transfer relatively clean, treated water from the condensate storage tank (CST) to the steam generators. The auxiliary feedwater pumps can also transfer untreated water from a nearby lake when the CST empties or cannot be used.



Source: NRC 2011s.

At Byron and Braidwood, this piping supplies water to auxiliary feedwater pumps needed during emergencies to prevent heat from damaging the reactor core. (The two photos are identical, except for the labels on the right-hand photo.) The yellow shading highlights piping drawing water from a nearby lake—the focus of an NRC special inspection.

The longstanding practice at Byron and Braidwood was to keep the pipes from the lake emptied of water, to prevent any untreated lake water from leaking past a closed valve and entering the steam generators, where it could hasten corrosion and rusting. The NRC inspectors questioned whether the volume of air in the empty portion of the piping would impair or disable the auxiliary feedwater pumps if they needed to draw water from the lake.

The plant owner initially referred the NRC team to a letter sent by the Byron engineering department in 1993 to station managers at Byron and Braidwood. Without any formal supporting evaluation, this letter simply stated that the auxiliary feedwater pumps would not be adversely affected by the ingestion of air if the system switched to draw water from the lake rather than the CST.

The NRC challenged this unsupported evaluation. The owner then asked the vendor of the auxiliary feedwater pumps to evaluate their performance under the air-and-water scenario. The vendor concluded that the configuration did not meet industry criteria, and thus could not be assured of working when needed.

In response, the company installed vent valves on the piping at Braidwood to allow air to vent as the piping filled with water. Vent valves already existed on this piping at Byron. Operators also revised procedures at both plants to keep these sections of piping filled with water instead of air.

During this NRC inquiry, maintenance on the annunciators, or alarms, in the Unit 2 control room at Braidwood on March 24, 2011, exposed an unrelated problem. The alarms had not been wired properly when installed. A single failure disabled all the control room alarms on Unit 2 that day, and a similar failure would have done so on Unit 1. Workers failed to respond properly to the alarm problem on Unit 2. The plant's emergency response procedures required operators to declare an Unusual Event—the least serious of the NRC's four emergency classifications—when 75 percent or more of control room alarms are disabled for 15 minutes or longer. Although this condition existed on March 24, operators failed to declare an Unusual Event (NRC 2011p).

NRC Sanctions

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

- Failure to adequately justify the continued operability of the auxiliary feedwater system, given that some of its supply piping remained emptied of water.
- Failure to properly install control room alarm systems.

The NRC classified both violations as Green—the least serious of the color-coded violations.

The SIT also identified a violation of requirements associated with the *emergency preparedness* cornerstone:

- Failure to declare an emergency when conditions for that declaration existed.

The NRC classified this violation as Green.

CALLAWAY, MO

The Near-Miss

The NRC sent an SIT to the site after workers found that the lubricating oil for a bearing on the turbine-drive auxiliary feedwater pump was degraded, potentially preventing the pump from working during an accident. The SIT found that periodic testing used to determine whether this emergency pump was functioning properly was actually damaging it. The NRC sanctioned the company for poor maintenance and testing practices (NRC 2011o).

How the Event Unfolded

On February 8, maintenance workers drained a sample of oil from the reservoir for a bearing on the turbine-driven auxiliary feedwater pump. They attributed the oil's very dark coloration to degradation, and initially attributed that to an insufficient oil level in the reservoir. This was the third consecutive time that workers had found degraded oil: they had also done so in September 2009 and May 2010.

Operators sent an oil sample to an offsite laboratory for further evaluation, which revealed high levels of iron, copper, aluminum, lead, and zinc. These results were inconsistent with the initial determination, and prompted workers to probe for the real cause.

The turbine-driven auxiliary feedwater pump is normally in standby mode, ready to provide makeup water to steam generators when the feedwater pumps cannot do so, such as during a station blackout, when no AC electrical power would be available for the motor-driven feedwater pumps. As noted, makeup water absorbs heat from water circulating between the reactor vessel and steam generators, preventing damage to the reactor core caused by overheating. Under accident conditions, steam for the turbine-driven auxiliary feedwater pump comes from steam generators B and C. The pump, in turn, supplies 1,145 gallons of water per minute to the steam generators.

During a test, steam for the pump's turbine comes from either the steam generators or—when the plant is shut down and not producing steam—the oil-fired auxiliary boiler. When the auxiliary boiler is used, its steam pressure cannot operate the pump at its design speed of 3,850 revolutions per minute, so workers tested it at lower speeds and lower flows. However, the vendor manual for the pump cautioned against operating it at low speeds, as doing so could damage its internal mechanisms, including the bearing.

Plant workers did not incorporate this guidance into the testing procedures, which allowed the pump to run at low speeds for long periods. NRC inspectors also discovered that the pump had run at even lower speeds during recent tests. Testing to determine whether the pump was working properly was therefore damaging it—just as the vendor manual suggested. The practice of sampling the lubricating oil every 18 months inadequate to guard against degradation was also deemed inadequate: workers must now sample the oil every 9 months.

On March 18, 2011, the plant owner evaluated the impact of a degraded bearing on the turbine-driven auxiliary feedwater pump. The evaluators found that the degradation would not have prevented the pump from performing all safety functions during an accident lasting 100 minutes.

However, the NRC questioned the mission time assumed for the pump, since it accounted only for the duration of the accident, and not for the associated recovery time of four hours. A second evaluation using the correct mission time concluded that the pump would not fulfill its safety function.

The SIT also identified a different maintenance problem with the turbine-driven auxiliary feedwater pump. Employees had scheduled the trip and throttle valve—which controls the amount of steam admitted to the pump's turbine—for overhaul during a refueling outage in 2008. But that work had been eliminated without adequate justification. In a 2009 test, the pump failed to start because its trip and throttle valve failed to open (NRC 2011o).

NRC Sanctions

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

- Failure to incorporate precautions and limitations from the vendor manual into testing procedures for the turbine-driven auxiliary feedwater pump, leading to its failure.
- Failure to follow established procedures during testing of the pump, contributing to its failure.
- Failure to properly evaluate the potential impact of a degraded bearing on the pump over its entire post-accident mission time.
- Failure to properly determine allowable leakage for emergency equipment because of the use of non-conservative mission time.
- Failure to properly perform preventive maintenance on the lubricating oil for the turbine-driven auxiliary feedwater pump.
- Failure to properly take corrective action, causing the pump to fail to start.

The NRC classified all six violations as Green.

COOPER, NE

The Near-Miss

The NRC sent an SIT to the plant after workers deviated from prescribed procedures for replacing two detectors used to monitor the power level in the reactor core, and had to abandon the job because of very high radiation levels.

The SIT identified six violations of regulatory requirements associated with poor planning and execution of the maintenance activity (NRC 2011m).

How the Event Unfolded

On April 2, 2011, workers entered the area beneath the reactor vessel while the plant was shut down during a refueling outage. The workers intended to replace source range monitor (SRM) B and intermediate range monitor (IRM) C, which monitor the power level in the reactor core when it is shut down or operating at low (less than 10 percent) power.

The procedure guiding this task called for the workers to disconnect the detectors from below, so that other workers could remove them from the top

of the reactor vessel. The workers beneath the vessel also wanted to install a nose cone, to prevent water from leaking as the detectors were removed.

These employees attended numerous briefings immediately before beginning the work. Most of these briefings focused on removing the detectors from the top of the vessel, per written instructions. Participants in one briefing did discuss removing the detectors from below, but the written instructions did not change.

Shortly before beginning the task, workers found two different nose cones, only one of which would work. Because the workers did not know which one was correct, they took both. After determining which one worked, they used it to replace SRM B. However, because the remaining nose cone would not work, they could not replace IRM C as planned.

The workers asked a manager how to proceed, and were told to pull IRM C out from the bottom of the reactor vessel. The manager later said he thought the metal tube encasing IRM C was made of titanium—which does not become radioactive if inserted into the reactor core. However, the tube was made of stainless steel, which becomes very radioactive when exposed to the core of an operating reactor.

As the workers pulled the tube holding IRM C from the bottom of the reactor vessel, their radiation detectors alarmed. They set the bottom end on the floor and quickly left the area.

The ensuing recovery operation found that the radiation level at the tip of the tube holding IRM C measured 3,226 rem per hour on contact, and 39 rem per hour at a distance of 30 centimeters. General radiation levels in the area ranged from 4.6 rem to 8.6 rem per hour at waist level. Federal regulations limit workers to an annual dose of 5 rem—meaning that the workers could have exceeded their yearly exposure limit in an hour or less.

Workers remotely cut the IRM C tube into one-foot segments, put them in a special shielded container, and placed the container in the spent fuel pool (NRC 2011m).

NRC Sanctions

The SIT identified six violations of regulatory requirements associated with the ROP's *occupational radiation safety* cornerstone:

- Failure to follow clear and accurate instructions governing a maintenance task, so that workers were unable to complete the task as instructed and lacked needed tools.
- Failure to implement procedures that properly recognized the risk associated with a maintenance task, and properly accounted for that risk.
- Failure to implement proper human performance procedures in not telling workers what to do when they encountered problems during the task.
- Failure to comply with procedural requirements mandating formal revisions to written work instructions when the scope or methodology of a job changed.
- Failure to adequately brief workers on radiation levels that they might encounter when pulling out the IRM C tube.
- Failure to properly implement radiation protection procedures, exposing workers to radiation dose rates as high as 39 rem per hour.

The NRC classified all six violations as Green.

MILLSTONE UNIT 2, CT

The Near-Miss

The NRC sent an SIT to the site after a test procedure led to an unplanned and uncontrolled increase in the reactor's power level.

The SIT identified two violations involving workers not following procedures and failing to properly control the reactor's power level as a result.

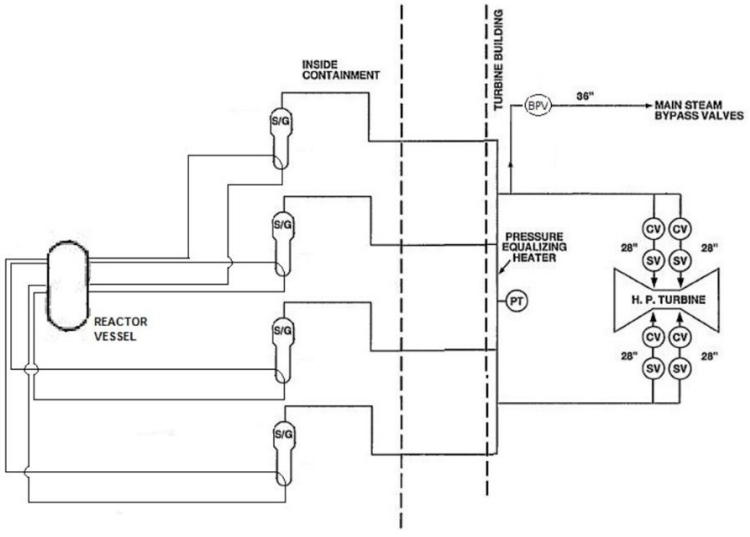
How the Event Unfolded

Operators in the control room of the Unit 2 pressurized water reactor (PWR) at Millstone reduced the power level from 100 percent to 88 percent on February 12, 2011, to perform a quarterly test of control valves for the main turbine. Because operating crews work in rotating shifts, this group had not conducted the test for many months. The crew took several steps to prepare for this infrequent operation and guard against any mistakes.

