U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Fact Sheet

Office of Public Affairs
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Three Mile Island Accident

The accident at the Three Mile Island Unit 2 (TMI-2) nuclear power plant near Middletown, Pennsylvania, on March 28, 1979, was the most serious in U.S. commercial nuclear power plant operating history(1), even though it led to no deaths or injuries to plant workers or members of the nearby community. But it brought about sweeping changes involving emergency response planning, reactor operator training, human factors engineering, radiation protection, and many other areas of nuclear power plant operations. It also caused the U.S. Nuclear Regulatory Commission to tighten and heighten its regulatory oversight. Resultant changes in the nuclear power industry and at the NRC had the effect of enhancing safety.

The sequence of certain events - - equipment malfunctions, design related problems and worker errors - - led to a partial meltdown of the TMI-2 reactor core but only very small off-site releases of radioactivity.

Summary of Events

The accident began about 4:00 a.m. on March 28, 1979, when the plant experienced a failure in the secondary, non-nuclear section of the plant. The main feedwater pumps stopped running, caused by either a mechanical or electrical failure, which prevented the steam generators from removing heat. First the turbine, then the reactor automatically shut down. Immediately, the pressure in the primary system (the nuclear portion of the plant) began to increase. In order to prevent that pressure from becoming excessive, the pilot-operated relief valve (a valve located at the top of the pressurizer) opened. The valve should have closed when the pressure decreased by a certain amount, but it did not. Signals available to the operator failed to show that the valve was still open. As a result, cooling water poured out of the stuck-open valve and caused the core of the reactor to overheat.

As coolant flowed from the core through the pressurizer, the instruments available to reactor operators provided confusing information. There was no instrument that showed the level of coolant in the core. Instead, the operators judged the level of water in the core by the level in the pressurizer, and since it was high, they assumed that the core was properly covered with coolant. In addition, there was no clear signal that the pilot-operated relief valve was open. As a result, as alarms rang and warning lights flashed, the operators did not realize that the plant was experiencing a loss-of-coolant accident. They took a series of actions that made conditions worse by simply reducing the flow of coolant through the core.

Because adequate cooling was not available, the nuclear fuel overheated to the point at which the zirconium cladding (the long metal tubes which hold the nuclear fuel pellets) ruptured and the fuel pellets began to melt. It was later found that about one-half of the core melted during the early stages of the accident. Although the TMI-2 plant suffered a severe core meltdown, the most dangerous kind of nuclear power accident, it did not produce the worst-case consequences that reactor experts had long feared. In a worst-case accident, the melting of nuclear fuel would lead to a breach of the walls of the containment building and release massive quantities of radiation to the environment. But this did not occur as a result of the Three Mile Island accident.

The accident caught federal and state authorities off-guard. They were concerned about the small releases of radioactive gases that were measured off-site by the late morning of March 28 and even more concerned about the potential threat that the reactor posed to the surrounding population. They did not know that the core had melted, but they immediately took steps to try to gain control of the reactor and ensure adequate cooling to the core. The NRC's regional office in King of Prussia, Pennsylvania, was notified at 7:45 a.m. on March 28. By 8:00, NRC Headquarters in Washington, D.C. was alerted and the NRC Operations Center in Bethesda, Maryland, was activated. The regional office promptly dispatched the first team of inspectors to the site and other agencies, such as the Department of Energy and the Environmental Protection Agency, also mobilized their response teams. Helicopters hired by TMI's owner, General Public Utilities Nuclear, and the Department of Energy were sampling radioactivity in the atmosphere above the plant by midday. A team from the Brookhaven National Laboratory was also sent to assist in radiation monitoring. At 9:15 a.m., the White House was notified and at 11:00 a.m., all non-essential personnel were ordered off the plant's premises.

By the evening of March 28, the core appeared to be adequately cooled and the reactor appeared to be stable. But new concerns arose by the morning of Friday, March 30. A significant release of radiation from the plant's auxiliary building, performed to relieve pressure on the primary system and avoid curtailing the flow of coolant to the core, caused a great deal of confusion and consternation. In an atmosphere of growing uncertainty about the condition of the plant, the governor of Pennsylvania, Richard L. Thornburgh, consulted with the NRC about evacuating the population near the plant. Eventually, he and NRC Chairman Joseph Hendrie agreed that it would be prudent for those members of society most vulnerable to radiation to evacuate the area. Thornburgh announced that he was advising pregnant women and pre-school-age children within a 5-mile radius of the plant to leave the area.

Within a short time, the presence of a large hydrogen bubble in the dome of the pressure vessel, the container that holds the reactor core, stirred new worries. The concern was that the hydrogen bubble might burn or even explode and rupture the pressure vessel. In that event, the core would fall into the containment building and perhaps cause a breach of containment. The hydrogen bubble was a source of intense scrutiny and great anxiety, both among government authorities and the population, throughout the day on Saturday, March 31. The crisis ended when experts determined on Sunday, April 1, that the bubble could not burn or explode because of the absence of oxygen in the pressure vessel. Further, by that time, the utility had succeeded in greatly reducing the size of the bubble.

Health Effects

Detailed studies of the radiological consequences of the accident have been conducted by the NRC, the Environmental Protection Agency, the Department of Health, Education and Welfare (now Health and Human Services), the Department of Energy, and the State of Pennsylvania. Several independent studies have also been conducted. Estimates are that the average dose to about 2 million people in the area was only about 1 millirem. To put this into context, exposure from a full set of chest x-rays is about 6 millirem. Compared to the natural radioactive background dose of about 100-125 millirem per year for the area, the collective dose to the community from the accident was very small. The maximum dose to a person at the site boundary would have been less than 100 millirem.

In the months following the accident, although questions were raised about possible adverse effects from radiation on human, animal, and plant life in the TMI area, none could be directly correlated to the accident. Thousands of environmental samples of air, water, milk, vegetation, soil, and foodstuffs were collected by various groups monitoring the area. Very low levels of radionuclides could be attributed to releases from the accident. However, comprehensive investigations and assessments by several well-respected organizations have concluded that in spite of serious damage to the reactor, most of the radiation was contained and that the actual release had negligible effects on the physical health of individuals or the environment.

Impact of the Accident

The accident was caused by a combination of personnel error, design deficiencies, and component failures. There is no doubt that the accident at Three Mile Island permanently changed both the nuclear industry and the NRC. Public fear and distrust increased, NRC's regulations and oversight became broader and more robust, and management of the plants was scrutinized more carefully. The problems identified from careful analysis of the events during those days have led to permanent and sweeping changes in how NRC regulates its licensees - - which, in turn, has reduced the risk to public health and safety.

Here are some of the major changes which have occurred since the accident:

- Upgrading and strengthening of plant design and equipment requirements. This includes fire protection, piping systems, auxiliary feedwater systems, containment building isolation, reliability of individual components (pressure relief valves and electrical circuit breakers), and the ability of plants to shut down automatically;
- Identifying human performance as a critical part of plant safety, revamping operator training and staffing requirements, followed by improved instrumentation and controls for operating the plant, and establishment of fitness-for-duty programs for plant workers to guard against alcohol or drug abuse;
- Improved instruction to avoid the confusing signals that plagued operations during the accident;
- Enhancement of emergency preparedness to include immediate NRC notification requirements for plant events and an NRC operations center which is now staffed 24 hours a day. Drills and response plans are now tested by licensees several times a year, and state and local agencies participate in drills with the Federal Emergency Management Agency and NRC;
- Establishment of a program to integrate NRC observations, findings, and conclusions about licensee performance and management effectiveness into a periodic, public report;

- Regular analysis of plant performance by senior NRC managers who identify those plants needing additional regulatory attention;
- Expansion of NRC's resident inspector program - first authorized in 1977 - whereby at least two inspectors live nearby and work exclusively at each plant in the U.S to provide daily surveillance of licensee adherence to NRC regulations;
- Expansion of performance-oriented as well as safety-oriented inspections, and the use of risk assessment to identify vulnerabilities of any plant to severe accidents;
- Strengthening and reorganization of enforcement as a separate office within the NRC;
- The establishment of the Institute of Nuclear Power Operations (INPO), the industry's own "policing" group, and formation of what is now the Nuclear Energy Institute to provide a unified industry approach to generic nuclear regulatory issues, and interaction with NRC and other government agencies;
- The installing of additional equipment by licensees to mitigate accident conditions, and monitor radiation levels and plant status;
- Employment of major initiatives by licensees in early identification of important safety-related problems, and in collecting and assessing relevant data so lessons of experience can be shared and quickly acted upon;
- Expansion of NRC's international activities to share enhanced knowledge of nuclear safety with other countries in a number of important technical areas.

Current Status

Today, the TMI-2 reactor is permanently shut down and defueled, with the reactor coolant system drained, the radioactive water decontaminated and evaporated, radioactive waste shipped off-site to an appropropriate disposal site, reactor fuel and core debris shipped off-site to a Department of Energy facility, and the remainder of the site being monitored. The owner says it will keep the facility in long-term, monitored storage until the operating license for the TMI-1 plant expires at which time both plants will be decommissioned. Below is a chronology of highlights of the TMI-2 cleanup from 1980 through 1993.

Date	Event
July 1980	Approximately 43,000 curies of krypton were vented from the reactor building.
July 1980	The first manned entry into the reactor building took place.
Nov. 1980	An Advisory Panel for the Decontamination of TMI-2, composed of citizens, scientists, and State and local officials, held its first meeting in Harrisburg, PA.
July 1984	The reactor vessel head (top) was removed.
Oct. 1985	Defueling began.
July 1986	The off-site shipment of reactor core debris began.

Aug. 1988	GPU submitted a request for a proposal to amend the TMI-2 license to a "possession-only" license and to allow the facility to enter long-term monitoring storage.
Jan. 1990	Defueling was completed.
July 1990	GPU submitted its funding plan for placing $229 million in escrow for radiological decommissioning of the plant.
Jan. 1991	The evaporation of accident-generated water began.
April 1991	NRC published a notice of opportunity for a hearing on GPU's request for a license amendment.
Feb. 1992	NRC issued a safety evaluation report and granted the license amendment.
Aug. 1993	The processing of accident-generated water was completed involving 2.23 million gallons.
Sept. 1993	NRC issued a possession-only license.
Sept. 1993	The Advisory Panel for Decontamination of TMI-2 held its last meeting.
Dec. 1993	Post-Defueling Monitoring Storage began.

Additional Information

Further information on the TMI-2 accident can be obtained from sources listed below. The documents can be ordered for a fee from the NRC's Public Document Room at 301-415-4737 or 1-800-397-4209; e-mail pdr@nrc.gov. The PDR is located at 11555 Rockville Pike, Rockville, Maryland; however the mailing address is: U.S. Nuclear Regulatory Commission, Public Document Room, Washington, D.C. 20555. A glossary is also provided below.

Additional Sources for Information on Three Mile Island

NRC Annual Report - 1979, NUREG-0690
"Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station," NUREG-0558
"Environmental Assessment of Radiological Effluents from Data Gathering and Maintenance Operation on Three Mile Island Unit 2," NUREG-0681
"Report of The President's Commission on The Accident at Three Mile Island," October, 1979
"Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600
"Three Mile Island; A Report to the Commissioners and to the Public," by Mitchell Rogovin and George T. Frampton, NUREG/CR-1250, Vols. I-II, 1980

"Lessons learned From the Three Mile Island - Unit 2 Advisory Panel," NUREG/CR-6252

"The Status of Recommendations of the President's Commission on the Accident at Three Mile Island," (A ten-year review), NUREG-1355

"NRC Views and Analysis of the Recommendations of the President's Commission on the Accident at Three Mile Island," NUREG-0632

"Environmental Impact Statement related to decontamination and disposal of radioactive wastes resulting from March 28, 1979 accident Three Mile Island Nuclear Station, Unit 2," NUREG-0683

"Answers to Questions About Updated Estimates of Occupational Radiation Doses at Three Mile Island, Unit 2," NUREG-1060

"Answers to Frequently Asked Questions About Cleanup Activities at Three Mile Island, Unit 2," NUREG-0732

"Status of Safety Issues at Licensed Power Plants" (TMI Action Plan Reqmts.), NUREG-1435

Walker, J. Samuel, Three Mile Island: A Nuclear Crisis in Historical Perspective, Berkeley: University of California Press, 2004.

Other Organizations to Contact

GPU Nuclear Corp, One Upper Pond Road, Parsippany, NJ, 07054, telephone 201-316-7249;

Three Mile Island Public Health Fund, 1622 Locust Street, Philadelphia, PA, 19103, telephone 215-875-3026;

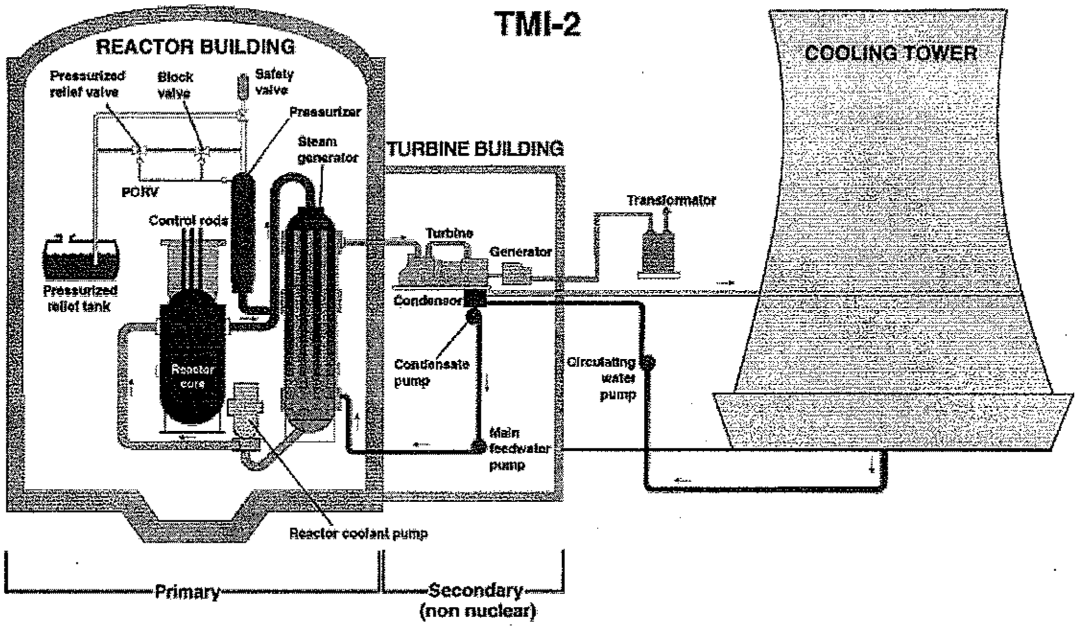
Pennsylvania Bureau of Radiation Protection, Department of Environmental Protection, Rachel Carson State Office Building, P.O. Box 8469, Harrisburg, PA, 17105-8469, telephone 717-787-2480.

