NUCLEAR ENERGY RISK MANAGEMENT

HEARING

BEFORE THE

SUBCOMMITTEE ON INVESTIGATIONS AND
OVERSIGHT
JOINT WITH THE
SUBCOMMITTE ON ENERGY AND ENVIRONMENT

COMMITTEE ON SCIENCE, SPACE, AND
TECHNOLOGY

HOUSE OF REPRESENTATIVES

ONE HUNDRED TWELFTH CONGRESS

FIRST SESSION

FRIDAY, MAY 13, 2011

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Friday, May 13, 2011

HEARING ON NUCLEAR ENERGY RISK MANAGEMENT

FRIDAY, MAY 13, 2011

HOUSE OF REPRESENTATIVES,
SUBCOMMITTEE ON INVESTIGATIONS AND OVERSIGHT
JOINT WITH THE
SUBCOMMITTEE ON ENERGY AND ENVIRONMENT,
COMMITTEE ON SCIENCE, SPACE, AND TECHNOLOGY,
Washington, DC.

The Subcommittees met, pursuant to call, at 9:04 a.m., in Room 2318 of the Rayburn House Office Building, Hon. Paul Broun [Chairman of the Subcommittee on Investigations and Oversight] presiding.

RALPH M. HALL, TEXAS
CHAIRMAN

EDDIE BERNICE JOHNSON, TEXAS
RANKING MEMBER

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COMMITTEE ON SCIENCE, SPACE, AND TECHNOLOGY

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Joint Hearing:
Subcommittee on Investigations & Oversight and
Subcommittee on Energy & Environment

Nuclear Energy Risk Management

Friday, May 13, 2011
10:00 a.m. to 12:00 p.m.
2318 Rayburn House Office Building

Witnesses

Dr. Brian Sheron, Director, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission

Mr. Lake Barrett, Principal, LBarrett Consulting, LLC

Dr. John Boice, Scientific Director, International Epidemiology Institute

Mr. Dave Lochbaum, Director, Nuclear Safety Project, Union of Concerned Scientists

HEARING CHARTER

COMMITTEE ON SCIENCE, SPACE, AND TECHNOLOGY

U.S. HOUSE OF REPRESENTATIVES

Nuclear Energy Risk Management

FRIDAY, MAY 13, 2011

10:00 A.M. TO 12:00 P.M.

2318 RAYBURN HOUSE OFFICE BUILDING

On Friday, May 13, 2011 at 10:00 a.m. the House Science, Space, and Technology Subcommittee on Investigations and Oversight & Subcommittee on Energy and Environment will hold a joint hearing entitled, *"Nuclear Energy Risk Management."* The Committee on Science, Space, and Technology has jurisdiction over all energy research, development, and demonstration projects and all federally owned or operated nonmilitary energy laboratories.[1] The purpose of the hearing is to examine nuclear energy safety, risk assessment, public health protection, and associated scientific and technical policy issues in the United States in light of the earthquake and tsunami in Japan.

Witnesses

- Dr. Brian Sheron, Director, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission
- Mr. Lake Barrett, Principal, LBarrett Consulting, LLC
- Dr. John Boice, Scientific Director, International Epidemiology Institute
- Mr. Dave Lochbaum, Director, Nuclear Safety Project, Union of Concerned Scientists

Overview

In the United States, 104 operating nuclear reactors currently supply approximately 20 percent of U.S. electricity.[2] The majority of nuclear reactors came online throughout the 1970's and 80's, with the newest nuclear plant beginning generation in 1996. Currently, the Nuclear Regulatory Commission (NRC) is considering license applications for several new nuclear plants that industry is seeking to bring online over the coming decade. Southern Company is furthest along in this process, and is seeking a license from NRC to construct and operate two new nuclear reactors at its Vogtle site near Augusta, Georgia. These reactors would be the first in a new generation of nuclear plants in the United States.

The U.S. nuclear industry has experienced significant advancements in reactor safety and risk mitigation since the construction of the previous reactor. Recent events have refocused attention to the need for continual attentiveness to these issues.

Review of Japan

[1] Additionally, the Committee has jurisdiction over all environmental research and development, and the commercial application of energy technology, as well as all scientific research, development and demonstrations and projects. In addition to its legislative jurisdiction, the Committee is also tasked with the special oversight function of reviewing and studying on a continuing basis laws, programs, and Government activities relating to nonmilitary research and development.

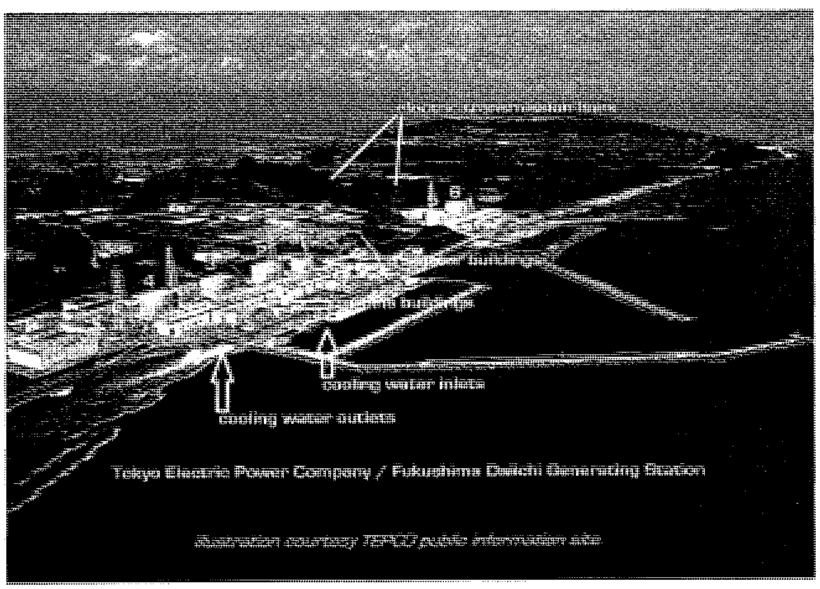
[2] "Nuclear Energy Quick Facts." Nuclear Energy Institute. 9 May 2011. http://www.nei.org/filefolder/Nuclear__Energy__Quick__Facts.pdf.

On March 11, 2011, a magnitude 9.0 earthquake struck just off Japan's east coast. The earthquake was the fourth largest recorded in the last century.[3] Compounding the devastation of the earthquake, a massive tsunami followed shortly after the initial earthquake and struck Japan's coast with little preparation time. The earthquake and resulting tsunami generated widespread destruction throughout the Japanese islands and is estimated to have killed over 10,000 people. Aftershocks continued for weeks impeding humanitarian response efforts.

The earthquake triggered the automatic shutdown of 11 of Japan's 55 operating nuclear power plants, as designed. Within close proximity to the earthquake's epicenter stood three sites with nuclear reactors, Onagawa, Fukushima Daiichi, and Fukushima Daini. Of the six nuclear units located at the Fukushima Daiichi site, three were in operation on March 11 while the remaining three units were shut down for inspections and maintenance.

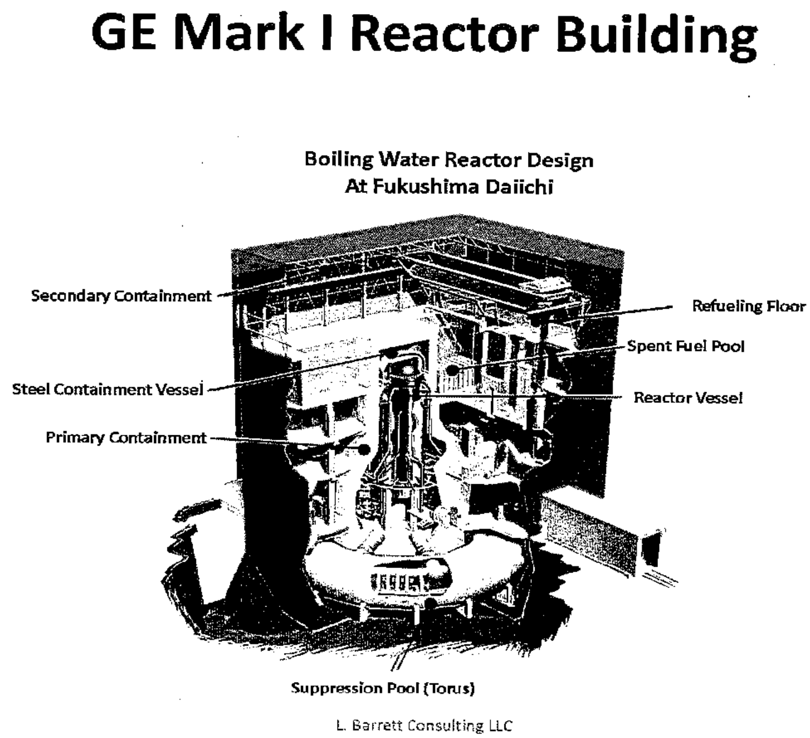
While further investigation is necessary to assess the specific consequences of the earthquake inside the reactors, it is believed all of the Daiichi reactors responded to the earthquake as intended. The site, cut off from the electric grid due to the earthquake, operated during this period as expected with the onsite backup diesel generators powering the cooling system for each reactor. Approximately one hour after the earthquake, an estimated 14 meter tsunami reached the Fukushima Daiichi site, overwhelmed the six meter high barrier, flooded the generators, swept away the diesel fuel tanks and eliminated all backup cooling systems located at the station (figure 1).

Figure 1 – Layout of the Fukushima Daiichi Nuclear Power Plant



[3] "Largest Earthquakes in the World Since 1900." *U.S. Geological Survey.* 9 May 2011. http://earthquake.usgs.gov/earthquakes/world/10__largest__world.php.

<u>Figure 2 – GE Mark 1 Reactor Building</u>



Lacking the ability to cool the reactors, Tokyo Electric Power Company (TEPCO), the owner of the Daiichi reactors, immediately began to experience severe difficulties associated with rising temperatures in the reactors. Absent primary and secondary cooling systems, TEPCO began to cool the reactor cores by pumping seawater into the reactors. Lacking the necessary information on the status of the reactor cores, water levels in the units dropped, resulting in partial exposure of fuel rods inside the reactor vessel (figure 2). As the fuel rods were exposed, the fuel rod's zirconium cladding reacted with water and generated hydrogen, which accumulated within the unit. The hydrogen buildup within the reactors ultimately led to explosions in Units 1, 2 and 3 within days of the tsunami and removed the secondary containment structures of those units.

In addition to the difficulties TEPCO faced stabilizing the cooling systems for Units 1, 2 and 3, the spent fuel pool located inside Unit 4 experienced problems. Unit 4 was undergoing maintenance at the time of the earthquake and had offloaded additional fuel rods in the spent fuel pool. While details are still not clear, in the days following the earthquake multiple fires ignited inside Unit 4 as a result of problems with the spent fuel pool. Investigation into the cause of the fires and specific spent fuel pool issues in Unit 4 are ongoing.

TEPCO continues to pump freshwater into the reactors at Units 1, 2 and 3. Further evaluation of the site's infrastructure is necessary prior to reconnecting electricity to the reactor and stabilizing the reactor cooling process. TEPCO is shooting water aimed at Unit 4's spent fuel pool to ensure the pool is adequately filled. Radiation levels surrounding the reactors remain elevated; however, they have notably decreased from spikes following the initial explosions.

Public Health Implications

Immediately following the tsunami and explosions at the Fukushima Daiichi reactors, the Japanese government ordered the evacuation of a 20 kilometer (12 mile) area surrounding the plant and directed those living within 30 kilometers (18 miles) to stay indoors. Japanese health authorities immediately began testing Japanese citizens, particularly children, for traces of radiation, but found only minimal levels of exposure. As of April 27, 2011, over 175,000 people have been screened. Radiation levels in the food supply were also evaluated and some restrictions were placed on distribution. Testing and evaluation of public health is ongoing and continue to be closely monitored. Workers at the Fukushima Daiichi plant were exposed to higher than normal radiation, though under the emergency dose limit set by the Japanese government and not enough to induce sickness. TEPCO rotates employees once the workers reach the permitted dose threshold.

As a consequence ofthe overheating of reactor fuel at Fukushima Daiichi Units 1, 2 and 3 and overheating within spent fuel storage areas, radiation was released into the atmosphere and environment. In the weeks following the release, traces of radiation were detected over portions of the United States. The trace amounts of radiation led to public discussion regarding the advisability of purchasing potassium iodide (KI) pills to prevent uptake of radioactive potassium and the possibility of radioactive material entering the food chain.[4] Of particular note, despite a lack of evidence suggesting human health would be impacted in the United States, U.S. Surgeon General Dr. Regina Benjamin noted in response to questioning about citizens stocking up on potassium iodide that such actions were "definitely appropriate" precautions to take.

The spread of radiation has refocused attention on the need for appropriate evacuation plans in the event of an accident or natural disaster at a nuclear facility, for appropriate plans for the return ofpopulations to evacuated areas, the efficacy of KI distribution and long-term health implications for exposure to low-dose radiation.[5]

Evaluations of U.S. nuclear safety

The nuclear industry and governmental bodies consistently review nuclear reactor safety and risk mitigation measures in the United States. However, the 1979 accident at Three Mile Island and the attacks of September 11, 2001, in particular, spurred significant reviews of and enhancements to nuclear reactor safety.

Previous reviews provide context for current and future evaluations of nuclear energy, such as the review currently underway by the NRC in response to the incident in Japan.

Three Mile Island

On March 28, 1979, a series of mechanical and human errors led to the most significant accident in the history of the U.S. nuclear power industry. For reasons still unknown, water pumps feeding the generator shut down. Because operators had closed valves on the secondary water system for routine maintenance, the system could not pump any water and the reactor began to overheat. A relief valve opened automatically to relieve primary system pressure; however, the valve failed to close once pressure had been released, allowing coolant water to escape. Compounding the problem was the failure of plant operators to recognize the opened valve and a misinterpretation of readings on the control panel.[6] Once operators realized the problem, serious damage had already occurred. When the core was opened four years later it was discovered that half the fuel rods had melted—a partial meltdown.[7]

In response to Three Mile Island, President Carter chartered the Kemeny Commission to investigate the accident. The Commission's recommendations covered a wide range of issues. One recommendation of note was for the nuclear power industry to establish a program that "specifies appropriate safety standards including those for management, quality assurance, and operating procedures and practices,

[4] For more information on radiation health implications and dose levels see Congressional Research Service Report titled, *"The Japanese Nuclear Incident: Technical Aspects,"* R41728

[5] Mason, Julie. "Fears Cause Run on Pills." *Politico* 16 Mar 2011. 9 May 2011. http://www.politico.com/politico44/perm/0311/a__run__on__iodide__9de5fce3-9807-44b1-9721-48d1b9abab2e.html.

[6] "Backgrounder on the Three Mile Island Accident." *Nuclear Regulator Commission.* http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html. Retrieved May 5, 2011.

[7] Gilinsky, Victor (March 23, 2009). "Behind the scenes of Three Mile Island". *Bulletin of the Atomic Scientists.* http://thebulletin.org/web-edition/features/behind-the-scenes-of-three-mile-island. Retrieved March 31, 2009.

and that conducts independent evaluations."[8] Further, "there must be a system gathering, review, and analysis of operating experience at all nuclear power plants coupled with an industry-wide international communications network to facilitate the speedy flow of this information to affected parties." [9]

As a consequence of that recommendation, the nuclear power industry established the Institute of Nuclear Power of Operations (INPO) and directed INPO to "promote the highest levels of safety and reliability—to promote excellence—in the operation of commercial nuclear power plants."[10] INPO continues to actively engage in a partnership with industry to provide valuable safety and risk mitigation expertise.

September 11, 2001

After the attacks of September 11, 2001 the NRC issued a series of orders and advisories to its license holders directing them on specific threats and security enhancements. For example, the NRC has issued orders requiring license holders to increase specific security measures, including: "increased patrols, augmented security forces and capabilities, additional security posts, installation of additional physical barriers, vehicle checks at greater stand-off distances, enhanced coordination with law enforcement and military authorities, and more restrictive site access controls." In addition, the NRC has made several changes to its Design Basis Threat (DBT), first implemented after the Three Mile Island accident in 1979. Although the DBT is not public, it outlines specific threats and characteristics of adversaries. In April 2003 and March 2006, the NRC made additions to the DBT with lessons learned from September 11. In January 2007, the DBT was further amended to consolidate previous additions and incorporate specific threat factors outlined in the Energy Policy Act of 2005.[11]

DOE and NRC Nuclear Energy Research Programs

Both the United States Department of Energy (DOE) and the NRC fund extensive research programs across a wide variety of topics. DOE and NRC conduct significant research focused on all components of nuclear facility safety, risk analysis, and reactor design. Given recent events, the manner in which government research programs inform reactor safety and regulations are integral to ensure public health and safety.

Nuclear Regulatory Commission

The Office of Nuclear Regulatory Research (NRR) is NRC's primary research entity, coordinating research and informing regulatory decisions for the organization. The NRR provides all encompassing research relating to reactor safety, operational regulations, environmental radiological impact, and performance and reliability. The NRR office consists of Program Management, Policy Development and Analysis Staff; the Division of Engineering; Division of Systems Analysis; and Division of Risk Analysis. The primary responsibility of NRR is to provide "leadership and plan, recommend, manage, and implement programs of nuclear regulatory research and interface with all NRC Offices and the Commission on research issues."[12]

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[8] *"Report Of The President's Commission On The Accident At Three Mile Island."* 1979. 9 May 2011. http://www.pddoc.com/tmi2/kemeny/utility__and__its__suppliers1.htm.

[9] Ibid

[10] "About." *Institute of Nuclear Power Operations.* Web. 9 May 2011. http://www.inpo.info/AboutUs.htm.

[11] "NRC's Response to the 9/11/01 Events." *Nuclear Regulatory Commission.* 25 Apr 2011. http://www.nrc.gov/security/faq-911.html.

[12] All NRR and Division responsibilities are summarized from: United States. *Office of Nuclear Material Safety and Safeguards*., 20 Apr 2011. Web. 9 May 2011. http://nrc.gov/about-nrc/organization/nmssfuncdesc.html.

Funding Levels (In Millions)

Major Programs	FY 2010 Enacted	FY 2012 Request
Operating Reactors-Research	72.7	70.4
New Reactors-Research	23.2	13.7
Nuclear Reactor Safety Research Subtotal	**95.9**	**84.1**
Fuel Facilities-Research	0.3	0.3
Nuclear Materials Users-Research	1.2	1.4
Spent Fuel Storage and Transportation-Research	1.3	5.9
Decommissioning and Low-Level Waste-Research	1.5	0.8
High-Level Waste Repository-Research	0.0	0.0
Nuclear Materials and Waste Safety Subtotal	**4.3**	**8.4**
Total	**100.2**	**92.5**

Among NRR's tasks, the Office:

- Recommends regulatory actions to resolve ongoing and potential safety issues for nuclear power plants and other facilities regulated by the NRC;
- Conducts research to reduce uncertainties in areas of potentially high safety or security risk or significance;
- Develops the technical basis for risk-informed, performance-based regulations in all areas regulated by the NRC;
- Leads the agency's initiative for cooperative research with DOE and other Federal agencies, the domestic nuclear industry, U.S. universities, and international partners;
- Maintains technical capability to develop information for resolution of nuclear safety and security issues and provides technical support and consultation to the Program Offices in the specialized disciplines involved in these issues and;
- Collects and analyzes operational data; assesses trends in performance from this data; evaluates operating experience to provide insights into and improve the understanding of the risk significance of events, precursors and trends; and produces and disseminates periodic performance indicator and Accident Sequence Precursor (ASP) Reports. [13]

The various divisions provide valuable, informative research relating to reactor safety and risk mitigation. For example, the Division of Systems Analysis conducts research to quantify margins, reduce unnecessary burden, and reduce uncertainties for areas of potentially high risk or safety significance, supports identification of accident phenomena and assessment of anticipated safety issues in new and advanced reactors, and develops technical bases for dose limits in regulations. The Division of Risk Analysis develops, recommends, plans, and manages research programs relating to probabilistic risk assessments (PRA); develops and uses PRA-based methodologies, models, and analysis techniques, as well as other risk assessment techniques to determine overall risk; and supports agency efforts to use risk information in all aspects of regulatory decision making.

Department of Energy—Office of Nuclear Energy

The primary mission of the DOE Office of Nuclear Energy (NE) is to "advance nuclear power as a resource capable of meeting the Nation's energy, environmental, and national security needs by resolving technical, cost, safety, proliferation resistance, and security barriers through research, development, and demonstration as

[13] Ibid

appropriate."[14] The Fiscal Year (FY) 2011 continuing resolution provided $737 million for the Office of Nuclear Energy.

Funding Levels (In Millions)

Major Programs	FY 2010 Enacted	FY 2012 Request
Reactor Concepts RD&D*	169.0	125.0
Generation IV Nuclear Energy Systems	212.9	0.0
Fuel Cycle R&D	131.9	155.0
LWR SMR Licensing Technical Support	0.0	67.0
Nuclear Energy Enabling Technologies	0.0	97.4
NE TOTAL	**870.0**	**852.0**

*FY10 Reactor Concepts RD&D was directed to Next Generation Nuclear Plant

Unlike the NRC, NE's research, development, and deployment programs are not consolidated within one office, but rather undertaken throughout all of NE's program offices. Safety and risk mitigation activities span fuel cycle research, advanced reactor research, and light water reactor sustainability research. For example, future reactor designs have passive cooling systems to cool nuclear reactor cores even in the absence of electricity. The Westinghouse AP1000 reactor design, currently under consideration for licensing by the NRC, has a passive cooling system and Small Modular Reactors also incorporate the technology.

Idaho National Laboratory (INL) is DOE's lead nuclear energy research and development facility. Primary NE tasks undertaken at INL include nuclear safety analysis, irradiation services, nuclear operations, management of spent nuclear fuel, and biocorrosion offuels.[15] These efforts are carried out through funding from the various NE research programs. Located at INL are a munber of facilities providing world class research capabilities for DOE, such as the Advanced Test Reactor Complex which is also a DOE National Scientific User Facility. Significant additional NE R&D is carried out at other Federal facilities, such as Oak Ridge National Laboratory, Argonne National Laboratory, Los Alamos National Laboratory, and Savannah River Site.

DOE's Office of Health, Safety and Security includes the Risk Assessment Technical Experts Working Group to assist DOE with the use of "quantitative risk assessment in nuclear safety related activities." These activities "help DOE ensure that risk assessments supporting nuclear safety decisions are conducted in a consistent manner, or appropriate quality, properly tailored to the needs of the decisions they are intended to support and documented."[16]

The Modeling and Simulation Energy Innovation Hub, located at Oak Ridge National Laboratory, will create a Virtual Reactor (VR) to model and simulate a nuclear reactor. The VR aims to enhance the scientific understanding of fission and

[14] "Mission Statement." *U.S. Department of Energy.* 9 May 2011. http://nuclear.energy.gov/neMission.html.

[15] "Nuclear Energy." *Idaho National Laboratory*. 9 May 2011. https://inlportal.inl.gov/portal/server.pt/community/nuclear__energy/277.

[16] "Risk Assessment Technical Experts Working Group." *U.S. Department of Energy, Office of Health, Safety and Security.* 9 May 2011. http://www.hss.energy.gov/nuclearsafety/ns/rawg/.

reduce uncertainties associated with safety and risk. The capabilities can be used to assess and improve safety of existing reactors.[17]

Need for future reactor safety research, risk assessment, and accident mitigation

The incident at the Fukushima Daiichi reactors has highlighted the need for continual examination of safety and risk assessment in the United States. Policies and priorities undergoing heightened assessment include:

- *Spent fuel management.* What is the best and most secure method of storing spent nuclear fuel? In a spent fuel pool or dry cask storage? In a single centralized storage facility, such as the proposed, but now cancelled Yucca Mountain repository, or onsite at individual reactor locations, including at sites containing decommissioned reactors?
- *Risk assessment modeling and risk mitigation.* How can risk uncertainty be reduced to the greatest degree and incorporated into risk mitigation measures? What are the necessary inputs to produce the most realistic risk assessment models?
- *Reactor design.* What design features may warrant incorporation into the new reactors to make nuclear reactors inherently more safe and resilient to natural disasters? Do different reactor technologies offer additional safety and risk mitigation benefits?
- *Emergency planning.* Are current Emergency Planning Zones adequate? Are the lines of communication between stakeholders clear and proper? Are additional steps to ensure public health safety necessary?
- *Response.* How can response capabilities be improved in the event of a disaster? What R&D is needed in this area?

[17] "Advanced Modeling and Simulation." *U.S. Department of Energy, Office of Nuclear Energy.* 9 May 2011. http://www.ne.doe.gov/AdvModelingSimulation/casl.html.

Chairman BROUN. Good morning. This joint hearing of the Subcommittee on Investigations and Oversight and the Subcommittee on Energy and Environment will come to order. I welcome everyone here to this hearing, "Nuclear Energy Risk Management." In front of you are packets containing the written testimony, biographies, and truth of testimony disclosures for today's witnesses.

Before we get started, since this is a joint hearing involving two Subcommittees, I want to explain how we will operate procedurally so all Members understand how the question and answer period will be handled. As always, we will alternate between the Majority and Minority Members, and allow all Members an opportunity for questioning before recognizing a Member for a second round of questions, if we have time for the second round. We will recognize those Members present at the gavel in order of seniority on the full Committee, and those coming in after the gavel would be recognized in the order of their arrival.

I now recognize myself for a five minute opening statement. I would first like to welcome our witnesses to today's hearing, and express my sincere appreciation for their effort in joining us here today.

Risk assessment and risk management associated with nuclear energy are timely and important topics for the Science Committee to address. This topic is clearly a priority for the Science Committee, as two of our Subcommittees are here today together. While the facts and implications of the Japanese earthquake, tsunami, and resulting nuclear disaster are still being determined, it is an opportunity for us to reassess our Nation's current safety posture here in this country.

After the Three Mile Island, Chernobyl, September 11, and several other incidents, the United States regularly revisited the state of our nuclear power infrastructure. Today's hearing is yet another opportunity to evaluate whether we, as a Nation, are doing everything that we can to ensure that nuclear energy is a safe component of our energy supply. This includes evaluating the current research and development portfolio for reactor safety, spent fuel storage, and public health monitoring.

The Department of Energy was invited to this morning's hearing and would have provided a valuable contribution to the hearing. Unfortunately, they were unable to provide a witness here today. DoE did provide written comments, but that does not substitute for actually appearing. Testifying is not a correspondence course. The Science Committee understands the many demands that agency officials have on their time. As Members of Congress, we have similar demands. Because of this, the Committee provided four weeks of notice and did not request a specific individual, leaving that determination to DoE. Unfortunately, it seems as though the entire Department only has one individual that they believe is qualified to speak on the issues that we are addressing here today, and he was otherwise engaged for multiple days.

While I find this troubling in and of itself, what is more frustrating is that this has now become a trend for this Administration. The TSA refused to testify at a hearing earlier this year before the I&O Subcommittee. Two days ago EPA refused to testify before the full Committee unless they could dictate the terms of their attendance.

Let me be clear. This Committee is willing to work with the Administration to reach neutral accommodations, but it will not allow it to obstruct our oversight efforts. We take our oversight responsibilities very seriously. This Administration's arrogance continues to undermine its claims of transparency and openness, particularly when they fail to be accountable to Congress and to the American people. If the Administration is not willing to work with this Committee, we have several options that can compel their cooperation. Unfortunately, it appears that we may have to exercise those options in the future.

For the witnesses that did appear today, I want to sincerely thank you for your cooperation.

[The prepared statement of Dr. Broun follows:]

PREPARED STATEMENT OF CHAIRMAN PAUL BROUN, M.D.

I would first like to welcome our witnesses to today's hearing and express my sincere appreciation for their effort in joining us here. Risk Assessment and Risk Management associated with Nuclear Energy are important and timely topics for the Science Committee to address. This topic is clearly a priority for the Science Committee as two of our Subcommittees are here together today. While the effects and implications of the Japanese earthquake, tsunami, and resulting nuclear disaster are still being determined, it is an opportunity for us to reassess our nation's current safety posture here in this country. After Three Mile Island, Chernobyl, September 11th, and several other incidents, the United States regularly revisited the state of our nuclear power infrastructure. Today's hearing is yet another opportunity to evaluate whether we, as a nation, are doing everything we can to ensure that nuclear energy is a safe component of our energy supply. This includes evaluating the current research and development portfolio for reactor safety, spent fuel storage, and public health monitoring.

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For the witnesses that did appear today, I want to sincerely thank them for their cooperation. I look forward to their testimony, and will now recognize Ms. Edwards, the Ranking Member of the Investigations and Oversight Subcommittee for an Opening Statement.

Chairman BROUN. The Chair now recognizes Ms. Edwards for an opening statement.

Ms. EDWARDS. Thank you, Mr. Chairman, and good morning. I look forward to today's hearing and thank the witnesses, because

I think for far too long we have heard just a drum beat about how nuclear energy is both safe and efficient, with electricity produced "too cheap to meter." I want to thank the Chairman for giving Members a chance to get to the bottom of these claims and others.

The idea of nuclear power as a cost effective source of power can be traced back to a statement in 1954 by the then-Chairman of the Atomic Energy Commission, who suggested that "Our children will enjoy in their homes electrical energy too cheap to meter." Unfortunately that same year, of course, General Electric ran an advertisement which I am attaching to my statement—it is quite interesting—from 1954 that optimistically trumpeted how the industry would be on its own two feet within five to ten years. That was in 1954. After suggesting that the big question on atomic energy was whether it could be done economically, the ad says, and I quote, "We already know the kinds of plants which will be feasible, how they will operate, and we can estimate what their expenses will be. In five years, certainly within ten, a number of them will be operating at about the same cost of those using coal. They will be privately financed and built without government subsidy." So here we are and it is 2011, and the reality is that nuclear power has always required government subsidies. In the almost 60 years since that ad appeared, the taxpayer has seen more than $80 billion spent on nuclear power research and development. In fact, it is the largest single energy research area since 1948. There are billions and billions and billions of dollars in other subsidies created through government actions designed to distort markets to give nuclear power a competitive edge over other sources of energy, although we are in a discussion now about how heavily subsidized the oil industry is.

Despite decades of support, nuclear power plants are still unable to operate competitively in the United States energy market, and now we are being asked for still more subsidies to build another generation of plants. According to an analysis by the Union of Concerned Scientists, these subsidies could be worth twice as much as the value of the electricity produced by the plant. That strikes me as throwing a lot of good money after bad.

We recently held a hearing on renewable energy in which the Majority seemed to want to make the point that subsidizing renewable energy would be picking winners and losers, and yet that same strategy that energy produced would not be competitive without government support is being used with respect to the nuclear industry.

Well, if you truly reject such support, the nuclear power industry should be the poster child for an industry that needs government to profit up, and profit up to the tune of billions of dollars. I support subsidies to help emerging energy sources such as wind and solar and battery technologies. They deserve at least as much of a chance as nuclear has had, and since nuclear cannot stand on its own feet after 60 years, it is time to say enough. The public gravy train has got to come to a stop for now for this mature industry, and it is indeed a mature industry, it just can't stand on its own, and its claims of safety, the events of Japan's Fukushima plant illustrate how safety is contingent on a complex set of systems all working perfectly. If those systems go down, system safety starts to slip beyond our control. Natural disasters and human folly know no national bounds, and it would be beyond arrogant to think that something similar to Fukushima could not happen here in the United States.

To avoid another accident requires aggressive regulators, safety-minded operators, and perfect luck. As was illustrated in a recent New York Times article, attached also to my statement, operators often confuse profit margins with safety margins and regulators are too passive or overwhelmed to always enforce accountability. In fact, there are claims that the regulatory agency is too cozy with the industry.

A recent report from the Union of Concerned Scientists documents 14 near-misses in just the past year, including one at Maryland's own Calvert Cliffs plant, located approximately 50 miles from where we sit today. Calvert Cliffs has two reactors. In February 2010, both reactors were automatically shut down. The cause of the shutdown was that water had shorted out a degraded piece of electric equipment that had neither been inspected or replaced. A subsequent study—investigation by the NRC revealed that the water resulted from chronic roof leaks. In fact, the NRC found that there were 58 outstanding work orders to repair roof leaks, and despite some of the orders being two years old, not one of them had even been scheduled for repair.

Each shutdown, like the one at Calvert Cliffs, caused plant owners and ultimately rate payers an average of more than $1.5 billion. Since the Three Mile Island accident, safety failures have resulted in plant shutdowns costing more than $80 billion. So we subsidize the energy—the industry's creation, the building of plants the production of electricity, and then we subsidize a failure of plant management. I think enough is enough, and with that, I yield.

[The prepared statement of Ms. Edwards follows:]

PREPARED STATEMENT BY THE HONORABLE DONNA F. EDWARDS

I look forward to today's hearing because for too long we have heard a drumbeat about how nuclear energy is both safe and efficient, with electricity produced "too cheap to meter." I want to thank the Chairmen for giving Members a chance to get to the bottom of these claims.

The idea of nuclear power as a cost-effective source of power can be traced back to a statement in 1954 by the then-Chairman of the Atomic Energy Commission who suggested that "Our children will enjoy in their homes electrical energy too cheap to meter. . ." That same year, General Electric ran an advertisement that optimistically trumpeted how the industry would be on its own two feet within five to ten years. After suggesting that the big question on atomic energy was whether it could be done *economically*, the ad says:

> "We already know the kinds of plants which will be feasible, how they will operate, and we can estimate what their expenses will be. In five years—certainly within ten—a number of them will be operating at about the same cost as those using coal. They will be privately financed, built without government subsidy."

The reality is that nuclear power has always required government subsidies. In the almost sixty years since that ad appeared, the taxpayer has seen more than $80 billion spent on nuclear power research and development. In fact, it is the largest single energy research area since 1948. And there are billions and billions and billions of dollars in other subsidies created through government actions designed to distort markets to give nuclear power a competitive edge. Subsidies include the Price-Anderson Act, which caps nuclear plant operators exposure to costs that would come from an accident, loan guarantees to underwrite the capital costs of plants, tax exempt bonds for construction of public plants, no charges to plants for their use of water and the list goes on and on.

Despite decades of support, nuclear power plants are still unable to operate competitively in the U.S. energy market. Now, we are being asked for still more subsidies to build another generation of plants. According to an analysis by the Union of Concerned Scientists, these subsidies could be worth twice as much as the value of the electricity produced by the plants. That strikes me as throwing good money after bad.

We recently held a hearing on renewable energy in which the Majority seemed to want to make the point that subsidizing renewable energy would be "picking winners and losers" or distorting the market and that the energy produced would not be competitive without government support. Well, if you truly reject such support, the nuclear power industry should be the poster child for an industry that needs government to prop it up.

I do not oppose subsidies to help new energy sources get on their feet I believe we should be investing in wind and solar and battery technologies and exploring other potential renewables to give them a chance to demonstrate their value to meeting our country's energy needs. They appear to be safer to the public and the environment than any other sources of electricity and they promise true energy independence without worries about proliferation of nuclear materials. They deserve at least as much of a chance as nuclear has had, and since nuclear cannot stand on its own feet after sixty years, it is time to say "enough." The public gravy train has got to come to a stop for this now mature industry.

As to c1airus of safety, the events at Japan's Fukushima plant illustrate how safety is contingent on a complex set of systems all working perfectly. If those systems go down, safety starts to slip beyond our control. Natural disasters and human folly know no national bounds and it would be beyond arrogant to think that something similar to Fukushima could not happen here.

The risks posed by nuclear power are unique in their potential health and environmental scope. In the last thirty years, we have had three catastrophic accidents of varying effect: Three Mile Island, Chernobyl, and Fukushima. To avoid another accident requires aggressive regulators, safety-minded operators, and perfect luck. As was illustrated in a recent New York Times article, operators often confuse profit margins with safety margins and regulators are too passive or overwhelmed to always enforce accountability.

To keep the public safe from disaster, you have to get nuclear plant safety right every second of every day of every year and everywhere. And natural disasters cannot be allowed to interfere or those carefully calibrated perfect systems can fail. I think that this is an impossible standard, but a failure once a generation or so is not acceptable to me. In fact, the Union of Concerned Scientists has issued a report documenting 14 near misses just in the past year, including one at Maryland's own Calvert Cliffs plant.

Located approximately 50 miles from where we sit today, Calvert Cliffs has two reactors. In February 2010, both reactors were automatically shut down. The cause of the shut-down was that water had shorted out a degraded piece of electrical equipment that had neither been inspected nor replaced. And the water, as a subsequent NRC investigation revealed, was the result of chronic roof leaks. In fact, the NRC found that there were 58 outstanding work orders to repair roof leaks. Despite some of the orders being two years old, not one of them had even been scheduled for repair.

I am sure that a nuclear advocate would point to Calvert Cliffs' automatic shutdown as a "success." But such successes, in which safety systems shut reactors down in the face of systems operating out of spec, are not cost free. Each shutdown costs plant owners, and ultimately rate payers, an average of more than $1.5 billion dollars. Since the Three Mile Island accident safety failures that resulted in plant shutdowns cost more than $80 billion.

So we subsidized the industry's creation, the building of plants, the production of electricity and then we subsidize the failures of plant managers.

I think enough is enough.

Chairman BROUN. Thank you, Ms. Edwards.

I now recognize the Chairman of the Subcommittee on Energy and Environment, Dr. Harris, for his opening statement.

Dr. Harris, you are recognized for five minutes.

Dr. HARRIS. Thank you very much, Mr. Chairman. I want to thank our witnesses also for being here today to testify on issues relating to nuclear energy risk management, and I do look forward to hearing from all your testimony.

First I would like to echo Dr. Broun's disappointment with the Department of Energy's inability to provide a witness for the hearing. I do recognize that the head of the Office of Nuclear Energy was unavailable due to international travel, but I would hope that in a program with a budget of over $850 million that the Department has more than one individual qualified to represent it before Congress.

The purpose of this hearing is to examine nuclear energy safety, risk assessment, and public health protection. Nuclear energy is clearly an integral piece of America's energy portfolio today, and will probably continue to be in the future.

In Maryland, my State, one-third of our electricity is generated by nuclear reactors, and the State is home to two reactors located near my district at Calvert Cliffs.

DoE's Energy Information Administration projects that U.S. electricity demand will increase by 31 percent over the next 25 years. We simply have to get this electricity from somewhere, and nuclear energy may indeed provide a clean, safe, and affordable source of base load power to meet this demand. However, as with all critical energy sources, producing nuclear energy is certainly not without risk, and we must take great care to appropriately manage those risks. The March earthquake and tsunami in Japan clearly serves as a stark reminder of this. However, it is important to note that the incident and response at Fukushima did not happen in a vacuum. Both the nuclear industry and government regulators continually assess safety measures and mitigate those kind of risks. Largely due to this diligence and attentiveness, nuclear facilities in this country are among the safest workplaces across all industries, and not a single death has been attributed to nuclear energy production here in the United States.

As I hope to hear today, continued improvements in reactor design and operating procedures will make what is already safe nuclear energy even safer. To this end, I am interested in learning how the Federal Government can best prioritize its nuclear energy research programs to further reduce these risks.

I am also interested in key policy questions associated with nuclear energy risk management. For example, is a Fukushima-like event even possible here in the U.S., given our regulatory environment and reactor design? Do facilities pre-stage the necessary equipment to manage unexpected incidents? What are the comparative risks associated with storage of spent nuclear fuel scattered throughout the country or consolidated into centralized storage, such as Yucca Mountain.

Finally, as a medical doctor by training, I believe it is important to objectively and responsibly discuss potential radiologic effects on public health. Senior government officials encouraging Americans to stockpile potassium iodide pills due to detection of miniscule traces of radiation is simply not responsible, since potassium iodide can obviously have harmful results if those pills are unnecessarily taken. This kind of alarmism also feeds unnecessary public fears about nuclear energy potentially harming its future viability.

I hope the witnesses can help provide perspective on this issue. I look forward to hearing today's discussion surrounding these topics. Again, I thank you all for appearing. I thank the Chairman for holding the hearing, and I yield back.

[The prepared statement of Dr. Harris follows:]

PREPARED STATEMENT OF CHAIRMAN ANDY HARRIS

I thank our witnesses for being here today to testify on issues relating to Nuclear Energy Risk Management and I look forward to hearing your testimony. First, I would like to echo Dr. Broun's disappointment with the Department of Energy's inability to provide a witness for this hearing. I recognize that the head of the Office of Nuclear Energy was unavailable due to international travel, but I would hope that with a program budget of over $850 million, the Department has more than one individual qualified to represent it before Congress.

The purpose of this hearing is to examine nuclear energy safety, risk assessment, and public health protection. Nuclear energy is an integral piece of America's energy portfolio today and will continue to be in the future. In Maryland, one third of our electricity is generated by nuclear reactors and the state is home to two reactors located near my district, at Calvert Cliffs.

DOE's Energy Information Administration projects that U.S. electricity demand will grow by 31 percent in the next 25 years. We have to get this electricity from somewhere, and nuclear energy provides a clean, safe, and affordable source of baseload power to meet this demand.

However, as with all critical energy sources, however, producing nuclear energy is not without risk, and we must take great care to appropriately manage these risks. The March earthquake and tsunami in Japan serves as a stark reminder of this.

However, it is important to note that both the incident and the response at Fukushima did not happen in a vacuum. Both the nuclear industry and government regulators continually assess safety measures and mitigate risk. Largely due to this diligence and attentiveness, nuclear facilities are among the safest workplaces across all industries, and not a single death has ever been attributed to nuclear energy production in the United States. As we will hear today, continued improvements in reactor design and operating procedures will make nuclear energy even safer. To this end, I'm interested in learning how the Federal government can best prioritize its nuclear energy research to further reduce risks.

I'm also interested in key policy questions associated with nuclear energy risk management. For example: Is a Fukushima-like event even possible in the U.S.? Do facilities pre-stage the necessary equipment to manage unexpected incidents? What are the comparative risks associated with storage of spent nuclear fuel-scattered throughout the country or consolidated into centralized storage, such as Yucca Mountain?

Finally, as a medical doctor by training, I believe it is important be responsible when discussing potential radiological effects on public health. Senior government officials encouraging American citizens to stockpile potassium iodide pills due to detection of miniscule traces of radiation is not responsible, and can have harmful results if those pills are unnecessarily taken. This alarmism also feeds unnecessary public fears about nuclear energy, potentially harming its future viability. I hope the witnesses can help provide perspective on this issue.

I look forward to hearing today's discussion surrounding these topics. Thank you and I yield back.

Chairman BROUN. Thank you, Dr. Harris.

If there are Members who would like to submit additional opening statements, your statements will be added to the record at this point.

At this time I would like to introduce our panel of witnesses. Dr. Brian Sheron, is that correct, Director, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission; Mr. Lake Barrett, Principal Consultant, Barrett Consulting, LLC; Dr. John Boice, Scientific Director, International Epidemiology Institute and Professor of Medicine, Vanderbilt University School of Medicine; and Mr. Dave Lochbaum, Director of Nuclear Safety Project, Union of Concerned Scientists.

As our witnesses should know, spoken testimony is limited to five minutes and I would ask you, because we are really pressed, we are going to have votes about 9:45 to 10 o'clock, so please limit your testimony to five minutes. If you can shave a few seconds off that, we would appreciate it, but we don't want to shortchange you, either. After your spoken testimony, Members of the Subcommittees will have five minutes each to ask questions. Your written testimony will be included in the record of the hearing.

It is the practice of the Subcommittee on Investigations and Oversight to receive testimony under oath, and we will use that practice today as well. Do any of you have any objection to taking an oath? If you shake your head, it will be fine.

Let the record reflect that all witnesses have shaken their heads from side to side, indicating that they have no objection to taking an oath.

You may also be represented by counsel. Do any of you have counsel here today?

Let the record reflect that none of the witnesses have counsel, indicated by them shaking their heads from side to side.

If you would now please stand and raise your right hand. Do you solemnly swear and affirm to tell the whole truth and nothing but the truth, so help you God?

Let the record reflect that all witnesses participating have taken the oath. Thank you. You may sit down.

I now recognize our first witness, Dr. Brian Sheron, Director of the Office of Nuclear Regulatory Research at the Nuclear Regulatory Commission, NRC. Dr. Sheron, you are recognized for five minutes.

STATEMENT OF DR. BRIAN SHERON, DIRECTOR, OFFICE OF NUCLEAR REGULATORY RESEARCH, NUCLEAR REGULATORY COMMISSION

Dr. SHERON. Thank you. Good morning Chairmen Harris and Broun, Ranking Members Miller and Edwards, Members of the Subcommittees. I am pleased to appear before you on behalf of the United States Nuclear Regulatory Commission, the NRC, to discuss the Agency's research program and our current activities in response to the events that have occurred at the Fukushima-Daiichi nuclear power plant site.

My name is Brian Sheron. I have been the Director of the NRC's Office of Nuclear Regulatory Research for the past five years, and have been at the NRC and its predecessor agency, the Atomic Energy Commission, for nearly 38 years.

The following testimony is intended to provide an overview of NRC's Office of Nuclear Regulatory Research, or RES, and its current activities, as well as provide a discussion of the Agency taskforce and research activities related to the Fukushima-Daiichi event in Japan.

Office of Research is a major NRC program office mandated by Congress and created along with the NRC in 1975. The NRC's regulatory research program addresses issues in the areas of nuclear reactors, nuclear materials, and radioactive waste. My office plans, recommends, and implements programs of nuclear regulatory research, standards development, and resolution of generic issues for nuclear power plants and other facilities regulated by the NRC. There are currently about 260 staff members in my office.

We do not conduct research for the primary purpose of developing improved technologies. That is a function that is more appropriately the nuclear industry's. Rather, the NRC conducts research to confirm that the methods and data generated by the industry ensure that adequate safety margins are maintained.

We work with the offices that are responsible for licensing activities within the NRC to develop appropriate regulatory actions to resolve potential safety issues for nuclear power plants and other facilities regulated by the NRC, including those issues designated as generic issues. Generic issues are potential technical or security issues that could impact two or more facilities.

My office coordinates the development of consensus and voluntary standards for agency use, including appointment of Agency staff to numerous domestic and international standards committees. Participation by the NRC staff in consensus standards development is essential because the codes and standards are an integral part of the Agency's regulatory framework.

We have implemented over 100 international cooperative agreements with other nuclear regulators and international organizations to share information and leverage resources. We also participate extensively in several International Atomic Energy Agencies, and Organization for Economic Cooperation and Development Nuclear Energy Agency committees and working groups that facilitate the exchange of information between countries on topics such as risk assessment, events, and best practices.

The NRC has a robust reactor operating experience program, and we have taken advantage of the lessons learned from previous operating experience to implement a program of continuous improvement for the U.S. reactor fleet. As you know, on Friday, March 11, 2011, an earthquake and subsequent tsunami occurred near the northeast coast of Japan, resulting in the shutdown of more than 10 reactors. From what we know now, it is likely that the earthquake caused the loss of normal alternating current power and it is likely that the reactor's response to the earthquake went as designed. The ensuing tsunami, however, caused the loss of emergency A/C power to four of the six units at the Fukushima site.

The phenomena associated with the events at Fukushima involved numerous disciplines in which my office has expertise and has done substantial research. I would now like to discuss some of these technical areas that have been raised since the events.

The Office of Research has a seismic research program that is currently addressing updated geological assessments, particularly in the central and eastern United States. We have also initiated a current tsunami research program in 2006, and our tsunami research leverages work being done at the United States Geological Survey and the National Oceanographic and Atmospheric Administration. This will help form the basis for NRC review of new license applications.

We have performed significant severe accident research since the TMI accident to better understand the phenomena and improve both accident prevention and mitigation.

The NRC has been using probabilistic risk assessment, or PRA methods to obtain estimates of risk associated with severe accidents since 1975.

The NRC has previously studied spent fuel pool issues and implemented additional requirements to minimize spent fuel pool vulnerabilities. Following the events in Japan, we have begun to update spent fuel pool studies to estimate the relative consequence of removing older fuel from the spent fuel pool and placing it into dry storage, versus leaving it in the spent fuel pool.

In conclusion, I want to reiterate that the NRC has a very robust regulatory research program that performs confirmatory research to allow the licensing offices to make technically-informed regulatory decisions. The research office has expertise in a multitude of technical disciplines and has performed significant research in the past related to reactors, materials, and waste.

In light of the events in Japan, the NRC has initiated a near-term evaluation of the event's relevance to reactors in the U.S. and we are continuing to gather the information necessary for us to take a longer, more thorough look at the events and their lessons for us. Based on the lessons learned from these efforts, we will pursue additional regulatory actions and research as needed to ensure the continuing safety of the U.S. fleet.

Thank you.

[The prepared statement of Dr. Sheron follows:]

PREPARED STATEMENT OF DR. BRIAN SHERON, DIRECTOR, OFFICE OF NUCLEAR REGULATORY RESEARCH, NUCLEAR REGULATORY COMMISSION

Good morning, Chairmen Harris and Broun, Ranking Members Miller and Edwards, and Members of the Subcommittees. I am pleased to appear before you on behalf of the United States Nuclear Regulatory Commission (NRC) to discuss the agency's research program and our current activities in response to the events that have occurred at the Fukushima-Daiichi nuclear power plant site.

My name is Dr. Brian Sheron, and I have been the Director of the NRC Office of Nuclear Regulatory Research for the past five years and have been at the NRC and its predecessor agency, the Atomic Energy Commission, for nearly 38 years.

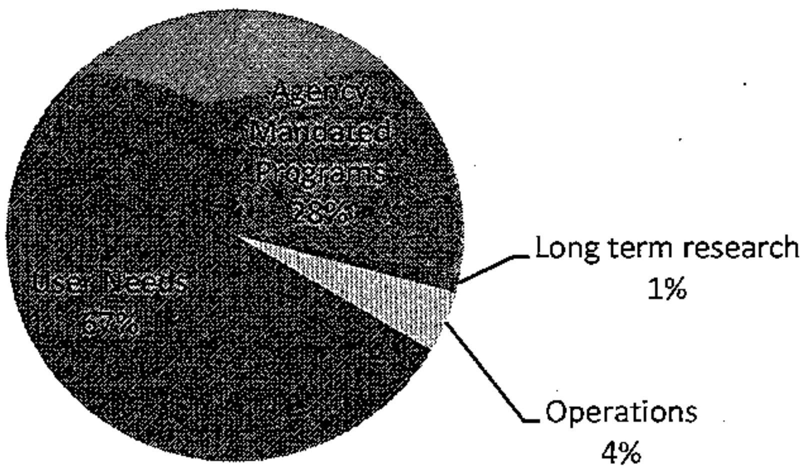
The following testimony is intended to provide an overview of the NRC's Office of Nuclear Regulatory Research (RES) and its current activities, as well as provide a discussion of the agency task force and research activities related to the Fukushima-Daiichi event in Japan.

As you are aware, the NRC is an independent Federal agency established to license and regulate the Nation's civilian use of production and utilization facilities, as well as the use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, to promote the common defense and security, and to protect the environment. The NRC currently licenses, inspects, and assesses the performance of 104 operating nuclear power plants, as well as many materials licensees, fuel cycle facilities, and research and test reactors.

The Office of Nuclear Regulatory Research (or RES) is a major NRC program office, mandated by Congress and created along with the NRC in 1975. RES is one of the offices that reports to the Executive Director for Operations. RES plans, recommends, and implements programs of nuclear regulatory research, standards development, and resolution of generic safety issues for nuclear power plants and other facilities regulated by the NRC. The Office coordinates research activities within and outside the agency, including NRC participation in national and international volunteer standards efforts. There are currently about 260 staff members in the office, which is organized into three technical divisions: the Division of Engineering, Division of Risk Analysis, and Division of Systems Analysis.

RES is responsible for developing methods, technical expertise and computer codes that are used by the NRC to assess safety and regulatory issues for materials licensees, fuel cycle facilities, operating reactors as well as new and advanced reactor designs. We develop the data needed to assess these codes by conducting experiments at national laboratories, universities, or in collaboration with international organizations.

The NRC regulatory research program addresses issues in the three arenas of nuclear reactors, nuclear materials, and radioactive waste. The research program is designed to improve the agency's knowledge where uncertainty exists, where safety margins are not well-characterized, and where regulatory decisions need to be confirmed in existing or new designs and technologies. Typically, the regulatory offices approach us with an issue, and we determine how to appropriately resolve it through research or analysis. The majority of our work is this user need driven work performed in response to requests from our regulatory offices, as shown in the following chart:



RES coordinates research activities with the other NRC program offices, as appropriate, and leads the agency's initiative for cooperative research with the U.S. Department of Energy (DOE) and other Federal agencies, the domestic nuclear industry, U.S. universities, and international partners. RES coordinates the development of consensus and voluntary standards for agency use, including appointment of agency staff to various standards committees. Based on research results and experience gained, we work with the regulatory offices to develop appropriate regulatory actions to resolve potential safety issues for nuclear power plants and other facilities regulated by the NRC, including those issues designated as Generic Issues (GIs). GIs are technical or security issues that could impact two or more facilities or licensees. RES also develops the technical basis for those areas regulated by the NRC that have risk-informed, performance-based regulations.

RES supplies technical tools, analytical models, and experimental data needed to support the agency's regulatory decisions. RES does not conduct research for the primary purpose of developing improved technologies, a function that is more appropriately that of the Department of Energy or the nuclear industry. Rather, the NRC conducts research to confirm that the methods and data generated by the industry ensure that adequate safety margin is maintained.

In addition to supporting regulation of the commercial use of radioactive materials to protect public health and safety and to protect the environment, RES is responsible for providing the technical basis for regulations to ensure the protection and safeguarding of nuclear materials and nuclear power plants in the interest of national security. Thus, while its primary focus is on supporting the licensing and regulatory process, the research conducted by and for the NRC plays an important role in supporting broad government-wide initiatives associated with national security.

The Office of Research's staff is very well qualified and educated, with 30% of staff holding PhDs, and 33% of staff with master's degrees. The staff continues to reflect diversity in education, demographics, and technical disciplines. The wide range of engineering and scientific disciplines includes expertise in nuclear engineering, materials science, human factors and human reliability, health physics, fire protection, and probabilistic risk assessment, to name a few. It is this diversity in highly technical and specialized disciplines that allows RES to support the licensing offices as they carry out their licensing and regulatory tasks. Given this internal expertise, we perform a significant amount of research in-house. However, because we have more work than RES staff's capacity, we use contractors to supplement our work to perform research that requires special skills or facilities. Our staff develops the work plan and is engaged in the research process with the contractor throughout the entire research effort.

In addition to conducting confirmatory research, RES also conducts forward-looking research. The objectives of forward-looking and long-term research are to develop the technical basis to support related regulatory decision making. We monitor areas where the regulated industry may be moving and determine the technical information needed for future regulatory decisions to prepare the agency to respond to anticipated future industry requests and initiatives.

These activities address new safety technologies or developments in analytical technologies or infrastructure. By their nature, these items span a wide range of disciplines, from risk assessment to structural integrity to fission product transport. Our development of data and assessment tools for these technologies will ensure that the agency is prepared to meet its future regulatory needs.

In addition to our research efforts, the NRC cooperates with professional organizations that develop voluntary consensus standards associated with systems, structures, equipment, or materials used by the nuclear industry. In fiscal year 2010, 184 NRC staff members participated in 325 standards activities, such as membership on a standards-writing committee. The organizations governing these committees include the American Society of Mechanical Engineers (ASME), the National Fire Protection Association (NFPA), the American Nuclear Society, the Institute of Electrical and Electronics Engineers, the American Concrete Institute, and the National Council on Radiation Protection and Measurements.

For example, ASME developed the Boiler and Pressure Vessel Code and the Operations and Management Code which are widely acknowledged as an acceptable set of standards used to design, construct, and inspect pressure-retaining components, including nuclear vessels, piping, pumps, and valves. Similarly, NFPA has developed consensus standards to define acceptable methods to design, install, inspect, and maintain fire protection systems. The NRC has incorporated into its regulations various standards from the groups discussed above.

The NRC's use of voluntary consensus standards is consistent with statutory requirements. Participation by the NRC staff in voluntary consensus standards development is essential because the codes and standards are an integral part of the agency's regulatory framework. The benefits of this active involvement include cost savings, improved efficiency and transparency, and regulatory requirements of high technical quality. The agency acknowledges the broad range of technical expertise and experience of the individuals who belong to the many consensus standards organizations. Thus, participation in standards development minimizes the expenditure of NRC resources that would otherwise be necessary to provide guidance with the technical depth and level of detail of voluntary consensus standards.

Over the past 35 years, RES has developed or sponsored over 40 computer codes for use in its safety analyses. These codes are used in many aspects of the NRC's mission and perform wide ranging tasks including modeling fuel and reactor systems behavior, radiation's health effects, atmospheric dispersion, probabilistic risk assessment and more. They are shared with domestic and international counterparts to capture the value of a larger expert user community, which adds robustness to the codes and certainty to their results.

NRC uses computer codes to model and evaluate fuel behavior, reactor kinetics, thermal-hydraulic conditions, severe accident progression, time-dependent dose for design-basis and beyond design-basis accidents, health effects, and radionuclide release and transport during various operating and postulated accident conditions. Computer codes are validated against scaled tests and actual plan data. Results from such code applications support regulatory decision making in risk-informed activities, confirmatory and exploratory analyses, review of licensees' codes, performance of audit calculations, and resolution of other technical issues to inform the NRC staff on a wide variety of emergent technical questions for ensuring the health and safety of the general public. NRC code development is focused on improving the realism, accuracy and reliability of code results while improving code usability. However, the modeling of some novel systems (e.g., medical isotopes production) and new and advanced reactor design (e.g., Next Generation Nuclear Plant) requires further code development and additional assessment against specific experimental data.

Some specific examples of codes and how they are more specifically used in the regulatory environment are the MELCOR and MACCS2 codes. The MELCOR code models the progression of severe accidents in light-water nuclear power reactors.

MELCOR models several phenomena including thermal-hydraulics, core heatup, containment performance, hydrogen production, and fission product release and transport behavior. The MACCS2 code is used to evaluate doses and health risks from the accidental atmospheric releases of radio nuclides. It is also used to confirm license renewal analyses regarding plant specific evaluation of Severe Accident Mitigation Alternatives (SAMAs) that is required as part of the environmental assessment for license renewal. The MACCS2 code is also routinely used in environmental impact statements (EIS) supporting early site permits (ESP).

The agency shares its codes with other organizations under various agreements and has organized user groups for some codes that are widely used. Two such programs are the Code Applications and Maintenance Program (CAMP) and the Cooperative Severe Accident Research Program (CSARP). CAMP, which has existed as a user community for almost 30 years, includes thermal-hydraulic codes, and has members from more than 25 nations. CSARP includes members from 20 nations who focus on the analysis of severe accidents using primarily the MELCOR code. Through the CAMP and CSARP programs, the NRC is able to share some of the codes' development and maintenance cost, while improving their quality and performance.

RES has implemented over 100 international cooperative agreements with other nuclear regulators and international organizations to share information and leverage resources. RES also participates in several International Atomic Energy Agency (IAEA) and Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) committees and working groups that develop safety standards and facilitate the exchange of information between countries on topics such as risk assessment, events and best practices. These include the IAEA Nuclear Safety Standards Committee, the Committee on the Safety of Nuclear Installations, the Working Group on Risk Assessment, and others. In addition, I serve as vice-Chair for the Committee on the Safety of Nuclear Installations at the OECD/NEA.

The NRC has a robust reactor operating experience program, and we have taken advantage of the lessons learned from previous operating experience to implement a program of continuous improvement for the U.S. reactor fleet. We have learned from experience across a wide range of situations, including, most significantly, the Three Mile Island (TMI) accident in 1979. As a result of those lessons learned, we significantly revised emergency planning requirements and emergency operating procedures for licensees, and made substantive improvements in NRC's incident response capabilities. We also addressed many human factors issues regarding control room indicators and layouts, added new requirements for hydrogen control to help prevent explosions inside of containment, and created requirements for enhanced control room displays of the status of pumps and valves.

Two particularly significant changes after TMI accident were the expansion of the Resident Inspector Program and the incident response program. Today, there are at least two Resident Inspectors at each nuclear power plant. The inspectors have unfettered access to all licensees' activities, and serve as NRC's eyes and ears at the power plant. The NRC headquarters operations center and regional incident response centers are prepared to respond to all emergencies, including any resulting from operational events, security events, or natural phenomena. Multidisciplinary teams in these centers have access to detailed information regarding licensee facilities, and access to plant status information through telephonic links with the Resident Inspectors, an automated emergency response data system, and directly from the licensee over the emergency notification system. NRC's response would include the dispatch of a site team to supplement the Resident Inspectors on site, and integration with the licensee's emergency response organization at their Emergency Offsite Facility. The program is designed to provide independent assessment of events, to ensure that appropriate actions are taken to mitigate the events, and to ensure that State officials have the information they would need to make decisions regarding protective actions.