For example, the crew reported to the training center at Millstone on February 10, to review test procedures and perform them on a full-scale control room simulator. To prevent error, a peer checker guided each operator as he or she manipulated switches on a control panel, to ensure that the operator turned the correct switches at the correct time in the correct direction.

The turbine control valves (labeled CV in Figure 3) regulate the flow from the steam generators into the high pressure turbine (labeled HP turbine), to maintain constant pressure at that point. The test involved closing each of the four control valves one at a time. When a valve is closed, operators manipulate switches to open the other three valves slightly to compensate.

Figure 3. Schematic of a Typical Main Steam System for a Pressurized Water Reactor



Source: NRC

Workers must maintain a constant steam flow to hold the reactor's power level steady during testing. The heat produced by the reactor is transferred to the steam generators, so they can make the steam that flows to the turbine. The steam generators are the balancing point. If testing changed the steam flow, it would upset the balance and change the reactor's power level.

The key to the test involved balancing opposing effects in the reactor. Thus, during the simulated test, operators first reduced the reactor's power level from 100 percent to 88 percent. An inherent result was the buildup of xenon, a fission byproduct, which further decreased the power level. To maintain a constant power level, the operators diluted the boron concentration in the reactor cooling water, offsetting the xenon effect.

To further ensure proper balance, the operators opened one of the turbine bypass valves (labeled BPV). The open bypass valve allowed steam to detour around the turbine control valves and the turbine, and flow directly into the condenser, to maintain a constant flow.

When the operators performed the practice test on the control room simulator, they successfully maintained a constant steam flow.

Two days later, the freshly prepared operators reported to the real Unit 2 control room for a repeat performance. The crew leader conducted a briefing to review the test procedures and revisit each individual's responsibilities. The operators then reduced the reactor's power level to 88 percent, as planned and practiced. They then diluted the boron concentration and opened the turbine bypass valve, as they had done on the simulator just two days earlier.

However, when testing the first control valve, an operator turned the dial for the remaining three control valves in the wrong direction. That operator's peer checker mistakenly believed the operator had turned the dial in the correct direction. The control room supervisor, who was watching the test, also mistakenly thought the operator had turned the dial in the correct direction. However, because it was actually turned in the wrong direction, it upset the steam flow balance. The operator immediately saw that the reactor was losing balance, but turned the dial improperly three more times—further upsetting the balance.

The turbine bypass valve—which operators had opened in case they lost balance between the closing and opening control valves—closed fully about a minute later, in a futile attempt to restore the balance. The crew leader noticed that the bypass valve had closed and directed an operator to reopen it about 45 second later, but it automatically reclosed within 6 seconds.

The imbalance caused more steam to flow into the main turbine, and the pressure—which all the balancing measures were supposed to hold constant—rose 10 percent. In a PWR, increasing the steam flow causes the reactor's power level to rise. Three minutes after the test began, the power level stabilized at 96 percent—8 percent higher than before the test.

The operators reduced the reactor's power level back down to 88 percent, and successfully completed testing of the turbine control valve about an hour later (NRC 2011q).

NRC Sanctions

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

- Failure by operators to implement written procedures delineating responsibilities for controlling power output from the reactor core.
- Failure by operators to implement written procedures for testing the turbine valves, producing an unplanned increase in the reactor's power level from 88 percent to 96 percent.

The NRC classified the first violation as White, and the second violation as Green (NRC 2011k).

MONTICELLO, MN

The Near-Miss

The NRC sent an SIT to the site after a periodic test of a portion of the fire sprinkler system revealed that rust particles were blocking the water flow. The SIT determined that the plant owner had received numerous reports on corroded piping in the fire protection system in recent years, but had not reviewed them for applicability to Monticello, as required.

How the Event Unfolded

The intake structure at Monticello houses two circulating water pumps, four residual heat removal service water (RHRSW) pumps, an electric-motor powered fire pump, four emergency service water (ESW) pumps, two makeup pumps, two seal water pumps, and the fire system jockey pump. The circulating water pumps draw water from the Mississippi River to handle waste heat rejected from the main condenser and cooling towers. The RHRSW pumps send river water through heat exchangers during an emergency, to cool the reactor core and containment building. The ESW pumps send river water throughout the plant to cool emergency equipment and buildings housing emergency equipment. For example, the ESW system supplies cooling water for the plant's emergency diesel generators.

To prevent a fire from disabling much or all of this equipment, the intake structure has a fire sprinkler system. The sprinkler piping is normally drained of water. When a fire is detected and in response to operator action, a valve opens to admit water into the sprinkler piping, so it can spray water to extinguish the fire.

On August 26, workers opened the valve to test the fire suppression system at the intake structure. They found that something was blocking the flow of water into the sprinkler piping. Maintenance workers removed debris from inside the piping.

On August 28, workers retested the fire sprinkler system at the intake structure. Once again, the flow was blocked. Maintenance workers found extensive corrosion inside the sprinkler piping, and on September 2, operators declared the fire suppression system at the intake structure inoperable. Workers were assigned to patrol the area looking for fires as a compensatory measure.

Workers replaced some of the fire sprinkler piping, and flushed all the lines to remove loose material. Workers then used a video borescope and radiography to inspect the piping, and restored the fire suppression system to service.

The SIT found that the plant owner had received seven reports about clogging and blocking of piping for the fire protection system from 2006 to 2011. The SIT determined that the owner had not evaluated those reports, or factored their lessons into the plant's inspection and testing procedures.

In August 2007, Monticello workers had found that corrosion particles were clogging piping for the fire protection system for the emergency diesel generators. Workers initiated a maintenance task to inspect and flush the piping for the fire sprinkler at the intake structure, because it was of similar design. However, they postponed that maintenance task 10 times—and had not performed it before the test failure in August 2011. In April 2009, workers had to remove a section of the fire sprinkler piping at the intake structure to allow other maintenance work. Workers testing the reinstalled piping found that the flow was blocked, but an engineering assessment accepted the degradation. The SIT concluded that "the assessment was limited and narrowly focused."

The SIT further determined that the owner believed that the fire sprinkler piping at the intake structure was invulnerable to aging because it was normally emptied of water. However, the piping had filled with water several times since 1983, owing to spurious actuations and testing activities. Workers had then drained the piping and pressurized it with oxygen. The SIT concluded that the frequent wetting and drying cycles accelerated the corrosion of the piping (NRC 2011a).

NRC Sanctions

The SIT identified one violation of regulatory requirements associated with the ROP's *mitigating systems* cornerstone:

- Failure to incorporate experience from recent corrosion of piping for fire protection into maintenance and testing procedures.

The NRC classified the violation as Green.

The SIT also reported four violations of regulatory requirements associated with the ROP's *mitigating systems* cornerstone —violations that the plant owner had identified:

- Failure to design and install the fire sprinkler system at the intake structure per fire protection standards, in that the slope of piping and drain connections did not allow for adequate drainage.
- Failure to implement timely and effective corrective actions to inspect and flush the fire sprinkler piping at the intake structure, after discovery of degradation in piping for the fire protection system for the emergency diesel generators in August 2007.
- Failure to adequately assess the operability of the fire sprinkler piping at the intake structure after identification of flow blockage in April 2009.
- Failure to develop and apply proper post-maintenance testing criteria for testing conducted in April 2009.

The NRC reviewed and concurred with the owner's classification of all four violations as Green.

A primary factor leading to the Green classifications was the small amount of combustible material in the intake structure. The small inventory meant that while a fire might have started in a component, it would likely not have propagated to other equipment.

NORTH ANNA UNITS 1&2, VA

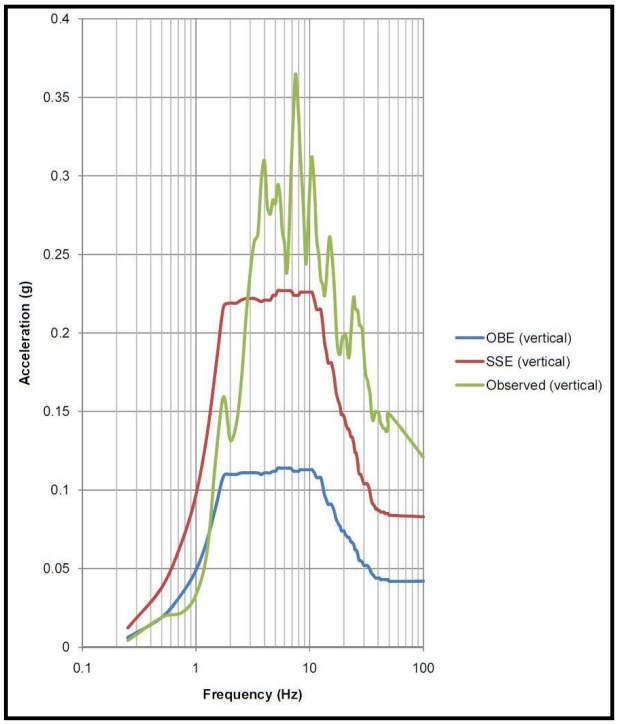
The Near-Miss

The NRC sent an AIT to the site following an earthquake that caused ground motion greater than the plant was designed to withstand (Figure 4).

The AIT determined that the earthquake caused no significant damage to safety systems, and the NRC authorized the plant owner to restart the reactors (NRC 2011c).

Figure 4. Ground Motion from an Earthquake at the North Anna Plant

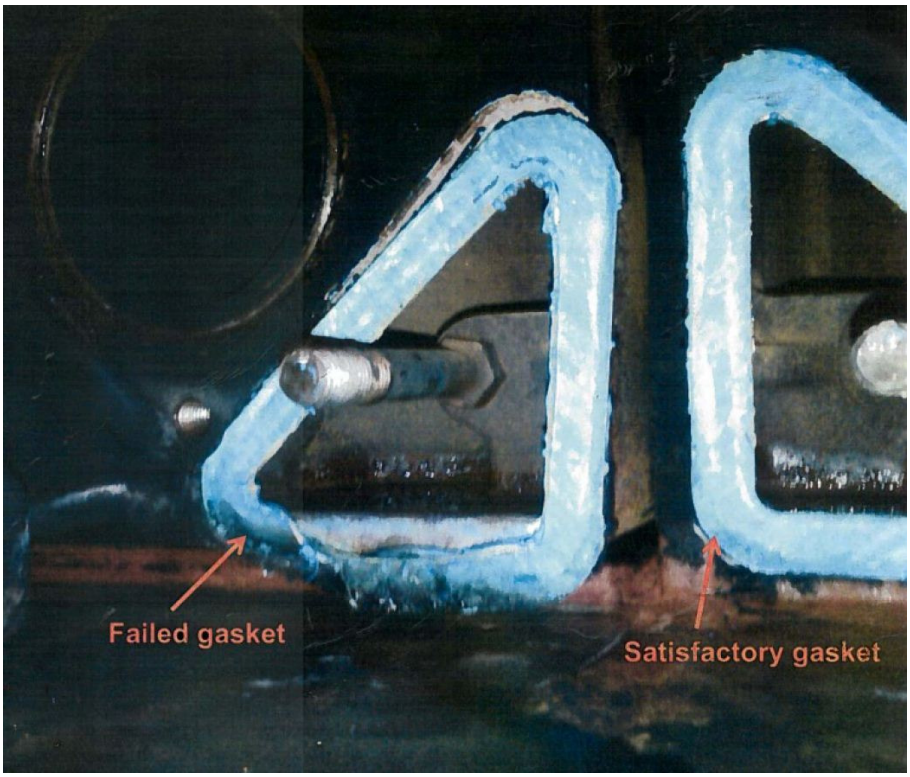
The blue line shows the ground motion (in g's), considered in the plant's design, from an operational-basis earthquake (OBE)—one severe enough to prevent the plant from operating. The red line shows ground motion from a safe shutdown earthquake (SSE)—one severe enough to potentially damage the main turbine/generator, but not emergency equipment needed to cool the reactor core. The green line is the actual ground motion measured at the plant during the August 2011 earthquake. The horizontal axis shows how fast the ground is shaking.