March 2004

Glossary

Auxiliary feedwater - (see emergency feedwater)
Background radiation - The radiation in the natural environment, including cosmic rays and radiation from the naturally radioactive elements, both outside and inside the bodies of humans and animals. The usually quoted average individual exposure from background radiation is 360 millirem per year.
Cladding - The thin-walled metal tube that forms the outer jacket of a nuclear fuel rod. It prevents the corrosion of the fuel by the coolant and the release of fission products in the coolants.
Aluminum, stainless steel and zirconium alloys are common cladding materials.
Emergency feedwater system - Backup feedwater supply used during nuclear plant startup and shutdown; also known as auxiliary feedwater.
Fuel rod - A long, slender tube that holds fuel (fissionable material) for nuclear reactor use. Fuel rods are assembled into bundles called fuel elements or fuel assemblies, which are loaded individually into the reactor core.
Containment - The gas-tight shell or other enclosure around a reactor to confine fission products that otherwise might be released to the atmosphere in the event of an accident.
Coolant - A substance circulated through a nuclear reactor to remove or transfer heat. The most commonly used coolant in the U.S. is water. Other coolants include air, carbon dioxide, and helium.
Core - The central portion of a nuclear reactor containing the fuel elements, and control rods.

Decay heat - The heat produced by the decay of radioactive fission products after the reactor has been shut down.
Decontamination - The reduction or removal of contaminating radioactive material from a structure, area, object, or person. Decontamination may be accomplished by (1) treating the surface to remove or decrease the contamination; (2) letting the material stand so that the radioactivity is decreased by natural decay; and (3) covering the contamination to shield the radiation emitted.
Feedwater - Water supplied to the steam generator that removes heat from the fuel rods by boiling and becoming steam. The steam then becomes the driving force for the turbine generator.
Nuclear Reactor - A device in which nuclear fission may be sustained and controlled in a self-supporting nuclear reaction. There are several varieties, but all incorporate certain features, such as fissionable material or fuel, a moderating material (to control the reaction), a reflector to conserve escaping neutrons, provisions for removal of heat, measuring and controlling instruments, and protective devices
Pressure Vessel - A strong-walled container housing the core of most types of power reactors.
Pressurizer A tank or vessel that controls the pressure in a certain type of nuclear reactor.
Primary System - The cooling system used to remove energy from the reactor core and transfer that energy either directly or indirectly to the steam turbine.
Radiation - Particles (alpha, beta, neutrons) or photons (gamma) emitted from the nucleus of an unstable atom as a result of radioactive decay.
Reactor Coolant System - (see primary system)
Secondary System - The steam generator tubes, steam turbine, condenser and associated pipes, pumps, and heaters used to convert the heat energy of the reactor coolant system into mechanical energy for electrical generation.
Steam Generator - The heat exchanger used in some reactor designs to transfer heat from the primary (reactor coolant) system to the secondary (steam) system. This design permits heat exchange with little or no contamination of the secondary system equipment.
Turbine - A rotary engine made with a series of curved vanes on a rotating shaft. Usually turned by water or steam. Turbines are considered to be the most economical means to turn large electrical generators.



1. The catastrophic Chernobyl accident in the former Soviet Union, in 1986, was by far the most severe nuclear reactor accident to occur in any country; it is widely believed an accident of that type could not occur in U.S.-designed plants. For more detail on the accident at Chernobyl, see Fact Sheet at http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/fschernobyl.html.

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Environmental Monitoring

Background

The discharge of radioactive effluents from routine nuclear power plant operations can have environmental impacts--on man, animals, plants, and sea life. During the licensing of a nuclear plant, NRC issues a Final Environmental Statement (FES) which identifies these potential impacts. As part of NRC's requirements for operating a nuclear power plant, licensees must:

- keep releases of radioactive material to unrestricted areas during normal operation as low as reasonably achievable (as described in the Commission's regulations in 10 CFR Part 50.36a), and
- comply with radiation dose limits for the public (10 CFR Part 20).

In addition, NRC regulations require licensees to have various effluent and environmental monitoring programs to ensure that the impacts from plant operations are minimized. The permitted effluent releases result in very small doses to members of the public living around nuclear power plants.

Regulations

Current regulations to limit offsite releases and their associated radiation doses are much more restrictive than those required for nuclear power plants licensed in the 1960s. In 1975, the NRC amended its regulations (in 10 CFR Parts 50.34 and 50.36 and a new Appendix I) to provide numerical guides for design objectives and limiting conditions for operation to meet the radiation dose criterion "as low as is reasonably achievable." Adoption of these regulations requires that plant releases be kept to doses well below the radiation exposure limits for the public in 10 CFR Part 20.

In late 1979, the Environmental Protection Agency (EPA) placed an additional radiation dose requirement on reactor licensees. This requirement established total body, thyroid, and other organ dose limits for radioactive effluents and direct radiation. The NRC incorporated EPA's regulation into 10 CFR Part 20 in 1981, and all plants must now meet these requirements.

Monitoring Environmental Impacts

The NRC requires licensees to report plant discharges and results of environmental monitoring around their plants to ensure that potential impacts are detected and reviewed. Licensees must also participate in an interlaboratory comparison program which provides an independent check of the accuracy and precision of environmental measurements.

In annual reports, licensees identify the amount of liquid and airborne radioactive effluents discharged from plants and the associated doses. Licensees also must report environmental radioactivity levels around their plants annually. These reports, available to the public, cover sampling from TLDs (thermoluminescent dosimeters); airborne radioiodine and particulate samplers; samples of surface, groundwater, and drinking water and downstream shoreline sediment from existing or potential recreational facilities; and samples of ingestion sources such as milk, fish, invertebrates, and broad leaf vegetation.

The NRC conducts periodic onsite inspections of each licensee's effluent and environmental monitoring programs to ensure compliance with NRC requirements. The NRC documents licensee effluent releases and the results of their environmental monitoring and assessment effort in inspection reports that are available to the public.

Over the past 25 years, radioactive effluents released from nuclear power plants have decreased significantly. During the early part of that period, a significant contributor to the reduction was the addition of special systems (augmented offgas systems) to boiling water reactors, which process some of the noncondensible gases formed in the reactor process to limit the radioactive gases released to the environment. In recent years, improved fuel performance and licensees' improved effluent control programs further contributed to reducing radioactive effluents.

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Safety and Security Improvements at Nuclear Plants

Post 9-11 Actions

The Nuclear Regulatory Commission (NRC) - responsible for protecting public health and the environment from potential hazards involved in using nuclear materials - took prompt action to enhance safety and security, and has comprehensively re-evaluated security at nuclear power plants and other facilities it regulates.

Since September 11, 2001, NRC has strengthened security at nuclear facilities by working with national experts using state-of-the-art structural and fire analyses to realistically predict the consequences of terrorist acts. These studies confirm that, given robust plant designs and the additional enhancements to safety, security, and emergency preparedness and response, it is unlikely that significant radiological consequences would result from a wide range of terrorist attacks, including one from a large commercial aircraft.

Actions taken by Federal aviation safety and security agencies - Federal Air Marshals, reinforced cockpit doors, airport passenger and baggage screening, improved ability to detect deviation from planned flight paths and greater military aircraft intercept capability - have reduced the likelihood that large commercial aircraft could be used to attack critical infrastructure, including a nuclear facility. Other actions, such as improved communication between military surveillance authorities, NRC, and its licensees, would allow plant operators to prepare the plant for safe shutdown should it be necessary. These actions, coupled with those taken by the NRC and the nuclear industry, are an integral part of the government's overall strategy for protecting the nation's critical infrastructure.

NRC has strengthened requirements at nuclear power plants and enhanced coordination with Federal, State and local organizations since 9-11

NRC major actions include:

- Ordered plant owners to sharply increase physical security programs to defend against a more challenging adversarial threat;
- Required more restrictive site access controls for all personnel;
- Enhanced communication and liaison with the Intelligence Community;
- Ordered plant owners to improve their capability to respond to events involving explosions or fires;

- Enhanced readiness of security organizations by strengthening training and qualifications programs for plant security forces;
- Required vehicle checks at greater stand-off distances;
- Enhanced force-on-force exercises to provide a more realistic test of plant capabilities to defend against an adversary force; and
- Improved liaison with Federal, State, and local agencies responsible for protection of the national critical infrastructure through integrated response training.

Safety and security studies show that a radiological release affecting public health and safety is unlikely from a terrorist attack, including large commercial aircraft

- Power plants are among the most hardened commercial structures in the country and are designed to withstand extreme events, such as hurricanes, tornadoes, and earthquakes;
- Power plants have redundant safety systems and are operated by highly trained staff;
- Multiple barriers protect the reactor and prevent or minimize off-site releases;
- With mitigation strategies and measures in place, the probability of damaging the reactor core and releasing radioactivity that could affect public health and safety is low;
- Significant releases due to a terrorist attack on a spent fuel pool are very unlikely;
- It is highly unlikely that a significant release of radioactivity would occur from a dry spent fuel storage cask; and
- No release of radioactive material is expected from an aircraft attack on a transportation cask.

Time is available to protect the public in unlikely event of a radiation release

- If a radiation release did occur, there would be time to implement mitigating actions and offsite emergency plans at power plants, spent fuel pools, and dry-cask storage installations; and
- Safety and security studies confirm that NRC's emergency planning basis remains valid.

NRC has taken action to strengthen security and safety

Increased aviation security and aggressive NRC action provide enhanced protection against terrorist attacks

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Storage of Spent Nuclear Fuel

What is Spent Nuclear Fuel?

Spent nuclear fuel refers to uranium-bearing fuel elements that have been used at commercial nuclear reactors and that are no longer producing enough energy to sustain a nuclear reaction. Once the spent fuel is removed from the reactor the fission process has stopped, but the spent fuel assemblies still generate significant amounts of radiation and heat.

For years, nuclear power plants have temporarily stored spent nuclear fuel in water-filled pools at the reactor site. The NRC has also authorized nuclear power plant licensees to store spent fuel at reactor sites in NRC-approved dry storage casks. Until a permanent repository for spent fuel and other high-level nuclear waste is available, spent nuclear fuel continues to be stored primarily in specially designed, water-filled pools and NRC-approved dry casks at individual reactor sites around the country. Periodically, about one-third of the nuclear fuel in an operating reactor needs to be unloaded and replaced with fresh fuel.

NRC regulations require stringent design, testing and monitoring in the handling and storage of spent nuclear fuel

The Nuclear Regulatory Commission (NRC) is an independent regulatory agency whose primary mission is to protect public health and safety, the common defense and security, and the environment in the use of nuclear materials. The agency regulates the possession, transportation, storage and disposal of spent fuel produced by nuclear reactors.

- For approval of cask designs, the NRC conducts tests and performs extensive analyses to ensure designs are safe and secure for use at any licensed nuclear power plant site in the country.
- The NRC's regulations are developed through a public process and provide a sound basis for determining whether use of a proposed storage system will protect public health and safety and the environment.
- The NRC regularly inspects the design, construction, and use of spent fuel pools and dry casks to ensure licensees and vendors meet NRC's radiation safety and security requirements.

Spent nuclear fuel pools adequately protect spent nuclear fuel

- Spent fuel pools are strong structures constructed of very thick steel-reinforced concrete walls with stainless steel liners located inside protected areas.
- Many fuel pools are located below ground level, many are shielded by other structures, and many have intervening walls that would obstruct an aircraft's or other object's impact.
- Spent fuel pools contain enormous quantities of water. Nuclear plants possess many other sources of water as backup supplies to the spent fuel pool.
- NRC has ordered licensees to develop guidance and strategies to maintain and restore spent fuel pool cooling using existing or available resources if cooling is lost for any reason. For many events, plant operators would have significant time to correct a problem, or implement fixes to restore cooling.

Spent nuclear fuel storage in casks is safe and environmentally sound

- Casks typically consist of a sealed metal cylinder containing the spent fuel enclosed within a metal or concrete outer shell. In some designs, casks are placed horizontally; in others, they are set vertically on a concrete pad.
- Casks are designed to resist situations such as floods, tornadoes, projectiles, and temperature extremes.
- Typically, the maximum heat generated in an hour from 24 fuel assemblies stored in a cask is less than that given off by a typical home heating system for the same amount of time.

Spent Nuclear Fuel Storage Facilities protect against sabotage, theft, and diversion

- The NRC sets the requirements and assesses compliance with the requirements, the licensees are responsible for providing the protection.
- The NRC has a threat assessment program to maintain awareness of the capabilities of potential adversaries and threats to facilities, material, and activities.
- The NRC's domestic safeguards program is focused on physically protecting and controlling spent nuclear fuel, against sabotage, theft, and diversion.
- Key features of the physical protection programs for spent nuclear fuel storage facilities include:
  - intrusion detection;
  - assessment of detection alarms to distinguish between false or nuisance alarms and actual intrusions and to initiate response;
  - response to intrusions; and
  - offsite assistance, as necessary, from local, State, and Federal agencies.
- Over the last 20 years, there have been no radiation releases which have affected the public and no known or suspected attempts to sabotage spent fuel casks or storage facilities.
- The NRC responded to the terrorist attacks on September 11, 2001, by promptly developing and requiring security enhancements for both spent nuclear fuel storage in spent nuclear fuel pools and dry casks.

NRC has taken action to ensure the safe and secure storage of spent nuclear fuel

April 2005

Comparison of the Reactor Oversight Process to the Independent Safety Assessment of Maine Yankee Atomic Power Company

1. Introduction

An Independent Safety Assessment (ISA) of Maine Yankee Atomic Power Company was performed in 1996. Since that time, many changes have occurred in the NRC regulatory oversight of the Nation's nuclear power plants, including the creation of the NRC's Reactor Oversight Process (ROP). This analysis provides a brief description of the events leading up to the ISA, describes the current ROP, and provides a comparison of the ISA to the ROP and other applicable regulatory processes.

2. Timeline of the Maine Yankee Events Leading Up to the ISA

The Maine Yankee (MY) facility was licensed in 1972 at 2440 megawatts thermal (MWt) power. In 1977, the NRC approved MY's application for a power uprate to 2630 MWt. In 1988, MY applied for a power uprate to 2700 MWt, which was approved in 1989. In December 1995, an allegation was made that the Yankee Atomic Electric Company (YAEC), acting as an agent for MY, had knowingly performed inadequate analyses to support the increase in power to 2700 MWt and further that the NRC staff may not have appropriately reviewed the MY power uprate request. The subsequent investigation by the Nuclear Regulatory Commission's (NRC's) Office of the Inspector General identified problems with the YAEC's use of computer codes as part of the power uprate analysis as well as weaknesses in the NRC review of the power uprates.[1] A confirmatory order was issued to MY limiting power operation to 2440 MWt. The regulatory oversight program at that time allowed for special inspections as a part of the process, called Diagnostic Evaluation Team (DET) inspections. In response to the above concerns, as well as those expressed by the Governor of the State of Maine, the NRC Chairman directed that an ISA be conducted.

The ISA was started in July 1996 and completed in October 1996. It focused on conformance of the facility to its design and licensing bases, operational safety performance, licensee self-assessments, corrective actions and improvement plans, and determination of the causes of safety-significant findings.