The NRC had a Boiling Water Reactor Mark I Containment Improvement Program in the 1990's, which resulted in the installation of hardened vent systems for containment pressure relief, as well as enhanced reliability of the automatic depressurization system.

As a result of the events of September 11, 2001, we identified important pieces of equipment that, regardless of the cause of a significant fire or explosion at a plant, we want licensees to have available and staged in advance, as well as new procedures, training requirements, and policies that would help deal with a severe situation.

As you know, on Friday, March 11, 2011, an earthquake and subsequent tsunami occurred near the northeast coast of Japan, resulting in the shutdown of more than 10 reactors. From what we know now, it appears possible that the reactors' response to the earthquake went according to design. The ensuing tsunami, however, likely caused the loss of emergency alternating current (AC) power to four of the six units at the Fukushima Daiichi site. It is these four units that have received the majority of our attention since that time. Units One, Two, and Three at the site were in operation at the time of the earthquake. Units Four, Five, and Six were in previously scheduled outages.

Our program of continuous improvement based on operating experience will include evaluation of the significant events in Japan and what we can learn from them. We have already begun enhancing inspection activities through temporary instructions to our inspection staff, including the Resident Inspectors at each nuclear power plant and the region-based inspectors in our four Regional offices, to look at licensees' readiness to deal with both the design basis accidents and the beyond-design basis accidents. The information that we gather will be used for additional evaluation of the industry's readiness for similar events, and will aid in our understanding of whether additional regulatory actions need to be taken in the immediate term.

The phenomena associated with the events at Fukushima-Daiichi involve numerous disciplines in which RES has expertise and are in areas where we have already done substantial analysis. I would now like to discuss some of these technical areas that have been raised since the events in Japan and discuss our related existing or planned research activities.

First, the NRC has an extensive seismic research program. Seismic safety in the design and operation of nuclear facilities has been evolving since the development of the first rules and guidance for seismic design by the NRC's predecessor, the Atomic Energy Commission. In 1998, the NRC issued a policy decision to move towards a risk-informed and performance-based regulatory framework. Risk-informed frameworks use probabilistic methods to assess not only what can go wrong, but also the likelihood of going wrong. Over the last decade, significant advances have been made in the ability to assess seismic hazards. The NRC is currently sponsoring several projects in support of both an updated assessment of seismic hazards in the Central and Eastern United States (CEUS) and an enhancement of the overall framework under which the hazard characterizations are developed. The products of these projects will be used in the determination of seismic hazard design levels for new reactors and are being used in a program to reassess seismic hazards at existing plant locations. Although no immediate safety issue has been identified, the NRC will take action if our further analysis shows that safety improvements can be justified.

Since the 2004 Indian Ocean tsunami, significant advances have been made in the ability to assess tsunami hazard globally. The NRC initiated its current tsunami research program in 2006. It focuses on bringing the latest technical advances to the regulatory process and exploring topics unique to nuclear facilities. The tsunami research program focuses on several key areas: landslide-induced tsunami hazard assessments, support activities associated with the licensing of new nuclear power plants in the United States, development of probabilistic methods, and development of the technical basis for new NRC guidance. This program, which includes cooperative work with the United States Geological Survey (USGS) and the National Oceanic and Atmospheric Administration (NOAA), has already resulted in several important publications on tsunami hazard assessments on the Atlantic and Gulf Coasts of the United States. The publications and research results help form the basis of NRC review of new license applications. Whether additional work is needed for operating reactors will also be examined.

The NRC has performed extensive research since the TMI accident to understand the phenomena associated with severe accidents and has developed analytical models that predict accident progressions and their consequences. This research includes test programs on zirconium fires, source term analysis, molten core-concrete interactions, and containment analyses.

The NRC is conducting research to estimate the possible public health and safety consequences in the unlikely event that a severe accident occurs at a commercial nuclear power plant in the United States. The State-of-the-Art Reactor Consequence Analysis (SOARCA) program takes maximum advantage of extensive national and international reactor safety research and reflects improved plant design, operation, and accident management implemented over the past 25 years. Using computer models and simulation tools, the NRC is developing a set of realistic consequence estimates of accidents at two U.S. reactor sites representative of different reactor and containment designs used in the United States. The two pilot plants are a General Electric boiling-water reactor (BWR) with a Mark I containment (Peach Bottom) and a Westinghouse pressurized-water reactor (PWR) with a dry, sub-atmospheric containment (Surry). The results of the analyses are showing thus far that analyzed scenarios could reasonably be mitigated, either preventing core damage or delaying or reducing the radiation release. For cases assumed to proceed unmitigated, accidents appear to progress more slowly than previously thought and usually result in smaller and more delayed radiological releases than previously predicted.

A Probabilistic Risk Assessment (PRA) is a structured analytical process that provides estimates of risk by (1) identifying potential initiating event scenarios that can challenge system operations, (2) estimating the likelihood of event sequences that lead to an adverse event such as core damage, containment failure, and offsite radiological effects; and (3) estimating the consequences associated with accident sequences. These rankings are very valuable in the sense that resources can be directed towards the major contributors to risk. There are three levels of PRA for nuclear power plants. Level 1 PRA covers the initiating event to the onset of core damage. Level 2 PRA covers the onset of core damage to radioactive material release to the environment. Level 3 PRA covers radioactive material release to offsite radiological consequences.

The first study to use PRA methods to obtain more realistic estimates of risk associated with severe reactor accidents was completed in 1975. In 1988 the NRC asked the licensees to conduct Individual Plant Examinations to ensure that NRC's regulations were adequate and no undue risk was posed to the public by any plant. In 1990, NRC completed a Level 3 PRA for five commercial nuclear power plants of different reactor and containment designs. Since this last NRC-sponsored Level 3 PRA, the design, operation, maintenance, testing, and inspection of NPPs and the state-of-the-art in PRA technology, and data have evolved considerably. Our staff therefore continues to improve NRC's PRA capability and risk understanding to enhance PRAs role in NRC's current risk-informed regulatory approach.

The NRC has developed independent confirmatory PRA models for operating and new reactor nuclear plants. The NRC maintains Standardized Plant Analysis Risk (SPAR) models that represent the 104 operating commercial plants in addition to 2 SPAR models for new reactor designs. These SPAR models are used to support a variety of NRC regulated activities including the reactor oversight and the accident precursor programs. The SPAR models are updated periodically to reflect plant modifications, new operating experience data, and improved risk modeling capabilities (e.g., support system initiating events, external hazards, and loss of offsite power).

As part of the PRA program, the NRC conducts human reliability analysis (HRA) research to assess the human contribution to risk. We study human performance because it can significantly influence the reliability and safety of nuclear plant operations. HRA research is key to understanding accident sequences and appropriately representing their relative importance to overall risk. Research is conducted both domestically and internationally in cooperation with other organizations. In addition, the NRC participates in and I am the Board Chairman of the OECD/NEA Halden Reactor Project. Halden is a research facility in Norway that advances HRA through research. Several regulatory agencies and private sector companies participate in Halden research activities. NRC continues to study human performance in nuclear power plants and improve the methods for assessing human reliability.

Another PRA based program that measures risk is the Accident Sequence Precursor (ASP) Program. The NRC established ASP in 1979 after the TMI accident. The ASP Program systematically evaluates U.S. nuclear power plant operating experience to identify, document, and rank the operating events most likely to lead to inadequate core cooling and severe core damage (precursors), given the likelihood of additional failures.

The ASP Program provides (1) a comprehensive, risk-informed view of nuclear power plant operating experience and a measure for trending core damage risk; (2) a partial check on dominant core damage scenarios predicted by probabilistic risk assessments; and (3) provides feedback to regulatory activities. The NRC also uses the ASP Program to monitor performance against the safety goal established in the agency's strategic plan and report significant precursors to Congress.

The NRC has previously studied spent fuel pool (SFP) issues and modified licensee requirements in various areas such as an aircraft impact assessment, loss of SFP cooling, modifications to assembly configurations, and additional requirements following the attacks of September 11, 2001. As a result of the recent events in Japan, an updated SFP safety study to estimate the relative consequences of removing older fuel from the SFP and placing it into dry storage versus leaving it in the spent fuel pool is being considered.

Beyond the initial steps to address the experience from the events in Japan, the NRC staff has established a senior level agency task force to conduct a methodical and systematic review of our regulatory processes to determine whether the agency should make any improvements to our regulatory system and to make recommendations to the Commission for its policy direction. This activity will have both near-term and longer-term objectives.

For the near-term effort, we have started a 90-day review. This review will evaluate the currently available information from the Japanese events to identify immediate or near-term operational or regulatory issues potentially affecting the 104 operating reactors in the United States, including their spent fuel pools. Areas of investigation will include: the ability to protect against natural disasters; response to station blackouts; severe accidents and spent fuel accident progression; and severe accident management issues. Over this 90-day period, the task force will develop recommendations, as appropriate, for changes to inspection procedures and licensing review guidance, and recommend whether generic communications, orders, or additional regulations are needed.

This 90-day effort includes a briefing to the Commission after approximately 30 days to provide a snapshot of the regulatory response and the condition of the U.S. fleet based on information it has available at that time. This briefing, which occurred on May 12, also ensured that the Commission is both kept informed of ongoing efforts and prepared to resolve any policy recommendations that surface. However, over the 90-day and longer-term efforts the task force will seek additional stakeholder input. At the end of the 90-day period, a report will be provided to the Commission and to the public in accordance with normal Commission processes, and it will be provided to the Advisory Committee on Reactor Safeguards for its review. The task force's longer-term review will begin as soon as the NRC has sufficient technical information from the events in Japan.

The task force will evaluate all technical and policy issues related to the event to identify additional potential research, generic issues, changes to the reactor oversight process, rulemakings, and adjustments to the regulatory framework that should be pursued by the NRC. The task force is also expected to evaluate potential interagency issues, such as emergency preparedness, and examine the applicability of any lessons learned to non-operating reactors and materials licensees. The task force is expected to seek input from stakeholders during this process. A report with appropriate recommendations will be provided to the Commission within 6 months of the start of this evaluation. Both the 90-day and final reports will be made publicly available in accordance with our regulatory decision making. The NRC has expertise in a multitude of technical disciplines and has performed significant research in the past related to reactors, materials, and waste. In light of the events in Japan, the NRC has initiated a near-term evaluation of the events' relevance to the U.S. nuclear power plants, and we are continuing to gather the information necessary for us to take a longer, more thorough look at the events and their lessons for us. Based on the lessons learned from these efforts, we will pursue additional regulatory actions and research, as needed, to ensure the continuing safety of the U.S. fleet.

Summary of Testimony of Dr. Brian Sheron, Director, Office of Nuclear Regulatory Research at the U.S. Nuclear Regulatory Commission to House Committee on Science, Space, and Technology, Subcommittee on Energy and Environment, and Subcommittee on Investigations and Oversight

May 13, 2011

The United States Nuclear Regulatory Commission (NRC) is an independent Federal agency established to license and regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, to promote the common defense and security, and to protect the environment. The Office of Nuclear Regulatory Research (RES) is a major NRC program office, mandated by Congress and created along with the NRC in 1975. The NRC regulatory research program addresses issues in the arenas of nuclear reactors, nuclear materials, and radioactive waste. RES plans, recommends, and implements programs of nuclear regulatory research, standards development, and resolution of generic issues for nuclear power plants and other facilities regulated by the NRC. We work with the regulatory offices to develop appropriate regulatory actions to resolve potential safety issues for nuclear power plants and other facilities regulated by the NRC.

RES coordinates the development of consensus and voluntary standards for agency use, including appointment of agency staff to numerous domestic and international standards committees. RES implements international cooperative agreements with other regulators and organizations to share information and leverage resources. RES participates extensively in several International Atomic Energy Agency (IAEA) and Organisation for Economic Cooperation and Development Nuclear Energy Agency (OECD/NEA) committees and working groups that facilitate the exchange of information.

The phenomena associated with the events at Fukushima-Daiichi involve numerous disciplines in which RES has expertise and has done substantial analysis. NRC has seismic and tsunami research programs that use current geologic information to assess the risks to each plant. The NRC has been using probabilistic risk assessment (PRA) methods to obtain estimates of risk for several years, and the Research staff continues to improve our PRA capabilities and risk understanding to enhance PRA's role in NRC's current risk-informed regulatory approach. RES has developed independent confirmatory PRA models for operating and new reactor nuclear plants, and we update the models to reflect the current design of the plant. As part of our PRA efforts, RES conducts human reliability analysis (HRA) research to assess the human impact to risk. This research is conducted both domestically and internationally in cooperation with other organizations.

The NRC has previously studied spent fuel pool (SFP) issues and implemented additional requirements to minimize SFP vulnerabilities. Following the event in Japan, we have begun an updated SFP study to estimate the relative consequences of removing older fuel from the SFP and placing it into dry storage versus leaving it in the SFP. In addition, Chairman Jaczko, with the full support of the Commission, directed the staff to establish an agency task force to conduct a systematic review of our regulatory processes to determine whether the agency should make additional improvements.

Brian W. Sheron
Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Brian W. Sheron is currently the Director of the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research.

In this position, Dr. Sheron oversees the agency's regulatory research programs, which help resolve important safety issues, evaluate industry initiatives, enhance understanding of new technologies, identify needed enhancements to NRC regulations and contribute to a risk-informed, performance-based regulatory framework. The office also works cooperatively with the U.S. Department of Energy and other federal agencies, the U.S. nuclear industry and universities, and international partners.

Dr. Sheron joined the NRC in 1976 as a nuclear engineer in the Office of Nuclear Reactor Regulation (NRR). From 1980 to 1987, he served in a number of progressively more responsible positions within NRR including Deputy Director, Division of Safety Review and Oversight. From 1987 to 1994, he worked in RES before returning to NRR, where he served as Director, Division of Engineering and Associate Director for Project Licensing and Technical Analysis. Most recently, Dr. Sheron served as Associate Director for Engineering and Safety Systems.

Prior to joining the NRC, Dr. Sheron worked on the Clinch River Breeder Reactor Project with the former Atomic Energy Commission and former Energy Research and Development Administration.

Dr. Sheron graduated from Duke University in 1969 with a B.S. degree in Electrical Engineering. He received his M.S. degree in Nuclear Engineering in 1971 and his Ph.D. in Nuclear Engineering in 1975, both from the Catholic University of America.

Enclosure 3

Committee on Science, Space, and Technology

U.S. House of Representatives

Witness Disclosure Requirement - "Truth in Testimony"
Required by House Rule XI, Clause 2(g)(5)

1. Your Name: BRIAN WALTER SHERON

2. Are you testifying on behalf of the Federal, or a State or local government entity?	Yes ✓	No
3. Are you testifying on behalf of an entity that is not a government entity?	Yes	No ✓

4. Other than yourself, please list which entity or entities you are representing:

U.S. NUCLEAR REGULATORY COMMISSION

5. Please list any Federal grants or contracts (including subgrants or subcontracts) that you or the entity you represent have received on or after October 1, 2008:

NONE.

6. If your answer to the question in item 3 in this form is "yes," please describe your position or representational capacity with the entity(ies) you are representing:

7. If your answer to the question in item 3 is "yes," do any of the entities disclosed in item 4 have parent organizations, subsidiaries, or partnerships that you are not representing in your testimony?	Yes	No

8. If the answer to the question in item 3 is "yes," please list any Federal grants or contracts (including subgrants or subcontracts) that were received by the entities listed under the question in item 4 on or after October 1, 2008, that exceed 10 percent of the revenue of the entities in the year received, including the source and amount of each grant or contract to be listed:

I certify that the above information is true and correct.

Signature: Brian W. Sheron Date: May 6, 2011

Chairman BROUN. Thank you very much, Dr. Sheron.

I now recognize our next witness, Mr. Lake Barrett, Principal of L. Barrett Consulting, LLC.

STATEMENT OF MR. LAKE BARRETT, PRINCIPAL, L. BARRETT CONSULTING, LLC

Mr. BARRETT. Thank you very much, Mr. Chairman, Ms. Edwards, and Chairman Hall. I appreciate the opportunity to be here before you today.

I would like to just quickly try to go through what happened at Fukushima-Daiichi plant. It is a large, six reactor facility on the northeast coast of Japan. On March 11, there was a huge earthquake. The earthquake was slightly beyond the design basis of the plant, but the safety systems all performed satisfactorily there. There was a greater-than-designed basis tsunami, a huge wave that surrounded the plant as you can see in the lower right-hand corner, and when it hit, it took out all the emergency A/C power in the plant. They were able to cool the core for about eight hours using a backup system that was operated with batteries. After about eight hours the battery power exhausted and there was no more cooling, and the core started to uncover and overheat. As the core overheated, it started to melt and there was a steam cloud interaction producing hydrogen. This led to an over-pressurization. The primary containment was vented to the secondary containment and there was hydrogen gas in that. That led to an explosion in the Unit 1 building, and then there was another explosion in the Unit 3 building.

Units 1, 2, and 3 were operating at the time of the earthquake and tsunami. The operators started to inject seawater to cool the core and through the feed and bleed operation, and they are doing that to this day now. They are working to restore recirculation cooling. They also have had to spray water up onto the spent fuel pools, which are in the upper areas, with fire trucks in the beginning. They now have an injection boom with a concrete injection pump.

Thirty years ago at Three Mile Island there was another accident that had core degradation also. There were entirely different reasons for the accident at Three Mile Island. It was the Unit 2 reactor which is in the foreground on this photo. At Three Mile Island, it was an operator misunderstanding of the reactor system. There was an abnormal shutdown and a valve stuck open. The operators thought it was closed and the operators thought there was too much water in the reactor, when in reality there was not enough. They turned off the emergency pumps and this led to the core being overheated. It melted approximately a little over half of the core. This is what I expect we will find at Fukushima when they eventually get inside. Hydrogen gas was generated. The hydrogen gas did have a deflagration event, but it was contained primarily within the reactor building. There was about a half a million gallons of highly radioactive water on the floor of the containment building. This would be a sequence of how the core would melt and redistribute down toward the bottom of the vessel, which again, as reported last night from Japan, is a situation like in Unit 1.

At Three Mile Island, sophisticated clean up systems were installed, and the spent fuel pool, which was empty. Special refueling tools were built and damaged fuel, the damaged fuel was placed in canisters. This was safely completed in about a decade, cost about $1 billion and about 3 million gallons of highly radioactive water was processed.

At Fukushima, they are still stabilizing the plant. It is not stable yet. They are looking to establish clean areas. They are working to mitigate the airborne releases, which are unmonitored. They are working to capture the 10-plus million gallons of highly radioactive water that is in the plant, and gain access. This is just an old picture of the four reactors that are severely damaged at Fukushima, another side angle where you can see some of the vapors coming off probably the spent fuel pools and the reactors, which are located down in the lower parts of the buildings. They are taking mitigative actions to mitigate the airborne effluents such as spraying resins and fixatives on the contaminated soil on the plant site. There is also the work to contain the tens of millions of gallons of highly radioactive water. They have robotic equipment trying to remove the highly radioactive debris from the site so they can gain access to the buildings inside. There is offsite contamination, but it is not that severe, but nonetheless it significant.

My observations on Fukushima: it is not a public health catastrophe, it certainly is an industrial plant catastrophe. The tsunami was the critical safety matter. I think Units 1 and 4 are a complete loss, but the cleanup, I believe, can be done. The technology is there. We had it 30 years ago at Three Mile Island, and it is much better today than it was back then. The Japanese have a strong technological society, and I believe they can handle this in the future, but they still have challenges. As far as U.S. plants, I believe they have adequate safety margins today. The tsunami risk was the main issue for safety. That is primarily limited to the northwest coast of the United States. We have no operating reactors there on the coast, but there are two shut down reactors that have spent fuel that is stored there, and that is a risk that probably shouldn't be there. But it is a small risk because it is in dry storage.

The United States has done a lot of work in severe response improvements over past decades, and I think that is a good basis for the United States, but we need to have a systematic, methodical risk informed, lessons learned evaluation. The industry is doing it, and so is the NRC. We should resist quick fix, emotional reactions to this until we get the facts and learn what has happened and what is the right course of action.

The lessons learned from Three Mile Island greatly improved U.S. nuclear safety and productivity. The most painful lessons are the most teachable lessons, and we had very painful lessons at Three Mile Island and we are undergoing one now with Fukushima. I believe history will probably look back, if we keep on a steady course, that Fukushima will improve our entire energy situation, improve safety and performance for the future, just like Three Mile Island did 30 years ago.

Thank you very much.

[The prepared statement of Mr. Lake Barrett, Principal of L. Barrett Consulting, LLC, follows:]

PREPARED STATEMENT OF MR. LAKE BARRETT, PRINCIPAL OF L. BARRETT CONSULTING, LLC

Chairman Broun, Chairman Harris, Ranking Member Edwards, and Ranking Member Miller, good morning and I am honored to appear before you today to present my views on the events surrounding the incident at the Fukushima Daiichi nuclear reactors in Japan, the current status of reactor safety in the United States, and how the events at Fukushima can inform policies and technology advancement to improve safety and risk management for nuclear facilities. I am presenting my views as a private person in the context of my experience as the Nuclear Regulatory Commission Site Director in charge of recovery and cleanup at Three Mile Island.

On March 11, 2011 a subduction slip fault, where the Pacific plate slides under the Japan plate, snapped and released a tremendous amount of energy causing a massive 9.0 earthquake that shocked the north east coast of Japan. The earthquake caused a massive. tsunami that hit the coast approximately one hour after the earthquake. This was reported as the largest earthquake to hit Japan in over the last 1,000 years. The earthquake and tsunami caused immense destruction throughout northern Japan destroying entire towns and killing over 20,000 persons with early damage estimates of over $300 billion.

The massive earthquake took down the northern Japan power grid causing the operating major power plants in the region to automatically shutdown. The Fukushima Daiichi power reactor complex was impacted by the earthquake and the three operating reactors there safely shutdown. Although the earthquake dynamic loading was reportedly slightly above the seismic design basis of the facility, there was no reported damage to safety systems and the shutdown appeared to function normally despite the massive earthquake. The emergency diesels started as designed and there was no reported significant structural damage to safety systems.

Approximately one hour after the earthquake, a massive 15 meter high tsunami hit the Fukushima Daiichi site and overwhelmed the tsunami protections that had a reportedly nominal design basis of 5.7 meters with the major facility buildings located approximately 10 meters high. This ultra high "mega" tsunami flooded all the emergency diesels, swept away their fuel supplies, and destroyed much of the electrical switch gear. This complete loss of AC power and destruction of electrical components resulted in an extended "station blackout" situation.

With the loss of all AC electric power, reactor Units 1, 2, and 3, which had automatically shutdown, were then cooled by their DC battery controlled backup cooling systems: an isolation condenser for the older Unit 1 reactor and the steam turbine driven Reactor Core Isolation Cooling system pumps at the newer larger Units 2 and 3. After approximately 8 hours these backup systems apparently failed and thus the operators were unable to remove the decay heat from the reactor cores. The operators and government officials declared a site emergency and initiated a phased evacuation and sheltering order in areas surrounding the site with a 30 KM radius.

With the loss of cooling, the reactor primary coolant system water in the reactor core started to boil away and increase the primary coolant system pressure. This led to either an automatic opening of the system overpressure protection relief valves or manual opening of the valves to relieve primary system pressure by releasing steam to the primary containment suppression pool in the basement of the reactor building, The continued loss of coolant lowered the reactor vessel water level such that the core became uncovered, but was bathed in superheated steam, With this loss of cooling water, the fuel cladding temperatures increased significantly until the zirconium alloy rods that encase the uranium fuel pellets over heated, became over pressurized, and likely burst As the temperatures further increased there was a chemical reaction between the zirconium alloy cladding material and the superheated steam, The chemical reaction was an oxidation of the zirconium metal by the oxygen in the steam which produced additional heat and also hydrogen gas, This release of additional gas and energy into the primary coolant system led to further over pressurization of the primary coolant system which in turn led to further release of steam, which now contained hydrogen and noble gas fission products, into the suppression pool and primary containment.

Since there were no cooling systems available to cool the primary containment system suppression pool, the water temperature of the suppression pool began to rise past the boiling point and the primary containment system pressure began to rise, At some point, likely around 5 atmospheres of pressure, the primary containment system was in danger of over pressurizing toward a possible structural failure, Although I do not know exactly what happened at this point, it appears that the operators manually released pressure from the primary containment to prevent a failure of the primary containment system, They were likely trying to vent the steam, hydrogen, and fission product gas mixture through filters and up the 100 meter ventilation stack, However, for some unknown reasons, there may have been leaks in the system or they may have intentionally vented the gas mixture into the reactor building (which serves as a secondary containment) trying to minimize releases of radioactive materials to the environment Regardless of the operator actions, the hydrogen gas apparently mixed with oxygen rich natural air in the reactor building resulting in an explosive gas mixture within the reactor building,

Some unknown ignition source ignited the explosive gas mixture resulting in the destruction of the roof and upper sides of the Unit 1 and Unit 3 reactor buildings, As expected, the hot gases rose toward the top of the reactor building doing the most damage to the upper areas, The primary containment system boundary in the lower levels of these reactor buildings seemed to not be seriously compromised and seemed to maintain their ability to contain and scrub fission products from hot radioactive effluents venting from the primary coolant system,

Although no details are yet available on specific mitigation actions that the operators were taking to cool the reactor cores and mitigate the release of radioactive releases, there was one heroic effort apparently made to prevent a hydrogen explosion in Unit 2 reactor building, The operators went into the Unit 2 reactor building and removed a side wall panel to allow hydrogen gas to naturally diffuse into the environment before it could build up to explosive levels and ignite, There was however, a reported explosion in the lower regions of the Unit 2 reactor building that likely damaged the primary containment; however information as to the situation there is not yet available,

Portable diesel power generators and fire engine pumps were brought into the site as soon as possible, however, the huge extent of earthquake and tsunami damage to the local area was a major delaying factor.

Eventually the operators were able to connect the fire truck pumps to directly inject seawater into the reactor cores of Units 1, 2 and 3 to start removing decay heat from the cores, thus likely preventing further core overheating and damage, Unfortunately, considerable damage was already done to the cores, but with the seawater and later freshwater injection to the cores, the situation seems to have stabilized.

At the time of the earthquake and tsunami the Unit 4 reactor was shut down for maintenance with its reactor core removed from the reactor vessel and placed in its spent fuel pool. Several days after the earthquake and tsunami there was one or more major explosions in the Unit 4 reactor building. At this point, I do not know the source of explosion energy. At an early time, it was theorized that the Unit 4 spent fuel pool may have overheated, but recent water samples from the Unit 4 pool do not indicate major fuel damage. So at this point, more information is necessary to determine what happened in Unit 4.

Although information is very sketchy, it seems based on water samples taken, there has been damage to spent fuel that is stored in the Units 2 and 3 spent fuel pools. Information as to what happened in these pools is still unavailable, so it is impossible to determine the significance at this time, but it certainly appears that something of significance occurred. Once information becomes available, a careful analysis should determine what happened and what are the appropriate lessons learned regarding spent fuel pool storage safety.

The U.S. Three Mile Island Unit 2 (TMI) accident back on the morning of March 28,1979 resulted in similar reactor core overheating and core damage similar to what has now happened to the cores in the Fukushima Units 1,2 and 3. The TMI accident led to localized core melting, hydrogen generation and release of radioactive materials from the reactor core, but for entirely different reasons. Although the physical core degradation mechanisms were similar for TMI and Fukushima, I expect that the primary safety lessons learned will be different because of different circumstances involved.

At Three Mile Island there was no natural catastrophe as at Fukushima. It was a major man-machine interface problem when, during an abnormal reactor shutdown, the reactor operators were not aware of a stuck open pressurizer relief valve, which had a faulty valve position indicator, resulting in the operators believing that there was too much water in the reactor when actually there was not enough. The operators stopped automatic water injection when they should not have done so. The lack of water injection led to the core becoming uncovered and grossly overheating. As in the Fukushima cores, the fuel cladding burst, chemically reacted with superheated steam, released hydrogen gas and melted approximately 50% of the reactor core. Again, as at Fukushima, the TMI primary coolant system over pressurized and radioactive steam, hydrogen gas, and radioactive fission products were released into the TMI reactor containment building. The hydrogen gas mixed with the oxygen in the air inside the TMI containment building and ignited in a deflagration burn wave pressure spike that was fully contained within the primary containment. At TMI there was no breach of the primary containment system.

Once the TMI operators realized what the reactor situation was, cooling water was immediately added and a sustainable core cooling function was restored later on the first day by operating the large main coolant pumps. Decay heat was then removed through the steam generators until cold shutdown was achieved.

The operation of the main coolant pumps required some highly contaminated primary coolant to be circulated into the Auxiliary Building which led to some radioactive gases being released into the Auxiliary Building ventilation system. Virtually all significant releases were contained within the reactor containment building.

There were approximately two and one half million gallons of highly radioactive water generated during the accident and recovery that needed to be cleaned up. All the accident related water was contained on site and special water processing systems were built to remove the radioactive fission products, primarily Cesium and Strontium. Eventually the processed accident water was safely discharged by evaporation.

The stabilization and cleanup of TMI took approximately a decade and cost approximately one billion dollars. Building accesses had to be established, ventilation system improvements made, high radiation areas mitigated, radioactive water removed, buildings decontaminated, building infrastructures (e.g. cranes) restored, access to the damaged reactor cores accomplished, special defueling systems deployed, packaging of damaged fuel and other highly radioactive waste products completed, temporary onsite storage facilities constructed, and eventual offsite shipment of the damaged fuel and radioactive wastes for lessons learned research and development accomplished. This was safely achieved with virtually no offsite environmental impacts.

There were no radioactive injuries or adverse health effects from the Three Mile Island accident and cleanup. Inadequate operator response to deficient control room instrumentation proved to be the root cause of the accident. The primary lessons learned from TMI was that a much better integration of the operator's understanding of the reactor systems was needed during off normal events. Major industry wide improvements were instituted which included creation of the Institute of Nuclear Power of Operations (lNPO) and risk informed regulatory processes. Thus the TMI lessons learned responses led to improved U.S. nuclear safety and improved reactor productivity.

It should be noted that the sister Three Mile Island Unit 1 reactor, which has a similar design to the damaged Unit 2, was restarted after a thorough lessons learned review and continues to operate safely today with one of the highest capacity factors in the country.

I believe Fukushima is nearing the end of their initial stabilization period and will hopefully soon be entering their recovery/cleanup and lessons learned phases. They are working to establish closed circuit core cooling for Units 1, 2 & 3 so that they do not continue to create large quantities of highly radioactive water containing fission products and continuing radioactive gas venting. In addition, they are working to mitigate the releases of contaminated water that has accumulated in all the reactor and turbine buildings by installing new water storage tanks and processing systems. Airborne releases are being mitigated by the installation of air filtration systems and spraying of resin fixatives to onsite areas that were highly contaminated by earlier airborne releases. Even though the radioactive effluent mitigation challenges are great, I expect they should be able to establish sufficient capability to minimize any future significant radioactive releases from the site.

In summary, it is my view that the public health consequences of the Fukushima accident should be infinitesimal when compared to the impact of the earthquake and tsunami. From a radiological perspective, this should be inconsequential from a national public health perspective. There are some areas to the northwest where Cesium and likely Strontium contamination has deposited and significant remediation challenges will have to be addressed.

From an overall reactor safety perspective, I expect that there will be much learned from Fukushima that will confirm present U.S. safety margins and should also provide information to further improve reactor safety in the coming years. The fundamental U.S. reactor safety level that exists today is likely to be demonstrated as adequate because there is a limited tsunami risk to most U.S. reactor facilities. The only significant tsunami risk area in the U.S. is in the Northwest Pacific coast where there are no exposed operating reactors, but there are two shutdown reactor sites which have stranded spent nuclear fuel in dry storage casks. In my view, I believe that all stranded spent fuel at shutdown reactor sites should be removed to completely eliminate 'all radiological risks at these decommissioned sites.

There are two other southern Pacific coast reactors, Diablo Canyon and San Onofre; however, I expect that further reviews will confirm there are adequate tsunami safeguards already in place at these sites that should demonstrate adequate facility safety.

In the U.S. a lot of attention has already been placed on severe accident mitigation over the last 25 years and especially since September 11, 2001. Many safety improvements have already been made which I believe will demonstrate that U.S. reactors are well prepared to withstand severe accidents regardless of the initiating event. So although a systematic methodical risk informed Fukushima lessons learned evaluation should be performed and enhancing improvements should be made, I expect that fundamental existing severe accident safety margins will basically be confirmed.

I strongly recommend that the U.S. lessons learned process be methodical, deliberate, risk informed and primarily led by private industry. The independent Nuclear Regulatory Commission will do their own safety reviews into the adequacy of their regulations with their own lessons learned function. NRC and DOE nuclear research programs should be adjusted as more is learned.

The NRC should resist political or emotional calls for quick actions in one area or another until a thoughtful, fully informed lessons learned analysis is completed based on facts and public health and safety significance. Of course, if some immediate safety issue is discovered requiring immediate action, the NRC has all the necessary authority to act as necessary, but only when a clear significant safety situation exists.

Three Mile Island lessons learned programs strengthened U.S. nuclear energy in many different ways. The most painful lessons are often the most teachable. Although we are just beginning to understand the Fukushima lessons, I firmly believe that they will further strengthen U.S. nuclear energy programs and other nuclear energy programs throughout the world.

Lake H. Barrett

Lake@Lbarrett.com

Lake Barrett is a part time independent consultant in the energy field. He has worked in the nuclear energy and nuclear materials management areas for over 4 decades, most recently as the former head of the US Department of Energy's Office of Civilian Nuclear Waste Management which is responsible for implementing the United Sates' programs for spent nuclear fuel and high-level radioactive waste, as mandated by the Nuclear Waste Policy Act. In that capacity, he lead the complex scientific Yucca Mountain Geologic Repository program through the statutory site selection process culminating with the Presidential site designation and following successful House and Senate votes.

Lake also served at U. S. Nuclear Regulatory Commission, where he was directly involved with the early response to the Three Mile Island reactor accident and became the Site Director, responsible for regulatory programs during the stabilization, recovery, and cleanup of the damaged reactor. He also has had extensive managerial and engineering experiences in DOE's Defence Programs and private industry at both Bechtel Power Corporation, with commercial nuclear power plants, and Electric Boat Division of General Dynamics with nuclear reactor and submarine systems design, operation, and decommissioning. He is a long time member of ANS and INMM.

Enclosure 3

Committee on Science, Space, and Technology
U.S. House of Representatives
Witness Disclosure Requirement - "Truth in Testimony"
Required by House Rule XI, Clause 2(g)(5)

1. Your Name: Lake H. Barrett		
2. Are you testifying on behalf of the Federal, or a State or local government entity?	Yes	No X
3. Are you testifying on behalf of an entity that is not a government entity? Testifying for myself only	Yes	No X
4. Other than yourself, please list which entity or entities you are representing: None		
5. Please list any Federal grants or contracts (including subgrants or subcontracts) that <u>you or the entity you represent have received</u> on or after October 1, 2008: Small contracts from Lawrence Livermore to teach & advise on International Fuel Mgmt & Savannah River as Independent reviewer on SNF long term storage R&D facilities tasks.		
6. If your answer to the question in item 3 in this form is "yes," please describe your position or representational capacity with the entity(ies) you are representing:		
7. If your answer to the question in item 3 is "yes," do any of the entities disclosed in item 4 have parent organizations, subsidiaries, or partnerships that you are not representing in your testimony?	Yes	No
8. If the answer to the question in item 3 is "yes," please list any Federal grants or contracts (including subgrants or subcontracts) that were received by the entities listed under the question in item 4 on or after October 1, 2008, that exceed 10 percent of the revenue of the entities in the year received, including the source and amount of each grant or contract to be listed:		

I certify that the above information is true and correct.
Signature: [signature] Date: May. 8, 2011

Chairman BROUN. Thank you very much, Mr. Barrett. We have been notified that we will be taking votes shortly, but what we are going to do is we are going to hear from the last two witnesses and then recess. We are going to go vote and we are going to come back for questions.

So I now recognize our next witness, Dr. John Boice, Scientific Director of the International Epidemiology Institute.

STATEMENT OF DR. JOHN BOICE, SCIENTIFIC DIRECTOR, INTERNATIONAL EPIDEMIOLOGY INSTITUTE

Dr. BOICE. Thank you, Mr. Chairman, ranking Members, and Members of the Subcommittee. I am a radiation epidemiologist, and I have spent my entire career studying populations exposed to radiation, from Chernobyl cleanup workers to populations living near nuclear power plants. I was in Hiroshima just a few days before the accident as a member of the Science Council of the Radiation Effects Research Foundation, reviewing the study of atomic bomb survivors.

Fukushima is not like Chernobyl. The Chernobyl accident resulted in massive radiation exposures. There was no containment vessel, and a fire burned for 10 days, spewing radioactive material into the environment. The first responders and the fire fighters received so much radiation that 28 died of acute radiation sickness within a few months. Radioactive iodines were deposited on large areas, and were ingested by grass-eating cows who gave milk that was drunk by children, and an epidemic of thyroid cancer resulted.

In contrast, Fukushima appears to have resulted in substantially lower worker and public exposures. The Japanese authorities raised the annual limit of worker exposure from 2 to 25 rem, but only 21 workers received more than 10 rem. These levels are far below the hundreds of rem needed to cause acute radiation sickness, but they are sufficient to increase the lifetime risk of developing cancer over their lifetimes by about 1 percent.

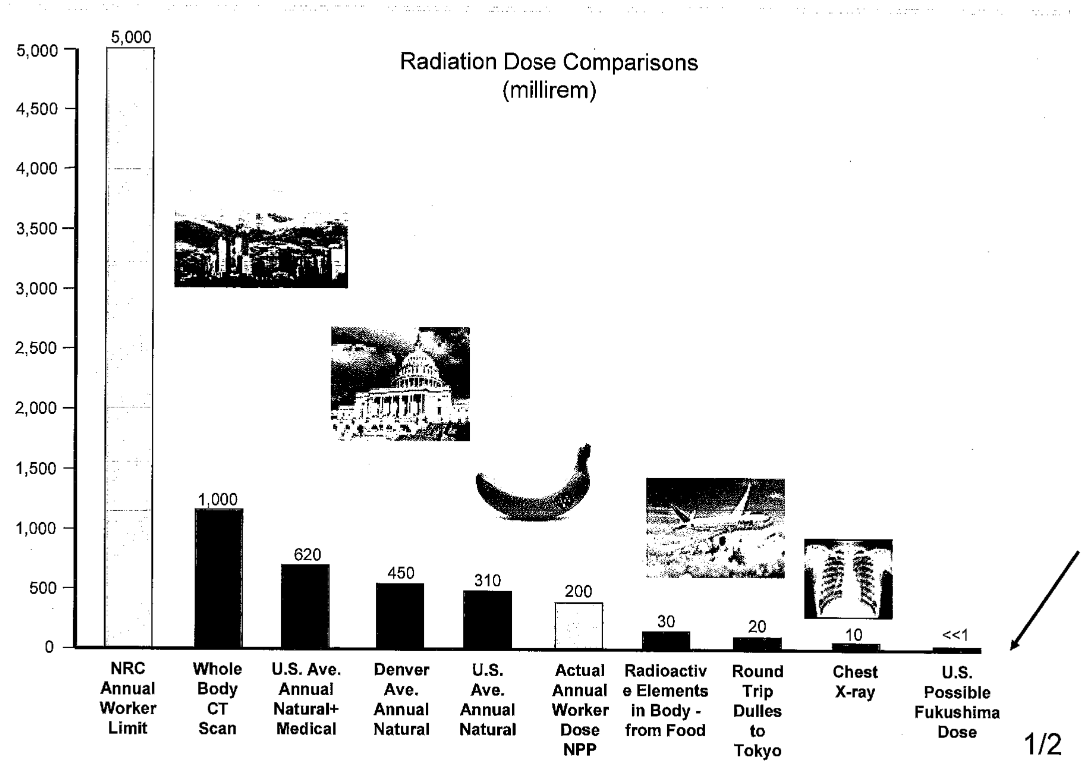
Exposure to the public was minimal in large part because the prevailing winds blew much of the radioactive releases toward the ocean, and because of the actions taken by the Japanese authorities. They evacuated people living within 20 kilometers of the Fukushima plant, and recommended that those within 30 kilometers stay indoors to minimize exposure. They monitored the food and water supplies, and banned the shipment of foodstuffs and milk when the radiation levels exceeded allowable standards. These protective measurements, including the distribution of stable iodine pills or syrup for children minimized public doses, and subsequently, there was unlikely to be any or minimal health consequences. This is borne out in a survey of over 1,000 children who had their thyroids measured for possible uptakes of radioactive iodine. Not one child had a measurement above normal. Nonetheless, some of the prevailing winds did blow toward populated areas and these areas will be a concern for remediation before allowing public access to return.

Fukushima is 5,000 miles away from the United States, and radiation is substantially diluted after traveling such a long distance. The detection of trace amounts of radiation speaks more about the sensitivity of our detectors than to the possible consequences to public health. They pose no threat to human health. They represent at most only a tiny fraction of what we receive each day from daily sources of radiation.

The minute levels of radioactive iodine detected in milk in Washington State were 5,000 times below the levels set by the FDA to trigger concern. An infant would have to drink hundreds of gallons of milk to receive a radiation dose equivalent to a day's worth of natural background radiation exposure. These trace levels are not a public health concern, and potassium iodide tablets should not be taken as a preventive measure to block the thyroid's uptake of such tiny levels. There are potential adverse effects from taking these tablets, and these risks have to be a balance against a non-existent benefit.

We live in a radioactive world. If I could have that first slide? [Slide]

We live in a radioactive world



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Needed Research When Exposure Spread Over Time

One Million U.S. Radiation Workers and Veterans

- U.S. DOE and Manhattan Project workers
- U.S. atomic veterans
- U.S. nuclear utility workers
- U.S. medical and other occupational groups
- U.S. nuclear navy workers

In comparisons might help place the radiation levels from Fukushima in context. Practically all the food we eat contains small amounts of naturally occurring radioactive elements. We breathe radioactive radon. Bricks and granite contain radioactive materials that emit gamma radiation. The Capitol building has some of the highest radiation levels in the United States. Water contains small amounts of radioactive radium, thorium, and uranium.

These examples are not to minimize the health consequences of high and moderate exposures, but just to place in perspective the tiny amounts from Fukushima which pose no public health problems to the United States.

The Fukushima accident, however, highlights the need for continued health research to fill important gaps in knowledge. We know much about the effects of high levels of radiation when received briefly, as was the case for the atomic bomb survivors whose exposure was in less than a second. However, the level of risk following exposures experienced gradually, over long periods of time, are uncertain and remains the major unanswered question in radiation epidemiology and risk assessment.

One untapped opportunity that should not be wasted is to study our own U.S. radiation workers and veterans. The Low Dose Radiation Program within the Department of Energy had the foresight to provide seed money to evaluate the feasibility of studying one million Americans, and this comprehensive work should continue. The studied populations include Department of Energy and Manhattan Project workers, atomic veterans who participated in nuclear weapons tests, nuclear utility workers, and others.

Thank you very much for this opportunity to appear before you.

[The prepared statement of Dr. Boice follows:]

PREPARED STATEMENT OF DR. JOHN BOICE, SCIENTIFIC DIRECTOR, INTERNATIONAL EPIDEMIOLOGY INSTITUTE

Good morning, Mr. Chairmen, ranking Members, and Members of the Subcommittee. I am pleased to discuss the possible health implications of radiation from the Fukushima Daiichi nuclear power plant accident in Japan. Just a few days before the natural disasters struck on March 11, 2011, I was in Hiroshima, Japan as a member of the Radiation Effects Research Foundation's Science Council, reviewing the study of atomic bomb survivors. I would like to begin by expressing my heartfelt sympathy for the families of the tens of thousands who lost their lives as a result of the tsunami and earthquake and for the hundreds of thousands who have been displaced from their homes and livelihoods. The health consequences associated with the radiation exposures emanating from the Fukushima Daiichi plant pale in comparison.

As background, I am a radiation epidemiologist and Professor in the Department of Medicine at Vanderbilt University and Scientific Director of the International Epidemiology Institute. I have spent my career studying human populations exposed to radiation, including Chernobyl clean-up workers, patients receiving diagnostic and therapeutic radiation, underground miners exposed to radon, nuclear energy workers, atomic veterans, persons living in areas of high background radiation and U.S. populations living near nuclear power plants and other facilities. I am also a commissioner of the International Commission on Radiological Protection, an emeritus member of the National Council on Radiation Protection and Measurements, a U.S. delegate to the United Nations Scientific Committee on the Effects of Atomic Radiation, and a member of the Congressionally-mandated Veterans Advisory Board on Dose Reconstruction.

My remarks will cover five areas:

- Fukushima is not Chernobyl.

- The health consequences for Japanese workers and public appear to be minor.
- The health consequences for United States citizens are negligible to non-existent.
- We live in a radioactive world.
- There is a pressing need to learn more about the health consequences of radiation in humans when exposures are spread over time at low levels and not received briefly at high doses such as in atomic bomb survivors.

Fukushima is not Chernobyl [Slide 1]

The Chernobyl accident on April 26, 1986, resulted in massive radiation exposures, both to the emergency workers putting out the ensuing fire and to the environment. There was no containment vessel and after the explosion a fire burned for ten days and spewed radioactive particles continuously into the environment. The emergency workers, the first responders and fire fighters, received so much radiation that 28 of them died of acute radiation sickness within a few months of exposure. Those who survived developed cataracts at a high rate and several subsequently died of myelodysplastic disorders. Radioactive iodines were deposited on large areas throughout the Ukraine, Belarus and Russian Federation and were ingested by cows who gave milk that was drunk by children, and an epidemic of thyroid cancer ensued beginning about five years after the accident. Over 520,000 recovery workers were sent to clean up the environment and build the so-called sarcophagus to contain the damaged nuclear reactor. To date there is little conclusive evidence for adverse health effects associated with radiation received during these clean-up operations. There have, however, been indications of severe psychological stress and increased rates of suicide.

In contrast, while the radiation releases from Fukushima [Slide 2] are estimated to be up to 10% of that from Chernobyl, there appears to be substantially less worker and public exposure. The Japanese authorities relaxed the allowable annual limit of worker exposure from 2 to 25 rem for this emergency situation, but only about 21 workers received more than 10 rem and only two workers received between 20 and 25 rem. These levels are far below the hundreds of rem needed to cause acute radiation sickness. Those workers who experienced levels over 10 rem to their entire body, however, have an increased lifetime risk of developing cancer of about 1-2% over the expected normal lifetime rate of about 42%. There were reports of high radiation fields in the vicinity of the damaged reactors and spent fuel storage ponds and with the contaminated water, but apparently the Japanese authorities rotated workers in such a way that cumulative exposures to individuals were minimized. Three workers received beta particle exposures to their legs from an estimated 200-300 rem to the skin, but the health consequences of these localized exposures were minimal and resulted in only a reddening of the skin.

Exposure to the public was minimal in large part because of the prevailing winds and the quick action taken by the Japanese authorities. The prevailing winds were generally to the east and over the ocean and thus did not result in meaningful radiation exposures to the Japanese public. In contrast to the circumstances around Chernobyl where the authorities failed to alert or evacuate the surrounding populations until several days had passed, the Japanese government quickly evacuated persons living within 20 km of the Fukushima Daiichi plant and recommended that those living within 30 km stay indoors to minimize any possible exposure to radioactive releases. In addition, they immediately monitored the food and water supplies and banned the shipment of foodstuffs and milk where the radiation levels exceeded allowable standards.

These protective action measures, including the distribution of stable iodine pills (or syrup for children), minimized public doses and suggest that there will be minimal health consequences associated with any radiation exposures to the Japanese public. This is borne out in one survey of over 1,000 children who had their thyroids measured for possible uptakes of radioactive iodine. Not one child had a measurement above detectable limits. This is in contrast to children living near Chernobyl for whom large numbers had extremely high levels of radioactive iodine detected in their thyroids from drinking contaminated milk shortly after the accident.

Nonetheless, some of the prevailing winds did blow toward populated areas shortly after the accident and during the hydrogen explosions, and to the north-west in particular. Rain, snow and hail deposited radioactive particles in certain regions, including some beyond 20 km, and these areas will be a concern for remediation before allowing public access or return. The Japanese authorities are considering regular medical examinations for workers and inhabitants who received more than 10 rem. To reduce anxiety, they are considering medical check-ups for those who may have received between 2 to 10 rem. They are also grappling with important issues as to when and how to allow evacuated inhabitants to return to their homes. Childhood exposures are of particular concern and topsoil is already being removed from some school playgrounds.

Thus, while Fukushima is clearly a major reactor accident, the potential health consequences associated with radiation exposures in terms of loss of life and future cancer risk are small, particularly in contrast with those resulting from the Chernobyl accident some 25 years ago.

For completeness, the 1979 reactor accident at Three Mile Island did not release appreciable amounts of radioactive substances into the environment, and public and even worker exposures were minimal. The average dose to people in the area was only about 1 millirem, or about what would be received in three days from sources of natural background radiation to the surrounding population.

The health consequences for United States citizens are negligible to nonexistent. [Slide 2]

Fukushima is 5,000 miles away from the United States and the radiation that has been detected was substantially diluted after traveling such a long distance. The detection of trace amounts of radiation speaks more about the potential health consequences from the radiation itself. In addition to EPA's RadNet system that monitors water, milk and the atmosphere, the Department of Energy has radiation monitoring equipment that can detect minute quantities of radioactive particles from the other side of the world as part of the Comprehensive Nuclear Test Ban Treaty. The tiny amounts of detected radioactive materials from Fukushima pose no threat to human health. They represent, at most, only a tiny fraction of what we receive each day from natural sources, such as the sun, the food we eat, the air we breathe and the houses we live in.

It is impressive that radiation monitors can detect levels of radioactive iodine-131 as low as 0.03 Bq/L (0.8 pCi/L) in milk in Washington State; this is the decay of one radioactive atom per second in about 33 gallons of milk. Such a level is 5,000 of times below the Derived Intervention Level set by the Food and Drug Administration to trigger concern over radionuclides in food. An infant would have to drink hundreds of gallons of milk to receive a radiation dose equivalent to a day's worth of natural background radiation exposure. Such tiny levels of radiation are inconsequential compared with the levels we experience in daily life.

Interestingly, the radiation monitoring stations in Washington State had to detect radionuclides other than iodine-131 in order to distinguish radiation from Fukushima from that at any local hospital in the area. Most nuclear medicine departments use radioactive iodine for imaging the thyroid and to treat thyroid diseases, and patients are discharged shortly after intake and remain radioactive for several months, releasing small but detectable levels of radioactive iodine into the environment.

The trivial levels of radiation from Japan, while detectable, should not be of a concern and Americans should not take stable iodine (potassium iodide pills, KI) as a preventive measure to block the thyroid's uptake of radioactive iodine. There are potential adverse health effects from taking KI pills and these risks have to be balanced against a nonexistent benefit.

We live in a radioactive world. [Slide 3]

To place the radiation levels from Fukushima in brief perspective, it is important to recognize that we live in a radioactive world. A banana, for example, has 10 Bq of activity, that is, 10 radioactive potassium atoms decay every second. All the foodstuffs we eat that contain potassium also contain a small amount of radioactive potassium, a primordial element with a billion year half-life. There are no concerns and no health consequences from such exposures.

We breathe radioactive radon which contributes over the year to about 210 millirem of natural background radiation. Bricks and granite contain radioactive materials that result in radiation exposures to the public (20 millirem). The Capitol Building was constructed with granite and is frequently cited as having some of the highest radiation levels in all of the United States, about 85 millirem per year. Water contains small amounts of radioactive radium, thorium and uranium, all within allowable limits.

Not only do we live in a radioactive world, our bodies are radioactive (30 millirem per year). Each second over 7,000 radioactive atoms in our bodies decay and can irradiate those sitting next to us. The atoms are largely radioactive potassium in our muscles and carbon-14 in our tissues. The amount of radiation we receive each year from medical sources (300 millirem), such as CT and medical imaging, equals the amount received from natural sources (300 millirem). International travel increases our exposure to cosmic rays and space radiation. A roundtrip from Dulles to Tokyo would result in 20 millirem. Living in Denver for a year results in 450 millirem of radiation dose, or 35% more than the U.S. average of 310 millirem from natural sources. About 2.5 million Americans (0.8% of the population) receive more than 2,000 millirem per year from natural sources.

These examples are not to minimize the health consequences of high-level exposures which are clearly demonstrable in human populations and include acute radiation sickness at very high doses in excess of 200 rem and an increase in cancer at moderate doses above about 10 rem (10,000 millirem). The examples do indicate, however, that we live in a world of exposures to the U.S. population from Fukushima are tiny and thousands of times below U.S. standards or guidelines where remedial action would be triggered.

What research is needed? [Slide 4]

Although we know much about the health effects of high levels of radiation when received briefly, as was the case for atomic bomb survivors, the risk following exposures experienced gradually over time is uncertain and remains the major unanswered question in radiation epidemiology.

One untapped opportunity is to study our own U.S. radiation workers and veterans. The Low Dose Radiation Program within the Department of Energy had the foresight to initiate pilot investigations of over one million such workers and this comprehensive work should continue. Cooperating agencies include the National Cancer Institute, the Department of Defense, the Department of Veterans Affairs, the Nuclear Regulatory Commission and others. The study populations include early DOE and Manhattan Project workers, atomic veterans who participated in nuclear weapons testing in the 1940s and 1950s, nuclear utility workers, medical workers and others involved in the development of radiation technologies, as well as nuclear navy personnel.

Such a large study in the United States is critically important to understand scientifically the health consequences of low-dose radiation experienced over time and is directly relevant to the setting of protection standards for workers and the public; the assessment of possible risks from enhanced medical technologies such as CT and nuclear medicine imaging; the expansion of nuclear power; the handling of nuclear waste; the compensation of workers with prior exposures to radiation; and even the possible consequences of the radiation released from reactor accidents such as at Fukushima. To date, no direct study of these issues has been exposures in 1945 have to be relied upon.

Summary [Slide 5]

Fortunately, the health consequences from the radiation releases from the Fukushima Daiichi power plant appear to be minimal and are of little importance with regard to the U.S. public. The Japanese authorities acted quickly to evacuate over'200,000 inhabitants living near the damaged reactors; they monitored food and water and took rapid action to ban foodstuffs with increased radiation levels; they distributed stable iodine pills and syrup; and they made measurements on over 175,000 persons. The lasting effects upon the Japanese population will most likely be psychological with increased occurrence of stress-related mental disorders and depression associated not necessarily with the concern about reactor radiation, but with the horrific loss of life and disruption caused by the tsunami and earthquake. There is a need for better public understanding and better communications on the health effects of radiation exposures. Finally, there is now the opportunity in the United States to learn directly about low-dose, long-term radiation health effects by studying our workers and veterans.

Thank you for this opportunity to testify. I welcome any questions that you may have.

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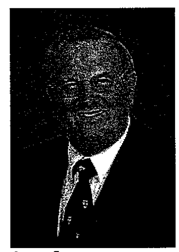
Vanderbilt-Ingram Cancer Center

The VICC.ORG Directory of Doctors, Healthcare Providers & Researchers

John D. Boice, Jr., Sc.D.

Professor, School of Medicine
VICC Member
Cancer Epidemiologist



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See also:

- Cancer Epidemiology, Prevention and Control Research
- Medicine: Division of Epidemiology
- Medicine: General Internal Medicine and Public Health
- PubMed Listing of Dr. Boice's Publications

Profile

John Boice is professor of Medicine at Vanderbilt University School of Medicine and scientific director of the International Epidemiology Institute (IEI), a biomedical research firm with offices in Rockville, Md., and Jacksonville, Fla., involved in the design, conduct and analysis of epidemiologic studies into the causes of cancer and other diseases. He is an international authority on radiation effects and currently serves on the Main Commission of the International Commission on Radiological Protection, the US delegation to the United Nations Scientific Committee on the Effects of Atomic Radiation, and the US Congressionally mandated Veterans' Advisory Board on Dose Reconstruction. He is a Distinguished Emeritus Member of the National Council on Radiation Protection and Measurements.

During 27 years of service in the US Public Health Service, Boice developed and became the first chief of the Radiation Epidemiology Branch at the National Cancer Institute (NCI). Boice has established programs of research in all major areas of radiation epidemiology, with major projects dealing with populations exposed to medical, occupational, military, and environmental radiation. These research efforts have aimed at clarifying cancer and other health risks associated with exposure to ionizing radiation, especially at low dose levels. Boice's seminal discoveries and over 420 publications have been used to formulate public health measures to reduce population exposure to radiation and prevent radiation associated diseases.

In March 2010, he delivered the Elis Berven Lecture at the Swedish Society of Oncology in Kalmar, Sweden on "Epidemiologic studies of second cancers following radiotherapy."

In 2009, Boice delivered the Lauriston Taylor Lecture at the National Council on Radiation and Protection and Measurements' annual meeting and the Fessinger-Springer Lecture at the University of Texas at El Paso. In 2008, Boice received the Harvard School of Public Health Alumni Award of Merit. He has also received the E.O. Lawrence Award from the Department of Energy an honor bestowed on Richard Feynman and Murry Gell Mann among others and the Gorgas Medal from the Association of Military Surgeons of the United States. In 1999 he received the outstanding alumnus award from the University of Texas at El Paso (formerly Texas Western College).

Boice has a bachelor's degree in Physics from Texas Western College (now the University of Texas at El Paso) and a master's degree in Nuclear Engineering from Rensselaer Polytechnic Institute. He received a master's degree in Medical Physics at Harvard and also a doctoral degree in Epidemiology at Harvard.

He currently directs the Genetic Consequences of Cancer Treatment study, supported by the NCI, the largest epidemiologic study yet undertaken to assess the possible risks of adverse pregnancy outcomes (malformation, neonatal death, stillbirth, cancer) related to the curative treatments received by cancer survivors who are able to become pregnant. In cooperation with the Department of Defense and the Department of Veterans Affairs, Boice recently initiated an NCI-funded study of atomic veterans who participated in any of the 230 atmospheric nuclear weapons tests between 1946 and 1958 at the Nevada Test Site or the Pacific Proving to examine the lifetime risk of cancer following relatively low-dose exposures received gradually over time.

Enclosure 3

Committee on Science, Space, and Technology

U.S. House of Representatives

Witness Disclosure Requirement - "Truth in Testimony"
Required by House Rule XI, Clause 2(g)(5)

1. **Your Name:**
   John D. Boice, Jr.

	Yes	No
2. **Are you testifying on behalf of the Federal, or a State or local government entity?**		X
3. **Are you testifying on behalf of an entity that is not a government entity?**		X

4. **Other than yourself, please list which entity or entities you are representing:**

5. **Please list any Federal grants or contracts (including subgrants or subcontracts) that <u>you or the entity you represent have received</u> on or after October 1, 2008:**

   See attached.

6. **If your answer to the question in item 3 in this form is "yes," please describe your position or representational capacity with the entity(ies) you are representing:**

	Yes	No
7. **If your answer to the question in item 3 is "yes," do any of the entities disclosed in item 4 have parent organizations, subsidiaries, or partnerships that you are not representing in your testimony?**		

8. **If the answer to the question in item 3 is "yes," please list any Federal grants or contracts (including subgrants or subcontracts) that were received by the entities listed under the question in item 4 on or after October 1, 2008, that exceed 10 percent of the revenue of the entities in the year received, including the source and amount of each grant or contract to be listed:**

I certify that the above information is true and correct.

Signature: John D. Boice, Jr. **Date:** 5 May 2011

Federal grants or contracts on which John D. Boice is an investigator on or after October 1, 2008

Award Number	Title	Sponsor
R01 CA104666	Genetic Consequences of Therapies for Cancer	NIH-National Cancer Institute
U24 CA055727	Childhood Cancer Survivors Study	NIH-National Cancer Institute
R01 CA97397	Interaction of Radiation, BRCA1/2, and Breast Cancer	NIH-National Cancer Institute
R01 CA114236	Breast Cancer, Radiation and ATM-CHEK2 Pathway	NIH-National Cancer Institute
R01 CA092447	Southern Community Cohort Study	NIH-National Cancer Institute
U01 CA137026	Cancer Mortality among Military Participants at U.S. Nuclear Weapons Tests	NIH-National Cancer Institute
DESC0004307	One Million U.S. Workers and Veterans Exposed to Radiation	U.S. Department of Energy

Dr. HARRIS. Thank you very much, Dr. Boice, and now I recognize our final witness, Dr. Dave Lochbaum, the Director of Nuclear Safety Project for the Union of Concerned Scientists.

