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United States General Accounting Office

GAO

Report to the Honorable
Timothy E. Wirth, U.S. Senate

November 1987

NUCLEAR SAFETY

Reactor Design, Management, and Emergency Preparedness at Fort St. Vrain

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GAO United States
General Accounting Office
Washington, D.C. 20548

Resources, Community, and
Economic Development Division

B-229002

November 13, 1987

The Honorable Timothy E. Wirth
United States Senate

Dear Senator Wirth:

This report responds to your request of August 29, 1986. It includes information on similarities and differences between the Fort St. Vrain nuclear power plant near Denver, Colorado, and the nuclear power plant that exploded near Chernobyl, U.S.S.R. This report also addresses management and emergency preparedness deficiencies at the Fort St. Vrain plant.

As agreed with your office, unless you publicly announce its contents earlier, we will not distribute this report until 14 days from the date of this letter. At that time, we will send copies to the Director, Office of Management and Budget; interested congressional committees; Members of Congress; the Nuclear Regulatory Commission; and the Federal Emergency Management Agency. Copies will be made available to others upon request.

Major contributors to this report are listed in appendix I.

Sincerely yours,

Keith O. Fultz

Keith O. Fultz
Associate Director

Executive Summary

Purpose

Following the 1986 accident at the nuclear power plant near Chernobyl, U.S.S.R., concern over the safety of nuclear power plants, particularly those of similar design, was heightened. In the United States, the Fort St. Vrain nuclear power plant near Denver, Colorado, is the only commercial facility with some similarities to the Chernobyl plant.

Because of these similarities, as well as Fort St. Vrain's history of management and operating problems, Senator Timothy E. Wirth requested GAO to provide information on the (1) design similarities, risks, and safety of Fort St. Vrain compared with the Chernobyl reactor; (2) basis for the Nuclear Regulatory Commission's (NRC) conclusion that a Chernobyl-like accident could not occur at the Fort St. Vrain power plant; (3) management problems reported at the plant; and (4) emergency preparedness program for the Fort St. Vrain reactor.

Background

On April 26, 1986, operators were conducting an experiment on a "graphite-moderated," water-cooled nuclear reactor near Chernobyl, U.S.S.R. (A graphite-moderated plant uses a graphite core to control the rate of fission within the reactor.) A combination of human error and poor reactor design resulted in a rise in power within 20 seconds, producing a large explosion in the reactor core. Subsequently, radiation escaped to the atmosphere, causing at least 31 deaths and spreading significant amounts of radiation over several countries lying to the north and northwest of Chernobyl.

Among the 98 operating commercial nuclear power plants in the United States, only Fort St. Vrain is graphite-moderated. It was built to demonstrate the "high-temperature, gas-cooled reactor" concept. It began operations in 1979; however, because of operational problems, it has operated intermittently for a total of only 38 months in the past 8 years. In addition, on the basis of overall plant performance, NRC, in mid-1986, named the Fort St. Vrain nuclear power plant as 1 of the 16 worst-managed nuclear power plants in the United States.

Results in Brief

Both the Fort St. Vrain and Chernobyl reactors have graphite cores and utilize basic nuclear reactor systems common to most nuclear power plants. However, there is very little similarity in the specific designs of the reactor systems. Fort St. Vrain's design provides a wider margin of safety than Chernobyl's, making it unlikely, according to a NRC review, that a Chernobyl-type accident could occur at Fort St. Vrain.

Fort St. Vrain's past operations, according to NRC, showed deficiencies in maintenance, licensing, security, outage management, quality assurance, and emergency preparedness that reduced safety and limited plant operations. Following a plant shutdown to upgrade the plant's electrical system, NRC refused to allow the plant to restart until management implemented a plan for improvement. Fort St. Vrain management has developed a program to rectify the deficiencies. This program is expected to be completely in place by mid-1988. In April 1987, NRC permitted the restarting of the Fort St. Vrain reactor.

Principal Findings

Chernobyl and Fort St. Vrain Reactors Differ Substantially

Fort St. Vrain and Chernobyl, like most nuclear reactors, have a reactor core, fuel systems, and cooling systems. The specific designs of these systems, however, are significantly different. While both reactors have massive amounts of graphite in the reactor core, the Fort St. Vrain reactor is designed so that the graphite absorbs most of the heat. The heat is then transferred to the helium gas that is circulated through the graphite. In contrast, the heat produced by the Chernobyl reactor is absorbed directly by cooling water, and the graphite acts only as a moderator. Thus, the consequences of a loss-of-coolant accident would be more severe at a Chernobyl-type reactor than at Fort St. Vrain.

The fuel of the Fort St. Vrain reactor is designed to withstand higher temperatures than the fuel at the Chernobyl reactor. The Fort St. Vrain fuel also has a much slower heat-up rate. Thus, the Fort St. Vrain fuel design provides hours to avoid a major accident after a problem occurs while the Chernobyl fuel design allowed only seconds.

Cooling systems at Fort St. Vrain and Chernobyl are vastly different. Fort St. Vrain's massive graphite core is cooled by helium gas that, according to Fort St. Vrain power plant officials, makes chemical reactions between the coolant and reactor components nearly impossible. Chernobyl was cooled by water. When the cooling system at Chernobyl became inadequate for a few seconds, fuel began to melt, and explosions occurred almost immediately.

The operating characteristics are also quite different. When the Chernobyl reactor did not receive a sufficient supply of coolant, it tended to increase in reactivity, thus increasing temperature. Fort St.

Vrain is designed to decrease in reactivity as temperature increases, thus making it safer under accident conditions.

Chernobyl-Like Accident at Fort St. Vrain Unlikely

The differences in design and operating characteristics argue against a Chernobyl-like accident occurring at the Fort St. Vrain reactor. In the aftermath of the Chernobyl accident, NRC reexamined the safety of the Fort St. Vrain reactor, including the consequences of a graphite fire and the accident analyses performed previously. It concluded that the probability of a Chernobyl-like accident at the Fort St. Vrain nuclear power plant is beyond the credible range.

Fort St. Vrain Operations Plagued by Management Deficiencies and Poor Performance

The NRC assessments of the Fort St. Vrain plant's operational safety showed a negative trend from 1982 to 1986. The assessment for the period ending April 1986 showed overall performance to be minimally satisfactory in 6 of 11 areas addressed. The plant was found deficient in maintenance, management of activities during periods when the reactor was not operating, adequacy and timeliness of licensing activities, quality assurance, security, and emergency preparedness. These deficiencies reduced the plant's margin of safety and contributed to its history of numerous periods when the reactor was shut down.

According to NRC officials, Fort St. Vrain management and employees appear to have had an attitude and morale problem that was at the root of Fort St. Vrain's poor performance. This problem resulted from (1) limited regulatory attention provided by NRC until about a year ago, (2) a history of intermittent operations, and (3) the reactor's unique design among U.S. reactors, which fostered a related belief that normal NRC operation and maintenance requirements should not apply. Following a shutdown to upgrade the plant's electrical system, NRC refused to allow the operators to restart the reactor until management demonstrated sufficient improvement in the deficient areas.

Management at Fort St. Vrain has developed a Performance Enhancement Program to improve the management and performance of the power plant. The program consists of projects to improve planning and scheduling, policies and procedures, conduct of operations, human resource management, quality assurance training, and preventive maintenance. Improvements in the power plant's emergency preparedness program have been underway since August 1986. NRC inspectors have noted improvements in programs for management involvement, plant performance, and problem-solving and gave permission for Fort St.

Vrain to restart in April 1987. In addition, the NRC's most recent assessment, released July 1, 1987, showed that only 1 of the 11 areas was minimally satisfactory.

Recommendations

This report provides information on a comparison of the Fort St. Vrain reactor design with the design of the Chernobyl reactor. It also provides information on operational and managerial deficiencies at the Fort St. Vrain reactor and the steps that have been taken or planned to improve those areas. Because a plan for improvement is being implemented and NRC has noted improvements sufficient to allow restart, GAO is making no recommendations.

Agency Comments

GAO discussed the facts presented in this report with NRC and Fort St. Vrain officials. These officials agreed with GAO's overall observations and provided information to clarify data contained in the report. As requested, GAO did not obtain official agency comments on the report.

Contents

Figures

Abbreviations

DODES	Division of Disaster Emergency Services
FEMA	Federal Emergency Management Agency
GAO	General Accounting Office
HTGR	high temperature gas-cooled reactor
NRC	Nuclear Regulatory Commission
NUS	Nuclear United Services
PSC	Public Service Company of Colorado
PSIG	pounds per square inch gauge
PSI	pounds per square inch
SALP	Systematic Assessment of Licensee Performance
SAR	Safety Analysis Report

Introduction

On April 26, 1986, an accident occurred at a nuclear power plant near the Soviet town of Chernobyl, which triggered a worldwide realization that nuclear accidents previously considered improbable could occur and produce catastrophic consequences. In the United States, nuclear safety has historically been a major concern to both the public and the Congress. The Chernobyl accident heightened this concern because at least two nuclear reactors in the United States, the N-Reactor near Richland, Washington, and the Fort St. Vrain reactor near Denver, Colorado, have features similar to the Chernobyl reactor. They are similar in that large amounts of graphite are used in the reactor core to moderate the nuclear reaction to allow it to be self-sustaining. In a report we issued on August 5, 1986,[1] concerning the safety of the N-Reactor, we pointed out that while there were problems associated with N-Reactor, it was basically safer than the Soviet reactor.

On August 29, 1986, we received a request from Senator Timothy E. Wirth to review safety aspects of the design, management, and emergency preparedness of the Fort St. Vrain nuclear power plant. We were also asked to compare the design and safety of Fort St. Vrain with that of Chernobyl. This report is our response to Senator Wirth's request. The remainder of this chapter provides background on the Chernobyl reactor, the Fort St. Vrain reactor, the Nuclear Regulatory Commission's (NRC) responsibilities concerning commercial nuclear power plants, and our objectives, scope, and methodology.