The MY ISA was unique in its scope, independence, and in its coordination with state representatives. The ISA was a modified DET that added a detailed review of analytic codes for transient and accident safety analyses. As noted in the ISA, use of application analytic codes was not typically inspected as part of the NRC regulatory process at the time, and additional focused resources were applied to this area. However, review of the codes was necessary specifically to address the allegations made against YAEC. While the exact data is no longer available, it is estimated that the ISA expended approximately 4000 hours (25 people times 4 weeks) of on-site inspection, where a typical DET expended approximately 1800 hours (15 people times 3 weeks) of on-site inspection. The difference in the on-site hours is directly related to the size of the ISA team (additional inspectors to address the highly technical and detailed allegation related to transient and accident safety analyses codes), the number of state representatives (3 on the ISA), and the extra week of on-site inspection. This can be compared to the ROP today, which utilizes approximately 2500 hours of on-site inspection time annually for a well performing single unit site. Under the ROP today, poorly performing plants may receive up to an additional 2000-2500 hours of inspection.

3. ISA Results

The results of the 25-member team inspection were that the licensee's performance was considered adequate for operation. There were a number of findings in the final report, many of which would be considered minor under today's more risk-informed ROP and would not be documented in an inspection report. However, the significant results were summarized as weak identification and resolution of problems; weak scope, rigor, and evaluation of testing; and declining material condition. These problems were caused in part by economic pressure to be a low-cost producer, which limited resources to address problems, and the lack of a questioning culture resulting in the failure to identify or promptly correct significant problems. The findings did not warrant or require a shutdown of the facility.

In December of 1996, the licensee shut down the plant. Soon afterward, the NRC issued a Confirmatory Action Letter (CAL) requiring specific actions to address licensee-identified safety-system electrical separation issues and logic circuit testing deficiencies. Follow-up inspections identified problems in five major categories: inoperability of safety related equipment, and inadequacies in testing, safety review, procedures, and corrective actions. Additional design and configuration control problems were identified by NRC inspectors and the licensee in 1997. Because of these and other economic considerations, the plant's owners voted to permanently shut down the reactor in August 1997. The diverse owners of the plant decided not to make the investments needed to restore the plant to good performance. The owners of other plants with similar (or, in some cases, worse) problems but with different ownership structure and different corporate governance chose to make the investments necessary to restore their plants' performance.

(Throughout the balance of this document many references are made to procedures used in the inspection process. To maintain brevity, in most cases these procedures are not called out in the body but are referenced with endnotes. A more detailed description of the ROP and links to Inspection Procedures can be obtained on the NRC web site: http://www.nrc.gov/reactors/operating/oversight.html.)

4. Description of the ROP

The reactor oversight process is anchored in the NRC's mission to ensure public health and safety in the operation of commercial nuclear power plants. To measure plant performance, the oversight process focuses on seven specific "cornerstones" that support the safety of plant operations: initiating events, mitigating systems, barrier integrity, emergency preparedness, occupational radiation safety, public radiation safety, and physical protection. These cornerstones are evaluated using both performance indicators (PIs) and direct inspections. The NRC assessment program collects information from inspections and performance indicators in each cornerstone to enable the NRC to arrive at objective conclusions about the licensee's safety performance. Inspection findings are evaluated for safety significance using a generally objective significance determination process. Performance indicator data is compared against prescribed risk-informed thresholds.

Based on this assessment information, the NRC determines the appropriate level of agency response, including supplemental inspections focusing on areas of declining performance and pertinent regulatory actions ranging from management meetings to orders for plant shutdown. The process uses four levels of regulatory response, with NRC regulatory review increasing as plant performance declines. The first two levels of heightened regulatory review are managed by the appropriate NRC regional office. The next two levels call for an agency response and involve senior management attention from both headquarters and regional offices. The scope of inspections are driven by plant performance. A poor performing plant having multiple or long-standing significant issues will be inspected using a procedure that incorporates processes and techniques originally used in the previous DET process that was applied at MY.[2] For comparison purposes, in 2006 there were three plants receiving increased regulatory attention. In each case, the plant warranted this major increase in NRC oversight because plant performance had met specific pre-defined criteria.

Even if there are no earlier signs of declining plant performance, should a plant experience operational problems or events that the NRC believes require greater scrutiny, there will be additional reactive inspections. The criteria for initiating these reactive inspections are described in the publicly available NRC Management Directive (MD) 8.3, "NRC Incident Investigation Program,"[3] and are typically used about a dozen times per year. In some instances the regulatory actions dictated by the ROP framework may not be appropriate. In these instances, the NRC may deviate from the prescribed program to allow modified regulatory oversight for a facility based on specific circumstances. Historically there have been 1-3 deviations each year. Use of the deviation process requires senior NRC management approval.

5. NRC Independence

The MY ISA used independent NRC inspectors to perform the assessment. The large multi-disciplined team was composed of 25 members, including three state representatives. To ensure independence, the NRC members were selected from offices other than the Office of Nuclear Reactor Regulation or Region I; six contractors were also part of the team. Only persons with no significant prior responsibility for regulating Maine Yankee were chosen.

In the ROP, the most closely related process to the ISA are the three levels of supplemental inspections. For a poor performing plant with multiple or repetitive degraded cornerstones, the highest level of supplemental inspection effort will be used.[4] This inspection has several objectives, including providing an independent assessment to aid in the determination of whether an unacceptable margin of safety exists; assessing the adequacy of licensee programs to identify, evaluate and correct performance issues; providing insight into the root and contributing causes of performance deficiencies; and independently assessing the licensee's safety culture. These objectives are taken together to provide additional information to be used in deciding whether continued operation of the facility is acceptable and whether additional regulatory actions are necessary to arrest declining plant performance. The inspection team is staffed, in part, with inspectors from other regions or headquarters to give a degree of independence to the effort. The approximate numbers of inspection hours for the three levels of supplemental inspections are, in increasing order, 24, 240, and 2400.

Another type of inspection is the reactive inspection described in MD 8.3, which is used to investigate incidents at plants. The scope and depth of the inspection is predicated on the significance of the event being investigated, with Incident Investigation being the highest level, followed by Augmented Inspection and then Special Inspection. Similar to the MY ISA, incident investigation team inspections require that the inspection team be composed of members who have not been significantly involved in the licensing and inspection of the facility. Approximate numbers of inspection hours for these efforts are similar to, in increasing order, the supplemental inspections in the preceding paragraph.

The concept of independence is institutionalized in NRC routine procedures and practices. Inspectors are not allowed to own securities, such as company stock, that could cause a conflict of interest during an inspection. NRC employees who have previously worked for a licensee (including the parent companies) are not assigned to inspect those facilities for at least a one year period, and this time frame may be extended according to individual office policy.

In addition to inspections conducted by inspectors located at the regional office, at least two resident inspectors are assigned full-time to each site. To maintain independence, the maximum time a resident inspector can be assigned to a site is seven years unless a longer period specifically approved by the EDO. The ROP inspections are also divided so that regional office-based inspectors perform a portion of the required inspection program independent of the resident inspectors and their associated management chain. Management site visits are conducted on a routine basis to assess the adequacy of the inspection effort. Finally, inspectors from headquarters or the regions are at times assigned to inspect plants in other regions.

The NRC also provides additional independence through the use of contractors. The NRC typically hires two contractors for all Component Design Basis Inspections. These contractors must be cleared concerning any potential conflict of interest.

In addition, the NRC allegation process allows individuals, including plant employees, to bring safety concerns directly to the NRC. Overall, the necessary level of inspector independence from the licensee is maintained by the processes and procedures described above.

6. Conformance to Design and Licensing Basis

During the ISA conducted at MY, the inspection team conducted an in-depth review of the plant's conformance to the design and licensing-basis. Because of allegations regarding computer codes used to justify previously approved power uprates, significant attention was placed on the transient and accident safety analyses. Further inspections focused on design review of two plant safety systems and their associated support systems (including the electrical system) to evaluate the ability of the systems to perform their design basis safety functions.

The current ROP provides a broad and in-depth ongoing assessment of licensee performance, as described below, using various inspection procedures and the performance indicator program. Specifically, the ROP verifies that the safety system design and performance are being maintained to the approved design and licensing bases. The extensive focus of the MY ISA on computer codes and transient analysis was specific to the allegations at the time and is not now normally the focus of routine baseline inspections. Currently, when a power uprate request is made, the NRC would review changes to computer codes used to justify the uprate and transient analyses affected by the power uprate.

a. Transient and Accident Safety Analyses

Facility operating licenses specify the maximum power level at which commercial nuclear power plants may be operated; NRC approval is required to increase the maximum power output of the facility. The NRC evaluates the licensee's operational, transient, and accident analyses that are affected by the uprate as part of the license amendment process prior to approving a power uprate (license amendment). The staff also reviews any changes made to the computer codes. The staff's power uprate review confirms that the reactor can be safety operated at the new power level. The staff's review focuses on multiple areas, including the nuclear steam supply system, instrumentation and control systems, electrical systems, accident evaluations, radiological consequences, operations and training, testing, and technical specification changes.

Transient and accident analyses were performed by YAEC to support two power uprates for MY. The NRC power uprate review process has improved significantly since the MY ISA. Power uprates are now controlled by two guidance documents, RS-001, "Review Standard for Extended Power Uprates," and RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."[5] As part of the review process, the NRC reviews changes to computer codes used to justify the power uprate.

Inspection Procedure (IP) 71004, "Power Uprates," is used to inspect facilities that increase their licensed power level by greater than 7.5 percent of original plant rated output. IP 71004 verifies that the licensee has taken the required actions to alleviate or prevent the effects of new or likely initiating events that were due to changes such as higher core power densities or increased flow in primary or secondary systems. IP 71004 also triggers other inspections to review design bases and safety margins, performance of heat exchangers for mitigating systems, erosion and corrosion programs, and modifications.[6] Power uprates of less than 7.5 percent also receive field inspections in the normal course of the ROP implementation.

b. Design Review of Selected Systems Including Electrical and Control Systems

The MY ISA focused on two safety systems (high-pressure safety injection and service water/component cooling water), the electrical power systems, and the instrumentation and controls area. These types of systems are also covered by the ROP in-depth as described below.

Currently, the NRC evaluates safety-related systems using several different inspection procedures to ensure the design adequacy and the ability of the systems to perform their intended functions. Design adequacy and system component margin of safety are evaluated under the NRC's inspection program during the Component Design Basis Inspection (CDBI).[7] The ISA team specifically evaluated the high-pressure safety injection, service water and component cooling water systems, the MY off-site power capability, the station batteries, the back-up emergency diesel generators, and the environmental qualification of components to determine their adequacy. Using the current process, each of these areas have potential components that can be selected for review during the CDBI. The component selection is based on risk significance and low margin relative to the design or licensing basis. The inspection is performed biennially at each operating reactor and represents approximately 650 hours of NRC direct inspection effort and additional effort for preparation and documentation time. The team consists of three engineering inspectors, one operations inspector, and typically two independent contractor design specialists with expertise in the mechanical and electrical disciplines. The team reviews the adequacy of 15-20 components typically covering 4-6 systems. The inspection includes reviews related to configuration control, design calculations, component testing, environmental qualification, and electrical component inputs/outputs.

Prior to the NRC's implementation of the CDBI in 2006, the Safety System Design Inspection (SSDI) was performed. This biennial inspection was similar to the ISA format, and two safety-related systems received a detailed inspection. The inspection was conducted by a six-person team that included a contractor about half of the time.

On a more frequent basis, the components inspected during the CDBI, as well as other safety related systems and components, are evaluated for their continued operability based on surveillance testing[8]; post maintenance testing[9]; operability determinations[10]; and modifications made to the system, component, or licensing basis.[11]

c. FSAR Inconsistencies

Discrepancies in the licensee's Final Safety Analysis Report (FSAR) were identified as a result of the MY ISA inspection. Currently, inspector preparation for most inspection procedures includes reviewing the FSAR; therefore, discrepancies may be identified during this process. Typically, FSAR inconsistencies are not significant and are not documented as findings. However, more importantly, the CDBI inspection focuses heavily on the licensing and design basis documentation and supporting calculations. Also, the licensee may make changes to the facility. Inspectors have a specific procedure to use when reviewing plant modification screenings, evaluations, and the resulting FSAR change to ensure their appropriateness.[12]

In summary, the ROP today includes elements in the baseline inspection program to assess key safety systems and conformance to the design and licensing basis either by inspection or with the performance indicator program. The MY ISA addressed transient analyses and codes to address allegations made regarding use of the codes. As noted earlier, transient analyses and related codes are not normally inspected as part of the ROP, and it was noted in the ISA that they were not normally addressed by the regulatory process at that time. However, when power uprates are now requested by licensees, affected transient analyses and changes to codes are reviewed, and current procedures typically require inspection of many other potentially impacted systems. Lessons learned from the MY ISA in the area of power uprates have been institutionalized to ensure similar problems do not recur.

7. Assessment of Operational Safety

Operational safety was inspected by the MY ISA team. The review included problem identification and resolution (PI&R); quality of operations; operational programs and procedures; and plant support programs related to operator training, radiation protection, and fire protection.

Assessment of operational safety in the ROP is done continuously by resident inspectors as well as periodically by regional inspectors using inspection procedures. Current procedures and practices are described below that compare significant aspects from the ISA to the current ROP. PI&R will be addressed in a later section.

The quality of operations is currently inspected by daily control room observations and inspector attendance at selected licensee meetings.[13] Continuous control room observations are not routinely performed; should concerns arise, a specific procedure exists for inspectors to use.[14] Safety system walkdowns are specifically performed by using two procedures as well as the requirement for the resident inspector to be cognizant of the plant status.[15] Additionally, most procedures require inspectors to enter the plant to perform the inspection and therefore observe ongoing activities and the material condition of the plant.

The MY ISA report discusses the team's effort to review Technical Specification (TS) interpretations. Inspectors monitor licensee compliance to the TS action statements, requirements, and license conditions as part of the plant status procedure.[16]

Online risk management and shutdown risk are evaluated using two procedures written for that specific purpose.[17] These procedures require inspectors to review the status of risk significant equipment and determine if the site has taken appropriate actions to reduce the overall station risk while equipment is out of service. When there are concerns regarding safety equipment performance, the licensee documents the operability of the equipment. These operability evaluations are inspected by the resident staff with a specific procedure written for that purpose; typically 15-30 reviews are performed per year per site.[18]

Operating procedures assessed during the MY ISA are also reviewed today as part of the CDBI inspection. The CDBI inspection procedure requires that several risk-significant operator actions be evaluated.[19] These actions are typically found in abnormal or emergency operating procedures. Additionally, regional inspectors and resident inspectors both use inspection procedures in the observation of simulator training scenarios to evaluate operator response to events and determine if procedures are adequate to address accident scenarios.[20]

The configuration control program is assessed using two inspection procedures, both of which are performed annually.[21] The Restart Readiness Program and the Operations Performance Assessment Program were programs specific to Maine Yankee and are therefore not currently part of the NRC baseline inspection program. However, a plant that is in a shutdown condition due to significant performance problems or operational events may be placed under the process prescribed in Inspection Manual Chapter (IMC) 0350, "Oversight of Reactor Facilities in a Shutdown Condition Due to Significant Performance and/or Operational Concerns." This IMC provides adequate assurance that a licensee that was placed into the IMC 0350 process is ready for a return to plant operation, and under this process, a plant's restart program would be reviewed.