Source: NRC

How the Event Unfolded

At 1:51 pm on August 23, 2011, an earthquake of magnitude 5.8 on the Richter scale, and an epicenter about 12 miles from the site, caused both reactors at North Anna—then running at 100 percent power—to automatically shut down. The earthquake disconnected the plant from its electrical grid, which prompted all four emergency diesel generators to automatically start and supply power to essential safety equipment. Because



Gaskets on the emergency diesel generators at North Anna. Source: NRC

of the loss of offsite power, operators declared an Alert.[3] Less than an hour later, they turned off one of the emergency diesel generators because of a cooling water leak, and declared another Alert because of the loss of the generator. Less than four hours after the earthquake, the plant was partially reconnected to the electrical grid. Around nine hours after the earthquake, offsite electrical power to the plant was fully restored.

Because the earthquake's magnitude exceeded the level considered in the plant's design, the NRC dispatched an AIT to investigate. The team identified several shortcomings with the instruments installed at North Anna to monitor earthquake activity. For example, an instrument failed that had been designed to sound an alarm when an earthquake's severity approached that considered in the plant's design. This instrument lost power when the earthquake disconnected the plant from its electrical grid. Workers later installed a battery backup, to allow the alarm to remain in service even if it lost its normal power supply. The AIT also found that measurements from earthquake monitors at the plant could not be readily compared, because workers had not adequately maintained and calibrated the monitors.

The AIT investigated the cooling water leak that prompted operators to shut down one of the four emergency diesel generators less than an hour after the earthquake. Plant workers attributed the leak to a fiber gasket improperly installed after maintenance in May 2010. The AIT concluded that workers had failed to follow the vendor's recommendations for installing the gasket. The AIT also questioned the finding of the plant owner that the failure of the emergency diesel generator was an isolated event, as the inadequate gasket

[3] The NRC's four categories of emergencies include Unusual Event, Alert, Site Area Emergency, and General Area Emergency.

installation introduced the potential for a common-cause failure of all the emergency diesel generators.

While those generators were powering emergency equipment on August 23, operators reported that the electrical output from generator 1J exhibited frequency oscillations from 59 hertz to 61 hertz. The AIT noted that the plant's technical specifications limited the frequency to a range of 59.5 hertz to 60.5 hertz. The engine speed of generator 1J also varied by nearly 100 revolutions per minute—compared with a range of 20 to 30 revolutions per minute at the other generators.

When workers tested generator 1J on September 5, they did not observe frequency oscillations, although the test conditions did not match the conditions on August 23. Because no instruments recorded the frequencies of generator 1J during the event, the AIT could not substantiate the apparent violation.

On September 3, workers observed a cooling water leak from another emergency diesel generator during a test run. Workers discovered that an orifice plate was missing from the discharge flange of the cooling water pump. They found it still attached to the discharge flange of a cooling water pump removed in 2004. Further investigation determined that the orifice plate was missing from the discharge flange of a cooling water pump on a third emergency diesel generator. The AIT concluded that the missing plates degraded the cooling capabilities of the emergency diesel generators.

Workers also discovered that 25 of the 27 dry casks of spent fuel at North Anna had slid ½ inch to 4½ inches during the earthquake. The workers did not observe any visual damage to the casks, and calculations showed that the force needed to move the casks would likely not have damaged their fuel or internal components.

NRC Sanctions

The AIT identified no violations of regulatory requirements.

OCONEE UNITS 1, 2, and 3, SC

The Near-Miss

The NRC sent an SIT to the plant after workers reported that the standby shutdown facility (SSF)—an emergency system designed to cool the reactor core after an accident—could be disabled by the high temperature inside the containment vessel resulting from the accident.

The SIT identified two violations. One stemmed from an inadequate modification to the facility in 1983. The other violation involved improper revision of operating procedures in 2011 after workers identified the problematic modification.

How the Event Unfolded

The three pressurized water reactors at Oconee have the same design as the reactor that experienced a partial meltdown at Three Mile Island in March 1979. The SSF, added in 1983, was among safety upgrades owners made at Oconee in response to that meltdown.

The SSF serves as a backup to other emergency systems. It protects the reactor core from damage caused by overheating during a station blackout

(the loss of all power except that supplied by batteries), a high-energy line break (the rupture of a pipe connected to the vessel housing the reactor core, which would allow its cooling water to rapidly discharge), or a fire or flood that disabled emergency equipment.

The SSF has its own emergency diesel generator, which supplies power to electric heaters in the pressurizer, among other components. Each of the three reactors at Oconee has a pressurizer attached to piping connected to the vessel housing the reactor core. In these reactors, water flowing through the reactor core is heated to more than 500° F, but does not boil because of the very high pressure inside the reactor vessel. The pressurizer accommodates the expansion of the water as it heats up, and its contraction when it cools down. The pressurizer also allows operators to control the pressure of the water: they can spray cool water into the pressurizer to reduce the water's temperature and pressure, or turn on heaters within the pressurizer to increase the water's temperature and pressure.

On June 2, 2011, workers determined that electrical breakers located within the reactor containment building had protective features that caused them to trip (open) if sensors detected overheating. Computer analyses of postulated accidents had indicated that the temperature inside the containment building could reach 267° F. Yet the protective devices were set to open the electrical breakers at lower temperatures. That meant that conditions inside the containment building during an accident would cause the electrical breakers to open, preventing the SSF from turning on the pressurizer heaters. Without the pressurizer heaters, operators' ability to cool the reactor core adequately would be impaired, if not prevented.

Workers finished replacing the electrical breakers on June 8 with breakers that did not include the protective devices. In parallel, a test laboratory subjected the replacement breakers to the environmental conditions that could exist inside the containment building during an accident. On June 24, the laboratory informed the plant owner that 75 percent of the tested breakers failed at temperatures below 267° F, even with their protective devices removed.

Operators had also revised the procedures they would use during an accident to cool the reactor core adequately even if the SSF were unable to control the pressurizer heaters. The SIT observed operators using the revised procedures on the Oconee control room simulator, and concluded that that approach could work. However, the SIT determined that federal regulations required prior NRC approval for this alternate means of cooling the reactor, and that the plant's owner had failed to properly obtain that approval (NRC 2011g).

NRC Sanctions

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

- Failure to install electrical breakers for the SSF pressurizer heaters that were properly qualified for their post-accident operating environment.
- Failure to properly evaluate the proposed revision to emergency operating procedures for use of the SSF without control of the pressurizer heaters.

The NRC classified the first violation as Red for Oconee Units 1 and 2, and as Yellow for Oconee Unit 3. Because of uncertainty in the risk assessments performed by the plant owner and the NRC, the agency elected to apply the less severe Yellow finding to all three units. The NRC classified the second violation as Green (NRC 2011b).

PALISADES, MI (first incident)

The Near-Miss

The NRC sent an SIT to the site after one of three pumps supplying cooling water to emergency equipment failed for the second time in two years. The SIT determined that workers had replaced internal parts of the pump in 2009 with materials susceptible to stress corrosion cracking. This susceptibility caused pump failures in September 2009 and August 2011 (NRC 2011d).

How the Event Unfolded

The service water system at Palisades has three pumps that use water from a nearby lake to cool safety equipment. This equipment includes emergency diesel generators, control room coolers, containment air coolers, and the component cooling water system.

Workers replaced the internal parts of service water pump P-7C in June 2009, because the original carbon steel parts were eroding. The replacement parts were made of stainless steel, which is more erosion resistant.

Pump P-7C failed in September 2009 after 2,414 hours of operation, owing to stress corrosion cracking of the recently installed internal parts. The plant's operating license gave the owner up to 72 hours to repair the pump and return it to service, or the reactor had to be shut down. Workers replaced the broken parts within the 72-hour deadline. However, they used new parts with the same design and composed of the same material as the old parts, so the parts remained vulnerable to stress corrosion cracking.

A similar pump at the Prairie Island nuclear plant in Minnesota, with internal parts supplied by the same vendor, had failed in July 2010 for the same reason as the Palisades failure—stress corrosion cracking.

On August 9, 2011, service water pump P-7C failed again owing to stress corrosion cracking of internal parts, which had operated for 14,114 hours. Workers replaced the broken parts with those made of a new material that was resistant to both erosion (the original problem) and stress corrosion cracking (the new problem).

The plant owner, had received a report in March 2011 from a consultant it had retained to examine the September 2009 pump failure. The consultant reported that the internal parts used for the pumps were not suitable for the operating conditions. However, the owner did not review and accept the report until August 2011—too late to prevent another pump failure.

The SIT chronicled a long list of warnings dating back to September 1991 concerning the use of the stainless steel parts. For example, the NRC had issued Information Notice 93-68 in September 1993, on stress corrosion cracking of pump internal parts made of stainless steel at the Beaver Valley plant. The industry's Institute for Nuclear Power Operations had issued a report in 2006 on 12 pump failures from 1998 to 2006—most caused by stress corrosion cracking of stainless steel parts. And the NRC had issued

Information Notice 2007-05 in February 2007, listing 23 service water pump failures since 1983 that stemmed from stress corrosion cracking of stainless steel parts.

Despite these repeated warnings, workers replaced the carbon steel parts of pump P-7C with stainless steel parts in June 2009. As had happened so often before, the unsuitable parts caused failure in September 2009 and again in August 2011 (NRC 2011d).

NRC Sanctions

The SIT identified no violations of regulatory requirements.

PALISADES, MI (second incident)

The Near-Miss

The NRC sent an SIT to the site after workers troubleshooting faulty indicator lights for the position of the emergency airlock door inadvertently shut down about half the power supply to instruments in the main control room. The power loss triggered an automatic shutdown of the reactor, as well as the automatic closure of the main steam and containment isolation valves.