The Chernobyl Reactor

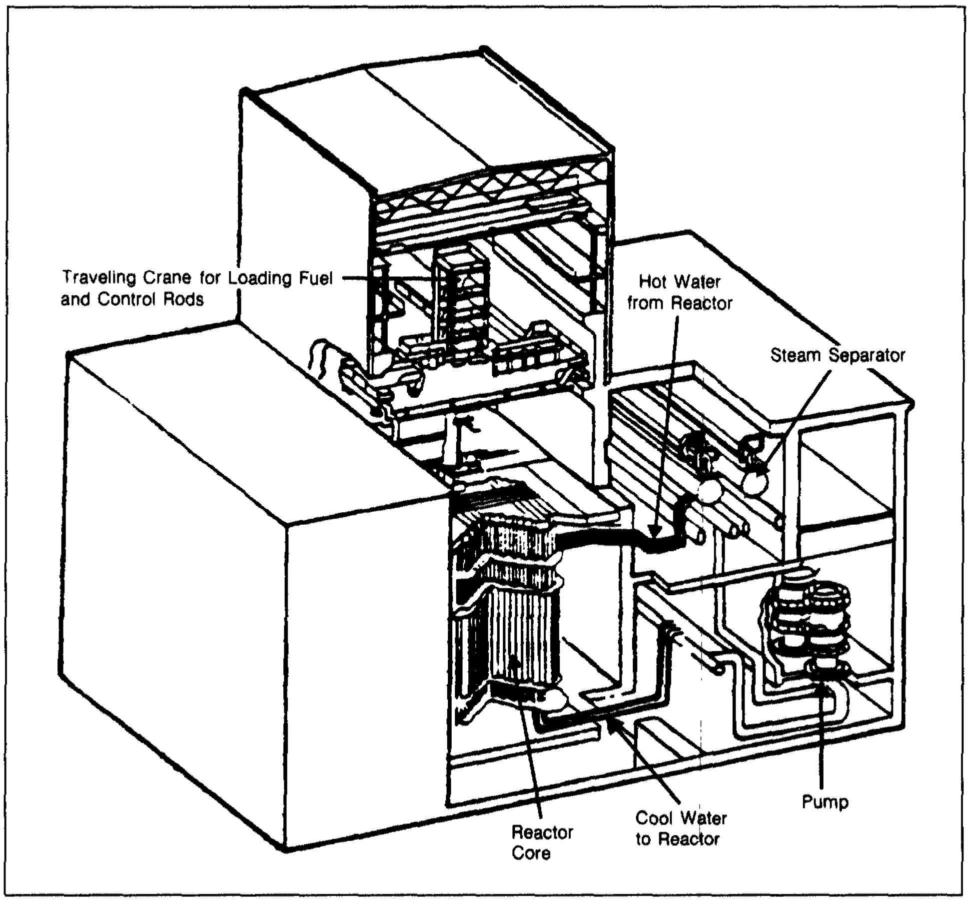
Chernobyl is a town in the western part of the U.S.S.R. known as the Ukraine. Approximately 150,000 people live within a 18-mile radius of the Chernobyl reactor. The accident that occurred at Chernobyl began while the Soviets were performing a test with the reactor. A combination of human error and poor reactor design resulted in a large explosion inside the reactor core that blew the core apart and destroyed the building housing the reactor. Subsequently, radiation escaped into the atmosphere and fallout from the accident occurred in several countries.

At least 31 deaths have resulted from the accident. In addition, several countries lying to the north and northwest of the accident received significant amounts of radiation. It is believed that adverse effects from the radiation may continue for decades.

[1] Nuclear Safety: Comparison of DOE's Hanford N-Reactor With the Chernobyl Reactor (GAO/RCED-86-213BR).

The Chernobyl reactor was housed in a building constructed mostly of concrete and metal. The reactor core was in the middle of the structure and provided steam for two large turbine generators located near the reactor core. Above the reactor was a large crane and support equipment used to refuel the reactor (see fig. 1.1).

Figure 1.1: Diagram of Chernobyl Reactor



Source: Department of Energy.

The core of the reactor was constructed of graphite blocks stacked together to form a cylinder approximately 23 feet high and 39 feet in diameter. The blocks used to construct the graphite stack were about 24 inches high and 10 inches wide. The core had at least 1,872 vertical channels, or holes, which accommodated 1,661 tubes containing fuel and coolant, and 211 rods to control the reactor. Each fuel tube could be opened at the top to remove old fuel and insert new fuel. The reactor control rods were inserted into the reactor from both the bottom and the top of the core to control the power level of the reactor.

The 1,661 fuel tubes in the reactor core were separate units that collectively produced steam to drive the turbine generators. Each tube had a water supply pipe attached at the bottom and a steam removal pipe at the top. The water was initially supplied by a main line that distributed and routed water to each tube through a series of headers. In order to produce steam, cool water entered the bottom of the tube and was heated as it ascended the tube.[2] About one-third of the distance up the tube, the water began to boil, creating steam. After passing through the tube, the steam was collected at the top of the reactor core in reverse fashion, with each line feeding into collectors until eventually all the steam was carried into main pipes to the two turbine generators.

The steam produced by heating the water in the tube was used to drive the turbine generators that produce electricity. After the steam was used, it was cooled back into water and routed back to the fuel tubes to repeat the cycle. This process is referred to as the "cooling loop" of the reactor. The Chernobyl reactor had two separate cooling loops that divided it in half. In other words, the tubes on one side of the reactor produced steam for one turbine generator, and the other half produced steam for the other turbine generator.

The Fort St. Vrain Nuclear Power Plant

The Fort St. Vrain nuclear power plant, located about 35 miles north of Denver, Colorado, is the only nuclear power plant owned and operated by the Public Service Company of Colorado (PSC). In addition to the Fort St. Vrain plant, PSC operates seven non-nuclear electric generating facilities that supply electric power to two states.

The Fort St. Vrain nuclear power plant is capable of generating 330 megawatts of electric power when operated at full power. This is a relatively small plant compared with other nuclear plants, which generate 800 to 1,000 megawatts of electricity. The Fort St. Vrain nuclear power

[2]The majority of the energy from the fission process is deposited in the fuel, which becomes hotter than the coolant and heats the coolant by ordinary heat transfer. In other words, when free neutrons smash into the uranium atoms, other neutrons are produced that smash into more uranium atoms. Thus, a chain reaction is established. However, to sustain this reaction the speed of the neutrons must be reduced to prevent them from going through the uranium atoms or deflecting without smashing the atoms. The graphite acts as a "moderator" by reducing the speed of the neutrons. In addition, the water in the tubes helps to slow the neutrons. The neutron activity produces energy or heat that is transferred to the water. Consequently, controls and limits must be established.

plant was built to demonstrate, on a commercial scale, the "high-temperature gas-cooled reactor" (HTGR) concept. Fort St. Vrain is cooled by helium gas, while all other commercial reactors in the United States are cooled with water (these reactors are referred to as light-water reactors). The HTGR concept was first tested at the 40 megawatt Peach Bottom reactor in Pennsylvania by the Philadelphia Electric Company from 1967 through 1974.

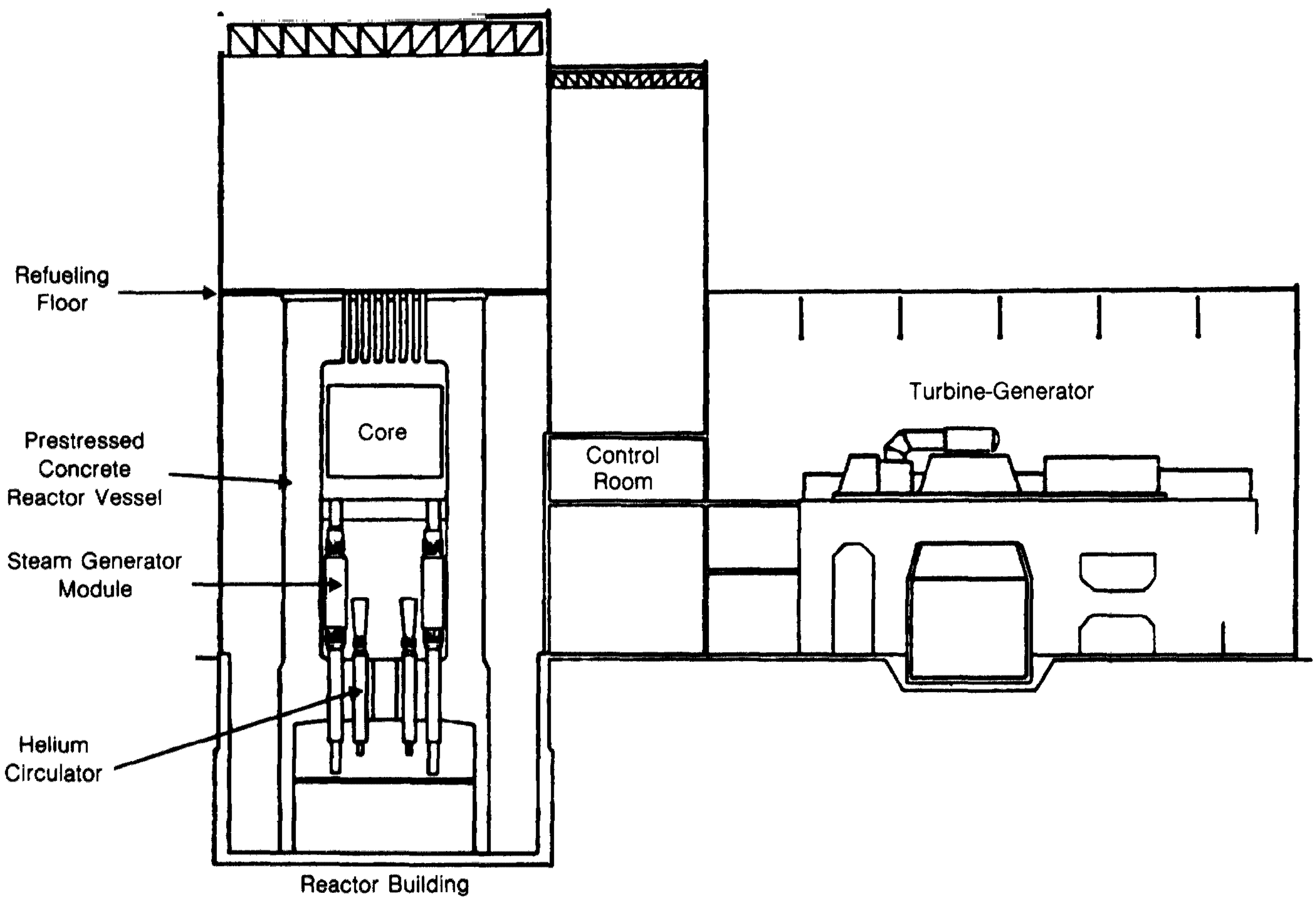
The Fort St. Vrain plant started commercial operation in 1979; however, because of various problems the reactor operated intermittently for a total of only 38 months, from mid-1979 to April 1987. The major problem during this period has been numerous equipment failures and malfunctions. A recent example was an oil fire in the turbine building that occurred on October 2, 1987, and forced the reactor operators to shut down the reactor. The fire started as a result of oil leaking onto a hot valve, which caused the oil to ignite. Although NRC's preliminary assessment of the accident indicates that neither the plant nor any of its systems were in jeopardy, the plant will remain shut down for at least 3 weeks for repairs.