The licensee's other plant support programs, such as Fire Protection, Radiation Protection, and Operator Training Programs, are all inspected by NRC inspectors using a number of specific inspection procedures for each program.[22] The frequency of performance of these inspection procedures ranges from quarterly to triennially.

Overall, the ROP provides a thorough assessment of station operational performance, including quality of operations, operational programs and procedures, and plant supporting programs, such as fire protection, training, and radiation protection.

8. Maintenance and Testing Assessment

The MY ISA reviewed maintenance and testing activities at the site. The team identified several issues, particularly in the testing area. Equipment performance, PI&R, quality of maintenance, and maintenance work order control were also discussed in the ISA report. The current ROP uses a number of procedures to evaluate each of these areas as described below. PI&R will be addressed in a later section.

Surveillance testing is inspected to verify satisfactory equipment performance by observing and reviewing the results of the surveillance tests. The surveillance tests acceptance criteria are also reviewed to verify that the licensing and design requirements are satisfied. The resident inspector staff performs quarterly reviews of 5-6 surveillance tests.[23] Additionally, CDBI teams review several years of test data on the selected components (typically pumps or valves). Integrated leak rate testing of valves to verify containment integrity is inspected during refueling outages.[24]

Also, failures of key safety systems are reported quarterly through the performance indicator program. The availability and reliability of safety systems reported on by the PIs include emergency AC power, high pressure injection, heat removal, residual heat removal and cooling water.[25] Inspectors verify that the licensee accurately reports the performance indicators. Should there be a discrepancy in reporting that cannot be readily resolved, the NRC performs an additional inspection to gather the performance indicator data.[26] The inspections and performance indicators are used together to ensure that safety system performance is assessed and indications of declining performance are identified for additional inspections.

Post maintenance tests (PMT) verify that equipment is operable prior to returning the equipment to service. The NRC has a specific procedure[27] for review of PMTs, and failure of equipment is evaluated by the licensee under 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Inspectors review the licensee conclusion as to the cause of the failure and the adequacy of current maintenance practices.[28]

Maintenance work order control as it relates to overall plant risk is inspected by the resident staff to ensure that the risk is fully understood by plant personnel prior to changing plant configuration.[29] Additionally, shutdown risk management is evaluated by the resident staff during refueling or forced outages.[30] In both cases, the NRC Regional Senior Reactor Analyst supports the evaluation.

In summary, the ROP has inspection procedures in place that are routinely used to assess the areas covered by the Maine Yankee ISA, including equipment performance, quality of maintenance, testing, and work order control as described above.

9. Engineering Assessment

General conclusions on problem identification and resolution (PI&R), the engineering programs, design basis information, and the quality of engineering were reached and reported on by the ISA team. The ROP includes a thorough set of inspections that encompass the MY ISA reviewed areas. PI&R will be discussed in a later section.

Engineering programs and the quality of engineering are reviewed as part of the review of modifications. Modifications are inspected by resident staff and regional inspectors to ensure that the modification maintained the design and licensing basis.[31] Service water systems are inspected by resident staff and regional inspectors to ensure that the components meet the requirements reiterated in Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."[32] Other inspection procedures encompass service water components.[33] Erosion/corrosion issues are currently inspected on a refueling outage basis.[34]

Design basis information is frequently reviewed during several NRC inspections, but most notably during the CDBI and to a lesser extent when reviewing modifications. These inspections are all performed biennially under the current inspection program, as described above, and are used to evaluate the quality of the engineering work performed.

The CDBI probes heavily into the engineering area to ensure compliance with the design and licensing basis, including review of calculations and design margin. This has already been described in an earlier section. In summary, the ROP has procedures and processes to assess licensee performance that encompass the ISA scope, which included engineering programs, the design basis, and the quality of engineering.

10. Self Assessment, Corrective Actions, Planning & Resources

The MY ISA inspection report included several sections that discussed the licensee's ability to self-assess and identify and correct problems. A specific section in the report discussed the adequacy of the corrective action program. This area has become one of the most important areas of inspection and is a significant focus for inspectors in recent years.

A fundamental goal of the ROP is to establish confidence that each licensee is detecting and correcting problems in a manner that ensures nuclear safety. One specific PI&R inspection objective is to provide for early warning of potential performance issues that could result in heightened NRC attention due to declining performance. As a result, significant inspection effort is devoted under the ROP to PI&R.

The current ROP inspects the licensee's corrective action program in every inspection procedure. Each procedure requires that 10 percent of the inspector's effort is focused on problem identification and resolution. As part of every inspection procedure, inspectors are tasked with identifying issues that the licensee has missed. Inspectors routinely perform plant and equipment walkdowns to facilitate this requirement. Additionally, inspectors review the licensee's evaluation of, and corrective actions for, selected identified deficiencies.

The licensee's corrective action program is specifically inspected by IP 71152, "Problem Identification and Resolution." This procedure is used to inspect the licensee's program at various points in the corrective action process as well as the licensee's self-assessments. Each day, resident inspectors are required to screen all written reports of licensee identified deficiencies or condition reports (CR). The purpose of the review is to provide early warning of potential performance issues. The resident inspectors evaluate all the CRs semi-annually to identify any trends in the identified deficiencies.

The second section of this inspection procedure requires that 3-6 specific deficiencies (samples) receive an in-depth review to assess the adequacy of the licensee's actions to correct the deficiencies and self-assessments. These samples are selected by resident inspectors, and the selection may have a supervisory review. Both resident and region-based inspectors perform this portion of the inspection.

The third portion of this inspection procedure requires that an inspection team perform a biennial in-depth review of the licensee's corrective action program. The four-member team conducts a two-week inspection that samples all seven of the ROP cornerstones. Additionally, the team assesses the adequacy of the licensee's corrective actions for all NRC-identified findings since the previous PI&R team inspection was performed.

The Service Water Operational Performance Inspection (SWOPI), which was performed by MY staff, was commented on by the ISA team in their report. This was a licensee-specific self-assessment. Self-assessments are reviewed during the PI&R inspection. The service water system, in general, is a safety system inspected under the ROP by a number of inspection procedures as noted previously.

The MY ISA also assessed the licensee in the area of planning and resources. The NRC assesses all NRC findings and violations to determine if there are cross-cutting aspects associated with the issue. Inspectors determine if a cross-cutting issue exists by evaluating the apparent or root cause of the issue. If the issue is determined to be caused by a problem identification or resolution failure, human performance failure, or a safety conscious work environment issue, it can be considered to have a cross-cutting aspect. The human performance area includes evaluating problems caused by lack of resources as an attribute suitable for inclusion. The ROP assessment process reviews findings identified in the previous year and could conclude that insufficient resources are available if several findings are identified with this attribute. A substantive cross-cutting issue would then be identified and discussed in an assessment letter sent to the licensee.[35]

The inspections described above demonstrate that the current ROP provides a thorough inspection of the licensee's self-assessment and corrective action programs. Also, the adequacy of resources is reviewed as a part of a human performance cross-cutting issue along with other potential cross cutting issues. This is a key focus area for the NRC staff because cross-cutting issues are systemic and can be an indicator of declining performance.

11. Conclusions From Comparison of MY ISA to ROP

The MY ISA did not require the shutdown of the facility because performance was considered adequate. However, it did result in an in-depth review of the licensee's operation, particularly in the areas of the design basis and problem identification and resolution. Similar to the MY ISA and past DET inspections, under the ROP a poor performing plant, as defined by objective criteria, receives an inspection using IP 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input." The inspection has several objectives and includes gathering additional information to be used in deciding whether continued operation of the facility is acceptable and whether additional regulatory actions are necessary to arrest declining plant performance. The inspection also provides insight into the overall root and contributing causes of performance deficiencies. To provide a diversity of talent and perspectives and to add a degree of independence to the effort, the inspection team is staffed, in part, with inspectors from other regional offices or headquarters. In the situation where a plant experiences an isolated operational event that meets the criteria described in MD 8.3, a reactive inspection will take place. Similar to the MY ISA, the highest level of reactive inspection requires that the inspection team be composed of members who are independent from significant involvement in the licensing and inspection of the facility.

Problems with the power uprate codes and processes used for MY were recognized and, based on the lessons learned, procedures now prescribe specific actions and inspections to ensure that design margins are maintained.

Weak identification and resolution of problems found during the ISA are now covered in depth by the PI&R inspections that are done continuously at every site by the resident inspectors and by more rigorous PI&R inspections performed biennially with inspection teams. Weak scope, rigor, and evaluation of testing, and declining material condition are inspected thoroughly in the surveillance testing reviews, walkdowns done by resident inspectors, and by the extensive component design basis inspections which are performed biennially. The causes of the problems identified in the ISA were economic pressure to be a low-cost producer limiting resources to address problems and lack of a questioning culture resulting in failure to identify or promptly correct significant problems. While the NRC does not directly assess economic pressure, as discussed above, inspectors may address resources as part of a human performance cross-cutting issue when categorizing findings. The lack of a questioning culture and not identifying and correcting problems is the direct focus of the PI&R inspection. These areas have also received heightened attention with the safety culture enhancements implemented in July 2006.

Overall, the current ROP inspection procedures and NRC review standards provide essentially full coverage of key aspects of the MY ISA, with greater attention to safety culture and better focus on potentially risk-significant problems. This is shown in a cross-reference between the ISA and the ROP in the Attachment. If the resources used to review the MY allegations are subtracted from the overall direct inspection effort for the ISA, the remaining resources are similar to those used for a single unit site under the ROP each year. The ROP is designed to be objective and predictable, meaning that given the same performance, different licensees will receive the same level of regulatory oversight. Plants that show symptoms of declining performance receive increased levels of inspection above the baseline. The tools available to the inspectors, regional and headquarters management, and the Executive Director of

Operations are extensive to ensure the health and safety of the public. As described in earlier sections, there are some facilities that are receiving increased oversight due to performance concerns. In summary, the current ROP is working to ensure the right level of oversight is provided based on licensee performance.

Attachment
Cross-Reference Between the MY ISA and the ROP

ISA	ROP	Comments
Transient and Accident Safety Analyses	N/A	Allegation related, not included in the baseline ROP. This area inspected when power uprates are requested.
Design Review of Selected Systems and Electrical and Instrument and Controls	IP 71111.21 "Component Design Basis Inspection" IP 71111.22 "Surveillance Testing" IP 71111.19 "Post-Maintenance Testing" IP 71111.15 "Operability Evaluations" IP 71111.17 "Permanent Plant Modifications"	Addressed by ROP
FSAR Inconsistencies	Inspector preparation for most inspection procedures includes reviewing the FSAR, therefore, discrepancies may be identified during this process. The CDBI inspection does reference the safety analysis report as a potential resource and input for inspectors. IP 71111.02 "Evaluations of Changes, Tests, or Experiments"	Addressed by ROP
Operations Assessment: Quality of Operations	IMC 2515, "Light-Water Reactor Inspections – Operations Phase," Appendix D, "Plant Status" IP 71111.04 "Equipment Alignment" IP 71111.21 "Component Design Basis Inspection" IP 71111.13 "Maintenance Risk Assessment and Emergent Work Control" IP 71111.20 "Refueling and Outage Activities" IP 71111.15 "Operability Evaluations"	Continuous control room observations are not routinely performed but the following IP is used as needed: IP 71715 "Sustained Control Room and Plant Observation."

ISA	ROP	Comments
Operations Assessment: Programs and Procedures	IP 71111.21 "Component Design Basis Inspection" IP 71111.11 "Licensed Operator Requalification Program" IP 71111.06 "Flood Protection Measures" IP 71111.04 "Equipment Alignment" IP 71111.23 "Temporary Plant Modifications"	Addressed by ROP
Operations Assessment: Plant Support	IP 71111.05A/Q "Fire Protection Annual/Quarterly" IP 71111.05T Fire Protection (Triennial)" IP 71121 "Occupational Radiation Safety" IP 71121.01 "Access Control To Radiologically Significant Areas" IP 71121.02 "ALARA Planning and Controls" IP 71121.03 "Radiation monitoring Instrumentation," IP 71122 "Public Radiation Safety" IP 71122.01 "Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems" IP 71122.02 " Radioactive Material Processing and Transportation" IP 71122.03 "Radiological Environmental Monitoring Program (REMP) And Radioactive Material Control Program" IP 71111.11 "Licensed Operator Requalification Training"	Plant restart readiness is not a part of the baseline ROP inspections. Instead the following guidance is used: IMC 0350 "Oversight of Reactor Facilities in a Shutdown Condition Due to Significant Performance and/or Operational Concerns."
Maintenance and Testing Assessment	IP 71111.22 "Surveillance Testing" IP 71151 "Performance Indicator Verification" IP 71111.19 "Post-Maintenance Testing" IP 71111.12 "Maintenance Effectiveness" IP 71111.13 "Maintenance Risk Assessment and Emergent Work Control" IP 71111.20 "Refueling and Outage Activities"	Addressed by ROP
Maintenance and Testing Assessment: Equipment Performance	IP 71151 "Performance Indicator Verification" IP 71111.22 "Surveillance Testing" IMC 2515, "Light-Water Reactor Inspections – Operations	Addressed by ROP

ISA	ROP	Comments
Maintenance and Testing Assessment: Quality of Maintenance	IP 71111.12 "Maintenance Effectiveness"	Addressed by ROP
Maintenance and Testing Assessment: Testing Weaknesses	IP 71111.22 "Surveillance Testing" IP 71111.21 "Component Design Basis Inspection" IP 71111.19 "Post-Maintenance Testing"	Addressed by ROP
Maintenance and Testing Assessment: Maintenance Work Order Control	IP 71111.13 "Maintenance Risk Assessment and Emergent Work Control" IP 71111.20 "Refueling and Outage Activities"	Addressed by ROP
Engineering Assessment: Programs	IP 71111.02 "Evaluations of Changes, Tests, or Experiments" IP 71111.17 "Permanent Plant Modifications" IP 71111.23 "Temporary Plant Modifications" IP 71111.07 "Heat Sink Performance" IP 71111.21 "Component Design Basis Inspection" IP 71111.15 "Operability Evaluations" IP 71111.12 "Maintenance Effectiveness" IP 71111.04 "Equipment Alignment" IP 71111.22 "Surveillance Testing" IP 71111.08 "Inservice Inspections"	Addressed by ROP
Engineering Assessment: Design-Basis	IP 71111.17 "Permanent Plant Modifications" IP 71111.23 "Temporary Plant Modifications" IP 71111.21 "Component Design Basis Inspection"	Addressed by ROP
Engineering Assessment: Quality of Engineering	IP 71111.17 "Permanent Plant Modifications" IP 71111.23 "Temporary Plant Modifications" IP 71111.21 "Component Design Basis Inspection"	Addressed by ROP
Self Assessment, Corrective Actions, Planning and Resources	IP 71152 "Problem Identification and Resolution" Inspection Manual Chapter 0305 "Operating Reactor Assessment Program"	Addressed by ROP

End Notes

1. A more detailed summary of the sequence of events leading up to the MY Independent Safety Assessment (ISA) is contained in the associated NRC Office of the Inspector General report, which is publicly available at http://www.nrc.gov/reading-rm/doc-collections/insp-gen/1996/96-04s.html.

2. IP 95003 "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input"

3. Management Directive (MD) 8.3, "NRC Incident Investigation Program" is publically available at: http://www.nrc.gov/reactors/operating/oversight/program-documents.html.