Mr. Lochbaum?

STATEMENT OF MR. DAVID LOCHBAUM, DIRECTOR, NUCLEAR SAFETY PROJECT, UNION OF CONCERNED SCIENTISTS

Mr. LOCHBAUM. Good morning, Mr. Chairman, Ranking Member Edwards, and other Members of the Subcommittees. On behalf of the Union of Concerned Scientists, I appreciate this opportunity to share our perspectives. My written testimony describes lessons already evident from the Fukushima disaster that are applicable to ensuring safer nuclear power plants in the United States. This morning, I would like to focus on three of those lessons.

The first lesson involves severe accident management guidance. In NRC terminology, a severe accident involves some fuel damage. The NRC and the nuclear industry representatives have claimed that the severe accident management guidelines developed after the Three Mile Island meltdown would protect us from the problems faced at Fukushima. They have not been telling the whole story. As broadcaster Paul Harvey used to say, here is the rest of the story.

The entry for severe accident management guidelines in NRC manual chapter 0308 states "The staff concluded that regular inspection was not appropriate because the guidelines are voluntary and have no regulatory basis." The NRC never checks the guidelines to determine if they might actually work under severe accident conditions. From March 2009 until March 2010, I worked for the NRC as an instructor at their technical training center. My duties included teaching the severe accident management guidelines to NRC employees. I and the other instructors emphasized that NRC inspectors were not authorized to evaluate the adequacy of the guidelines. Plant owners are required to have the guidelines, while NRC inspectors are required not to assess them.

If the NRC continues to rely on these guidelines to protect public health, it must evaluate their effectiveness. It would be too late and too costly to find out after a nuclear plant disaster that the guidelines were missing a few key steps or contained a handful of missteps.

The second lesson involves upgraded guidance for spent fuel pool events. As I mentioned, the NRC and the nuclear industry upgraded the procedures used by the operators during reactor core accidents. The upgraded procedures provide the operators with a full array of options available to deal with the reactor core accident, not just those options relying on emergency equipment. In addition, the upgraded procedures would help the operators handle problems like unavailable or misleading instrumentation readings. No such procedures and associated training are available to help the operators deal with spent fuel pool events. The NRC must require robust procedures for spent fuel pool problems comparable to those available for reactor core problems so that operators can prevent fuel damage from occurring, or mitigate its consequences when those efforts fail.

The last lesson involves additional regulatory requirements for defueled reactors. When the earthquake and tsunami happened in Japan, the reactor core in Fukushima Unit 4 was fully offloaded into the spent fuel pool. This configuration is termed a defueled condition. There is a gaping hole in the regulatory safety net when reactors are defueled. When the NRC issues operating licenses for reactors, appendix A to that license contains the technical specifications. These specifications establish "the lowest functional capability of performance levels of equipment required for safe operation of the facility," along with the scope and frequency of testing required to demonstrate that capability.

The operational condition of the reactor determines which requirements are applicable when. When the entire reactor core has been offloaded into the spent fuel pool, very few requirements still apply. For example, the containment structure surrounding the spent fuel pool is no longer required to be available to be intact. This containment significantly reduces the amount of radioactivity reaching the environment from damaged fuel in the spent fuel pool, but only when it is intact. Likewise, the specifications do not require normal power, backup power, or even battery power to be available.

When the fuel is in the reactor core, the specifications mandate safety measures to protect Americans from that hazard, but when that hazard is entirely relocated to the spent fuel pool, nearly all those safety measures can be removed. The NRC must fix this deficiency as soon as possible to provide adequate protection of public health when reactor cores are defueled. In the interim, the NRC should seriously consider banning full core reactor offloads into the spent fuel pool.

In conclusion, the measures we have recommended will lessen the chance of a disaster at a U.S. nuclear power plant, but if it happens anyway, the Federal Government would be able to look Americans in the eye and say we took every reasonable measure to protect you.

Thank you.

[The prepared statement of Mr. Lochbaum follows:]

PREPARED STATEMENT OF MR. DAVID LOCHBAUM, DIRECTOR, NUCLEAR SAFETY PROJECT, UNION OF CONCERNED SCIENTISTS

For nearly four decades, the Union of Concerned Scientists has been a nuclear power safety and security advocate. Neither anti- nor pro-nuclear power, UCS strives to ensure that the technology's inherent risks are minimized to the extent that is practically achievable.

The tragic events at the Fukushima Dai-Ichi nuclear plant in Japan have already revealed areas of elevated risk that should be rectified. Over the ensuing months and years, additional lessons will undoubtedly surface as workers conduct CSI Nuclear to assess what failed due to various causes, including the earthquake, the tsunami, the extended power outage, the hydrogen explosions, the torrents of water dropped from above and sprayed from below, and the submersion of equipment in water. Today, UCS would like to share six of the lessons already evident from Fukushima Dai-Ichi that are applicable to ensuring safer nuclear power plants in the United States:

- Better protection against extended power outages
- Adequate severe accident management guidance
- Safer storage of spent fuel
- Upgraded guidance for spent fuel pool events
- Additional regulatory requirements for defueled reactors

BETTER PROTECTION AGAINST EXTENDED POWER OUTAGES

Some may argue that what happened at Fukushima Dai-Ichi cannot happen here—that our nuclear power plants are not vulnerable to extended power outages caused by the one-two punch of an earthquake and tsunami. In June 1998, a tornado disabled the normal power supply for the Davis-Besse nuclear plant in Ohio, just as the earthquake had done for Fukushima Dai-Ichi. Outside air temperatures exceeding 90°F caused the backup power supply to overheat and fail, just as the tsunami had done at Fukushima Dai-Ichi. The difference was that workers restored the normal power supply for Davis-Besse an hour before the backup power supply failed while more extensive damage prevented workers at Fukushima Dai-Ichi from restoring its normal power supply for nearly a week, days too late to prevent fuel damage.

Enclosure 1 provides Tables B-1 and B-2 from a 2003 report issued by the NRC on power outages at U.S. nuclear power plants. When both the normal and backup power supplies are lost, a condition called station blackout occurs. As at Fukushima Dai-Ichi, the only source of power during a station blackout is a bank of batteries. The fourth column of data in Tables B-1 and B-2 provides the percentage of overall risk of reactor core damage (called core damage frequency or CDF) due to station blackouts as calculated by the plant owners themselves. For example, station blackouts constitute 80.6 percent of the overall core damage risk at the LaSalle nuclear plant in Illinois. In other words, the risk from station blackouts is roughly four times the risk from all other causes combined. And LaSalle is located far away from the earthquake faults of California and the tsunami risks of both coasts, so clearly an earthquake and tsunami is not the only path to a station blackout disaster.

The three reactors at Fukushima Dai-Ichi operating at the time of the earthquake were each equipped with banks of batteries having 8-hour capacities. As reflected by the data in the fifth column of Tables B-1 and B-2, the majority of U.S. reactors have equal or shorter station blackout coping durations. This means that workers at a U.S. reactor experiencing a station blackout would essentially be playing a very high stakes version of "Beat the Clock." If they restore normal or backup power within a few hours, they win. If not, many may lose.

Requiring nuclear plants to have 16 hours of battery capacity would give workers a greater chance of bearing the clock. But what if, as at Fukushima Dai-Ichi, it takes longer than 16 hours to restore the normal and backup power supplies? The world has been watching what happens, and it isn't pretty or worth emulating.

UCS believes a better way to ensure victory in station blackout "Beat the Clock" is to evaluate how long it will likely take for replacement batteries and/or portable generators to be delivered to each nuclear power plant site. For some plant sites, the current situation is fine because nearby reinforcements exist and it will be possible to supply replacement batteries or portable generators within the existing 4-hour or 8-hour station blackout coping duration. However, for other plants reinforcements are not likely to arrive in time, and reactor owners should increase the battery capacity and/or pre-stage battery replacements and portable generators closer to the site.

ADEQUATE SEVERE ACCIDENT MANAGEMENT GUIDANCE

In NRC terminology, a severe accident is one in which at least some of the fuel melts. In testimony at Congressional hearings, NRC and nuclear industry representatives have claimed that the severe accident management guidelines (SAMGs) developed in the wake of reactor meltdown at Three Mile Island would provide reliable protection against the problems faced at Fukushima Dai-Ichi. They have not been telling the whole story. As newscaster Paul Harvey used to say, here's the rest of the story.

Enclosure 2 provides part of Table 2 from NRC Manual Chapter 0308 on its reactor oversight process (ROP). The fourth column for the severe accident management guidelines entry states:

> *The [NRC] staff concluded that regular inspection of SAMG was not appropriate because the guidelines are voluntary and have no regulatory basis.*

The NRC never checks—repeat, never checks—the guidelines to see if they would be effective under severe accident conditions.

From March 2009 until March 2010, I worked for the NRC as a Boiling Water Reactor technology instructor at their Technical Training Center. My duties included teaching the severe accident management guidelines to NRC employees for their initial qualifications and re-qualifications. I and the other instructors emphasized that NRC inspectors were not authorized to evaluate the adequacy of the guidelines. Plant owners are required to have the guidelines while NRC inspectors are required not to assess their effectiveness. It's like maritime inspectors ensuring that passenger liners have lifeboats, but not checking to see that there's sufficient capacity for all passengers and crew members.

If NRC continues to rely on these guidelines to protect public health, it must evaluate their effectiveness.. It would be too late and too costly to find out after a U.S. nuclear plant disaster that the plant's severe accident management guideline was missing a few key steps or contained a handful of missteps.

SAFER STORAGE OF SPENT FUEL

Much has been reported about the problems with the fuel in the spent fuel pools at Fukushima Dai-Ichi Units 3 and 4. Helicopters dropped tons of water from above while water cannons on fire trucks sprayed water from below. And yet it appears that fuel in at least two spent fuel pools has been damaged.

Virtually nothing has been reported about the fuel stored in dry casks at Fukushima Dai-Ichi. It experienced the earthquake. It experienced the tsunami. It experienced the prolonged power outage. It did not overheat. It was not damaged. It did not produce hydrogen that later exploded. It did not cause the evacuation of a single member of the public. It did not cause a single worker to receive radiation over-exposure.

The spent fuel pools at nuclear plants in the United States are significantly fuller than those in Japan. As a result, the chances of a spent fuel accident are higher and the consequences would be greater.

For the first five years after being taken out of the reactor core, spent fuel generates too much heat to be placed into dry casks. After five years, the heat generation rates have dropped low enough to permit dry cask storage.

It takes no pumps, no power, no switches, and no forced circulation of water to protect spent fuel in dry casks from damage. Instead, air enters an inlet in the bottom of the dry cask, gets warmed by the heat from the spent fuel, and flows out an outlet in the top of the dry cask via the chimney effect. It's the "passive" safety system that worked at Fukushima Dai-Ichi and would work here, if we bothered to use it.

Instead, spent fuel pools in America are filled nearly to capacity. Then and only then is spent fuel transferred into dry casks. But the amount of spent fuel transferred is just enough to free up the space needed for the next fuel discharged from the reactor core. This practice maintains the spent fuel pool risk at a level about as high as can be achieved, and exposes millions of Americans to elevated and undue risk.

The safer way to store spent fuel is to transfer it into dry casks as soon as possible following the five year cooling off period in a spent fuel pool. That's the "passive" safety system Americans need most.

UPGRADED GUIDANCE FOR SPENT FUEL POOL EVENTS

Following the March 1979 accident at Three Mile Island Unit 2 in Pennsylvania, the NRC and the nuclear industry significantly upgraded the procedures used by operators during reactor core accidents. The upgraded procedures provide the operators with the full array of options available to deal with a reactor core accident, not just those relying on emergency equipment. In addition, the upgraded procedures would help the operators handle problems like unavailable or misleading instrument readings.

No such procedures, and associated training, are available to help operators deal with spent fuel pool accidents. After the water level in the Unit 4 spent fuel pool at Fukushima Dai-Ichi dropped below the top of the fuel assemblies, the fuel rods heated up, producing large amounts of hydrogen gas. That hydrogen exploded, destroying the reactor buildings walls and roof and creating a pathway for radioactivity to freely escape to the environment. To lessen the likelihood of similar explosions, workers cut openings in the roofs and walls of the reactor buildings on Units 2, 5, and 6. Their efforts were ad hoc and reactive.

The NRC should require robust procedures for spent fuel pool problems, comparable to those for reactor core problems, to help operators either prevent fuel damage or mitigate its consequences should such damage occur.

ADDITIONAL REGULATORY REQUIREMENTS FOR DEFUELED REACTORS

When the earthquake and tsunami happened, the reactor core on Fukushima Dai-Ichi Unit 4 was empty of fuel, with the fuel having been transferred to its spent fuel pool. That configuration is termed a defueled operating condition. There's a gaping hole in the regulatory safety net when reactors are defueled.

Enclosure 3 contains pages excerpted from the NRC's Standard Technical Specifications for boiling water reactors. When the NRC issues, or renews, licenses to operate nuclear power reactors, Appendix A to these licenses are the technical specifications. These specifications establish "the lowest functional capability or performance levels of equipment required for safe operation of the facility"[1] along with the scope and frequency of testing required to verify that capability. The operational condition of the reactor (also called its MODE and defined by the Reactor Mode Switch Position and the temperature of the reactor cooling water) determines which requirements are applicable when. However, technical specification requirements only apply when one or more fuel assemblies are located in the reactor core. When the entire reactor core inventory has been offloaded to the spent fuel pool, almost no technical specification requirements still apply.

For example, technical specification 3.6.4.1 no longer requires secondary containment to be intact. Secondary containment, which is the reactor building, houses the spent fuel pool and acts as a barrier to prevent any radioactivity released from fuel in the spent fuel pool from reaching the environment—but only when it is intact. Likewise, technical specification 3.8.2 does not require normal or backup power supplies to be available. And technical specification 3.8.5 does not even require battery power to be available.

When one or more fuel assemblies is in the reactor core, the technical specifications mandate safety measures to protect Americans from that hazard. But when that hazard is entirely relocated to the spent fuel pool, the technical specifications allow all of those safety measures to be taken away. Technical specification 3.7.8 would even allow all the water to be drained from the spent fuel pool with all the irradiated fuel in it.

The NRC must fix this technical specification deficiency to provide adequate protection of public health when reactor cores are defueled.

CONCLUSION

The measures we have recommended will lessen the chance of a disaster at a U.S. nuclear power plant. But if it happens anyway, the federal government would be able to look Americans in the eye and say, "we took every reasonable measure to protect you." Americans expect that protection. We urge the Congress to ensure the NRC provides Americans the protection they deserve.

Enclosures:

1. Pages from NRC NUREG-1776, "Regulatory Effectiveness of the Station Blackout Rule, August 2003.
2. Pages from NRC Inspection Manual Chapter 0308, "Reactor Oversight Process (ROP) Basis Document," October 16, 2006.
3. Pages from NRC NUREG-1433, Volume 1, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4," December 2005.
4. Executive Summary from UCS's report "Nuclear Power: Still Not Viable without Subsidies," February 2011.

[1] 10 CFR 50.36, Technical Specifications. Available online at http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0036.html

Enclosure 1

NUREG-1776

Regulatory Effectiveness of the Station Blackout Rule

Manuscript Completed: August 2000
Date Published: August 2003

Prepared by
W.S. Raughley

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-1 Operating pressurized-water reactors

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Arkansas Nuclear One Unit 1	4.67E-05	1.58E-05	33.8	4/.95/10/1	Added 1 DG and crosstie	3.58E-02	2	1			
Arkansas Nuclear One Unit 2	3.40E-05	1.23E-06	3.6	4/.95/10/1	Added crosstie	5.84E-02	1	1			
Beaver Valley Unit 1	2.14E-04	6.51E-05	30.4	4/.975/60/1	Added crosstie	6.64E-02	2				
Beaver Valley Unit 2	1.92E-04	4.86E-05	25.3	4/.975/60/1	Added crosstie	7.44E-02	1				
Braidwood Units 1&2	2.74E-05	6.20E-06	22.6	4/.95/10/1		4.53E-02	2				
Bryon Units 1&2	3.09E-05	4.30E-06	13.9	4/.95/10/1		4.43E-02					
Callaway	5.85E-05	1.80E-05	30.8	4/.975/-/1		4.60E-02					
Calvert Cliffs Units 1&2	2.40E-04	8.32E-06	3.4	4/.975/60/4	Added 1 EDG and one 1 DG	1.36E-01	3				
Catawba Units 1&2	5.80E-05	6.0E-07	10.3	4/.95/10/1		2.0E-03	1			330	
Comanche Peak Units 1&2	5.72E-05	1.5E-05	26.2	4/.95/-/1							

The battery capacity for each reactor is the first number provided in the 5th column of this table. For Arkansas Nuclear One Unit 1, the battery capacity is 4 hours. The fourth column shows fraction of overall risk from reactor core damage that station blackout represents. For example, station blackout represents 33.8% of the risk of reactor core damage at Arkansas Nuclear One Unit 1. NOTE: These risk values only consider the hazard of reactor core damage. The hazard of spent fuel pool accidents is neglected here.

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-1 Operating pressurized-water reactors (Cont.)

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Crystal River Unit 3	1.53E-05	3.28E-06	21.5	4/.975/-/4	dc load shed. Added nonclass 1E battery	4.35E-01	3				
Davis-Besse	6.6E-05	3.50E-05	53	4/.95/10/2	Added 1 DG	3.50E-02	2	1		1680	
DC Cook Units 1&2	6.2E-05	1.13E-05	18.1	4/.975/-/2	dc load shed	4.0E-02	1				
Diablo Canyon Units 1&2	8.8E-05	5.0E-06	5.68	4/.95/-/1	Added 1 DG	9.1E-02	1				261 917
Farley Units 1&2	1.3E-04	1.22E-05	9.4	4/.95/10/3	Service water to Aac, auto load shedding	4.70E-02	2				
Fort Calhoun	1.36E-05	NA	–	4/.95/-/2	DC load shed	2.17E-01	2				
Ginna	8.74E-05	1.0E-06	1.14	4/.975/-/1		3.50E-03	4				
Harris	7.0E-05	1.71E-05	24.4	4/.95/-/3	Lighting in several areas, ladder to isolation valve						
Indian Point Unit 2	3.13E-05	4.47E-06	14.3	8/.95/60/2	Added a DG for gas turbine auxiliaries	6.91E-02	2		3	390	

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-2 Operating boiling-water reactors

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Browns Ferry Units 2&3	4.80E-05	1.30E-05	27	4/.95/-/1	dc load shed	1.12E-01					
Brunswick Units 1&2	2.70E-05	1.80E-05	66.7	4/.975/60/5	Modified controls for existing crosstie	7.40E-02	3				1508 814
Clinton	2.66E-05	9.8E-06	36.8	4/.95/10/1	Added gas fans for selected room cooling	8.40E-02					
Cooper	7.97E-05	2.77E-05	34.8	4/.95/-/2		3.50E-02					
Dresden Units 2&3	1.8E-05	9.30E-07	5.03	4/.95/60/2	Added 2 DGs	1.12E-01	3	1		240	
Duane Arnold	7.84E-06	1.90E-06	24.2	4/.975/-/2	dc load shed, RCIC insulation & main control room lighting	1.17E-01			1		
Fermi	5.70E-06	1.3E-07	NMN	4/.95/60/1		1.88E-01					
FitzPatrick	1.92E-06	1.75E-06	NMN	4/.95/-/1	dc load shed, instrumentation and power supply mods	5.70E-02					
Grand Gulf	1.77E-05	7.46E-06	36.8	4/.95/-/2	dc load shed	6.80E-02					

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-2 Operating boiling-water reactors (Cont.)

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Hatch Unit 1	2.23E-05	3.30E-06	14.8	4/.95/60/2	Replaced battery chargers	2.20E-02					
Hatch Unit 2	2.36E-05	3.23E-06	13.7	4/.95/60/2	Replaced battery chargers	2.20E-02					
Hope Creek	4.63E-05	3.38E-05	73	4/.95/-/2	Valve modifications	3.4E-02					
LaSalle Units 1&2	4.74E-05	3.82E-05	80.6	4/.975/-/1	dc load shed, New batteries	9.60E-02	1				
Limerick Units 1&2	4.30E-06	1.0E-07	NMN	4/.95/60/3	Upgraded cross-ties	5.9E-02					
Monticello	2.60E-05	1.20E-05	46.2	4/.95/-/1	dc load shed	7.90E-02					
Nine Mile Point Unit 1	5.50E-06	3.50E-06	NMN	4/.975/-/1	dc load shed, added two safety related batteries	5.00E-02	4			595	
Nine Mile Point Unit 2	3.10E-05	5.50E-06	17.7	4/.975/-/1	dc load shed	1.20E-01					

Plant-Specific Station Blackout Information by Reactor Type and Operating Status

Table B-2 Operating boiling-water reactors (Cont.)

Plant	Plant CDF	SBO CDF	Percent SBO CDF of Plant CDF	Coping time in hours/EDG reliability/Aac access time in minutes/ extremely severe weather	Modification summary including dc load shed procedural modifications	SBO factors					
						PRA LOOP initiating event frequency	Number of LOOP events at power since commercial operation			LOOP event recovery times ≥ 240 minutes	
							Plant	Weather	Grid	Power	Shutdown
Oyster Creek	3.90E-06	2.30E-06	NMN	4/.975/60/1	Added crosstie & reactor pressure indication	3.26E-02	3				240
Peach Bottom Units 2 & 3	5.53E-06	4.81E-07	8.7	8/.975/60/3	Cross-tie to hydro unit	5.9E-02					
Perry	1.30E-05	2.25E-06	43.4	4/.95/10/1	Replaced selected cables	6.09E-02					
Pilgrim	5.80E-05	1.0E-10	NMN	8/.975/10/4	Alarms to line-up Aac	6.17E-01	1	5			1263 534
Quad Cities Units 1&2	1.2E-06	5.72E-07	NMN	4/.95/60/1	Added 2 DGs	4.81E-02	2				
River Bend	1.55E-05	1.35E-05	87.5	4/.95/-/2	Minor structural mod	3.50E-02	1				
Susquehanna Units 1&2	1.7E-05	4.2E-11	NMN	4/.975/-/2	dc load shed	-	1				
Vermont Yankee	4.30E-06	9.17E-07	21.3	8/.975/10/4	Modified incoming line and controls	1.0E-01	2			277	
Washington Nuclear Plant Unit 2	1.73E-05	1.07E-05	61.1	4/.95/-/1	dc load shed, replaced inverters	2.46E-02					

NRC INSPECTION MANUAL

MANUAL CHAPTER 0308

REACTOR OVERSIGHT PROCESS (ROP)
BASIS DOCUMENT

0308-01 PURPOSE

To describe the basis for the significant decisions reached by the U.S. Nuclear Regulatory Commission (NRC) staff during the development and implementation of the Reactor Oversight Process (ROP) for operating commercial nuclear power plants. This document shall serve as the source information for all applicable program documents such as manual chapters, performance indicator guidance, and assessment guidance.

0308-02 OBJECTIVES

02.01 To discuss significant developmental steps and decisions reached.

02.02 To describe in general how the processes work and why they are setup the way they are.

02.03 To summarize the history of, and reasons for, significant changes made to the oversight processes.

02.04 To explain those significant attributes that were considered but not used in the ROP, and the basis for the decision not to include them in the process.

0308-03 DEFINITIONS

None stated.

0308-04 RESPONSIBILITIES AND AUTHORITIES

None stated.

0308-05 GENERAL REQUIREMENTS

05.01 <u>Introduction</u>

On April 2, 2000, the NRC implemented a new ROP at all operating commercial nuclear power plants. The objectives of the staff in developing the various components of this new

Table 2 Other Inspection Program Elements Considered But Not Included (continued)

Inspectable Area or Program Attribute	Cornerstone	Scope	Basis for Not Including in Baseline Inspection Program
Severe Accident Management Guidelines (SAMG)	Emergency Preparedness	SAMGs include strategies for dealing with accidents that impact RCS integrity. SAMGs are sometimes implemented during EP drills and must be written is such a manner as to not impede implementation of the Plan.	The staff concluded that regular inspection of SAMG was not appropriate because the guidelines are voluntary and have no regulatory basis. The emergency response organization that would implement SAMGs is inspected through EP baseline inspection and performance is covered by two PIs.
Radiation Worker Performance	Occupational and Public Rad Safety	The objective of this area is to verify that workers understand the radiological hazards associated with nuclear plant operation, effectively identify and control these hazards, identify and resolve adverse trends or deficiencies, and maintain proper oversight of work.	Worker performance is a cross cutting area. Since the PIs are performance based, problems in this area should result in an operational occurrence that meets the definition of a PI.

Enclosure 3

Standard Technical Specifications General Electric Plants, BWR/4

Specifications

This electronic text represents the Commission's current Standard Technical Specifications. This document is updated periodically to incorporate NRC approved generic changes to the Standard Technical Specifications.

The last Standard Technical Specification NUREGs were published as Revision 3 of NUREG-1430, NUREG-1431, NUREG-1432, NUREG-1433, and NUREG-1434 in June 2004.

1.1 Definitions

LEAKAGE (continued)

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE,

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE, and

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

[LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.]
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
[MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)	The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.]
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core [for each class of fuel]. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

3.6 CONTAINMENT SYSTEMS

3.6.4.1 [Secondary] Containment

LCO 3.6.4.1 The [secondary] containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of [recently] irradiated fuel assemblies in the [secondary] containment,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. [Secondary] containment inoperable in MODE 1, 2, or 3.	A.1 Restore [secondary] containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. [Secondary] containment inoperable during movement of [recently] irradiated fuel assemblies in the [secondary] containment or during OPDRVs.	C.1 ---------------NOTE-------------- LCO 3.0.3 is not applicable. ----------------------------------- Suspend movement of [recently] irradiated fuel assemblies in the [secondary] containment. AND C.2 Initiate action to suspend OPDRVs.	 Immediately Immediately

3.7 PLANT SYSTEMS

3.7.8 Spent Fuel Storage Pool Water Level

LCO 3.7.8 The spent fuel storage pool water level shall be ≥ [23] ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1	---------------NOTE------------- LCO 3.0.3 is not applicable. ------------------------------------- Suspend movement of irradiated fuel assemblies in the spent fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify the spent fuel storage pool water level is ≥ [23] ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown" and

b. One diesel generator (DG) capable of supplying one division of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 4 and 5,
During movement of [recently] irradiated fuel assemblies in the [secondary] containment.

ACTIONS

--NOTE--

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	--------------------NOTE-------------------- Enter applicable Condition and Required Actions of LCO 3.8.10, with one required division de-energized as a result of Condition A. ---	
	A.1 Declare affected required feature(s), with no offsite power available, inoperable.	Immediately
	<u>OR</u>	

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5

[DC electrical power subsystems shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown."]

[One DC electrical power subsystem shall be OPERABLE.]

-----REVIEWER'S NOTE-----
This second option above applies for plants having a pre-ITS licensing basis (CTS) for electrical power requirements during shutdown conditions that required only one DC electrical power subsystem to be OPERABLE. Action A and the bracketed optional wording in Condition B are also eliminated for this case. The first option above is adopted for plants that have a CTS requiring the same level of DC electrical power subsystem support as is required for power operating conditions.

APPLICABILITY: MODES 4 and 5,
During movement of [recently] irradiated fuel assemblies in the [secondary] containment.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
[A. One [or two] battery charger[s on one division] inoperable. AND The redundant division battery and charger[s] OPERABLE.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage. AND	2 hours

Enclosure 4

Union of Concerned Scientists
Citizens and Scientists for Environmental Solutions

NUCLEAR POWER: Still Not Viable without Subsidies

Executive Summary

February 2011

Conspicuously absent from industry press releases and briefing memos touting nuclear power's potential as a solution to global warming is any mention of the industry's long and expensive history of taxpayer subsidies and excessive charges to utility ratepayers. These subsidies not only enabled the nation's existing reactors to be built in the first place, but have also supported their operation for decades.

The industry and its allies are now pressuring all levels of government for large new subsidies to support the construction and operation of a new generation of reactors and fuel-cycle facilities. The substantial political support the industry has attracted thus far rests largely on an uncritical acceptance of the industry's economic claims and an incomplete understanding of the subsidies that made—and continue to make—the existing nuclear fleet possible.

Such blind acceptance is an unwarranted, expensive leap of faith that could set back more cost-effective efforts to combat climate change. A fair comparison of the available options for reducing heat-trapping carbon emissions while generating electricity requires consideration not only of the private costs of building plants and their associated infrastructure but also of the public subsidies given to the industry. Moreover, nuclear power brings with it important economic, waste disposal, safety, and security risks unique among low-carbon energy sources. Shifting these risks and their associated costs onto the public is the major goal of the new subsidies sought by the industry (just as it was in the past), and by not incorporating these costs into its estimates, the industry presents a skewed economic picture of nuclear power's value compared with other low-carbon power sources.

SUBSIDIES OFTEN EXCEED THE VALUE OF THE ENERGY PRODUCED

This report catalogues in one place and for the first time the full range of subsidies that benefit the nuclear power sector. The findings are striking: since its inception more than 50 years ago, the nuclear power industry has benefited—and continues to benefit—from a vast array of preferential government subsidies. Indeed, as Figure ES-1 (p. 2) shows, subsidies to the nuclear fuel cycle have often exceeded the value of the power produced. This means that buying power on the open market and giving it away for free would have been less costly than subsidizing the construction and operation of nuclear power plants. Subsidies to new reactors are on a similar path.

Since its inception more than 50 years ago, the nuclear power industry has benefited—and continues to benefit—from a vast array of preferential government subsidies.

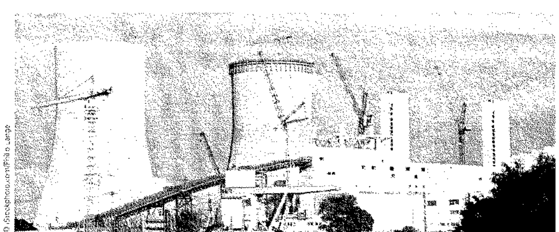
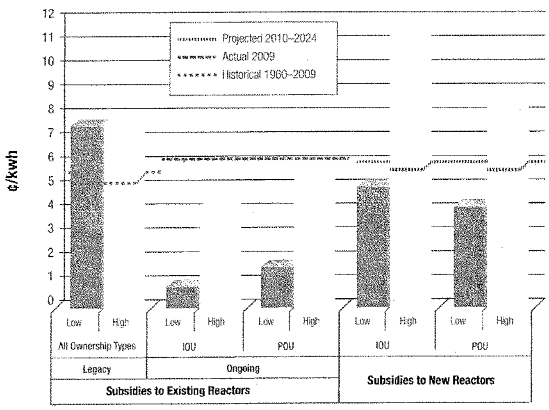


Figure ES-1. Nuclear Subsidies Compared to EIA Power Prices



Note: Legacy subsidies are compared to the Energy Information Administration (EIA) average 1960–2009 industrial power price (5.4 ¢/kWh). Ongoing subsidies are compared to EIA 2009 actual power prices for comparable busbar plant generation costs (5.9 ¢/kWh). Subsidies to new reactors are compared to EIA 2009 reference-case power prices for comparable busbar plant generation costs (5.7 ¢/kWh).

Throughout its history, the industry has argued that subsidies were only temporary, a short-term stimulus so the industry could work through early technical hurdles that prevented economical reactor operation. A 1954 advertisement from General Electric stated that, "In five years—certainly within ten," civilian reactors would be "privately financed, built without government subsidy." That day never arrived and, despite industry claims to the contrary, remains as elusive as ever.

The most important subsidies to the industry do not involve cash payments. Rather, they shift construction-cost and operating risks from investors to taxpayers and ratepayers, burdening taxpayers with an array of risks ranging from cost overruns and defaults to accidents and nuclear waste management. This approach, which has remained remarkably consistent throughout the industry's history, distorts market choices that would otherwise favor less risky investments. Although it may not involve direct cash payments, such favored treatment is nevertheless a subsidy, with a profound effect on the bottom line for the industry and taxpayers alike.

Reactor owners, therefore, have never been economically responsible for the full costs and risks of their operations. Instead, the public faces the prospect of severe losses in the event of any number of potential adverse scenarios, while private investors reap the rewards if nuclear plants are economically successful. For all practical purposes, nuclear power's economic gains are privatized, while its risks are socialized.

Recent experiences in the housing and financial markets amply demonstrate the folly of arrangements that separate investor risk from reward. Indeed, massive new subsidies to nuclear power could encourage utilities to make similarly speculative, expensive investments in nuclear plants—investments that would never be tolerated if the actual risks were properly accounted for and allocated.

While the purpose of this report is to quantify the extent of past and existing subsidies, we are not blind to the context: the industry is calling for even more support from Congress. Though the value of these new subsidies is not quantified in this report, it is clear that they would only further increase the taxpayers' tab for nuclear power while shifting even more of the risks onto the public.

LOW-COST CLAIMS FOR EXISTING REACTORS IGNORE HISTORICAL SUBSIDIES

The nuclear industry is only able to portray itself as a low-cost power supplier today because of past government subsidies and write-offs. First, the industry received massive subsidies at its inception, reducing both the capital costs it needed to recover from ratepayers (the "legacy" subsidies that underwrote reactor construction through the 1980s) and its operating costs (through ongoing subsidies to inputs, waste management, and accident risks). Second, the industry wrote down tens of billions of dollars in capital costs after its first generation of reactors experienced large cost overruns, cancellations, and plant abandonments, further reducing the industry's capital-recovery requirements. Finally, when industry restructuring revealed that nuclear power costs were still too high to be competitive, so-called stranded costs were shifted to utility ratepayers, allowing the reactors to continue operating.

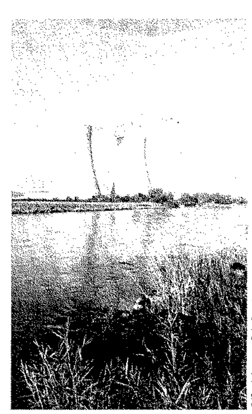
These legacy subsidies are estimated to exceed seven cents per kilowatt-hour (¢/kWh)—an amount equal to about 140 percent of the average wholesale price of power from 1960 to 2008, making the subsidies more valuable than the power produced by nuclear plants over that period. Without these subsidies, the industry would have faced a very different market reality—one in which many reactors would never have been built, and utilities that did build reactors would have been forced to charge consumers even higher rates.

ONGOING SUBSIDIES CONTRIBUTE TO NUCLEAR POWER'S PERCEIVED COST ADVANTAGE

In addition to legacy subsidies, the industry continues to benefit from subsidies that offset the costs of uranium, insurance and liability, plant security, cooling water, waste disposal, and plant decommissioning. The value of these subsidies is harder to pin down with specificity, with estimates ranging from a low

Massive new subsidies to nuclear power could encourage speculative, expensive investments in nuclear plants that would never be tolerated if the actual risks were properly accounted for and allocated.

of 13 percent of the value of the power produced to a high of 98 percent. The breadth of this range largely reflects three main factors: uncertainty over the dollar value of accident liability caps; the value to publicly owned utilities (POUs) of ongoing subsidies such as tax breaks and low return-on-investment requirements; and generous capital



Legacy and ongoing subsidies to existing reactors are not sufficient to attract new investment in nuclear infrastructure. Thus an array of new subsidies was rolled out during the past decade, targeting not only reactors but also other fuel-cycle facilities

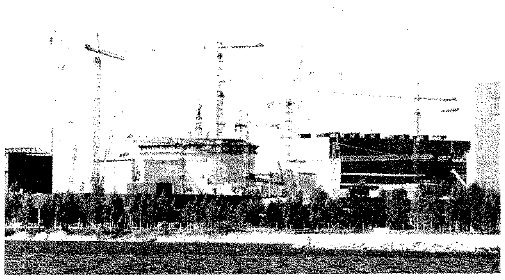
subsidies to investor-owned utilities (IOUs) that have declined as the aging, installed capacity base is fully written off.

Our low-end estimate for subsidies to existing reactors (in this case, investor-owned facilities) is 0.7 ¢/kWh, a figure that may seem relatively small at only 13 percent of the value of the power produced. However, it represents more than 35 percent of the nuclear production costs (operation and maintenance costs plus fuel costs, without capital recovery) often cited by the industry's main trade association as a core indicator of nuclear power's competitiveness; it also represents nearly 80 percent of the production-cost advantage of nuclear relative to coal. With ongoing subsidies to POUs nearly double those to IOUs, the impact on competitive viability is proportionally higher for publicly owned plants.

SUBSIDIES TO NEW REACTORS REPEAT PAST PATTERNS

Legacy and ongoing subsidies to existing reactors may be important factors in keeping facilities operating, but they are not sufficient to attract new investment in nuclear infrastructure. Thus an array of new subsidies was rolled out during the past decade, targeting not only reactors but also other fuel-cycle facilities. Despite the profoundly poor investment experience with taxpayer subsidies to nuclear plants over the past 50 years, the objectives of these new subsidies are precisely the same as the earlier subsidies: to reduce the private cost of capital for new nuclear reactors and to shift the long-term, often multi-generational risks of the nuclear fuel cycle away from investors. And once again, these subsidies to new reactors—whether publicly or privately owned—could end up exceeding the value of the power produced (4.2 to 11.4 ¢/kWh, or 70 to 200 percent of the projected value of the power).

It should be noted that certain subsidies to new reactors are currently capped at a specific dollar amount, limited to a specific number of reactors, or available only in specific states or localities. Therefore, although all the subsidies may not be available to each new reactor, the values shown in Figure ES-1 are reasonably representative of the subsidies that will be available to the first new plants to be built. Furthermore, it is far from clear whether existing caps will be binding. Recent legislative initiatives would expand eligibility for these subsidies to even more reactors and extend the period of eligibility during which these subsidies would be available.



KEY SUBSIDY FINDINGS

Government subsidies have been directed to every part of the nuclear fuel cycle. The most significant forms of support have had four main goals: reducing the cost of

Methodology: How We Estimated Nuclear Subsidies

Identifying and valuing subsidies to the nuclear fuel cycle for this report involved a broad review of dozens of historical studies and program assessments, industry statements and presentations, and government documents. The result is an in-depth and comprehensive evaluation that groups nuclear subsidies by type of plant ownership (public or private), time frame of support (whether the subsidy is ongoing or has expired), and the specific attribute of nuclear power production the subsidy is intended to support.

Plant ownership

Subsidies available to investor-owned and publicly owned utilities are not identical, so were tracked separately.

Time frame of support

The data were organized into:

- **Legacy subsidies**, which were critical in helping nuclear power gain a solid foothold in the U.S. energy sector but no longer significantly affect pricing
- **Ongoing subsidies to existing reactors**, which continue to affect the cost of electricity produced by the 104 U.S. nuclear reactors operating today
- **Subsidies to new reactors**, which are generally provided in addition to the ongoing subsidies available to existing reactors

A further set of subsidies proposed for the nuclear sector but not presently in U.S. statutes is discussed qualitatively but not quantified.

Attribute of production

The following subcategories were modeled on the structure commonly used internationally (as by the Organisation for Economic Cooperation and Development):

- **Factors of production**—subsidies intended to offset the cost of capital, labor, and land
- **Intermediate inputs**—subsidies that alter the economics of key inputs such as uranium, enrichment services, and cooling water
- **Output-linked support**—subsidies commensurate with the quantity of power produced
- **Security and risk management**—subsidies that address the unique and substantial safety risks inherent in nuclear power
- **Decommissioning and waste management**—subsidies that offset the environmental or plant-closure costs unique to nuclear power

To enable appropriate comparisons with other energy options, the results are presented in terms of levelized cents per kilowatt-hour and as a share of the wholesale value of the power produced. Inclusion of industry and historical data sources for some component estimates means that some of the levelization inputs were not transparent. Where appropriate, a range of estimates was used to reflect variation in the available data or plausible assumptions.

capital, labor, and land (i.e., factors of production), masking the true costs of producing nuclear energy ("intermediate inputs"), shifting security and accident risks to the public, and shifting long-term operating risks (decommissioning and waste management) to the public. A new category of subsidy, "output-linked support," is directed at reducing the price of power produced. Table ES-1 (p. 6) shows the estimated value of these subsidies to existing and new reactors. The subsequent sections discuss each type of subsidy in more detail.

A. Reducing the Cost of Capital, Labor, and Land (Factors of Production)

Nuclear power is a capital-intensive industry with long and often uncertain build times that exacerbate both the cost of financing during construction and the market risks of misjudging demand. Historically, investment tax credits, accelerated depreciation, and other capital subsidies have been the dominant type of government support for the industry, while subsidies associated with labor and land costs have provided lesser (though still relevant) support.

Legacy subsidies that reduced the costs of these inputs were high, estimated at 7.2 ¢/kWh. Ongoing subsidies to existing reactors are much lower but still significant, ranging from 0.06 to 1.94 ¢/kWh depending on ownership structure. For new reactors, accelerated depreciation has been supplemented with a variety of other capital subsidies to bring plant costs down by shifting a large portion of the capital risk from investors to taxpayers.

The total value of subsidies available to new reactors in this category is significant for both POUs and IOUs, ranging from 3.51 to 6.58 ¢/kWh. These include:

- **Federal loan guarantees.** Authorized under Title 17 of the Energy Policy Act (EPACT) of 2005, federal loan guarantees are the largest construction subsidy for new, investor-owned reactors, effectively shifting the costs and risks of financing and building a nuclear plant from investors to taxpayers. The industry's own estimates, which we have used despite large subsequent increases in expected plant costs, place the value of this program between 2.5 and 3.7 ¢/kWh. Total loan guarantees are currently limited to

Federal loan guarantees are the largest construction subsidy for new, investor-owned reactors, effectively shifting the costs and risks of financing and building a nuclear plant from investors to taxpayers.

$22.5 billion for new plants and enrichment facilities, but the industry has been lobbying for much higher levels.

Loan guarantees not only allow firms to obtain lower-cost debt, but enable them to use much more of it—up to 80 percent of the project's cost. For a

Table ES-1. Subsidies to Existing and New Reactors

	Subsidies to Existing Reactors (¢/kWh)			Subsidies to New Reactors (¢/kWh)	
	Legacy	Ongoing			
Subsidy Type	All Ownership Types	IOU	POU	IOU	POU
Factors of production	7.20	0.06	0.96–1.94	3.51–6.58	3.73–5.22
Intermediate inputs	0.10–0.24	0.29–0.51	0.16–0.18	0.21–0.42	0.21–0.42
Output-linked support	0.00	0.00	0.00	1.05–1.45	0.00
Security and risk management	0.21–0.22	0.10–2.50	0.10–2.50	0.10–2.50	0.10–2.50
Decommissioning and waste management	No data available	0.29–1.09	0.31–1.15	0.13–0.48	0.16–0.54
Total	**7.50–7.66**	**0.74–4.16**	**1.53–5.77**	**5.01–11.42**	**4.20–8.68**
Share of power price	139%–142%	13%–70%	26%–98%	84%–190% (high)	70%–145% (high)
				88%–200% (reference)	74%–152% (reference)

Note: A range of subsidy values is used where there was a variance in available subsidy estimates. To determine the subsidy's share of the market value of the power produced, legacy subsidies are compared to the Energy Information Administration (EIA) average 1960–2009 industrial power price (5.4 ¢/kWh). Ongoing subsidies are compared to EIA 2009 power prices for comparable busbar plant generation costs (5.9 ¢/kWh). Subsidies to new reactors are compared to EIA 2009 high- and reference-case power prices for comparable busbar plant generation costs (6.0 and 5.7 ¢/kWh, respectively); using the low case would have resulted in even higher numbers.

single 1,600-megawatt (MW) reactor, the loan guarantee alone would generate subsidies of $495 million per year, or roughly $15 billion over the 30-year life of the guarantee.

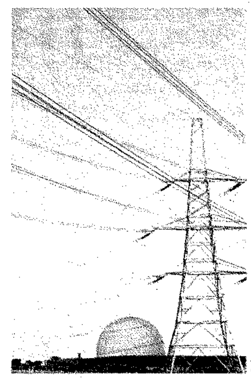
- **Accelerated depreciation.** Allowing utilities to depreciate new reactors over 15 years instead of their typical asset life (between 40 and 60 years) will provide the typical plant with a tax break of approximately $40 million to $80 million per year at current construction cost estimates. Rising plant costs, longer service lives, and lower capacity factors would all increase the value of current accelerated depreciation rules to IOUs. This subsidy is not available to POUs because they pay no taxes.
- **Subsidized borrowing costs to POUs.** The most significant subsidy available to new publicly owned reactors is the reduced cost of borrowing made possible by municipal bonds and new Build America Bonds, which could be worth more than 3 ¢/kWh.

The most significant subsidy available to new publicly owned reactors is the reduced cost of borrowing made possible by municipal bonds and new Build America Bonds.

- **Construction work in progress.** Many states allow utilities to charge ratepayers for construction work in progress (CWIP) by adding a surcharge to customers' bills. This shifts financing and construction risks (including the risk of cost escalations and/or plants being abandoned during construction) from investors to customers. CWIP benefits both POUs and IOUs and is estimated to be worth between 0.41 and 0.97 ¢/kWh for new reactors.
- **Property-tax abatements.** Support for new plants is also available through state and local governments, which provide a variety of plant-specific subsidies that vary by project.

B. Masking the True Costs of Producing Nuclear Energy (Intermediate Inputs)

A variety of subsidies masks the costs of the inputs used to produce nuclear power. Uranium fuel costs, for example, are not a major element in nuclear economics, but subsidies to mining and enrichment operations contribute to the perception of nuclear power as a low-cost energy source. In addition, the under-pricing of water used in bulk by nuclear reactors has significant cost implications. The value of such legacy subsidies to existing reactors is estimated between 0.10 and 0.24 ¢/kWh, and the value of ongoing subsidies is estimated between 0.16 and 0.51 ¢/kWh. The value of



such subsidies to new reactors is estimated between 0.21 and 0.42 ¢/kWh. Subsidized inputs include:

- **Fuel.** The industry continues to receive a special depletion allowance for uranium mining equal to 22 percent of the ore's market value, and its deductions are allowed to exceed the gross investment in a given mine. In addition, uranium mining on public lands is governed by the antiquated Mining Law of 1872, which allows valuable ore to be taken with no royalties paid to taxpayers. Although no relevant data have been collected on the approximately 4,000 mines from which uranium has been extracted in the past, environmental remediation costs at some U.S. uranium milling sites actually exceeded the market value of the ore extracted.

- **Uranium enrichment.** Uranium enrichment, which turns mined ore into reactor fuel, has benefited from substantial legacy subsidies. New plants that add enrichment capacity will receive subsidies as well, in the form of federal loan guarantees. Congress has already authorized \$2 billion in loan guarantees for a new U.S. enrichment facility, and the Department of Energy has allocated an additional \$2 billion for this purpose. While we could not estimate the per-kilowatt-hour cost of this subsidy because it depends on how much enrichment capacity is built, the \$4 billion represents a significant new subsidy to this stage of the fuel cycle.
- **Cooling water.** Under-priced cooling water is an often-ignored subsidy to nuclear power, which is the most water-intensive large-scale thermal energy technology in use. Even when the water is returned to its source, the large withdrawals alter stream flow and thermal patterns, causing environmental damage. Available data suggest that reactor owners pay little or nothing for the water consumed, and are often given priority access to water resources—including exemption from drought restrictions that affect other users. While we provide a low estimate of water subsidies (between \$600 million and \$700 million per year for existing reactors), more work is needed to accurately quantify this subsidy—particularly as water resources become more constrained in a warming climate.

Nuclear power is the most water-intensive large-scale thermal energy technology in use. The large withdrawals alter stream flow and thermal patterns, causing environmental damage.

C. Reducing the Price of Power Produced (Output-Linked Support)

Until recently, subsidies linked to plant output were not a factor for nuclear power. That changed with the passage of EPACT in 2005, which granted new reactors an important subsidy in the form of:

- **Production tax credits (PTCs).** A PTC will be granted for each kilowatt-hour generated during a new reactor's first eight years of operation; at present, this credit is available only to the first plants to be built, up to a combined total capacity of six gigawatts. While EPACT provides a nominal PTC of 1.8 ¢/kWh, payments are time-limited. Over the full life of the plant, the PTC is worth between 1.05 and 1.45 ¢/kWh. Under current law, PTCs are not available to POUs (since POUs do not pay taxes), but there have been legislative efforts to enable POUs to capture the value of the tax credits by selling or transferring them to other project investors that do pay taxes.

D. Shifting Security and Accident Risks to the Public (Security and Risk Management)

Subsidies that shift long-term risks to the public have been in place for many years. The Price-Anderson Act, which caps the nuclear industry's liability for third-party damage to people and property, has been a central subsidy to the industry for more than half a century.

Plant security concerns have increased significantly since 9/11, and proliferation risks will increase in proportion to any expansion of the civilian nuclear sector (both in the United States and abroad). The complexity and lack of data in these areas made it impossible to quantify the magnitude of security subsidies for this analysis. But it is clear that as the magnitude of the threat increases, taxpayers will be forced to bear a greater share of the risk. Subsidies that shift these risks are associated with:

- **The Price-Anderson Act.** This law requires utilities to carry a pre-set amount of insurance for off-site damages caused by a nuclear plant accident, and to contribute to an additional

The Industry's Shopping List: New Subsidies Under Consideration

The following nuclear subsidies, as proposed in the American Power Act (APA) and the American Clean Energy Leadership Act (ACELA), would not necessarily be available to every new reactor, but their collective value to the industry would be significant:

- A clean-energy bank that could promote nuclear power through much larger loans, letters of credit, loan guarantees, and other credit instruments than is currently possible
- Tripling federal loan guarantees available to nuclear reactors through the Department of Energy, from $18.5 billion to $54 billion
- Reducing the depreciation period for new reactors from 15 years to five
- A 10 percent investment tax credit for private investors or federal grants in lieu of tax payments to publicly owned and cooperative utilities
- Expanding the existing production tax credit from 6,000 to 8,000 megawatts, and permitting tax-exempt entities to allocate their available credits to private partners
- Permitting tax-exempt bonds to be used for public-private partnerships, which would allow POUs to issue tax-free, low-cost bonds for nuclear plants developed jointly with private interests
- Expanding federal regulatory risk insurance coverage from $2 billion to $6 billion (up to $500 million per reactor), which would further shield plant developers from costs associated with regulatory or legal delays

pool of funds meant to cover a pre-set portion of the damages. However, the law limits total industry liability to a level much lower than would be needed in a variety of plausible accident scenarios. This constitutes a subsidy when compared with other energy sources that are required to carry full private liability insurance, and benefits both existing and new reactors.

Only a few analysts have attempted to determine the value of this subsidy over its existence, with widely divergent results: between 0.1 and 2.5 ¢/kWh. More work is therefore needed to determine how the liability cap affects plant economics, risk-control decisions, and risks to the adjacent population.

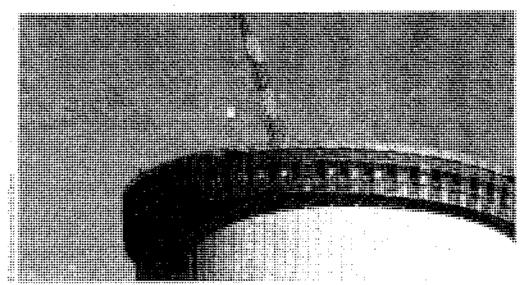
- **Plant security.** Reactor operators must provide security against terrorist attacks or other threats of a certain magnitude, referred to as the "design basis threat." For threats of a greater magnitude (a larger number of attackers, for example), the government assumes all financial responsibility, which constitutes another type of subsidy. It is difficult to quantify the value of this taxpayer-provided benefit because competing forms of energy do not carry similar risks. But it is important that plant security costs be reflected in the cost of power delivered to consumers, rather than supported by taxpayers in general.
- **Proliferation.** The link between an expanded civilian nuclear sector and proliferation of nuclear weapons or weapons technology is fairly widely accepted. It is also consistently ignored when assessing plant costs—much as investors in coal plants ignored the cost of carbon controls until recently. Though quantifying proliferation costs may be difficult, assuming they are zero is clearly wrong. These ancillary impacts should be fully assessed and integrated into the cost of nuclear power going forward.

E. Shifting Long-Term Operating Risks to the Public (Decommissioning and Waste Management)

The nuclear fuel cycle is unique in the types of long-term liabilities it creates. Reactors and fuel-cycle facilities have significant end-of-life liabilities associated with the proper closure, decommissioning, and decontamination of facilities, as well as the safe management of nuclear waste over thousands of years. The industry has little operational experience with such large and complex undertakings, greatly increasing the likelihood of dramatic cost overruns. In total, the subsidies that shift these long-term operating risks to the public amount to between 0.29 and 1.09 ¢/kWh for existing reactors and between 0.13 and 0.54 ¢/kWh for new reactors. The specific subsidies that do the shifting are associated with:

- **Nuclear waste management.** The federal Nuclear Waste Repository for spent fuel is expected to cost nearly $100 billion over its projected operating life, 80 percent of which is attributed to the power sector. A congressionally mandated fee on nuclear power consumers, earmarked for the repository, has collected roughly $31 billion in waste-disposal fees through 2009. There is no mechanism other than investment returns on collections to fully fund the repository once reactors close.

  The repository confers a variety of subsidies to the nuclear sector. First, despite its complexity and sizable investment, the repository is structured to operate on a break-even basis at best, with no required return on investment. Second, utilities do not have to pay any fee to secure repository capacity; in fact, they are allowed to defer payments for waste generated prior to the repository program's creation, at interest rates well below their cost of capital. Third, the significant risk of delays and cost overruns will be borne by taxpayers rather than the program's beneficiaries. Delays in the repository's opening have already triggered a rash of lawsuits and taxpayer-funded waste storage at reactor sites, at a cost between $12 billion and $50 billion.



Reactors and fuel-cycle facilities have significant end-of-life liabilities associated with the proper closure, decommissioning, and decontamination of facilities, as well as the safe management of nuclear waste over thousands of years.

- **Plant decommissioning.** While funds are collected during plant operation for decommissioning once the plant's life span has ended, reduced tax rates on nuclear decommissioning trust funds provide an annual subsidy to existing reactors of between $450 million and $1.1 billion per year. Meanwhile, concerns persist about whether the funds accrued will be sufficient to cover the costs; in 2009, the Nuclear Regulatory Commission (NRC) notified the operators of roughly one-quarter of the nation's reactor fleet about the potential for insufficient funding. We did not quantify the cost of this potential shortfall.

CONCLUSIONS AND POLICY RECOMMENDATIONS

Historical subsidies to nuclear power have already resulted in hundreds of billions of dollars in costs paid by taxpayers and ratepayers. With escalating plant costs and more competitive power markets, the cost of repeating these failed policies will likely be even higher this time around. Of equal importance, however, is the fact that subsidies to nuclear power also carry significant opportunity costs for reducing global warming emissions because reactors are so expensive and require such long lead times to construct. In other words, massive subsidies designed to help underwrite the large-scale expansion of the nuclear industry will delay or diminish investments in less expensive abatement options.

Other energy technologies would be able to compete with nuclear power far more effectively if the government focused on creating an energy-neutral playing field rather than picking technology winners and losers. The policy choice to invest in nuclear also carries with it a risk unique to the nuclear fuel cycle: greatly exacerbating already thorny proliferation challenges as reactors and ancillary fuel-cycle facilities expand throughout the world.

As this report amply demonstrates, taxpayer subsidies to nuclear power have provided an indispensable foundation for the industry's existence, growth, and survival. But instead of reworking its business model to more effectively manage and internalize its operational and construction risks, the industry is pinning its hopes on a new wave of taxpayer subsidies to prop up a new generation of reactors.

Future choices about U.S. energy policy should be made with a full understanding of the hidden taxpayer costs now embedded in nuclear power. To accomplish this goal, we offer the following recommendations:

- **Reduce, not expand, subsidies to the nuclear power industry.** Federal involvement in energy markets should instead focus on encouraging firms involved in nuclear power—some of the largest corporations in the world—to create new models for internal risk pooling and to develop advanced power contracts that enable high-risk projects to move forward without additional taxpayer risk.
- **Award subsidies to low-carbon energy sources on the basis of a competitive bidding process across all competing technologies.** Subsidies should be awarded to those approaches able to achieve emissions reductions at the lowest possible cost per unit of abatement—not on the basis of congressional earmarks for specific types of energy.
- **Modernize liability systems for nuclear power.** Liability systems should reflect current options in risk syndication, more robust requirements for the private sector, and more extensive testing of the current rules for excess risk concentration and counterparty risks. These steps are necessary to ensure coverage will actually be available when needed, and to send more accurate risk-related price signals to investors and power consumers.
- **Establish proper regulation and fee structures for uranium mining.** Policy reforms are needed to eliminate outdated tax subsidies, adopt market-level royalties for uranium mines on public lands, and establish more appropriate bonding regimes for land reclamation.
- **Adopt a more market-oriented approach to financing the Nuclear Waste Repository.** The government should require sizeable waste management deposits by the industry, a repository fee structure that earns a return on investment at least comparable to other large utility projects,

Other energy technologies would be able to compete with nuclear power far more effectively if the government focused on creating an energy-neutral playing field rather than picking technology winners and losers.

and more equitable sharing of financial risks if additional delays occur.

- **Incorporate water pricing to allocate limited resources among competing demands, and integrate associated damages from large withdrawals.** The government should establish appropriate benchmarks for setting water prices that will be paid by utilities and other consumers, using a strategy that incorporates ecosystem damage as well as consumption-based charges.
- **Repeal decommissioning tax breaks and ensure greater transparency of nuclear decommissioning trusts (NDTs).** Eliminating existing tax breaks for NDTs would put nuclear power on a similar footing with other energy sources. More detailed and timely information on NDT funding and performance should be collected and publicized by the NRC.
- **Ensure that publicly owned utilities adopt appropriate risk assessment and asset management procedures.** POUs and relevant state regulatory agencies should review their internal procedures to be sure the financial and delivery risks of nuclear investments are appropriately compared with other options.
- **Roll back state construction-work-in-progress allowances and protect ratepayers against cost overruns by establishing clear limits on customer exposure.** States should also establish a refund mechanism for instances in which plant construction is cancelled after it has already begun.
- **Nuclear power should not be eligible for inclusion in a renewable portfolio standard.** Nuclear power is an established, mature technology with a long history of government support. Furthermore, nuclear plants are unique in their potential to cause catastrophic damage (due to accidents, sabotage, or terrorism); to produce very long-lived radioactive wastes; and to exacerbate nuclear proliferation.
- **Evaluate proliferation and terrorism as an externality of nuclear power.** The costs of preventing nuclear proliferation and terrorism should be recognized as negative externalities of civilian nuclear power, thoroughly evaluated, and integrated into economic assessments—just as global warming emissions are increasingly identified as a cost in the economics of coal-fired electricity.
- **Credit support for the nuclear fuel cycle via export credit agencies should explicitly integrate proliferation risks and require project-based credit screening.** Such support should require higher interest rates than those extended to other, less risky power projects, and include conditions on fuel-cycle investments to ensure the lending does not contribute to proliferation risks in the recipient country.

The full text of this report is available on the UCS website at [illegible]

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Nuclear Power: Still Not Viable without Subsidies was prepared for UCS by Doug Koplow, president and founder of Earth Track.

The Union of Concerned Scientists is the leading science-based nonprofit working for a healthy environment and a safer world.

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Citizens and Scientists for Environmental Solutions

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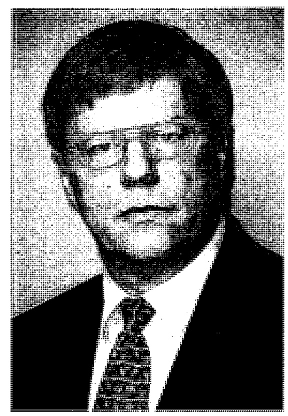
Union of Concerned Scientists
Citizens and Scientists for Environmental Solutions

David Lochbaum is the Director of the Nuclear Safety Project for the Union of Concerned Scientists.



David Lochbaum is one of the nation's top independent experts on nuclear power. At UCS, he monitors safety issues at the nation's nuclear power plants, raises concerns with the Nuclear Regulatory Commission, and responds to breaking events, such as current concerns over aging power plants and plant fire safety.