While the Fort St. Vrain plant is the only operating commercial HTGR in the United States, gas-cooled graphite reactors have achieved wide acceptance in other countries. Britain and France started developing gas-cooled graphite reactors in the 1950s and 1960s, and today Britain has 40 such reactors in operation or under construction, and France has 4. More than 800 reactor operating years have been achieved by these two countries. The reactors are similar to Fort St. Vrain, except they operate at a lower temperature.

The reactor at the Fort St. Vrain power plant, like the Chernobyl reactor, uses graphite as its major core construction material. The graphite blocks used at Fort St. Vrain are hexagonal —about 14 inches across and 31 inches high—and are stacked vertically to form the reactor core, which is 15.6 feet high and 19.5 feet across. The blocks contain vertical channels that allow the helium gas coolant to flow through the core from top to bottom. The blocks also contain nuclear fuel and vertical channels for the reactor control rod system.

The reactor core, coolant circulators, and steam-generating system are enclosed in a cylindrical prestressed concrete vessel with walls 9 to 15 feet thick and 106 feet high. The inside of the vessel is lined with steel. The liner acts as a seal to maintain the gas coolant. The liner and the concrete walls are cooled with circulating water. There are 12 penetrations through the vessel for steam pipes that run to and from the steam generators. Unlike light-water reactors, the entire cooling system and steam generators are encased in the concrete vessel. A more detailed description of the Fort St. Vrain design is presented in chapter 2. A diagram of the Fort St. Vrain nuclear power plant is shown in figure 1.2.

Figure 1.2: Diagram of the Fort St. Vrain Nuclear Power Plant



Source: PSC

NRC Responsibilities

The Energy Reorganization Act of 1974 (42 U.S.C. 5801) created NRC to regulate commercial nuclear activities, including commercial nuclear power plants. Headquartered in Washington, D.C., NRC also has five regional offices. It is responsible for assuring that commercial nuclear facilities in the United States are safely designed and operated. To fulfill its responsibility, NRC reviews and approves nuclear facility designs, licenses construction, and inspects all phases of the construction for compliance. Once a facility becomes ready for operation, NRC licenses the operation and inspects it using its rigid technical specifications and standards. Utilities that design, construct, and operate nuclear power plants must meet NRC's specifications and standards or face possible fines and/or plant shutdowns.

As of October 1986, there were 98 commercial nuclear power plants licensed to operate and regulated by NRC. At least one NRC resident inspector is assigned to each nuclear power plant to assure compliance with NRC regulations. NRC also periodically inspects the operations of each nuclear power plant, using its regional personnel with expertise in specific areas of safety and operations. These inspections are conducted on all aspects of plant operations, including reactor operations, operator training and qualification, physical security, emergency preparedness, and maintenance. The inspections cover a broad and voluminous range of specifications, procedures, standards, and requirements. For example, reactor operations require periodic calibration of numerous gauges that record various reactor operating conditions, surveillance of plant equipment, and monitoring of other operating parameters such as coolant temperature, reactor vessel pressures, position of valves, and water chemistry to assure that all are within established limits.

All NRC-licensed nuclear power plants are required to report to NRC any events that adversely affect plant safety. NRC studies these events in an effort to improve the safety at all plants. When an event that adversely affects plant safety occurs, NRC may require corrective action by the utility before the plant is permitted to operate. This action is usually inspected for compliance.

Because Fort St. Vrain uses graphite as a major core component, NRC felt it necessary to review the safety of the plant in view of the Chernobyl accident. In addition, NRC also asked PSC to perform safety-related studies at Fort St. Vrain in view of the events that had occurred at Chernobyl.

Objectives, Scope, and Methodology

In his August 29, 1986, request, Senator Timothy E. Wirth asked that we answer the following five questions pertaining to the safety and operations of the Fort St. Vrain commercial nuclear power plant:

- How does the Fort St. Vrain reactor design compare with the design of the Chernobyl reactor?
- What risks are associated with any similarities between the two plants, or with any other Fort St. Vrain design features?
- What are the safety features at the Fort St. Vrain reactor, and how would those safety features function in the series of events that may have occurred at Chernobyl?
- What was the basis for NRC's recent assurance that there is no danger of a Chernobyl-like accident occurring at Fort St. Vrain? Are NRC's findings sound?
- In what areas has NRC judged management at the Fort St. Vrain plant to be deficient? Have those deficiencies affected operations or safety?

In addition, Senator Wirth asked that we assess the adequacy of emergency planning for the Fort St. Vrain reactor.

To answer the questions concerning the design, similarities, risks, and safety of Fort St. Vrain compared with the Chernobyl reactor, we drew upon the Chernobyl design and operation information contained in our August 1986 report. We also reviewed a Soviet report released at an International Atomic Energy Agency meeting in Vienna, Austria, in August 1986.[3] The Soviet report included information on both the design of the Chernobyl reactor and the accident. In addition, we reviewed a Department of Energy report, prepared by a team of technical experts, on the accident sequence at Chernobyl.[4]

To develop an understanding of the design and operation of the Fort St. Vrain reactor and evaluate NRC's basis for concluding that a Chernobyl-like accident could not occur there, we reviewed pertinent information on the reactor design and interviewed a consultant working at Department of Energy's Oak Ridge National Laboratory. In addition, we interviewed NRC officials at headquarters in Washington, D.C.; the regional office in Arlington, Texas; and at Fort St. Vrain. We also interviewed PSC officials, visited the reactor site, and discussed PSC operations with site

[3]U.S.S.R. State Committee on the Utilization of Atomic Energy—The Accident at the Chernobyl Nuclear Power Plant and Its Consequences, August 1986.

[4]Report of the U.S. Department of Energy's Team Analyses of the Chernobyl-4 Atomic Energy Station Accident Sequence (DOE/NE-0076) November 1986.

personnel. Further, we interviewed officials of GA Technologies, the company that designed the Fort St. Vrain reactor.

To evaluate management deficiencies and their effect on operations and safety at Fort St. Vrain, we obtained and reviewed past NRC assessments, interviewed NRC officials, and discussed the NRC-cited deficiencies with PSC officials. We also reviewed management improvements presently being implemented by PSC and attended NRC hearings in Washington, D.C., in October 1986 and February 1987, held by the NRC commissioners concerning management deficiencies at Fort St. Vrain.

To determine the adequacy of on-site emergency preparedness, we interviewed officials at NRC headquarters in Washington, D.C., and at the regional office in Arlington, Texas. We also reviewed NRC's inspection reports, evaluations of PSC's management of Fort St. Vrain's emergency preparedness program, and reports on PSC's annual emergency preparedness exercise. We interviewed PSC officials responsible for emergency preparedness and reviewed PSC's emergency preparedness plan to determine related roles and responsibilities.

To determine the adequacy of emergency preparedness beyond the Fort St. Vrain boundaries, we interviewed Federal Emergency Management Agency (FEMA) officials in the headquarters' offices in Washington, D.C., and at FEMA's regional office in Denver, Colorado. We reviewed results of the biennial emergency preparedness exercises conducted in 1983 and 1985 and the results of a survey of the effectiveness of a public notification system. We also reviewed state and county agencies' emergency preparedness plans to determine their roles and responsibilities and interviewed state and county officials responsible for emergency preparedness around Fort St. Vrain.

We discussed the facts presented in this report with PSC officials and the NRC resident inspector at Fort St. Vrain; and the NRC Project Manager for Fort St. Vrain at NRC headquarters. We incorporated their comments where appropriate. However, as agreed with your office, we did not ask PSC or NRC officials to comment officially on this report.

Our work was conducted between August 1986 and June 1987 and was performed in accordance with generally accepted government auditing standards.

Chapter 2 of this report compares the designs of the Fort St. Vrain and Chernobyl reactors and the safety significance of the differences. It also describes the Chernobyl accident, its implications for Fort St. Vrain, and the NRC basis for concluding that a Chernobyl-type accident could not occur at Fort St. Vrain. Chapter 3 discusses management deficiencies at Fort St. Vrain and PSC's progress in correcting the deficiencies. Chapter 4 addresses the emergency preparedness program at Fort St. Vrain.