4. IP 95003 "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input"

5. RS-001, "Review Standard for Extended Power Uprates" and RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications" are publically available at: http://www.nrc.gov/reactors/operating/licensing/power-uprates.html#relatedregs.

6. IP 71111.21 "Component Design Bases Inspections," IP 49001 "Inspection of Erosion/Corrosion Monitoring Programs," IP 71111.17 "Permanent Plant Modifications," IP 71111.02 "Evaluations of Changes, Test, and Experiments"

7. IP 71111.21 "Component Design Basis Inspection"

8. IP 71111.22 "Surveillance Testing"

9. IP 71111.19 "Post-Maintenance Testing"

10. IP 71111.15 "Operability Evaluations"

11. IP 71111.17 "Permanent Plant Modifications," IP 71111.23 "Temporary Plant Modifications," IP 71111.02 "Evaluation of Changes, Tests, or Experiments"

12. IP 71111.02 "Evaluations of Changes, Tests, or Experiments"

13. Inspection Manual Chapter 2515, "Light-Water Reactor Inspections – Operations Phase," Appendix D, "Plant Status"

14. IP 71715 "Sustained Control Room and Plant Observation"

15. IP 71111.04 "Equipment Alignment," IP 71111.21 "Component Design Basis Inspection," Inspection Manual Chapter 2515, "Light-Water Reactor Inspections – Operations Phase," Appendix D, "Plant Status"

16. Inspection Manual Chapter 2515, "Light-Water Reactor Inspections – Operations Phase," Appendix D, "Plant Status"

17. IP 71111.13 “Maintenance Risk Assessment and Emergent Work Control,” IP 71111.20 “Refueling and Outage Activities”

18. IP 71111.15 “Operability Evaluations”

19. IP 71111.21 “Component Design Basis Inspection“

20. IP 71111.11 “Licensed Operator Requalification Program,” IP 71111.06 “Flood Protection Measures”

21. IP 71111.04 “Equipment Alignment,” IP 71111.23 “Temporary Plant Modifications”

22. IP 71111.05A/Q “Fire Protection Annual/Quarterly,” IP 71111.05T Fire Protection (Triennial),” IP 71121 “Occupational Radiation Safety,” IP 71121.01 “Access Control To Radiologically Significant Areas,” IP 71121.02 “ALARA Planning and Controls,” IP 71121.03 “Radiation monitoring Instrumentation,” IP 71122 “Public Radiation Safety,” IP 71122.01 “Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems,” IP 71122.02 “ Radioactive Material Processing and Transportation,” IP 71122.03 “Radiological Environmental Monitoring Program (REMP) And Radioactive Material Control Program,” IP 71111.11 “Licensed Operator Requalification Training”

23. IP 71111.22 “Surveillance Testing”

24. IP 71111.22 “Surveillance Testing”

25. IP 71151 “Performance Indicator Verification,” Inspection Manual Chapter 0608 “Performance Indicator Program”

26. IP 71150 “Discrepant or Unreported Performance Indicator Data”

27. IP 71111.19 “Post-Maintenance Testing”

28. IP 71111.12 “Maintenance Effectiveness”

29. IP 71111.13 “Maintenance Risk Assessment and Emergent Work Control”

30. IP 71111.20 “Refueling and Outage Activities”

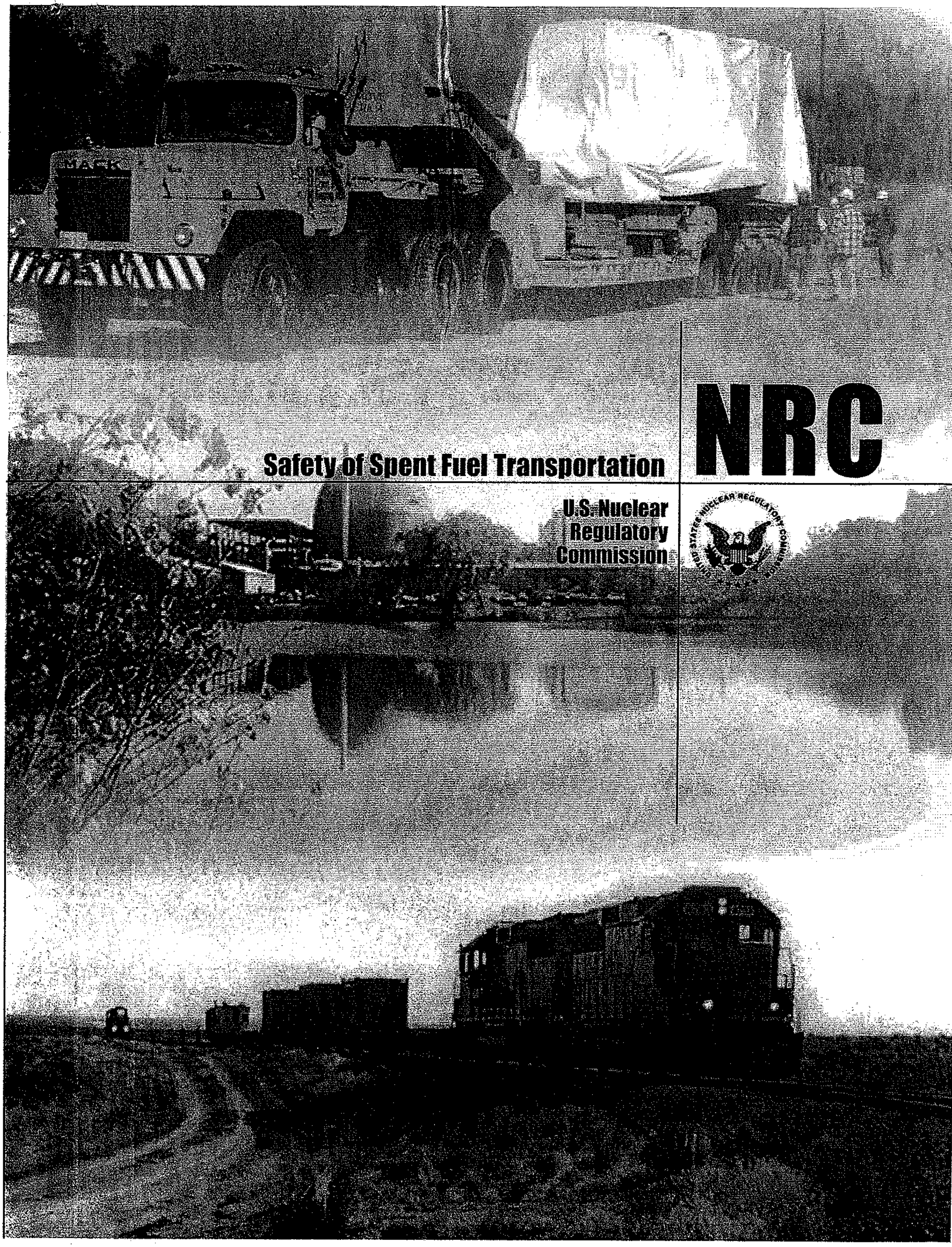
31. IP 71111.02 “Evaluations of Changes, Tests, or Experiments,” IP 71111.17 “Permanent Plant Modifications,” IP 71111.23 “Temporary Plant Modifications”

32. IP 71111.07 “Heat Sink Performance”

33. IP 71111.21 “Component Design Basis Inspection,“ IP 71111.15 “Operability Evaluations,” IP 71111.12 “Maintenance Effectiveness,” IP 71111.04 “Equipment Alignment,” IP 71111.22 “Surveillance Testing”

34. IP 71111.08 “Inservice Inspections”

35. Inspection Manual Chapter 0305 “Operating Reactor Assessment Program”

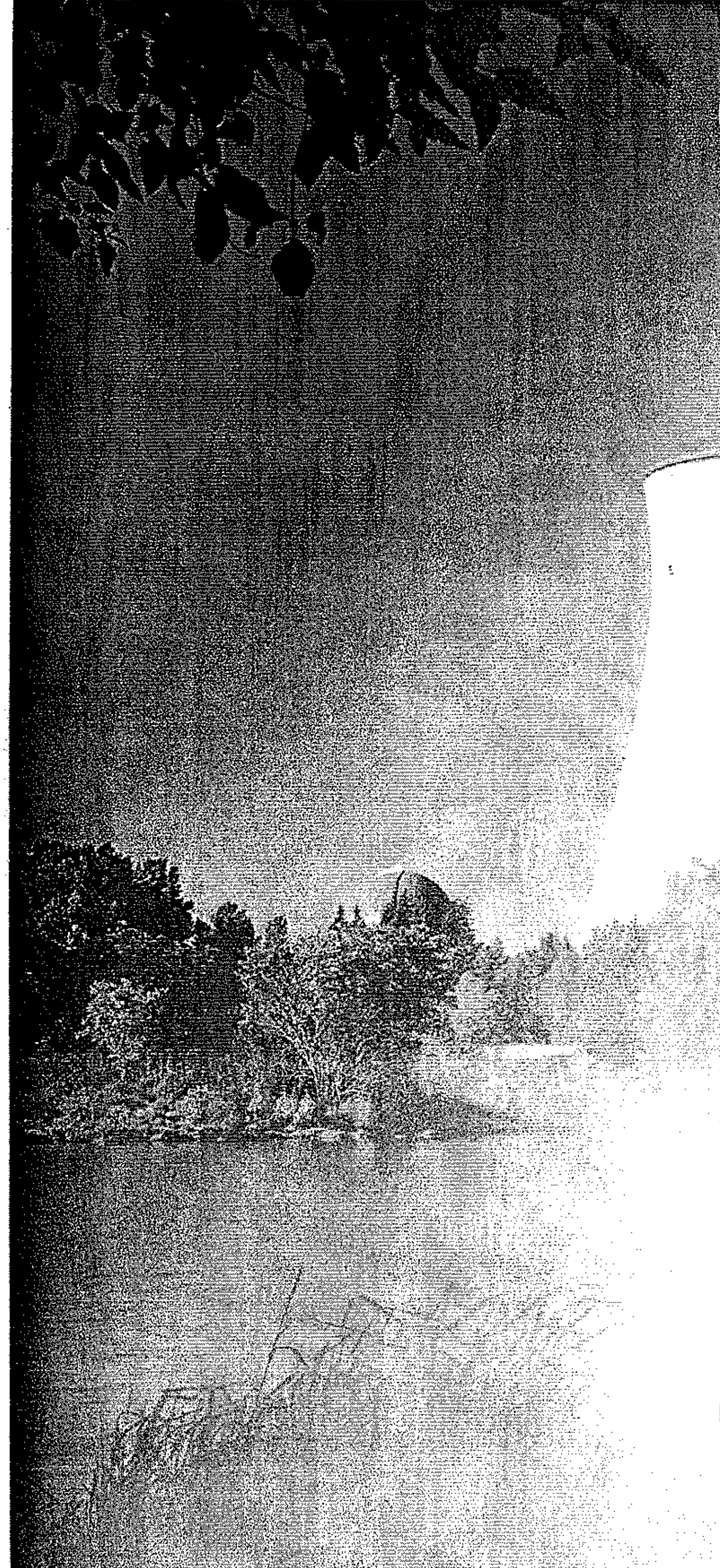


NRC

Safety of Spent Fuel Transportation

U.S. Nuclear Regulatory Commission

The Agencies: Who does what?



The U.S. Nuclear Regulatory Commission (NRC) is an independent agency established by the U.S. Congress in 1974 to ensure adequate protection of public health, safety, and the environment in the use of nuclear materials. The NRC regulates commercial nuclear power reactors; non-power research, test, and training reactors; and fuel cycle facilities. The NRC also regulates medical, academic, and industrial uses of nuclear materials, as well as packaging for the transport, storage, and disposal of nuclear materials and waste. In addition, the NRC regulates the design, manufacture, use, and maintenance of containers for high-level radioactive shipments.

The U.S. Department of Transportation (DOT), in coordination with the NRC, sets rules governing the packaging of nuclear materials. With NRC and the affected states, DOT regulates the transport of nuclear materials. The DOT also regulates carriers of nuclear materials, sets standards for transportation routes, and is responsible for international agreements on the transport of all hazardous materials.

The U.S. Department of Energy (DOE), among other things, oversees the development of disposal systems for spent nuclear fuel from the nation's nuclear power plants. This activity is entirely funded by fees collected from nuclear power plant companies and ultimately from rate payers.

The International Atomic Energy Agency (IAEA) serves as the world's principal intergovernmental forum for scientific and technical cooperation in the nuclear field. An agency of the United Nations, the IAEA published regulations for transporting nuclear materials. These regulations serve as a model for the United States and other nations.

Cover photo of tarped cask on trailer courtesy of GE Nuclear Energy. Train photo courtesy of NAC International. Back cover photo of IF-300 cask courtesy of GE Nuclear Energy.

The Nuclear Regulatory Commission

The U.S. Nuclear Regulatory Commission (NRC) regulates the nuclear materials cycle from beginning to end. This cycle begins with the mining of uranium. It continues through the manufacture of fuel, its use in reactors, any temporary storage, and (ultimately) with permanent geologic disposal.