Mr. Lochbaum is a nuclear engineer by training and worked in nuclear power plants for 17 years. In 1992, he and a colleague identified a safety problem in a plant where they were working, but were ignored when they raised the issue with the plant manager, the utility, and the Nuclear Regulatory Commission. They decided to go to Congress, and the problem was eventually corrected at the original plant and at plants across the country. Concerned about nuclear safety and frustrated with the NRC's complacency, Mr. Lochbaum joined UCS in 1996. Mr. Lochbaum left UCS in 2009 to accept a position as a reactor technology instructor at the NRC's Technical Training Center. Mr. Lochbaum returned to UCS in his old position in March 2010.

He has written numerous reports, including *The Good, the Bad, and the Ugly: A Report on Safety in America's Nuclear Power Industry, Three Mile Island's Puzzling Legacy,* and the book *Nuclear Waste Disposal Crisis.* He is widely quoted in the media and a frequent guest on network news programs.

Enclosure 3

Committee on Science, Space, and Technology
U.S. House of Representatives
Witness Disclosure Requirement - "Truth in Testimony"
Required by House Rule XI, Clause 2(g)(5)

1. Your Name: DAVID LOCHBAUM		
2. Are you testifying on behalf of the Federal, or a State or local government entity?	Yes	No X
3. Are you testifying on behalf of an entity that is not a government entity?	Yes X	No
4. Other than yourself, please list which entity or entities you are representing: Union of Concerned Scientists.		
5. Please list any Federal grants or contracts (including subgrants or subcontracts) that you or the entity you represent have received on or after October 1, 2008: I have received none. Lisbeth Gronlund, my supervisor, told me that UCS has received none.		
6. If your answer to the question in item 3 in this form is "yes," please describe your position or representational capacity with the entity(ies) you are representing: Director, Nuclear Safety Project		
7. If your answer to the question in item 3 is "yes," do any of the entities disclosed in item 4 have parent organizations, subsidiaries, or partnerships that you are not representing in your testimony?	Yes	No X
8. If the answer to the question in item 3 is "yes," please list any Federal grants or contracts (including subgrants or subcontracts) that were received by the entities listed under the question in item 4 on or after October 1, 2008, that exceed 10 percent of the revenue of the entities in the year received, including the source and amount of each grant or contract to be listed: None, per item 5.		

I certify that the above information is true and correct.

Signature: David A. Lochbaum   Date: 5-9-2011

Dr. HARRIS. Thank you very much, Mr. Lochbaum. The Committee will now recess so we can go and vote. We will reconvene five minutes after the last vote.

Committee is in recess.

[Recess.]

Chairman BROUN. I want to thank the witnesses for your indulgence and apologize for the break, but we will try to expedite this. I want to thank the panel for your testimony. I remind Members that the Committee rules limit questioning to five minutes.

The Chair, at this point, will open the round of questions. The Chair recognizes himself for five minutes.

I am concerned nuclear groups will exploit the tragedy in Fukushima as an excuse to halt not only future expansion of nuclear power, but restrict relicensing of existing plants. Dr. Sheron, are you—I have a hard time pronouncing it—Chairman Jaczko and NRC committed to continue moving forward with reviewing the application—license application for the Vogtle plant in Georgia? What commitment can you provide that your office will continue to provide the necessary information for these licenses to advance?

Dr. SHERON. Right now the Agency does not believe that there are any impediments to the continued either licensing of new plants or the renewed license of existing plants, such as the Vogel plant. So the Agency, as I understand, is moving forward with the relicensing of the plant, the review, and provided that the licensee provides all of the required information, I believe they will maintain on the agreed upon schedule.

Chairman BROUN. Well, I certainly hope so. It is absolutely critical for us to go forward in as expeditious a manner as possible, and I would encourage you to do so.

The impetus for this hearing was the tragic event in Japan. Since then, the American south has experienced a tragedy of its own, in fact, even in my north Georgia district, several of my counties have been hit by that tragedy. Recent tornados in Alabama and the flooding of the Mississippi River unfortunately provide another opportunity for us to learn. How has the NRC incorporated in lessons learned from the recent events in the South? It has been reported that some reactors were taken offline as a result of the extreme weather. To your knowledge, were there any problems with any of these? How will this impact NRC's research portfolio, and how did the previous safety reviews prepare the U.S. for these events?

Dr. SHERON. The events, the tornados that took place in the South did take down some transmission lines at some plants, which did cause loss of offsite power. My understanding is the emergency diesel generators at those sites did work as designed.

We look at all natural phenomena that occur in the United States. We confer with other agencies, as I said before, like USGS, to determine if there is any new information that we need to take into account in the design of these plants. Nuclear plants are designed for tornados, for high winds, for storms. We look at floods that might occur in the vicinity when these plants are licensed to make sure that they are designed such that they can handle them.

If we learn anything new that says the current design base for these plants is not adequate, then obviously the Agency will take action to make the plant—install, you know, whatever corrections are necessary.

Chairman BROUN. The answer is no problem at this point?

Dr. SHERON. Yes.

Chairman BROUN. Okay, very good. Now that this Administration has decided to ignore the law and clear congressional direction, our Nation has no long-term storage plans for radioactive waste. Where is spent fuel stored at Fukushima? Where is the U.S. currently storing its spent fuel? How many sites have currently filled their available storage space? Have any waivers been granted or regulatory changes made to allow greater onsite storage, and has any comparative risk analysis been done to compare centralized storage with dispersed storage? Doctor?

Dr. SHERON. The spent fuel at the Fukushima plant, as I understand, was stored on the site in the pools. I do not know if they had any dry cask storage. At the U.S. right now, plants store their fuel either at—in their spent fuel pools which have been designed to handle the amount of fuel that they can put in, that they can hold, or to independent spent fuel storage facilities, ISFS, they are called. Usually these are dry casks that are stored onsite or nearby, and are basically—require air cooling.

Chairman BROUN. Are you going to allow expansion of those local pools since the Administration has closed down the Yucca Mountain storage facility?

Dr. SHERON. Some licensees have come in and proposed to rerack the pools, which is to "do" a more dense configuration where they can hold more fuel. Licensees have to come in and present a safety analysis to demonstrate why that is acceptable and safe. I can't tell you which ones have done that so far. I don't have that information with me. I know there are some plants that do have the high density fuel racks.

With regard to a comparative risk study, with regard to—let me call it a minimally loaded spent fuel pool versus a fully loaded one, my office is beginning to undertake a comparative risk study to see what the differences are in risk to public health and safety between the two. My personal opinion is that pools have a lot of water in them, and regardless of the amount of fuel, it takes a very long time, if there was an accident, to actually drain the pool to the point where there would be an uncovering of the fuel, which gives licensees ample time to bring in either emergency equipment or to restore whatever did fail.

Usually—and even if it was a drain down that was occurring or a boil off, the amount of time that is available before one actually starts as a release of radioactivity provides ample time for evacuation in the vicinity of the site so that people could be evacuated and there wouldn't be any harmful radiation effects.

Chairman BROUN. Thank you, Doctor. Your answer just further points out the need to open up Yucca Mountain for the Administration to start obeying the law.

I now recognize Ms. Edwards for five minutes, and I will give you some leeway on that, Ms. Edwards. You are recognized for five minutes.

Ms. EDWARDS. Thank you, Mr. Chairman, and thank you to the witnesses for your patience. Before I begin questions, I would like to ask the Chairman for unanimous consent to enter two Nuclear Regulatory Commission reports relating to the shutdown at Calvert Cliffs that I referenced earlier, and a report by Mr. Lochbaum at the Union of Concerned Scientists on the 14 near-misses at U.S. power plants and their safety.

Chairman BROUN. Any objections? Hearing no objections, so ordered.

[The information follows:]

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PA 19406-1415

June 14, 2010

EA-10-080

George H. Gellrich, Vice President
Calvert Cliffs Nuclear Power Plant, LLC
Constellation Energy Nuclear Group, LLC
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT - NRC SPECIAL INSPECTION REPORT 05000317/2010006 AND 05000318/2010006; PRELIMINARY WHITE FINDING

Dear Mr. Gellrich:

On April 30, 2010, the U. S. Nuclear Regulatory Commission (NRC) completed a Special Inspection of the February 18, 2010, dual unit trip at Calvert Cliffs Nuclear Power Plant (CCNPP) Units 1 and 2. The enclosed report documents the inspection results, which were discussed on April 30, 2010, with you and other members of your staff.

The special inspection was conducted in response to the dual unit trip with complications on February 18, 2010. The complications included loss of a 500 kilovolt (kV) offsite power supply to each unit, loss of power to a 4 kV safety bus on each unit, failure of the 2B emergency diesel generator (EDG) to reenergize a 4 kV safety bus, loss of power to the Unit 2 4 kV non-safety buses, loss of Unit 2 forced reactor coolant system (RCS) flow, and loss of the Unit 2 normal heat sink. The NRC's initial evaluation of this event satisfied the criteria in NRC Inspection Manual Chapter 0309, "Reactive Inspection Decision Basis for Reactors," for conducting a special inspection. The Special Inspection Team (SIT) Charter (Attachment 2 of the enclosed report) provides the basis and additional details concerning the scope of the inspection.

The special inspection team (the team) examined activities conducted under your license as they relate to safety and compliance with Commission rules and regulations and with conditions of your license. The team reviewed selected procedures and records, observed activities, conducted in-plant equipment inspections, and interviewed personnel. In particular, the team reviewed event evaluations (including technical analyses), causal investigations, relevant performance history, and extent-of-condition to assess the significance and potential consequences of issues related to the February 18 event.

The team concluded that, overall, station personnel maintained plant safety in response to the reactor trips. Nonetheless, the team identified several issues related to equipment performance and human performance which complicated the event. The enclosed chronology (Attachment 3 of the enclosed report) provides additional details on the sequence of events and event complications.

This report documents one self-revealing finding that, using the reactor safety Significance Determination Process (SDP), has preliminarily been determined to be White, a finding with low to moderate safety significance. The finding is associated with the failure to perform appropriate maintenance activities to ensure 2B EDG reliability. Specifically, safety related time delay relays in the EDG low lube oil pressure trip circuit were used beyond the manufacturer recommended service life, without an associated test or monitoring program to demonstrate their continued reliability. Consequently, when called upon to reenergize the 24 4 kV safety bus, the time delay relay failed and the 2B EDG prematurely tripped in response to a low lube oil pressure signal. The 24 4 kV safety bus was reenergized from an alternate feed source approximately 30 minutes into the event. The significance determination of the event was performed assuming that similar time-delay relays on other systems have not failed due to this performance deficiency. Subsequent corrective actions included replacing and retesting the associated time delay relays on all three EDGs susceptible to the low lube oil pressure trip. There is no current immediate safety concern due to this finding, because all EDGs have subsequently been demonstrated operable and long term corrective actions are being implemented through the Calvert Cliffs corrective action program to address the extent-of-condition and extent-of-cause. The final resolution of this finding will be conveyed in a separate correspondence addressing the final risk significance and disposition of any violations.

As discussed in the attached inspection report, the finding is also an apparent violation (AV) of NRC requirements, involving Technical Specification 5.4.1, and is therefore being considered for escalated enforcement action in accordance with the Enforcement Policy, which can be found on NRC's Web site at http://www.nrc.gov/reading-rom/doc-collections/enforcement/.

In accordance with NRC Inspection Manual Chapter (IMC) 0609, we will complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of the date of this letter. The significance determination process encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination.

Before we make a final decision on this matter, we are providing you with an opportunity (1) to attend a Regulatory Conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance, or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of your response to this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of your receipt of this letter. If you decline to request a Regulatory Conference or submit a written response, you relinquish your right to appeal the final SDP determination, in that by not doing either, you fail to meet the appeal requirements stated in the Prerequisite and Limitation sections of Attachment 2 of IMC 0609. We request that if you decide to attend a Regulatory Conference or provide a written response, that you address the apparent violation, and that you also address the length of time that the 2B EDG was considered inoperable.

Please contact Glenn Dentel at (610) 337-5233 in writing within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

Because the NRC has not made a final determination in this matter, no Notice of Violation is being issued for these inspection findings at this time. In addition, please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

In addition, the report documents two NRC-identified findings and two self-revealing findings, each of very low safety significance (Green). Three of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspector at Calvert Cliffs Nuclear Power Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Senior Resident Inspector at Calvert Cliffs Nuclear Power Plant. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

David C. Lew, Director for
Division of Reactor Projects

Docket Nos.: 50-317, 50-318
License Nos.: DPR-53, DPR-69

Enclosure: Inspection Report 05000317/2010006 and 05000318/2010006
w/Attachments: Supplemental Information (Attachment 1)
Special Inspection Team Charter (Attachment 2)
Detailed Sequence of Events (Attachment 3)

cc w/encl: Distribution via ListServ

Enclosure: Inspection Report 05000317/2010006 and 05000318/2010006

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Sincerely,
/RA/
David C. Lew, Director
Division of Reactor Projects

Docket Nos.: 50-317, 50-318
License Nos.: DPR-53, DPR-69
Enclosure: Inspection Report 05000317/2010006 and 05000318/2010006
w/Attachments: Supplemental Information
Special Inspection Team Charter
Detailed Sequence of Events

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-317 50-318
License No.:	DPR-53, DPR-69
Report No.:	05000317/2010006, 05000318/2010006
Licensee:	Constellation Generation Company
Facility:	Calvert Cliffs Nuclear Power Plant (CC)
Location:	Lusby, Maryland
Dates:	February 22, through April 30, 2010
Team Leader:	D. Kern, Senior Resident Inspector, Division of Reactor Projects (DRP)
Team:	W. Cook, Senior Reactor Analyst, Division of Reactor Safety (DRS) M. Patel, Reactor Inspector, DRS P. Presby, Reactor Inspector, DRS B. Smith, Resident Inspector, DRP R. Montgomery, Reactor Engineer, Nuclear Safety Professional Development Program, DRP (added subsequent to issuance of the Inspection Charter)
Observers:	S. Gray, Power Plant Research Program Manager, Department of Natural Resources, State of Maryland M. Griffen, Nuclear Emergency Preparedness Coordinator, Department of the Environment, State of Maryland
Approved By:	Glenn T. Dentel, Chief Projects Branch 1 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000317/2010006 and 05000318/2010006; 02/22/2010 - 04/30/2010; Constellation Generation Company, Calvert Cliffs Nuclear Power Plant; Special Inspection for the February 18, 2010, Dual Unit Trip; Inspection Procedure 93812, Special Inspection.

A six-person NRC team, comprised of resident inspectors, regional inspectors, and a regional senior reactor analyst conducted this Special Inspection. The team was accompanied by two engineers from the State of Maryland, Department of Natural Resources and Department of the Environment. One apparent violation with potential for greater than Green safety significance and four Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (IMC) 0609, 'Significance Determination Process' (SDP); the crosscutting aspect was determined using IMC 0310, 'Components Within the Cross Cutting Areas;' and findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

NRC Identified and Self Revealing Findings

Cornerstone: Initiating Events

- **<u>Green</u>:** A self-revealing non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI "Corrective Actions," was identified, because auxiliary building roof leakage into the Unit 1 and Unit 2 45 foot switchgear rooms was identified on several occasions from 2002 to 2009, but was not thoroughly evaluated and corrective actions to this condition adverse to quality were untimely and ineffective. This degraded condition led to the failure of the auxiliary building to provide protection to several safety related systems from external events, a ground on a reactor coolant pump (RCP) bus, and ultimately a Unit 1 reactor trip. Immediate corrective actions included: repair of degraded areas of the roof; walk downs of other buildings within the protected area that could be susceptible to damage to electrical equipment due to water intrusion; issuance of standing orders to include guidance regarding prioritizing work orders due to roof leakage; and identifying further actions to take during periods of snow or rain to ensure plant equipment is not affected. Constellation entered the issue into their corrective action program (Condition Report (CR) 2010-001351). Long-term corrective actions include implementation of improved plant processes for categorization, prioritization and management of roofing issues.

  The finding is more than minor because it is associated with the protection against external factors attribute of the Initiating Events Cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The team determined the finding had a very low safety significance because, although it caused the reactor trip, it did not contribute to the likelihood that mitigation equipment or functions will not be available. The cause of the finding is related to the crosscutting area of Problem Identification and Resolution, Corrective Action Program aspect P.1(c) because Constellation did not thoroughly evaluate the problems related to the water intrusion into the auxiliary building such that the resolutions addressed the causes and extent-of-condition. This includes properly classifying, prioritizing, and evaluating the condition adverse to quality. (Section 2.1)

- **Green:** The team identified a finding for failure to translate the design calculations of phase overcurrent relays on 13 kV feeder breakers into the actual relay settings. The overcurrent relays protect the unit service transformer against faults in the primary or secondary side windings. The design specified limit of 1200 amps was determined based on the breaker rating of the feeder breakers. Constellation determined the as-found relay setting for the feeder breakers was 1440 amps which exceeded the rating of the feeder breakers. The team determined that due to the as-found relay setting, certain phase overcurrent conditions could potentially cause the breakers to fail prior to the phase overcurrent relay sensing the degraded condition. This condition could affect the recovery of the safety buses from the electrical grid. Constellation entered this issue into the corrective action program (condition report 2010-002123).

  The finding is more than minor because it affected the Initiating Events Cornerstone attribute of equipment performance for ensuring the availability and reliability of systems to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Also, this issue was similar to Example 3j of IMC 0612, Appendix E, "Examples of Minor Issues," because the condition resulted in reasonable doubt of the operability of the component, and additional analysis was necessary to verify operability. This finding was determined to be of very low safety significance because the design deficiency did not result in an actual loss of function based on Constellation's determination that the maximum load current possible would not challenge the feeder breaker ratings. Enforcement action does not apply because the performance deficiency did not involve a violation of a regulatory requirement. The finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency was not reflective of current licensee performance. (Section 2.3)

Cornerstone: Mitigating Systems

Preliminary White: The NRC identified an apparent violation of Technical Specification 5.4.1 for the failure of Constellation to establish, implement, and maintain preventive maintenance requirements associated with safety related relays. The team identified that Constellation did not implement a performance monitoring program specified by the licensee in Engineering Service Package (ES200100067) in lieu of a previously established (in 1987) 10-year service life replacement PM requirement for the 2B EDG T3A time delay relay. As a consequence, the 2B EDG failed to run following a demand start signal on February 18, 2010. Following identification of the failed T3A relay, it was replaced and the 2B EDG was satisfactorily tested and returned to service. In addition, time delay relays used in the 1B and 2A EDG protective circuits, that also exceeded the vendor recommended 10-year service life, were replaced. Constellation entered this issue, including the evaluation of extent-of-condition, into the corrective action program.

This finding is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely impacted the objective of ensuring the availability, reliability, and capability of the safety related 2B EDG to respond to a loss of normal electrical power to its associated safety bus. This finding was assessed using IMC 0609, Appendix A and preliminarily determined to be White (low to moderate safety significance) based upon a Phase 3 Risk Analysis with an exposure time of 323 days which resulted in a total (internal and external contributions) calculated conditional core damage frequency (CCDF) of 7.1E-6. The cause of this finding is related to the crosscutting area of Human Performance, Resources aspect H.2(a) because preventive maintenance procedures for the EDGs were not properly established and implemented to maintain long term plant safety by maintenance of design margins and minimization of long standing equipment issues. (Section 2.2)

- **Green:** The team identified a NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," because Constellation did not thoroughly evaluate and correct a degraded condition of a CO-8 relay disc sticking or binding issues which can adversely impact the function of the EDGs and the electrical distribution protection scheme. Specifically, following the February 18, 2010 event, Constellation did not identify and adequately evaluate the recent CO-8 relay failures due to sticking or binding of the induction discs in the safety related and non-safety related applications. Constellation entered this issue into the corrective action program (CR 20100004673).

  The finding is more than minor because it is associated with the equipment reliability attribute of the Mitigating Systems Cornerstone, and it adversely affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). This finding was determined to be of very low safety significance because these historical relay failures did not result in an actual loss of system safety function. The cause of the finding is related to the crosscutting area of Problem Identification and Resolution, Corrective Action Program aspect P.1(c) because Constellation did not thoroughly evaluate the previous station operating experience of CO-8 relay induction disc sticking and binding issues such that resolutions addressed the causes and extent-of-condition. (Section 2.3)

- **Green:** A self-revealing NCV of Technical Specification (TS) 5.4.1.a, "Procedures" was identified for failure to establish adequate procedures for restoration of Chemical and Volume Control System (CVCS) letdown flow. On February 18, 2010, an electrical ground fault caused a Unit 1 reactor trip, loss of the 500 kV Red Bus, and CVCS letdown isolation as expected on the ensuing instrument bus 1Y10 electrical transient. Deficient operating instructions prevented timely restoration of letdown flow following the initial transient. Pressurizer level remained above the range specified in Emergency Operating Procedure (EOP)-1 for an extended period because of the operators' inability to restore letdown. This ultimately led to exceeding the TS high limit for pressurizer level. CVCS Operating Instruction OI-2A was subsequently revised, providing necessary guidance for re-opening the letdown system excess flow check valve to restore letdown flow. This event was entered into the licensee's corrective action program (CR 2010-001378).

  The finding is more than minor because it is associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding is of very low safety significance because it is not a design or qualification deficiency, did not represent a loss of a safety function of a system or a single train greater than its TS allowed outage time, and did not screen as potentially risk significant due to external events. This finding has a crosscutting aspect in the area of human performance, resources aspect H.2(c), because Constellation did not ensure that procedures for restoring CVCS letdown were complete and accurate. (Section 3.1)

REPORT DETAILS

1. Background and Description of Events

In accordance with the Special Inspection Team (SIT) charter (Attachment 2), team members (the team) conducted a detailed review of the February 18, 2010, dual unit trip with complications at Calvert Cliffs Nuclear Power Plant including equipment and operator response. The team gathered information from the plant process computer (PPC) alarm printouts, interviewed station personnel, performed physical walkdowns of plant equipment, and reviewed procedures, maintenance records, and various technical documents to develop a detailed timeline of the event (Attachment 3). The following represents an abbreviated summary of the significant automatic plant and operator responses which began at 8:24 a.m. on February 18, 2010, and ended on February 22, 2010, with both Unit 1 and Unit 2 in cold shutdown:

On February 18, 2010, at 8:24 a.m., the Unit 1 reactor automatically tripped from 93 percent reactor power in response to a reactor coolant system (RCS) low flow condition. Water had leaked through the auxiliary building roof into the 45' elevation switchgear room, causing an electrical ground on bus 14 which tripped the 12B reactor coolant pump (RCP), thereby initiating the reactor protection system trip on RCS low flow. Three of the four Unit 1 RCPs continued operating.

Ground overcurrent (O/C) relay 2RY251G/B-22-2 failed to actuate as designed, permitting the Unit 1 ground O/C condition to reach the Unit 2 22 13 kV RCP bus and the associated 500 kV/13 kV transformer (P-13000-2). Ground O/C protection for the P-13000-2 transformer actuated which deenergized the 500 kV "Red Bus" offsite power supply, the 22 bus, and all four RCPs. At 8:24 a.m., the Unit 2 reactor automatically tripped from full reactor power in response to the associated reactor protection system trip on RCS low flow.

The P-13000-2 isolation also deenergized the 21 13 kV service bus, which deenergized the Unit 1 14 4 kV safety bus, the Unit 2 24 4 kV safety bus, and several Unit 2 non-safety related 4 kV busses. The 1B emergency diesel generator (EDG) started as designed and reenergized the Unit 1 14 bus. The 2B EDG started, but tripped 15 seconds later due to a low lube oil pressure signal and the 24 bus remained deenergized. The electrical transient deenergized 120 volt instrument buses 1Y10 and 2Y10, which isolated the chemical volume control system (CVCS) and RCS letdown for both units and complicated operators' control of pressurizer level.

Loss of power to the Unit 2 non-safety related buses resulted in loss of the normal RCS heat removal path (main feedwater pumps, circulating water pumps, and condenser). Operators used the turbine driven auxiliary feedwater pump and atmospheric steam dump valves for decay heat removal.

At 8:48 a.m., Unit 2 operators exited emergency operating procedure (EOP)-0, "Reactor Trip" and entered EOP-2, "Loss of Flow and Loss of Offsite Power." At 8:57 a.m., operators reenergized the 24 bus via the alternate feeder breaker. At 9:00 a.m., Unit 2 operators restored RCS letdown and maintained appropriate pressurizer level control.

At 11:17 a.m., Unit 2 operators started the 23 motor driven auxiliary feedwater (AFW) pump and secured the turbine driven AFW pump. At 11:18 a.m., Unit 2 operators exited the EOPs and returned to normal operating procedures. As of 12:02 p.m., Unit 1 operators remained unsuccessful at restoring RCS letdown and exceeded the pressurizer high level limits specified by both EOPs and TS. At 1:09 p.m., Unit 1 operators restored RCS letdown and restored normal pressurizer level control. At 1:38 p.m., Unit 1 operators exited the EOPs and returned to normal operating procedures.

At 2:07 p.m., Unit 1 vital 4 kV bus 14 was aligned to its alternate offsite source and the 1B EDG was secured. At 5:13 p.m., Unit 2 operators started 21B and 22A RCPs to restore forced RCS circulation. On February 19, 2010, at 12:05 p.m., operators verified two offsite power supplies were available, with the 21 13 kV service bus energized from an alternate offsite source. On February 20, 2010, at 10:31 p.m. repairs on the 2B EDG were completed and the diesel generator was declared operable.

Unit 1 achieved cold shutdown at 5:38 a.m. on February 21, 2010, and 500 kV Red Bus was restored at 5:50 a.m. Unit 2 achieved cold shutdown at 5:00 a.m. on February 22, 2010.

2. Equipment Performance

2.1 Untimely Corrective Actions to Unit 1 45 Foot Elevation Switchgear Room Roof Leak Caused Reactor Trip

a. Inspection Scope

Water leakage through the Unit 1 auxiliary building roof into the 45' elevation switchgear room, caused an electrical ground on Bus 14 which tripped the 12B RCP, thereby initiating a reactor protection system trip on RCS low flow. The team interviewed station personnel, performed field walkdowns, and reviewed various records including maintenance backlogs, maintenance history, operating logs, condition reports, and maintenance rule program records to independently determine the cause of the event and assess associated corrective actions. Constellation determined the root cause of the event was that Calvert Cliffs lacked sensitivity to the consequences associated with degraded roof conditions which led to a reactive rather than preventive strategy for dealing with roof leaks. The team independently reviewed Constellation's Root Cause Analysis Report (RCAR) for the Unit 1 reactor trip to determine the adequacy of the evaluation, the extent-of-condition review, and associated corrective actions.

b. Findings

Introduction: A self-revealing non-cited violation (NCV) of very low safety significance associated with 10 CFR Part 50, Appendix B, Criterion XVI "Corrective Actions," was identified because Constellation did not promptly identify and correct degraded conditions associated with the Unit 1 auxiliary building (45-foot elevation switchgear room) roof leakage. These degraded conditions led to the failure of the auxiliary building to provide adequate protection to numerous safety related systems from external events (adverse weather conditions) resulting in a ground on a reactor coolant pump (RCP) bus and a consequential Unit 1 reactor trip on February 18, 2010.

Description: On February 18, 2010, Unit 1 tripped due to water from a roof leak entering into the Unit 1 45-foot elevation switchgear (SWGR) room and causing a phase to ground short near a current transformer (CT) for the 12B RCP bus 14P differential/ground current protection devices. The ground fault was not isolated close to the source, due to a failed ground protection relay in the feeder breaker to the Unit 1 RCP bus. The consequential trip of the 12B RCP led to the Unit 1 reactor protection system (RPS) trip due to the a low reactor coolant system (RCS) flow signal.

While conducting a review of the dual unit trip, the team noted that in July of 2008, condition report (CR) IRE-032-766 was written regarding rain water which had fallen onto and into the emergency shutdown panel (ESDP) 1C43, which is located in the Unit 1 45' elevation SWGR room. Immediate actions were taken to notify the control room supervisor of the condition as well as to clean up the pooled water around the panel. Corrective actions were initiated to establish a program to maintain weather tight building integrity. In June of 2009, CR 2009-004060 documented water dripping inside the SWGR room just east of the No. 12 motor generator set. No immediate actions were taken; however, recommended actions were to repair the roof. On August 8, 2009, a third CR (CR 2009-005508) was written, again regarding water leaking into the SWGR room and onto the ESDP. Immediate actions were taken to cover the panel with herculite and to direct the leaking water into a plastic bucket, as well as mopping up the standing water. Despite the immediate actions taken to address the three rain water issues, no additional actions were taken to properly prioritize, identify, and correct the roof leakage. This is evident due to the fact that each CR was given the lowest priority (category 4) as well as none of the work orders written to address the roof leakage had been approved. Additional safety related SWGR equipment in the SWGR room included power supply breakers for the "B" train auxiliary feed water pump, high pressure safety injection pump, low pressure safety injection pump and EDG.

Based on the review of the RCAR, the team noted several missed opportunities from 2002 to 2009 to identify and evaluate the degraded condition prior to the dual unit trip. During a periodic bus inspection in 2004, repairs were made to insulating material on the power cables inside the 14P01 cubicle to correct a water spot on the "B" phase of the 12B RCP bus. This cubicle is in the same SWGR enclosure as the 14P02 cubicle where the water intrusion occurred that resulted in the February 18, 2010 trip. The work was completed under the bus inspection work order; however, no CR was written documenting the indicated water intrusion. This preventive maintenance activity should have led to an investigation into the cause of the water intrusion as well as the extent of the degraded condition. An apparent cause (IRE-007-705) was also completed in 2005 in response to a CR written by quality assurance personnel noting that there were 33 leaks identified during a walk down but no trend CR was written. Corrective actions were proposed; however they were not adequately implemented.

The Calvert Cliffs' maintenance rule scoping document states that the function of the auxiliary building is to provide structural support and separation to safety and non-safety related equipment while accounting for the effects of certain external events. Rain storms and heavy snowfall are examples of external events for which the auxiliary building is designed to provide protection against. The Calvert Cliffs' structure monitoring program did not effectively use the corrective action process to ensure this function of the auxiliary building would be maintained. At the time of this special inspection, 58 work orders were open to repair roof leaks. None of these work orders were planned or scheduled. Several of these work orders were over 2 years old.

Immediate corrective actions included: repairing degraded areas of the auxiliary building roof; performing walk downs of other protected area buildings that could be susceptible to damage to electrical equipment due to water intrusion; issuing standing orders to include guidance regarding prioritizing work orders due to roof leakage; and identifying further actions to take during periods of snow or rain to ensure plant equipment is not affected. Long-term corrective actions include implementing improved plant processes for categorization, prioritization, and management of degraded roof and water leakage issues.

The team concluded that Constellation had numerous opportunities to have thoroughly evaluated, classified, and prioritized the roof leakage, such that corrective actions could have addressed the full extent of the auxiliary building roofing degraded condition and prevented the water intrusion event and subsequent plant trip on February 18, 2010. The team concluded that station personnel did not properly inspect and maintain the roofs of several safety related structures to ensure the internal safety related and non-safety related components were protected from effects of the external environment (i.e., rain, snow).

Analysis: The failure of Constellation to promptly identify and correct conditions adverse to quality, associated with the auxiliary building roof leakage, is a performance deficiency. The finding is more than minor because it is associated with the Initiating Events Cornerstone and affects the cornerstone objective to limit the likelihood of those external events that upset plant stability and challenge critical safety functions during shutdown, as well as power operations. The inspectors evaluated this finding using IMC 0612 Attachment 4, "Phase 1- Initial Screening and Characterization of Findings." The team determined the finding to have very low safety significance because, although it contributed to a reactor trip, it did not contribute to the likelihood that mitigation equipment would not be available.

The cause of this finding is related to the Problem Identification and Resolution cross-cutting area, corrective action program, because Constellation did not thoroughly evaluate the problems related to the water intrusion into the auxiliary building such that the resolutions addressed the causes and extent-of-condition. This included properly classifying, prioritizing, and evaluating the condition adverse to quality (P.1(c)).

Enforcement: 10 CFR Part 50, Appendix B, Criterion XVI "Corrective Action," states, in part, that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. Contrary to the above, from 2002 to February 18, 2010, Constellation did not thoroughly evaluate and promptly correct degraded conditions associated with auxiliary building roof leakage. This led to the failure of the auxiliary building to provide protection to several safety related systems from external events (i.e. flooding), a ground on a reactor coolant pump bus, and ultimately a Unit 1 reactor trip. Because this violation was of very low safety significance and was entered into the licensee's corrective action program as CR 2010-001351, this violation is being treated as an NCV, consistent with the NRC Enforcement Policy." **(NCV 0500317/318/2010006-01: Failure to Thoroughly Evaluate and Correct Degraded Conditions Associated with Auxiliary Building Roof Leakage)**

2.2 Deficient Preventive Maintenance Program Procedures and Implementation for EDG Agastat Time Delay (TD) Relays

a. Inspection Scope

On February 18, 2010, Unit 2 experienced an automatic reactor trip, loss of the P-13000-2 Service Transformer, and loss of the 500 kV Red Switchyard Bus. The loss of the Red Bus resulted in loss of power to the No. 24 4 kV safety bus which caused an automatic start of the 2B EDG. The 2B EDG tripped due to low lube oil (LO) pressure after running for 15.2 seconds. The team reviewed the timing sequence, design requirements, relay schematics, and surveillance and maintenance history for the 2B EDG. Failure of a T3A time delay (TD) relay coincident with the 2B EDG LO low pressure protection logic not having reset caused the low LO pressure protective trip of the engine. Constellation identified two root causes for the EDG failure: (1) station personnel failed to recognize and quantify the low margin in all aspects of the low lube oil pressure trip set feature for the EDG; and, (2) station personnel did not rigorously assess all failure modes of the Agastat relays in the EDG protection circuitry prior to extending its service life beyond the vendor qualified life.

The team reviewed Constellation's evaluation of the 2B EDG's failure, the adequacy of proposed and completed corrective actions, and the appropriateness of the extent-of-condition review. Independent reviews of design documents, mock-up testing, drawings, surveillance testing, and field walk-downs were performed by the team to evaluate the cause of the 2B EDG failure. In addition, the team reviewed Constellation's preventive maintenance (PM) history and associated PM programs.

b. Findings

Introduction. The NRC identified an apparent violation of Technical Specification 5.4.1 for the failure of Constellation to establish, implement, and maintain preventive maintenance requirements associated with safety related relays. The team identified that Constellation did not implement a performance monitoring program in lieu of a previously established 10-year service life replacement PM requirement for the 2B EDG T3A TD relay. As a consequence, the 2B EDG failed to run following a demand start signal on February 18, 2010. This apparent violation is preliminarily determined to be of low-to-moderate safety significance (White).

Description. The purpose of the T3A (Agastat 7000 series) TD relay in the EDG protective circuit is to bypass the low lube oil trip on the EDG start to allow the EDG lube oil pressure to initially build up to operating conditions. The relay begins timing when the EDG speed reaches 810 rpm (approximately 6 seconds after EDG start). The relay functions to bypass the low LO pressure trip (<17 pounds pressure sensed in the EDG upper crankcase) for 15 seconds (a total of 21 seconds from EDG start). This time delay allows LO pressure to build-up in the EDG upper crankcase high enough to reset the trip logic (2 of 3 pressure switches reset at >20 pounds). The Unit 2 February 18, 2010, sequence of events printout revealed that the T3A relay timed out early (after 9.2 seconds) at 15.2 seconds following the EDG start and prior to the low LO pressure sensing trip logic being reset. Constellation determined that a typical fast, non-pre-lubricated EDG start results in LO pressure exceeding 20 pounds pressure approximately 13 seconds following the start of the EDG. Accordingly, the early timeout of the T3A relay was not the only degraded 2B EDG condition that presented itself on February 18, 2010. Constellation attributed the February 18 delayed reset of the pressure switches to "sticky lubrication oil" in the ½-inch stainless steel pressure sensing line to the pressure switches, vice an actual low LO pressure condition in the diesel engine upper crankcase.

The team determined that the T3A relay, which timed-out early, had been in-service on the 2B EDG for approximately 13.5 years, 3.5 years beyond its vendor recommended 10-year service life. In 2001, Constellation engineering discontinued the vendor recommended 10-year replacement PM and substituted a performance monitoring program envisioned to ensure Agastat relays (approximately 100 safety related applications and 500 to 600 non-safety related applications in the two Calvert Cliffs units) were appropriately monitored and replaced prior to failure (reference Engineering Service Package ESP No. ES200100067, approved 03/06/2001). The team identified that a relay performance monitoring program had not been established since 2001 at Calvert Cliffs. Constellation initiated CR 2010-04493 to address this performance issue. The Shift Manager reviewed the immediate operability and determined that the other safety-related components using Agastat relays remain operable because these relays are installed in less harsh operational environments (e.g. vibrations) then the EDG Agastat relays, and therefore, are less susceptible to age-related degradation. In addition, CR 2010-01784 was written to address the extent-of-condition of Agastat relays used in other safety-related applications.

Constellation replaced the 2B EDG failed T3A relay and, via a single 'as-found' bench test, validated its February 18, 2010, in-service failure, when the relay failed again, timing out early at 11.6 seconds. Subsequent attempts by Constellation to adjust the relay to within calibration tolerance were unsuccessful. The failed relay was shipped to an independent laboratory for diagnostic testing and destructive examination. The laboratory identified that, exercised over its full range of operation, >40 percent of the TD actuation results were out of tolerance. Internals examination identified three of six screws on the flexible diaphragm retaining ring were loose, suggesting that the early time-out of the relay was possibly due to excessive air bleed off (leakage passed the diaphragm seal). Constellation concluded that the TD relay failure was a relatively recent event (within the last 47 days) and attributable to the three 2B EDG starts and approximately seven cumulative hours of operation that occurred in early January 2010. The team concluded that Constellation provided no evidence to support the approximate time of failure of the TD relay. However, the team determined that the failure and probable failure mechanism may have occurred between the last successful calibration of the TD relay (May 13, 2008) and the observed failure on February 18, 2010. In addition, the team concluded that the TD relay early time-out was most likely a latent failure and masked by the monthly EDG surveillance test. Accordingly, the TD relay failure was revealed by the fast, non-pre-lubrication, demand start on February 18, 2010.

The basis for the team's conclusion was as follows:

- Constellation's troubleshooting results were not conclusive regarding the lubricating oil pressure sensing line "sticky oil" theory, based upon the following: 1) the "sticky oil" drained from the sensing line was not saved or analyzed for consistency or contaminants (Constellation did not exercise appropriate quarantine practices); 2) the ½-inch LO pressure sensing line was not backfilled with oil and was therefore susceptible to trapped air pockets that may tend to dampen accurate pressure sensing and may result in a delayed pressure response; and, 3) Constellation's routine (two-year calibration cycle) and post-event calibration checks of the pressure switches did not record "as-found" values of the pressure switch reset values; this information may have assisted in ruling out possible pressure switch setpoint drift or malfunction.

The team acknowledged that Constellation's subsequent mock-up testing of the pressure sensing line did show that lubricating oils of heavier viscosity tend to delay the pressure sensing response. However, the 100W oil used to demonstrate the phenomena (approximate 3 second pressure sensing delay) was considerably heavier than the lubricating oil used in the 2B EDG (40W) and may or may not have reflected the "sticky oil" viscosity observed by the technician responsible for the pressure switch troubleshooting.

- The fast, non-pre-lube start of the 2B EDG contributed to the identification of the failed relay; whereas the monthly pre-lube EDG starts likely masked the failure of the TD relay. The team determined that for a typical fast, pre-lubricated EDG start, a small pre-lube pump is run for 3 to 5 minutes prior to the EDG starting and fills the upper crankcase with lubricating oil, but is not of sufficient capacity to pressurize the upper crankcase. When the EDG starts, the engine driven LO pump functions to complete the upper crankcase fill and pressurization (>20 pounds pressure) in approximately 8 seconds. Accordingly, any relay failure (timing out early, <12 seconds) is masked by the fast, pre-lube EDG start because the relay actuates at 6 seconds and only has to satisfactorily function (block the low lube oil trip signal) for >2 seconds. The team noted that by the low LO pressure protective system design, the fast pre-lube EDG starts allow for a significant margin to satisfactory build-up of lube oil pressure before the TD relay times out (a margin of approximately 13 seconds). For the fast non-pre-lube start, LO pressure typically exceeds 20 pounds pressure at 13 seconds after EDG start. This 13 second time interval similarly translates to the TD relay having to function for >7 seconds from the time it actuates at 6 seconds from EDG start. This 7 seconds minimal TD function also, by design, provides margin (an additional 8 seconds) for satisfactory LO pressure build-up.

The team concluded that the last known satisfactory relay calibration (setpoint) check of the T3A relay was the two-year calibration check completed on May 13, 2008. Based upon Constellation records, the as-found setting was 17.5 seconds and the as-left was 16.5 seconds. All monthly surveillance tests of the 2B EDG since May 13, 2008, were fast, pre-lube starts. There were no demand starts of the 2B EDG between May 13, 2008, and February 18, 2010, that would have proved or disproved that the T3A relay was operable, and that the LO pressure sensing line issue was coincidental or precipitous of a fast, non-pre-lube start.

Following identification of the failed T3A relay, the licensee replaced the relay, satisfactorily tested the 2B EDG, and returned the 2B EDG to service. In addition, time delay relays used in the 1B and 2A EDG protective circuits, that also exceeded the vendor recommended 10-year service life, were replaced. Constellation is evaluating the continued use of Agastat relays beyond their vendor recommended 10-yr service life. As previously noted, there are approximately 100 safety related applications and 500-600 non-safety related applications at the two Calvert Cliffs units.

Analysis. The team identified that the failure of Constellation to perform preventive maintenance in accordance with vendor recommendations without adequate performance monitoring on safety related Agastat 7000 series TD relays used in safety related applications is a performance deficiency and violation of Technical Specifications (TS). This violation of TS is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely impacted the objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the early timeout of the T3A relay caused the 2B EDG to trip prior to the low lube oil pressure trip signal clearing (resetting) after a demand fast start on February 18, 2010. The failure of the 2B EDG to run resulted in the continued loss of alternating current to the No. 24 4 kV safeguards bus and its associated emergency core cooling systems.

In accordance with Table 4a of IMC 0609, Attachment 04, "Phase 1 – Initial Screening and Characterization of Findings," this performance deficiency required a Phase 2 or 3 risk analysis because the issue resulted in an actual loss of safety function of a single train for greater than its TS allowed outage time. A Phase 3 risk assessment was performed by a Region I Senior Reactor Analyst (SRA) using the SAPHIRE software and Calvert Cliffs Unit 2 Standardized Plant Analysis Risk (SPAR) model, Revision 3.46, dated February 2010.

To conduct the Phase 3 analysis, the SRA made the following modeling assumptions:

- Exposure time was based upon a T/2 approximation. The team determined that the 2B EDG exposure time is best approximated by a T/2 value, per the usage rules of IMC 0308, Appendix A, "Technical Basis for At Power Significance Determination Process." Specifically, if the inception of a condition is unknown, the use of the mean exposure time (T/2) is a statistically valid time period because it represents one-half of the time since the last successful demonstration of the component's function and the time of discovery or known failure. The last successful demonstration of the T3A relay was the calibration check performed on May 13, 2008. The total time (T) between May 13, 2008 and February 18, 2010 is 646 days. Therefore, T/2 represents an approximate exposure time of 323 days or 7752 hours.

- SPAR model basic event EPS-DGN-FS-2B, representing "Diesel Generator 2B Failure to Start" was set to TRUE. The basis for the TRUE, vice a failure probability of 1.0, is that common cause failure of the remaining Fairbanks-Morris EDGs could not be conclusively ruled out. The same type Agastat 7000 series TD relays, with comparable greater than 10 years in-service times were installed on the 1B and 2A EDGs.

- SPAR model basic event AFW-XHE-XM-FC8, representing operator failure to open the Turbine Building to turbine driven auxiliary feedwater (TDAFW) pump room door within 12 hours of a station blackout event, was set to FALSE. The basis for this change is that recent engineering analysis of the TDAFW pump room heat-up (post Appendix R fire, LOOP/LOCA, SBO) identified no dependency on operator action to open the door to the turbine building to ensure adequate cooling of the TDAFW pumps.

- No additional 2B EDG recovery credit was applied to the model based upon this event. The SRA noted that 2B EDG non-recovery probability (0.772) in the SPAR model is based upon industry statistical data. The SRA notes that Constellation procedures have operators align the OC EDG (within 45 minutes) vice attempt to troubleshoot and restart the failed EDG. Accordingly, any subsequent attempts to restart the 2B EDG, after an approximate one hour delay (aligning the OC EDG) would likely have the same result because all LO would have drained from the upper crankcase.

- Even though Agastat 7000 series relays are used in multiple safety related applications (some beyond their vendor recommended service life), no broad-based increase in safety related systems' or components' failure probabilities was applied for this Phase 3 risk assessment. As a consequence, the calculated risk estimate for this condition may be a non-conservative value because the Agastat relays are used in multiple other safety related applications beyond the manufacturer's recommended 10-year service life.

- Truncation for the SPAR model analysis was set at 1E-13.

Using the above stated assumptions, the increase in internal risk (core damage frequency) associated with the 2B EDG failure of February 18, 2010, was estimated at 6.0E-6. The dominant core damage sequence involves the loss of Facility B (13 kV Service Bus No. 21), loss of steam generator cooling (main feedwater and auxiliary feedwater), and the subsequent loss of once through cooling (feed and bleed, using the charging system and a power operated relief valve).

Based upon the absence of an NRC external risk quantification tool, the SRA used Constellation's calculated external risk values to approximate the external risk contribution. Constellation's estimated external risk is based upon a RISKMAN fire modeling tool and was calculated at 1.1E-6 for the T/2 exposure period. No appreciable external risk contributions were identified for flooding or seismic events. The dominant core damage external events include turbine building fires (involving the steam generator main feedwater pump area) and high wind/hurricane events. The dominant turbine building fire scenarios involve the failure of the available EDGs (2B and 1B) and a spurious initiation of the safety feature actuation system (SFAS). The dominant high wind/hurricane event core damage scenarios involve the assumed failure of the OC EDG, the subsequent failure of the remaining safety related EDGs, and a spurious SFAS.

Based upon the SRA's calculated internal events risk estimate and Constellation's estimated external events risk contribution, the total increase in Unit 2 core damage frequency for this finding is approximately 7.1E-6. Accordingly, this finding is of low to moderated safety significance (WHITE). This finding and the associated risk analysis was reviewed by a Significance and Enforcement Review Panel (SERP) conducted on June 1, 2010. The SERP concluded that the stated Technical Specification violation and associated risk characterization were appropriate. The violation does not represent an immediate safety concern because the licensee took prompt corrective actions to replace the Agastat relays in use beyond their service life for all three Fairbanks-Morris EDGs and ensured the LO pressure sensing lines were properly backfilled. Subsequent testing of all three EDGS verified operability, including a non-pre-lubricated fast start of the 2B EDG.

The Constellation PRA staff performed a risk assessment of the 2B EDG failure using their CAFTA internal events model and RISKMAN external events model. Constellation assumed the same exposure time as the Region I SRA of T/2 equal to 323 days. Constellation's total risk estimate was 3.1E-6 CDF. Based upon discussions with the Constellation PRA staff, their risk estimate and dominant core damage sequences compare favorably with the NRC results.

The cause of this finding is related to the crosscutting area of Human Performance, resources aspect because preventive maintenance procedures for the EDGs were not properly established and implemented to maintain long term plant safety by maintenance of design margins and minimization of long standing equipment issues (H.2(a)).

Enforcement. Technical Specification 5.4.1 states, in part, that written procedures specified in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, shall be established, implemented, and maintained. Section 9.b. of Appendix A to Regulatory Guide 1.33 states, in part, that preventive maintenance schedules should be developed to specify replacement of parts that have a specific service life. In March 2001 Constellation replaced their original 10-year relay replacement preventive maintenance with a proposed performance monitoring program, to ensure the continued reliability and operability of Agastat relays installed in safety related applications beyond the vendor recommended 10-year service life, via Engineering Change Package No. ES200100067.

Contrary to the above, the team identified that Constellation did not establish a performance monitoring program, and all Agastat relays installed in safety related applications at Calvert Cliffs have been subject to "run to failure" preventive maintenance/replacement interval. Constellation took prompt corrective action to replace Agastat relays used in service, beyond their 10-year service life, in the 2B, 2A and 1B EDGs. The remaining Agastat relays, used in safety related applications beyond their vendor recommended service life, are under evaluation by Constellation. Constellation has initiated several CRs (see Attachment 1 to this report) associated with this performance deficiency. Pending final significance determination, the finding is identified as **Apparent Violation (AV) 05000318/2010006-02, Inadequate Preventive Maintenance Results in the Failure of the 2B Emergency Diesel Generator**.

2.3 Ground Fault Relay 251G/B-22-2 Did Not Actuate on Ground Overcurrent to Trip Open Breaker 252-2202

a. Inspection Scope

The team reviewed design requirements, drawings, and maintenance history of the 251G/B-22-2 relay. Failure of this relay to actuate and trip open the 252-2202 breaker resulted in a loss of the P-13000-2 service transformer, which resulted in loss of power to the Unit 2 RCPs and a Unit 2 trip with loss of normal decay heat removal. Unit 2 remained on atmospheric dump valves and auxiliary feedwater for heat removal for approximately 68 hours. Constellation determined the most likely cause of the relay failure was premature coil aging due to the operating environment and the magnitude of the current seen, which caused insulation breakdown and shorting of the magnetizing coil. Even though Constellation could not conclusively identify the cause of the insulation breakdown and magnitude of the signal that coincided with the breakdown, they did note that the relay in this particular application is located in non-environmentally controlled space which would impact aging mechanisms due to the temperature extremes. Additionally, the 251G/B-22-2 relay age was 39 years at the time of the event, which is only 1 year within the 40-60 year service life.

The team reviewed Constellation's root cause analysis report (RCAR) for the 251G/B-22-2 relay to determine the adequacy of the evaluation and the appropriateness of the extent-of-condition review. Independent reviews of the design documentation, drawings, maintenance history, and field walk-downs were performed to validate the cause of the relay failure. The team reviewed the design requirement and the relay setting information of the 13.8 kV fault protection relaying scheme to ensure proper equipment protection during transient and steady state conditions. The team also reviewed the history of the 251G/B-22-2 relay, along with other protective relays in the 13.8 kV system that were required during the event, to verify that the applicable test acceptance criteria and maintenance frequency requirements were met.

b. Findings

Deficient Evaluation and Untimely Corrective Action Associated with Induction Disc Binding on CO-8 Type Relays

Introduction: The team identified a finding of very low safety significance (Green) that involved a NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," because Constellation did not thoroughly evaluate and correct a degraded condition of CO-8 relay disc sticking or binding issues which can adversely impact the function of the EDGs and the electrical distribution protection scheme. Specifically, following the February 18, 2010, event Constellation did not identify and adequately evaluate the recent CO-8 relay failures due to sticking or binding of the induction discs in the safety related and non-safety related applications.

Description: The team reviewed Constellation's RCAR for the relay 2RY251G/B-22-2 on breaker 2BKR252-2202 which failed to trip open the breaker. The relay was a CO-8 ground fault over-current relay which had been in service for the life of the plant. The relay consists of an electromagnet and an induction disc which rotates to close a moving contact to a stationary contact to complete the breaker trip circuitry. The root cause analysis concluded that the magnetizing coil had shorted out the majority of the windings in a manner that current would pass but the induction disc would not rotate.

The team reviewed Constellation's maintenance and corrective action history of the CO-8 relay failures and noted that the induction disc type relays had a failure history associated with disc binding and sticking conditions. The team also noted that CO-8 relays and other induction disc type relays had a high failure rate for out of tolerance conditions during the performance of relays calibration procedures. The team determined that failures of the relay due to binding, sticking, and out of tolerance conditions can potentially impact the breaker trip operation and affect breaker coordination.

The failure history for binding, sticking, and out of tolerance conditions for the induction type relays were reviewed since 2007. The team found 40 failures since 2007 and 5 failures of the CO-8 type relays. Constellation has a total of 68 CO-8 type relays installed in safety related and non-safety related applications, all of which have been scheduled to be calibrated every 2 years since 2005. The team noted that from 1999 to

2005 as-found testing and calibration of the relays were performed every 4 years. The team reviewed the failure data of the CO-8 and other induction disc type relays prior to 2005 and concluded that the failure rate did not change significantly subsequent to the increase in calibration frequency. The CO-8 relay failures were noted to be 10 percent from 1999-2005.

Constellation replaced or cleaned the relays with sticking or binding conditions; however, the licensee did not place the relays in any system or component monitoring program. The relays were also not part of the system health tracking report. The team reviewed the historical failures of the CO-8 relays and noted that for some of the testing conditions, the induction disc needed to be mechanically agitated to free it from the binding or sticking conditions. The team reviewed the vendor and Electric Power Research Institute (EPRI) calibration and maintenance manual and determined that Constellations' calibration and inspection procedure did not include all of the recommended practices specified in the EPRI guideline related to inspection and cleaning of the induction disc units. Constellation entered this issue into the corrective action program (CRs 2010-004672 and 2010-004673).

Analysis: The team reviewed Constellation's root cause evaluation, which concluded the cause of the relay failure to be premature coil aging due to its operating environment and the magnitude of the current seen by the relay. The team concluded that there was no direct correlation between the coil failure and the historical binding and sticking conditions of the CO-8 relay discs. However the team determined that Constellation's failure histories of the CO-8 type relays were significant and the failure to evaluate the degraded conditions and implement timely and effective action to correct this condition adverse to quality was a performance deficiency. The CO-8 relays are used in multiple safety related and non-safety related applications.

The finding was more than minor, in accordance with NRC IMC 0612, Appendix B, "Issue Screening," (IMC 0612B) because, while it was not similar to any examples in IMC 0612, Appendix E, "Examples of Minor Issues" (IMC 0612E), it was associated with the equipment reliability attribute of the Mitigating Cornerstone and it adversely affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The team evaluated this finding using IMC 0612 Attachment 4, "Phase 1-Initial Screening and Characterization of Findings." The finding is of very low safety significance (Green) because it is not a design or qualification deficiency, did not represent a loss of a safety function of a system or a single train greater than its TS allowed outage time, and did not screen as potentially risk significant due to external events. The historical relay failures did not result in an actual loss of system safety function.

The cause of the finding is related to the crosscutting area of Problem Identification and Resolution, Corrective Action Program because Constellation did not thoroughly evaluate the previous station operating experience of CO-8 relay induction disc sticking and binding issues such that resolutions addressed the causes and extent-of-condition (P.1(c)).

Enforcement: 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, Constellation did not adequately evaluate and correct the degraded condition of CO-8 relays which can potentially impact the function of multiple safety related systems or component. Because the finding was of very low safety significance and has been entered into Constellation's corrective action program (CR 2010-004673), this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: **NCV 05000317 & 318/2010006-03, Failure to Evaluate Degraded Conditions Associated With CO-8 Relays and Implement Timely and Effective Action to Correct the Condition Adverse to Quality.**

Deficient Offsite Power Distribution Tripping Scheme Design Control

Introduction: The team identified a finding having very low safety significance (Green) for failure to translate design calculation setpoint standard listed in calculation E-90-058 and E-90-061 of phase overcurrent relay (250) on feeder breakers 252-1101, 1102, 1103, 2101, 2102, and 2103 into the actual relay settings.

Description: During the relay settings review, the team identified that the service transformer 251G/ST-2 and service bus 251G/SB-21 ground overcurrent relays settings specified in the relay setting sheets did not support the values listed in the relay setting calculation E-90-61 for the 500/14 kV Service Transformer (P-13000-2). The value listed in the calculations for the 251G/ST-2 ground overcurrent relay tap settings was 2.5 amps and the actual field setting, which is set in accordance with the relay setting sheets, was found to be at 2 amps. For the service bus 251G/SB-21 the calculation setting of the time delay value was 4 seconds and the actual field settings was found to be at 3 seconds. Due to these discrepancies Constellation's engineering staff conducted an evaluation to determine if the actual field settings as specified in the relay setting sheets for the two overcurrent relays provided adequate coordination to ensure selective tripping. The relays are designed to detect ground faults on the 13.8 kV system which have not been cleared by the 500 kV transmission system relays and separate the station service transformer P-13000-2 from the grid. The team reviewed Constellation's evaluation and determined that there was no selective tripping coordination impact due to the relay setting discrepancies on 251G/ST-2 and 251G/SB-21. However, due to these discrepancies identified between the relay setting sheets and the design calculations, Constellation conducted an extent-of-condition review for the 13.8 kV systems to determine if other similar relay settings discrepancies exist.

As a result of the extent-of-condition review, Constellation identified that the phase overcurrent relay (250) pickup value for the six unit service transformers feeder breakers 252-1101, 1102, 1103, 2101, 2102, and 2103 were set at 1440 amps in accordance with the relay setting sheets and the values specified in the calculations E-90-058 and E-90-061 were 1200 amps.

The normal system operation design when offsite power is available, is the 4.16 kV system being supplied by the 13.8 kV system through six unit service transformers. The unit service transformers have overcurrent protection to protect against transformer faults in the primary or secondary side windings. This overcurrent protection per calculations E-90-058 and E-90-061 was limited to be at 1200 amps due to the breaker rating of all of the feeder breakers. Due to the as found relay setting of 1440 amps exceeding the breaker ratings of 1200 amps, Constellation conducted an operability analysis and performed a calculation which determined that the maximum load current possible during the worst case electrical distribution line-up condition would be 982 amps. The calculation demonstrated that the maximum load current possible during the worst case electrical distribution line-up would not challenge the feeder breaker ratings, and therefore would not cause the breaker to fail prior to the trip operation (tripping).

Analysis: The team determined that the failure to translate the design calculation setpoint standard values listed in the calculation E-90-058 and E-90-061 of phase overcurrent relay (250) on feeder breakers 252-1101, 1102, 1103, 2101, 2102, and 2103 into the actual relay settings was a performance deficiency.

The team determined that this finding was more than minor because it affected the Initiating Events Cornerstone attribute of equipment performance for ensuring the availability and reliability of systems to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Also, this issue was similar to Example 3j of IMC 0612, Appendix E, "Examples of Minor Issues," because the condition resulted in reasonable doubt of the operability of the component, and additional analysis was necessary to verify operability. The failure to translate adequate design calculation setpoint of phase overcurrent relays on the feeder breakers resulted in an as-found relay setting that exceeded the rating of the feeder breakers. The team determined that due to the as-found relay setting exceeding the breaker ratings, certain phase overcurrent conditions could have potentially caused the breaker to fail prior to the phase overcurrent relay sensing the degraded condition. The team determined that this condition could affect the recovery of the safety buses from the electrical grid. The team evaluated this finding using IMC 0612 Attachment 4, "Phase 1- Initial Screening and Characterization of Findings." This finding was determined to be of very low safety significance (Green) because these inadequate relay settings did not result in an actual loss of system safety function and Constellation also performed an evaluation and determined that the maximum load current possible would not challenge the feeder breaker ratings. The finding did not have a cross-cutting aspect because the most significant contributor to the performance deficiency was not reflective of current licensee performance.

Enforcement: This finding was not a violation of regulatory requirements because the unit service transformers and the overcurrent protection relays are not a system or component covered under 10 CFR Part 50, Appendix B. The issue has been entered into the licensee's corrective action program (CR 2010-002123. Because this finding does not involve a violation and has very low safety significance, it is identified as **FIN 05000317 & 318/2010006-04: Failure to Translate Design Calculation Setpoint of Phase Overcurrent Relay on Feeder Breakers.**

2.4 Breaker 2BKR152-2501 (4 kV Bus 25 Normal Feed) Failed to Trip Open

a. Inspection Scope

The team reviewed design requirements, drawings, and maintenance history of the 2BKR152-2501 breaker. The breaker inspection reviewed the maintenance practice and procedure of overhauling the 4 kV breakers to determine if adequate test acceptance criteria were established and followed vendor recommendations. The team reviewed Constellation's root cause analysis report for the 2BKR152-2501 to determine the adequacy of the evaluation and the appropriateness of the extent-of-condition review. Independent reviews of the design documentation, drawings, maintenance history, and field walkdowns were performed to validate the cause of the breaker failure.

Additionally, operations, maintenance, and engineering staff were interviewed to confirm the observations and causes cited in Constellation's evaluation of this issue. The team reviewed the adequacy of associated preventive maintenance, corrective actions, and post maintenance testing performed on the 2BKR152-2501 breaker. Bus 25 supplies power to three Unit 2 circulating water pumps.

b. Findings

No findings of significance were identified for this equipment issue. The team determined that this failure of 2BKR152-2501 to open had no adverse consequence during this event.