Comparison of Chernobyl and Fort St. Vrain Reactor Designs

A comparison of the various components, systems, and operating parameters of the Fort St. Vrain and Chernobyl reactors indicates that while both reactors have graphite cores and utilize the same basic reactor systems that perform the same functions, there is very little similarity in the specific designs of the systems. Consequently the two reactors differ considerably. Fort St. Vrain's design provides a much wider margin of safety than the Chernobyl design, according to information provided by PSC management and studies performed by NRC. In addition, the differences in design appear to lend support to NRC's conclusion that there is no danger of a Chernobyl-type accident at Fort St. Vrain.

Major Systems and Components of Fort St. Vrain and Chernobyl

While the two reactors employ components and systems that perform the same basic functions—reactor fuel, cooling system, reactor core, and systems to prevent release of radioactive materials—they are quite different in specific design and operation. The specific design and operation of these components and systems determine the safety of the reactor during normal operations and accident conditions. In addition, the specific design of a reactor also determines its reactivity coefficient[1] and reactor controllability, which are especially important under accident conditions. The reactor systems, which are summarized in table 2.1, are discussed in this chapter.

Table 2.1: Comparison of Fort St. Vrain and Chernobyl Reactor Systems

Reactor Systems	Fort St. Vrain	Chernobyl
Fuel system	Coated fuel particles; slow to heat and able to withstand high temperatures	Fuel pellets in zirconium fuel rods; low heat capacity, subject to melting in a few seconds if cooling is lost
Cooling system	Helium coolant; coolant does not contact fuel, and helium provides an inert, dry atmosphere	Pressurized boiling water cooled; will produce hydrogen gas if fuel comes in contact with coolant
Core	Graphite; designed to absorb large quantities of heat in an accident situation	Graphite; not designed to be the primary heat absorber in an emergency
Radiation release protection	Pressurized vessel system designed to withstand pressures created by an accident	No containment or confinement system as used in the U.S.; design incorporated some pressure suppression and retention features.
Reactivity coefficient	Predominately negative coefficient; increase in core temperature reduces rate of nuclear reaction	Positive power coefficient; increase in core temperature caused by insufficient coolant increases rate of nuclear reaction
Controllability	37 separate control regions; monitored and adjusted from the control room	2 control regions that react differently in accident situations

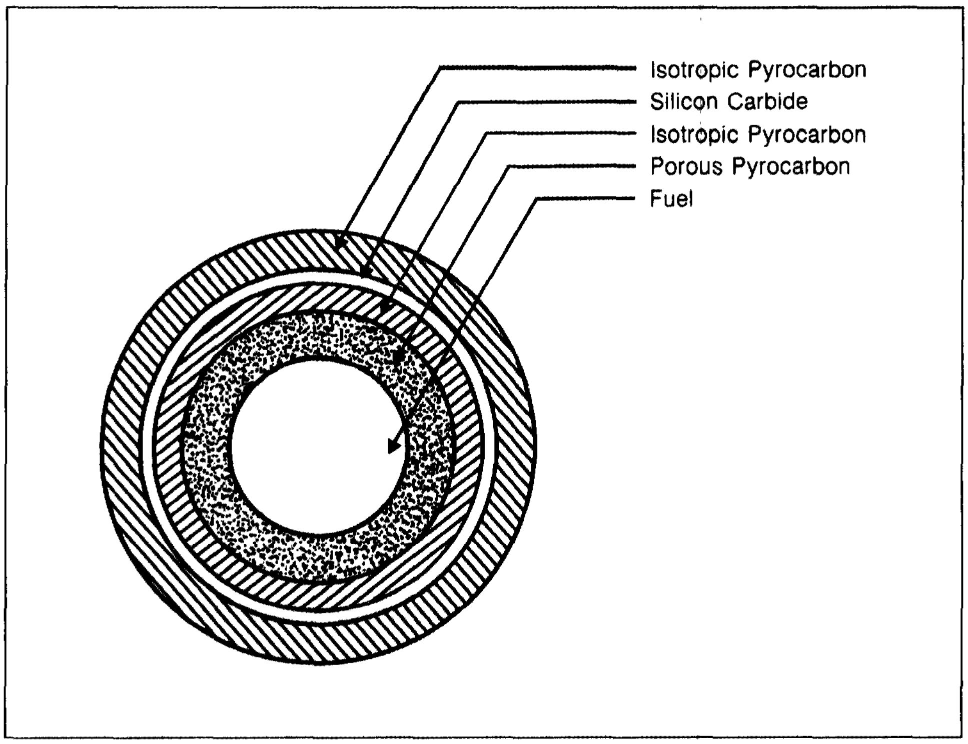
[1] Reactivity coefficient measures the change in nuclear reaction caused by change in power.

Reactor Fuel

The reactor fuel is important from a safety standpoint because it contains radioactive materials that must be prevented from escaping into the environment. The more resistant the fuel is to degradation by high, prolonged temperatures, the less likely radioactive materials will escape into the environment. In addition, during accident conditions the longer it takes the fuel to heat to the point where it begins to deteriorate, the more time is available to respond to the problem.

The fuel used in the Fort St. Vrain reactor is fabricated by applying four separate coatings around micro particles of uranium and thorium fuel (see fig. 2.1).[2]

Figure 2.1: Diagram of Fort St. Vrain Fuel



Source: PSC

[2]Thorium is a radioactive metallic element used to produce fissionable fuel.

Each of the coatings has a separate function in protecting the fuel from damage during normal operations and accident conditions. The coating directly adjacent to the fuel is a porous pyrocarbon[3] material that allows for expansion and accommodates radioactive gases created when thorium and uranium atoms are split. The second coating, isotropic pyrocarbon,[4] provides the first barrier to release of fission products. The third coating, silicon carbide, is another form of carbon that resists high temperatures and prevents metallic fission products from escaping outside the fuel. The outer coating is the same as the second, isotropic pyrocarbon, and is used to compress the silicon layer to prevent it from cracking.

The finished fuel particles, each of which is about the size of a grain of sand, are molded into graphite rods about 1.5 inches long and 0.4 inches in diameter. These small rods are inserted into drilled holes in the graphite core blocks to form the fuel elements. Each graphite block contains 210 fuel holes that accommodate the small graphite rods.

Peak temperatures obtained in the Fort St. Vrain reactor during normal operation are about 2,000° F. The pyrocarbon and silicon carbide fuel particle coatings or claddings will retain fission products to temperatures of over 3,200° F or about 1,200° F above normal peak temperature. In addition, the fuel at Fort St. Vrain is surrounded by graphite, which has a high heat capacity. If the gas coolant is lost, the heat-up rate is slow because the graphite absorbs the heat.

In contrast, the Chernobyl reactor contained 189 metric tons of uranium fuel. This fuel was made by fabricating uranium dioxide into small pellets and inserting them into rods. The rods were about 1/2 inch in diameter and about 40 inches long. Thirty-six of these rods were bundled in each of the 1,661 pressure tubes through which the coolant flowed. The coolant was in direct contact with the zirconium cladding of the fuel rods, and the pressure tubes were vertical and extended through the reactor core from top to bottom.

During operation, the temperature of the fuel at Chernobyl ranged from 2,500° F to 3,000° F. The fuel melting temperature of the Chernobyl fuel was 3,300° F, or only 300° F to 800° F above the operating temperature.

[3]Pyrocarbon refers to carbon that is fabricated using very high temperatures, thus making it resistant to similarly high temperatures.

[4]Isotropic means the material has the same properties (strength, heat resistance, etc..) in all directions.

Further, the fuel at Chernobyl was enclosed in zirconium, which has a low heat capacity when compared with graphite. Thus, if cooling was lost, the zirconium would heat up immediately and melt within a short period unless cooling was quickly restored.

In summary, the fuel of the Fort St. Vrain reactor is designed to withstand higher temperatures and transfers its heat to the graphite, thus providing a much slower heat-up-rate than the Chernobyl reactor. Translated into safety, the Fort St. Vrain fuel design provides hours to avert a major accident, while the Chernobyl fuel design provided only seconds.

Cooling Systems

The cooling system in a nuclear power reactor serves two purposes. First, it removes heat produced by the fuel from the reactor core, thereby preventing the fuel from becoming too hot. Second, it transfers the heat obtained from the fuel to some sort of system to generate electricity. The design of the Fort St. Vrain cooling system precludes a buildup of explosive gases created by the interaction of the fuel and coolant during accident conditions. However, during the Chernobyl accident, the fuel and coolant did interact and produced explosions that possibly added to the dilemma at Chernobyl.

The coolant used in the Fort St. Vrain reactor is helium gas that is pressurized to increase its density and enable it to absorb more heat. The normal operating pressure of the gas is about 688 pounds per square inch gauge (psig),[5] and the normal operating temperature of the helium ranges from a high temperature of 1,400° F at the bottom or gas outlet region of the core to a low temperature of 700° F after being circulated through the steam generators. The helium gas flowing through the core removes the heat being generated by the activity of the neutrons in the core.

The gas is circulated by four large fans through the thousands of small holes in the graphite blocks. After passing through the graphite core, the heated helium is then cycled through 12 steam generators, which cool the helium by transferring its heat to water and/or steam. This is a continuous process, so the fans, which circulate the helium gas, must run constantly.

[5]Pounds per square inch gauge is a measure of the difference between actual pressure and atmospheric pressure.

Unlike the Fort St. Vrain reactor, the reactor at Chernobyl used pressurized boiling water, circulated through each of the 1,661 pressure tubes, to cool the fuel contained in the pressure tubes. The reactor had two main cooling loops, with each loop providing coolant to one-half of the reactor. Four main pumps were used to circulate the coolant through each loop. A main line supplied coolant to the bottom of the reactor where it went through a series of headers that separated it into smaller lines until each of the pressure tubes in the loop had its own individual inlet line.

The water entered the bottom of the pressure tubes and moved upward. About one-third of the way up the pressure tubes, the water began to boil, producing steam that was collected through a series of headers at the top and routed to the steam separator tank. The steam separated from the water was then routed to the steam turbine generators. After leaving the generators, the steam was cooled and condensed back into water and returned to the reactor. The water removed in the separator was also circulated back to the reactor.