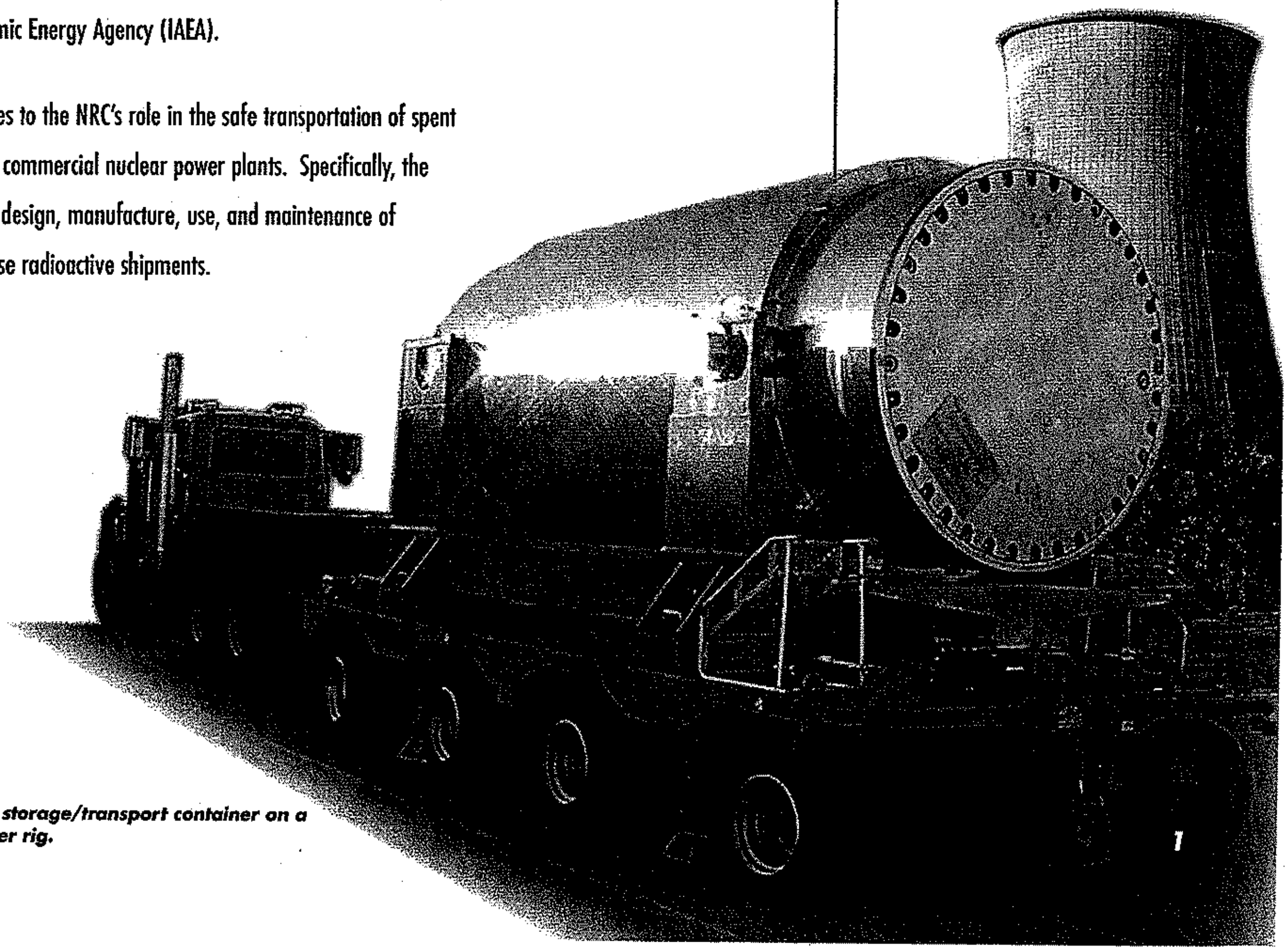
The NRC is dedicated to maintaining public health and safety, protecting the environment, and ensuring our national security in ways that increase public confidence in the agency. The NRC plans to achieve these goals by making its activities more effective, efficient, and realistic, and by reducing unnecessary regulatory burden on all those involved in the use, handling, transport, and disposal of nuclear materials.

The NRC believes that proper handling of nuclear materials will help to ensure the safety of the public and plant workers. Toward that end, the NRC works with other agencies, such as the U.S. Department of Transportation (DOT), the U.S. Department of Energy (DOE), and the International Atomic Energy Agency (IAEA).

This booklet relates to the NRC's role in the safe transportation of spent nuclear fuel from commercial nuclear power plants. Specifically, the NRC oversees the design, manufacture, use, and maintenance of containers for these radioactive shipments.

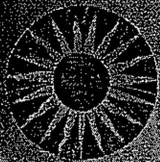
The NRC has three principal functions:

1. ***to set standards and develop regulations;***
2. ***to issue licenses for nuclear facilities and nuclear materials users; and***
3. ***to inspect facilities to ensure that NRC regulations are being met.***



This is an empty storage/transport container on a semi tractor-trailer rig.

Radiation



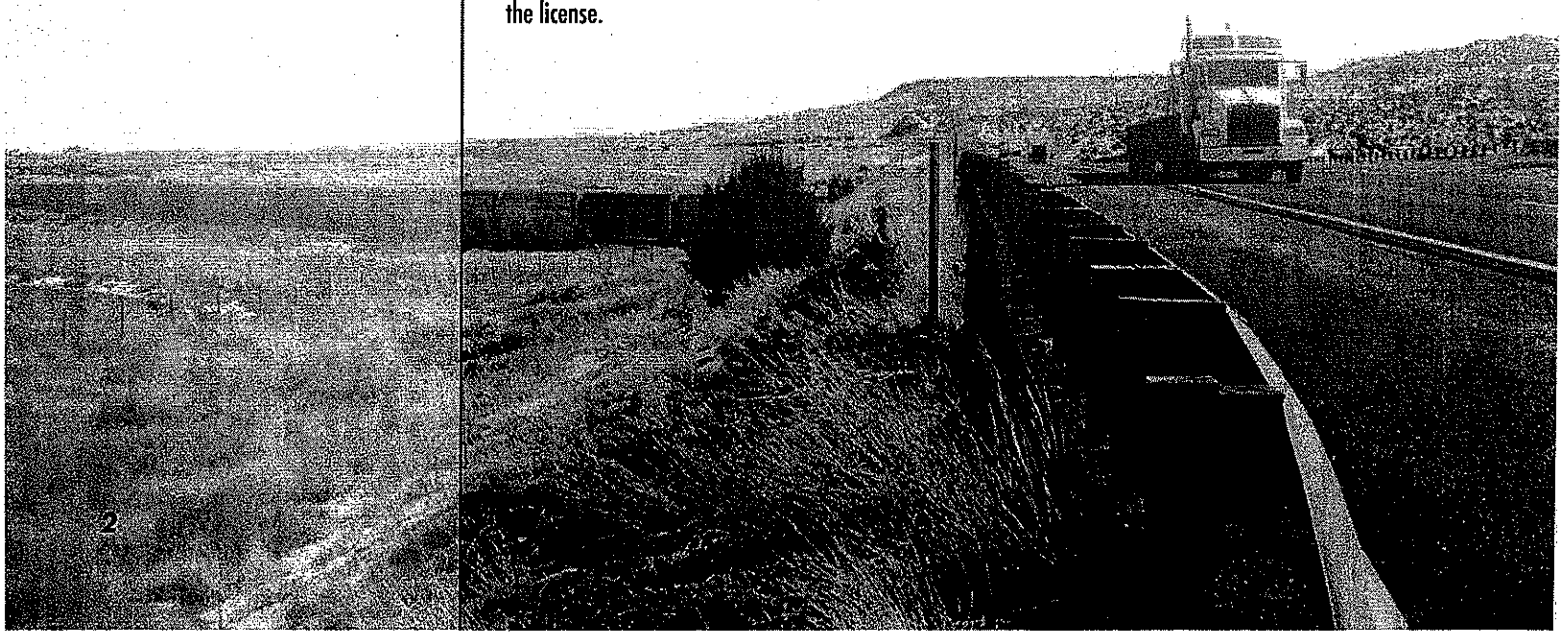
Scientists estimate that nearly 90 percent of all radiation exposure comes from natural sources: radon gas, the human body, outer space, rocks, and soil. Background radiation is naturally present, but its levels can vary greatly. People living in areas with a significant amount of granite, for example, receive more earth-based radiation. Those living or working at high altitudes receive more cosmic radiation. Most natural exposure is from radon, a gas that seeps from the earth's crust into the air we breathe.

The remaining 10 percent of all radiation exposure comes from man-made sources, primarily medical x-rays. Natural and artificial radiation are similar in kind and effect.

What is Spent Fuel?

Nuclear reactors produce electricity and, as a waste product, spent fuel. Uranium fuel powers reactors for a number of years, until its potential to produce electrical power is exhausted. The used uranium fuel is then referred to as "spent fuel." Nuclear power plants store spent fuel in enclosed cooling pools and, in some cases, in dry storage casks to await shipment to a temporary storage or permanent disposal facility. The Nuclear Waste Policy Act (NWPA), enacted by Congress in 1992, calls for spent fuel to be moved to a temporary storage facility or to a permanent DOE repository.

The NWPA sets a national policy for safe, permanent disposal of spent nuclear fuel and other radioactive wastes in an underground repository. The action by Congress and the President in July 2002 approving Yucca Mountain will permit the DOE to apply to NRC to construct the repository. The NRC's role under the NWPA is to use its independent judgment as an expert technical agency to decide whether to grant DOE a license to construct a high-level waste repository at Yucca Mountain. Only after extensive review of a DOE application will the NRC be able to judge whether DOE has satisfied the demands of the regulations. The NWPA gives NRC up to four years to decide whether to grant the license.



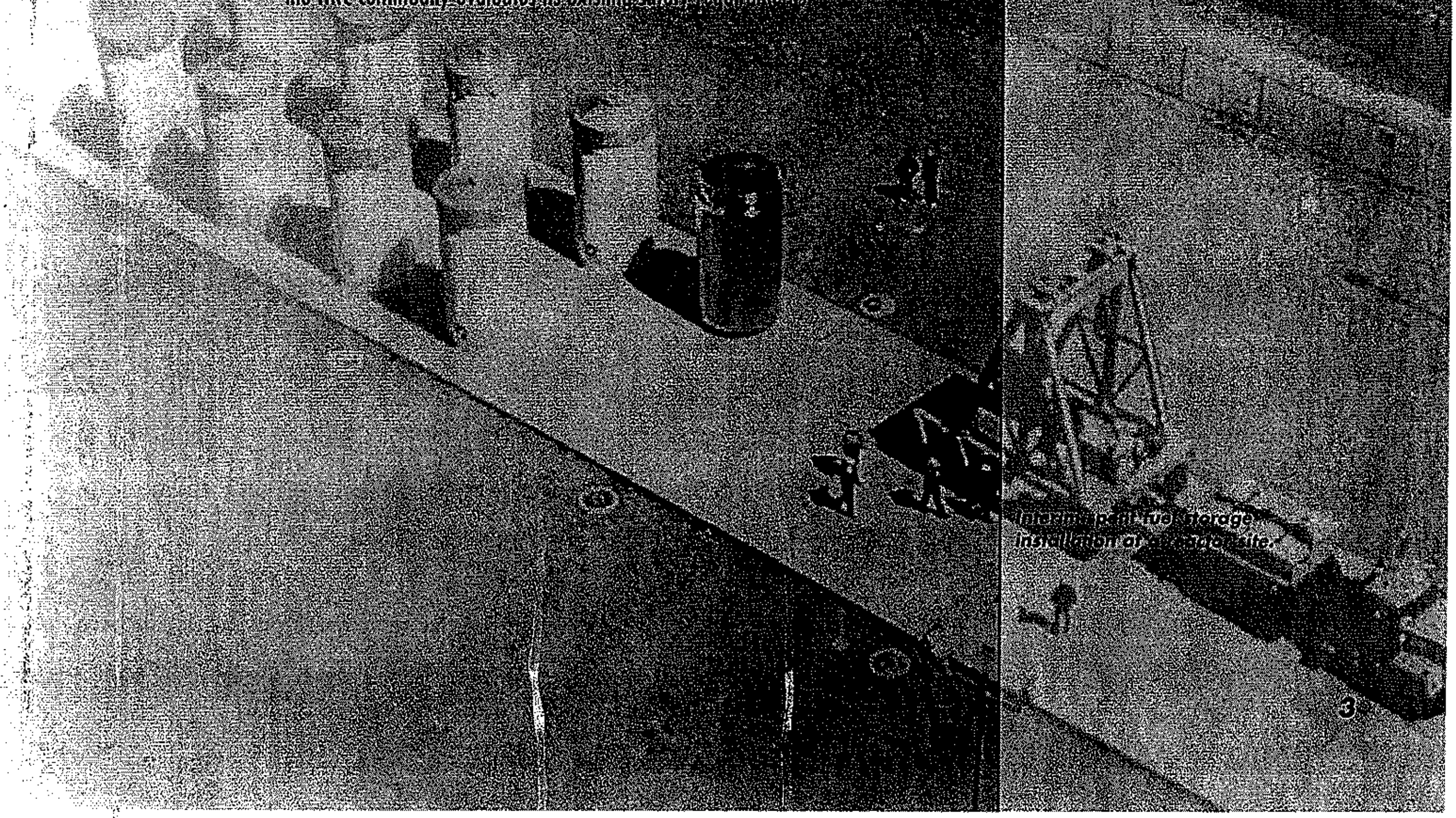
Because a repository won't be available for some time, some nuclear power plants are implementing plans for temporary storage on site. Other plants plan to store spent fuel away from the reactor at a temporary site until a permanent repository is built.

Given the widespread locations of power reactors, if a disposal site is finally approved, licensees will need to transport spent fuel to that site safely. These shipments would likely be made on railroads and on public highways.

Spent fuel is highly radioactive and must be transported in large, heavy containers that shield the public from exposure. This raises the following frequently asked questions in connection with such shipments:

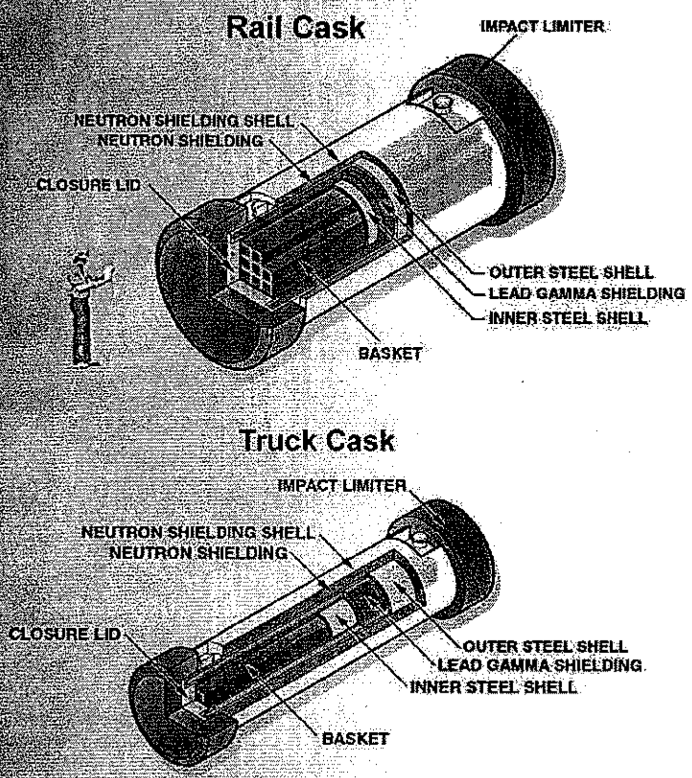
- How does the NRC protect the public from radioactive waste that is being transported?
- What is the likelihood of these shipments being involved in an accident?
- How well can the transportation containers withstand an accident and prevent the release of nuclear materials?

The NRC addresses these and other questions as a part of its ongoing efforts to ensure safe transport. As new technologies and information become available, the NRC continually evaluates its existing safety requirements.

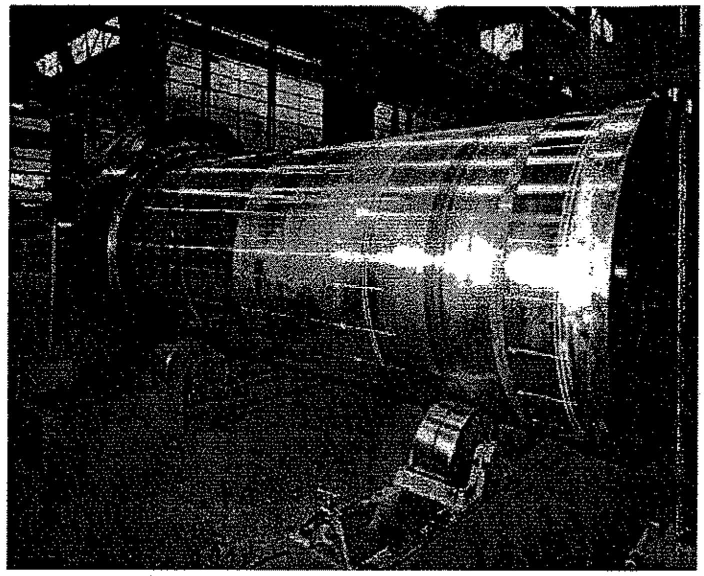


Interim spent fuel storage installation at a reactor site.