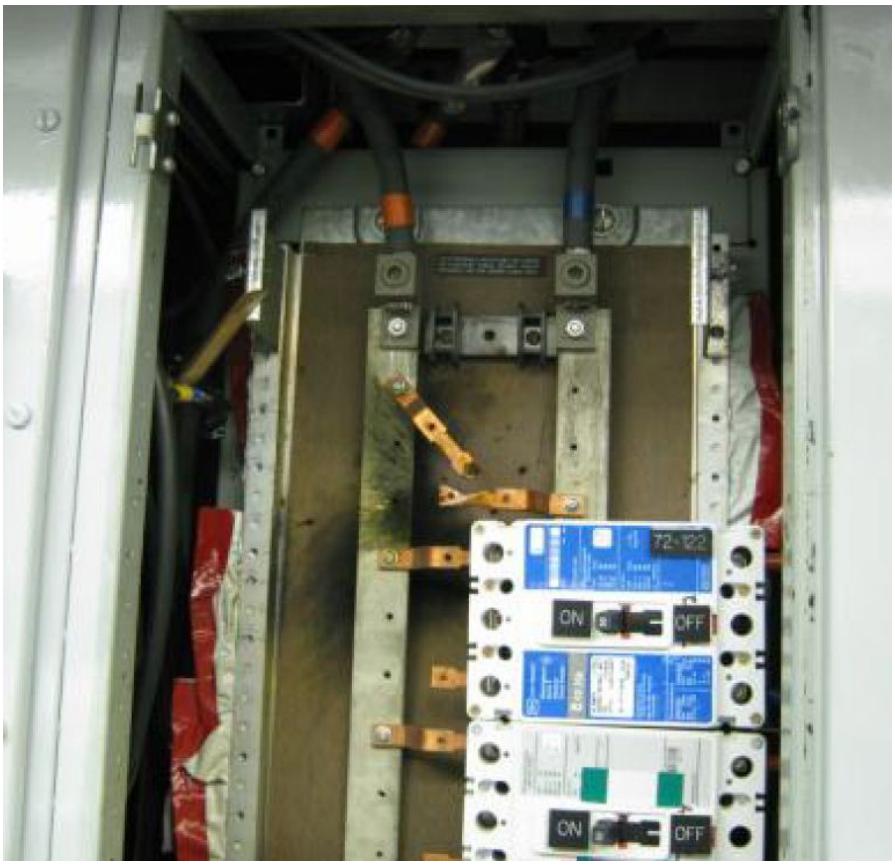
The SIT identified eight violations of safety requirements. The most serious involved the failure to adequately plan for and conduct maintenance on equipment inside the control room.

How the Event Unfolded

The lights in the main control room indicating that the emergency airlock door was closed failed. A periodic test of the airlock door was due to be performed soon, so maintenance workers were troubleshooting the reason for the failure of the indicator lights. The workers traced the problem to a faulty electrical breaker inside a distribution panel that connected power from one of the two sets of station batteries to plant equipment. Workers replaced the faulty breaker on September 23.

After completion of this maintenance task, control room operators observed flickering lights for some of their instruments. The next day maintenance workers reopened the distribution panel and identified four electrical breakers that might have been improperly installed, causing the intermittent power fluctuations. Managers decided to reinstall the four suspect breakers.

On September 25, a worker loosened a screw inside the distribution panel to gain access to an electrical breaker. A flash from an electrical spark caused the worker to quickly pull away his hands. The right end of a copper bar—which the tightened screw and the worker's hand had held horizontal—fell toward other energized copper bars. The proximity of the bars caused an electrical spark to jump across the gap, and the spark cut power to the area. The electrical short also shut down about half the power supply to instruments and controls in the main control room.



The distribution panel at Palisades where the disruption in electrical power started. Source: NRC.

By design, that loss of power automatically triggered the rapid shutdown of the reactor core, as well as the closure of the main steam and containment isolation valves. The operators' response to the reactor shutdown was complicated by the unexpected and unwanted opening of a relief valve inside the containment building. The open valve allowed reactor cooling water to leak onto the floor, the water level inside the pressurizer to rise to 98 percent full, the water level in one of the steam generators to rise to 98 percent full, and relief valves on the charging pumps to open and leak water into the auxiliary building. The power loss also disabled many indicators, chart recorders, and alarms in the main control room, further complicating operators' response. Despite these complications, operators succeeded in taking the reactor to cold shutdown by 6:33 am on September 27.

The NRC's SIT examined the preparation for and execution of the maintenance work for the failed indicator lights. The team "concluded that the work on September 25, 2011, was performed with a focus on completion of the tasks on schedule, without ensuring all safety policies were followed."

NRC Sanctions

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

- Failure to provide adequate instructions for maintenance work, and to ensure that workers followed approved procedures, as required by Appendix B, Quality Assurance, to 10 CFR Part 50.
- Failure to implement procedures for responding to reactor incidents when operators did not take steps specified in the approved procedure for loss of control room alarms, designed to ensure that the main generator was disconnected from the offsite electrical grid.

The NRC classified the first violation as Yellow and the second as Green.

The SIT also identified six violations of regulatory requirements associated with the ROP's *mitigating systems* cornerstone:

- Failure to conduct a pre-job briefing for workers performing the breaker maintenance on September 25, 2011, as required by plant procedures.
- Failure to limit the working hours of staff, as required by plant procedures. Specifically, the SIT reported that the duty station manager had worked for 25 straight hours, and more than 72 hours in the prior seven days, and that an electrical superintendent had worked more than 72 hours in the prior seven days.
- Failure to develop adequate procedures for operators to implement in response to a design and licensing bases event: namely, the loss of a single train of battery power.
- Failure to properly screen proposed modifications to the plant, as required by Appendix B to 10 CFR Part 50 as well as 10 CFR 50.59.
- Failure to comply with Criterion IV, Design Control, of Appendix B to 10 CFR Part 50, in that two electrical breakers were installed in the battery distribution panel with automatic protective trips, when the design bases required the breakers to be actuated manually.
- Failure to notify the NRC within eight hours of an event or condition that put the plant in an unanalyzed condition that significantly degraded safety.

The NRC classified the first five violations as Green and the last one as a Severity Level IV violation.

PERRY, OH

The Near-Miss

The NRC sent an SIT to the plant after workers replacing a detector used to monitor the power level in the reactor core had to abandon the job because of very high radiation levels.

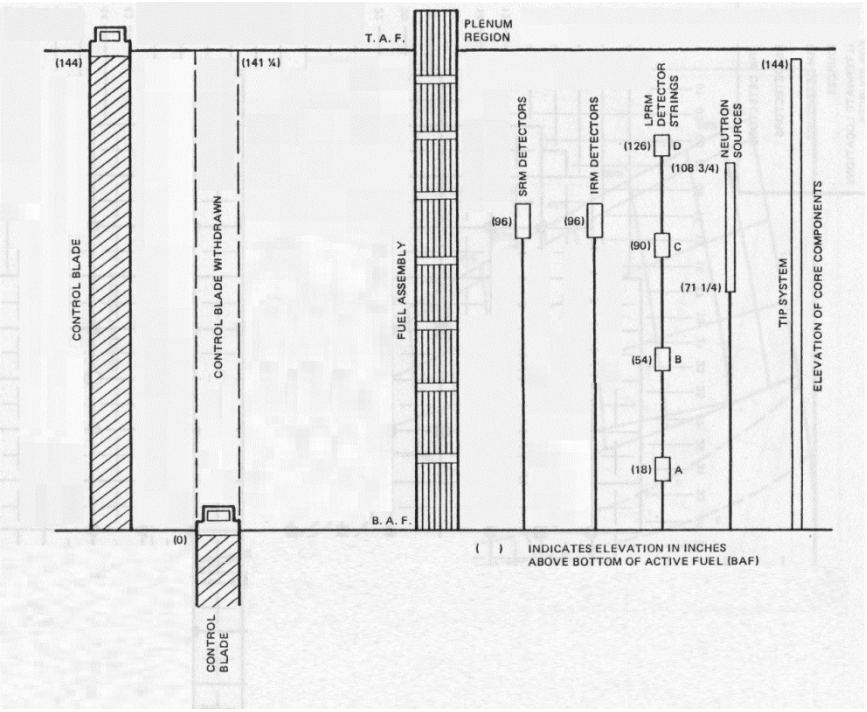
The SIT identified two violations of regulatory requirements associated with the planning and execution of the maintenance activity.

How the Event Unfolded

Operators shut down the reactor on April 18, 2011, for refueling. On April 22, workers entered the area beneath the reactor vessel to replace source range monitor (SRM) C. The boiling water reactor at Perry has four movable

detectors called SRMs that monitor the reactor's power as it is starting up and shutting down. The SRMs are intended for use only at power levels below about 1 percent of rated power. The SRM detectors are about 1 inch long and about 0.16 inch in diameter, and are attached to cables that are nearly 40 feet long. The detectors and their cables are located within hollow metal tubes that rise vertically from the domed bottom of the reactor vessel (Figure 5). Electric motors beneath the reactor vessel insert the SRMs to a position 18 inches above the mid-point of the reactor core when the reactor is shut down or operating at very low power levels. Once intermediate range monitors (IRMs) begin tracking the power level during startup, the SRMs are withdrawn to a position 2 ½ feet below the bottom of the reactor core.

Figure 5. Side View of a Boiling Water Reactor Core Showing the Positions of Its Components



Workers had replaced SRM detector C in February 2009. SRMs are replaced periodically when their sensitivities decrease because the uranium they contain depletes, or because of wear and tear on their cables. During a reactor startup in May 2010, operators were unable to retract SRM C from the reactor core, and it was quickly disabled as it remained in the core during high-power operation. Workers were scheduled to replace SRM C during the refueling outage in 2011.

The workers entering the area beneath the reactor vessel on April 22, had instructions to cut the cable for SRM C, and attach the cut end to a spool in a portable take-up cartridge. After connecting the cable to the spool, workers planned to exit the area and remotely turn the spool to wind the cable and SRM C into the cartridge. They would then place the cartridge in a lead cask for shielding against high levels of radiation from the unit. The plant owner calculated that the radiation level one inch from the SRM detector was 66

rem per second. That yields a lethal exposure to radiation in about seven seconds—hence the remote-handling provisions.

The take-up cartridge could hold only 30 feet of cable and the detector, but the cable for SRM C was 39 feet long. To work around that shortcoming, workers planned to cut the cable, pull the SRM down 9 feet, and cut the cable again. They would then attach the 30-foot cable to the spool in the take-up cartridge, and remotely retract the cable and detector.

The written procedure instructed them to cut the cable "when radiation level rises and/or there is approximately 30' of cable left" in the tube. However, the cable does not have markings like a measuring tape indicating its length. Instead of pulling the SRM detector 9 feet down, the workers pulled nearly 22 feet of cable from the tube. In doing so, they pulled the highly radioactive SRM detector much closer to themselves than planned. Radiation levels in the area skyrocketed from about 2 rem per hour to more than 1,000 rem per hour. (Instruments in use in the area could not record values above 1,000 rem per hour.)

The SRM cable take-up cartridge at the Perry plant. *Source: NRC*

Two radiation protection specialists were present at the job site. One was in the area with the workers. The second specialist was just outside the area, monitoring radiation readings from the individual workers and other locations. When these specialists advised the workers about the extremely high radiation levels, they exited the area extremely quickly. Fortunately, electronic dosimeters worn by the workers revealed that their maximum radiation exposure was 0.098 rem—far below the NRC's limit of 5 rem per year for nuclear workers (NRC 2011n).