The major advantages of the Fort St. Vrain design are that the helium gas is dry, does not come in direct contact with the fuel, and provides an inert atmosphere throughout the pressurized reactor vessel. Thus, according to a PSC official, chemical reactions between the coolant and reactor components are nearly impossible. In addition, if cooling is lost at Fort St. Vrain, ample time is available to analyze problems and take action.

The Chernobyl reactor, however, was not designed to withstand loss of coolant without the use of the emergency cooling system. If cooling is inadequate for a very short period (seconds), voids in the cooling would be created, and the fuel would begin to melt.[6] This would result in over-pressurization of the fuel tubes and explosions. In addition, if damaged fuel cladding came in contact with hot water, hydrogen gas would be formed. This scenario is thought to be similar to the series of events that actually occurred. For a hydrogen explosion to occur, however, oxygen must also have been present. While the Chernobyl reactor design employed a system to prevent oxygen from entering the reactor area, that system was made inoperable by the initial explosion.

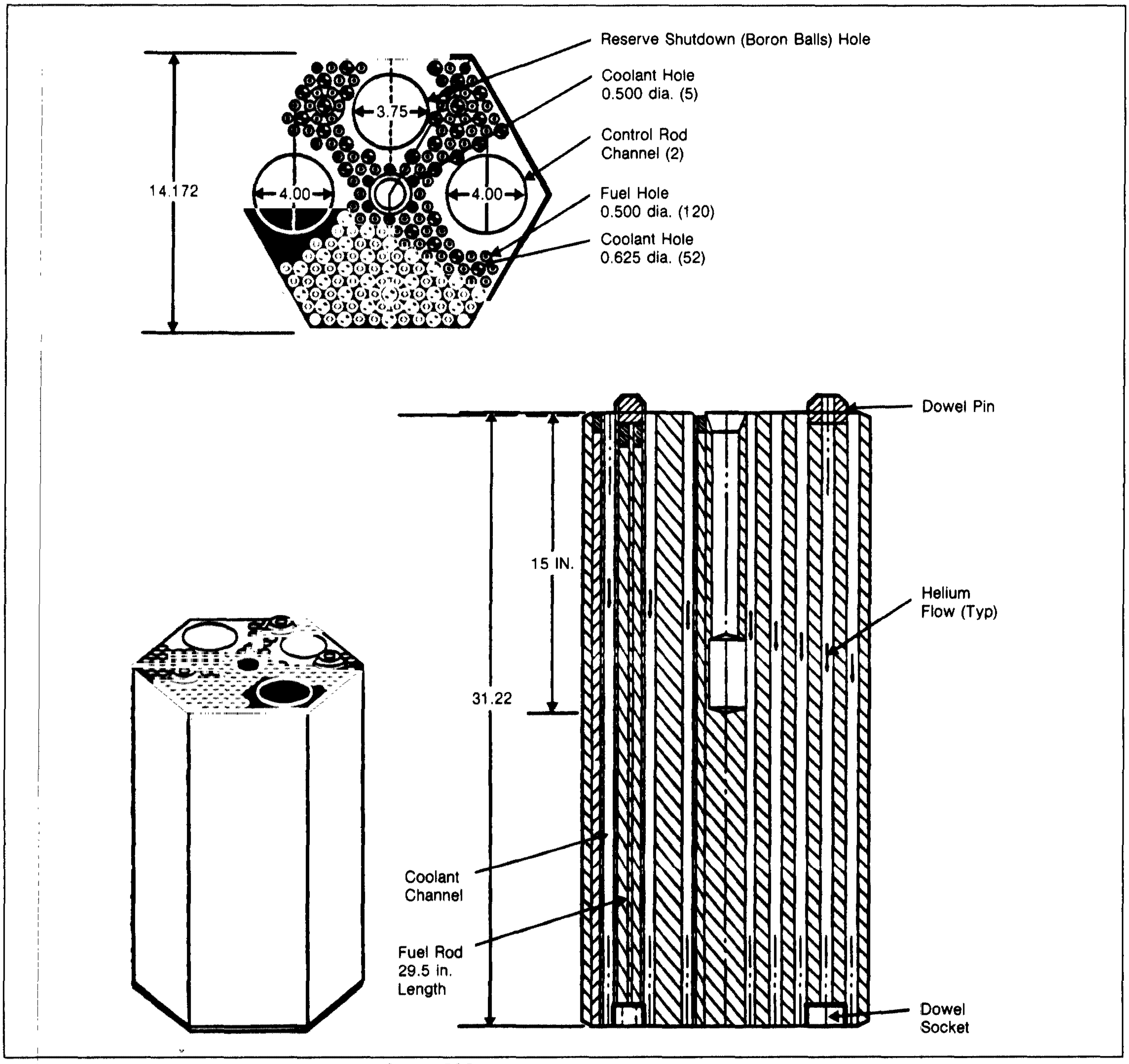
[6] A coolant void occurs when the coolant (water) boils and creates air bubbles.

Reactor Core

The reactor core is the primary component of a reactor. It contains the fuel to produce heat, a cooling system to remove the heat, and a moderator and a control rod system to assure proper reactivity. The reactor cores at Chernobyl and Fort St. Vrain both use massive amounts of graphite. However, the design of the Fort St. Vrain core and related systems allows the graphite to act as the primary heat-absorbing agent. In contrast, the Chernobyl reactor core and related systems were designed so that the coolant (water) was the primary heat absorber. Under accident conditions, the graphite core at Fort St. Vrain acts as a heat absorber, providing hours to correct a problem, whereas the core design at Chernobyl provided only seconds.

The reactor core at Fort St. Vrain is constructed primarily of hexagon-shaped graphite blocks stacked in columns to form a structure approximately 15.6 feet high and 19.5 feet in diameter. The structure contains about 1,500 tons of graphite. Each block is about 31 inches high and 14 inches across (see fig. 2.2).

Figure 2.2: Diagram of a Fort St. Vrain Reactor Graphite Core Block



Source: PSC

The blocks are stacked in columns in a dowel and pin arrangement that aligns each block. The reactor core consists of 247 vertical columns that are about 6 blocks high. The coolant holes extend vertically through each column, thus permitting the gas coolant to flow completely through the core structure.

In contrast, the core of the Chernobyl reactor was a cylinder-shaped graphite structure 39 feet across the top and 23 feet high. It was surrounded by a steel shroud with plates on the top and bottom that sealed the reactor. Enveloping the core area were two more cylinders, one surrounding the other, with about 4 feet of space between them. These two cylinders formed a water jacket that removed heat produced by the reactor. The jacket was divided vertically into 16 sections, and water was circulated by the main cooling system. Outside the water jacket was a layer of ordinary sand and a final exterior wall made of concrete. The water and sand helped reduce neutron bombardment on the concrete walls.

The reactor core contained 2,488 columns of graphite blocks. Total weight of the graphite in the Chernobyl core was about 1,875 tons. The blocks were about 24 inches high and 10 inches across. They were stacked end to end to form the columns, and the outermost blocks had steel rods through them, welded at the top and bottom to hold the core in place. The center of the blocks or columns had holes 4-1/2 inches in diameter that accommodated the fuel and control rod system.

One of the major differences between the two reactor cores is the amount of graphite and the ability of the Fort St. Vrain design to absorb heat directly into the graphite. The Fort St. Vrain reactor is rated at 842 megawatts of thermal power and has about 1,500 tons of graphite in its core.[7] In contrast, Chernobyl was rated at 3,140 megawatts thermal power and had about 1,875 tons of graphite. Thus, on a per megawatt basis, the Fort St. Vrain reactor contains more graphite. In addition, the Fort St. Vrain design allows the graphite to serve as the primary absorber of heat, whereas in the Chernobyl design the coolant water is the primary absorber of heat.

[7]Megawatt thermal is a measure of heat while a megawatt electric is a measure of electric power. About 3 megawatts thermal are required for each megawatt electric produced.