2.5 Breaker 2BKR252-2201 (13 kV Unit 2 RCP Buses Normal Feed) Failed to Trip Open

a. Inspection Scope

The team reviewed design requirements, drawings, and maintenance history of the 2BKR252-2201 breaker. The team reviewed the maintenance practice and procedure of overhauling the 13.8 kV breakers to determine if adequate test acceptance criteria were established and followed vendor recommendations. Constellation concluded the cause of the breaker failing to open was infant mortality (i.e., manufacturing defect). The team reviewed Constellation's root cause analysis report for the 2BKR252-2201 to determine the adequacy of the evaluation and the appropriateness of the extent-of-condition review. Independent reviews of the design documentation, drawings, maintenance history, and field walkdowns were performed to validate the cause of the breaker failure. Additionally, operations, maintenance, and engineering staff were interviewed to confirm the observations and causes cited in Constellation's evaluation of this issue. The team reviewed the adequacy of associated preventive maintenance, corrective actions, and post maintenance testing performed on the 2BRK252-2201 breaker.

b. Findings

No findings of significance were identified.

3. Human Performance

3.1 Event Diagnosis and Crew Performance

a. Inspection Scope

The team interviewed the operations crew that responded to the February 18, 2010, event, including three senior reactor operators, the shift manager, the control room supervisor, the shift technical advisor, two reactor operators, and three equipment operators to determine whether the operators performed in accordance with procedures and training. The team also reviewed narrative logs, post-transient reports, condition reports, PPC trend data, and procedures implemented by the crew.

b. Findings/Observations

Deficient Procedure Guidance for CVCS Letdown Restoration

Introduction: A self-revealing Green NCV of TS 5.4.1.a, "Procedures," was identified because Constellation did not establish adequate procedures for restoration of CVCS letdown flow. Deficient operating instructions prevented timely restoration of letdown flow following letdown isolation, which ultimately led to exceeding the TS high limit for pressurizer level.

Description: On February 18, 2010, Unit 1 was operating at 93% reactor power in preparation for main steam safety valve testing with the 11 and 13 charging pumps operating and increased letdown flow balanced with charging flow. At 8:24 a.m., a phase to ground overcurrent fault on 12B RCP switchgear resulted in an automatic reactor trip on Unit 1. Protective relaying isolated plant service transformer P-13000-2, which de-energized Unit 1 4 kV bus 14. Instrument Bus 1Y10, which is normally fed from 4 kV Bus 14, de-energized, isolating CVCS letdown by closing letdown isolation valve 1-CVC-515. The 1B EDG automatically started on bus undervoltage and re-powered 4 kV Bus 14 about 8 seconds later.

Charging pump 13 stopped on loss of power when 14 Bus de-energized and charging pump 11, powered from 4 kV Bus 11, continued running. At 8:31 a.m., operators re-started charging pump 13. Charging pumps remained running and pressurizer level increased as expected. Operators performed makeup to the CVCS Volume Control Tank (VCT) from 8:50 a.m. to 9:11 a.m. in order to maintain VCT inventory while the two running charging pumps transferred VCT contents into the pressurizer. At 8:58 a.m., 34 minutes after the reactor trip, and with pressurizer level approaching the high end of the EOP pressurizer level control band (180"), operators turned off charging pump 13. Charging pump 11 continued to run in anticipation of restoring letdown. At 9:02 a.m., operators stopped charging pump 11 because pressurizer level was above the EOP high level limit.

At 9:12 a.m., operators made their first attempt to restore letdown in accordance with OI-2A, "Chemical and Volume Control System", Section 6.7, "Starting Charging and Letdown" by re-starting charging pump 11 and shortly thereafter opening letdown isolation valves. They were not successful in restoring letdown. Subsequent post-event analysis of system parameter data stored on the plant computer indicated that excess flow check valve 1-CVC-343 was closed. Inadequate procedural guidance prevented operators from re-opening the check valve to establish letdown flow. The procedure for starting letdown consisted of setting letdown downstream control valves at 20% open in manual, starting a charging pump to cool the letdown stream, then opening letdown upstream isolations 1-CVC-515 and 1-CVC-516 to establish letdown flow. OI-2A did not contain any information related to the possibility that excess flow check valve 1-CVC-343 might be closed and did not provide direction for opening the valve.

Operators were confused by indicated letdown flow remaining downscale and took about 7 minutes re-confirming the system lineup and monitoring their instrumentation before stopping charging pump 11. They did not use OI-2A, Section 6.6, "Securing Charging and Letdown" to stop charging and letdown because letdown was not yet established. Initial conditions for using Section 6.6 were not met. Operators did not recognize a need for simultaneously stopping charging and letdown in accordance with the general methodology of Section 6.6. An additional 17 minutes elapsed from the time operators stopped the charging pump 11 until they closed the upstream letdown isolation valves.

Post-event data analysis showed the downstream letdown piping temperature steadily increased into the 400° to 500°F range during the 17 minutes between stopping the charging pump and closing the upstream letdown isolation valves because of hot reactor coolant flowing in the letdown line through the 10 gallons per minute (gpm) orifice which bypasses around the excess flow check valve. Typically, reactor coolant is cooled by charging flow through the letdown regenerative heat exchanger to about 220°F in the letdown line. It is postulated that during letdown restoration attempts, the RCS which was greater than 2000 psi pressure, re-pressurized the letdown line which rapidly collapsed steam voids in the hot (400°F-500°F) letdown piping and re-closed the excess flow check valve because of water hammer. A differential pressure was then established across the check valve, maintaining it closed. The restoration method provided by procedure OI-2A did not contain actions necessary for pressure equalization across this spring-loaded check valve.

During the second letdown restoration attempt at 10:44 a.m., letdown continued to flow through the bypass orifice for 21 minutes after stopping charging pump 11. This action again heated the letdown line to near reactor coolant temperature. On the third attempt at 11:39 a.m., operators closed letdown isolation valves just 2 minutes after stopping the charging pump, which left the letdown line in a relatively cool state, such that the transient conditions on the fourth and final attempt did not re-close the excess flow check valve. Operators made a total of four attempts to restore letdown over 5 hours before letdown was finally restored at 1:17 p.m.

Pressurizer level remained above the specified limit in EOP-1 for all but a few minutes of approximately 5 hours following the reactor trip. Throughout this period, operators attempted to control pressurizer level from the EOP high level limit of 180" to the normal full power level of 215". This range was based on the constraints of controlling pressurizer level below the TS high limit of 225" and high enough to prevent overfilling the VCT. With letdown unavailable, operators were only able to lower pressurizer level through the 6 gpm reactor coolant pump seal bleed off that returns to the VCT.

The team observed that unnecessarily conservative procedural requirements for ensuring adequate shutdown margin in NEOP-301, "Operator Surveillance Procedure" contributed to the operating crew's sense of urgency for letdown restoration. Operators recognized that the 2400 gallon RCS boration required to satisfy the requirements of NEOP-301 would cause pressurizer level to significantly exceed the TS high level limit if performed with letdown isolated.

Other options existed for controlling VCT level such that bleed off could be allowed to reduce pressurizer level to within the EOP band. These included intentionally draining the VCT to the liquid waste system and aligning bleed off flow to return to the reactor coolant drain tank instead of to the VCT. However, the station does not have an abnormal operating procedure for responding to a sustained loss of letdown and therefore no procedural guidance existed for using other methods to control VCT level.

Around noon, shortly after the third attempt to restore letdown, operators became involved in shifting main turbine gland sealing steam supply from main steam to auxiliary steam and failed to control RCS temperature. Loop temperature rose approximately 5°F, causing pressurizer level, already high at 215", to rise and peak at 231." Pressurizer level remained above the TS 3.4.9 high limit of 225" for approximately 7 minutes until operator actions which were taken to lower RCS temperature succeeded in reducing level to below the TS limit.

The excess flow check valve did not re-close on the fourth restoration attempt. Letdown was successfully re-established at 1317, approximately 5 hours after event initiation. Constellation has established procedure guidance relating to letdown restoration following closure of the excess flow check valve. The issue was entered into their CAP for further evaluation as CR 2010-001378.

Analysis: The performance deficiency is that Constellation did not establish adequate procedures for restoring letdown. Multiple factors contributed to pressurizer level exceeding the TS high limit. These included time pressure from overly conservative procedure requirements related to maintaining shutdown margin, filling the pressurizer above the EOP band when RCS temperature was below its nominal no-load value, makeup to the VCT to the high end of its control band when pressurizer level was already high, the absence of proceduralized options for controlling VCT level, and inattentiveness to reactor coolant temperature control. However, inadequate procedure guidance for letdown restoration is the primary reason which led to operation outside of EOP pressurizer level limits for an extended period of time and unnecessarily challenged operators in their attempts to maintain pressurizer level control.

The team determined this finding is more than minor because it is associated with the procedure quality attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding is of very low safety significance (Green) because it is not a design or qualification deficiency, did not represent a loss of a safety function of a system or a single train greater than its TS allowed outage time, and did not screen as potentially risk significant due to external events. This finding has a crosscutting aspect in the area of Human Performance, resources, because Constellation did not ensure that procedures for restoring CVCS letdown were complete and accurate (H.2(c)).

Enforcement: TS 5.4.1.a requires, in part, that written procedures be established, implemented, and maintained for activities described in Appendix A of Regulatory Guide (RG) 1.33, "Quality Assurance Program Requirements (Operation)." Specifically, Section 3 of RG 1.33, Appendix A, "Instructions for energizing, filling, venting, draining, startup, shutdown, and changing modes of operation should be prepared, as appropriate, for the following systems," includes the Letdown/Purification System. Contrary to the above, on February 18, 2010, the operators were unable to restore charging and letdown using the existing instructions of OI-2A, "Chemical and Volume Control System," due to inadequacy of the procedure. Because this issue is of very low safety significance (Green) and Constellation entered this issue into their corrective action program as CR 2010-001378, this finding is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000317/318/2010006-05, Failed to Establish Adequate Procedures for Letdown Restoration).**

3.2 Communications and Emergency Plan Applicability

a. Inspection Scope

This event involved an automatic reactor trip of both units with multiple complicating degraded equipment issues. Each unit lost one 500 kV offsite power supply (the Red Bus). In addition, Unit 2 lost forced RCS circulation when all four RCPs tripped, the 2B EDG failed to reenergize the Unit 2 24 4 kVafety bus, and the Unit 2 normal heat removal sink (main condenser) was unavailable for an extended time. Operators notified the NRC of the event at 11:47 a.m. on February 18 in accordance with 10 CFR 50.72. Operators determined that emergency action level (EAL) entry criteria were not met and accordingly did not declare an emergency event. The team reviewed operator logs, emergency procedures, the Emergency Plan, plant operating data, and interviewed station personnel to verify operators properly assessed the EAL entry criteria and notified the NRC of the event.

b. <u>Findings</u>

No findings of significance were identified.

4. <u>Organizational Response</u>

4.1 <u>Immediate Response and Restart Readiness Assessment</u>

a. <u>Inspection Scope</u>

The team interviewed personnel, reviewed various procedures and records, observed plant operators and station meetings, and performed plant walkdowns to assess station personnel's immediate response to the event and restart readiness assessment. The licensee restart readiness assessment was performed in accordance with CNG-OP-1.01-1006, Post-Trip Reviews, Rev. 1.

b. <u>Findings</u>

No findings of significance were identified.

Operators promptly announced the event, implemented the appropriate emergency operating procedures, and correctly assessed EALs. However, human performance deficiencies and/or procedure deficiencies led to Unit 1 exceeding the TS pressurizer level limit (Section 3.1) and untimely verification of offsite power source availability. Constellation augmented the on-shift staff promptly to support initial diagnosis and corrective actions to address the numerous degraded equipment problems.

The post-trip review was sufficient to ensure operator performance issues and significant equipment issues were identified and addressed. Notwithstanding, the team identified several deficiencies which posed challenges to the effectiveness of the licensee restart readiness assessment (CR 2010-004502). The team discussed each issue with licensee management who entered the issues into the corrective action program, as applicable. One notable issue was that station personnel did not quarantine several failed components (breaker 152-2501, 2B EDG oil sensing line contents, relay 251G/B-22-2). This adversely limited the as-found information available to diagnose the failure mechanisms.

4.2 <u>Post-Event Root Cause Analysis and Actions</u>

a. Inspection Scope

The team reviewed the RCAR for the 2010 Dual Unit Trip to determine whether the causes of the event and associated human performance and equipment challenges were properly identified. Additionally, the team assessed whether interim and planned long term corrective actions were appropriate to address the cause(s).

b. Findings

No findings of significance were identified.

The RCAR properly evaluated causes and appropriate corrective actions for several equipment challenges. For example, evaluation and corrective actions for the Unit 1 roof leakage which initiated the ground fault event were comprehensive. In addition to the root cause, the RCAR identified several contributing causes including deficient maintenance rule implementation and performance monitoring, over reliance and inadequate vendor oversight, incomplete incorporation of Quality Assurance findings, and insufficient engineering involvement in roof construction. Interim corrective actions were appropriate and long term actions were being developed through the corrective action program.

In several other areas the team determined the RCAR lacked depth and technical rigor in identifying and assessing potential causes. In each case the RCAR developed an explanation for what may have caused the event or equipment response, but did not fully develop other potential causes. Examples included:

- RCAR did not identify the failure to implement an Agastat relay monitoring program when the 10 year replacement PM was eliminated (2B EDG failure);
- RCAR conclusion that loose diaphragm retaining ring screws on the Agastat relay were caused by vibration and were the result of a manufacturing defect were not well supported by the contracted failure analysis or data evaluation (2B EDG failure);
- Information that the relay induction disc did not freely rotate back to the original position during bench troubleshooting, was not incorporated into the RCAR (relay 2RY251G/B-22-2 failure);
- RCAR did not thoroughly review previous internal OE regarding induction disc failure on CO-8 type relays. Station personnel did not recognize the sensitivity of the induction disc to sticking/binding (relay 251G/B-22-2 failure);
- RCAR did not include or address the 2008 as-found inspection results which found the armature linkage misaligned and the trip coil loose. This was an unexpected and infrequent occurrence (breaker 152-2501 failure); and
- RCAR concluded the 152-2501 breaker failure was due to mechanical binding in the trip linkage caused by human error during the October 2008 trip armature bolt replacement. However, corrective actions did not investigate other breaker maintenance performed by these technicians during that time period.

The team reviewed these issues and determined that none of these issues involved violations of regulatory requirements or were already described as part of the previously discussed violations in this report.

4.3. Review of Operating Experience

a. Inspection Scope

The team reviewed Constellation's use of pertinent industry and station operating experience (OE), including evaluation of potential precursors to this event.

b. Findings

No findings of significance were identified.

The team identified several instances where Constellation had not effectively evaluated or initiated actions to address related station or industry operating experience issues. Examples included:

- Unit 1 and Unit 2 45 foot switchgear room roof leakage onto electrical switchgear had been identified numerous times since 2002, but not corrected. Fifty-eight open work orders for roof leaks, several > 24 months old, had not been implemented (Section 2.1).
- Industry OE has reported numerous problems with Agastat series 7000 relays; several affecting reliability of the actuation setpoint. Yet engineers extended both the service life and calibration periodicity of the EDG lube oil pressure trip time delay relays beyond the vendor specified periods without adequate technical basis (Section 2.2).
- Technicians routinely did not consider relay actuation outside of the acceptance band to be a test failure. Often no condition report was initiated and no drift/performance trending was performed. Corrective action was often limited to adjusting the as-left setpoint to within the acceptance band (e.g, agastat 7000 series time delay relays, CO-8 overcurrent protection relays) (CR 2010-004090).

The team reviewed these issues and determined that none of these issues involved violations of regulatory requirements or were already described as part of the previously discussed violations in this report.

5. Risk Significance of the Event

a. Initial Assessment

The initial risk assessment for this event is documented in the enclosed SIT charter.

b. Final Assessment

Onsite follow-up and discussions with the Constellation PRA staff verified that there were no additional plant conditions or operator performance issues that significantly alter the initial event risk assessments performed for both units. The Unit 1 reactor trip estimated conditional core damage probability (CCDP) was calculated to be 2.6 E-6 for the February 18, 2010 reactor trip. The Unit 2 reactor trip CCDP, accounting for a loss of reactor coolant forced circulation (all RCPs tripped), loss of heat sink (main condenser), and failure of the 2B EDG to run, was estimated to be 1.5 E-5 for the February 18, 2010 event.

4OA3 Follow-up of Events

.1 (Closed) Licensee Event Report (LER) 05000317/2010-001, Reactor Trip Due to Water Intrusion into Switchgear Protective Circuitry

On February 18, at 8:24 a.m., the Unit 1 reactor automatically tripped from 93 percent reactor power in response to a RCS low flow condition. Water had leaked through the auxiliary building roof into the 45' switchgear room, causing an electrical ground which tripped the 12B RCP, thereby initiating the reactor protection system trip on RCS low flow. Three of the four Unit 1 RCPs continued operating. The electrical ground and failure of a ground fault protection relay caused service transformer P-13000-2 to isolate, thereby deenergizing the 14 4 kV safety bus and the 1Y10 120 volt instrument bus. The 1B EDG automatically started and reenergized the 14 bus as designed. The LER accurately described operator response to the event. The team reviewed the LER and identified no findings of significance beyond those previously documented in this report (NRC Inspection Report No. 05000317/2010006). This LER stated a supplemental LER will document a complete description of corrective actions after the event analysis and cause determination is complete. This LER is closed.

.2 (Closed) Licensee Event Report (LER) 05000318/2010-001, Reactor Trip Due to Partial Loss of Offsite Power

On February 18, at 8:24 a.m., the Unit 2 reactor automatically tripped from 99.5 percent reactor power due to a loss of power to all four RCPs and the associated reactor protection system RCS low flow trip. The event emanated from a ground fault on Unit 1 (see Section 2.1). A ground O/C relay failed to actuate as designed, permitting the Unit 1 ground O/C condition to reach Unit 2. Unit 2 electrical protection responded by deenergizing the 500 kV "Red Bus" offsite power supply and multiple onsite electrical buses including the 24 4 kV safety bus. The 2B EDG started as designed, but tripped on low lube oil pressure (see Section 2.2). The LER accurately described operator response to the event. The team reviewed the LER and identified no findings of significance beyond those previously documented in this report (NRC Inspection Report No. 05000317/2010006). This LER stated a supplemental LER will document a complete description of corrective actions after the event analysis and cause determination is complete. This LER is closed.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 30, 2010, the team presented their overall findings to members of Constellation management led by Mr. G. Gellrich, Site Vice President, and other members of his staff who acknowledged the findings. The team confirmed that proprietary information reviewed during the inspection period was returned to Constellation.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel	
G. Gellrich	Site Vice President
K. Allor	Senior Operations Instructor
P. Amos	Performance Improvement
P. Darby	Principal Assessor, Engineering Quality Performance Assessment
S. Dean	Manager, Maintenance
M. Draxton	Manager, Nuclear Training
D. Fitz	Communications
M. Flynn	HR Director
D. Frye	Manager, Operations
M. Gahan	GS, Design Engineering
G. Gellrock	Supervisor
S. Henry	Manager, Work Management
J. Koebel	PRA
D. Lauver	Director, Licensing
W. Mahaffee	Supervisor, Chemistry Operation
J. McCullum	Supervisor, Instrumentation and Controls
K. Mills	Assistant Operations Manager
P. O'Malley	Quality Performance Assessment
T. Riti	GS, System Engineering
K. Roberson	Manager, NSS
A. Simpson	Engineering/Licensing
R. Stark	Design Engineering
T. Trepanier	Plant General Manager
Others	
S. Gray	Power Plant Research Program Manager, Department of Natural Resources, State of Maryland
M. Griffin	Nuclear Emergency Preparedness Coordinator, Department of the Environment, State of Maryland

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000317/318/2010006-01	NCV	Failure to Thoroughly Evaluate and Promptly Correct Degraded Conditions Associated with Auxiliary Building Roof Leakage (Section 2.1)
05000317/318/2010006-02	AV	Inadequate Preventive Maintenance Results in the Failure of the 2B Emergency Diesel Generator (Section 2.2)
05000317/318/2010006-03	NCV	Failure to Evaluate Degraded Conditions Associated with CO-8 Relays and Implement

		Timely and Effective Action to Correct the Condition Adverse to Quality (Section 2.3)
05000317/318/2010006-04	FIN	Failure to Translate Design Calculation Setpoint of Phase Overcurrent Relay on Feeder Breakers (Section 2.3)
05000317/318/2010006-05	NCV	Failed to Establish Adequate Procedures for Letdown Restoration (Section 3.1)

Opened and Closed

05000317/2010-001	LER	Reactor Trip Due to Water Intrusion into Switchgear Protective Circuitry (Section 4OA3.1)
05000318/2010-001	LER	Reactor Trip Due to Partial Loss of Offsite Power (Section 4OA3.2)

LIST OF DOCUMENTS REVIEWED

Drawings
61004, Single Line Meter & Relay Diagram 13 kV System, Rev. 26
61001SH0001, Electrical Main Single Line Diagram FSAR Fig. No. 8-1, Rev. 42
63070SH0009, Schematic Diagram 13 KV Service Bus 22 RCP Bus Feeder Breaker 252-2201, Rev. 11
63049, AC Schematic Diagram Service Bus 22 & Service Transformer P-13000-2, Rev. 17

Condition Reports (CR)

IRE-000-433	CR 2009-008115	CR 2010-001707
IRE-004-399	CR 2009-008537	CR 2010-001779
IRE-004-400	CR 2009-008635	CR 2010-001780
IRE-011-621	CR 2010-001330	CR 2010-001781
IRE-011-769	CR 2010-001340	CR 2010-001782
IRE-020-768	CR 2010-001351	CR 2010-001783
IRE-020-769	CR 2010-001355	CR 2010-001784
IRE-020-776	CR 2010-001381	CR 2010-001787
IRE-022-227	CR 2010-001516	CR 2010-001813
IRE-026-951	CR 2010-001517	CR 2010-001888
IRE-028-751	CR 2010-001544	CR 2010-002875
IRE-031-691	CR 2010-001553	CR 2010-004411
IRE-032-766	CR 2010-001586	CR 2010-004493
CR 2008-001582	CR 2010-001592	CR 2010-004502
CR 2008-002458	CR 2010-001682	CR 2010-004613
CR 2009-004060	CR 2010-001685	CR 2010-004652
CR 2009-004074	CR 2010-001690	CR 2010-004672
CR 2009-004606	CR 2010-001691	CR 2010-004673
CR 2009-005508	CR 2010-001671	CR 2010-004674
CR 2009-005629	CR 2010-001699	
CR 2009-006187	CR 2010-001700	

Maintenance Orders
MO #1200801597, Replace Flex Hoses on the 1B EDG
MO #2199901416, Calibrate 2B EDG Lube Oil Pressure Gauge, 2-PI-4796
MO #2200000476, Perform E-19 on 2B EDG Agastat Relays
MO #2200201832, 2B EDG Engine Stop Relay
MO #2200401152, 2B EDG Engine Stop Relay
MO #2200501401, 2B EDG Engine Stop Relay
MO #2200700554, Replace Flex Hoses on the 2A EDG
MO #2200700555, Replace Flex Hoses on the 2B EDG
MO #2200700852, 2B EDG Engine Stop Relay

Operability Evaluation
OE-2009-003712

Procedures
Auxiliary Building Walkdown Results, MN-1-319 "Structure and System Walkdowns," Rev. 5
Auxiliary Building Walkdown Results, MN-1-319 "Structure and System Walkdowns," Rev. 7
10-02 "Rain/Snow Water Intrusion Compensatory Measures," Rev. 1
CNG-AM-1.01-2000, "Scoping and Identification of Critical Components," Rev. 00200
CNG-CA-1.01-1000, "Corrective Action Program," Rev. 0200
CNG-OP-1.01-1006, "Post Trip Reviews," Rev. 00001
CNG-OP-1.01-2000, "Operations Logkeeping and Station Rounds," Rev. 00100
CNG-QL-1.01-1007, "Quality Performance Assessment Process," Rev. 00201
CNG-PR-1.01-1009, "Procedure Use and Adherence Requirements," Rev. 00400
FTE-87, "Powell 13.8 kV Type PVDH Vacuum Circuit Breaker Inspection," Rev. 00101
FTE-51A, "General Electric Cubicle Inspection," Rev. 2
FTE-59, "Periodic Maintenance, Calibration and Functional Testing of Protective Relays," Rev. 5
MN-1-319 "Structure and System Walkdowns," Rev. 7
NO-1-200, "Control of Shift Activities, Rev. 04401
NO-1-201, "Calvert Cliffs Operating Manual," Rev. 02000
OI-2A, "Chemical and Volume Control System," Rev. 55/Unit 1

Miscellaneous
Control Room Operations Narrative Logs
Operations Administrative Policy 90-7, Guidelines, System Expert and Shift Crew Ownership Program Guidelines and Expectations, January 27, 2010, Change 15
Plant Areas System 102 Walkdowns, 1- Unit 1 performed January 5, 2010, & March 31, 2010
System 102 "Plant Areas," Maintenance Rule Scoping Document, Rev. 30
Site Roof Leakage Condition Report Scoping Document
U-1 Alarm History Printout for February 18, 2010
U-2 Alarm History Printout for February 18, 2010
U-1 Sequence of Events Recorder Printout for February 18, 2010
U-2 Sequence of Events Recorder Printout for February 18, 2010

- Engineering Service Package ES200100067, Revision 1, Delete Requirement in E-406 Sec 234.0.1 to Change Out Agastat Prior to Ten Years and Remove Testing Recommendations to VTM 15-167-001
- Procedure E-406, Rev. 0, Installation and Replacement for Agastat Relays
- RO01617, Revision 4, Guideline for Testing Agastat Relay Models
- Constellation Nuclear Generation Fleet Administrative Procedure CNG-CA-1.01-1004 Root Cause Analysis, Revision 00301

- Procedure FTI-328, Revision 1, Calibration Check/Calibration of Allen-Bradley Pressure Switches
- Rover Maintenance Approval and Closeout Form, MN-1-101, Revision 03601, 2A EDG Oil Sensing Line Flush
- Calvert Cliffs Surveillance Test Procedure, STP O-8B-2, Revision 26, Test of 2B DG and 4 kV Bus 24 LOCI Sequencer
- Calvert Cliffs Surveillance Test Procedure, STP O-8A-2, Revision 26, Test of 2A DG and 4 kV Bus 24 LOCI Sequencer
- Operating Experience OE13852 – Inadequate Venting of the Emergency Diesel Generator Lubricating Oil System
- Schematic Diagram Diesel Generator No. 2B Engine Control, No. 63086SH0010, Revision 39
- Work Order C90791765, 2B Diesel Generator Failed to Start and Load on the 24 4 kV Bus on an ESFAS UV Signal
- Operating Experience, ACE 013617, Surry EDG Agastat Relay Failure
- Constellation Nuclear Generation Fleet Administrative Procedure CNG-AM-1.01-1018 Preventive Maintenance Program, Revision 00400
- Vendor Manual 15167-001-1001, Agastat Timing Relays 7000 Series
- Vendor Manual 15167-001-1005, Tyco Electronics
- Herguth Labratories Crankcase Oil Sample Data
- Troubleshooting Data Sheet to Determine Cause of 2B EDG Trip after Closing onto 24 4 kV Bus
- CCNPP Procurement Engineering Specification, PES – 25180, Revision 17, Agastat Relays and Associated Hardware
- Maintenance Strategy 2RY2DG2BA/T3A Relay
- 2-PS-4798 Master Calibration Data Package, 2/19/10

Root Cause Analysis
CNG-CA-1.01-1004 "Root Cause Analysis" Dual Unit Trip, Rev 00301

Apparent Cause Evaluation
IRE-007-705

Calculations/Engineering Evaluation Reports
E-90-058, Breaker 252-1101, 1102, 1103, Rev. 2
E-90-061, Breaker 252-2101, 2102, 2103, Rev. 2
E-90-062, Breaker 252-2201, Rev. 2
RCS Letdown Line Evaluation for Potential Water Hammer dated 3/16/10

Completed Tests/Surveillances
E-30, 4.16 kV Magne-Blast Circuit Breaker Overhaul Procedure, Performed 10/04/04
FTE-51, 4 kV General Electric Magne-Blast Circuit Breaker Inspection, Performed 11/18/08, 4/14/05
FTE-59, Periodic Maintenance, Calibration and Functional Testing of Protective Relays, Performed 04/06/00, 03/26/03, 05/03/04, 10/01/05, 05/08/07, 10/10/07, 03/08/08, 11/20/08, 02/28/09
FTE-87, Powell 13.8 kV Type PVDH Vacuum Circuit Breaker Inspection, Performed 3/15/07
STP-O-90-1 and STP-O-90-2, "AC Sources and Onsite Power Distribution Systems 7 Day Operability Verification, Rev. 22

LIST OF ACRONYMS

AV	Apparent Violation
CC	Calvert Cliffs
ΔCDF	Increase in Core Damage Frequency
CFR	Code of Federal Regulations
CR	Condition Report
CVCS	Chemical and Volume Control System
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EAL	Emergency Action Level
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ESDP	Emergency Shutdown Panel
gpm	Gallons per Minute
IMC	Inspection Manual Chapter
kV	Kilovolt
ΔLERF	Increase in Large Early Release Frequency
LER	Licensee Event Report
LO	Lube Oil
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
OC	Overcurrent
OE	Operating Experience
PM	Preventive Maintenance
PORC	Plant Onsite Review Committee
PPC	Plant Process Computer
PRA	Probabilistic Risk Assessment
RCAR	Root Cause Analyses Report
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RPS	Reactor Protection System
SDP	Significance Determination Process
SM	Shift Manager
SPM-A	Woodward SPM-A Synchronizer
SRA	Senior Reactor Analyst
SIT	Special Inspection Team
SPAR	Standardized Plant Analysis Risk
ST	Surveillance Test
TD	Time Delay
TS	Technical Specification
UV	Under-Voltage
VCT	Volume Control Tank

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PA 19406-1415

SPECIAL INSPECTION TEAM CHARTER

February 22, 2010

MEMORANDUM TO:	Glenn Dentel, Manager Special Inspection Team	
	David Kern, Leader Special Inspection Team	
FROM:	David C. Lew, Director Division of Reactor Projects	**/RA/**
	Darrell J. Roberts, Director Division of Reactor Safety	**/RA/**
SUBJECT:	SPECIAL INSPECTION TEAM CHARTER - CALVERT CLIFFS PARTIAL LOSS OF OFFSITE POWER AND DUAL UNIT TRIP WITH COMPLICATIONS ON FEBRUARY 18, 2010	

In accordance with Inspection Manual Chapter (IMC) 0309, "Reactive Inspection Decision Basis for Reactors," a Special Inspection Team (SIT) is being chartered to evaluate a Calvert Cliffs dual unit trip with complications which occurred on February 18, 2010. The decision to conduct this special inspection was based on meeting multiple deterministic criteria (multiple failures in equipment needed to mitigate an actual plant event, significant unexpected system interactions, and events involving safety related equipment deficiencies) specified in Enclosure 1 of IMC 0309 and the event representing a preliminary conditional core damage probability in the low E-6 range for Unit 1 and low E-5 range for Unit 2.

The SIT will expand on the inspection activities started by the resident team immediately after the event. The team will review Constellation's organizational and operator response to the event, equipment and design deficiencies, and the causes for the event and subsequent issues. The team will collect data, as necessary, to refine the existing risk analysis. The team will also assess whether the SIT should be upgraded to an Augmented Inspection team.

The inspection will be conducted in accordance with the guidance contained in NRC Inspection

Procedure 93812, "Special Inspection," and the inspection report will be issued within 45 days following the final exit meeting for the inspection.
The special inspection will commence on February 22, 2010. The following personnel have been assigned to this effort:

Manager:	Glenn Dentel, Branch Chief, Projects Branch 1, Division of Reactor Projects (DRP), Region I
Team Leader:	David Kern, Senior Resident Inspector DRP, Region I
Full Time Members:	Peter Presby, Operations Inspector Division of Reactor Safety (DRS), Region I
	Manan Patel, Electrical Inspector DRS, Region I
	Brian Smith, Resident Inspector DRP, Region I
Part Time Member:	William Cook, Senior Reactor Analyst DRS, Region I

Enclosure: Special Inspection Charter

G. Dentel, D. Kern 2-3

The special inspection will commence on February 22, 2010. The following personnel have been assigned to this effort:

Manager:	Glenn Dentel, Branch Chief, Projects Branch 1, Division of Reactor Projects (DRP), Region I
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	Brian Smith, Resident Inspector DRP, Region I
Part Time Member:	William Cook, Senior Reactor Analyst DRS, Region I

Enclosure: Special Inspection Charter

cc w/encl:
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B. Boger, NRR
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SUNSI Review Complete: ___NP___ **(Reviewer's Initials)**

Non-Public Designation Category: MD 3.4 Non-Public __B.1__ **(A.3 - A.7 or B.1)**

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DATE	02/22/10	02/22/10	02/22/10	02/22/10	02/22/10

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Special Inspection Team Charter
Calvert Cliffs Nuclear Power Plant
Dual Unit Trip with Complications due to a Partial Loss of Offsite Power
on February 18, 2010

Background:

At 8:24 a.m. on February 18, 2010, Calvert Cliffs Unit 1 experienced an unexpected loss of the 12B reactor coolant pump (RCP). The loss of the RCP trip resulted in a valid reactor protection system (RPS) actuation on low reactor coolant system flow and a Unit 1 trip.

At approximately the same time, Unit 2 experienced a loss of the 500 kV to 13.8 kV transformer for the "Red Bus" (500 kV). The Red Bus is the feeder for offsite power for the Unit 1 "14" and Unit 2 "24" 4 kV safety buses. Unit 2 experienced the following system/component responses based on the loss of the Red Bus: loss of the non-safety related buses, a loss of load RPS trip signal, a loss of all RCPs, and a Unit 2 trip. The loss of the non-safety related buses resulted in the loss of the circulating water pumps, the main feedwater pumps, and condensate pumps, and the subsequent loss of the normal heat sink. Bus 21, the other Unit 2 safety 4 kV bus, normally aligned to the Black Bus, remained energized.

The loss of power to the "14" and "24" 4 kV safety buses resulted in a valid start signal for the 1B and 2B EDGs, respectively. The 1B EDG started and re-powered the "14" safety bus; however, the 2B EDG tripped during loading resulting in the loss of the "24" safety bus. This resulted in the unavailability of the "B" safety train. Calvert Cliffs subsequently restored power to the "24" safety bus via the Black Bus alternate power supply.

Unit 1 was cooled down and entered a refueling outage that was originally scheduled to begin on February 20, 2010. Unit 2 was stabilized on natural circulation, and normal decay heat removal was subsequently restored; the plant has entered a forced outage.

At the time of the event, the resident team responded to the control room and monitored licensee actions to stabilize the plant and restore offsite power. An NRC regional inspector was also deployed to the site to supplement the resident staff.

Basis for the Formation of the SIT:

The IMC 0309 review concluded that three deterministic criteria were met. The deterministic criteria met included: 1) multiple failures of plant equipment in systems used to mitigate an event; 2) significant unexpected system interactions; and 3) events involving safety related equipment deficiencies. These criteria were met based on the partial loss of offsite power due to the transformer loss, and the subsequent failure of the 2B EDG to start and restore a safety bus. In addition, the system interactions between the 12B RCP trip and the transient, which resulted in the opposite 500 kV transformer loss, were unexpected. The Unit 2 transformer loss also resulted in a complete loss of forced flow to Unit 2 due to the expected loss of all four RCPs, and the loss of the Unit 2 main condenser as a heat sink.

The event was also evaluated for risk significance because the IMC 0309 review concluded that at least one deterministic criteria was met. Based upon best available information, the Region I Senior Risk Analyst (SRA) conducted a preliminary risk estimate for each unit on February 18. Using the Graphical Evaluation Module initiating event quantification tool and the Calvert Cliffs Unit 1 and Unit 2 Standardized Plant Analysis Risk (SPAR) models, the conditional core damage probability (CCDP) for Unit 1 was estimated to be in the low E-6 range, and the Unit 2 estimated CCDP was in the low E-5 range. On February 19, 2010, the SRA discussed these results with the Constellation PRA staff and determined that the risk estimates (CCDP) performed by Constellation favorably compared to the NRC SPAR model generated values.

Based upon the preliminary CCDP estimates, and in accordance with IMC 0309, the Unit 1 and Unit 2 events fall within the overlap ranges of No Additional Inspection and Special Inspection Team (SIT) for Unit 1, and SIT and Augmented Inspection Team (AIT) for Unit 2. After consultation with NRC headquarters personnel, an SIT was initiated.

Objectives of the Special Inspection:

The SIT will review Constellation's organizational and operator response to the event, equipment and design deficiencies, and the causes for the event and subsequent issues. The team will collect data, as necessary, to refine the existing risk analysis. The team will also assess whether the SIT should be upgraded to an Augmented Inspection Team. Additionally, the team leader will review lessons learned identified during this special inspection and, if appropriate, prepare a feedback form on recommendations for revising the Reactor Oversight Process (ROP) baseline inspection procedures.

To accomplish these objectives, the team will:

1. Develop a complete sequence of events including follow-up actions taken by Constellation.

2. Review and assess the equipment response to the event. This assessment should include an evaluation of the consistency of the equipment response with the plant's design and regulatory requirements. In addition, review and assess the adequacy of any operability assessments, corrective and preventive maintenance, and post maintenance testing.

3. Review and assess operator performance including procedures, logs, communications (internal and external), and emergency plan implementation.

4. Review and assess the effectiveness of Constellation's response to this event. This includes overall organizational response, failure modes and effect analysis developed for the equipment challenges, causal analyses conducted, and interim and proposed longer term corrective actions taken.

5. Evaluate Constellation's application of pertinent industry operating experience and evaluation of potential precursors, including the effectiveness of any actions taken in response to the operating experience or precursors; and

6. Collect any data necessary to refine the existing risk analysis and document the final risk analysis in the SIT report.

Guidance:

Inspection Procedure 93812, "Special Inspection", provides additional guidance to be used by the Special Inspection Team. Team duties will be as described in Inspection Procedure 93812. The inspection should emphasize fact-finding in its review of the circumstances surrounding the event. It is not the responsibility of the team to examine the regulatory process. Safety concerns identified that are not directly related to the event should be reported to the Region I office for appropriate action.

The team will conduct an entrance meeting and begin the inspection on February 22, 2010. While on site, the team Leader will provide daily briefings to Region I management, who will coordinate with the Office of Nuclear Reactor Regulation, to ensure that all other parties are kept informed. A report documenting the results of the inspection will be issued within 45 days following the final exit meeting for the inspection.

This Charter may be modified should the team develop significant new information that warrants review.

DETAILED SEQUENCE OF EVENTS
February 18, 2010 Dual Unit Trip with Complications

The sequence of events was constructed by the team from review of Control Room Narrative Logs, corrective action program condition reports, post transient review report, process plant computer (PPC) data (alarm message file and plant parameter graphs) and plant personnel interviews. The sequence of events is listed separately by Unit 1 and Unit 2.

UNIT 1 EVENT TIMELINE		
Clock Time	Event Time	Description
02/18/2010		
08:24:25:225	0.000 sec	A phase to ground fault occurs on the 13 kV supply line to Unit 1 Reactor Coolant Pump (RCP) 12B Motor, upstream of 12B RCP Breaker 252-14P02, which is already open (normal lineup).
08:24:25:225	0.000 sec	RCP 12B Breaker 252-14P01 trips open on differential overcurrent relay actuation, stopping 12B RCP.
08:24:27:251	2.026 sec	Feeder Breaker 252-2104 to 13 kV Service Bus 21 trips open, de-energizing Unit 2 Non-vital balance of plant, Unit 2 Vital 4 kV Bus 24 and Unit 1 Vital 4 kV Bus 14.
08:24:27:421	2.196 sec	208/120 V/AC Bus 12 de-energizes, resulting in isolation of the Unit 1 RCS letdown flowpath in the Chemical and Volume Control System (CVCS).
08:24:28:803	3.578 sec	13 kV Service Bus 22 Supply Breaker 252-2202 to Unit 1 RCPs trips open. Unit 1 RCPs are not affected as they are aligned to their normal power supply from 13 kV Station Service Transformer P-13000-1 through 13 kV Service Bus 12.
08:24:28:781	3.556 sec	500 kV Switchyard Red Bus Isolation Breaker 552-41 trips open.
08:24:28:783	3.558 sec	500 kV Switchyard Red Bus Isolation Breakers 552-21 and 552-61 trip open, completing the high side isolation 13 kV Station Service Transformer P-13000-2.
08:24:29:110	3.885 sec	Unit 1 automatic reactor trip on reactor coolant low flow signal from 93% initial reactor power level. 3 of 4 Unit 1 reactor coolant pumps are still operating.
08:24:29:146	3.921 sec	Unit 1 reactor trip breakers open.
08:24:29:417	4.192 sec	Unit 1 turbine trip.
08:24:29:423	4.198 sec	Undervoltage signal actuates on Unit 1 4 kV Vital Bus 14, initiating the 1B Emergency Diesel Generator start sequence.
08:24:29:948	4.723 sec	Unit 1 4 kV Vital Bus 14 Normal Feeder Breaker 152-1414 trips open.
08:24:33:818	8.593 sec	13 kV Service Bus 21 Supply Breaker 252-2103 to Transformer U-4000-22 opens.
08:24:33:818	8.593 sec	13 kV Service Bus 21 Supply Breaker 252-2102 to Transformer U-4000-21 opens.
08:24:33:819	8.594 sec	13 kV Service Bus 21 Supply Breaker 252-2101 to Transformer U-4000-23 opens.
08:24:36:101	10.876 sec	Emergency Diesel Generator 1B reaches 810 rpm.
08:24:37:255	12.030 sec	Emergency Diesel Generator 1B Output Breaker 152-1403 to 4 kV Vital Bus 14 closes.
08:24:37:267	12.042 sec	Shutdown Sequencer on 4 kV Vital Bus 14 actuates, to re-start bus loads.

UNIT 1 EVENT TIMELINE		
Clock Time	**Event Time**	**Description**
08:24:37:748	12.523 sec	208/120 V/AC Bus 12 re-energizes.
08:24:37:774	12.549 sec	Undervoltage signal clears on Unit 1 4 kV Vital Bus 14.
08:24:42:015	16.790 sec	Reactor Operator backs up the automatic reactor trip signal by depressing manual reactor trip pushbuttons.
08:24:55	30 sec	Crew enters EOP-0, Post-Trip Immediate Actions
08:26:35	2.17 min	Component Cooling Pump 11 is manually started. Component Cooling system pressure and flow are restored.
08:31	7 min	Charging Pump 13 re-started.
08:40	16 min	Crew exits EOP-0 and enters EOP-1, Reactor Trip.
09:00	36 min	Pressurizer level out of EOP control band high, >180 inches.
09:02	38 min	Charging Pump 11 stopped.
09:12	48 min	Operators attempt to restore CVCS letdown (1st attempt). Charging Pump 11 started. Letdown Isolations CVC-515 and 516 opened.
09:20	56 min	Charging Pump 11 stopped.
09:37	73 min	Letdown Isolation Valves CVC-515 and 516 closed.
10:41	2.28 hrs	Pressurizer level returns within EOP control band, <180 inches.
10:44	2.33 hrs	Operators attempt to restore CVCS letdown (2nd attempt). Charging Pump 11 started. Letdown Isolations CVC-515 and 516 opened.
10:47	2.38 hrs	Pressurizer level out of EOP control band high, >180 inches.
11:07	2.72 hrs	Charging Pump 11 stopped.
11:28	3.07 hrs	Letdown Isolation Valves CVC-515 and 516 closed.
11:39	3.25 hrs	Operators attempt to restore CVCS letdown (3rd attempt). Charging Pump 11 started. Letdown Isolations CVC-515 and 516 opened.
11:47	3.38 hrs	Completed 4 hr report to NRC, as required per 10CFR50.72.
11:50	3.43 hrs	Charging Pump 11 stopped.
11:52	3.47 hrs	Letdown Isolation Valves CVC-515 and 516 closed.
12:02	3.63 hrs	Pressurizer level above Tech Spec limit, >225 inches.
12:07	3.72 hrs	Pressurizer level returns within Tech Spec limit, <225 inches.
12:07	3.72 hrs	Completed STP-O-90-1, AC Sources and Onsite Power Distribution Systems 7 Day Operability Verification.
12:11	3.78 hrs	Disconnects for 500 kV Switchyard Breaker 552-21 are opened.
12:14	3.83 hrs	Disconnects for 500 kV Switchyard Breaker 552-61 are opened.
12:15	3.85 hrs	Disconnects for 500 kV Switchyard Breaker 552-23 are opened.
12:17	3.88 hrs	Disconnects for 500 kV Switchyard Breaker 552-22 are opened.
12:18	3.90 hrs	Disconnects for 500 kV Switchyard Breaker 552-63 are opened.
13:06	4.70 hrs	Pressurizer level returns within EOP control band, <180 inches.
13:09	4.75 hrs	Operators attempt to restore CVCS letdown (4th attempt). Charging Pump 11 started. Commenced RCS boration from 11 Boric Acid Tank.
13:11	4.77 hrs	Pressurizer level out of EOP control band high, >180 inches.
13:17	4.88 hrs	Letdown Isolations CVC-515 and 516 opened. CVCS letdown restored. Letdown Excess Flow Check Valve 1-CVC-343-CV opened on 4th letdown restoration attempt.
13:30	5.10 hrs	Pressurizer level returns within EOP control band, <180 inches.

UNIT 1 EVENT TIMELINE		
Clock Time	**Event Time**	**Description**
13:38	5.23 hrs	Crew exits EOP-1 and enters OP-5, Plant Shutdown From Hot Standby to Cold Shutdown.
13:46	5.37 hrs	Boration stopped, charging suction from VCT to lower VCT level.
13:58	5.57 hrs	Boration re-commenced from 11 Boric Acid Tank.
14:07	5.72 hrs	4 kV Vital Bus 14 Alternate Feeder Breaker 152-1401 closed.
14:13	5.82 hrs	Emergency Diesel Generator 1B Output Breaker 152-1403 to 4 kV Vital Bus 14 opened.
14:15	5.85 hrs	Emergency Diesel Generator 1B shutdown.
14:16	5.87 hrs	Boration completed. Approximately 2420 gallons of boric acid injected.
14:37	6.22 hrs	RCS sampled for boron. Concentration at 529 ppm.
16:00	7.6 hrs	RCS sampled for boron. Concentration at 622 ppm.
21:50	13.4 hrs	Disconnects for 500 kV Switchyard Breaker 552-22 closed.
22:00	13.6 hrs	500 kV Switchyard Breaker 552-22 closed.
22:01	13.6 hrs	Disconnects for 500 kV Switchyard Breaker 552-23 closed.
22:07	13.7 hrs	500 kV Switchyard Breaker 552-23 closed.
02/19/2010		
12:01	27.6 hrs	SMECO now credited to 4 kV Bus 24.
02/20/2010		
17:05	56 hrs	Started 12B RCP.
19:20	59 hrs	Commenced RCS cooldown to MODE 5 per OP-5.
02/21/2010		
05:38	69 hrs	Unit 1 in MODE 5, RCS temperature < 200°F.
05:50	69.5 hrs	Divorced from SMECO, re-energized 500 kV Red Bus.

UNIT 2 EVENT TIMELINE		
Clock Time	**Event Time**	**Description**
02/18/2010		
08:24:25:225	0.000 sec	A phase to ground fault occurs on the 13 kV supply line to Unit 1 Reactor Coolant Pump (RCP) 12B Motor, upstream of 12B RCP Breaker 252-14P02, which is already open (normal lineup).
08:24:25:225	0.000 sec	RCP 12B Breaker 252-14P01 trips open on differential overcurrent relay actuation, stopping 12B RCP.
08:24:27:251	2.026 sec	Feeder Breaker 252-2104 to 13 kV Service Bus 21 trips open, de-energizing Unit 2 Non-vital balance of plant, Unit 2 Vital 4 kV Bus 24 and Unit 1 Vital 4 kV Bus 14.
08:24:27:478	2.253 sec	208/120 V/AC Bus 22 de-energizes, resulting in isolation of the Unit 2 RCS letdown flowpath in the Chemical and Volume Control System (CVCS).
08:24:28:803	3.578 sec	13 kV Service Bus 22 Supply Breaker 252-2202 to Unit 1 RCPs trips open. Unit 1 RCPs are not affected as they are aligned to their normal power supply from 13 kV Station Service Transformer P-13000-1 through 13 kV Service Bus 12.
08:24:28:781	3.556 sec	500 kV Switchyard Red Bus Isolation Breaker 552-41 trips open.

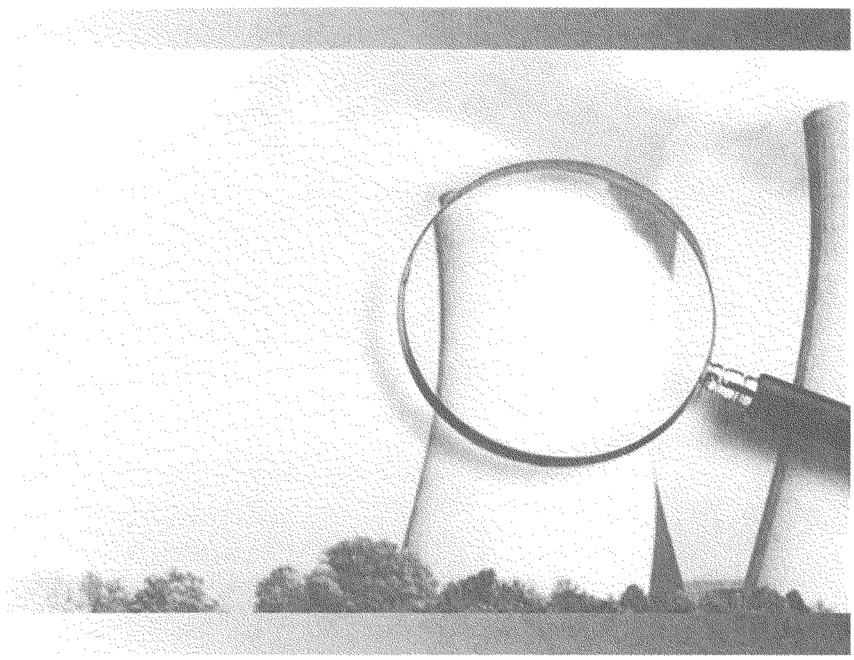
UNIT 2 EVENT TIMELINE		
Clock Time	**Event Time**	**Description**
08:24:28:783	3.558 sec	500 kV Switchyard Red Bus Isolation Breakers 552-21 and 552-61 trip open, completing the high side isolation 13 kV Station Service Transformer P-13000-2.
08:24:29:451	4.226 sec	Undervoltage signal actuates on Unit 2 4 kV Vital Bus 24, initiating the 2B Emergency Diesel Generator start sequence.
08:24:29:511	4.286 sec	Unit 2 4 kV Vital Bus 24 Normal Feeder Breaker 152-2401 trips open.
08:24:29:788	4.563 sec	Unit 2 automatic reactor trip on reactor coolant low flow signal from 100% initial reactor power level. All Unit 2 reactor coolant pumps have stopped.
08:24:29:827	4.602 sec	Unit 2 reactor trip breakers open.
08:24:30:019	4.794 sec	Unit 2 turbine trip.
08:24:32:122	6.897 sec	Emergency Diesel Generator 2B reaches 250 rpm.
08:24:33:818	8.593 sec	13 kV Service Bus 21 Supply Breaker 252-2103 to Transformer U-4000-22 opens.
08:24:33:818	8.593 sec	13 kV Service Bus 21 Supply Breaker 252-2102 to Transformer U-4000-21 opens.
08:24:33:819	8.594 sec	13 kV Service Bus 21 Supply Breaker 252-2101 to Transformer U-4000-23 opens.
08:24:33:889	8.664 sec	4 kV Non-Vital Bus 22 Feeder Breaker 152-2201 opens.
08:24:33:909	8.684 sec	4 kV Non-Vital Bus 23 Feeder Breaker 152-2311 opens.
08:24:35:988	10.763 sec	Emergency Diesel Generator 2B reaches 810 rpm.
08:24:37:306	12.081 sec	Emergency Diesel Generator 2B Output Breaker 152-2403 to 4 kV Vital Bus 24 closes.
08:24:37:785	12.560 sec	208/120 V/AC Bus 22 re-energizes.
08:24:37:887	12.662 sec	Undervoltage signal clears on Unit 2 4 kV Vital Bus 24.
08:24:45:155	19.930 sec	Emergency Diesel Generator 2B trips.
08:24:45:185	19.960 sec	Emergency Diesel Generator 2B Output Breaker 152-2403 to 4 kV Vital Bus 24 opens.
08:24:45:320	20.095 sec	208/120 V/AC Bus 22 de-energizes.
08:24:47:315	22.090 sec	Undervoltage signal actuates on Unit 2 4 kV Vital Bus 24.
08:24:55:133	29.908 sec	21 and 22 Steam Generator Feed Pumps low suction pressure trip.
08:24:56:335	31.110 sec	Reactor Operator backs up the automatic reactor trip signal by depressing manual reactor trip pushbuttons.
08:24:55	30 sec	Crew enters EOP-0, Post-Trip Immediate Actions
08:26	2 min	Commenced boration because of loss of power to rod position indication. Aligned gravity feed flowpath from boric acid storage tanks to charging pump suction through 2-MOV-508 and 2-MOV-509.
08:32	8 min	Manually closed 2-MS-343, Main Steam (MS) Isolation to 22 Moisture Separator Reheater (MSR) as alternate action because 2-MS-4019-CV, MS to 22 MSR 2nd Stage failed to close.
08:33	9 min	Steam-driven AFW Pump 21 started to maintain SG heat sink, feeding approximately 150 gpm to each steam generator.
08:34	10 min	2Y10 tied to 2Y09. Power restored to 2Y10.
08:38	14 min	Crew exits EOP-0 and enters EOP-2, Loss of Offsite Power / Loss of Forced Circulation
08:47	23 min	Report of smoke and acrid odor, vicinity of MCC-207

UNIT 2 EVENT TIMELINE		
Clock Time	**Event Time**	**Description**
08:53	29 min	Unit 2 main steam isolation valves closed.
08:57	33 min	4 kV Vital Bus 24 Alternate Feeder Breaker 152-2414 closed. Shutdown sequencer is manually initiated per EOP Attachment 16. The undervoltage signal clears on Unit 2 4 kV Vital Bus 24.
09:00	36 min	Restored Unit 2 CVCS letdown.
09:08	44 min	Low condenser vacuum.
09:10	46 min	VCT Outlet MOV-501 opened. Boration stopped. Approximately 1936 gallons of boric acid injected.
09:20	56 min	Electricians report acrid odor coming from closed 4 kV Non-vital Bus 23 Supply Breaker 152-2501 (cause later diagnosed as a burnt breaker trip coil).
10:46	2.37 hrs	Chemistry samples RCS for boron concentration.
11:00	2.60 hrs	Completed verification of required shutdown margin per NEOP-301 Attachment 3. Required concentration determined to be 1297 ppm.
11:17	2.88 hrs	Started 23 AFW Pump (motor-driven) and stopped 21 AFW Pump (turbine-driven).
11:18	2.90 hrs	Crew exits EOP-2 and enters OP-5, Plant Shutdown From Hot Standby to Cold Shutdown.
11:30	3.10 hrs	Chemistry reports RCS boron 1479 ppm. Initial concentration was 1129 ppm prior to the event.
11:47	3.38 hrs	Completed 4 hr report to NRC, as required per 10CFR50.72.
12:11	3.78 hrs	Disconnects for 500 kV Switchyard Breaker 552-21 are opened.
12:14	3.83 hrs	Disconnects for 500 kV Switchyard Breaker 552-61 are opened.
12:15	3.85 hrs	Disconnects for 500 kV Switchyard Breaker 552-23 are opened.
12:17	3.88 hrs	Disconnects for 500 kV Switchyard Breaker 552-22 are opened.
12:18	3.90 hrs	Disconnects for 500 kV Switchyard Breaker 552-63 are opened.
12:55	4.52 hrs	Completed STP-O-90-2, AC Sources and Onsite Power Distribution Systems 7 Day Operability Verification. This was a missed action requirement of TS 3.8.1, required to be completed within 1 hour of the event.
13:30	5.10 hrs	Commenced RCS Cooldown # 87 using Natural Circulation to target temperature of 445°F per OP-5 to protect RCP seals.
14:45	6.35 hrs	Stopped RCS Cooldown # 87 based on decision to start two RCPs and go on forced circulation. RCS temperature at 505°F.
17:13	8.82 hrs	Started 21B and 22A RCPs. Forced RCS circulation restored.
21:50	13.43 hrs	Disconnects for 500 kV Switchyard Breaker 552-22 closed.
22:00	13.60 hrs	500 kV Switchyard Breaker 552-22 closed.
22:01	13.62 hrs	Disconnects for 500 kV Switchyard Breaker 552-23 closed.
22:07	13.72 hrs	500 kV Switchyard Breaker 552-23 closed.
02/19/2010		
00:29		Started 21 Condensate Pump
02:56		Started 21 Circulating Water Pump
06:00		Restored Gland Sealing Steam
07:10		Performed fast speed start test of EDG 2A.
07:37		EDG 2A paralleled to 4 kV Bus 21.
07:49		EDG 2A at full load on 4 kV Bus 21.
10:08		Aligned SMECO to 13 kV Bus 21.

UNIT 2 EVENT TIMELINE		
Clock Time	**Event Time**	**Description**
11:01		Energized U-4000-21 from 13 kV Bus 21 (SMECO feeding).
11:02		Energized U-4000-22 from 13 kV Bus 21 (SMECO feeding).
12:05	27.6 hrs	Two offsite power sources verified OPERABLE with SMECO supplying 13 kV Bus 21 and available to Unit 2 4 kV buses.
12:28		Unloaded EDG 2A.
12:32		Shutdown EDG 2A. Completed 4 hour loaded test run.
13:52		Restored normal power supply alignment for 208/120 Instrument Bus 22 (2Y10). 2Y09 and 2Y10 are un-tied.
02/20/2010		
17:19	57 hrs	Performed fast speed start test of EDG 2B.
17:36		EDG 2B paralleled to 4 kV Bus 24.
17:46		EDG 2B at full load on 4 kV Bus 24.
21:57		Unloaded EDG 2B.
22:02		Shutdown EDG 2B. Completed 4 hour loaded test run.
22:31	62 hrs	EDG 2B declared OPERABLE.
02/21/2010		
04:24		Commenced drawing main condenser vacuum.
05:50	69.5 hrs	Divorced from SMECO, re-energized 500 kV Red Bus.
09:24		Opened 21 and 22 Main Steam Isolation Valves
09:25	73 hrs	Recommenced RCS Cooldown # 87 to MODE 5 per OP-5.
17:16	81 hrs	Unit 2 in MODE 4, RCS temperature < 350°F.
20:12	84 hrs	Stopped RCS cooldown to degas RCS.
02/22/2010		
01:30	89 hrs	Recommenced RCS cooldown.
05:00	92.6 hrs	Unit 2 in MODE 5, RCS temperature < 200°F.

The NRC and Nuclear Power Plant Safety in 2010

A BRIGHTER SPOTLIGHT NEEDED



Union of Concerned Scientists
Citizens and Scientists for Environmental Solutions

The NRC and Nuclear Power Plant Safety in 2010: A Brighter Spotlight Needed

DAVID LOCHBAUM

Union of Concerned Scientists
March 2011

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Executive Summary

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

In 2010, the NRC reported on 14 special inspections it launched in response to troubling events, safety equipment problems, and security shortcomings at nuclear power plants. This report provides an overview of each of these significant events—or near-misses.

This overview shows that many of these significant events occurred because reactor owners, and often the NRC, tolerated known safety problems. For example, the owner of the Calvert Cliffs plant in Maryland ended a program to routinely replace safety components before launching a new program to monitor degradation of those components. As a result, an electrical device that had been in use for longer than its service lifetime failed, disabling critical safety components.

In another example, after declaring an emergency at its Brunswick nuclear plant in North Carolina, the owner failed to staff its emergency response teams within the required amount of time. That lapse occurred because workers did not know how to activate the automated system that summons emergency workers to the site.

Outstanding Catches by the NRC

This report also provides three examples where onsite NRC inspectors made outstanding catches of safety problems at the Oconee, Browns Ferry, and Kewaunee nuclear plants—before these impairments could lead to events requiring special inspections, or to major accidents.

At the Oconee plant in South Carolina, the owner fixed a problem with a vital safety system on Unit 1 that had failed during a periodic test. However, the owner decided that identical components on Units 2 and 3 could not possibly have the same problem. NRC inspectors persistently challenged lame excuse after lame excuse until the company finally agreed to test the other two units. When it did so, their systems failed, and NRC inspectors ensured that the company corrected the problems.