The NRC's SIT found many faults with the radiation protection provided to these workers, including these:

- In replacing SRM C, workers followed radiation protection instructions used in replacing IRM C during a refueling outage in

2005. These instructions did not account for the much higher radiation levels from SRM C—which reflected the fact that it had resided inside the core of an operating reactor for nearly a year.
- The instrumentation used to monitor radiation during the work was not sufficient to record the range of potential radiation levels.
- The guidance provided to workers was not sufficient to prevent them from removing too much cable.
- The radiation protection specialists wore headsets to enhance communication with each other, but none of the workers had access to the channel the specialists were using.
- The work area had many traps, including open grating that workers could have fallen through and tripping hazards that could have delayed them from exiting the area before they received radiation exposures exceeding federal limits.

NRC Sanctions

The SIT identified two violations of regulatory requirements associated with the ROP's *occupational radiation safety* cornerstone:

- Failure to properly evaluate the radiological hazards associated with replacing a source range monitor. The SIT "determined that a substantial potential for an overexposure did exist, in that, it was fortuitous that the resulting exposure did not exceed the limits of 10 CFR Part 20."
- Failure to provide workers with adequate procedural guidance to ensure that the cable for SRM C was correctly attached to the take-up cartridge.

The NRC classified the first violation as White and the second as Green (NRC 2011i).

PILGRIM, MA (first 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 classified the violation as more serious than Green (NRC 2011y).

PILGRIM, MA (second incident)

The Near-Miss

The NRC sent an SIT to the site after employees encountered a problem during routine startup of the reactor after a refueling outage. The problem caused an emergency system to intervene and automatically shut down the reactor.

The SIT identified a violation involving failure to properly control the reactor's power level.

How the Event Unfolded

On May 10, 2011, operators restarted the reactor after a refueling outage. They did so by withdrawing control rods from the reactor core until it attained criticality—a sustained nuclear chain reaction. The operators continued withdrawing control rods to increase the reactor's power level, within various limits, to full power.

After operators had withdrawn five control rods by 1 foot each (a control rod is 12 feet long, and the Pilgrim reactor has 145 control rods, so this withdrawal was minor), the computer indicated that the water temperature inside the metal vessel housing the reactor had risen 18° F in five minutes. The computer projected that if this rate of increase was maintained for a full hour, the water temperature would rise by 216° F—exceeding the legal maximum heat-up and cool-down rate of 100° F per hour. That limit protects the metal vessel from damage caused by excessive expansion and contraction.

Concerned that they might violate the maximum heat-up rate, the operators reinserted the five control rods one foot each, reducing the reactor's power level. The computer showed the operators that the water temperature was no longer rising.

Because they were restarting the reactor, the operators needed both the power level and the water temperature to steadily increase. So they again withdrew the five control rods by one foot each, and also withdrew a sixth control rod by a foot. The reactor's power level began doubling every 20 seconds. That meant that a reactor operating at 1 percent power would be at 2 percent power 20 seconds later, 4 percent power 40 seconds later, 8 percent power 60 seconds later, 16 percent power 80 seconds later, 32 percent power 100 seconds later, 64 percent power 120 seconds later, and 128 percent power 140 seconds later.

In theory that's what it meant. In practice, the reactor protection system sensed that the reactor was out of control and responded by reinserting all the control rods within seconds, to terminate the runaway nuclear chain reaction.

No equipment malfunction contributed to this outcome—just many operator malfunctions. The problem was self-inflicted. The operators had overreacted to an apparently high heat-up rate by reinserting control rods. They then overreacted to an apparently low heat-up rate by again withdrawing control rods. In doing so, they increased the reactor's power level too rapidly and triggered an automatic reactor shutdown.

What the operators should have done was absolutely nothing. The initial indication of an excessive heat-up rate (the 18° F increase over a five-minute period) occurred because the operators had just withdrawn five control rods. Had they exercised some patience, that virtue would have been rewarded: the heat-up rate would have slowed by itself—as has happened tens of thousands of times at boiling water reactors and control room simulators. Minutes later, the operators would have had to withdraw more control rods to maintain the heat-up rate in the desirable range of 0° F to 100° F per hour. Instead, the operators overcorrected and then overcorrected again.

NRC Sanctions

The SIT identified one violation of regulatory requirements associated with the ROP's *initiating events* cornerstone:

- Failure to implement operating and reactivity control procedures during a reactor startup, "which contributed to an unrecognized sub-criticality followed by an unrecognized return to criticality and subsequent scram."

The NRC classified this violation as White (NRC 2011h).

TURKEY POINT UNIT 3, FL

The Near-Miss

The NRC sent an SIT to the site after a normally open valve in piping for heat exchangers in the cooling water system uncontrollably closed. This single failure disabled all cooling water flow, which allowed some equipment and water in the spent fuel pool to heat up.

The SIT determined that workers responded properly to the event, and identified no violations of regulatory requirements.

How the Event Unfolded

On August 11 2011, workers were inspecting one of three heat exchangers for the component cooling water (CCW) system. To prepare for this task, the workers had closed valves to stop the flow of water through this heat exchanger. Water continued to flow through the remaining two heat exchangers to handle that unit's equipment cooling needs.

An intake cooling water (ICW) system pumps saltwater from the Atlantic Ocean through thousands of metal tubes inside these three heat exchangers. Heat from the CCW is conducted through the tube walls and carried away by the ICW. The cooled CCW is routed through the plant to cool vital and emergency equipment. This arrangement fulfills the dual objectives of minimizing saltwater corrosion of plant components, and providing a barrier between potentially radioactive water and the environment.

The workers heard a loud noise and observed water leaking from the ICW connection to one of the two CCW heat exchangers still in service. They also noted that the ICW flow through both of those heat exchangers had stopped.

Over the next 20 minutes, as the CCW stopped transferring its heat to the ICW, its temperature rose from 95° F to 111° F. Because cooling water supplied to equipment throughout the plant was now warmer, that equipment heated up, too. For example, alarms indicated that the temperature of the motor bearings for the reactor coolant pumps was abnormally high.

During that short period, an operator and the CCW system engineer manually opened another valve in an alternate piping pathway for ICW through the CCW heat exchangers. After turning a wheel to crack open the valve, they used a wrench to crank it fully open. Their efforts restored ICW flow through the two in-service CCW heat exchangers. Less than 10 minutes later, the CCW temperature had dropped to 97° F. Opening this parallel ICW

flow path also reduced the rate at which water was leaking from the CCW heat exchanger to less than 10 gallons per minute.

Workers determined that butterfly valve 3-50-406 in the common ICW discharge piping—a valve that is normally open—had failed. The preliminary reason was cyclic fatigue from vibrations caused by water flowing through the valve. Its closure had stopped the flow of ICW from the CCW heat exchangers back to the discharge. It closure also increased ICW pressure in the heat exchanger, causing a leak from one of the two in-service heat exchangers.

In 1994, workers had identified flow-induced vibration in valve 3-50-406, and concluded that failure of the valve could mean the complete loss of CCW. In December 2007 workers again identified flow-induced vibrations in valve 3-50-406, but did not implement a recommendation to replace the actuator. Workers also identified flow-induced vibration of valve 3-50-406 in February 2011, and found that the indicator had detached from the valve stem. No repairs were made to the valve before its failure on August 11, 2011.

The solution to the problem was to leave the valves in both the ICW pipes to the CCW heat exchangers open, to avoid subjecting them to flow-induced vibration damage (NRC 2011f).

NRC Sanctions

The SIT identified no violations of regulatory requirements.

WOLF CREEK, KS

The Near-Miss

During periodic testing of the component cooling water (CCW) system on June 1, 2010, workers observed unusual pressure on the discharge side of an operating pump. In the ensuing investigation, they found 22.5 cubic feet of air on the CCW side of the train B residual heat removal (RHR) heat exchanger. Similarly, during periodic testing of the CCW system on July 1, 2010, workers observed erratic behavior of a flow-control valve. This time they found 9.5 cubic feet of air on the RHR side of the train B RHR heat exchanger.

The SIT determined that these problems stemmed from a change in November 2002 to the procedure governing how the RHR system piping and components would fill with water and vent air. The SIT also documented numerous opportunities for workers to have detected and corrected the problems before the summer of 2010. The NRC identified four violations related to the identified shortcomings.

How the Event Unfolded

The RHR system is one of Wolf Creek's emergency core cooling systems, designed to cool the reactor core and containment building in the event of an accident. The CCW system supports the RHR system by providing cooling water to the RHR heat exchangers. The CCW system removes heat from water in the RHR system while cooling the reactor core and containment.

In October 2002, the Callaway plant in Missouri—nearly an identical twin to Wolf Creek—reported a problem caused by nitrogen gas leaking into

the piping of the RHR system. The displacement of water by the gas, called voiding, can impair and even disable the system. For example, if a gas bubble is large enough and is carried to the system's pump, it can cause the pump's impeller to freewheel rather than move water through the piping.

In response to the Callaway problem, workers at Wolf Creek revised procedures for filling piping and components with water and removing gas from the piping via venting. Workers sometimes intentionally drain water from systems to allow maintenance on pumps and valves. The revisions implemented in November 2002 omitted a vital requirement for successful refilling: they did not specify the minimum water flow rate needed to "sweep" voids out of pockets in the piping and carry them along to vent locations.

In fall 2009, employees shut down Wolf Creek for refueling and partially drained the RHR system. They then used the deficient fill and vent procedures to refill the piping.

In December 2009, a flow-control valve unexpectedly closed and then reopened during a test run of RHR pump A. The RHR system engineer observed the anomalous behavior but initiated no follow-up.

In March 2010, a flow-control valve again closed unexpectedly and then reopened during a test run of RHR pump A. The RHR system engineer again observed the anomalous behavior but initiated no follow-up. Later analysis revealed that voids passing through the RHR piping had caused the flow-control valve to close and reopen during both the December and March tests.

During testing on May 24, 2010, the backup CCW pump started unexpectedly, and a vibration technician observed that the CCW pipe was shaking. The water level in the CCW surge tank dropped by 68 gallons. This large tank is partially filled with water and connected to CCW system piping, to accommodate expansion and contraction of water caused by temperature changes. The technician incorrectly characterized the pipe vibrations as a normal system response, and attributed the unplanned pump start to a faulty pressure switch. Analysis later determined that voids in the CCW piping caused water to drain out of the surge tank, the backup pump to start, and the piping to shake.

During testing on June 1, 2010, the water level in the CCW surge tank dropped 65 inches when operators started CCW pump C. This time, workers finally attributed the anomalous behavior to voiding within the system, and vented 22.4 cubic feet of gas from the CCW side of the train B RHR heat exchanger.

On June 28, 2010, a flow-control valve unexpectedly closed and then reopened during a test run of RHR pump A. The RHR system was declared inoperable, and workers again identified voiding inside the heat exchanger and piping as the problem.

NRC Sanctions

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

- Failure to promptly identify and correct the accumulation of gas voids within piping for the CCW and the RHR systems, despite numerous signs, such as unexpected CCW pump starts and unexplained movements of flow-control valves.

- Failure to develop and implement an adequate procedure for filling safety system piping with water and venting gases from it.
- Failure to adequately evaluate the presence of gas voids within safety system piping and components.
- Failure to promptly identify and correct deficiencies in the procedures used to fill safety system piping with water and remove gases.

The NRC classified all four violations as Green (NRC 2011bb).