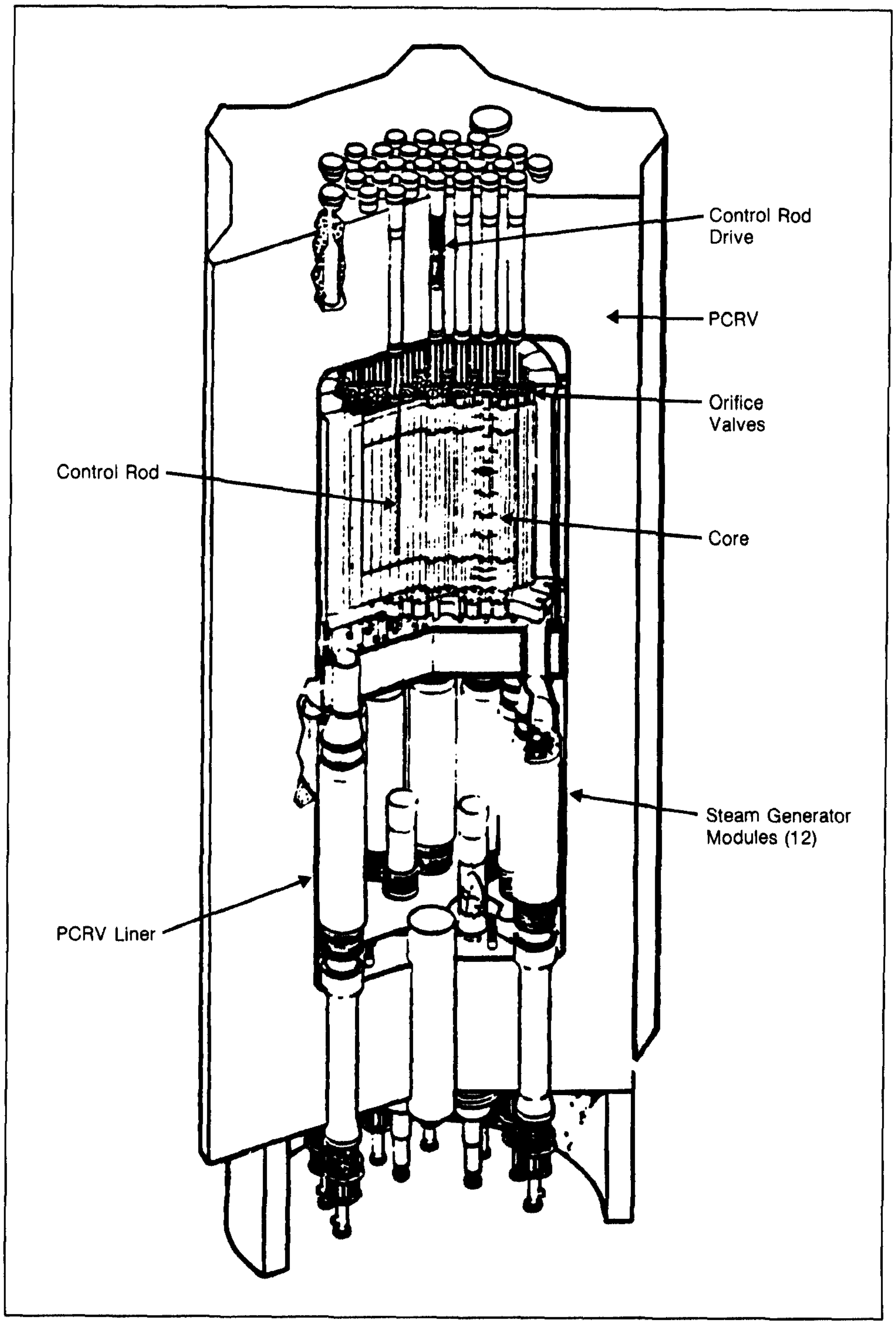
Radiation Release Protection Systems

To provide safety, nuclear power plants must have a system to prevent radioactivity from escaping into the environment if the fuel becomes degraded and releases radioactive material. This system usually consists of a structure around the core that either contains or provides for controlled releases of radioactive materials through vents and filters.

Fort St. Vrain uses a pressurized vessel system that acts as a containment and, in addition, it has a confinement system. The Chernobyl plant employed only a partial confinement system. The main difference between the systems is that containment does not allow for planned, monitored releases, while a confinement system makes use of this type of release during an accident.

The pressurized vessel that acts as a containment at Fort St. Vrain houses the core, coolant circulators, steam generators, and other components in a large reinforced concrete pressure vessel. The reactor core and other components are located in a cylindrical cavity inside the vessel, which is 31 feet in diameter and 75 feet high. The walls around the cavity are a minimum of 9 feet thick, and the top and bottom walls are 15-1/2 feet thick (see fig. 2.3).

Figure 2.3: Diagram of Fort St. Vrain's Pressurized Vessel



Source: PSC

The helium gas coolant, which fills the cavity inside the pressure vessel, is pressurized to about 688 psig, thus requiring the concrete to be reinforced to contain such pressures. This is accomplished by a network of 448 wire or cable tendons that run vertically through the vessel, around it, and across the top and bottom. The tendons are attached to steel plates located on the outside walls of the vessel. The tendons are stretched tight and fastened to the steel plates, which actually compress the entire concrete vessel. The tendons limit any expansion or distortion that might occur during reactor operation and enable the pressure vessel to accommodate up to 1,875 psig. In addition, there are two pressure rupture discs that are designed to release at 845 psig to serve as protection against overpressurization.

In contrast, the Chernobyl reactor was enclosed in a steel shroud that was designed to withstand the pressure resulting from a single pressure tube failure. In addition, the area immediately outside the shroud was pressurized to a higher level than the inside so that any leaks would be into instead of out of the reactor space. The reactor was further surrounded by steel, concrete, and other materials. The top of the structure provided access to the fuel in the pressure tubes for refueling purposes.

Reactivity Coefficient

Reactivity coefficients measure the inherent physical response—the changes in the rate of nuclear reaction—to changes in temperatures of the moderator, coolant, and fuel. In all graphite-moderated reactors, the nuclear reaction tends to increase as the temperature of the graphite increases (over a limited temperature range). This is an undesirable feature in graphite reactors and is referred to as a "positive coefficient." In addition, if the coefficient of the coolant and fuel are not negative enough to overcome the positive effect of the graphite, there is an overall positive coefficient that is extremely undesirable. A reactor with an overall positive coefficient can result in an uncontrollable power rise and a runaway reactor.

In the United States, however, there are no commercial nuclear power reactors with positive coefficients. NRC's design criteria require that reactors be designed to prevent a rapid increase in reactivity in the power operating range. According to an NRC official, these criteria require that commercial reactors operate with a negative coefficient. This important safety aspect is carefully reviewed by NRC before approving the licenses of any utility to operate a commercial nuclear power reactor.

Despite the positive coefficient of its graphite, the Fort St. Vrain reactor has an overall negative coefficient, which means that any increase in core temperature reduces the rate of nuclear reaction and tends to shut the reactor down. This overall negative coefficient is possible because the fuel has a large negative coefficient that offsets both the positive coefficient of the graphite and the small positive coefficient of the coolant. Thus, the net effect is a negative coefficient.

At Chernobyl, this was not the case. While the fuel in the Chernobyl reactor had a negative coefficient, the coolant had a positive coefficient. Thus, with the positive power coefficient of the graphite, the net effect was an overall positive coefficient. In this respect, when temperature increases and insufficient cooling exists, voids or air spaces are created that increase the rate of nuclear activity and thus temperature. These increases result in additional voids and have a spiral effect. It is believed that a sudden increase in thermal power experienced at Chernobyl just seconds before the reactor exploded was caused, in part, by the effect of the positive coefficient.

Reactor Controllability

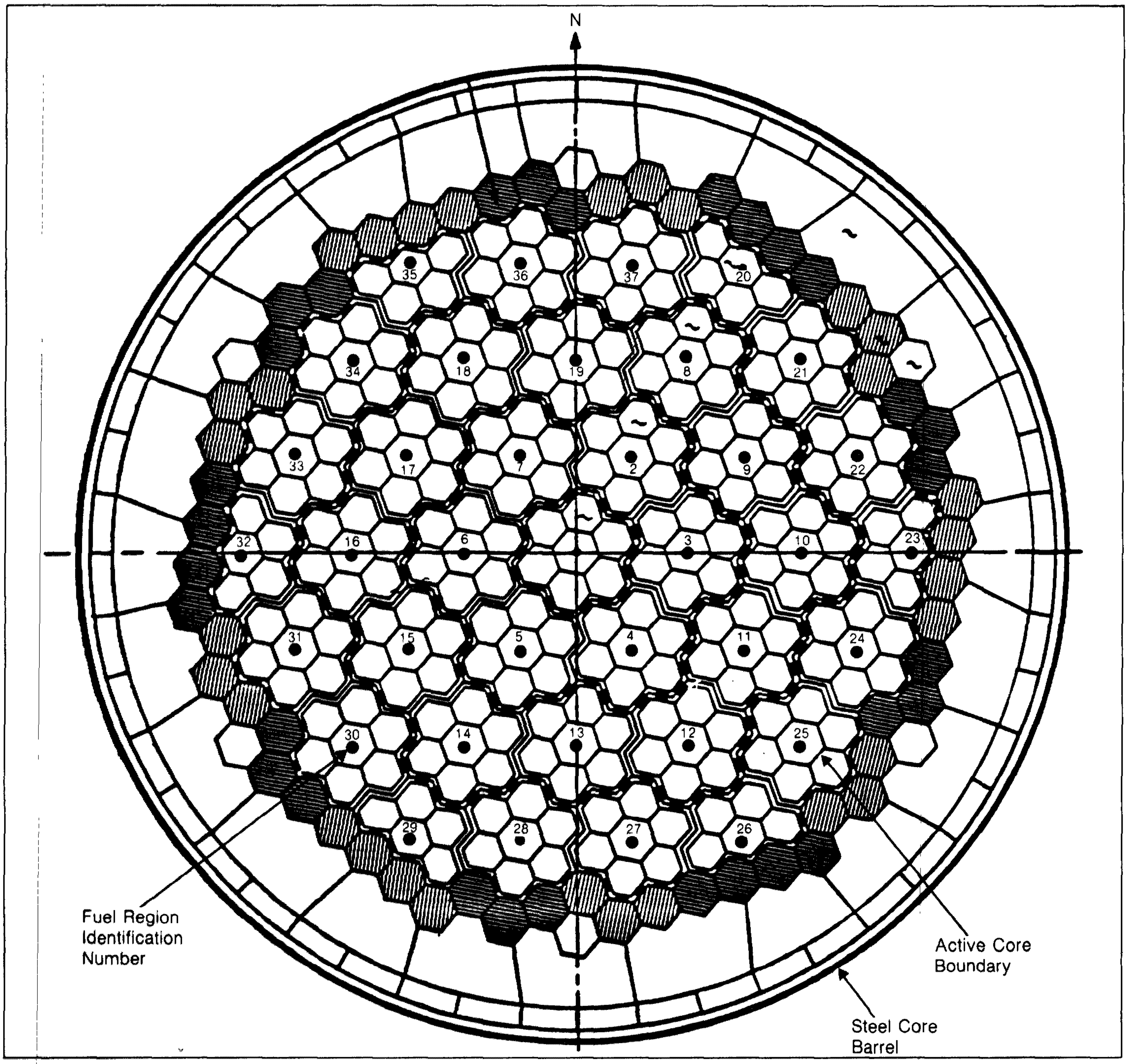
Reactor controllability is how effectively the reactivity in the reactor can be controlled. The variables that affect reactor controllability are coolant circulation and control rod position. Coolant circulation helps control the fuel's temperature and prevents overheating of the reactor. Control rod position regulates the nuclear reaction. When control rods are inserted into the reactor core, the nuclear reaction is slowed or stopped. As the control rods are withdrawn, the reaction increases.