Poor NRC Oversight

However, the NRC did not always serve the public well in 2010. This report analyzes serious safety problems at Peach Bottom, Indian Point, and Vermont Yankee that the NRC overlooked or dismissed. At Indian Point, for example, the NRC discovered that the liner of a refueling cavity at Unit 2 has been leaking since at least 1993. By allowing this reactor to continue operating with equipment that cannot perform its only safety function, the NRC is putting people living around Indian Point at elevated and undue risk.

The NRC audits only about 5 percent of activities at nuclear plants each year. Because its spotlight is more like a strobe light—providing brief, narrow glimpses into plant conditions—the NRC must focus on the most important problem areas. Lessons from the 14 near-misses reveal how the NRC should apply its limited resources to reap the greatest returns to public safety.

Because we have not reviewed all NRC actions, the three positive and three negative examples do not represent the agency's best and worst performances in 2010. Instead, the examples highlight patterns of NRC behavior that contributed to these outcomes. 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 of nuclear power that the public deserves.

Findings

Overall, our analysis of NRC oversight of safety-related events and practices at U.S. nuclear power plants in 2010 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 harm to employees and the public.

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

- When onsite NRC inspectors discover a broken device, an erroneous test result, or a maintenance activity that does not reflect procedure, they too often focus just on that problem. Every such finding should trigger an evaluation of why an owner failed to fix a problem before NRC inspectors found it.

- The NRC can better serve the U.S. public and plant owners by emulating the persistence shown by onsite inspectors who made good catches while eliminating the indefensible lapses that led to negative outcomes.

- Four of the 14 special inspections occurred at three plants owned by Progress Energy. 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 plant are low. When the NRC finds safety problems and ensures that owners address them—as happened last year at Oconee, Browns Ferry, and Kewaunee—it keeps the risk posed by nuclear power to workers and the public as low as practical. But when the NRC tolerates unresolved safety problems—as it did last year at Peach Bottom, Indian Point, and Vermont Yankee—this lax oversight allows that risk to rise. The more owners sweep safety problems under the rug and the longer safety problems remain uncorrected, the higher the risk climbs.

While none of the safety problems in 2010 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 Three Mile Island in 1979 and Chernobyl in 1986 occurred when a handful of known problems—aggravated by a few worker miscues—transformed fairly routine events into catastrophes. That plant owners could have avoided nearly all 14 near-misses in 2010 had they corrected known deficiencies in a timely manner suggests that our luck at nuclear roulette may someday run out.

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, assaults, 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 when even minor violations occur.

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, we decided that another report chronicling only the latest examples of lax enforcement would be futile. Instead, this report—the first in an annual series on NRC performance— chronicles what the agency is doing right as well as what it is doing wrong.

The Reactor Oversight Process

When an 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 (see Table 1). In this process, the NRC's inspectors continually 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 in 2010 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 report, which they usually make public.

Both routine inspections and those of the special teams identify violations of NRC regulations. The NRC classifies these violations into five categories, with Red denoting the most serious, followed by Yellow, White, Green, and Non-Cited Violations (NCVs).

The Focus of This Report

Chapter 2 investigates all 14 "near-misses" at nuclear reactors that the NRC reported on in 2010: events that spurred the NRC 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 these events, this chapter shows how one problem led to another in more detail. The chapter then describes the "tickets" the NRC wrote for the numerous safety violations that contributed to each near-miss. Finally, the chapter suggests how the NRC can prevent plant owners from accumulating problems that will conspire to cause next year's near-misses—or worse.

This review of near-misses 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 stem from failed equipment, such as emergency diesel generators, the NRC could focus its efforts in that area until it arrests declining performance.

With these near-misses attesting to why enforcement is vital to the safety of nuclear power, the next two chapters highlight NRC performance in monitoring safety through the onsite reactor oversight process. Chapter 3 describes three occasions in which effective NRC oversight produced three positive outcomes—preventing safety problem from snowballing into even more dangerous near-misses. Chapter 4, in turn, describes three occasions in which ineffective NRC oversight failed to prevent negative outcomes.[1]

Chapter 5 summarizes findings from the near-misses in Chapter 2, the examples of positive outcomes in Chapter 3, and the examples of negative outcomes in Chapter 4. This chapter notes which oversight and enforcement strategies worked well for the NRC in 2010 and which did not. This chapter also recommends steps the agency should take to reinforce behavior patterns leading to commendable outcomes, and steps it should take to avoid condemnable outcomes.

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.

[1] The utility of the examples as models was more important than the number. Future reports may include a different number of examples.

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 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, such that workers 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 2010

In 2010, the NRC reported on 14 significant safety- and security-related events at nuclear reactors that resulted in special or augmented inspections (see Table 2). (Some of the events actually occurred in 2009, but the reports appeared in 2010.) Thirteen of these events triggered an SIT, one triggered an AIT, and none triggered an IIT.

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. Lessons from these 14 near-misses reveal how the NRC can apply its limited resources to reap the greatest returns to public safety.

Table 2: Nuclear Near-Misses in 2010

Reactor and Location	Owner	Highlights
Arkansas Nuclear One Russellville, AR	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.
Braidwood Joilet, IL	Exelon	SIT: The plant owner knew about several problems but did not correct them, leading to a near-miss. The problems included a poor design that led to repeated floods in buildings with safety equipment, a poor design that allowed vented steam to rip metal siding off containment walls, and undersized electrical fuses for vital safety equipment.
Brunswick Southport, NC	Progress Energy	SIT: Equipment failure prompted the plant owner to declare an emergency. Workers did not know how to operate the computer systems that automatically notified offsite workers to report immediately to emergency response facilities. Staffing and preparing these facilities took far longer than required.

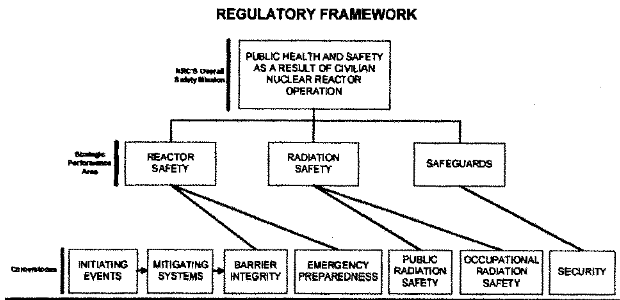
Reactor and Location	Owner	Highlights
Calvert Cliffs Annapolis, MD	Constellation Energy	SIT: A roof known for years to leak when it rained allowed rainwater to short out electrical equipment. One reactor automatically shut down. A worn-out protective device that workers had not replaced because of cost-cutting efforts allowed the electrical problem to trigger an automatic shutdown of a second reactor.
Catawba Rock Hill, SC	Duke Energy	SIT: Security problems prompted the NRC to conduct a special inspection. Details of the problems, their causes, and their fixes are not publicly available.
Crystal River 3 Crystal River, FL	Progress Energy	SIT: Workers severely damaged thick concrete reactor containment walls when they cut a hole to replace steam generators. The ensuing inquiry concluded that the workers had applied more pressure than the concrete could withstand—a mistake that cost more than $500 million.
Davis-Besse Toledo, OH	FirstEnergy	SIT: Workers discovered through-wall cracks in metal nozzles for control rod drive mechanisms in a replacement reactor vessel head. These cracks leaked because workers did not properly account for peak temperatures inside the reactor vessel.
Diablo Canyon San Luis Obispo, CA	Pacific Gas & Electric	SIT: A misguided repair to valves that would not open fast enough prevented other key valves from opening. Tests after the valve repairs failed to detect the problem. The reactor operated for nearly 18 months with vital emergency systems disabled.
Farley Dothan, AL	Southern Nuclear	SIT: A replacement pump had a part with a manufacturing defect. Excessive vibration levels caused the pump to fail when workers did not ensure that it met key parameters specified in the purchase order.
Fort Calhoun Omaha, NE	Omaha Public Power District	SIT: Pumps in an emergency water makeup system failed repeatedly over several years. The plant owner never identified the true cause of the failures, and therefore did not take the right steps to prevent their recurrence.

Reactor and Location	Owner	Highlights
HB Robinson Florence, SC	Progress Energy	AIT: On the 31st anniversary of Three Mile Island, this event revisited nearly all the problems that caused that meltdown: bad design, poor maintenance of problematic equipment, inadequate operator performance, and poor training.
HB Robinson Florence, SC	Progress Energy	SIT: The same problems (see above) caused this reactor's second near-miss in six months: bad design, nonconforming equipment, inadequate operator performance, and poor training. This baggage reflected years of programmatic failures.
Surry Newport News, VA	Dominion Generation	SIT: After an inadvertent shutdown of the Unit 1 reactor, a fire began in the control room due to an overheated electrical component. A similar component in the Unit 2 control room had overheated and started a fire six months earlier. The company did not take steps to protect Unit 1 from the problem identified in Unit 2.
Wolf Creek Burlington, KS	Wolf Creek Nuclear	SIT: Seven hours after the reactor shut down automatically because of a problem with the electrical grid, an NRC inspector found water leaking from the system that cools the emergency diesel generators and virtually all other emergency equipment. An internal study in 2007 had forecast such leakage, and a leak had actually occurred after a reactor shutdown in April 2008. However, the owner had taken few steps to correct this serious safety problem.

In 2010, SIT/AIT reports identified 40 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 2010 by Cornerstones of the Reactor Oversight Process



[illegible]	0	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0
Yellow	0	0	0	0	0	0	0
White	0	2	0	1	0	0	0
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
NCV	2	1	0	0	0	0	0
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

Source: NRC (top half of figure).

Two of the NRC's regulatory cornerstones accounted for most of the near-misses in 2010. And most near-misses drew a Green finding—the weakest color-coded sanction—from the agency. NCV = Non-Cited Violations.

The most significant near-miss occurred on March 28, 2010—coincidentally, the 31[st] anniversary of the Three Mile Island accident—at the HB Robinson nuclear plant in South Carolina. The most costly event forced the owner of the Crystal River 3 reactor in Florida to shut it down for the entire year.

Arkansas Nuclear One, AR

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 for withholding information, the SIT report on the problem(s) 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 uncovered no violations (NRC 2010k).

Braidwood, IL

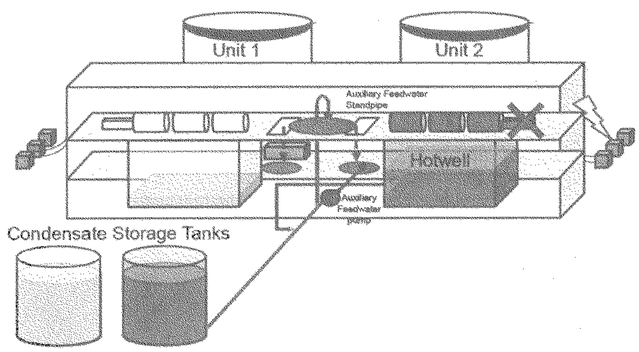
The Near-Miss

The NRC sent an SIT to the site after an unplanned shutdown of both reactors on August 16, 2010—complicated by problems with an emergency pump for Unit 2 and the steam pressure control valve for Unit 1 (NRC 2010d).

The SIT found that these complicating factors had all occurred individually at least once before, and that they combined this time to create serious risks. The NRC sanctioned the owner for having known about these problems but not correcting them. Yet the NRC also knew or should have known about them, but did nothing to compel their resolution until after this near-miss.

How the Event Unfolded

On August 16, 2010, both reactors at the Braidwood nuclear plant in Illinois were operating at full power. The Unit 2 reactor automatically shut down at 2:16 am, when an electrical ground caused the main generator to turn off. The pumps of the auxiliary feedwater (AFW) system started automatically after the reactor shutdown, to transfer water from the condensate storage tank to the steam generators.

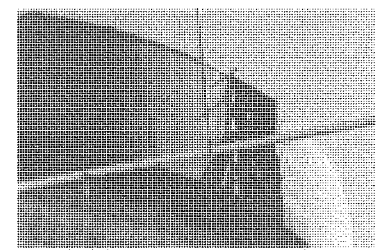


NRC drawing of the key components involved in the Braidwood near-miss. The red "X" indicates where the event started, when the main generator shut down.

However, the flow-control valve for one AFW pump failed in the open position, and the water level in the main condenser hotwell rose until valves opened to send some of this water back to the condensate tank. Nearly 12,000 gallons of water spilled onto the floor of the turbine building, from open standpipes installed on the piping between the outdoor tank and the AFW pumps (NRC 2010j).

Some of the spilled water flowed through holes in the floor and rained down on equipment on lower floors. Water leaked into an electrical panel housing controls for Unit 1 equipment. Two large pumps that circulate water between a nearby river and the main condenser stopped running because of electrical shorts. The reduction in cooling water flow through the main condenser impaired the condensation of steam inside the condenser. This impairment degraded the condenser's vacuum, triggering an automatic shutdown of the Unit 1 reactor about 15 minutes after the Unit 2 reactor shut down.

After the Unit 1 reactor shut down, the main steam safety valves (MSSVs) automatically opened to relieve pressure in the piping carrying steam from the steam generators to the main turbine. One MSSV stuck open after pressure dropped back below the opening set points. The operators did not realize that the MSSV was open until a worker arriving at the site 40 minutes later told them. Meanwhile steam passing through this open value dislodged sheet-metal siding around the top of the Unit 1 containment building. Some of the siding landed on power lines for the Unit 1 off-site power transformer.



NRC photo of the metal siding torn from the containment building at the Braidwood nuclear power plant in Illinois.

Although two large circirculating water pumps for Unit 1 had shut down because of electrical shorts, other pumps continued to run. These pumps sit in a concrete structure on the banks of the nearby river. The piping on the discharge of each pump contains a valve that closes when the pump is not running, to prevent backflow. However, the loss of electrical power that shut down the pumps also prevented their motor-operated valves from closing. Water flowing back through the idle pumps stirred up organic growth and debris. The pumps carried this material into the piping of the service water system, which supplies cooling water to essential plant equipment. The debris impaired but did not disable the system and the equipment it supported.

A second spill then complicated the Unit 1 reactor shutdown. The seal on a condensate booster pump failed, allowing water to spray onto another electrical panel. Operators stopped the pump and closed its valves to isolate the leak.

NRC Sanctions

The SIT identified two violations of regulatory requirements of the ROP's *initiating events* cornerstone. The first violation involved the failure to correct the condition that allowed water to spill onto the turbine building floor. Operators had observed such spills several times before, but had evaluated them only from a worker safety perspective.

The second violation involved failure to properly evaluate operating experience. Workers had evaluated an event at another nuclear plant where steam had dislodged metal siding, and had concluded that it did not apply to Braidwood. They failed to evaluate a previous event at Braidwood in which steam had dislodged metal siding. The NRC classified both violations as Green—the least serious of the color-coded violations.

The SIT identified two other violations of requirements associated with the *mitigating systems* cornerstone. The first involved a failure to properly inspect and clean the pump intake structure, to prevent fouling that could disable the essential service water system.

The second violation involved inadequate corrective actions. In 2008, workers had found that they needed to replace 1.5-amp fuses in safety-related electrical panels with 3.0-amp fuses. However, the workers did not do so, and the fuses failed in 2009. After the failures, workers replaced the blown fuses with 1.5-amp fuses, and these failed again during the August 2010 event. The NRC classified both violations as Green.

Brunswick, NC

The Near-Miss

The NRC sent an SIT to the site after the inadvertent discharge of Halon gas—a fire suppression agent—on June 6, 2010, into the basement of the building housing the emergency diesel generator. The release of the toxic gas into a vital area prompted control room operators to declare an Alert—the third-most-serious of four emergency classifications. The SIT investigated delayed responses to the emergency declaration.

The SIT found that workers did not know how to activate the computer systems that automatically notified emergency responders, so the responders took longer than required to staff emergency facilities. Luckily, this event was not an actual emergency, or the delay could have put people in harm's way.

How the Event Unfolded

On June 6, operators declared an Alert at 11:37 am, after Halon discharged into the building housing the plant's emergency diesel generator. Halon extinguishes fires by reducing the concentration of oxygen in the air. In this case, no fire had occurred, and the Halon discharge was spurious. While the Halon discharge was inadvertent, it prevented ready access to the diesel generator building. This restriction prompted the Alert declaration.

The Alert should have prompted operators to activate three onsite emergency response facilities within 75 minutes: the Technical Support Center, the Operations Support Center, and the Emergency Operations Facility. Specialists at the Technical Support Center help control room operators diagnose problems and take steps to mitigate them.

Specialists at the Operations Support Center help repair broken or malfunctioning safety equipment. Specialists at the Emergency Operations Facility liaise with local, state, and federal officials responding to the emergency. The Alert is also supposed to activate an emergency response data system (ERDS) within 60 minutes, which provides continuous, real-time information on conditions at the plant to local, state, and federal authorities. These activations all occurred late.

Twenty-five minutes after the Alert declaration, the control room site emergency coordinator (CR-SEC) notified the plant's security department to initiate the emergency callout system, which notifies off-duty personnel to report to their assigned emergency response facilities promptly. Security personnel made five unsuccessful attempts to initiate the callout system, and then informed the CR-SEC that they were unable to do so. The CR-SEC then directed the control room emergency communicator to initiate the callout, who made three unsuccessful attempts.

An hour after workers declared the Alert, an emergency preparedness supervisor initiated the callout from home on the first attempt, and off-duty personnel began receiving notification to report to the plant because of an emergency. Two hours and thirty minutes after operators declared the Alert, onsite emergency response facilities were fully staffed and activated. That response time was twice as long as specified in the plant's emergency response procedures.

The CR-SEC directed the shift technical advisor (STA) to activate the ERDS 28 minutes after the Alert declaration. After several unsuccessful attempts, the STA contacted the on-call nuclear information technologist (NIT) for help in activating the ERDS. The NIT did not know how to do so, but contacted another NIT who did. The second NIT initiated the ERDS from home on the first attempt—80 minutes after operators had declared the Alert. That was 20 minutes longer than specified in the plant's emergency response procedures (NRC 2010g).

NRC Sanctions

The SIT identified two violations of regulatory requirements associated with the ROP's *emergency preparedness* cornerstone. The first violation involved the failure to activate the onsite emergency response facilities within 75 minutes, as specified in the plant's emergency response procedures. The NRC classified that violation as White—one step up from Green (NRC 2010a).

The second violation involved the failure to activate the emergency response data system within 60 minutes, as specified in the plant's emergency response procedures. The NRC classified that violation as Green.

Calvert Cliffs, MD

The Near-Miss

The NRC sent an SIT to the site after an unplanned shutdown of both reactors on February 18, 2010 (NRC 2010s). The SIT determined that two factors had complicated this event. One was the longstanding flow of rainwater through a leaky roof. The second was a problem created by the plant's replacement program for safety equipment.

The plant owner had originally replaced devices on safety equipment before they reached the end of their service life. To save money, the company decided to test the performance of the devices rather than replacing them automatically. However, the company stopped the routine replacement program before instituting the new regime for testing actual conditions. As a result, a worn-out device failed to prevent electrical problems caused by rainwater from propagating throughout the plant.

How the Event Unfolded

This event began when water leaking through the roof of Unit 1's auxiliary building caused an electrical short that shut down one of the four large pumps circulating water through the reactor core. The reduced flow of cooling water triggered the Unit 1 reactor to shut down automatically.

The failure of an electrical protection device on Unit 1 then created an overcurrent condition in Unit 2's power distribution system. In response, an electrical protection device on Unit 2 shut down all four pumps circulating water through the reactor core, and the loss of cooling water triggered the automatic shutdown of the Unit 2 reactor.

The problems with the power distribution system prompted emergency diesel generators for both reactors to start automatically. However, an emergency generator for Unit 2 shut down after only 15 seconds, because of a signal indicating low lubricating oil pressure. Loss of that emergency diesel generator de-energized equipment needed by the operators to control the water level in the pressurizers.

The pressurizers are large tanks partially filled with water that are connected to the pipes running between the reactor vessel and the steam generators. By heating or cooling the water inside the pressurizer, the operators can control the pressure of the water flowing through the reactor core. The pressurizer also accommodates the swelling and shrinking of water caused by temperature changes during changes in reactor power.

To supplement the pressurizer's ability to handle the expansion of water during temperature increases, water can be removed from the system via drain pipes called letdown paths. The SIT discovered that procedural problems prevented the operators from restoring the letdown paths in time to prevent water levels in the pressurizers from exceeding their safety limits.

The power distribution problems at Unit 2 eliminated the normal means of removing decay heat from the reactor core after shutdown. Operators relied instead on the turbine-driven auxiliary feedwater pump, and atmospheric steam dump valves, to remove decay heat.

The SIT found that roof leakage had been a recurring problem since 2002, and that the company knowingly tolerated it. For example, in 2005 plant workers noted 33 roof leaks. When rainfall leaked through the roof in July 2008, workers notified control room operators and mopped up the puddle. In August 2009, workers responded to water leaking through the roof onto an electrical panel by covering the panel with a plastic sheet and catching the leakage in a bucket. The plant owner discussed corrective actions but never took them.

The SIT reported that the company attributed the failure of the electrical protection device to premature aging of its coil. The device had a 40-year service lifetime but failed after 39 years, because high temperatures aged it more rapidly. The SIT discovered that 68 devices at Calvert Cliffs had a 10 percent failure rate between 1999 and 2005, and that the owner's calibration and inspection procedures lacked common industry practices specified in a manual from the Electric Power Research Institute.

The SIT determined that Unit 2's emergency diesel generator did not run because of a failed time-delay relay. The relay prevents a shutdown stemming from low oil pressure until the pressure has first risen to the normal operating range after the emergency generator has started.

On February 18 the relay timed out too soon, shutting down the emergency generator. The SIT found that the failed relay had been in service for 3.5 years longer than the 10-year service lifetime recommended by the vendor. In 2001, the company had discontinued the practice of replacing the relays after 10 years of service. The owner substituted a performance-monitoring program for about 100 relays with safety functions, and more than 500 relays with non-safety functions. However, the owner had never developed the monitoring program, much less implemented it.

NRC Sanctions

The SIT documented two violations of regulatory requirements associated with the ROP's *initiating events* cornerstone. The first involved the company's failure to respond to recurring roof leakage with timely and effective corrective action. The second violation involved failure to properly evaluate and correct degraded electrical protection devices. The NRC classified both violations as Green.

The SIT also identified three violations of regulatory requirements associated with the *mitigating systems* cornerstone. The first violation involved failure to implement a preventive maintenance program for electrical relays with safety functions. The second violation involved failure to properly evaluate and correct recurring binding and sticking problems with electrical protective devices.

The third violation involved failure to establish procedures for restoring the primary system's letdown flow function. The NRC classified the first violation as White, and the remaining two violations as Green.

Catawba, SC

The Near-Miss

The NRC sent an SIT to the site in response to security-related problems. Reflecting post-9/11 procedures, the SIT report explaining 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 NRC identified one Green violation (NRC 2010r).

Crystal River Unit 3, FL

The Near-Miss

The NRC sent an SIT to the site after discovery of a gap in the concrete containment walls on October 2, 2009, near an opening cut to allow workers to replace the steam generators (NRC 2010h).

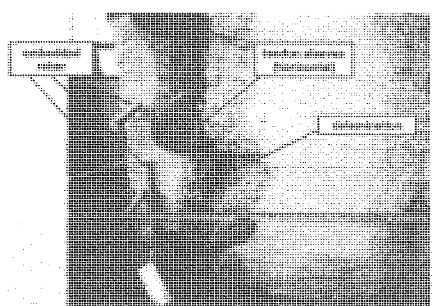
The SIT found that the method used to cut through the thick concrete walls created so much pressure that thick metal reinforcing bars in the walls acted like the San Andreas fault. The SIT's computer simulations showed that the outer half of the walls had separated from the inner half along the reinforcing bars.

This finding raises several questions: Why didn't the company do such homework before embarking on this ill-fated experiment, and why did the NRC allow it to happen? Even more fundamentally, why did the owner design and build a massive structure with doors smaller than the equipment it houses, given the potential need to replace the equipment?

How the Event Unfolded

The pressurized water reactor (PWR) at Crystal River Unit 3 (CR3) has large heat exchangers, called steam generators. Water heated to nearly 600° F in the reactor core flows through thousands of thin metal tubes in the steam generators. This water is maintained at high pressure to keep it from boiling. Heat conducted through the walls of the tubes boils water at lower pressure outside the tubes. The resulting steam is piped to the turbine to generate electricity.

When originally installed inside the reactor containment structure in the 1970s, the steam generators were expected to last the plant's entire operating lifetime. However, corrosion, vibration-induced wear, and stress cracking degraded the generators' thin metal tubes. Thus, work performed during a scheduled refueling outage in September 2009 included replacing the steam generators. Because they were larger than the equipment hatch for the reactor containment building, workers had to cut a 25-by-27-foot opening through the 42-inch-thick containment wall to get the old steam generators out and the new ones in.



NRC picture of the crack (delamination) in the concrete containment wall at Crystal River 3 caused when workers cut a square opening to replace the steam generators.

CR3's dome-shaped reactor containment structure is lined with a 3/8-inch layer of steel, reinforced with 282 horizontal 5-inch-thick metal cables, called tendons, and 144 vertical tendons embedded inside the concrete. The tendons are stretched, or tensioned, to strengthen the containment structure.

The SIT found that workers had loosened 10 vertical and 17 horizontal tendons where they planned to cut through the containment walls, and had then used a high-pressure jet of water to make the cut. A significant crack in the concrete running vertically between the horizontal tendons then appeared. Further investigation revealed a 60-by-82-foot hourglass-shaped delamination around the opening.

The SIT confirmed that the containment structure had been intact while the reactor operated, and concurred with the owner that seven factors had combined to produce more force than the concrete could withstand. Fortunately, the delamination occurred during an outage, when safety did not require integrity of the containment walls.

NRC Sanctions

The SIT identified no violations of NRC requirements. From a regulatory perspective, damaging the reactor's containment building is perfectly acceptable if the reactor is not operating, and it is not restarted until the building is fixed. The CR3 reactor remained shut down for more than a year—punishment enough for this miscue.

Davis-Besse, OH

The Near-Miss

The NRC sent an SIT to the site after the discovery on March 12, 2010, of cracks in nozzles on the control rod drive mechanism (CRDM) that had penetrated through the head of the reactor vessel. Borated reactor cooling water leaked through some of the cracks (NRC 2010f).

This situation was déjà vu all over again, as an SIT had visited Davis-Besse in 2002 after a cracked and leaking CRDM nozzle caused extensive damage to the reactor vessel head. After replacing the damaged head and correcting numerous other safety problems, operators had restarted the reactor in March 2004.

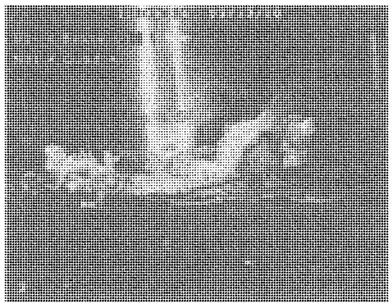
That episode had revealed that higher temperatures in the CRDM nozzles create more stress, allowing cracks to form and hastening their propagation. Despite that finding, the 2010 SIT learned that workers did not accurately track temperatures inside the reactor vessel, assuming instead that they were the same as the temperature of the water leaving the vessel. However, some temperatures inside the vessel were nearly 7° F higher.

Given that the water is at about 600° F, this error may seem minor. However, those seven degrees are the difference between detecting cracks in the CRDM nozzles before they leak and experiencing a déjà vu moment.

How the Event Unfolded

The March 2001 discovery of similar cracking and leakage at the Oconee nuclear plant in South Carolina prompted the NRC to require more extensive inspections of CRDM nozzles. The nozzles are four-inch-diameter hollow metal tubes that penetrate through the six-inch-thick steel heads atop the reactor pressure vessel. The nozzles connect the control rods used to regulate the power level of the reactor core to electric motors on a platform above the reactor vessel head.

When workers performed Oconee-inspired inspections at Davis-Besse in March 2002, they found extensive cracking in the nozzles, and that leaking borated water had significantly degraded the reactor vessel head. Workers replaced this damaged head with one from the closed Midland nuclear plant in Michigan, and restarted Davis-Besse in March 2004. Inspections of the CRDM nozzles during refueling outages in 2006 and 2008 revealed no evidence of leakage.



White-crystalline boric acid leaked through a cracked nozzle in the head of the reactor vessel at the Davis-Besse plant in Ohio. NRC photo.

However, inspections during the March 2010 refueling outage revealed that two cracked CRDM nozzles had leaked borated reactor cooling water, and that many other nozzles had apparent cracks. Although the reactor vessel head did not need repair or replacement, workers repaired 24 of the 69 CRDM nozzles.

The SIT identified three violations of regulatory requirements associated with the ROP's *initiating events* cornerstone. The first involved workers' failure to control water rinse time after applying a liquid dye penetrant to the CRDM nozzles and welds. The penetrant makes cracks more apparent during a visual inspection. The uncontrolled rinse time could have allowed the penetrant to wash away before the inspection.

The second violation cited control room operators for failing to provide specific guidance to ensure that workers examined the entire affected area on camera. The third violation involved a defective welding process used to repair one of the two leaking CRDM nozzles. The procedure failed to control temperature during the welding process. Welding temperature is important to ensuring high-quality results: too low a temperature can allow the metal to cool before strong bonds form, while too high a temperature can damage the metal. The NRC classified all three violations as Green.

Diablo Canyon Unit 2, CA

The Near-Miss

The NRC sent an SIT to the site after operators could not open valves that provide emergency cooling water to the reactor core and containment vessel during a test on October 22, 2009 (NRC 2010x).

The SIT found that a misguided fix of an earlier problem had caused this even larger problem. When the valves failed to open and close within specified time limits, workers shortened their "travel distance." The workers did not realize that this meant that these valves no longer reached their finish lines. Interlocks prevented other safety valves from opening until the first valves were fully open. The NRC sanctioned the company for a bad "fix," and for inadequate post-fix testing that should have identified the unintended consequences but failed to do so.

How the Event Unfolded

In July 2005, workers became aware that motors for valves that provide emergency cooling water to the reactor core could not move against pressure inside the cooling system's pipes under certain accident conditions. In October, workers revised the emergency operating procedure to have control room operators establish cooling water flow within 30 minutes of an accident, to reduce pressure on the valves. However, operators needed to ensure that the valves would function under all credible accident conditions.

In February 2008, therefore, workers changed the gear ratios on the motors for the valves, to enable them to move against any pressures that might occur. The workers then tested the valves to verify that they could move from fully closed to fully open in 25 seconds or less, as required. However, the valves failed the test. To fix that problem, an engineer shortened the travel distance between the two positions, and both valves passed retests.

Eighteen months later, when operators tried to open the valves to allow pumps to provide flow inside the containment building, they would not open. That meant operators would be unable to provide cooling water to the reactor core and containment vessel at a key point during an accident.

The SIT found that three pairs of valves were interlocked, and that the first pair had to open fully before the other pairs could do so. The February 2008 modification to shorten the travel distance of the first pair meant that they stopped moving before they reached the fully open position. That is, the fix for the problem that some valves might not open when required meant that other valves definitely would not open.

NRC Sanctions

The SIT identified three violations of regulatory requirements associated with the ROP's *mitigating systems* cornerstone. The first violation involved the improperly analyzed change that shortened travel distances for the valves. The second violation involved inadequate post-modification testing of the valves. The NRC classified both violations as Green. Although the February 2008 modification impaired the emergency core cooling systems, workers could have opened the valves manually, so that mitigated the severity of the violations.

A third violation involved the October 2005 revision to emergency operating procedures that introduced a manual action into an accident response. The SIT determined that workers failed to conduct a safety evaluation to determine whether this change required NRC review and approval. The NRC classified this violation as Severity Level IV, the least serious sanction.

Farley, AL

The Near-Miss

The NRC sent an SIT to the site after a vendor notified the agency about a defective coating on a pump shaft journal (a device used to maintain the shaft alignment as it rotates at high speed), which contributed to the failure of a service water pump at Unit 2 in August 2009 (NRC 2010u).

The SIT found that the company had replaced the failed pump just three years earlier. The purchase order for the replacement pump specified key parameters, including some intended to protect it from damage caused by excessive vibration. However, the installed pump did not satisfy those parameters, and it failed after excessive vibration exacerbated the defect in the journal coating.

How the Event Unfolded

The service water system provides cooling water to safety equipment, such as emergency diesel generators, during an accident. Each of two reactors at Farley has five service water pumps. Four pumps must be available to allow each reactor to operate safely, with the fifth pump acting as a spare.

In April 2006 the company issued a purchase order for 11 service water pumps to replace the originals. Workers then replaced five of the original pumps over the ensuing three years. The first one replaced was the 2E pump on Unit 2. However, the new pump failed in August 2009, and was replaced again and sent back to the vendor for evaluation. The vendor found that a defective coating on the pump shaft's bearing journal had led to bearing damage and fracture of the wear ring.

The SIT found that purchase specs for the replacement pumps required that the critical speed of the rotor be at least 25 percent above the pump's normal speed, but that the replacement pumps failed to meet that requirement. Operating the pumps contrary to this specification increased their susceptibility to vibration, contributing to the August 2009 failure.

NRC Sanctions

The SIT identified one violation of regulatory requirements associated with the ROP's *mitigating systems* cornerstone. The violation involved the failure to ensure that service water pumps conformed to purchase specifications. The NRC classified the violation as Green.

Fort Calhoun, NE

The Near-Miss

The NRC sent an SIT to the site after the turbine-driven auxiliary feedwater (AFW) pump automatically shut down shortly after operators started the pump during a monthly test. The AFW system is an emergency system. During normal plant operation, it is in standby mode.

However, although the AFW system plays a vital role in an accident, the SIT found that the pump had failed numerous times over many years. The owner had never found the cause of the problem, and therefore had never taken steps to prevent it.

How the Event Unfolded

On February 17, 2010, workers manually started the turbine-driven AFW pump, to test whether it could deliver the required flow of water within the time frame assumed in safety studies for the plant. The pump automatically shut down shortly after it started because of high pressure in the turbine's exhaust. When pressure in the exhaust line rises to nearly 10 times normal, a piston unlatches a trip lever, which shuts down the turbine.

There were no indications that pressure in the turbine exhaust line had actually exceeded the normal range during the test. This prompted workers to check the calibration and functioning of the device that triggers the automatic shutdown. They found nothing wrong with the calibration, but they did observe that minor bumping of the equipment unlatched the trip lever. When they tried to start the AFW pump with the trip lever already unlatched, it soon shut down, just as it had during the February 17 test. The company responded by restricting access to the area around the trip device, and by requiring shift managers to brief workers needing access to that area before entry.

The SIT identified four violations of regulatory requirements associated with the ROP's *mitigating systems* cornerstone. The first violation involved five instances where workers bumped the AFW and the pressure trip lever had unlatched, preventing the pump from starting when required. The second violation involved the company's failure to develop procedures to verify that the trip device for the AFW pump was properly latched.

The third violation involved an inadequate procedure in which workers did not properly vent air from the oil system for the AFW pump control after maintenance. As a result, the AFW pump failed to start during a test on February 26, 2009.

The fourth violation involved failure to properly translate information in the plant's design into its equipment, which led to the automatic shutdown of the AFW pump during a test on April 6, 2009. The NRC classified all four violations as Green (NRC 2010n).

HB Robinson, SC

The Near-Miss

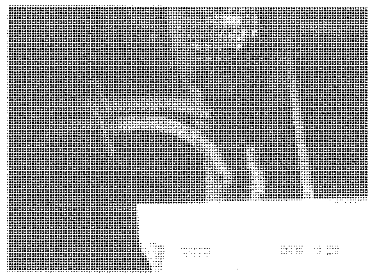
The NRC sent an SIT to the site to investigate electrical fires, which had caused an unplanned reactor shutdown and declaration of an Alert—the third-most-serious emergency classification—on March 28, 2010. The SIT found so many problems that the NRC upgraded it to an AIT after a few days (NRC 2010q).

The AIT documented numerous problems in many areas—including design and procurement of safety equipment, maintenance, operations, and training—over many years. There is simply no excuse for the fact that the company and the NRC had not detected and corrected at least some of these problems before this event.

How the Event Unfolded

The event began when a 4,160-volt electrical cable shorted out and started a fire. An electrical breaker designed to automatically open and deenergize power to the shorted cable failed to do so.

The failed electrical breaker allowed electricity to flow from a circuit through the shorted cable into the ground, reducing the circuit's voltage. This circuit powered a large motor-driven pump circulating water through the reactor core, among other components. As the circuit's power dropped, the pump's output also dropped low enough to trigger the reactor to shut down automatically.



Conduit for electrical cables damaged by the fire at the HB Robinson plant. NRC photo.

The electrical problems damaged the main power transformer between the plant and its electrical grid. When the reactor shuts down, this transformer usually allows the electrical grid to supply power to the plant's equipment. However, the damage to this transformer meant that another transformer had to provide the sole connection to the electrical grid. Other electrical breakers opened to isolate the faulted cable. This stabilized the plant's electrical conditions, but left roughly half of its equipment without power.

The equipment without power included valves on drain lines from the main steam lines. Although these valves normally close when a reactor shuts down, they opened fully on loss of power, as designed. That meant that heat escaped from the reactor more rapidly than normal, exceeding the cool down safety limit of 100° F per hour. The large reactor vessel and its piping have strict limits on how fast they can heat up or cool down to prevent thermal stress from cracking the metal. The operators did not notice the open drain valves or abnormally fast cool down. Another power failure 33 minutes later closed the drain valves.

The electrical problems interrupted the supply of cooling water to the pump seals for the reactor coolant system. When seals are damaged by overheating, cooling water leaks into the containment building. Control room operators did not notice the lack of cooling for more than 30 minutes.

After the reactor shut down, the operators started two pumps that transferred water from a tank in the auxiliary building to the reactor vessel. When this tank emptied, the pumps were supposed to automatically realign to obtain water from the refueling water storage tank. This realignment failed to happen. The operators did not notice this failure for nearly an hour.

About four hours into the event, the operators attempted to restore power to the de-energized circuit, but they did not check first to ensure that workers had fixed the original fault—and they had not. When the operators closed the electrical breaker to repower the circuit, they reenergized the shorted cable, and it caused another fire. The electrical disturbance also triggered alarms on both sets of station batteries, prompting the operators to declare an emergency Alert.

The AIT documented an incredibly long series of mistakes that first caused this event and then made it more severe. For example, the cable that started the first fire, installed in 1986, did not meet several parameters specified in the plant design. The design called for providing coated copper conductors for the cable, but it had uncoated conductors. The design also called for an outer jacket on the cable, but it did not have one. And finally, the design called for insulating the cable with self-extinguishing and non-propagating material. However, rather than extinguishing when the cable was de-energized, the fire actually spread along its length.

The non-conforming cable was connected to an electrical breaker that was supposed to open if the cable failed to isolate the problem. But with the breaker closed, a light bulb thought to indicate that the breaker was closed would not illuminate. Workers had replaced the bad light bulb in November 2008, but the new bulb also failed to illuminate. These workers thought that meant the bulb was good but the socket was bad, so they requested that other workers repair it. The second group of workers never made the trip, thinking it merely concerned an annoying problem with an unnecessary light bulb. But that bulb, when lit, actually indicated that control power was available to automatically open the electrical breaker. With the bulb not lit, the electrical breaker did not open.

Control room operators joined this error-fest with errors of omission and commission. First, they failed to stay aware of key plant parameters. For example, they did not note that the cool down rate of the reactor coolant exceeded the safety limit of 100° F per hour. Second, as noted, they failed to ensure that workers had corrected the original electrical fault before reenergizing the electrical circuits. Because the problem remained uncorrected, their misguided actions started another fire.

NRC Sanctions

The AIT identified 14 unresolved problems (NRC 2010e; NRC 2010i). Follow-up reports documented resolution of these problems. The NRC also identified six violations associated with the ROP's *initiating events* cornerstone:

- One violation involved a deficiency in the systems approach to training. This training weakness manifested itself in the operators' failure to mitigate a loss of cooling water to the seals on reactor coolant pumps during this event.

- A related violation involved the company's failure to develop emergency procedures to guide operators in ensuring cooling of the seals of the reactor coolant pump.

- One violation involved inadequate work and post-maintenance testing that prevented the cbarging pump from automatically switching from the volume-control tank to the refueling water storage tank.

- One violation involved inadequate design control that enabled installation of an out-of-specification electrical cable. Failure of this cable initiated the March 2010 fire.

- One violation involved inadequate configuration of the control room simulator: Some valves modeled in the simulator behaved exactly opposite to those in the actual plant after a loss of electrical power. Operators received misleading training in how to handle this scenario.

- One violation involved inadequate corrective actions for a degraded control power condition for an electrical breaker, which prevented it from opening when required to isolate an electrical fault during the March 2010 event.

The NRC classified four violations as Green, and deferred classification of the other two.

The NRC also identified two violations of regulatory requirements associated with the ROP's *mitigating systems* cornerstone. The first involved inadequate corrective actions for a degraded condition on the output breaker for emergency diesel generator B. A stuck control relay link caused the emergency diesel generator to fail in October 2008, and again in April 2009, before workers identified and corrected the problem.

The second violation involved the failure to provide the NRC with complete and accurate information on the problem with the breaker for the emergency diesel generator. The plant owner informed the NRC, in writing, that certain diagnostic and testing activities had been performed when in fact they had not. The NRC classified the first violation as being preliminarily White, and deferred classification of the second violation.

HB Robinson, SC

The Near-Miss

The NRC sent an SIT to the site after an automatic shutdown of the reactor on October 7, 2010, followed by equipment failures and operator miscues (NRC 2010b). This was the second near-miss at Robinson in six months (see the preceding case).

The SIT found many of the same shortcomings that had played a role in the earlier near-miss: bad design, nonconforming parts, inadequate operator performance, and poor training. The SIT should not have been surprised: an owner cannot correct years of programmatic deficiencies overnight.

How the Event Unfolded

The problems began shortly after midnight, when one of four pumps that supply cooling water to the reactor vessel experienced a motor failure and automatically shut down. That shutdown, in turn, triggered an automatic shutdown of the reactor and main turbine, per the plant design. One of the two feedwater pumps normally supplying makeup water to the steam generators also shut down automatically.

About a minute after the reactor shut down, relief valves opened in the steam system to protect piping and components from damage caused by excessive pressure. The shutdown of the turbine stopped steam from entering it. The steam vented directly into the turbine building, where its high temperature triggered the fire protection system for the main turbine's lubricating oil system. Water began spraying inside the turbine building to extinguish a nonexistent fire. About a minute later, two-inch piping in the fire protection system ruptured, adding to the flooding. Workers dispatched to the turbine building manually closed valves within 10 minutes, stopping the water flow.

About 11 minutes after the reactor shutdown, the second feedwater pump supplying makeup water to the steam generators automatically shut down because of high water level in the steam generators. The auxiliary feedwater (AFW) system—a backup to the normal system—had started after the trip of the first feedwater pump, and continued to provide makeup water.

Concerned that continued reliance on the AFW system rather than the normal feedwater system might prompt the NRC to issue a Red violation, the operators attempted to restart one of the normal feedwater pumps about four hours after the reactor shut down. Although they restarted the pump, it automatically shut down right away because they had improperly reset the parameters that had caused it to shut down in the first place. Not understanding the normal feedwater system, the operators gave up trying to restore it.

About 10 hours after the reactor shut down, day-shift operators tried to restart one of the normal feedwater pumps. They succeeded in doing so, but only because they improperly defeated safety interlocks. That meant they operated without required safety protection for the next 3 hours and 11 minutes. After realizing this mistake, the operators restarted the AFW system and reinserted the safety interlocks. About 30 minutes later, the operators successfully restarted the normal feedwater pump with safety interlocks.

NRC Sanctions

The SIT determined that the motor failure that initiated this event had stemmed from age-related degradation of the insulation on the motor winding. The reactor owner had been aware of this problem, and developed a plan in 2003 to deal with it. However, the motor that failed on October 7 had not yet been fixed.

The SIT determined that operators' procedures and training did not allow them to recover from the automatic reactor shutdown. They had encountered similar problems in trying to recover from the automatic shutdown six months earlier.

The SIT also determined that the fire protection system for the lubricating oil system for the main turbine had started up because steam vented into the turbine building after the turbine shut down falsely simulated a fire condition. Events at the plant on May 15, 2007, and November 6, 2009, had shown that this would occur, but the company had done nothing to correct the problem. In response to this event, workers installed piping to carry steam vented from the relief valves outside the turbine building.

The SIT determined that the pipe in the fire protection system ruptured because workers had improperly welded two different types of metal together. This failure reinforced the large inventory of information showing that welding two different materials together simply does not work.

The SIT identified two violations of regulatory requirements associated with the ROP's *mitigating systems* cornerstone. The first involved the violation of safety requirements when day-shift operators improperly bypassed safety interlocks to restart a pump in the normal feedwater system.

The second violation involved regulations requiring owners to correct known deficiencies in equipment in a timely manner. Specifically, the owner knew that steam vented after turbine shutdowns inadvertently initiated the fire protection system in the turbine building, but had done nothing to correct it. The NRC classified both violations as Green.

Surry, VA

The Near-Miss

The NRC sent a SIT to the site after a loss of power to instrumentation caused the Unit 1 reactor to shut down automatically on June 8, 2010, with ensuing complications (NRC 2010l).

The SIT found that an overheated electrical device had started a fire in the Unit 1 control room about 90 minutes after the reactor shut down. A similar device had overheated and started a fire in the Unit 2 control room the previous November. The NRC sanctioned the company for not taking steps to prevent a fire at Unit 1 that it had taken to prevent another fire at Unit 2.

How the Event Unfolded

The event began when workers removed one of two power supplies to an electrical bus service—an electrical connection—for planned maintenance. The electrical bus powered circuits controlling plant equipment, as well as devices for monitoring them.

During the maintenance, a worker dropped a tool, causing an electrical short that disabled the remaining power supply to the electrical bus. That, in turn, caused various valves in the feedwater system to either lock up or fully open. The result was an imbalance between the amount of steam flowing from the steam generators and the amount of water supplied to the steam generators by the feedwater system. Less than 90 seconds later, low water level in one steam generator triggered the automatic shutdown of the reactor and the turbine.

The imbalance also triggered two standby emergency pumps to begin supplying makeup water to the reactor vessel. This measure was precautionary, as no piping had ruptured, and the reactor vessel was not losing water. About 20 minutes later, the unnecessary makeup water increased pressure in the reactor vessel to the point where a relief valve opened automatically, to protect the system. That relief valve opened and closed 14 times during the next 20 minutes. A similar relief valve, which stuck open the first time it opened, contributed to the partial meltdown of the Unit 2 reactor core at Three Mile Island in March 1979.

About 90 minutes after the reactor shut down, overheated electrical resistor/capacitor (RC) filters inside a control room cabinet caught fire. The operators put out the fire within three minutes. Shortly afterward, electrical fuses blew to de-energize some instrumentation monitoring key plant parameters. The operators restored power within minutes.

NRC Sanctions

The SIT learned that overheated RC filters had caused a fire in a control room cabinet at Unit 2 in November 2009. After putting out the fire and replacing the scorched filter, workers wrote a condition report asking technicians to investigate why the RC filter had overheated. However, the company closed the condition report without any investigation or evaluation. After the similar fire in Unit 1, workers tested all the RC filters in cabinets in both control rooms. They found many in a degraded condition, including some that produced visible electrical sparks during testing. Workers replaced all RC filters in all applicable cabinets.

The SIT identified one violation of regulatory requirements associated with the ROP's *initiating events* cornerstone. The violation involved failure to correct degraded RC filters in Unit 1 instrumentation cabinets after discovery of the same situation at Unit 2. The NRC classified the violation as Green.

Wolf Creek, KS

The Near-Miss

The NRC sent an SIT to the site after a nearby lightning strike on August 19, 2009, disconnected the plant from the electrical grid. The reactor and turbine automatically shut down in response, as designed. Onsite emergency diesel generators started automatically, to provide electrical power to essential safety equipment. Essential service water (ESW) pumps also started automatically. However, a pressure spike in the ESW system after the pumps started created a 3/8-inch-diameter hole in the piping. The SIT investigated the loss of offsite power and the ensuing damage to the ESW system (NRC 2010y).

The SIT found that a 2007 internal study had forecast leakage in the ESW piping, and that leakage had actually occurred in April 2008 in an event similar to that in August 2009. The NRC sanctioned the company for having identified this safety problem but having failed to correct it.

How the Event Unfolded

The SIT found that Wolf Creek personnel had little responsibility for the plant's electrical switchyard. Most responsibility rested with Westar Energy, an independent electricity provider. This division of responsibility meant that workers at Wolf Creek did not enter all switchyard-related problems into the plant's corrective action program, which determines the root causes of equipment failures and proper fixes.

For example, one or more transmission lines between the plant and the electrical grid had failed 31 times since 2004, but workers had not entered 20 percent of those failures into the corrective action program. The SIT also learned that when Wolf Creek workers received accounts of switchyard problems at other nuclear facilities, they did not effectively communicate that information to Westar Energy. The plant was therefore more vulnerable to offsite power interruptions than necessary.

The loss of offsite power triggered several fire protection alarms. Plant procedures called for workers to monitor areas triggering the alarms, to compensate for the disabling of automatic fire detection and suppression circuits owing to the loss of power. NRC inspectors discovered that more than a dozen areas lacked the required fire watches.

The plant's response to the loss of offsite power, and the resulting rupture in the ESW piping, led to a sizable leak in the auxiliary building—discovered by an NRC inspector seven hours later. During an accident or a loss of offsite power, this plant's ESW system draws water from a nearby lake for numerous cooling systems, including one used to remove heat from the reactor core and containment.

The SIT found that similar leakage in ESW system piping had occurred after another loss of offsite power in April 2008. The SIT concluded that the company's evaluations after these two events were too narrow to determine the causes and consequences of the problem. Specifically, the SIT found that the company had not adequately evaluated the damage caused by internal corrosion of ESW system piping and components.

The SIT also found that a 2007 assessment of the ESW system found that lake water was causing pitting and other corrosion. The study recommended better chemistry control and monitoring measures to prevent damage. However, managers opted to delay "repairs until such degradations (pitting) had become through-wall leaks" (NRC 2010y).

NRC Sanctions

The SIT documented two violations of regulatory requirements associated with the ROP's *initiating events* cornerstone. One violation involved the failure to enter electrical switchyard problems into the corrective action program. The second violation involved failure by the operators to control the water level in the steam generator after the reactor shut down. The NRC classified both violations as Green.

The SIT identified five other violations of regulatory requirements associated with the ROP's *mitigating systems* cornerstone. The first involved the failure to assess the impact of the through-wall leaks caused by internal corrosion of ESW piping on the system's operability.

The second violation involved inadequate corrective action following damage to ESW piping after the loss of offsite power in April 2008. The third violation involved inadequate corrective action related to the corrosion problems identified by the ESW assessment in 2007.

The fourth violation involved failure to develop and implement needed procedures. Wolf Creek required operators to visually examine systems subject to water-hammer forces during electrical events for structural damage. However, the company did not include the ESW system in such inspections, despite the fact that a water hammer after the loss of offsite power in April 2008 damaged ESW piping and components.

The fifth violation involved a violation of the plant's operating license reflected in the inadequate response to fire protection alarms. The NRC classified all five violations as Green.

Observations on Near-Misses in 2010

Nearly all 14 near-misses in 2010 resulted from known safety problems that went uncorrected. With luck, such impairments do not interact to turn a bad day into a catastrophe. However, Three Mile Island and countless other nuclear and non-nuclear technological catastrophes show what can happen when luck runs out.

Many excuses underlie owners' failures to correct these safety problems. For example, each time the roof at Calvert Cliffs leaked without serious consequences, that outcome encouraged the owner to continue to tolerate the problem rather than fixing it before luck ran out. At Surry, operators considered the electrical component that overheated and caused a fire in the Unit 2 control room an isolated failure—until the same component overheated and caused a fire in the Unit 1 control room.

At Wolf Creek, an internal 2007 study predicted through-wall corrosion of piping in the emergency cooling system, and an event when the piping actually leaked validated that prediction in April 2008. Yet the owner took inadequate steps to correct the safety problem until the piping leaked again in August 2009. None of these excuses are defensible, particularly in an industry that so often claims to place safety first.

Shortcomings in NRC Oversight

A majority of the SIT and AIT findings in 2010 fell into two of the ROP's seven cornerstones: initiating events and mitigating systems. The NRC already devotes considerable resources to these cornerstones through the efforts of its onsite inspectors. These near-misses therefore do not suggest that the agency needs to reallocate resources from other cornerstones.

However, NRC inspectors—full-time personnel at each nuclear plant, supplemented by employees from regional offices and headquarters—conduct about 6,300 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 14 near-misses?

Agency inspectors audit only about 5 percent of the activities at each plant each year. That means each device examined, each test result reviewed, and each maintenance activity witnessed represents 19 unaudited devices, tests, and activities.

Limiting audits to only 5 percent makes sense if and only if the NRC views the findings as insights into the bigger picture. Instead, the agency treats them as if they stem from 100 percent, full-scope audits. When inspectors find a broken device, an erroneous test result, or a maintenance activity that does not reflect procedure, they simply require companies to fix the device, correct the problem and rerun the test, or perform the maintenance activity correctly.

The NRC simply cannot be an effective regulator if it continues to treat limited-scope audits as full-scope audits. Instead, every NRC finding should trigger a formal evaluation of why an owner failed to fix a problem before NRC inspectors found it. Such an evaluation would answer questions such as:

- Did plant workers identify the device as broken?
  - If so, did they attempt to repair it?
    - If so, why wasn't the repair successful?
    - If not, was the reason for the deferral justified?

  - If workers did not identify the device as broken, why didn't the plant's tests and inspections work?
    - Are tests and inspections adequate to detect this kind of failure?
    - Do workers conduct tests and inspections often enough?

- What other devices might also be broken but undetected?

- What assurances can the owner give that uninspected devices will work?

Owners of the top-performing nuclear plants do not wait for the NRC to ask such questions: they already ask and answer them. For example, workers at the South Texas Project discovered that reactor cooling water had leaked from instrumentation lines on the bottom of a reactor in spring 2003.

To prepare for public meetings between the NRC and the owner, UCS reviewed the agency's inspection reports as well as company documents. This owner answered all our questions—plus dozens more we had not considered asking—during its own presentations at the meetings. Unfortunately, not all reactor owners back up their safety-first assertions with such solid homework. The NRC must ask the questions that the underperformers are not asking.

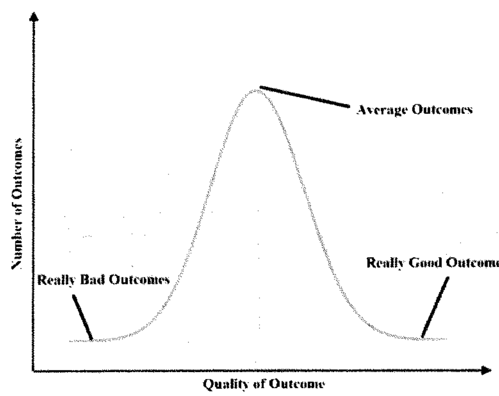
This is especially important because 4 of the 14 near-misses in 2010 occurred at reactors owned by Progress Energy. Progress owns less than 5 percent of the U.S. nuclear fleet, yet experienced more than 28 percent of the significant events that year. These near-misses occurred at three different Progress-owned sites—Robinson, Crystal River 3, and Brunswick: only one Progress site did not have a near-miss.

While these events may have nothing in common other than the same owner, the corporate hand may have played a role. Companies with multiple reactors at various sites develop fleet-wide standards and procedures intended to improve performance through the sharing of best practices. However, even good intentions can contribute to bad outcomes in the face of insufficient resources, or resistance to change among employees. The NRC should take formal, documented steps to confirm that four near-misses at three Progress Energy sites in the same year is coincidence, or identify common causes and ensure that the company eliminates them.

CHAPTER 3

POSITIVE OUTCOMES FROM NRC OVERSIGHT

This chapter describes situations where resident NRC investigators acted to bolster the safety of nuclear plants before problems spiraled into significant events that prompted the agency to send in an outside team to provide more in-depth analysis. 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 actions to make that claim. Nor are these outcomes the only positive ones the NRC achieved last year—far from it.



UCS's review focused on really good and really bad outcomes from the larger population of average NRC outcomes.

Instead, in choosing these situations, we focused on especially good outcomes. We also found two important instances in which the NRC expanded public access to agency officials and information on reactor safety. These results show that the NRC can be an effective and accessible regulator, and provide insights into how onsite investigators can emulate these results in other situations.

Oconee Letdown Flow

On October 9, 2009, workers shut down the Oconee nuclear plant in South Carolina for scheduled refueling. On October 11, they conducted a routine test to verify that the letdown line of the reactor coolant system for Unit 1 had adequate flow. The letdown line prevents the pressurizer from overfilling during an accident. If it does, the system can leak more water than the emergency makeup pump can compensate for.

No water flowed through the letdown line during the test. Workers found that gasket material from a valve had broken apart and completely clogged a filter in the line. Workers replaced the valve and cleaned the filter, and completed a successful test of the letdown flow rate before restarting Unit 1 in December (NRC 2010t).

Workers installed the same type of valves in Units 2 and 3 around the same time. However, they did not test their letdown flow rates, citing two primary reasons: (1) the degradation of the Unit 1 valve was an isolated occurrence unlikely to happen in Units 2 and 3; and (2) even if the filters in those units were blocked, control room operators could bypass them to establish a flow path. In the face of these lame excuses, resident NRC inspectors could have easily asked a few questions about the Unit 1 test results and moved on to other concerns. Instead, they peeled away the claims and found serious problems.

First, the inspectors found that the manufacturer of the failed valve had informed the plant owner in November 2009 that valves in other units were equally vulnerable to degradation. Second, the inspectors found that the alternate flow path would not be available during an accident. To create that path, workers would have had to open closed valves within the reactor containment buildings—which they could not do in the dangerous conditions existing in the wake of an accident.

On February 20, 2010, spurred by NRC inspectors, workers reduced the power level of Unit 2 to test the letdown flow rate—and found that debris from a degraded valve had indeed clogged the filter. Three days later they found the same problem in Unit 3.

The NRC issued a Yellow finding to the plant owner in August 2010—not for the failure at Unit 1, but for allowing the same degraded conditions to impair Units 2 and 3 for nearly three months after discovery of the first clogged filter (NRC 2010m). If the NRC inspectors had not taken the hard route and persisted with their questioning, Oconee Units 2 and 3 would have operated with a key safety system significantly impaired.

NRC managers supported these inspectors by issuing the Yellow finding. Had the plant owner reacted when workers first revealed the problem, the agency would not have needed to issue any sanction. And had the owner reacted sooner to pointed questioning by the inspectors, the NRC would probably have levied a lighter Green or White sanction. The Yellow finding deservedly called attention to the unsafe condition sustained for three months because of the owner's recalcitrance.

Browns Ferry Oil Leak

On July 24, 2009, workers conducted a routine test to verify the performance of the high pressure coolant injection (HPCI) system for the Unit 1 reactor at the Browns Ferry plant in Alabama. The HPCI system is an emergency system that is normally in standby mode. If an accident drains cooling water from the metal vessel housing the reactor core, the system provides makeup water to protect the core from damage caused by overheating.

During the test, an oil leak of 0.25 to 0.50 gallons per minute developed. The HPCI system uses oil pressure to regulate the position of valves that control the flow of makeup water to the reactor vessel. The plant owner initially reported this condition to the NRC as degradation that could prevent the HPCI system from fulfilling its safety function during an accident. However, the owner later retracted this report, claiming that further evaluation had revealed that the oil leak was too small to impair valve control.

However, the NRC resident inspectors at Browns Ferry asked an important question. The HPCI system operates for just minutes during a test, but might have to operate for hours during an accident. Would the oil reservoir have enough capacity to sustain the valves during that entire time? After reevaluating the situation, the owner answered no, and formally reported the problem with the HPCI system to tbe NRC.

The inspectors' efforts produced much more than a mea culpa from the plant owner. They refocused the company's workers on all the potential consequences of a degraded condition. The inspectors' efforts also produced another significant outcome. HPCI systems at other U.S. nuclear reactors also contained the part that broke at Browns Ferry, and the vendor recalled it. The ripple effect from the actions of these NRC inspectors yielded safety dividends at nuclear plants across the country.

In contrast to the Oconee case, the NRC did not issue a Yellow finding (or any finding) for the problem with the HPCI system at Browns Ferry. That is because the owner fixed the HPCI problem within hours—although the "what-if" analysis required NRC intervention and took much longer. At Oconee, the flawed what-if analysis delayed correction of safety hazards at Units 2 and 3 for months.