OBSERVATIONS ON THE NEAR-MISSES IN 2011

While they did not receive the harshest NRC sanctions—appropriately, given that safety systems still functioned—the near-misses at Millstone Unit 2 and Pilgrim were the most ominous. Each involved an infrequent operation: a test of valves controlling steam flow to the turbine at Millstone, and a startup of the reactor at Pilgrim. Although operators perform these tasks only occasionally, they practice them routinely on full-scale control room simulators. Neither case involved the failure of any equipment. Neither case involved inadequate procedures. Both cases involved problems self-inflicted by plant operators. And both cases involved more than a single error by a single operator—the near-misses occurred after multiple mistakes by multiple people.

These cases are troubling because of what they suggest about operators' performance when facing larger challenges. What if safety equipment had been disabled before the incidents, or malfunctioned during them? What if procedures contained errors that directed the operators to take the wrong steps, or not to take the right ones? What if the incidents had escalated into severe accidents for which operators receive little or no training? Two near-misses involving inadequate operator performance during fairly routine activities does not instill confidence that their performance would be better during accidents.

A majority of the SIT and AIT findings in 2011 fell into two of the ROP's seven cornerstones: mitigating systems and occupational radiation exposure. The NRC already devotes considerable resources to these cornerstones through its onsite inspectors. These findings therefore do not suggest that the agency needs to reallocate resources from other cornerstones.

Full-time onsite NRC inspectors, supplemented by employees at regional offices and headquarters, conduct nearly 6,000 person-hours of oversight at each plant each year. Why didn't this NRC inspection army identify all, some, or at least one of the problems contributing to these 15 near-misses?

The simple answer is that it is not the job of these inspectors. Nor should it be their job.

It is not the NRC's job to install the right safety equipment properly, or to develop sound procedures that are faithfully implemented. The NRC's job—through its special inspection teams sent in response to safety-related events, and its myriad other inspectors—is to determine if plant owners are fulfilling their legal obligations by conforming to all applicable safety regulations.

In theory, the NRC's inspectors should rarely identify any problems,[4] and few problems should be self-revealing: that is, they should not appear during routine NRC inspections. After all, federal regulations[5] require that plant owners find and fix safety problems in a timely manner. Because the NRC lacks the resources to inspect every inch of piping, and peer over the shoulder of every worker performing maintenance tasks, the NRC must place its inspection findings in proper context.

That is, any time an NRC inspector identifies a safety problem, it means two things: (1) a broken widget needs to be fixed, and (2) a deficiency in the owner's inspection and testing regimes needs to be fixed, too. The NRC is not doing its job properly when it allows owners to fix only broken widgets and not their inspection and testing regimes—the procedures that did not prompt workers to realize that widgets were broken.

Every NRC finding should trigger a formal evaluation of why an owner failed to find and fix the problem before NRC inspectors found it. Because NRC inspectors cannot examine every widget, the agency and the public must have confidence that owners have adequate inspection and testing regimes. When those exist, NRC inspectors find fewer problems in the small samples they examine, and—more importantly—there are fewer problems in the larger samples that NRC inspectors do not examine. When the regimes are inadequate, the proliferating number of preexisting, undetected safety problems means that fewer things have to go wrong to trigger the perfect nuclear storm.

Adequate inspection and testing regimes are more than a good idea—they are the law. The NRC must enforce that law. It's theirs.

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[4] We use "rarely" instead of "never" because some problems, such as a leak that develops through a bad gasket, can be found by the next person walking through an area who sees a puddle on the floor. That person will sometimes be an NRC inspector rather than a plant worker.

[5] Specifically, Appendix B to 10 CFR Part 50, online at *http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-appb.html*.

CHAPTER 3. POSITIVE OUTCOMES FROM NRC OVERSIGHT

This chapter describes situations in 2011 where the NRC, working through the reactor oversight process, 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, in choosing these situations, we focused on especially good outcomes. These results show that the NRC can be an effective regulator and provide insights into how the agency can emulate these outcomes more broadly and consistently.

FLOODING AT FORT CALHOUN

In June 2011, severe flooding relocated the Fort Calhoun nuclear plant in Nebraska from being beside the Missouri River to being within it. The plant withstood this challenge intact in large part because of commendable performance by NRC inspectors, analysts, and managers the previous year.

The Fort Calhoun plant survived severe flooding in June 2011.
Source: Associated Press 06/14/2011

NRC inspectors discovered that the agency had issued an operating license for Fort Calhoun based on representations by the owner that the facility could withstand flooding up to 1,014 feet above mean sea level. However, NRC inspectors found that flooding above 1,008 feet could disable vital equipment in several structures.

The plant's procedures called for stacking sandbags above existing floodgates to protect against flooding above 1,008 feet. The NRC inspectors determined this measure to be unreliable because the half-inch-wide top of the floodgates provided an inadequate foundation upon which to stack five or six feet of sandbags. The NRC inspectors concluded that this deficiency was significant because the company's own risk assessment found that "severe core damage results if either intake or auxiliary building sandbagging fails" (NRC 2010c).

NRC analysts and managers backed up the inspectors' findings when the company argued that existing flood protection measures were sufficient. The company contended that the chances that a flood would exceed 1,007 feet were so remote that the sandbagging measures were justified, despite their potential unreliability. However, the NRC stood firm: it had issued the plant's license based on flood protection up to 1,014 feet, and the company had to take steps to comply with that requirement (NRC 2010b).

With the NRC's spotlight on flood protection, workers at Fort Calhoun also identified many other shortcomings. For example, they found several penetrations through walls that would have allowed floodwater to enter the intake and auxiliary buildings and disable vital safety equipment. The owners corrected these safety shortcomings (OPPD 2011).

We cannot say that the NRC's efforts to identify and correct flood protection deficiencies at Fort Calhoun averted nuclear disaster when the site experienced unusually severe flooding in June 2011. But we can say that these NRC efforts did mean that the site met its flood protection requirements, which came in handy last June.

MISTAKE AT THE HATCH PLANT

By its own admission, the NRC made a flawed decision in 1995 regarding safety at the Hatch nuclear plant in Georgia. However, the NRC found its mistake, and in 2011 took steps to correct it and defend its actions from an appeal by the plant's owner maintaining that the 1995 decision should remain unchanged.

The problem dated back to July 1976, when electricity supplied to emergency equipment at the Millstone Unit 1 reactor in Connecticut had a voltage too low to protect the equipment from damage. For example, a motor for a large pump that is not supplied with adequate voltage may be unable to turn the pump's shaft. Instead, the motor continues to draw current until its windings overheat and the motor burns up.

To avoid such damage, the NRC required plant owners to install under-voltage protection devices. These devices monitor voltage levels and turn off power when the voltage drops too low for too long. The NRC approved changes to the operating licenses for the two reactors at Hatch in 1982 governing the under-voltage protection methods.

In 1991 an NRC inspection team determined that Hatch's under-voltage setpoints were too low to protect emergency equipment from damage in some cases. The plant owner was concerned that raising the setpoints could lead to spurious power isolations—automatic shutdowns of some equipment—during short-lived voltage fluctuations. The owner therefore asked the NRC's permission to leave the setpoints alone, while adding an alarm to warn operators that voltage had fallen below desired levels, so they could take corrective action before automatic power isolations occurred. The NRC approved this request in February 1995.

In 2009, NRC inspectors conducting a Component Design Bases Inspection at Hatch determined that the voltage protection configuration did not ensure that emergency equipment would function as needed during accidents. The NRC examined the history of voltage protection at Hatch and concluded that:

> *…the [NRC] staff made an error in 1995 in approving manual actions to control voltage on the offsite circuits in order to demonstrate compliance with the applicable provisions of GDC 17, in establishing adequate voltages to the safety-related controls. This reliance on manual actions to control voltage was determined to be clearly inconsistent with current as well as staff guidance established in 1995.*

In June 2011 the NRC overturned its 1995 decision, and required the owner to correct the voltage protection problem (NRC 2011r).

The owner formally appealed the NRC's reversal, contending that the mandate to correct the voltage protection problem constituted a new regulatory requirement that the agency could impose only if a cost-benefit analysis showed that the safety benefits justified the cost (Southern Company 2011).

The NRC denied the appeal, informing the owner that:

> *The staff maintains its position that SNC's [Southern Nuclear Operating Company] electrical analysis for HNP [Hatch Nuclear Plant] must show that the existing setpoints and time delays are adequate to ensure that all safety-related loads have the required minimum voltage measured at the component terminal to start and operate safety-related equipment necessary to mitigate the consequences of the worst-case design basis event (DBE), without any credit for administratively controlled bus voltage levels* (NRC 2011e).

The NRC could have easily dismissed the concerns raised during its 2009 inspection on the grounds that it had reviewed and approved the voltage protection scheme in 1995. But the NRC discarded the easy decision for the right one. It conceded making a mistake with the 1995 decision, and ruled that public safety required the owner to rectify the problem. The NRC then defended its corrected position from a formal appeal by the owner.

One might criticize the NRC for making the mistake in 1995, and for not catching it over the ensuing 14 years. But it is better to commend the NRC for finding the mistake and taking steps to correct it. The U.S. public is far

better served by an agency that remedies mistakes rather than one that pretends to be perfect.

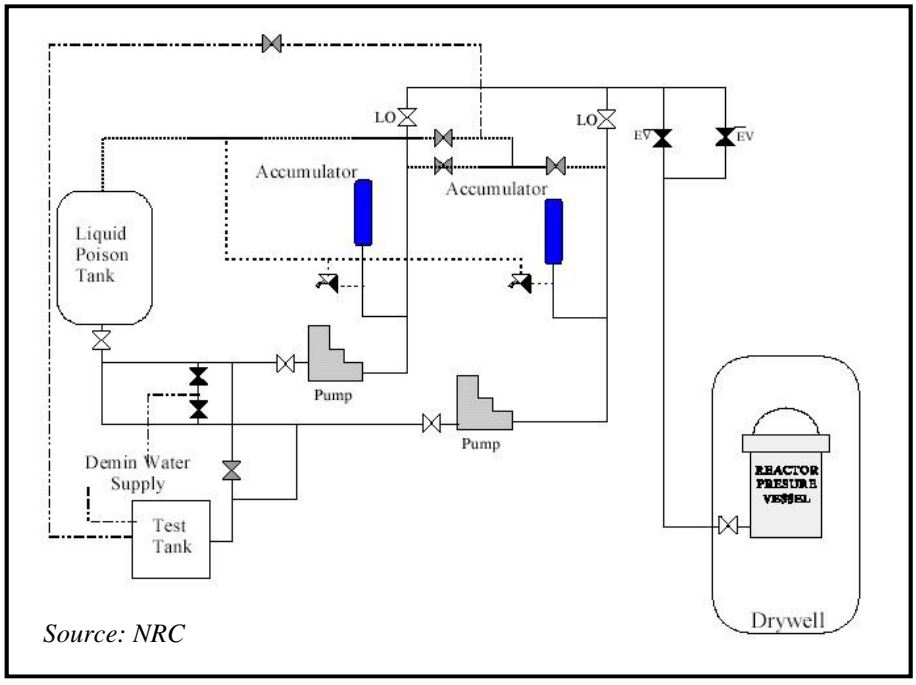
EARTHQUAKE HAZARD AT LASALLE

Nuclear plants employ a defense-in-depth approach to safety. That means that when a plant relies on a function to maintain safety, two or more independent methods—as different as possible—must be available to perform that function.