The Key to Ensuring Safety: the Spent Fuel Shipping Container

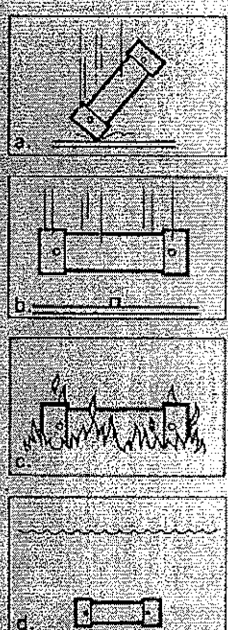


Spent fuel containers are specially designed to protect the public by withstanding accident conditions without releasing their radioactive contents.



The manufacture of spent fuel casks is carefully regulated by the NRC.

Spent fuel is highly radioactive and must be heavily shielded and tightly contained to be transported safely. An essential component for any safe shipment is a robust spent fuel container, or "cask."



The impact (free drop and puncture), fire, and water-immersion tests are considered in sequence to determine their cumulative effects on a given package.

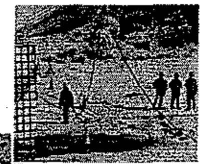
The NRC establishes regulations and standards for the design and construction of robust casks as the primary way to protect the public during transport. Containers used to move spent fuel by rail or highway are designed to withstand severe accidents. U.S. and international regulations require that these containers must pass a series of tests that mimic accident damage. The NRC conducts rigorous reviews to certify that spent fuel containers meet the design standards and test conditions in the regulations.

These containers must be shown, by test or analysis, to survive a sequence of four simulated accident conditions involving impact, puncture, fire, and submersion. During and after the tests, the containers must contain nuclear material, limit doses to acceptable levels, and prevent nuclear reaction.

To protect workers and the public, containers have walls five to 15 inches thick, made of steel and shielding materials, and a massive lid. Truck containers weigh about 25 tons when loaded with 1 to 2 tons of spent fuel. Rail containers can weigh as much as 150 tons and can carry up to 20 tons of spent fuel. The ends of these transportation containers are encased in structures called impact limiters. In the event of an accident, these limiters would crush, absorbing impact forces and protecting the container and its cargo.

Spent fuel containers are tightly sealed and provide shielding for most radiation. However, it is not possible to eliminate all radiation with shielding. Containers provide enough shielding to reduce external radiation to low levels that meet DOT and NRC radiation standards for the radiation dose to individuals who might be near the cask during transport.

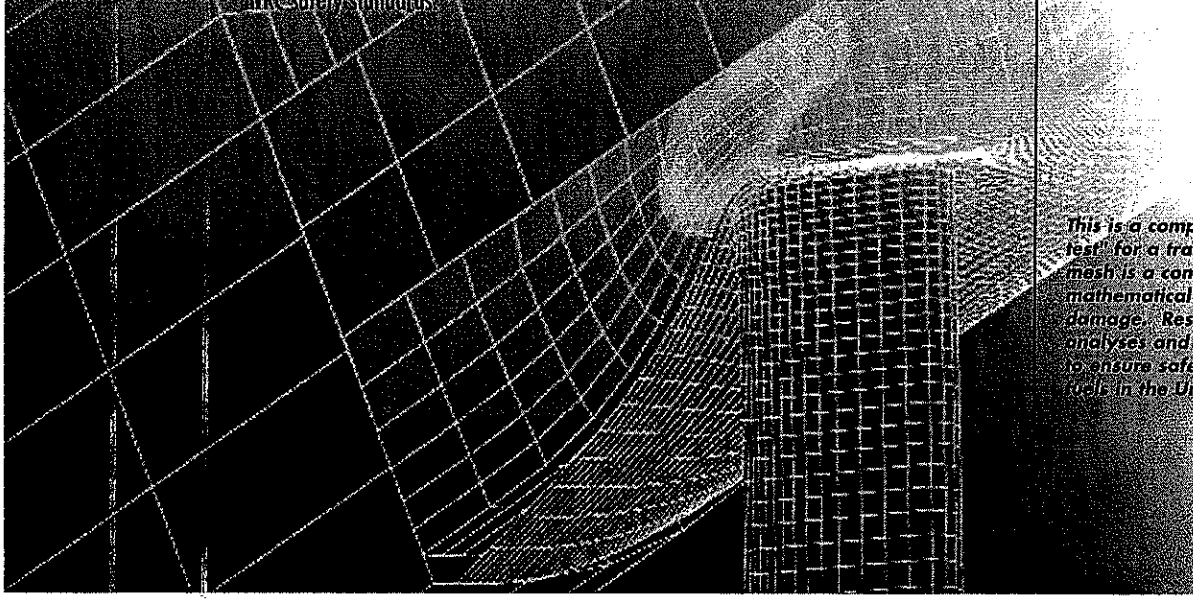
Container designers may use computer analyses, comparisons with other designs, component testing, scale-model testing, or a combination of these techniques to demonstrate that containers are safe. Most often, they use a combination of computer analyses and physical testing. NRC evaluates each application for a container design, examines the information in depth, and then performs its own calculations. NRC reviewers include structural and materials engineers and safety specialists with advanced degrees and many years of experience.



Once the NRC issues a Certificate of Compliance for a spent fuel container design, fabricators make the containers. Manufacturers and shippers must adhere to a program that ensures the containers continuously meet design specifications.

A scale-model container "drop test" helps researchers understand the forces involved in typical and unusual crash situations.

NRC and DOT regulations also require a number of safety determinations before each spent fuel shipment. These include checks for leaks and tests to ensure that radiation levels and contamination levels are within safe limits. These actions are designed to ensure that all aspects of every spent fuel shipment meet the applicable NRC safety standards.



This is a computer simulation of a "punch test" for a transportation container. The mesh is a computer-constructed mathematical device to help calculate cask damage. Results from a variety of analyses and tests like this one help NRC to ensure safe transportation of spent fuels in the United States.

Brief History of Spent Fuel Shipments and Studies

More than 1,300 spent fuel shipments regulated by the NRC have been completed safely in the U.S. during the past 25 years. Although there have been four accidents involving those shipments, none have resulted in a release of radioactive material.

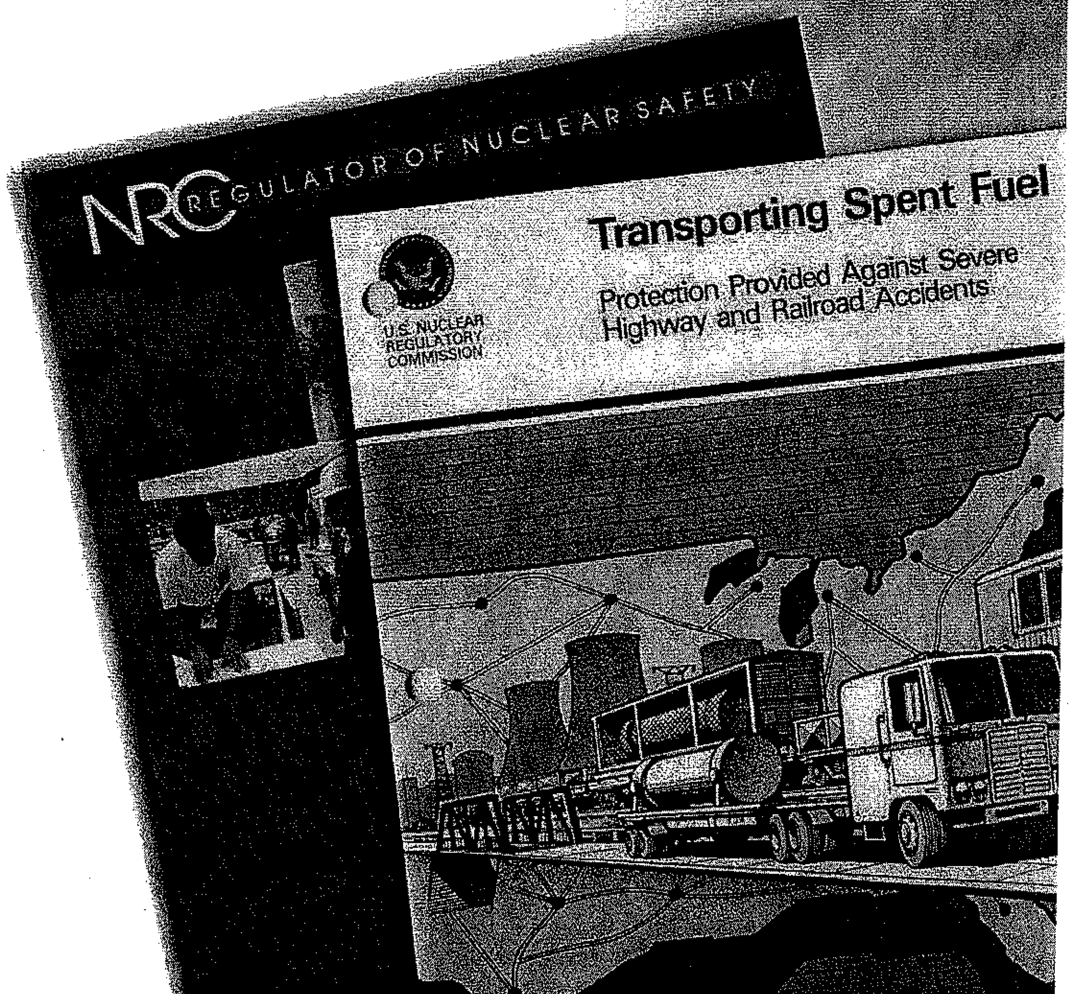
Experience with past shipments confirms that the fundamental safety system is sound. The question becomes, "What might happen if there are *thousands* of future shipments?" The NRC continuously evaluates risks associated with spent fuel transport in a methodical and scientific way. To provide additional confidence, the NRC has sponsored several risk studies related to spent fuel transportation on highways and railroads.

In 1977, the NRC completed a study that has since become the "baseline" for comparison with new information and studies completed since then.

In 1987, the NRC used improved research methods to evaluate how shipping containers react in accidents and to estimate the risk of releasing radioactive materials. The study results added assurances about the ability of shipping casks to withstand an accident and confirmed results of the 1977 study.

Another study, released in March 2000, used improved technology to analyze the ability of containers to withstand an accident. This study concluded that the risk from the increased number of spent fuel shipments that could occur in the first half of this century would be even smaller than originally estimated in 1977.

On the basis of these studies, operational experience, and its own technical reviews, the NRC concluded that the shipment of spent fuel is safe at projected shipment levels. The NRC is continuing to follow developments in spent fuel shipping, including the performance of additional analyses and testing of spent fuel casks, to ensure that the risks remain low.



Researchers use a four-step process to study actual and potential accidents and their effects on a container.

Step 1. Experts use historic records to determine what might happen.

- *They also gather data on how many spent fuel shipments are likely each year.*
- *They look at the rate of accidents for rail and highway shipments.*
- *Researchers look at a large number of accidents that are conceivable.*
- *They also look at crash impact forces, fires, or punctures that are more severe than those covered by NRC standards.*

Understanding the Risks

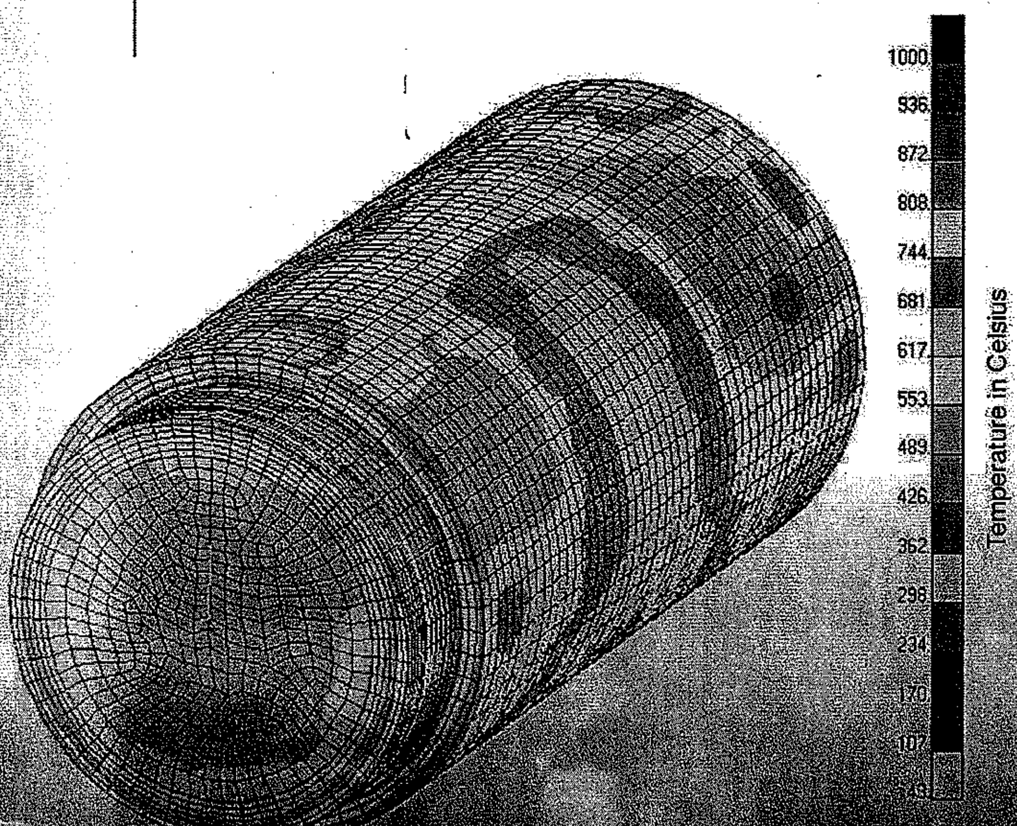
Risk is generally understood to be the possibility of injury, damage, or some kind of loss.

Given that understanding, the spent fuel shipment record in the U.S. has been outstanding to date. Many more shipments have been successfully completed internationally under the same basic safety standards.

While shipping spent fuel does involve risk, NRC studies indicate that this risk is low. As a part of its safety effort, minimizing risk is an important concept to the NRC. The NRC's risk assessment asks the following three questions and then converts the answers into numbers to arrive at a risk value:

- What can go wrong?
- How likely is it?
- If something goes wrong, what are the consequences?

Although the overwhelming majority of spent fuel shipments are accident-free, researchers calculate radiation risks to the public using two scenarios. One scenario involves a journey during which an accident occurs; the other covers the vast majority of journeys that do not involve an accident.



Shown is a computer simulation of the response of a cask to a severe fire environment. Analyses like this and tests are used by NRC to assure safe transportation of spent fuel.