Kewaunee Emergency Pumps

When the reactor at the Kewaunee nuclear plant in Wisconsin is operating normally, two emergency safety injection (SI) pumps are in standby mode. If cooling water drains out of the reactor vessel because of a pipe break or other accident, these pumps automatically start to transfer cooling water from the refueling water storage tank to the reactor vessel.

However, under some conditions, the pressure inside the reactor vessel is initially higher than that created by tbe SI pumps, which prevents them from supplying water to the vessel. In that situation, if the pumps operate but water does not flow through them, the water would heat up and could damage the pumps. To protect them, a small pipe recirculates water back to the refueling water safety tank, until the pressure inside the reactor vessel drops low enough to allow the pumps to deliver the cooling water.

At Kewaunee, NRC resident inspectors found that workers were routinely closing valves in the recirculation pipes while testing the safety injection system—despite the fact that the reactor was still operating (Dominion 2010). The inspectors noted that this practice disabled both SI pumps because they share a common recirculation line. In response, the company changed the testing procedure to avoid disabling the key emergency pumps while the reactor was operating.

This was a good catch by NRC inspectors for several reasons:

- The problem occurred only during infrequent tests. The inspectors might have focused just on practices during normal operation or accidents.
- The problem reflected an atypical plant design at Kewaunee. At most plants, SI pumps have separate recirculation lines back to the refueling water safety tank. The inspectors caught a problem that they probably had not encountered in their training or other experience.
- Closing the valves during testing had been standard practice since the reactor began operating in 1973. That the problem existed for nearly 40 years testifies to its subtlety. Numerous plant workers and NRC inspectors who had reviewed the safety injection system had overlooked it.
- The SI pumps would not need the recirculation line during most accidents. If a pipe ruptures, the SI pumps automatically start when pressure inside the reactor vessel drops from about 2,235 pounds per square inch (psi) to 1,815 psi. The discharge pressure of the SI pumps is nearly 2,195 psi. Thus the pumps would typically supply makeup water immediately to the reactor vessel, without the need for the recirculation lines.

However, operators may manually start the SI pumps in response to events such as a rupture in a steam generator tube. Depending on the size of the tube, the pressure in the reactor vessel could remain close to normal long enough for SI pumps to sustain damage.

How Top NRC Officials Served the Public Interest

The NRC chair and commissioners visit several nuclear plants each year. These visits typically involve a tour of the facility and a brief presentation by the owner on plant safety. The visits also often feature updates by resident NRC inspectors on the plant's performance. The agenda may even include a press conference or a meeting with local elected officials.

Although not unprecedented, an NRC chair or commissioner rarely meets face to face with residents who live near nuclear plants, to listen to their concerns and explain what the agency is doing about them. In 2010, the NRC chair and a commissioner took the time to do just that.

NRC Chair Gregory B. Jaczko visited the Vermont Yankee nuclear plant on July 4. His visit included a 90-minute roundtable meeting with several members of the public, at which Jaczko heard their concerns and offered his views (NRC 2010p). The NRC arranged a telephone call-in so stakeholders from around the country could listen to the discussion.

Similarly, when NRC Commissioner William D. Magwood IV visited the Braidwood nuclear plant in Illinois on November 16, he met with local citizens to hear their concerns about the more than 6 million gallons of radioactively contaminated water that had leaked from the plant. One attendee told UCS that it was the most meaningful dialogue the community had had with the NRC since the leaks were first reported in late 2005.

These officials impressed members of the public by telling them exactly what they most wanted to hear—the truth. For example, Chair Jaczko shared concerns that senior NRC managers expressed to him about Vermont Yankee, and the measures they planned to address those concerns. When those senior NRC managers spoke at public meetings in Vermont weeks and months earlier, they remained silent about those concerns, instead conveying only rosy assurances. Chair Jackzo and Commissioner Magwood provided spin-free commentary on conditions at these plants.

Expanding Public Access to NRC Records

Members of the public can gain access to NRC records in several ways. For example, they can search the Agencywide Documents Access and Management System (ADAMS), which includes hundreds of thousands of records.[3] They can also submit requests for information to the NRC under the Freedom of Information Act (FOIA). The NRC significantly improved public access to its records via both these avenues in 2010.

The agency introduced Web-Based ADAMS (WBA), a new interface that greatly enhances public access to NRC records.[4] WBA lacks the firewall barriers of earlier interfaces, and allows users to find, view, and download records more easily. The system also allows NRC staff to make changes to it more quickly. For example, after some users told the NRC that the interface had made some routine searches more difficult, employees revised WBA within days to allow the requested searches.

The NRC also recently added a search tool to its website that greatly facilitates public access to licensee event reports (LERs).[5] Federal regulations require plant owners to submit LERs on the causes of problems with safety equipment and corrective actions taken. The new search tool allows users to find LERs for a specific cause at a specific reactor during a specific time frame, and provides many other search options. The LER database also extends back decades—long before records stored in ADAMS.

The NRC also significantly improved its response time to FOIA requests. UCS has often waited months and sometimes more than a year for NRC responses to FOIA requests. In 2010, UCS received complete responses to FOIA requests of comparable scope within weeks.

Unlike the Oconee, Browns Ferry and Kewaunee catches, these gains in public access to information do not immediately affect plant safety. However, they deserve equal recognition. The NRC prides itself on being transparent. When it backs up good intentions with action, everyone wins.

Observations on Effective NRC Oversight

At Oconee, Browns Ferry, and Kewaunee, some information suggested that the status quo was acceptable, but onsite NRC inspectors probed deeper.

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[3] See *http://www.nrc.gov/reading-rm/adams.html*.

[4] See *http://www.nrc.gov/reading-rm/adams/web-based.html*.

[5] See *https://lersearch.inl.gov/Entry.aspx*.

Resident inspectors at other plants can improve plant safety by asking similar kinds of questions:

- Could workers actually perform critical but dangerous safety-related actions inside a reactor containment vessel during an accident?

- Could a degraded safety system work reliably for the entire essential period if an accident occurs?

- Even if problems with a safety system might not limit its performance during many accidents, could the system perform as required during *all* such events?

In all three of these cases, plant owners were initially satisfied that reactor safety was adequate, but NRC inspectors revealed that the owners were wrong. These owners should have ensured plant safety without NRC assistance—and in fact were legally required to do so. Given this record, the NRC must insist that plant owners find out why their own testing, inspection, and evaluation methods fail to uncover safety-related problems.

CHAPTER 4

NEGATIVE OUTCOMES FROM NRC OVERSIGHT

This chapter describes situations where lack of effective oversight by on-site NRC inspectors led to negative outcomes. As Chapter 3 noted, these outcomes are not necessarily the worst the NRC achieved last year. Rather, they provide insights into practices and patterns that prevent the NRC from achieving the return it should from its investment in oversight.

Peach Bottom's Slow Control Rods

The NRC was aware of a serious safety problem at the Peach Bottom nuclear plant in Pennsylvania in 2010, and an even more troubling response by the plant owner, yet did nothing except watch.[6]

The Peach Bottom plant includes two boiling water reactors (BWRs), both with 185 control rods. The power level in these reactors can spike under certain conditions. If that occurs, all control rods can be fully inserted within seconds to stop the nuclear chain reaction—a vital response. Fatal accidents at the Chernobyl nuclear plant in Ukraine in April 1986, and the SL-1 nuclear plant in Idaho in January 1961, occurred when unchecked increases in reactor power caused massive steam explosions.

The operating licenses for the Peach Bottom BWRs require the owner to test the control rods periodically, to verify that their insertion times are within required safety margins. Each control rod travels 12 feet from the fully withdrawn to the fully inserted position. The licenses require that each control rod begin moving within 0.44 second, and finish moving within 3.35 seconds, after operators initiate this response. Because each BWR features 185 control rods, some can be "slow" if their neighbors are "fast." The operating licenses and associated safety studies limit the share of slow control rods to 7 percent of tested control rods.

On January 29, Peach Bottom workers tested the insertion times of 19 control rods at Unit 2, and found that three took longer than 0.44 second to begin moving. The workers then tested other control rods, to try to reduce the share of slow ones to less than 7 percent of those tested. However, they in-

[6] For more information on this Peach Bottom event, see Union of Concerned Scientists. 2010. Artful dodgers at Peach Bottom. Cambridge, MA. Online at http://www.ucsusa.org/nuclear_power/nuclear_power_risk/safety/brief-on-slow-control-rods-at.html.

stead found more slow ones. Workers ultimately tested all 185 control rods and found that 21 were slow.

The operating license for Unit 2 requires workers to shut down the reactor within 12 hours if more than 13 control rods are slow. However, workers did not shut down Unit 2. Instead, the team testing the control rods slowed its pace to match that of the team repairing the slow ones. That meant the plant never officially had more than 13 slow control rods. However, because of the foot-dragging, tests of all 185 control rods took longer than two days—a task I have performed in a single 12-hour shift at similar reactors.

The control rods were slow because of a part found to be faulty in the 1990s. The vendor offered free replacement kits at the time, and other BWR owners fixed the problem. However, 39 of the 185 control rods at Peach Bottom Unit 2—including the 21 slow ones—still had the defective part.

As soon as workers traced the cause to the defective part, the safe and legal move would have been to shut down the reactor. Instead, the workers conspired to keep the reactor operating despite known safety flaws. Had Unit 2 encountered an event that required rapid insertion of the control rods before employees finished playing their games, the results could have resembled those at Chernobyl and SL-1.

Onsite NRC inspectors were fully aware of the shenanigans at Peach Bottom but simply stood by. The NRC later issued a Green citation to the plant owner for replacing the defective parts only belatedly (NRC 2010w). However, the agency could and should have examined earlier tests of the control rods to show that testing all 185 does not take two days, and then asked the owner to justify the foot-dragging. The NRC also should have forced the plant owner to comply with federal safety requirements rather than scoff at them.

The NRC's reaction contrasts sharply with that in 1987, when the agency fined both individual Peach Bottom operators and the company after finding that operators routinely slept on duty. The NRC did so because they demonstrated "a total disregard for performing licensed duties and a lack of appreciation for what those duties entail," and because supervisors and senior plant managers knew or should have known about the rampant sleeping (NRC 1987). In so doing, the NRC noted:

The NRC expects licensees to maintain high standards of control room professionalism. NRC licensed operators in the control room at nuclear power plants are responsible for assuring that the facility is operated safely and within the requirements of the facility's license, technical specifications, regulations and orders of the NRC.

Because both operators and managers deliberately circumvented safety requirements again in 2010, the NRC should have levied similar sanctions. When the agency condones egregiously poor performance, it is being unfair on many levels. First and foremost, that response is unfair to the people living around Peach Bottom, who deserve protection. A lax response is also unfair to the owners of other plants, who sometimes pay a price for doing the right thing.

For example, the owner of the North Anna nuclear plant in Virginia voluntarily shut down the Unit 2 reactor in September 2010. The owner took this step after workers at Unit 1—which had shut down on September 12 for refueling—discovered 58 cubic feet of Microtherm insulation and 8 cubic feet of calcium-silicate insulation inside the containment building.

In 2007, to resolve a safety problem, workers had removed Microtherm and calcium-silicate insulation from the containment buildings for North Anna Units 1 and 2. During an accident, such insulation could block the flow of water to emergency pumps used to cool the reactor core and the containment building. The owner replaced the Microtherm and calcium silicate with another type of insulation less likely to impair the performance of emergency pumps.

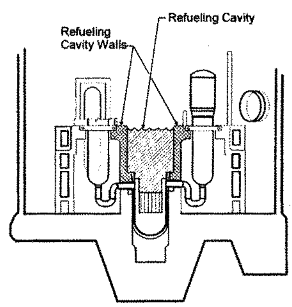
In 2010, rather than arguing that unlike Unit 1, Unit 2 did not contain leftover Microtherm and calcium-silicate insulation, or that Unit 2 could operate safely until its next scheduled refueling outage, the owner voluntarily shut down Unit 2 and fixed the problem (NRC 2010c). The owner did the right thing despite the fact it carried a price tag reflecting lost revenue from electricity sales and the higher cost of replacing insulation on short notice. North Anna's owner clearly placed safety ahead of production.

This owner took a financial hit for doing the right thing—only to watch as the NRC allowed Peach Bottom's owner to avoid a financial hit by doing the wrong thing. North Anna's owner has a long track record of putting safety first.[7] Not all owners can match that record. The NRC must deprive owners of the option of placing safety second, third, or lower.

Indian Point's Leaking Refueling Cavity Liner

The Indian Point nuclear plant in New York features two pressurized water reactors (PWRs). To refuel a PWR, workers flood the refueling cavity with water, which allows them to remove irradiated fuel assemblies from the reactor core and replace them with fresh fuel assemblies. The water both removes decay heat from the irradiated fuel assemblies and shields the radiation they emit, protecting the workers.

The Final Safety Analysis Reports (FSARs) submitted by the plant owner with the application for an operating license for Unit 2 stated that the refueling cavity was "designed to withstand the anticipated earthquake loadings," and that "the liner prevents



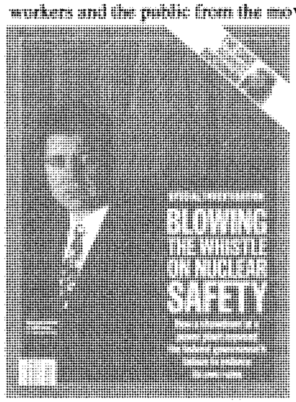
NRC drawing showing refueling cavity walls and the fuel rods located at the bottom of the cross-hatched refueling cavity volume.

[7] In fall 2001, North Anna's owner voluntarily shut down a reactor months before a scheduled refueling outage, to inspect the nozzles on the reactor's control rod drive mechanism (CRDM). The owner of the Davis-Besse plant in Ohio, in contrast, resisted NRC pressure to conduct these inspections, and operated a reactor into 2002 with cracked and leaking CRDM nozzles. The NRC later found that this near-miss of a reactor accident was the most serious event since the Three Mile Island meltdown in 1979.

leakage in the event the reinforced concrete develops cracks." When the NRC issued the operating license for Unit 2, the leakage prevention function of the liner for the refueling cavity became part of the licensing basis.

However, NRC inspectors at Indian Point recently found that the liner has been leaking 2 to 20 gallons per minute since at least 1993 (NRC 2010v), and that the plant owner has not yet delivered on repeated promises to fix the leak. That means the device installed to prevent leakage after an earthquake is leaking before an earthquake even occurs. The liner has no other safety function. Yet NRC managers have dismissed the longstanding problem, noting that the refueling cavity leaks only when it is filled with water (NRC 2010o).

These inspectors are repeating the very same mistakes the NRC made at the Millstone nuclear plant in Connecticut 15 to 20 years ago. In March 1996 the NRC made the cover of *Time* magazine—and not as regulator of the year. *Time* called the NRC out for failing to enforce its own rules. Workers at Millstone routinely transferred all the fuel from the reactor core to the spent fuel pool during each refueling outage, despite a regulatory requirement to do so only under abnormal conditions. Workers also nearly always violated a regulatory requirement to wait a few hours before transferring fuel out of the reactor core, to allow radiation levels to drop, thus lowering the threat to workers and the public from the movements.



After being embarrassed on the cover of *Time*, the NRC found that the Millstone reactors had been operating outside their design and licensing bases, and ordered the owner to shut them down (NRC 1996). The NRC also fined the owner a then-record $2.1 million, for "several failures to assure that the plants were operated in accordance with design requirements in the plants' Final Safety Analysis Report (NRC 1997a).

To prevent another Millstone, the agency also required its inspectors to review "the applicable portions of the FSAR during inspection preparation and verify that the commitments had been properly incorporated into plant practices, procedures, or design (NRC 1997b). The resident inspectors at Indian Point were expressly carrying out this prevent-another-Millstone mission when they discovered that the degraded refueling cavity liner no longer conformed to the plant's licensing basis.

The Millstone debacle also prompted the NRC to develop specific guidance on what plant owners should do when they find degraded or nonconforming conditions (NRC 2008).

This guidance allows owners to resolve nonconforming conditions via any one of three options: (1) full restoration to the FSAR condition; (2) a change in the licensing basis to accept the new condition; or (3) some modification of the facility or licensing basis other than restoration.

That means the Indian Point owner could fix the refueling cavity liner so that it no longer leaks. Or the company could seek NRC approval for leaving the cavity liner as is, if an evaluation shows that the plant would then maintain required safety margins. Or the owner could seek NRC's approval to modify the plant or its procedures to compensate for the leaking liner.

However, the Indian Point owner has chosen option 4: to do absolutely nothing to resolve the safety nonconformance, daring the NRC to respond. That was the very same option the Millstone owner chose in the early 1990s—which led to the reactor shutdown and the NRC's efforts to prevent such a situation from ever happening again.

The laissez-faire approach to safety at Indian Point contrasts sharply with the approach at Turkey Point Unit 3 in Florida, after a similar problem surfaced in 2010. On July 29, workers at that plant detected a through-wall crack in the drain pipe from the refueling cavity transfer canal (FPL 2010). Workers could not repair the crack until they drained the refueling cavity, but the owner committed to making the repair immediately after they did so.

The owner also committed to "daily walkdowns for increased leakage or new leak locations while the transfer canal is filled." In other words, workers would inspect that area each day for water leaking from the damaged drain pipe. Rather than fall back on the NRC's apparent indifference to leaks from the refueling cavity, this owner took steps to manage the risk until workers could correct the degraded condition.

The NRC's performance at Indian Point is worse than that 15 to 20 years ago at Millstone, for the simple reason that the agency has put measures in place to prevent the next such fiasco. The NRC has explicitly directed resident inspectors to determine whether nuclear plants are operating within their licensing bases, and whether they are adhering to the agency's guidance given any discrepancies.

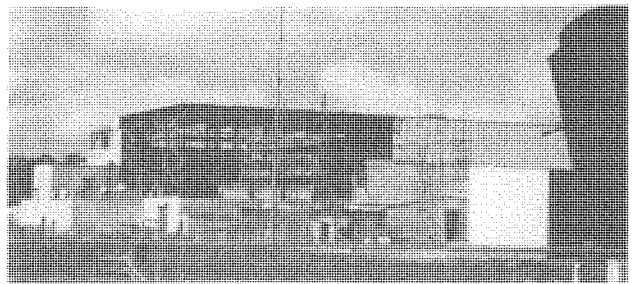
The resident NRC inspectors at Indian Point did their job by flagging the degradation of the liner for Unit 2's refueling cavity, and the fact that the plant does not conform to its licensing basis. However, NRC managers have deviated from their own post-Millstone guidance by accepting the degraded, nonconforming condition without any analysis showing that the plant has critical safety margins. There is just no excuse for the NRC to revert back to its pre-Millstone nonchalance regarding nuclear reactors that operate outside their licensing bases.

Curbing Illegal Radioactive Effluents

NRC regulations permit owners to routinely release air and water contaminated with radioactivity from their nuclear facilities. However, owners must monitor and control the pathways for such effluents, and the total inventory must remain below federal limits. These regulations are intended to protect the public from radiation-induced health problems.

The NRC has enforced these regulations inconsistently over the past decade. Examples at two plants—one positive and one negative, both at plants owned by Entergy—illustrate this baffling inconsistency.

In September 2008, Hurricane Gustav caused considerable damage to the River Bend nuclear plant outside Baton Rouge, La. High winds tore sheet metal siding from three sides of the turbine building. The company repaired some damage and prepared to restart the reactor—planning to replace the walls of the turbine building later.



The turbine building at the River Bend plant after Hurricane Gustav removed its metal siding. NRC photo.

If the radioactivity level of air flowing through ventilation ducts in the turbine building rises too high, radiation detectors sound alarms and dampers close, to stop any release to the environment. Because the River Bend turbine building lacked walls, any radioactively contaminated air that had leaked into the building would have reached the environment via uncontrolled and unmonitored pathways.

The potential for unmonitored and uncontrolled releases spurred the NRC to take steps to prevent River Bend from restarting. Only after reinstalling the walls and complying with regulations could the owner restart the plant.

In January 2010, Entergy informed the NRC that it had detected tritium—radioactively contaminated water—in an onsite monitoring well at the Vermont Yankee nuclear plant. The company thought the tritium was coming from a leak in an underground pipe, but was uncertain about the location, size, and nature of the leak. The NRC allowed the company to continue operating Vermont Yankee while workers searched for the leak. Weeks later they found holes in two underground drain pipes that carried radioactively contaminated water to a tank inside the turbine building.

At River Bend, the mere potential for an unmonitored and uncontrolled release of radioactively contaminated air prompted the NRC to prevent the reactor from operating until the owner eliminated that potential. Yet at Vermont Yankee, an actual unmonitored and uncontrolled release of radioactively contaminated water from spurred no response from the NRC.

The agency did the right thing at River Bend by enforcing its regulations and not allowing Entergy to intentionally violate them. The agency did the wrong thing at Vermont Yankee—and at Pilgrim in Massachusetts, Oyster Creek in New Jersey, Brunswick in North Carolina, and many other plants by pretending that those same regulations did not exist.[8]

The people living in Vermont and other states expect and deserve the same protections as those the NRC provided to residents of Louisiana. By

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[8] See Lochbaum, David. 2010. Regulatory roulette: The NRC's inconsistent oversight of radioactive releases from nuclear power plants. Cambridge, MA: Union of Concerned Scientists.

failing to enforce regulations designed to protect public health and safety, the NRC let millions of Americans down.

Observations on Lax NRC Oversight

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

When workers at Oconee sought to narrow a problem to Unit 1, NRC inspectors expanded the shortcoming to two other reactors. When workers at Peach Bottom sought to narrow a problem to a handful of control rods at Unit 2, NRC inspectors passively accepted that response.

When workers at Browns Ferry justified a degraded safety system by saying that it satisfied all requirements at that moment, NRC inspectors questioned whether the system could respond throughout an emergency. When workers at Indian Point noted that a critical safety liner leaked only when filled with water, NRC managers meekly nodded.

When workers at Kewaunee explained that they had been testing a safety system a certain way for nearly four decades, NRC inspectors asked whether the system could do its job if the reactor remained in operation during testing. When workers at Indian Point explained that a safety device had been leaking for more than two decades, NRC managers simply accepted that deviance.

When River Bend's owner sought to restart a reactor without the ability to monitor and control releases of radioactively contaminated air from the turbine building, the NRC stepped in to prevent that scenario. When Vermont Yankee's owner sought to continue operating the reactor while releasing radioactively contaminated water from an uncontrolled and unmonitored pathway, the NRC stepped aside and allowed it.

NRC inspectors cannot examine every inch of piping or every foot of cabling. They cannot look over the shoulder of every worker to verify that he or she is following every procedure faithfully, and that the result of every test is valid.

NRC staff informed commissioners some 15 years ago that inspectors could audit 5–10 percent of all activities at each reactor each year. Every safety problem found during a 10 percent sample audit represents 9 safety problems in areas not sampled. Each safety problem found during a 5 percent sample audit represents 19 other safety problems in areas not sampled.

The NRC cannot be blamed for safety problems in areas it does not examine, but the agency deserves considerable blame for failing to correct safety problems it has identified. When the agency's limited-scope audits find broken devices, the failures of the plants' testing and inspection regimes to find and fix these devices are the true safety problems. By failing to insist that owners correct these true safety problems, the NRC does nothing about the 90–95 percent of conditions and activities in nuclear plants that it does not audit.

Peach Bottom, Indian Point, and Vermont Yankee are all in the NRC's Region I. All the negative outcomes in 2010 involved Region I reactors, while none of the positive outcomes involved Region I reactors. Those outcomes may simply be statistical anomalies. Or they might indicate where the agency most needs to reform its own efforts and those of plant owners—and soon.

CHAPTER 5

SUMMARY AND RECOMMENDATIONS

In UCS's view, the 14 near-misses reported at nuclear power plants in 2010 are too many, for several reasons:

- Two of the near-misses occurred at the HB Robinson plant in South Carolina. These events shared contributing causes, including design flaws complicated by known but uncorrected equipment problems—and inadequate operator performance. Neither the plant owner nor the NRC should have allowed conditions to deteriorate so deeply and broadly that they set the stage for near-miss after near-miss.

- Four of the near-misses occurred at three plants owned by Progress Energy. This company owns only four plants. Better corporate governance and NRC oversight likely would have prevented the company's fleet from having such a bad year.

- Reactor owners could easily have avoided many of the near-misses in 2010 simply by correcting known problems. For example, one Calvert Cliffs reactor was known to have a leaking roof, with frequent reminders occurring when it rained. But the problem remained uncorrected until rainwater triggered a series of events that ultimately shut down both reactors.

- Similarly, workers at Wolf Creek predicted in 2007 that piping in a vital cooling system was vulnerable to leaking, and actual leakage in April 2008 validated that prediction. Yet the company merely patched the leak, allowing the degraded piping to leak further in August 2009.

The NRC identified 40 violations of federal safety regulations in these near-misses. Some of these violations resulted from problems arising during the event itself, but most were for safety problems known for months if not years. When known problems combine to cause near-misses, they are not surprises: these were accidents waiting to happen.

The NRC enables lax behavior to occur again and again. For example, the NRC sanctioned the Calvert Cliffs owner for not having fixed the leaky roof. When the owner finally fixed it, NRC inspectors verified the repair.

However, they let the owner off the hook by not probing whether other known safety problems remain uncorrected. Nor did the NRC ask the owner to explain why it had allowed the leaking roof to go unrepaired for so long, or to describe measures it would use to prevent future roof leaks from going uncorrected. In short, the NRC did little to prevent known safety problems from causing future near-misses at Calvert Cliffs and other sites.

The NRC must draw larger implications from narrow findings for the simple reason that it 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.

Each NRC finding therefore has two important components: identifying a broken device or impaired procedure, and revealing deficient testing and inspection regimes that prevented workers from fixing a problem before the NRC found it. The NRC's recurring shortcoming is that it focuses nearly exclusively on the first part. It is good that the NRC assured that the leaking roof at Calvert Cliffs no longer leaks even when it rains. But the NRC failed in the larger sense by not ensuring that Calvert Cliffs patched leaks in its testing and inspection regimes that allowed this known problem to languish for so long. The NRC simply has to do better in tackling this larger picture.

The NRC can do better because the NRC did do better in some cases last year. Agency inspectors uncovered safety problems at the Oconee, Browns Ferry, and Kewaunee plants that their owners initially misdiagnosed or dismissed. NRC resident inspectors kept asking questions until the true picture came into focus. Their commendable efforts meant that owners corrected safety problems, making these plants less vulnerable to near-misses. The intangible dividends from these efforts are very likely lessons learned by these plant 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 the NRC in the Oconee, Browns Ferry, and Kewaunee cases is not yet the rule. The NRC did not flag comparable safety problems at the Peach Bottom, Indian Point, and Vermont Yankee nuclear plants.

At Indian Point, the liner for the refueling cavity has been leaking for nearly 20 years. The only reason the liner was installed is to prevent leakage during an earthquake. That means the chances that the liner could fulfill its only safety function are nil. The NRC tolerates this longstanding safety violation. However, if an earthquake caused a near-miss at Indian Point, the NRC would sanction the company for having violated safety regulations for so long—even though the agency is essentially a co-conspirator in this crime.

By boosting its commendable performance and shrinking its poor performance, the NRC would strengthen safety levels at nuclear plants across the country, reducing the risks of near-misses—and full-blown accidents.

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The NRC and Nuclear Power Plant Safety in 2010

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Ms. EDWARDS. Thank you very much, Mr. Chairman. I have a question, actually, for Mr. Lochbaum. I know that the Union of Concerned Scientists in that report that we just entered into the record concluded that "Nuclear subsidies effectively separate risk from reward, shifting the burden of possible losses onto the public and encouraging speculative investment by masking the true cost of nuclear power, and that subsidies also allow the industry to exaggerate its economic competitiveness. Consequently, they diminish or delay support for more economical and less risky alternatives, like energy efficiency and renewable energy." That is a direct quote from your report. Do you believe that the nuclear power companies would be economically viable without substantial federal subsidies they receive from taxpayers?

Mr. LOCHBAUM. Based on the work we have done and the industry's own request for loan guarantees and other subsidies, the answer seems pretty clear that they would not be.

Ms. EDWARDS. I wonder if you could elaborate on how public subsidies, and especially at the levels at which we subsidize the industry to nuclear power distort risk in the nuclear power industry?

Mr. LOCHBAUM. I think the best example of that would be the Price-Anderson Federal Liability Protection. Plant owners have to get approximately $375 million of private liability insurance, and the Price-Anderson Act protects against liability costs above that. That is a big savings for the plant owners, but more importantly, it discourages the reactor vendors from developing designs that are less risky and much safer, because there is no incentive—while the higher cost of those safety features may be borne out, because they don't get a break on the insurance protection, the liability insurance that they get, it is hard to sell that into a marketplace when you are competing with cheaper, less safe reactors. So the federal subsidies are actually discouraging reactor designers from coming up with safer reactors that better protect Americans.

Ms. EDWARDS. So this leads me to a question about identifying and fixing safety risks. Do you think the NRC does what it needs to in both identifying safety risks and forcing fixes to these known safety problems at our power plants in a timely way?

Mr. LOCHBAUM. During my ten years experience with UCS and during my predecessor's 20 years, we find that the NRC does a very good job at setting the safety bar at the right height. They establish regulations that provide adequate protection of public health. They don't do a very good job of enforcing those regulations. Too many plant owners are limboing beneath the safety bar for too long, putting Americans at higher risk, and additionally driving the costs of nuclear power upwards inexplicably.

Ms. EDWARDS. So in my colleague Mr. Harris's opening statement, he indicated that Fukushima is a type of accident that really is not possible here, given our nuclear regulatory environment, and so Mr. Lochbaum, I wonder if you could respond to that question as to whether there is a major nuclear—whether a major nuclear accident is actually possible here in the United States, given the NRC's oversight of our reactors?

Mr. LOCHBAUM. I think, again, the best proof that it is possible is the fact that the nuclear industry cannot operate nuclear power plants without federal liability protection. If there wasn't a chance of such a catastrophic accident, they could go down to State Farm and get private liability insurance. The fact that they can't means that they themselves recognize that these plants are unusual hazards of unprecedented nature.

Ms. EDWARDS. So let us go to something as simple as battery backup. At Fukushima, you indicated that the battery—I think it was in Mr. Barrett's testimony—I apologize, I probably got it all wrong. Whoever had the slides up there—that the battery backup at Fukushima was eight hours of battery backup, and compared to U.S. plants, what is the backup like at U.S. plants in the event of some catastrophic disaster?

Mr. LOCHBAUM. The battery backup is basically the same for U.S. reactors. Some reactors only have four hours of battery backup, so they would be even more vulnerable to that situation. Studies done by the Nuclear Regulatory Commission show that many of our reactors, the station blackout where you are relying on battery, the De Salle plant in Illinois, for example, that is 80 percent of the overall risk of core meltdown. It is equal to four times the risk of all other things leading to meltdown combined, so it is station blackout and battery dependence at our U.S. reactor.

Ms. EDWARDS. Yes, so let us take away the fact that we might have a hurricane or tornado, or some other thing, simple blackout that could be caused by any number of factors actually poses a strong vulnerability for risk, isn't that right?

Mr. LOCHBAUM. That is absolutely right. I mean, when you get down to station blackouts, you only have one safety system working. If something causes that to go away, you played beat the clock and lost, like they did in Japan.

Ms. EDWARDS. Thank you very much, and thank you, Mr. Chairman.

Chairman BROUN. Thank you, Ms. Edwards.

Now I recognize Mr. Rohrabacher for five minutes.

Mr. ROHRABACHER. Thank you very much, Mr. Chairman, and thank you for your leadership in holding this hearing.

Let me first ask, you just made a statement about the subsidies, and without the subsidies nuclear power would not be able to compete. Is that not also true of solar and most of the other renewables? By the way, we subsidize them to the tune of billions every day, so here we are—are those subsidies not necessary? Is this a new revenue source for us to defund those subsidies for the renewables?

Mr. LOCHBAUM. There is no such thing as a free lunch. Everybody gets a shot at the apple. I think the point we were trying to make was that nuclear power has been subsidized so heavily over so many years and has built in subsidies that it is not a level playing field.

Mr. ROHRABACHER. I got you, but you reach a certain plateau and subsidies are still necessary for the nuclear energy, but let me just note, as we stand today, we are subsidizing perhaps even heavier these new supposed renewable sources of energy.

How many people—I am just asking the panel—how many people have died in nuclear power accidents over the last 50 years here in the United States? Anybody?

Mr. LOCHBAUM. There is the one that is buried in Arlington from the January 3, 1961 accident, so——

Mr. ROHRABACHER. So since 1961 has there been anybody? I mean, there is one guy back in 1961. Anybody else?

Mr. LOCHBAUM. He had two colleagues.

Mr. ROHRABACHER. Okay. How many people have died in the production of coal during that time period? I think we are talking about hundreds of people, are we not, maybe thousands.

Mr. LOCHBAUM. Probably in the thousands.

Mr. ROHRABACHER. Probably in the thousands, because we are also talking about lung disease that people get from coal, et cetera. So there is a place for that, too, although coal isn't subsidized, or is it subsidized? Yes. Perhaps we are taking care through the black lung whatever fund that we have and that we fund federally, so there are subsidies for coal even as well.

So what strikes me today is, of course, we have seen the crisis over in Japan, this horrible accident which we now seem to say that there are not large numbers of people dying, but this puts people at risk. Have people lost their lives in Japan already? Has anybody been—and I mean, I know in Chernobyl they certainly did. Is the Japanese accident resulted in loss of life?

Mr. SHERON. We are not aware of any nuclear related deaths from the Fukushima.

Mr. ROHRABACHER. Well the tsunami, of course. Right.

Mr. SHERON. Yeah, the tsunami, obviously people died there, but if not——

Mr. ROHRABACHER. Any nuclear-related deaths? All right. Now let me just say that this is—all of this is happening while we are utilizing 50-year-old technology. All the complaints that we hear and the risk that is being taken, if there is a risk, is happening because we are utilizing 50-year-old nuclear technology. Light water reactors were put in place in the '60s, were they not?

Mr. SHERON. Even sooner than that.

Mr. ROHRABACHER. Right, even sooner than that. There is a new generation of nuclear power plants that come to grips with many of the challenges that exist that require subsidies, et cetera, for the nuclear industry, and that new technology is actually focused on small modular reactors and high temperature gas cool reactors. Should we not then start focusing our efforts on these new technologies rather than making the light water reactors a bit safer? Shouldn't we be focusing our research and energy on putting in place high temperature gas cool reactors which cannot melt down and maybe these small modular reactors, which would be dramatically safer?

Mr. SHERON. I will take a shot at that. At the NRC, we don't really pass judgment on what kind of reactors should be built. We leave that up to the industry and the Department of Energy to determine that. Our job is to determine if what is put in front of us meets our regulations and is safe.

Mr. ROHRABACHER. Let me just—for the record, Mr. Chairman, state that in various studies that I have made and hearings that I have been at, it is very clear that we have now the capability of overcoming many of the challenges that nuclear energy 50 years ago posed to us. For example, the elimination of waste, you actually have some of these new reactors that will bring the level of waste being stored in Yucca Mountain down, rather than bundle it up, meaning that it actually burns used fuel as part of its own fuel cycle.

So as we look at the safety and the challenges of nuclear energy, I would hope that we keep in mind that a lot of the challenges and a lot of the criticisms are the old technology, and we have a great new opportunity to move forward with new technology and solve these problems.

Thank you very much, Mr. Chairman.

Chairman BROUN. Thank you, Mr. Rohrabacher.

Chair now recognized Mr. Miller for five minutes.

Mr. MILLER. Thank you. Dr. Lochbaum mentioned the Price-Anderson Act as a substantial subsidy for the industry, and in traditional economic theory, the market mechanism for safety is liability, that if you cause harm to others, then you are responsible for it. You make them whole, you compensate them for their losses. Does anyone dispute that a cap on liability is a subsidy to an industry?

Okay, so you all agree with that. I understand that Price-Anderson limits the liability to $375 million. What relationship does that have to the actual risk? Dr. Lochbaum?

Mr. LOCHBAUM. It is pretty much decoupled from that. That was set as a number that has been upped over the years. The way Price-Anderson works, if there is offsite damages that exceed that number, whatever it is, then the rest of the surviving reactors are invoiced to make up the difference. In the old days when they regulated utility companies, that secondary pool was pretty much guaranteed. Today, many of the reactors are limited liability corporations that may shut down and not be available to pay into that secondary pool.

Mr. MILLER. I guess my question is will the actual lawsuits of Fukushima or a similar accident be anywhere in the neighborhood of $375 million, or whatever the liability is under Price-Anderson, or it could be substantially more?

Mr. LOCHBAUM. If Fukushima is any indication, they drive by that almost the first day, very quickly. Much higher.

Mr. MILLER. And the subsidy is perhaps not borne by taxpayers, but it is borne by random depending on which way the wind blows.

Mr. Rohrabacher mentioned the experience since 1961, and usually lawsuits go into actuarial considerations and underwriting and insurance is pretty good at that. That is their business. Even with no deaths since 1961, do any of you think that industry could get insurance—liability insurance without a cap, given the 50 years of no deaths? No one thinks that? I mean, so the industry continues to say that the risk is acceptable, so long as someone else bears it? If the risk is on them, it is unacceptable.

Dr. Lochbaum, there sometimes is a tradeoff between safety and profits. In your work in the industry, have you identified any shortcuts that might be—might make operations more profitable but less safe?

Mr. LOCHBAUM. There are those opportunities. For example, we are aware of right now that the industry knows of about half the plants operating in the United States don't need fire protection regulations who were adopted after the 1975 fire at Brown's Ferry. The plant owners who have consciously spent the money to come into compliance are actually at a cost disadvantage to their neighbors who are outlaws, nuclear outlaws. The Nuclear Regulatory Commission is basically enabling bad behavior that drives cheaper plants to be less safe plants. If the industry were to enforce its regulations, those fire protection regulations, people would be protected, but more importantly, the people wouldn't benefit from violating the law as they have in the past.

Mr. MILLER. All right. Mr. Rohrabacher also mentioned new technologies. Do you think the new technology will dramatically change the potential risk of nuclear accidents? Could a nuclear power company—a company operating a nuclear power plant go to insurance companies and say look, we have got this new technology, now will you write us some coverage?

Mr. LOCHBAUM. At a House hearing back in I think it was 2006, there was a vendor, reactor vendor at the table who was asked that question, and he said his company was so—could stand behind their reactor design and opt out of Price-Anderson. No other reactor vendors I have heard have said that, and reactor operators haven't said that either.

Mr. MILLER. Okay. Do you—Dr. Lochbaum, how safe do you think these plants will be if—compared to the old technology? Will there be a dramatic difference in safety?

Mr. LOCHBAUM. What we have in the new reactors is that the chances of an accident are less with the new reactors, but any time a safety gain is made in that regard, the containment is made less robust and there are savings done, so that the cost remains the same. As a result, the number of accidents would be fewer, but the number of dead bodies will be greater.

Mr. MILLER. My time is almost expired. I will yield back the little bit that I have got.

Chairman BROUN. Thank you, Mr. Miller.

Chair now recognizes Dr. Harris for five minutes.

Dr. HARRIS. Thank you very much, Mr. Chairman. I want to thank the panel for your patience as we went and voted.

Mr. Lochbaum, from what I understand, you are making the availability to get liability insurance kind of a guide as to how safe something is. I know a lot of obstetricians and neurosurgeons in some States who just can't get liability insurance from commercial companies. They literally couldn't get it, so the State had to form insurance companies. How is that different from what is going on? I mean, I assume that there are people who still think it is safe to go to an obstetrician, safe to go to a neurosurgeon, but in fact, there are instances where you can't conduct normal business, because look, there is tort in this world. What can I say? Isn't that true? I mean, aren't there other circumstances where the government has to step in to insure things that people consider pretty safe, I mean, going to an obstetrician, going to a neurosurgeon?

Mr. LOCHBAUM. There is, but if you look in the energy technology sector, nuclear power is the only one that is so hazardous that it needs——

Dr. HARRIS. Oh, I understand that, but in the medical sector it is only OB/GYNs and neurosurgery. That doesn't mean that it is dangerous to go to a neurosurgeon, that is my only point. I mean, to use that as—you know, because you do represent the Union of Concerned Scientists, I mean, I don't think that is a very scientific way to look at it, to be honest with you, because we know from other areas where tort law is an issue that that just doesn't work. The world just doesn't work that way. It is not that simple.

Dr. Boice, there has been—you know, part of the discussion and I think Mr. Lochbaum's testimony actually brings it up, part of the problem is with spent fuel and the risks with spent fuel. You know, an issue that I think is probably going to come before us at some point is the getting spent fuel out of these plants and eventually getting to a central location. Have you looked into at all the risks associated on populations with using a central repository like Yucca Mountain? Do you have any writings that you can provide me or provide the Committee?

Dr. BOICE. No, not specifically with regard to spent fuel and enhanced levels of radiation in the background. We have done a number of studies of people that lived in areas of enhanced background radiation in China and other countries where they have been exposed to increased levels that might, in some sense, be relevant. We have also done studies that evaluated cancer risks around all the nuclear power facilities, including those with proximal spent fuel storage in the United States and all the DoE facilities. So there actually are a number of studies in counties and areas close by that we could provide for you that might be somewhat relevant, but not specific to spent fuels and the levels of radiation from those exposures.

Dr. HARRIS. And is it because—I mean, is the reason because that is a—is probably a much, much lower risk than the risks associated with the plant, which is already low enough, than the storage of spent fuels in a facility like Yucca Mountain?

Dr. BOICE. I just have not had an opportunity to look at that issue, except indirectly since many nuclear power plants have their spent fuel stored in areas close to the operating reactor.

Dr. HARRIS. Okay, thank you very much.

Mr. Barrett, as you are aware, the Department of Energy does and has been moving forward with their next generation nuclear plant project for some time. My understanding is that the high temperature gas cooled reactors may have some very specific safety advantages, some of which—mitigating some of the risks we have been talking about today. Could you speak to the safety characteristics of that kind of reactor?

Mr. BARRETT. I am not an expert on gas cooled reactors, but I know a little bit about them. They have very excellent physics. They have a lot of very valuable safety aspects. They have developmental challenges ahead of them, economics and other things as well. But gas cooled reactors are a very good, safe technology. It is very passive, it doesn't heat up as quickly as some of the others do.

Dr. HARRIS. In your opinion, would that be a reason perhaps for the Department of Energy to more aggressively pursue research into that, because it does address some of those problems with things like passive cooling?

Mr. BARRETT. The Department of Energy has many worthy projects that they are working in their R&D program, that certainly is a worthy project and it is there. Relative to other R&D projects, I am afraid I can't really judge from where I am today.

Dr. HARRIS. Okay, thank you very much. I am going to yield back my time, Mr. Chairman.

Chairman BROUN. Thank you, Dr. Harris.

Now the Chair recognizes Mr. McNerney for five minutes.

Mr. MCNERNEY. Thank you, Mr. Chairman. One of the advantages of being the last Member to ask questions, I may have a little extra leeway in terms of my time, so I don't feel quite as pressured.

We are a little lucky here in Congress to have so many physicians that can expand their experience to the rest of life, so I really appreciate the wisdom that we get very frequently from the other side of the aisle in that regard.

Let us talk about the backlog at the NRC. Would—Dr. Sheron, can you describe how long it would take for someone or an organization that submits a design to get the decision on that design?

Mr. SHERON. I presume you are talking about a new plant design?

Mr. MCNERNEY. Yes.

Mr. SHERON. I believe that the Agency has identified a schedule. I can't remember exactly what the time is. I believe it is on the order of maybe several years.

Mr. MCNERNEY. Five years?

Mr. SHERON. I think it is less than that. A lot of it is dependent upon the quality of the submittal, however, whether the licensee—the applicant has adequately addressed all of the safety issues and is providing a strong technical basis to support them.

Mr. MCNERNEY. So is the new design evaluation in competition for resources, for NRC resources with safety evaluations of existing plants or new issues that come up in that regard?

Mr. SHERON. No. When there was an indication that there would be new designs coming in, the Agency purposely split the Office of Nuclear Reactor Regulation into two separate offices. One is the Office of New Reactors, and the other is the Office of Nuclear Reactor Regulation. The Office of Nuclear Reactor Regulation focuses solely on the safety of the current 104 operating plants. The Office of New Reactors focuses solely on the licensing of the new applications.

Mr. MCNERNEY. So would you say that your modeling—and I am not trying to throw arrows here or anything—that your modeling capabilities are state of the art, you have the best computers, the best numerical techniques and so on in doing modeling, both of the design and of the nuclear fuel rod modeling, safety modeling?

Mr. SHERON. Yes, I would probably say that the NRC has the best—some of the best models in the world, which is evidenced by the fact that most of the other nuclear countries—developed nuclear countries request our models, and we have a numerous cooperative programs where we provide our models to others to use.

Mr. MCNERNEY. Okay, good. Now with regard to failsafe, in my mind, failsafe means fail safe. It doesn't mean fail badly. We have had a couple of cases lately, one in Fukushima, one in the Gulf Coast last year where failsafe really didn't mean fail safe. Is your modeling able to predict any of these failures that were supposed to be failsafe that actually weren't failsafe?

Mr. SHERON. We don't—I don't think there are any designs right now that are totally failsafe. Obviously one can postulate failures that are going to, you know, lead to an accident. Our computer codes are able to model those failures and to predict the consequences. If we see that the consequences are too high or that there are other mitigative things that could be done, then we certainly pursue them with the industry or through regulation.

Mr. MCNERNEY. So I mean, you basically have said—I think you just said that the current design is not really failsafe. That is basically the situation, isn't it?

Mr. SHERON. Well, what I am saying is that there are low probability events that one could postulate, okay? In other words, if one postulates enough failures, which again, become very low probability——

Mr. MCNERNEY. Well they sound like they are low probability until they happen, and then they say geez, that wasn't as unlikely as we thought it was. Obviously, no one predicted a 14-meter tsunami in Japan. That was completely unforeseen.

Mr. SHERON. It was not unforeseen. I have heard reports that there was some prediction that the design basis at Fukushima was not adequate, but I am not at liberty or I am not really going to speculate on whether that is appropriate, you know, in other words whether or not the TEPCO organization designed the plant properly. What I will say is that we have looked at the design of U.S. plants against tsunamis and earthquakes, and we have concluded that we believe that our plants, you know, are adequately designed for those. In other words, we can predict fairly well, for example, the wave height of any tsunami that might occur and we make sure that the plants are adequately designed.

Mr. MCNERNEY. But we have heard this morning that a simple blackout, which could last, depending on what the cause is, for a week or a month if there is a significant transformer that goes down at a substation, which puts these plants at significant risk.

Mr. SHERON. You have got to be careful when you say it is a simple blackout. It is not a simple blackout. What the—what you are concerned about is first that you lose the offsite power source, which is the preferred source of power to the plant. The plant has two independent diesel generators that are designed to start and provide electricity to power the safety systems. You now have to postulate that both of those diesels don't start, not just one, but both don't start. Then there are additional backup systems that will run for some period of time. We do——

Mr. MCNERNEY. That sounds good, but we just saw at Fukushima that that wasn't necessarily the case.

I just want to make a little plug here. You know, you talk about the current generation of nuclear being safer—the current technology being safer than 50-year-old technology, and maybe that is the case, you know. I don't really know, I am not a nuclear engineer. But there is fast neutron technology that would be inherently failsafe, is that correct, Dr. Lochbaum?

Mr. LOCHBAUM. I am not aware of that, I would have to look. I don't know offhand if that is true or false.

Mr. MCNERNEY. Well, I find myself in agreement with Mr. Rohrabacher, and I am going to take just a few more moments here. We need to be aware of the new technology and make sure that if there is a fourth generation or fast neutron technology that it gets proper attention, and meanwhile, be very skeptical of claims of failsafe or highly improbable incidents.

Thank you, Mr. Chairman, for your indulgence.

Chairman BROUN. You are quite welcome.

We will now undertake a second round of questions, and I yield myself five minutes.

Mr. Barrett, Chairman Jaczko made a recommendation or made a judgment of a 50-mile evacuation to U.S. citizens at Fukushima. Were you involved in evacuation actions during the Three Mile Island accident?

Mr. BARRETT. Yes, I was at Three Mile Island, not at Fukushima.

Chairman BROUN. I should just ask you about Three Mile Island. Were there any NRC lessons learned from that experience?

Mr. BARRETT. Yes, the NRC and everybody learned a lot from that experience. One of the lessons learned from that is it is not just the nuclear computer codes and the "what if" calculations that are made at that time, but it is also what are the conditions on the ground, what is the situation with the people? Because you are trying to make a judgment call, whoever is making these evacuation recommendation decisions, to do the best thing for the people at that time under those conditions.

Chairman BROUN. Do you think this was done properly at Three Mile Island?

Mr. BARRETT. At Three Mile Island, at the time we made the decisions, and I was part of that, I thought it was the right thing to do at that time. However, I went and lived there for four years and I saw what the impact of that was and what the practicality of what an evacuation does to the people. After I learned from that experience, I felt it was inappropriate that we did that evacuation at Three Mile Island in the early days.

Chairman BROUN. After your experiences at Three Mile Island, do you believe that the 50-mile judgment by NRC Chairman Jaczko made for U.S. citizens at Fukushima was appropriate, and if you would please explain?

Mr. BARRETT. No, I don't think that really was Chairman Jaczko's judgement in the net sense. I believe that decision was a poor judgment decision, insofar as it was counterproductive and detrimental to all the people in Japan, the Japanese people as well as the Americans, because I don't think it appropriately considered the horrendous conditions that the people of Japan were under at that point with the tsunami and the earthquake. I mean, people were freezing in the north. A 50-mile evacuation radius hinders the ability of the people in the unaffected south to bring lifesaving supplies and things to people in the north. So I think it did not appropriately consider the situation on the ground. It was my understanding it was more of a worst case computer analysis "what if" type of projection. So my sense is there was not a sufficient evaluation of the conditions in Japan. I think it put a lot of confusion and uncertainty in the minds of people between the 12-mile official Japanese radius and the 50-mile U.S. one. People would ask each other "Why is yours different from mine?" In my view, I think one country should not second guess another country from 10,000 miles away as to what is the best thing for the citizens at that point.

Chairman BROUN. Thank you, Mr. Barrett.

Next question is for Dr. Boice. In the days following the Japanese disaster, U.S. Surgeon General Regina Benjamin responded to questioning about citizens stocking up on potassium iodide—actions were "definitely appropriate" cautions to take. What is your reaction to this suggestion, and is there any scientific basis for such recommendation, given the radiation levels that were detected?

Dr. BOICE. I believe—when the surgeon general mentioned that, it was shortly after the accident and all the evidence wasn't in about the radiation releases. When we found that the levels were so tiny, it certainly is an inappropriate action to make the statement that we should be distributing potassium iodide pills. I concur with the public health department from California and also the director of our own CDC that potassium iodide should not be given. There are adverse health effects, and particularly dangerous for people who have sensitivities to iodine, people who have thyroid disease and also people who are allergic to shellfish. Then if it is taken inappropriately, there can be serious effects such as heart abnormalities, nausea, and diarrhea. So the benefit, which is almost nonexistent because the levels of radiation are so incredibly small, is not sufficient with regard to these potential adverse health effects.

Chairman BROUN. Dr. Boice, I appreciate your efforts to put radiation risk in perspective. I was struck that you note that the U.S. Capitol building is frequently cited as having some of the highest radiation levels in the United States at 85 millirem per year. Could you put that level, which Members and employees of Congress are exposed to every day, in perspective with amount of elevated radiation that Americans on the West Coast might have been exposed to as the result of Fukushima?

Dr. BOICE. Certainly. The Capitol building for long term exposures of over a year might be on the order of 85 millirem from the gamma rays from the granite that was used in the building. From the Fukushima radiation, the potential exposure even to California is much, much less than one unit, 1 millirem. So it is a very tiny, inconsequential exposure. It is much less than just what we get every day from normal radiation exposures from natural background.

Chairman BROUN. Thank you, Dr. Boice. I now recognize Ms. Edwards for five minutes.

Ms. EDWARDS. Thank you, Mr. Chairman, and thank you very much for this second round of testimony.

You know, we heard from Mr. Rohrabacher that since we haven't had scores of dead bodies from past nuclear accidents, we shouldn't be worried about the future safety of nuclear plants, and I think the jury is still out, frankly, on what the long-term consequences are of even Fukushima, and I would note that as yet, nobody has done one of those longitudinal studies because we haven't actually had sufficient time pass. And yet, I keep seeing claims also that accidents like Fukushima couldn't happen here and that health effects of the disaster in Japan were inconsequential. Again, I think the jury is still out, but it does seem to be a bit of a mixed message that suggests that we are safe and nothing bad has happened anyway, and so, you know, let us just wait. And then here I see the cover of this week's New York Times on the Wednesday edition, and here you have got people—a couple in Japan in radiation protection gear, clearing out their precious possessions from a home that they may never be able to return to because of an accident. So I don't think we should have to wait until the accident happens before we figure out the safety of our plants.

Mr. Lochbaum, I wonder if you could tell us whether we understand the full impacts of Fukushima on health and safety in the communities around the plant, and if you could, elaborate on the 14 near-misses that occurred at U.S. power plants last year alone, and your key findings about what your biggest concerns are regarding nuclear safety here in this country?

Mr. LOCHBAUM. To address the first part of that question, I don't think we know what the human fallout from Fukushima will be. For example, because of the contamination they have had to increase the dose at schools in the area. Essentially, they have drafted all the school children into the nuclear workforce and the school children are now applied to the same radiation limits as nuclear plant workers. They had to do that, they really had no choice. The radiation levels are so high. The radiation elevation could cause problems for those children down the road, and we won't know that, unfortunately, for a while.

The study we did, we looked at the 14 near-misses, and the near-misses were times—events that occurred at nuclear power plants where the NRC had to send out a special team to look. What we found was that most of those 14, there were warning signs that were missed by the plant's owner and by the Nuclear Regulatory Commission that had they been heeded, the near-miss would have been avoided.

What concerns us about that is if you continue to miss—overlook the near-misses, the warning signs, you are setting the stage for preexisting conditions to cause that very bad day, should they be challenged. So the fact that we got lucky on those near-misses is great, but we need to remove luck from the equation to the extent we can.

Ms. EDWARDS. Thank you very much.

Lastly, I think we have heard a lot about—and Mr. Rohrabacher alluded to this—that we haven't built a new plant in this country for 30 years. New plants are being built overseas. I understand that they are being build in Finland and France and those have been pointed to as examples of where we need to go in terms of the technology, but I wonder if you can tell us, particularly Mr. Lochbaum, how the construction of those reactors is going? Are they on schedule, are they on budget, are we going to see them come online at any time, because it underscores, I think, the question about whether it makes sense to invest in these kind of long-term huge costs for a new plant without having the most aggressive regulatory scheme in place to make sure that they are safe.

Mr. LOCHBAUM. Well, both the nuclear plant under construction in Finland and in France are over budget and behind schedule. It is more than 25 percent over budget in Finland and several years behind schedule. They had trouble pouring concrete. They got bad concrete as a result. They had trouble with pipes, basic stuff that is nuclear 101 we didn't learn from the first go around and they are paying the price, not us.

Ms. EDWARDS. So what was—the cost of the French plant was what, initially?

Mr. LOCHBAUM. I don't—$6 billion.

Ms. EDWARDS. And it is 25 percent over budget?

Mr. LOCHBAUM. So far, they are not done yet.

Ms. EDWARDS. Okay, and they are not done yet, so in the end we could be talking about a $10 billion plant, and we still can't assure all of the safety considerations will be made.

Mr. LOCHBAUM. Going back to an earlier question, one of the things that France is doing—France is the vendor of that reactor that is being built. In order to try to market it elsewhere, they are taking some of the safety features out to reduce the price tag of the plant.

Ms. EDWARDS. Thank you, and with that, I yield back.

Chairman BROUN. Thank you, Ms. Edwards.

I now recognize Dr. Harris for five minutes.

Dr. HARRIS. Thank you very much, Mr. Chairman. Thank you for a second round so we can clear up some of these questions.

Dr. Sheron, very briefly in a minute or less, can you outline the evidence that our plants are not safe?

Mr. SHERON. Our plants are not safe?

Dr. HARRIS. Yeah, because there has been discussion that our plants aren't safe. Is there any evidence, scientific evidence, any evidence, injuries in the United States in the, you know, use of nuclear power for civilian use, anything like that. Is there any evidence that our plants are not safe?

Mr. SHERON. I am not aware of any.

Dr. HARRIS. Okay, thank you. Now, your office is considering updating spent fuel safety studies to estimate the relative consequence of removing older fuel from the spent fuel pool and placing it in dry storage. Have you specifically studied the additional risk associated with storing spent fuel onsite at operating reactors, as well as not operating and decommissioned reactors versus storing the spent fuel in a centralized geologic repository?

Mr. SHERON. No, that we haven't.

Dr. HARRIS. And is that something you think deserves closer examination, to answer that question about spent fuel? Where is it safer to store?

Mr. SHERON. I am probably not qualified to answer that. I think, you know, what we look at is if there is not a repository, is it safe to store the fuel in an interim, you know, location such as onsite in dry casks?

Dr. HARRIS. But not a question of whether—because you, I guess, make the practical assumption there may not be another repository, so that is probably why you haven't looked at it, I imagine, because it is simply a theoretical possibility?

Mr. SHERON. Well, I just don't know.

Dr. HARRIS. Okay, thank you.

Mr. Lochbaum, thank you for coming and testifying. You stated up front, you know, in your testimony that your organization's goal was to minimize the inherent risk of nuclear energy, and I take it that if you could make it safe that it would be something that would kind of satisfy your organization's search for something to minimize climate change, for instance.

But with that in mind that your organization wants to minimize the inherent risks, what is the organization's position or your position on how to manage our stockpile of nuclear waste? Do you think it is safer to leave it onsite at the 100-plus individual sites we have, or put it at a single location that is geographically isolated, away from population centers, underground, miles underground?

Mr. LOCHBAUM. We talked to that subject to the president of the Blue Ribbon Commission on the American Nuclear Future last august, and what we recommended was centralized interim storage for the permanently shut down plants where the only hazard left is spent fuel. Transfer that to some centralized location. We didn't specify it was above ground or below ground, but——

Dr. HARRIS. So you do think that is a good idea?

Mr. LOCHBAUM. Yes.

Dr. HARRIS. Mr. Barrett, given your experience at the Department's Office of Civilian Radioactive Waste Management, what do you think about that?

Mr. BARRETT. I fully agree. I believe we also need a geological repository for the permanent disposal of the waste that our generations have been making now for 40, 50 years and not just give this problem to our great grandchildren. So I think this country needs to move forward with Yucca Mountain, or if it has a better facility, let us have the better facility, but let us move forward while we are alive.

Dr. HARRIS. And I take it you feel—that is not only for decommissioned plants, but that is even for the spent fuel when it cools down enough even to be shipped from operating plants.

Mr. BARRETT. The decommissioned plants should be the first to go, and we have the two plants up in the northwest where there is a tsunami risk, even though it is in dry storage, that risk should not be there at all. There are almost a dozen of these old facilities. These should be the first to move but then we also need to start removing spent fuel from the operating plants, too, and reduce that risk as well.

Dr. HARRIS. Sure.

And finally, Dr. Sheron, would you please describe where the NRC currently is in its licensing efforts for the next generation nuclear plant project? How long do you think it might be or would take the Commission to issue a combined license?

Mr. SHERON. For the NGNP, we have already started doing research at the NRC to support our licensing reviews of that design when it is submitted by the Department of Energy. The last schedule I saw was that of the application, again, is complete and technically defensible. We had a three year review schedule, in which case—I'm sorry, at the end of three years we would issue the combined operating license.

Dr. HARRIS. Okay, thank you very much. I will yield back the balance of my time.

Chairman BROUN. Thank you, Dr. Harris.

The chairman will recognize Mr. Miller for five minutes.

Mr. MILLER. Thank you.

Quickly on the fact that there have been no new nuclear power plants in 30 years, in North Carolina 30 years ago, almost all of the cities and municipalities that had municipal power systems invested in a piece of one of Duke Power's nuclear facilities, and almost all of those cities came very close to bankruptcy as a result. It was hideously more expensive, even with all the subsidies that we have discussed. So it probably is not a regulatory burden, it probably was truly just much more expensive, even with the dramatic—very substantial subsidies that we have gotten.