The Fort St. Vrain reactor is controlled by regulating the coolant flow and positioning the boron control rods in 37 separate areas called "fuel regions."[8] Each fuel region, which normally contains seven vertical columns (with the center column containing channels for two control rods and an emergency safety system) is covered at the top with a hood-like arrangement that seals the fuel region. (See fig. 2.4.)

The control rod mechanism, which penetrates the hood, is also sealed (see fig. 2.5).

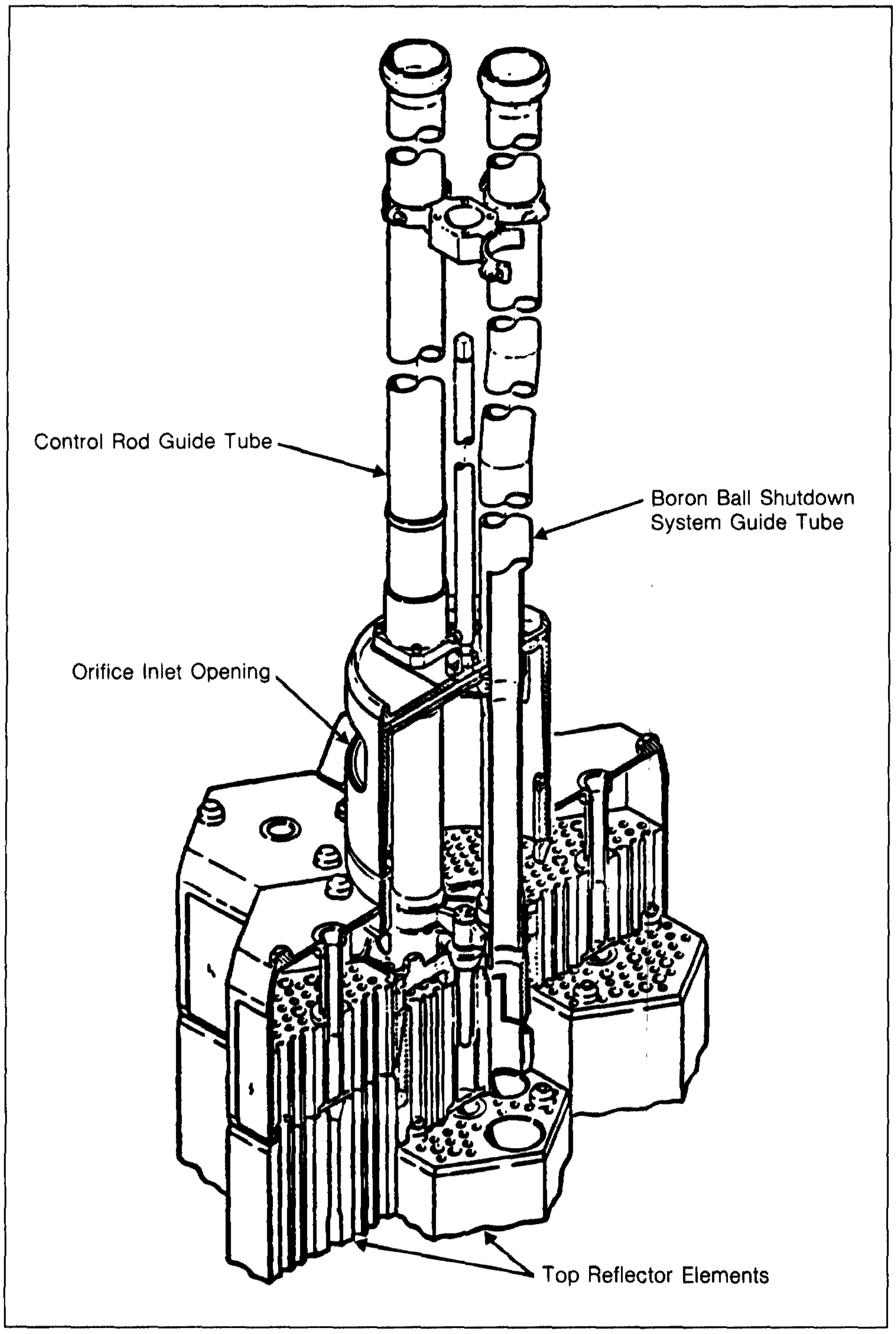
[8]Boron is an element with the ability to absorb neutrons in atoms and therefore regulate or stop nuclear reactions.

Figure 2.4: Diagram of Fort St. Vrain Fuel Regions



Source: PSC

Figure 2.5: Diagram of Control Rod Mechanism



Source: PSC

Coolant to each fuel region is controlled by an inlet hole or orificing device that is monitored and regulated from the reactor control room. Control rods are also monitored and regulated from the control room.

Each of the 37 regions in the Fort St. Vrain reactor has its own set of control rods. These control rods contain boron, which controls the rate of the nuclear reaction by absorbing neutrons. By inserting the control rods, operators can stop the operation of the reactor. Removing the control rods increases the rate of nuclear reaction.

The Chernobyl design has two coolant loops. Each loop is routed into one-half of the 1,661 pressure tubes and is totally independent of the operation of the other loop. According to Soviet technical papers, it is possible with this configuration that each half of the reactor could respond differently if a problem occurred in one of the two loops. For example, an inadvertent activation of one of its two emergency cooling systems, each of which supplies one-half of the core, could cause a change in power in that half of the reactor. This would result in an automatic adjustment of the other half, which can cause a rise in power and imbalance of the two halves of the reactor.

While the Fort St. Vrain reactor has 37 separate control areas, it is still easier to control than the Chernobyl reactor. The 37 areas are constantly monitored by a computer and adjusted by control room operators. The temperature of the coolant exiting the 37 regions is constantly averaged, and if any region significantly differs from the average, the reactor is adjusted. In the Chernobyl reactor, there are only two control areas. Thus, an imbalance of coolant or power in one affects the other.

Fort St. Vrain's Design Precludes a Chernobyl-Type Accident

According to the Soviets, the Chernobyl accident was directly related to and partially caused by the design of the reactor. In addition, the reactor physics (reactivity coefficient) played a major role in the accident. Because of the differences in the design features and the reactor physics of Fort St. Vrain and the Chernobyl reactors, as discussed previously in this chapter, the possibility of a Chernobyl-type accident occurring at Fort St. Vrain is considered beyond the credible range by PSC and NRC officials.

This section provides an account of events leading to the Chernobyl accident and demonstrates the inability of the design to provide any margin of safety for operator error. In addition, the design of Fort St.

Vrain is discussed in light of what would be the consequences of major systems degradation.

The Chernobyl Accident

Following the Chernobyl accident, many questions arose concerning the events leading to it and its probable cause. As a result, the Soviet Union formed a "team of experts" to review the accident and determine its cause. Their report was released in August 1986 and was the subject of a meeting held by the International Atomic Energy Agency in Vienna, Austria. Soviet nuclear experts presented their report at the meeting and answered other nuclear experts' questions.

According to the information provided in the Soviet report and Department of Energy's detailed analysis of the information, the accident occurred while the reactor operators were conducting a test to determine the ability of a turbine generator to provide emergency electrical power after they shut off the steam that normally drives the generator. The test was designed to demonstrate that generator rundown or inertia could provide sufficient electrical power until emergency diesel generators could start up and provide electric power.

To perform the test, several safety systems were turned off or overridden, including the emergency core cooling system, automatic shutdown system, and an automatic control rod safety system. In addition, the test required the reactor to be at a stable low power. Attempts to achieve a stable low thermal power were never successful, and compensatory actions by the reactor operators had already placed the reactor in a very unstable condition when the test was initiated.

When the steam flow to the generator was turned off, initiating the test, four of the main coolant pumps powered by the generator began to rundown. This reduced the coolant flow and increased steam pressure, so that the thermal power level more than doubled in 3 seconds. The power level continued to increase in a runaway fashion as the coolant flow continued to decrease because of the generator rundown.

In less than 20 seconds, the reactor power level increased to 100 times its designed power level, thus producing increases in temperature that destroyed the integrity of the fuel, caused coolant voids, and created sharp pressure increases in fuel channels. In addition, after the initial explosion, the steam interacted with the zirconium fuel cladding and produced other explosive gases. However, it is not clear whether these gases contributed significantly to the destruction of the reactor.

In summary, the accident at Chernobyl started while the reactor was operating at less than 10 percent power; it occurred with incredible speed; and it ended with a devastating thermal explosion. Almost all the control rods had been withdrawn from the reactor, and the coolant circulation was inadequate. The power surge was so rapid that neither the control rods nor the coolant could be adjusted fast enough to stop the accident. Because the fuel was adjacent to the coolant, the coolant's temperature also immediately increased once the power level and temperature started to increase. Thus, coolant voids were created, and the fuel began to melt. The expansion of heat in the sealed pressure tubes resulted in an explosion and the destruction of the reactor. The steam then interacted with the failed fuel cladding and created explosive gases. It was less than 30 seconds from the time it was realized the reactor was in trouble until the explosion occurred.

Consequences of Major Systems Failure at Fort St. Vrain

NRC requires all owners of commercial nuclear power plants to submit a "Safety Analysis Report" (SAR) for NRC review and approval before the plant can operate. The SAR addresses the consequences of major credible accidents that might occur as a result of the loss of one or more of a reactor's major components or systems necessary for safe operation. For example, one of the more significant events usually analyzed is a loss-of-cooling accident.

The SAR for Fort St. Vrain analyzes a loss-of-cooling accident, along with several other less significant accidents. In addition, since the Chernobyl accident, PSC has conducted two studies that address accidents considered beyond the credible range.