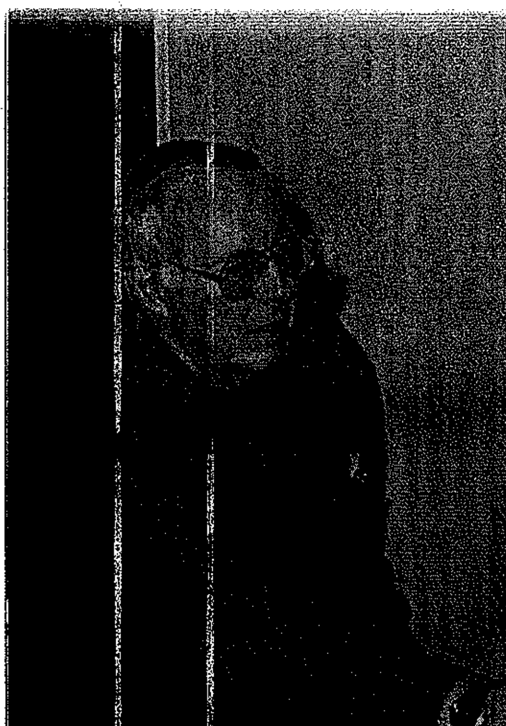
The Accident Scenario

NRC studies show that fewer than 1 in 100 accidents involving a spent fuel container will be more severe than the conditions of the design standards. However, if a very unlikely chain of events occurs, the accident might be severe enough to cause a radioactive release.

To estimate the likelihood and consequences of unusually severe accidents, researchers use a multi-step approach to calculate risk. That approach uses accident data and their experience with past trucking and rail accidents involving other hazardous materials. This also involves determining what kinds of accidents could happen and looking at their potential effects.

According to the DOE Final Environmental Impact Statement (FEIS) for the Yucca Mountain Project, about 11,000 rail or 53,000 truck shipments might be expected during the 24 years of operation of the repository, should it be approved. The chances that any accident would occur during a spent fuel shipment are about 1 in 10,000 for rail shipments and 1 in 1,000 for highway transport. Put another way, these estimates indicate that 1 to 50 accidents involving casks are conceivable in the process of moving all current spent fuel to a permanent repository.

Looking at these conceivable accidents, the chance that even one would be serious enough to lead to even a small release is about 1 in 1,000. The chance of a large release is estimated to be less than one in 1,000,000.



NRC engineer evaluating results of a container test.

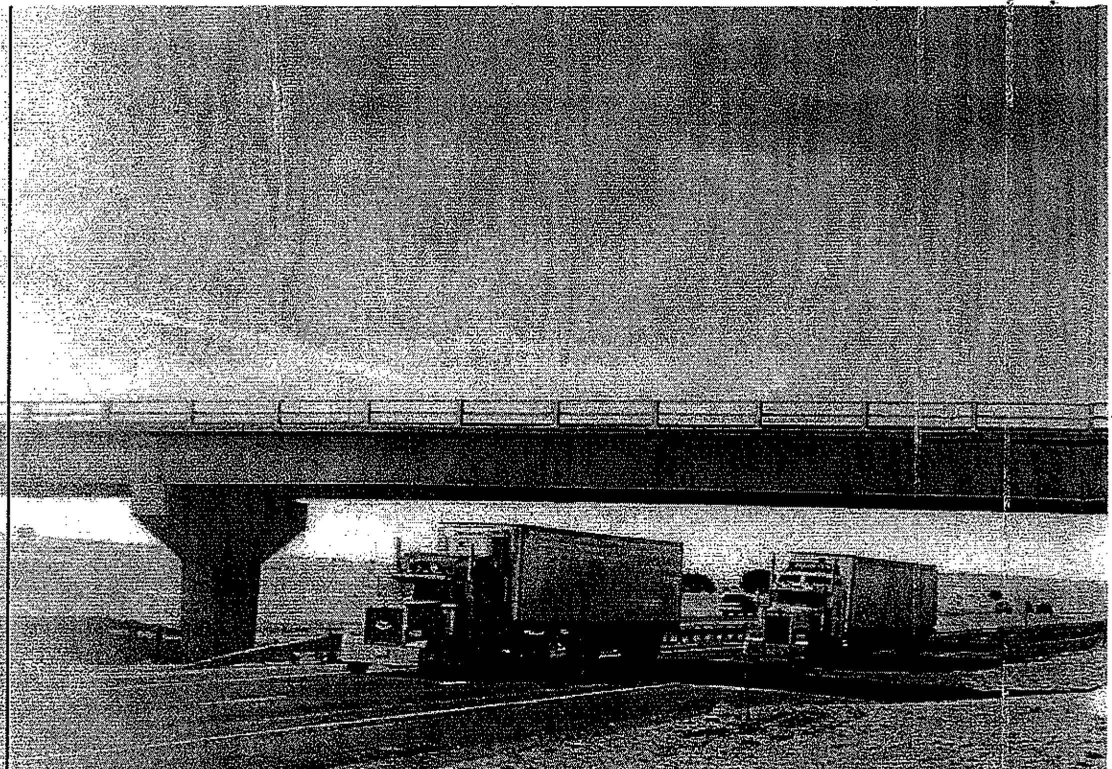
Step 2. ***Engineers use complex computer programs to estimate how the parts of a shipping container might be damaged by collisions or fires.***

- ***They gather data on how much spent fuel each container will carry.***
- ***They analyze how the fuel might respond in a given type of accident.***
- ***They calculate the temperature of the container and the spent fuel itself during a long-term fire.***

This information provides estimates on the size of any potential leak and how much nuclear material might escape.

Step 3. ***Researchers match accident scenarios from Step 1 with the assessments from Step 2 to determine the chance of severe damage to the container or its contents.***

Step 4. ***Researchers compute a risk estimate with a special computer program. The program takes accident probability estimates, expected numbers of shipments, route data (like population densities), weather data (to estimate how any release might be spread by wind), and radiological dose data to produce a risk estimate.***



The Accident-Free Scenario

In an accident-free journey, nothing goes wrong and no nuclear material is released from the container. In this scenario, the total of all radiological exposures, or doses, that could be received by all people along the transportation route is calculated. Because spent fuel, even fully contained, still emits low levels of radiation through the container walls, researchers use route and population information to estimate the number of people who could be exposed and the total radiation dose that they might receive.

The risk to the public from an accident-free journey results from the low-level radiation field that surrounds the spent fuel container. If the container is moving past a person, perhaps someone standing along the highway or railroad track, the exposure is brief and well below regulatory limits. Exposure will vary depending upon the speed of the train or tractor-trailer rig and the distance the person is standing from the highway or track. The very low dose to each person along the route is added to obtain the total population dose. As a basis for comparison, a passenger traveling round-trip by air from New York to Los Angeles receives a background radiation dose that is 25 times greater than the dose to persons closest to a typical spent fuel shipment.

The Bottom Line

The NRC believes that shipments of spent fuel in the U.S. are safe. This belief is based on the NRC's confidence in the shipping containers that it certifies and its ongoing research in transportation safety.

- The NRC ensures that shipping containers are robust by:
  - Regulating the design and construction of shipping containers.
  - Reviewing designs and independently checking a container's ability to meet accident conditions.
  - Ensuring that containers are built, maintained, and used properly.
- The NRC also follows an aggressive program to investigate and assess the risks involved in spent fuel shipments:
  - Analyzing spent fuel transportation records to understand safety issues better.
  - Evaluating new transportation issues, such as increased shipment levels, denser populations along some routes, and other factors.
  - Using new technology to estimate current and future levels of potential risk to the public.

Although there will always be a slight chance that an accident will cause a release of nuclear material, the NRC has found that the likelihood of such an event and the associated risk to the public are extremely low. Even so, the NRC will continue to be vigilant about public safety as an essential part of its mission.

Spent Fuel Transport Security

The NRC also regulates the physical protection of spent nuclear fuel in transit against sabotage or other malicious acts. The NRC's current physical protection regulations for spent fuel transportation include:

- Pre-shipment coordination with law enforcement agencies
- Pre-shipment notice of States and NRC
- In-transit shipment call-in to communications center
- Shipment monitoring
- Armed escorts (in populated areas)
- Immobilization devices

Since September 11, 2001, the NRC has taken additional steps to protect the public. These steps involve a heightening of the security posture, including new measures taken to protect nuclear facilities and regulated activities, such as spent fuel transportation, and orders that NRC has issued to licensees.

For Additional Information Contact:

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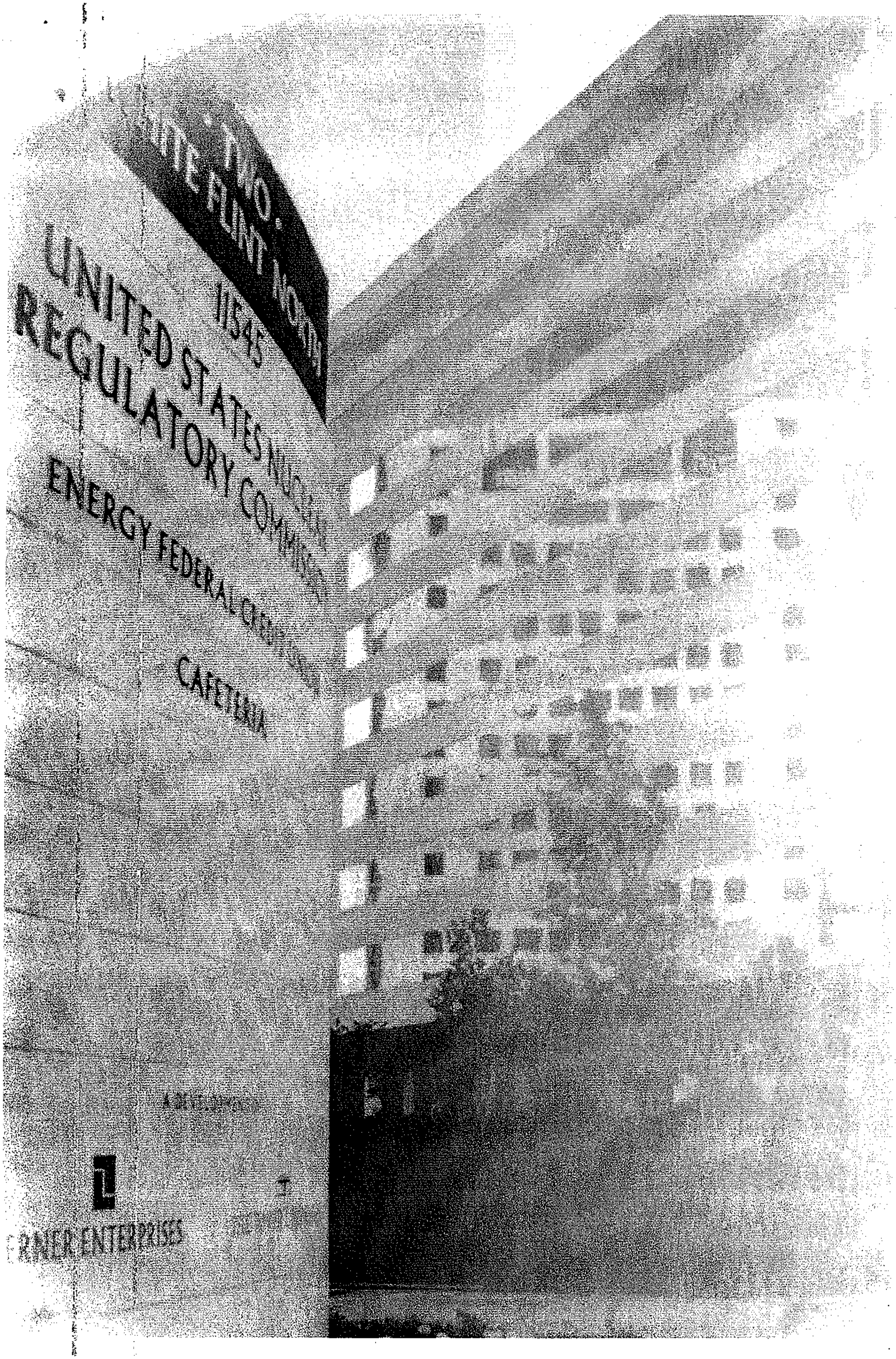
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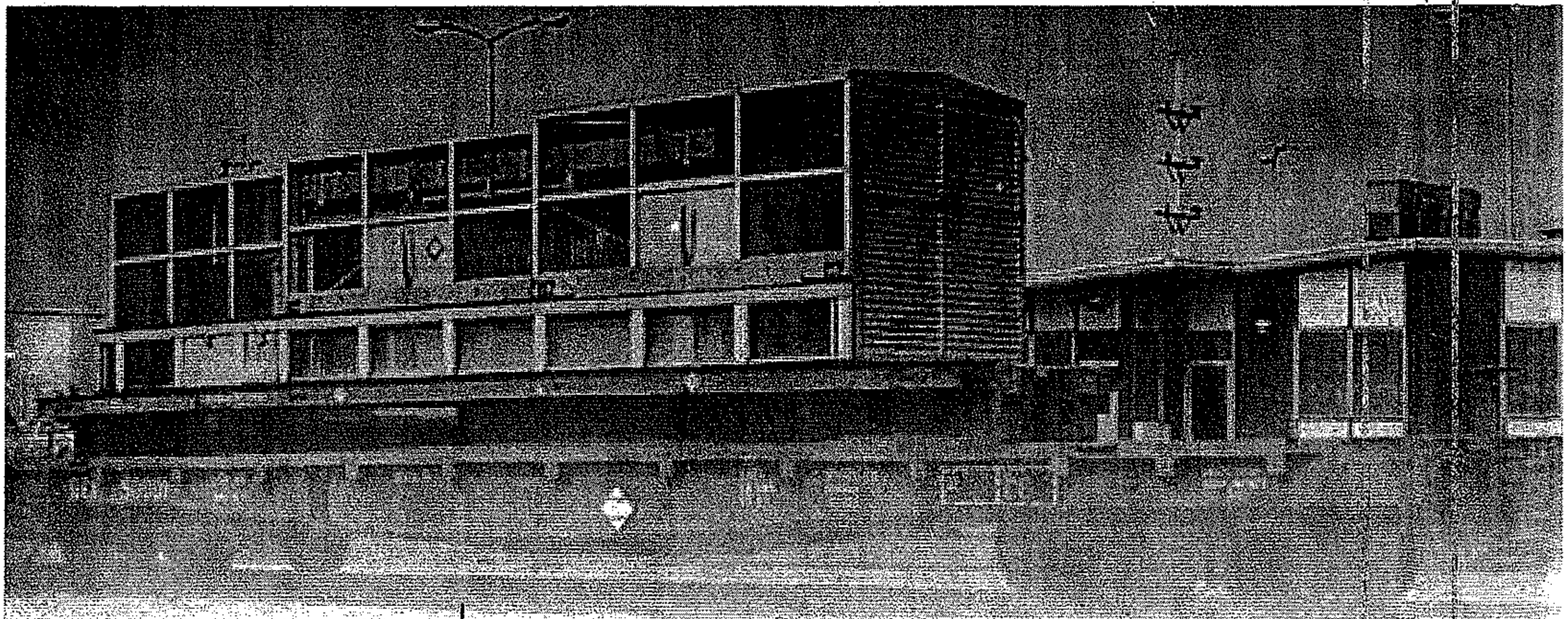

U.S. Nuclear
Regulatory
Commission

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