For example, two different systems can shut down a reactor when safety requires a shutdown. The primary method is the control rod drive system, which inserts rods filled with neutron absorbers into the reactor core within a few seconds. The control rods soak up neutrons, preventing them from interacting with uranium and plutonium atoms, thus interrupting the nuclear chain reaction.

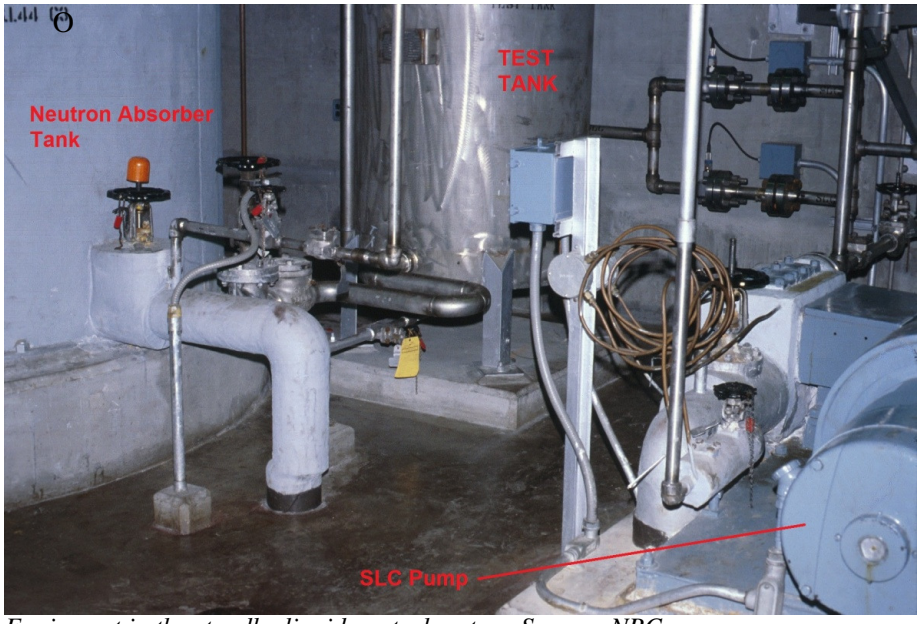
The standby liquid control (SLC) system backs up the control rod drive system (Figure 6). The SLC system can inject neutron absorbers in liquid form into the metal vessel housing the reactor core. The SLC system takes minutes to achieve what the control rod drive system can achieve in seconds. However, the SLC system can shut down the reactor from full power and keep it shut down, even if the control rod drive system experiences a total failure.

Figure 6. Schematic of the Standby Liquid Control System



Source: NRC

The SLC system features a large tank containing the neutron absorber in liquid form and two redundant pumps. Operators can manually start an SLC pump to transfer the neutron absorber into the reactor vessel.



Equipment in the standby liquid control system. Source: NRC

Operators periodically test the SLC system to verify that the pumps and other equipment are ready to function if needed. During this test, to avoid the delay and expense of adding the neutron absorber to water inside the reactor vessel and then filtering it out, the SLC pumps move liquid from the neutron absorber tank to a test tank. This procedure determines whether the pumps and control switches work, and whether liquid can be readily transferred out of the neutron absorber tank. One of the last steps in the test procedure is to pump the liquid from the test tank back into the tank storing the neutron absorber.

NRC inspectors examining the SLC system at the LaSalle nuclear plant in Illinois found that standard practice was to maintain the test tank about 75 percent filled with water when it was not being used for testing. In response to the NRC's questions, employees found that the original seismic design analysis of the SLC system addressed the test tank only in the empty condition.

A later assessment concluded that the weight of the water could cause the test tank to collapse during an earthquake, and that this collapse could disable the pumps and other components of the SLC system. The owner took steps to drain water from the test tank, to restore the configuration to what design studies had assumed (NRC 2011aa).

It was a good catch by NRC inspectors. They identified a practice that could have prevented a safety system from functioning when needed. And they identified this safety shortcoming before the Fukushima disaster heightened the world's sensitivity to such problems.

The NRC's good catch had a ripple effect. The owner of the Duane Arnold nuclear plant in Iowa read about the SLC test tank problem among the event notifications posted on the NRC's website. A quick check by the owner verified that Duane Arnold shared this shortcoming. Workers at that plant drained water from the test tank and revised procedures to require that the test tank be empty when the SLC system must be operable (NextEra 2011).

OBSERVATIONS ON EFFECTIVE NRC OVERSIGHT

In these cases, the NRC ensured adequate safety by enforcing owners to comply with existing requirements. At Fort Calhoun, the license issued by the NRC specified that the owner would protect the facility against flooding up to a certain elevation. NRC inspectors found that flood protection at the plant did not satisfy that requirement. Rather than acquiesce to arguments by the owner that flood protection was good enough, as shown over the past two decades, the NRC insisted on compliance with the requirements.

At Hatch, NRC inspectors found that a protection scheme the NRC had accepted in 1995 did not satisfy longstanding safety requirements. Rather than sustain the status quo, the NRC admitted its earlier miscalculation and compelled the owner to meet proper safety standards.

At LaSalle, NRC inspectors identified a safety deficiency that had been overlooked for decades. It involved a very subtle catch—an improperly analyzed and unacceptable configuration established during infrequent testing and then left in place. This subtlety was easy to miss, as evidenced by the many years it had been ignored. An NRC event notification was then instrumental in prompting the owner of another plant to correct the same problem.

CHAPTER 4. NEGATIVE OUTCOMES FROM NRC OVERSIGHT

This chapter describes situations where lack of effective oversight by the NRC, again working through the reactor oversight process, 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.

MISSED OPPORTUNITIES FROM NRC INSPECTION INSIGHTS

One of the NRC's major oversight efforts is its Component Design Bases Inspection (CDBI). In this inspection, the NRC audits a plant's design and associated worker training and procedures and compares those to the design requirements. Each plant receives one CDBI every three years.

The CDBI's roots go back to problems revealed at the Millstone nuclear plant in Connecticut in 1996. Units 2 and 3 were shut down from 1996 to 1998 so workers could correct a large number of deviations between the plant's actual physical configuration and what safety studies had assumed.

NRC inspectors identified 86 safety problems in 20 CDBI reports in 2011, illustrating the benefits of these inspections. For example:

- NRC inspectors found that the mandated maintenance program at the Brunswick nuclear plant in North Carolina did not include control switches for the torus hardened vent valves (NRC 2011z).
- NRC inspectors found that workers at the Salem nuclear plant in New Jersey had not tested the capacity of the backup batteries, as required, when voltage monitoring indicated that their performance had degraded by more than 10 percent (NRC 2011w).
- NRC inspectors found errors and omissions in design calculations for pumps used to transfer fuel oil from underground storage tanks to emergency diesel generators at the Peach Bottom nuclear plant in Pennsylvania (NRC 2011v).
- NRC inspectors found that emergency procedures for reversing a station blackout—the complete loss of normal and backup sources of AC electrical power—at the Summer nuclear plant in South Carolina failed to consider the compressed air needed to start emergency diesel generators (NRC 2011t).

The tragedy at the Fukushima plant illustrated the importance of reliable vent valves, battery power, and emergency diesel generators to nuclear safety. And NRC inspectors identified many other deficiencies in equally important safety equipment during CDBIs in 2011.

Given the long list of safety benefits resulting from CDBIs, why does this chapter include them? The primary reason is that the CDBIs do not audit the full range—or even a significant fraction—of design requirements for nuclear plants. Instead, they evaluate a minimum of "15 risk significant samples regarding engineering support of systems and components regardless of the number of units at the site" (NRC 2010a).

Consider the CDBI conducted by the NRC at the Harris nuclear plant in North Carolina. That CDBI examined 31 items:

- 17 components
- 7 operator actions related to those components
- 7 "operating experience items"—reports of good and bad safety outcomes issued by the NRC, vendors, or other plant owners

From this relatively small sample—less than 1 percent of the thousands of safety-related components, operator actions, and operating experience items at the Harris plant—the NRC identified 10 safety problems, an alarming percentage (NRC 2011j).

In the past, the NRC defended its small-scope CDBI process on grounds that its onsite inspectors select the highest-risk items to audit, rather than a random subset. That is true enough. However, that criterion suggests that the chosen items should be among the best maintained at each plant. After all, these items are subject to the most frequent and extensive testing and inspection owing to their elevated risk factors. If 10 of the 31 best-maintained items at a plant are found wanting, what does that suggest about the thousands of safety items that are less well maintained and less frequently inspected? Chances seem high that an equal or greater percentage of those safety items will also be impaired.

The NRC's determination that a large fraction of the few high-risk safety items examined during a CDBI have shortcomings speaks volumes about the plant owner's testing and inspection regimes. In theory, a CDBI should confirm that a reactor owner is complying with federal regulations by finding and fixing design bases safety issues in a timely and effective manner. Clearly and unequivocally, a CDBI identifying 10 safety issues among 31 high-risk items does not demonstrate compliance with federal regulations.

That also means that the NRC's CDBI program is seriously deficient, because when inspectors find safety problems, they ask owners to fix only those specific problems. That's all. They do not ask owners to address the bigger question that should be a primary objective of the CDBIs: to identify why owners' testing and inspection programs failed to identify and correct the design bases shortcomings.

Each plant in the country has undergone three or more CDBIs by now. Plant owners have had several opportunities to see where their own processes are failing to prevent or detect safety problems. However, the number of CDBI findings has been fairly consistent, with no signs of improvement (Doerflein 2011). That suggests that owners are not addressing programmatic

weaknesses revealed by the CDBIs but merely fixing specific problems. In essence, they are treating the symptoms rather than the cause.

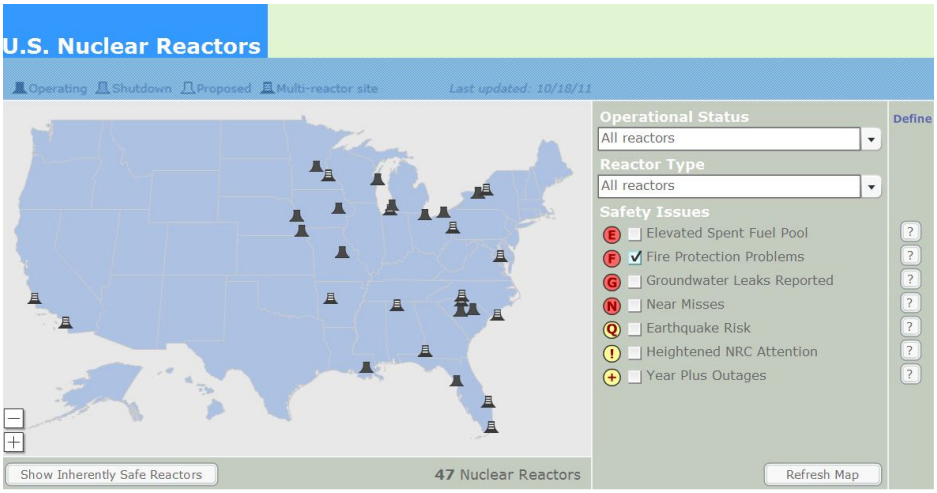
The NRC is similarly ignoring the most valuable insights from its CDBIs. And because a primary objective of the CDBIs should be to obtain those insights, ignoring them is inexcusable.