We have had some discussions of storage of spent fuel, about Yucca Mountain, about what to do with the closed down facilities, but you know—and we haven't even discussed the transportation of that fuel. There is not a star Trek transporter technology. It will not be beamed from the plant to a permanent facility. There are risks and transportation and on and on.

Part of the story coming out of Fukushima has been the spent fuel pools. Dr. Lochbaum, what have we learned—should we have learned form the Fukushima experience with spent fuel.

Mr. LOCHBAUM. When the event occurred at Fukushima, there were seven spent fuel pools that contained the radiated fuel. There were also radiated fuel in dry casks. That doesn't make the news very much because it survived without a problem. There was—the spent fuel in dry casks is safe, secure, not leaking radioactivity, so the dry cask endured that challenge that the spent fuel pools did not. I think it was a reminder—it wasn't so much a lesson—that dry cask storage is less vulnerable both from a security standpoint and a safety standpoint, and we should act upon that lesson rather than just continue to document it.

Mr. MILLER. Okay, and you think that that should be required by regulation, by the NRC?

Mr. LOCHBAUM. It should happen. It would be nice if the plant owners did it for safety reasons; if not, then the NRC should do it for — to protect the American public safety, and if not then the Congress should make it happen. However it happens, we need to make that happen.

Mr. MILLER. All right. Dr. Lochbaum, again, you are very familiar with this industry. How would you characterize the level of candor of the industry with respect to safety issues that have arisen.

Mr. LOCHBAUM. Well, the industry does release a lot of information. There is very little dirty laundry that is withheld from the American public, that is why we know about the tornado that hit Surry plant at Brown's Ferry. With the exception of security information, there is very little withheld from the public, so I think the industry deserves credit for that candor.

I think the candor issue is really internally. There is a failure within the plants themselves sometimes to recognize problems, that is why Calvert Cliffs had roof leaking for years that they tolerated but didn't fix, because there was this complacency or lack of candor about realizing what that could do that led them to—not to solve the problem they kept seeing happen over and over again.

Mr. MILLER. I yield back the balance of my time.

Chairman BROUN. Thank you, Mr. Miller.

I thank the witnesses for you all's valuable testimony. If you are not from the South, you all means all of you all. And the Members for all of you all's questions. Members of either Subcommittee may have additional questions of you all, and we ask that you respond to those in writing. The record will remain open for the two additional weeks for additional comments or questions from Members. Witnesses are excused, and the hearing is now adjourned.

[Whereupon, at 12:15 p.m., the Subcommittees were adjourned.]

Appendix I

ANSWERS TO POST-HEARING QUESTIONS

Answers to Post-Hearing Questions

Responses from Dr. Brian Sheron, Director, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission

Hearing on *"Nuclear Energy Risk Management"*
Friday, May 13, 2011

Questions for the Record Submitted to Dr. Brian Sheron,
Director, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission

Questions submitted by Dr. Paul Broun, Chairman
Subcommittee on Investigations and Oversight

QUESTION 1. **In the days following the Japanese tsunami, the NRC recommended evacuation of all U.S. citizens in a 50 mile radius surrounding the Fukushima Daiichi plant. Information released since the recommendation was made indicates the NRC recommendation originated from a computer model known as "RASCAL."**

a) What inputs were used in the RASCAL model to generate such a recommendation? To your knowledge, was the source of those inputs based on actual event-specific information or were the inputs simply "worst-case scenario" numbers?

ANSWER.

The 50-mile evacuation recommendation that the U.S. Nuclear Regulatory Commission (NRC) made to the U.S. Ambassador in Japan was made in the interest of protecting the health and safety of U.S. citizens in Japan. The assessment was based on the conditions as understood at the time. Since communications with knowledgeable Japanese officials were limited and there was a large degree of uncertainty about plant conditions at the time, it was difficult to accurately assess the potential radiological hazard.

In order to determine the proper evacuation distance, the NRC staff performed a series of calculations using NRC's Radiological Assessment Code for Consequence Analysis (RASCAL) computer code to assess possible offsite consequences. The computer models used meteorological model data appropriate for the Fukushima Daiichi vicinity. Source terms were based on hypothetical, but not unreasonable, estimates of fuel damage, containment integrity, and other release conditions. Certain calculations showed the potential for the Environmental Protection Agency's (EPA's) Protective Action Guidelines to be exceeded at a distance of up to 50 miles from the Fukushima site, if a large-scale release occurred from the reactors and/or spent fuel pools (SFPs).

The NRC's decision to recommend an evacuation area around the Fukushima Daiichi reactor site out to 50-miles was informed in part by computer calculations that were conservative, rough estimates that would not necessarily characterize an actual release. The "Calculation 1" assessment assumed an ex-vessel, unfiltered release from a totally failed containment and actual meteorological conditions during early morning hours. The assumed total release to the atmosphere for this assessment was 1.7E+08 Curies (Ci). The "Calculation 2" assessment represented multiple unit failures using an increased inventory of radionuclides in Unit 2 as a surrogate for 30 percent core damage at Units 2 and 3 and a 100 percent damaged Unit 4 SFP. The assumed total release to the atmosphere for this assessment was 2.1E+08 Ci.

The simulated computer calculations and subsequent recommended protective action decisions were based on conservative estimates of fuel damage, containment integrity, and other release and meteorological conditions. Conservative "durations of release" were also assumed for each of the computer calculations and assessments.

Since communications were limited, there was a large degree of uncertainty about plant conditions and associated meteorological data at the time of the decision-making process. The computer models did, however, use "Actual Observations" as meteorological model data appropriate for the Fukushima Daiichi vicinity.

- "Calculation 1" assumed meteorological data during early morning hours, when there were lower wind speeds, relatively stable air, and light precipitation – conditions that would degrade dispersion and increase downwind doses.
- "Calculation 2" assumed meteorological data during a period of higher wind speeds, less stable atmospheric conditions, and no precipitation – conditions that would enhance atmospheric dispersion and decrease downwind doses.

QUESTION 2. **Mr. Lochbaum's testimony implies the risk associated with station blackout at U.S. nuclear reactors as too high and suggests further regulations are needed. Please discuss NRC's previous evaluations of risk assessment associated with station blackout. In the NRC's opinion are current station blackout probabilistic risk assessments inadequate?**

ANSWER.

NRC probabilistic risk analysis (PRA) studies since WASH-1400, "Reactor Safety Study" (1975), have analyzed station blackout (SBO) accident sequences, and have found these sequences to be relatively important contributors to the overall risk of nuclear power plants.

In 1988, recognizing the relative importance of SBO as a risk contributor, the NRC issued the SBO rule (10 CFR 50.63), which requires nuclear power plants to be able to cope and recover from an SBO of specified duration. The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The required duration is based on redundancy/reliability of onsite emergency alternating current (AC) power sources (e.g., emergency diesel generators - EDGs), frequency of loss-of-offsite-power (LOOP), and probable time needed to restore AC power.

In 1990, the PRA study contained in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," provided a snapshot-in-time assessment of the severe accident risks associated with five commercial nuclear power plants of different reactor and containment designs. The scope of the NUREG-1150 study included a variety of initiating event hazard groups and plant operational conditions. This study dealt extensively with SBO sequences initiated by various hazards such as grid failure, earthquake, fire, and equipment failure.

As a landmark study that advanced the state-of-the-art in PRA, the NUREG-1150 models, results, and risk perspectives were subsequently used to help inform a variety of regulatory applications, including but not limited to:

- Development and implementation of the Commission's PRA Policy Statement in 1995.
- Validation of regulatory analysis guidelines (for rulemaking and backfit bases).
- Validation of subsidiary numerical acceptance criteria (for risk-informed license amendments).
- Prioritization of generic safety issues and nuclear safety research programs.
- Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities.

In 1997, after reviewing licensee PRA submittals in response to NRC Generic Letter 88-20, the staff published NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance" (1997), noting that the average SBO contribution to overall core damage frequency reported by licensees was about 20%, although wide variation was reported among the plants. However, the overall estimated risk of core damage for the fleet of U.S. commercial nuclear power plants was sufficiently small to be considered consistent with the Commission's Safety Goal Policy Statement. In addition, some licensees reported estimated reductions in SBO contributions to core damage frequency due to implementing the SBO Rule.

In 2005, the staff published NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants," which was a statistical and engineering study of approximately 75 of

LOOP events at US commercial nuclear reactors. It analyzed data from 1986 through 2004, building upon the information contained in NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants," which analyzed LOOPs from 1968 through 1986. In NUREG/CR-6890 the LOOP characteristics (frequency, duration, etc.) were input into PRA models for all plants to estimate the risks. In both NUREG-1032 and NUREG/CR-6890, LOOPs from all causes were analyzed. NUREG/CR-6890 compared the current SBO core damage frequency estimates with historical estimates from approximately 1980 to the present. This study indicated a downward trend in SBO core damage frequency, from an average of about 2×10^{-5} per reactor critical year to the current average of about 3×10^{-6} per reactor critical year, or about 18% of the total core damage frequency. NUREG/CR-6890 states that this trend is the result of many changes – plant modifications made in response to the SBO rule, improvements in plant risk modeling and improved component performance. The major contributor for this historical drop appears to be improved emergency diesel generator (EDG) performance.

It should be noted that licensee strategies such as severe accident mitigation guidelines (SAMGs) established in the 1990s and extreme damage mitigation guidelines (EDMGs) developed in the early 2000s provide pre-determined strategies to mitigate long-term SBO. These equipment enhancements and procedures are not generally includ d in PRA models, but if they were, they would tend to reduce the risk estimates from SBO events.

Although regulatory and licensee operational improvements over the years have led to a decrease in SBO risk, it remains a relatively important contributor to core damage frequency, and, therefore, the NRC continues to study SBO using state of the art tools, including risk and consequence models (e.g., State of the Art Reactor Consequence Analysis).

In response to the Fukushima event, the Commission directed the NRC staff to establish a senior-level task force to conduct a methodical and systematic review of our processes and regulations to determine whether the agency should make improvements to our regulatory system. The task force plans to provide a report to the Commission in July 2011.

QUESTION 3. **Your testimony notes NRC's research office is considering updating spent fuel pool safety studies to "estimate the relative consequences of removing older fuel from the spent fuel pool and placing it into dry storage." Has the NRC specifically studied the additional risk associated with storing spent fuel onsite at operating reactors, as well as non-operating and decommissioned reactors, versus storing the spent fuel in a centralized geological repository?**

a. (If yes) What were the results of the study?

b. (If no) Please conduct the study and make available to the Committee upon completion.

ANSWER.

The NRC has studied the risks associated with storing fuel onsite in spent fuel storage casks. NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," summarizes this pilot study that was done for one specific reactor site and cask design. The results of this analysis indicated that the risk is solely from latent cancer fatalities, and no prompt fatalities are expected. The risk is dominated by accident sequences occurring when the spent fuel is being loaded into the dry casks. Once spent fuel has been loaded into the dry casks, the risk from earthquakes is extremely small (about 10 billion times lower than the Commission's Safety Goal Policy Statement[1]) because the dry casks are seismically rugged (i.e., designed to withstand earthquakes).

The NRC also completed a study in February 2001 (NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants") that indicated that the risk

[1] See Safety Goals for the Operation of Nuclear Power Plants (51 FR 28044; August 4, 1986, as corrected and republished at 51 FR 30028; August 21, 1986.

from spent fuel pool accidents is well below the Commission's Safety Goal Policy Statement for both the individual risk of early fatality and the individual risk of latent cancer fatality.

The NRC has not performed a comparison between storing spent fuel onsite versus storing spent fuel in a centralized geologic repository, although the Department of Energy did perform such an analysis as part of the environmental impact statement for the Yucca Mountain site, which the NRC reviewed and adopted. As a result of the recent events in Japan, an updated spent fuel pool (SFP) safety study to estimate the relative consequences of removing older fuel from the SFP and placing it into dry storage versus leaving it in the SFP is being considered.

The senior-level task force is conducting a methodical and systematic review of our processes and regulations to determine whether the agency should make improvements to our regulatory system. The longer-term review will include an examination of spent fuel storage practices in light of recent events to determine whether changes to our regulations are necessary to ensure continuing protection of public health and safety.

QUESTION 4. **The NRC has seismic and tsunami research programs that attempt to quantify the risks to each plant here in the U.S. Please describe those risks and place in context compared to the Fukushima Daiichi reactors.**

ANSWER.

All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate for the specific site and plant design. Even those plants that are located in areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. Each plant is designed to protect against a ground motion level that is conservatively determined for its location, given the possible earthquake sources that may affect the site and its tectonic environment. Ground motion is a function of both the magnitude of the earthquake and the distance from the fault plane to the specific site. The seismic responses of the structures, systems, and components associated with these facilities are site specific. The plants are analyzed for any identified faults and tectonic capabilities in the area in addition to any active seismic zones.

Many nuclear plants are located in coastal areas that could potentially be affected by a tsunami. Two nuclear plants, Diablo Canyon and San Onofre, are on the Pacific Coast, which is known to be susceptible to tsunamis. Two nuclear plants on the Gulf Coast, South Texas and Crystal River, could also be affected by tsunami. There are many nuclear plants on the Atlantic Coast or on rivers that may be affected by rising water levels resulting from a tsunami. These include St. Lucie, Turkey Point, Brunswick, Oyster Creek, Millstone, Pilgrim, Seabrook, Calvert Cliffs, Salem/Hope Creek, and Surry. Tsunami on the Gulf and Atlantic Coasts occur, but are very rare. Generally the flooding anticipated from hurricane storm surges exceeds the flooding expected from a tsunami for nuclear plants on the Atlantic and Gulf Coast, and the hurricane storm surge actually sets the design basis for these plants.

The NRC has investigated the risk to U.S. commercial nuclear power plants from natural hazards (including earthquakes) for over 20 years:

- In December 1990, the NRC issued NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," which included an assessment of seismic risks at two of the five nuclear power plants. The results of NUREG-1150 indicated that the most probable core-damage accidents initiated by earthquakes involve seismically induced loss of offsite power coupled with onsite equipment failures (e.g., emergency diesel generators, and cooling water systems).
- Through the issuance of Supplement 4 to Generic Letter 88-20 (June 28, 1991), the NRC requested all nuclear power plants to perform an individual plant examination of external events (IPEEE) for severe accident vulnerabilities. This extensive effort included licensee development of analyses for seismic, fire, severe weather, floods, and site-specific external hazards. The staff reviewed each licensee submittal and issued an associated staff evaluation report for each. The results of the IPEEE program are summarized in NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program." Seismic events were found to be relatively important contributors to core-damage risk for a majority of plants; in fact, the core-damage risk contribution from seismic events for some plants is of the same order of magnitude as that from internal events. As a result of the IPEEE program, about 70% of plants made seismic-related improvements (such as hardware modifications, improved procedures and training, and enhanced maintenance and housekeeping).

- In June 2005, the NRC established Generic Issue 199 (GI-199), "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants." The Safety/Risk Assessment of GI-199, which was completed in September 2010, estimated the frequency of seismically induced core-damage accidents at all nuclear power plants east of the Rocky Mountains by using an approximate method that combined seismic hazard information developed by the U.S. Geological Survey in 2008 and plant seismic response information developed from the IPEEE program. Overall seismic core damage risk estimates are consistent with the Commission's Safety Goal Policy Statement because they are within the subsidiary objective for core damage frequency. The GI-199 Safety/Risk Assessment indicates adequate protection exists and that the current seismic design of operating reactors provides a safety margin to withstand potential earthquakes that exceed the original design basis, however, the agency continues to examine whether cost-justified backfits are possible that would further lower seismic risk.
- The NRC completed a study in February 2001 (NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants") that indicated that the risk from spent fuel pool accidents is well below the Commission's Safety Goal Policy Statement for both the individual risk of early fatality and the individual risk of latent cancer fatality.
- The NRC has made estimates of the risk from spent fuel stored in dry casks at a boiling water reactor site (NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," March 2007). The results of this analysis indicated that the risk is solely from latent cancer fatalities, and no prompt fatalities are expected. The risk is dominated by accident sequences occurring when the spent fuel is being loaded into the dry casks. Once spent fuel has been loaded into the dry casks, the risk from earthquakes is extremely small (about 10 billion times lower than the

Commission's Safety Goal Policy Statement) because the dry casks are seismically rugged (i.e. designed to withstand earthquakes).

The Fukushima-Daiichi reactors are located in a region of much higher seismic and tsunami potential than any areas of existing US reactors. Any discussion of risk context would be premature until the sequence of events at Fukushima-Daiichi is fully understood.

Following 9/11, the NRC required additional mitigation capability for events involving large fires and explosions. The NRC task force on the Fukushima Daiichi accident is evaluating this additional mitigation capability, which could be useful for mitigating station blackouts (SBOs). The NRC task force is evaluating whether the current requirements for withstanding and mitigating seismic and tsunami events and SBOs at U.S. reactors are sufficient or if more needs to be done.

QUESTION 5. **How have the events in Fukushima informed or changed NRC research priorities with respect to reactor safety, response, public health, etc.?**

ANSWER.

The phenomena associated with the events at Fukushima-Daiichi involve numerous disciplines in which the Office of Nuclear Regulatory Research has expertise and where we have already done substantial analysis, including seismic and tsunami hazards, and severe accidents.

The NRC has previously studied spent fuel pool (SFP) issues and augmented licensee requirements in various areas such as an aircraft impact assessment, loss of SFP cooling, modifications to assembly configurations, and additional requirements following the attacks of September 11, 2001. As a result of the recent events in Japan, an updated SFP safety study to estimate the relative consequences of removing older fuel from the SFP and placing it into dry storage versus leaving it in the spent fuel pool is being considered.

The NRC created a senior level agency task force to evaluate technical and policy issues related to the event to identify additional potential research, generic issues, changes to the reactor oversight process, rulemakings, and adjustments to the regulatory framework that should be pursued by the NRC. A report with recommendations will be provided to the Commission in mid-July. The results of the task force, followed by the Commission's review of the report and the longer-term review, will help define which research actions the agency will pursue in the future.

Questions Submitted by Ms. Donna F. Edwards, Ranking Member,
Subcommittee on Investigations and Oversight

QUESTION 1. **Dr. Sheron, during the hearing you indicated that you believed U.S. nuclear power plants were safe.**

a. How does the NRC measure nuclear power plant safety?

ANSWER.

The performance of US commercial nuclear power plants is assessed under the Reactor Oversight Process (ROP). ROP inspections are performed by NRC inspectors. Through the ROP, inspection findings and a variety of performance indicators are assessed continuously. There is a minimum set of baseline inspections done for all reactor licensees. As conditions merit, additional inspections are performed.

If performance declines, the NRC increases its involvement with the specific licensee to ensure performance continues to be safe. This is summarized on a publicly available listing known as the Action Matrix. At present, there are no plants in the unacceptable performance column. Continued operation is not allowed for plants in the unacceptable performance column.

b. What metrics does NRC utilize to determine if a plant is functioning safely or being operated safely?

ANSWER.

A combination of inspection results and performance indicators are used to access licensee performance under the Reactor Oversight Process. The performance of individual plants is assessed continuously, and information related to these assessments is made publicly available (with the exception of security related information).

Overall industry performance is also assessed to determine if there are undesirable industry trends that need to be corrected. This information is summarized annually, and reported publicly. There are a variety of processes that can be used to address issues requiring immediate action(s).

c. **Does compliance with NRC regulations factor in to NRC's assessment of a plant's safety performance?**

<u>ANSWER</u>.

Yes. All inspection findings are evaluated for non-compliance with NRC regulations and enforcement action taken as appropriate. However, the Reactor Oversight Process focuses on safety and security performance, and is not limited to verifying compliance with regulatory requirements.

QUESTION 2. **In your testimony to the Committee regarding the potential of Spent Nuclear Fuel accidents, you said: "My personal opinion is that pools have a lot of water in them, and regardless of the amount of fuel, it takes a very long time, if there was an accident, to actually drain the pool to the point where there would be an uncovering of the fuel, which gives licensees ample time to bring in either emergency equipment or to restore whatever did fail."**

a. How long did it take the Spent Nuclear Fuel pools at the Fukushima reactor to become drained after the earthquake and tsunami struck?

ANSWER.

At this time, the NRC does not have information indicating that any of the pools at Fukushima Daiichi completely drained. Consequently, we have no estimate of time that would be required to completely drain the pools.

b. The Fukushima plant had just 8 hours of battery back-up available on site. How long did it actually take after they lost power for the site to get electricity restored to the site?

ANSWER.

On March 11, 2011 at 14:46, the earthquake struck the Fukushima Nuclear Power Station and offsite power was lost. Back-up diesel generators started; however all alternating current (AC) power supplies were lost when the tsunami struck the site, except for the one diesel generator that continued to provide power to units 5 and 6 throughout the emergency.

According to *The Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety - The Accident at TEPCO's Fukushima Nuclear Power Stations* (starting on pg. IV-51), offsite power was restored to Unit 2 on March 20, with power restored to all other units (Units 1, 3,and 4) by March 23 at 01:40.

It is our understanding from the Report that it took nine days for the Japanese to begin AC power restoration and 12 days to complete their efforts.

c. In the U.S. 93 of the 104 operating nuclear power plants have only 4 hours of battery back-up power on hand in the event off-site power is lost. The other 11 reactors have 8 hours of battery power available. Given the fact that the Fukushima reactor had 8 hours of battery back-up power but it took days, not hours, to restore electricity to the site does the NRC plan to recommend that U.S. nuclear reactors dramatically increase the amount of battery back-up power they have available in the event of unpredictable and extended loss of off-site power?

ANSWER.

Of the 104 operating nuclear power plants in the US, 44 plants adopted the alternate current (AC) independent method and have battery power for 4 hours; 43 plants use the Independent Alternate Power source (AAC) methodology and are designed to restore AC (Emergency or Offsite) power within 4 hours. Hence, they have a coping duration of 4 hours. 14 plants use the AAC methodology and can restore AC (Emergency or Offsite) power within 8 hours; 3 plants use AAC and have a 16 hour duration for restoration of AC (Emergency or Offsite) power. The latter 3 plants, all at one site, had originally assumed a 4 hour duration but emergency diesel generator reliability and loss of offsite power events affected the calculated duration that these plants had to consider.

In response to recent events in Japan, the Commission directed the NRC staff to establish a senior-level task force to conduct a methodical and systematic review of our processes and regulations to determine whether the agency should make improvements to our regulatory system. The review will examine SBO requirements in light of recent events to determine whether changes to our regulations are necessary to ensure continuing protection of public health and safety.

Question Submitted by Mr. Brad Miller, Ranking Member,
Subcommittee on Energy and Environment

QUESTION 1. **I assume that the NRC has safety regulations in place for a reason. They do not simply write regulations for the sake of writing regulations.**

a. **Why is it then that the NRC has provided exemptions from complying with fire code regulations to nearly half of the operating nuclear reactors in the United States?**

ANSWER.

Exemptions granted to plants were based on NRC findings that public health and safety would be protected based on an alternative approach to NRC regulations. The plants that have received exemptions from 10 CFR Part 50, Appendix R, Sections III.G, J, and O, were licensed to operate before January 1, 1979. The rule was published in 1980 and backfit to the plants already licensed to operate. This prescriptive fire protection rule provided generic requirements that were not reflective of the already built plants' features. These plants have wide variations regarding physical layout of plant equipment and fire hazards.

In 1982, the Connecticut Light and Power Company challenged the legality of the stringent fire protection rule being backfit on the industry. The court ruled, "Exemptions are to be granted by the (NRC) Commission upon a showing by the licensee that the required plant modification would not enhance fire protection safety in the facility or that such modifications may be detrimental to overall facility safety." Therefore, licensees of plants with certain features that did not meet the rule were permitted to request specific exemptions under 10 CFR 50.48(c)(6). (This section of the rule was later removed and replaced with a performance-based fire protection rule and the exemption request rule now in 10 CFR 50.12, "Specific exemptions.") As a result, many specific exemptions were issued to plants that were licensed prior to the issuance of Appendix R to 10 CFR Part 50.

The NRC Staff reviews all exemptions to ensure that safety is maintained. Only following NRC staff review and acceptance is an exemption considered part of a plant's licensing basis. The reviews are documented and placed on the public record.

The other half of the nuclear fleet was licensed to operate after January 1, 1979, and was not required by 10 CFR 50.48 to meet 10 CFR Part 50, Appendix R, Section III.G, J, and O unless directed to do so in specific license conditions. The NRC typically reviewed these plants using the guidelines of NUREG-0800, Standard Review Plan Section 9.5.1, "Fire Protection Program," which incorporate the criteria specified in Appendix R.

b. Doesn't this increase the potential hazards at these plants that are not in compliance with these regulations?

ANSWER.

No. Plant hazards are considered in the review of exemptions, therefore exemptions do not increase the potential hazards in plants. A point worth noting is with the term "Specific Exemption." In many cases, the licensee is actually proposing an "alternative" approach to fire safety rather than the prescribed method in Appendix R. This was recognized and documented as an acceptable approach in the 1982 legal decision.

c. **If the NRC does not believe nuclear power plants need to comply with fire code regulations why does NRC have these fire codes in the first place?**

ANSWER.

The NRC believes nuclear power plants need to comply with fire code requirements. (See 10 CFR 50.48 and Appendix R to 10 CFR Part 50.) The fire codes (10 CFR 50.48) and their related guidance documents provide a baseline fire protection program. With plants, especially ones that predate the rules, there are special circumstances to be considered on a case-by-case basis. The staff has a rigorous process for reviewing exemptions.

It should be noted that 10 CFR 50.48 and 10 CFR Part 50, Appendix R, are deterministic or prescriptive rules, such that a change from the rule requires an exemption. Through the national consensus standard process, the NRC worked with stakeholders on the development of National Fire Protection Association Standard 805 (NFPA 805) that is referenced in the risk-informed-performance-based, alternative rule in 10 CFR 50.48(c). The new performance-based rule reduces the need for specific exemptions and increases safety and the Commission has urged licensees to transition to this voluntary rule. The older fire protection rules were not performance-based and therefore, many exemptions for alternative fire protection features and systems were requested and granted by the NRC.

QUESTION 2. **Dr. Sheron, in your testimony you said the NRC has performed significant severe accident research since the Three Mile Island (TMI) accident to improve accident prevention and mitigation. Nuclear plant operators are supposed to have Severe Accident Management guidelines (SAMG), for instance, that they can rely on in the event of an emergency or accident. But according to the NRC's own Inspection Manual a review of these safety critical plans is not included in the NRC's "Baseline Inspection Program." In fact, NRC staff, according to the NRC Inspection Manual: "concluded that regular inspection of SAMG was not appropriate because the guidelines are voluntary and have no regulatory basis."**

a. If these guidelines are deemed important to ensuring that nuclear power plants operate safely and have appropriate accident mitigation and response guidelines in place before an accident occurs why is the SAMG a "voluntary" measure and why has the NRC intentionally opted not to inspect these safety guidelines?

ANSWER.

In January, 1989, the NRC staff considered requiring severe accident management programs at all nuclear power plants (NPPs), and issued SECY 89-012, "Staff Plans for Accident Management Regulatory and Research Programs." The document included essential elements of a utility accident management (AM) plan that would include the continual evaluation of severe accident information, including operating experience and research and inclusion of a training program for operators, technical support staff, and managers on the AM procedures.

The industry proposed to put AM programs into place as a voluntary initiative using an industry-wide set of closure actions and implementation guidance. In 1995, the NRC staff accepted this industry commitment to complete these actions, follow this guidance, and implement appropriate improvements identified during the process no later than December 31, 1998. All licensees met this commitment.

The industry AM programs are based on a technical basis developed for the industry by the Electric Power Research Institute (EPRI) for systematically evaluating and enhancing the ability to deal with potential severe accidents. Vendor-specific AM guidelines were developed from the technical basis for use by individual utilities in establishing plant-specific procedures, guidance, and training.

In response to recent events in Japan, the Commission directed the NRC staff to establish a senior-level task force to conduct a methodical and systematic review of our processes and regulations to determine whether the agency should make improvements to its regulatory system. The review will examine AM requirements in light of recent events to determine whether changes to the regulations are necessary to ensure continuing protection of public health and safety.

b. **How can the NRC, the regulatory body charged with overseeing the U.S. nuclear power industry, be assured that plant operators have appropriate Severe Accident Management Guidelines (SAMG) in place if NRC's own inspection manual tells NRC's investigators that it is "not appropriate" for them to inspect these guidelines?**

ANSWER.

In SECY-96-088, the staff outlined plans to perform a limited number of pilot inspections to develop confidence in licensee accident management (AM) implementation, and to perform less detailed evaluations of AM performance for the balance of plants. Long-term evaluation of AM plan maintenance would be performed on a for-cause basis. As part of this inspection process, AM demonstrations were conducted at four plants.

Based on the demonstrations, plants appeared to be addressing the key elements of the industry wide process, i.e., plant-specific severe accident management guidelines (SAMG), severe accident training, initial training and drills, and administrative programs to maintain capabilities. The staff concluded that closure had been achieved on a voluntary basis using industry-developed guidance and methods approved by the staff, and that a formal inspection program was not necessary. The capabilities are maintained by conducting utility AM drills and self-assessments, and are periodically updated by the utilities to incorporate new information.

The NRC recently carried out inspections to ensure that SAMGs, training, and drills remain current. In these inspections, it was determined that, in some cases, the guidelines needed to be updated, or plant workers require more training on their implementation. These findings are being considered in the NRC's review of the Fukushima accident.

c. **How can Congress and the public be confident nuclear power operators are implementing these guidelines appropriately and effectively or have them at all if NRC refuses to investigate these guidelines?**

ANSWER.

The NRC carried out Severe Accident Management Guideline (SAMG) inspections at the request of the agency task force examining the lessons to be learned from the March 11 earthquake and tsunami in Japan, and the resulting damage to the Fukushima nuclear power plant. The NRC directed its resident inspectors at every U.S. nuclear power plant to examine the plants' SAMGs, which are meant to contain or reduce the impact of accidents that damage a reactor core. All plants put these guidelines in place voluntarily in the late 1990s.

The resident inspectors examined where the plants keep the SAMGs, how the guidelines are updated, and how the plants train their personnel to carry out the guidelines. The inspectors found that all plants have implemented the guidelines, with 97 percent of the plants keeping the SAMG documents in their Technical Support Center, generally considered the best location for properly implementing the guidelines. Our inspectors did identify that a number of plants do have additional work to do in either training their staff on these procedures or ensuring the guidelines are appropriately updated.

d. **Does the NRC have any plans currently in place to inspect the Severe Accident Management Guidelines (SAMG) or make them regulatory and not voluntary guidelines?**

ANSWER.

As noted above, the NRC recently carried out SAMG inspections at the request of the agency task force examining the lessons to be learned from recent events in Japan. The task force will incorporate the SAMG inspection results into its short-term review to help determine if any immediate changes to NRC requirements are called for in light of events at Fukushima. The inspection results also will help inform the NRC's long-term review of possible revisions to agency licensing and oversight processes.

Responses from Mr. Lake Barrett, Principal, L. Barrett Consulting, LLC

Joint Hearing
Subcommittee on Investigations & Oversight
Subcommittee on Energy & Environment
Committee on Science, Space, and Technology

"Nuclear Energy Risk Management"

Friday, May 13, 2011
10:00 a.m.

Questions for the Record Submitted to Mr. Lake Barrett,
Principal, L. Barrett Consulting, LLC

Questions Submitted by Dr. Paul Broun, Chairman
Subcommittee on Investigations and Oversight

1. The accidents at Three Mile Island and Fukushima produced highly radioactive water. Your testimony states, "All the accident related water was contained on site and special water processing systems were built to remove the radioactive fission products, primarily Cesium and Strontium."
   a. Was contaminated water at the Fukushima plant contained or released into the environment?

      A: The utility, TEPCO, has so far contained most of accident water, but some has leaked from the plant basement areas through pipe and cable trenches into the sea.

   b. Are there design mechanisms at nuclear plants to capture and contain contaminated water or other liquid spills?

      A: Yes. Usually accident water can be contained, but at Fukushima Daiichi Units 1-4, the existing radioactive water treatment system has been overwhelmed. Supplementary treatment systems are being developed. Three Mile Island was able to contain the accident liquid spills. US plants have even improved further since that time.

   c. What is the cleanup process for radioactive liquids if released into the soil or water?

      A: Once radioactive water is released into the soil or sea water, it is hard to recover. It is possible to capture some radioactivity by pumping

contaminated water through zeolite filters. This is currently being done at Fukushima.

2. How might private industry leadership incorporate efforts to simultaneously improve public awareness and rational communication regarding the progress of the integration of safety measures learned from Fukushima?

   A: Communications with the public is a challenge when responding to an emergency situation at a nuclear facility because most utility resources are dedicated to addressing the situation at hand to protect the public and environment. Public awareness and communication can be addressed by having an effective standing emergency preparedness plan with State and Local governments and electronic and printed media organizations. Through advanced planning, improved communication links can widely disseminate prompt and accurate information concerning the plant situation if needed. The Fukushima experience indicates that the advent of electronic social media methods is a new communication avenue that has the potential to possibly improve traditional emergency communications plans.

Questions Submitted by Ms. Donna F. Edwards, Ranking Member, Subcommittee on Investigations and Oversight

1. You testified that you believe the decision to recommend American citizens evacuate from within 50 miles of the plant at Fukushima did not "appropriately consider the situation on the ground." You also said, "I think one country should not second guess another country from 10,000 miles away as to what is the best thing for the citizens at that point."
   a. If a government has reason to believe that a nuclear accident in another country has the potential to expose its own citizens in that foreign country to levels and types of radiation that will endanger their citizens' health, do you believe that the government should not act to protect its citizens?

      A: I believe that a country should protect its citizens if there is reason to believe that they are in a realistically dangerous foreign place. However, in my view, the risk to US citizens in the area beyond the Japanese exclusion zone was unnecessary, unwarranted, and a net detrimental impact to all concerned. Although there were minor health risks from uncontrolled releases and much was not known about plant conditions at that time, there was no indication of actual or imminent offsite releases to warrant such a large costly detrimental evacuation. Such a overly cautious action could have resulted in thousands more people dying in the stricken north because the 50 mile radius cut the main road and rail supply routes that were transporting desperately needed life saving supplies from the south. There does not appear to be any safety rational for such a wide evacuation except assumed scenarios that do not seem reasonable and never happened. Indeed, Chairman Jaczko stated in a Congressional hearing that it appeared that the Unit 4 spent fuel pool had boiled dry. This has been proven to have not happened.

If Chairman Jaczko had been aware of the Three Mile Island evacuation experience, as I was, he would have more completely considered the total impact of his actions and most likely concluded differently. In my view his decision, although well meaning I am sure, was a reflection of inexperience and poor judgment. Chairman Jaczko was only eight years old at the time of Three Mile Island and his professional career has only been inside the Washington beltway.

b. At Chernobyl, the Russian government did not reveal either the extent or nature of the disaster for several days. Your testimony suggests that had the American government known of the situation, that it should have left American citizens in place, and in harm's way, simply because that is what the Russian government desired. Is that what you believe?

A: Absolutely not. The Soviet Chernobyl behavior was completely unacceptable and in that case, if there had been an opportunity for a US precautionary evacuation of US citizens, I would have supported it. Chernobyl was a completely different technical situation with an explosion of dry reactor core materials directly into the air. Such a gross airborne dispersal mechanism was not technically possible at Fukushima when the US evacuation was issued. In addition, I believe the Japanese utility and Japanese governments took responsible actions in alerting the nearby population of the accident at the earliest possible time. Although Fukushima internal plant information was uncertain, there is no reason to believe that the Japanese acted like the Soviets. To imply such similarities is an insult to a trusted ally and good friend. Adding to their burden in their time of need with an overly cautious hypothetical event scenario was very inappropriate and damaging to all.

Responses from Dr. John Boice, Scientific Director, International Epidemiology Institute

Joint Hearing
Subcommittee on Investigations & Oversight
Subcommittee on Energy & Environment
Committee on Science, Space, and Technology

"Nuclear Energy Risk Management"

Friday, May 13, 2011
10:00 a.m.

Questions for the Record Submitted to Dr. John Boice, Scientific Director, International Epidemiology Institute

Questions Submitted by Dr. Paul Broun, Chairman
Subcommittee on Investigations and Oversight

1. Please outline the need for further studies of low-dose long-term exposure to radiation. Have similar studies been undertaken previously? What specific unknowns deserve closer examination?

Much is known about the health effects of high levels of radiation when received briefly. However, the single most important question in understanding radiation health effects is determining the level of risk associated with low-dose long-term exposure to radiation, i.e., when the radiation is delivered at a low rate over a long period of time.

There have been attempts to address the issue of risk following gradual exposures over time, but they have fallen short. These include an international study of radiation workers in 15 countries (Cardis et al. 2007). Although the study was very large, the workers were young and received very little radiation exposure. Also, there were serious biases and methodologic limitations that unfortunately rendered the results uninterpretable (Boice 2010a). A second important study deals with the health effects of Russian citizens who lived downstream from the Mayak nuclear facility on the Techa River. In the late 1940s and early 1950s the former Soviet Union dumped their radioactive waste into this river and many tens of thousands of persons living downstream were contaminated unknowingly (Krestinina et al 2007). Although results from this study are important, the inability to accurately estimate the amount of radiation dose received, coupled with the complexities of conducting and tracing subjects in the former Soviet Union, make

this investigation limited in providing uncertain estimates of radiation risk (UNSCEAR 2008). A third study involves the evaluation of cancer risk in populations that live in areas of high natural background radiation, such as areas in China and in Kerala, India. These studies of lifetime radiation exposure may prove informative, but to date have not provided useful estimates of risk (Boice et al. 2010b). A final study that has potential for providing information on the risks of low-dose long-term exposure to radiation are those associated with the Chernobyl accident that occurred in 1986. The study of childhood exposure to radioactive iodines and increased thyroid cancer risk has already made an important contribution, but it appears that the doses associated with other environmental exposures were too small to result in a detectable radiation health effect. Similarly, the study of the large number of Chernobyl clean-up workers, so-called liquidators, has failed to reveal a consistent increased risk in leukemia or other cancers as might have been expected if the doses had been higher (UNSCEAR 2011).

The specific unknowns in radiation risk assessment that deserve closer examination are then to learn what happens when radiation dose is experienced gradually over time and not briefly as was the case for Japanese atomic bomb survivors and for patients treated with medical radiation. This gradual exposure to radiation is the type of exposure that most of us experience throughout life, whether from medical, occupational or environmental circumstances (NCRP 2009).

A tremendous amount has been learned from the study of atomic bomb survivors and from patients who received medical radiation. But these exposures are of relatively high dose and over relatively short periods of time. These studies are not optimal for use in risk assessments for exposure circumstances that are of current concern in America, such as radiation associated with diagnostic medicine, occupation, compensation schemes for past exposures, nuclear power, nuclear waste, and even the small exposures from fallout experienced from the Fukushima Dai-ichi reactor accident.

There have been studies conducted in the United States in years past that have the potential to address risks following protracted exposures to radiation. The problem is that they were conducted some 20-30 years ago, and the opportunity should not be lost to bring them up to date (Boice 2011). These studies of workers and veterans in the United States should be then combined because one study by itself is just too small to provide the statistically powerful answers that are needed. The One Million Worker and Veterans Study is over ten times as large as the atomic bomb survivor study, has higher radiation exposures cumulated over many years and could be extended now.

2. What is the current understanding of cancer risk among persons living near nuclear power plants in the United States?

In the late 1980s and early 1990s, I initiated a comprehensive survey of cancer risk among persons living near nuclear facilities in the United States. This study was conducted by the National Cancer Institute and included all nuclear power plants and U.S. Department of Energy national laboratories (Jablon et al. 1991). A special focus was on childhood leukemia because of reports of cancer clusters around nuclear facilities coming from the United Kingdom. Our study findings were reassuring in that there was no evidence that persons living in the United States near nuclear power plants were at increased risk of cancer compared with other citizens who lived in nearby counties but with limited potential for exposure from nuclear facilities. These results were not entirely unexpected given that standards are set so that population exposures do not exceed 100 mrem per year for an individual, a level way below were risks have been demonstrated in epidemiologic studies.

The results of studies around nuclear facilities in the United States have been confirmed and enhanced in studies conducted in a number of countries over the years as recently summarized in a comprehensive committee report from the Health Protection Agency in the United Kingdom (COMARE 2011). It is my understanding that the Nuclear Regulatory Commission has an interest in extending the previous survey conducted in the United States and has requested the National Research Council to conduct preliminary evaluations as to how this might be done (NAS 2011).

3. What are the international guidelines for populations to return to areas that have been contaminated with radiation? What are the health effects likely to be for the citizens surrounding the Fukushima Daiichi plant?

The international guidelines for populations to return to areas that have been contaminated with radiation are taken from the International Commission on Radiological Protection, of which I am a commissioner. Our recent publication is entitled, "Application of the Commission's recommendations to the protection of people living in long-term contaminated areas after a nuclear accident or a radiation emergency" (ICRP 2009). These guidelines are being used by the Japanese authorities in their "provisional idea" for allowing the evacuated populations and exposed populations to return to contaminated areas from the Fukushima Daiichi nuclear power plant. Specifically:

> **"The reference level for the optimization of protection of people living in contaminated areas should be selected in the lower part of the 1-20 mSv per year ban." (ICRP 2009)**

Because the nuclear reactors at Fukushima are not completely under control, it is unlikely that any populations will be returning to the evacuated areas by the end of this year. The provisional recommendation has raised concern that children might be allowed to receive the maximum level if they spent eight hours a day in certain schools and certain school yards. Accordingly, remediation has already begun by removing topsoil in those areas of radionuclide deposition, primarily cesium-137, that might result in levels approaching the 20 mSv per year limit.

It is unlikely that any radiation-related health effects will be detectable in the citizens surrounding the Fukushima Dai-ichi plant. As is often said in health research, the poison is in the dose (Paracelsus, 15th century physician) and it appears at this time that the population has not received meaningful exposures. This is because of the quick action taken by the Japanese authorities in evacuating citizens who lived close to the nuclear power plants, in monitoring and prohibiting the distribution of any food sources for which the levels of radioactivity were higher than standards, in screening the population, and in distributing masks and potassium iodide pills (or syrup for children) to the population. It was encouraging that one survey of over 1,000 children did not indicate any detectable levels of radioactivity that could be attributed to inhaled or ingested radioactive iodine, the major source of population exposure from the Chernobyl accident (Wakeford 2011). In large part, the winds blew towards the ocean and away from the populated areas, but nonetheless, there are areas near the Fukushima Dai-ichi power plant where radionuclide depositions are high and it will be some time before populations will be allowed to return. These evaluations and dose determinations are ongoing, and it appears that even some areas beyond the evacuation zone have unacceptably high levels that will have to be dealt with. On the other hand, the workers who have so effectively handled the radiation accident are experiencing radiation doses that will put them at increased risk for developing cancer later in life. Over 7,000 workers have been involved in the emergency and some 30 have received over 100 mSv external dose and a few over 600 mSv effective dose. The workers' exposures will continue to be of concern as they deal with the very high levels of radioactivity in the tens of thousands of gallons of radioactive water that need to be remediated, as well as the surrounding radioactive debris from reactor emissions and spent fuel (IAEA 2011; IRSN 2011). Finally, if the Chernobyl experience is used as a guide, the lasting effects upon the Japanese population will most likely be psychological with increased occurrence of stress-related mental disorders and depression associated not necessarily with the concern about reactor radiation, but with the horrific loss of life, evacuation from homes, separation from families and disruption caused by the tsunami and earthquake.

4. Are there any opportunities in the United States to study the effects of radiation when the exposure is received gradually and over long periods of time? Would such studies help in risk assessment and management decisions?

There are untapped opportunities in the United States to study the effects of radiation when the exposures are received gradually and over long periods of time. The early U.S. radiation workers and atomic veterans can be studied at relatively low cost by extending previous investigations that had ended some 20 to 30 years ago. The study populations include early Manhattan Project workers from the 1940s, atomic veterans who participated in nuclear weapons testing in the 1940s and 1950s, early nuclear power plant workers from the late 1950s, medical workers and others involved in the development of radiation technologies as well as nuclear navy personnel (Boice 2011). Pilot investigations have been initiative by the Low Dose Radiation Program within the Department of Energy (Hall et al. 2009) and this comprehensive work should continue. The effort is truly an interagency one with cooperating agencies including the National Cancer Institute, the Department of Defense, the Department of Veterans Affairs, and the Nuclear Regulatory Commission.

Such an important human radiation research effort would return the United States to a leadership role worldwide that it once held some years ago. The One Million Radiation Workers and Veterans Study would provide the needed answers to the major question remaining in radiation epidemiology: what is the level of risk associated with low-dose long-term exposure to radiation? Such a study would provide critically importance guidance in the area of risk assessment and management decisions. Currently, we rely upon the estimates from a Japanese population exposed in 1945 to atomic weapons and living in a war-torn country with associated problems of nutrition, hygiene, and health care. It is questionable whether these results can be validly generalized to United States populations of workers and citizens who are healthy, exposed differently and living during a different time. Yet extrapolation of health effects from the Japanese experience in 1945 to the United States circumstances in 2011 is what is done (BEIR 2006). The One Million Radiation Workers and Veterans Study is directly relevant to the setting of protection standards for workers and the public (ICRP 2009); risk assessment and management decisions; assessment of the possible risk from nuclear power plants and enhanced medical technologies such as CT and nuclear medicine imaging; the expansion of nuclear power; the handling of nuclear waste; the compensation of workers with prior exposure to radiation; and even the possible consequences of the radiation released from reactor accidents, such as at Fukushima.

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Appendix II

ADDITIONAL MATERIAL FOR THE RECORD

Statement of Peter B. Lyons
Assistant Secretary for Nuclear Energy
U.S. Department of Energy

Committee on Science, Space and Technology
Energy & Environment and Investigations & Oversight Subcommittees
U.S. House of Representatives

Nuclear Energy Risk Management Hearing
May 13, 2011

Chairman Broun, Chairman Harris, Ranking Member Edwards, Ranking Member Miller, and Members of the Committee, thank you for the opportunity to submit a statement on the Department of Energy's (DOE) Office of Nuclear Energy's R&D portfolio related to the safety of nuclear power plants.

The safety of our nuclear fleet is of paramount importance to this Administration, and we are committed to ensuring that nuclear plants in the United States continue to operate safely. As a former Commissioner of the Nuclear Regulatory Commission (NRC), I can attest to the dedication of the NRC staff and their focus on assuring safe operations. Many parameters tracked by NRC show the excellent safety of our plants and significant improvements in safety over the years.

The safety of nuclear plants is maintained through a series of barriers to prevent exposing the public to health-significant doses of radiation from release of radioactive materials. The principle of Defense-in-Depth is practiced in the industry, wherein multiple barriers are employed to provide this defense. Barriers in typical Light Water Reactors (LWR) include the fuel cladding, the reactor pressure vessel, and the containment building. Improvements in safety may result from increased robustness of existing barriers or from new barriers. Many programs of the Office of Nuclear Energy are focused on improvements in safety of operating and future new plants.

Passive Safety

Current generation II plants rely on extensive operator actions to place the plants into safe configurations in the event of off-normal conditions. In contrast, passive designs require far fewer operator actions and rely on extensive use of natural phenomena, like large quantities of stored water, gravity feed of cooling water rather than pumps, and convective cooling in accident conditions. They do not require operator actions for long periods after an accident, several days in some cases, and do not depend on offsite or emergency diesel generator power to maintain safety. Passive safety is one way of enhancing safety of nuclear plants. Another approach uses increased redundancy of active equipment. However, plants relying on additional active equipment may present challenges with the increased maintenance and operator actions required to keep the additional equipment in optimal condition and to effectively utilize it, and they depend on the availability of adequate power supplies. Plants with passive safety, in my opinion, offer the greatest promise for still safer operations in the future.

The global nuclear industry is currently building a new generation of reactors, most of which are so-called Generation III or III+ plants. These plants, among other features, use either passive safety or increased redundancy of active systems to enhance safety. In the United States, the Nuclear Power 2010 program, now concluded, was focused on bringing two Generation III+ plants, the Westinghouse AP1000 and the General Electric ESBWR, through design certification and issue of Construction and Operating Licenses (COLs). The Nuclear Regulatory Commission (NRC) is now in the final stages of evaluations of design certifications for both systems and COLs for the AP1000. Both of these plants make extensive use of passive safety features. NRC currently expects to be making final decisions regarding the AP 1000 design certification amendment and two combined licenses referencing that design certification amendment by the end of the calendar year. Final decisions regarding the ESBWR design certification are also expected by the end of the calendar year.

Fifteen licenses for these two passive reactor designs are currently pending at the NRC. Four AP1000 reactors are under construction in China. Extensive construction activities, short of safety-grade work, are in progress for two AP1000 reactors in Georgia and two in South Carolina.

Small Modular Reactors (SMRs) are another area of strong interest at DOE, and current designs offer some notable safety advantages. LWR SMR designs incorporate passive safety features that utilize gravity-driven systems rather than engineered, pump-driven systems to supply backup cooling in unusual circumstances. Because of their smaller size, current SMR designs have less water withdrawal requirements and some designs can make use of air cooling options. Some concepts use natural circulation for normal operations, requiring no primary system pumps and providing a still more robust safety case. In addition, many SMR designs utilize integral designs for which all major primary components are located in a single pressure vessel. That feature results in a much lower susceptibility to certain potential events, such as a loss of coolant accident, because there is no large external primary piping. Lastly, because of their lower power level, SMRs have a much lower level of decay heat and therefore require less cooling after reactor shutdown.

DOE has proposed the LWR SMR Licensing Technical Support program to help improve the timeline for the commercialization and deployment of light water SMRs and also proposed a longer-range program to conduct research on advanced SMR designs.

Improved Fuels and Cladding

All LWRs in the United States utilize uranium oxide pellets contained within zirconium alloy tubes or cladding. The cladding functions as one key barrier to confine the gaseous and volatile fission products that build up during reactor operation. But zirconium will react with steam at about 1,200 degrees C, generating hydrogen. If not properly controlled, hydrogen can ignite with oxygen leading to significant damage, as occurred in the Fukushima accident. Such damage may impair the cladding, releasing gaseous and volatile fission products into the reactor pressure vessel. Additionally, at somewhat lower temperatures, the zirconium cladding can distort and partially rupture, leading to some leakage of these radioactive materials. Similar scenarios can occur in pool storage of used fuel.

The Office of Nuclear Energy has been working with industry to develop a new silicon carbide ceramic cladding technology that could offer improved safety and performance as well as address additional current issues with zirconium cladding (such as rod fretting or abrasive damage to the cladding where cladding tubes pass through support structures) that results in occasional fuel failures in current plants. Ceramic cladding still needs substantial research and development efforts to determine if it can be successful. However, if successful, it has the potential to provide nuclear fuel with a higher tolerance to accidents. For example, silicon carbide cladding would not generate hydrogen in steam at high temperatures.

The sintered ceramic uranium oxide pellets used in current LWRs are very robust, but they do not retain volatile or gaseous fission products – they have significant porosity, and thus depend on the claddings for retention. An alternative fuel type was developed years ago for the original gas-cooled high temperature reactors, like Fort St. Vrain, but that fuel did not perform up to its original expectations. More recently we have revisited this fuel type (so-called TRISO fuel) with impressive results.

TRISO (TRi-ISOtropic) particle fuel has three layers surrounding a kernel of fissionable material like uranium oxide that act as primary, nearly impervious "containment" barrier for fission products during normal operations and accidents. This fuel retains its integrity and confines fission products up to extremely high temperatures. TRISO fuel has been suggested for utilization in other types of reactors, perhaps even in LWRs, because it would virtually eliminate release of fission products from the fuel itself. Significant issues would be associated with use of TRISO fuel in an LWR, and we plan to explore these issues in future research efforts.

High Temperature Gas Reactors (HTGR), as a class of reactors, can offer interesting safety features. Current concepts incorporate additional inherent physical characteristics that enhance safety and do not rely on active engineering systems or operator actions during accident scenarios. In addition to using the TRISO fuel, HTGRs can be cooled without the use of active systems which rely on electrical power for pumps, valves, instrumentation and control systems or operator actions. During such an event, it takes several days (2-4 days) for the TRISO fuel temperature to rise and peak at almost 1600°C, far slower than with LWR loss-of-coolant events. Even if the HTGR's control rods fail to insert during an accident, the TRISO fuel stays below its fuel failure temperature limits (<1800°C). Future HTGRs will be inherently, passively safe—no electricity, engineered safety systems or operator actions will be needed to control or cool the reactor.

The ongoing research and development (R&D) focuses on: (a) TRISO fuel development and irradiation performance experiments, post-irradiation evaluations (PIEs), and safety heat-up tests to demonstrate fission product retention, and (b) irradiation tests to demonstrate the strength of nuclear structural graphite at high temperatures. All R&D activities are being coordinated with the NRC to support the simultaneous development of regulatory requirements. DOE laboratories, universities, industrial collaborators and international experts are involved in the integrated R&D program activities.

Modeling and Simulation

The Office of Nuclear Energy has extensive programs building and utilizing modeling and simulation of complex nuclear-related phenomena, many of which have significant potential to improve safety. The most visible of these programs is the Energy Innovation Hub for Modeling and Simulation established by the Department through the Consortium for Advanced Simulation of Light Water Reactors (CASL) centered at Oak Ridge National Laboratory. The CASL collaboration is applying leading edge computational capabilities to create a new state-of-the-art in nuclear reactor simulations. The first set of problems to be tackled is related to fuel performance and safety in pressurized water reactors, in particular: pellet-clad interactions, CRUD (a term-of-art standing for Chalk River Unidentified Deposits, which plagued early reactors and occur today to lesser extents) deposition, and support grid-to-rod fretting. By improving our understanding of key mechanical, chemical, and nuclear interactions, the Hub may enable enhanced safety, prolonged life of nuclear fuel, increased power outputs, and enhanced reliability. The goal of the Hub is to have a highly realistic, virtual reactor capable of such simulations within its first five years. Less than one year after being established, the Hub has already released its first version of the virtual reactor code, which will undergo significant refinement in coming years.

As a multi-lab collaboration with industry and academia, the CASL mission is to create a useable tool set that is embraced by industry and researchers to advance the performance and safety of light water reactors. As such, the CASL vision is to embrace the full range of light water reactors including boiling water reactors and the new light-water-based small modular reactors that are being designed. These additional capabilities would be implemented during the first extension of the CASL award. The decision to extend the award or not will be made in 2015, subject to appropriations.

Seismic Evaluations

DOE-NE is currently involved in two projects related to seismic activity on nuclear power plants and plans to conduct further research on mitigating the effects of earthquakes. These two improved computational models will enable nuclear facility operators to determine the probabilistic seismic hazard at any point at ground level in the central and eastern United States with better accuracy:

(1) Generic Seismic Hazard Model for Central and Eastern United States (CEUS).

The objective of the ongoing CEUS Seismic Source Characterization Project, conducted by the Electric Power Research Institute, is to update the consensus seismic source hazards model for the CEUS based on new seismic activity data and comprehensive expert consensus on interpretation of said data. This model will be used to support nuclear plant site-specific Probabilistic Seismic Hazard Analyses (PSHA) anywhere in the CEUS. This project is scheduled to conclude by the end of 2011.

(2) Next Generation Attenuation – East (NGA-East) Seismic Project.
The NGA-East project is being conducted by the Pacific Earthquake Engineering Research Center at the University of California at Berkeley and will develop new ground motion attenuation relationships for the central and eastern U.S. Ground motion and the resultant attenuation of seismic motion is a function of soil type and rock structures at a site or location of interest as well as the location, depth, and magnitude of the seismic event. Ground motion attenuation as a result of an earthquake is a primary source of uncertainty in determining specific site seismic hazards. This project is scheduled for completion in 2014.

In addition to DOE funding, collaboration on these projects involves the Nuclear Regulatory Commission Office of Research, United States Geological Survey, and the nuclear industry through the Electric Power Research Institute.

In FY 2011, the Office of Nuclear Energy will solicit additional research and technology development on seismic isolation systems under its Nuclear Energy Enabling Technology program.

Dry Cask Storage

The present regulatory basis established by the NRC for dry cask storage is 60 years beyond the operating life of a reactor. This mode of storage is an integral part of the nation's current used fuel management and might be useful as an option for longer terms than the current regulatory limits. To evaluate such longer utilization, additional understanding of degradation mechanisms will be essential.

To help resolve issues associated with such extended used fuel storage, DOE is employing a competitive process to fund in Fiscal Year 2011 an Integrated Research Project (IRP) consortium consisting of a lead university, one or more partner universities, and potentially national laboratory and industry partners. Funding will be up to $1.5 million per year for up to 3 years and any potential industry partners would be required to cost share. This competition is now in progress.

The IRP focus will be on research and development (R&D) of accelerated aging techniques to better understand long-term degradation mechanisms, especially with high burnup used nuclear fuel (burnups above 45 gigawatt-days/metric ton). Specific issues to resolve include: long-term integrity of fuel cladding and canisters; maintaining fuel assembly configuration and associated components; canister leakage; hydride diffusion and embrittlement; creep, corrosion and stress corrosion cracking; accelerated degradation in a marine environment; and degradation of concrete. Project success will be measured by demonstration of laboratory-scale accelerated aging techniques that may eventually inform the technical basis providing technical justification for extended UNF storage of used nuclear fuel.

Detailed Modeling of the Fukushima Accident

The safety of nuclear power has been improved over the years by, among other approaches, careful learning from any accident or incident. Three Mile Island (TMI) led to major improvements in safety and new requirements for nuclear power plants in the United States. Modeling of TMI has helped enhance the current generation of severe accident modeling codes. Modeling and study of other accidents or near accidents, like Chernobyl or Davis-Besse, have led to further insights. The Fukushima accident must be understood in great detail in order to extract lessons that may further enhance safety or which may inform future regulatory decisions.

The Office of Nuclear Energy is organizing several of the DOE national laboratories for a joint study of accident progression at Fukushima. Our intent will be to learn from that analysis and to supplement any current analysis with future data that will become available once the damaged fuel can be examined. In addition, the implementation plans supporting the 2010 Nuclear Energy R&D roadmap are being studied by federal and laboratory staff to see if adjustments in R&D are needed in the aftermath of Fukushima.

Nuclear Energy University Program

The Office of Nuclear Energy devotes up to 20 percent of its R&D budget for research at universities. Much of this research has a significant nexus with safety issues, frequently related to further enhancement or understanding of the barriers that make up Defense-in-Depth. In addition, this research program plays an essential role in training the future generation of nuclear energy professionals upon whom the safety of future nuclear power systems will depend.

In Fiscal Years 2009 and 2010, NE funded via its Nuclear Energy University Programs (NEUP) a number of university-based safety-related research and development projects, including:

- Improved LWR Cladding Performance,
- Develop Advanced Models of LWR Pressure Vessel Embrittlement for Low Flux-High Fluence Conditions,
- Development of Diffusion Barrier Coatings and Deposition Technologies Mitigating Fuel Cladding Chemical Interactions,
- TRISO-Coated Fuel Durability Under Extreme Conditions,
- Evaluation of materials for interim storage of spent fuel for more than 100 years,

- Failure Predictions for Very High Temperature Reactor (VHTR) Core Components using a Probabilistic Continuum Damage Mechanics Model,
- Fission Product Transport in TRISO Particle Layers under Operating and Off-Normal Conditions,
- Multi-scale Concrete Modeling for Aging Degradation, and
- Investigation of Laser Shock Peening for Enhancing Fatigue and Stress Corrosion Cracking Resistance of Nuclear Energy Materials

In FY 2011, NE has solicited proposals for additional university-based, safety-related R&D projects in many areas, including:

- VHTR TRISO fuel development and qualification activities focused on producing robust fuel particles that can retain fission products during normal and accident conditions and have very low failure rates, as demonstrated by irradiation and accident safety testing programs,
- R&D addressing the Risk-Informed Safety Margin Characterization (RISMC) methodology,
- Development of new reactor concepts using advanced technologies or innovative engineering to provide improved safety and system performance, and
- R&D on sensors and infrastructure technology to address critical technology gaps to monitor and control new advanced reactors.

Conclusion

I began this statement with the well substantiated statement that the nation's nuclear reactors are safe today. But safety of nuclear energy, just as with any mature high technology endeavor, can always be improved through careful study and investigation. The Office of Nuclear Energy will maintain a focus on safe operations and on research and development that will provide still safer systems for future generations.

After the earthquake and tsunami in Japan, Deputy Secretary Poneman stated that: "We view nuclear energy as a very important component to the overall portfolio we are trying to build for a clean energy future." I fully concur with his view, and the programs of the Office of Nuclear Energy are focused on ensuring that the option for safe nuclear power remains open to the nation.