If the identification of design bases safety shortcomings spurred the NRC and owners to not only remedy those shortcomings but also to repair owners' inspection and testing regimes, this topic would have appeared in the chapter on positive outcomes. Until that occurs, it sadly belongs in this chapter.

STALLING FIXES TO KNOWN SAFETY PROBLEMS

A few years ago, UCS added a feature to its website called the Nuclear Power Information Tracker.[6] The tracker allows users to identify U.S. nuclear reactors that the NRC knows do not fulfill safety criteria. For example, one of the criteria is compliance with fire protection regulations. According to the tracker, 47 of 104 operating U.S. reactors do not meet this standard (Figure 7).

Figure 7. Operating Reactors with Known Fire Protection Problems



Source: UCS

Three of these reactors are at the Browns Ferry nuclear plant in Alabama. This is ironic, as the NRC adopted fire protection regulations in 1980 in response to a March 1975 fire at Browns Ferry that disabled all emergency core cooling systems for the Unit 1 reactor, and most of those for Unit 2. In ensuing inspections around the country, the NRC found that many reactors failed to meet these regulations, which led the agency to adopt an alternative set in 2004. Owners then had the choice of complying with either the 1980 or the 2004 regulations.

[6] See *http://www.ucsusa.org/nuclear_power/reactor-map/embedded-flash-map.html*.

The owners of the 47 reactors not now in compliance with the 1980 regulations have informed the NRC that they will fulfill the 2004 regulations. The current timetable calls for compliance by 2016. However, all past deadlines have slipped, suggesting that this deadline is also pliable.

NRC officials have stated that these reactors are sufficiently safe in the interim. These officials may sincerely believe that, but such a position is legally andethically unacceptable.

The NRC's Atomic Safety and Licensing Appeal Board best articulated the legal objections to this position in 1973:

> As a general rule, the Commission's regulations preclude a challenge to applicable regulations in an individual licensing proceeding. This rule has frequently been applied in such proceedings to preclude challenges to intervenors to Commission regulations. Generally, then, an intervenor cannot validly argue on safety grounds that a reactor which meets applicable standards should not be licensed. By the same token, neither the applicant nor the [NRC] staff should be permitted to challenge applicable regulations, either directly or indirectly. Those parties should not generally be permitted to seek or justify the licensing of a reactor which does not comply with applicable standards. Nor can they avoid compliance by arguing that, although an applicable regulation is not met, the public health and safety will still be protected. For, once a regulation is adopted, the standards it embodies represent the Commission's definition of what is required to protect the public health and safety (Farrar 1973).

NRC officials cannot legally redefine safety to include reactors known to be in violation of fire protection regulations promulgated in 2004. The only acceptable legal standard for safety is compliance with the regulations. These 47 reactors do not satisfy that legal standard.

It is also unacceptable from a ethical perspective to claim that these reactors are sufficiently safe despite violation of the legal standard. If these 47 reactors were truly safe enough today, why is the NRC making their owners—and stockholders and ratepayers, by extension—pay millions of dollars to comply with regulations?

The NRC cannot have it both ways. The agency cannot contend that these 47 reactors are sufficiently safe, and then require their owners to spend money on further fire protection measures. The NRC is on shaky ground when it tells the U.S. public that the nation's fleet of nuclear plants is safe.

Speaking of shaky ground, the NRC also knows that 27 reactors are operating with seismic protection levels below seismic hazard levels (Figure 8).

Figure 8. Operating Reactors with Known Seismic Protection Problems

Source: UCS

In 1996, the NRC revised its regulations governing seismic hazards in central and eastern United States. The NRC based this revision on information from the U.S. Geological Survey that the frequency and magnitude of earthquakes in these regions were greater than previously understood. The 1996 regulations required applicants seeking to build and operate new reactors to design protection levels for the heightened seismic hazards.

However, the NRC did nothing about the 27 reactors already operating in these regions. Their seismic protection levels—now known to be deficient—remain unchanged today.

In August 2011, an earthquake causing ground motions of greater magnitude than considered in the original design occurred at the North Anna nuclear plant in Virginia. The two reactors at North Anna are among the 27 the NRC has known since 1996 to be operating with less earthquake protection than needed.

North Anna's owner had applied to the NRC in 2003 for permission to build a new reactor. The owner designed the new reactor to comply with the NRC's 1996 regulations: its seismic protection levels are reportedly for ground motion of 0.5g's or more. By comparison, seismic protection levels for North Anna Units 1 and 2 remain for ground motion of 0.18 g's or less.

In the past, the NRC has required owners of existing reactors to upgrade seismic protection measures when the agency has upgraded seismic hazard levels. For example, the owner of the San Onofre nuclear plant in California shut down the Unit 1 reactor from February 26, 1982, until November 28, 1984, while workers installed additional pipe supports and other measures to close the gap between the plant's protection levels and known threat levels.

The NRC has also required owners to shut down operating reactors because of questions about seismic protection levels. On March 13, 1979, the NRC ordered owners to shut down five reactors within 48 hours—and to keep them shut down—until the owners responded to safety questions. The NRC had learned that errors in a computer code used to analyze the plants' response during seismic events made the results "non-conservative." In other

words, the piping supports and equipment restraints installed based on the computer studies might not prevent damage caused by earthquake movements. Because the safety of the reactors was no longer adequately assured, the NRC ordered the owners to shut them down (NRC 1979).

Today the NRC knows that 27 reactors operate with seismic protection levels less than their seismic threat levels. The NRC adopted regulations in 1996 requiring owners to equip new reactors with seismic protection levels at or above known seismic threat levels. In the past, the NRC has required reactors with similar safety deficiencies to correct them. That was the right thing to do then, and it is the right thing to do now.

Eight reactors fail to comply with both fire protection and seismic regulations. Making matters worse, these two safety threats share a link: earthquakes can trigger fires. These reactors may therefore experience an earthquake of greater magnitude than they are designed to withstand that can trigger a fire that employees are not capable of handling. These eight reactors are:

- Crystal River Unit 3 in Red Level, FL
- Duane Arnold in Palo, IA
- Farley Units 1 and 2 in Dothan, AL
- Perry in North Perry, OH
- St. Lucie Units 1 and 2 in Hutchinson Island, FL
- Summer in Parr, SC

People living around these plants face unnecessarily high risks because the NRC has not resolved known safety shortcomings.

OBSERVATIONS ON INEFFECTIVE NRC OVERSIGHT

Unsurprisingly, the common elements that produced negative NRC outcomes are essentially mirror images of the elements responsible for positive NRC outcomes.

At Fort Calhoun, the NRC compelled the plant owner to rapidly comply with safety standards. Yet the NRC allows 47 reactors around the country to operate even though they are out of compliance with fire protection regulations.

At Hatch, the NRC admitted that it made a mistake in 1995, and compelled the plant owner to rapidly comply with safety standards. At LaSalle, the NRC compelled the plant owner to rapidly comply with seismic protection requirements. Yet the NRC allows 27 reactors around the country to operate even though they are out of compliance with seismic protection standards adopted in 1995.

Even when nuclear power plants fully meet all applicable safety regulations, they may still experience an accident causing extensive harm to workers and the public. That reality is why the industry receives liability protection under the Price-Anderson Act, as amended—protection not needed by other private industries in the United States. When nuclear power plants operate despite violating safety regulations, the odds that a catastrophic reactor accident will occur increase.

The NRC simply is not doing its job when it writes tickets for safety violations identified during CDBIs, instead of using those insights to compel owners to reform their testing and inspection regimes so they can find and fix safety problems themselves.

The NRC simply is not doing its job when it tolerates widespread and longstanding violations of federal safety regulations.

And Congress simply is not doing its job when it allows the NRC to force Americans to face higher risks unnecessarily. Chernobyl and Fukushima are vivid reminders of what happens when safety requirements are not met.

The NRC simply must aggressively enforce its safety regulations. Anything less is unacceptable.

CHAPTER 5. SUMMARY AND RECOMMENDATIONS

In our view, the 15 near-misses reported at U.S. nuclear power plants in 2011 are too many, for several reasons:

- Two of the near-misses occurred at the Palisades nuclear plant in Michigan. These events shared contributing causes from improper maintenance. Maintenance is supposed to sustain safety margins, not compromise them.
- Two of the near-misses occurred at the Pilgrim nuclear plant in Massachusetts. (Comparisons between the two events cannot be drawn, as one involved a security matter about which the NRC provides scant public information.)
- Two of the near-misses involved no equipment failures and no procedure inadequacies—just problems self-inflicted by control room operators. When operators stumble over fairly routine tasks, it raises serious doubts about their ability to successfully respond to greater challenges under the stress of accidents.
- Two of the near-misses involved poor protection of workers from radiation exposure during removal of components from within the reactor core. While neither near-miss is defensible, the second is even less so given that the first sounded an alarm that apparently went unheeded.

The NRC identified 39 violations of federal safety regulations entailed in these near-misses. Some of these violations stemmed from problems arising during the event itself, but most were for safety problems that had been recognized for years. When known problems combine to cause near-misses, they are not surprises—they are accidents waiting to happen.

The NRC must draw larger implications from the findings of the reactive inspections summarized in Chapter 2 and its CDBIs. The NRC audits only about 5 percent of activities at every nuclear plant each year. The agency's limited-scope audits are designed to spot-check whether an owner's testing and inspection regimes are ensuring that a plant complies with regulations. Those regimes, if fully adequate, should find and correct any and all safety problems, leaving none for NRC inspectors to identify.

Consider this analogy. An inspector looks at only the left front tire of a car and finds it flat. What does that observation reveal about the other three tires? If the tire is flat because it has been punctured by a nail, it is reasonable to conclude that—absent a roadway littered with nails—the other three tires

are inflated. But if that tire is flat because it has been used so long that its tread is worn down to the point of exposing steel belts, it is reasonable to suspect that the other three tires are in similar condition. The NRC has to move beyond making plant owners fix flat tires to also determining why the tires became flat, and how other tires may be affected.

We know the NRC can do better because it did do better, in some cases, last year. Agency inspectors uncovered safety problems at the Fort Calhoun, Hatch, and LaSalle plants that their owners initially misdiagnosed or dismissed. NRC resident inspectors kept asking questions until the true picture came into focus. Their commendable efforts forced owners to correct safety problems, making these plants less vulnerable to near-misses. The intangible dividends from these efforts are very likely lessons learned by these owners about the kinds of questions they should be asking themselves. If so, the ripple effect from these NRC efforts will further reduce the risks of near-misses.

Unfortunately, the stellar performance exhibited by NRC inspectors in those cases is not the norm. The NRC did not extract the proper insights from its Component Design Bases Inspections last year. And it allows dozens of reactors to operate despite known safety impairments. As a direct result, millions of Americans are unnecessarily and unknowingly exposed to greater risk. If the NRC does not begin to consistently enforce its own safety requirements, the potential exists for tragic consequences that might have been avoided.

By expanding the behavior yielding positive outcomes and reducing the behavior leading to negative outcomes, the NRC would strengthen safety levels at nuclear plants across the country, decreasing the risks of near-misses—and of full-blown accidents.

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