The loss-of-cooling accident analyzed in the SAR assumed a loss of all cooling circulators. It also assumed that the reactor was operating at 100 percent power, the liner cooling system was working, and the control rods were inserted. Under these assumptions, according to the SAR, there would be 5 hours available to correct problems and restart the circulators. However, after 5 hours the circulators could not be restarted because the high internal temperature would damage the steam generators. After 6 hours the fuel would begin to fail and release radiation. However, the pressurized concrete vessel would remain intact and prevent significant release of radiation to the atmosphere. During the first 83 hours, about 28 percent of the fission products in the reactor would be released from the fuel.

All analyses and studies of the Fort St. Vrain reactor show that a rapid rise in reactivity, temperature, and power cannot occur because of the reactor's design. The Fort St. Vrain reactor has a negative reactivity coefficient, does not employ pressure tubes, and can absorb large amounts of heat under accident conditions. In addition, the reactor can be safely shut down in a loss-of-cooling accident if the cooling can be reestablished within 5 hours. The reactor can also be safely shut down with an indefinite loss of helium cooling if it is being operated at 30 percent power or less and the liner cooling system is operating.

In contrast, the Chernobyl accident was initiated by a rapid rise in reactivity, power, and heat. The reactor was destroyed when the coolant was displaced by heat expansion, which resulted in ruptured pressure tubes. The accident developed and occurred so fast that the control rods, which require 20 seconds to insert, could not be dropped effectively, and adequate cooling could not be established.

NRC Concludes Fort St. Vrain Is Sufficiently Safe for Continued Operation Following Chernobyl

Following the Chernobyl accident, NRC reexamined the safety of the Fort St. Vrain reactor because Fort St. Vrain, like Chernobyl, contained massive amounts of graphite in the reactor core. NRC reexamined the Fort St. Vrain SAR in light of the Chernobyl events. NRC also requested PSC, the owner of Fort St. Vrain, to analyze the consequences of a rapid oxidation (burning) of the graphite at Fort St. Vrain.

After its reexamination and the PSC analysis, NRC concluded that continued operation of Fort St. Vrain was justified and could find no reason to take any action regarding such operation.

NRC's reexamination of Fort St. Vrain's SAR included a review of the plant design. The principal design features reviewed were

- coatings on the fuel particles,
- the helium cooling system,
- the steel-lined reactor vessel, and
- the pressurized vessel that houses the reactor.

The reexamination of the SAR also included review of the accident analyses performed at the time of licensing. The worst postulated accident or scenario reviewed was a permanent loss of all forced circulated cooling along with the failure of one of the double sealed penetrations in the vessel. NRC found that the consequences of these accidents were within the limits set in 10 C.F.R. 100 for radiation exposures both within and beyond the facility's boundaries.[9]

The reexamination methodology consisted of reviewing, in light of the Chernobyl event, the SAR and information submitted during the initial licensing process at Fort St. Vrain in 1973. Generally, no new information was identified for further studies. However, because of early reports that the graphite in the reactor core at Chernobyl may have become so hot that it actually produced a self-sustaining fire, NRC felt it necessary to obtain additional information on graphite oxidation. Rapid graphite oxidation had not been analyzed in the original SAR because it was believed there was no credible way it could occur. As a result, NRC requested PSC to conduct an analysis of rapid oxidation of graphite. PSC obtained the services of GA Technologies, Inc. to help with the analysis.

In a scenario for rapid oxidation of graphite, two ingredients have to be present. One is intense heat and the other is an abundant supply of oxygen. Since there was no credible accident that could occur at Fort St. Vrain to provide these ingredients, one had to be invented. Thus, a hole in the bottom of the pressurized vessel and one at the top were postulated to set up the necessary "chimney" effect that would allow a large flow of oxygen through the graphite reactor. With the holes in the vessel, the forced-air cooling system would be ineffective.

Such an accident is viewed as not credible because the only mechanical way a double-sealed penetration can fail is by overpressurization of the seals. The vessel is protected by pressure ruption discs that release when pressure exceeds 845 psig. The double-sealed penetrations are designed for pressures up to about 1,700 pounds per square inch (psi). Even if the vessel would pressurize to 1,700 psi, two of the double-sealed penetrations would have to fail exactly at the same instance because once one failed the pressure would be relieved. Consequently, such an accident is considered beyond the credible range.

However, to analyze the rapid oxidation of graphite, it was assumed the large penetrations at the bottom and top failed, allowing the maximum volume of air flow through the reactor core. Air flow and reactor core heat-up rates were calculated for a 24-hour period. The calculation performed by GA Technologies staff was separately reviewed by others at GA Technologies. This review was done to verify the accuracy of the

[9]The Code of Federal Regulations establishes limits for radiation exposures to the whole body and certain organs.

calculations. In addition, an NRC consultant independently reviewed all the calculations to assure the accuracy of the analysis.

The results showed that about 2.5 percent (weight) of the graphite would oxidize in 24 hours, and because the air flow is limited by the openings, there would be no self-sustaining burning. This rate of oxidation would release 12 percent of the fission products in a 24-hour period.

The 24-hour time period is used because it represents the maximum time necessary to stop the air flow. The scenario calls for flooding the bottom 3-1/2 floors of the entire reactor building with water. This would effectively stop the air flow by stopping the bottom opening.

The off-site dose calculations of this severe accident show that the consequences are within 10 C.F.R. 100 guidelines for the low population zone (10-mile radius) and are exceeded in the immediate area of the plant. However, the incredible nature of the accident precludes it from being considered by NRC in the licensing of Fort St. Vrain.

Fort St. Vrain's Past Management Deficiencies and Poor Performance Are Being Corrected

From August 1982 through May 1986, Fort St. Vrain's plant performance had received increasingly lower ratings from NRC. In its annual assessment of Fort St. Vrain's operational safety for the period ending May 1986, NRC rated the plant's overall performance as minimally satisfactory. In particular, NRC judged PSC's management of the Fort St. Vrain reactor as deficient in the following areas: maintenance, outages, licensing, quality assurance, security, and emergency preparedness. According to NRC, these deficiencies had resulted in decreased plant safety and contributed to the plant's history of shutdowns.

To rectify these deficiencies and address their underlying causes, PSC has developed a broad-based Performance Enhancement Program. Since the program's inception in April 1985, NRC has noted significant operational improvements at the plant but continues to closely monitor the plant activities to ensure continued performance improvement.

The plant had been shut down to upgrade its electrical system, and NRC would not allow restart until it was satisfied that operational improvements were implemented. Because of operational and other improvements implemented as part of the Performance Enhancement Program, NRC commissioners, in April 1987, authorized the restarting of the plant. Since the completion of our audit work, NRC performed an overall assessment covering performance from May 1986 to April 1987, which confirmed significant improvement in five of the seven areas previously rated as deficient.

NRC's Assessment Process

NRC evaluates the performance of nuclear power plant licensees to assure that at least minimum safety levels are achieved and maintained. To accomplish this goal, NRC routinely inspects plants, reviews plant performance data, and meets with licensee management. Routine safety inspections, such as adherence to control room procedures, are made daily by the NRC resident inspectors. Inspections are also made on a weekly, quarterly, or annual basis—or whenever judged appropriate by the cognizant NRC regional office. Other performance-related data are also routinely reviewed by NRC inspectors, such as adherence to NRC regulations and logs of control room activities.

In addition to these routine inspections and reviews, NRC periodically assesses each licensee's overall performance. The purpose of this assessment, called the Systematic Assessment of Licensee Performance (SALP), is to (1) collect information to evaluate licensee performance, (2) provide the basis for NRC resource allocation, and (3) provide guidance to improve licensee performance. These assessments are generally conducted every 18 months, unless NRC determines that a licensee's performance requires more frequent evaluations.

To conduct this assessment, a board of regional and headquarters NRC officials convenes to review performance observations and other data collected during the assessment period. On the basis of their review, the board rates the licensee's performance in 11 functional areas considered essential to nuclear safety and the environment. These areas are maintenance, licensing, security, outages, quality assurance, plant operations, radiological controls, surveillance, fire protection, training and qualification effectiveness, and emergency preparedness. If deficiencies are identified, NRC may meet with the licensee's management and request a written response outlining the licensee's corrective actions.

NRC Assessments Show Operations at Fort St. Vrain Were Minimally Satisfactory

Plant performance at Fort St. Vrain had declined from August 1982 through May 1986. NRC rated plant performance minimally satisfactory in only 1 of the 11 functional areas from 1979 (when the plant opened) until 1982. However, for the assessment period from September 1, 1982, through September 30, 1983, NRC rated the plant minimally satisfactory in four functional areas—plant operations, licensing, design changes and modifications, and management control. In the assessment for the period ending May 1986, NRC rated the plant's performance minimally satisfactory in six functional areas. According to two NRC commissioners, the report for the period ending May 1986 was the worst SALP report they had ever seen.

1986 SALP Assessment Shows Declining Performance at Fort St. Vrain

In its assessment of Fort St. Vrain for March 1, 1985, through May 6, 1986, NRC judged the overall performance as minimally satisfactory in six functional areas: maintenance, licensing, security, outages, quality assurance, and emergency preparedness. Emergency preparedness will be discussed further in chapter 4. The remaining five areas—plant operations, radiological controls, surveillance, fire protection, and training and qualification effectiveness—were rated at either a satisfactory or high level of performance.[1]

Two of the functional areas that were rated as minimally satisfactory, maintenance and licensing, demonstrated a strong positive trend in performance towards the end of the assessment period. In the maintenance area, NRC noted the facility lacked an effective preventive maintenance program and violated NRC regulations. Eight violations resulted either from staff failing to follow maintenance procedures or from inadequate maintenance procedures. NRC concluded that although there were many violations, a strong management involvement to correct the problems became evident toward the end of the assessment period. For example, according to NRC officials, PSC was in the process of initiating effective preventive maintenance in all areas. In addition, PSC was implementing a computerized program to generate and track maintenance work. NRC recommended that PSC complete its revision of maintenance procedures and install the preventive maintenance program in a timely manner.

In the licensing area, which involves the adequacy and timeliness of licensee efforts to comply with NRC regulations and initiatives, NRC recommended that

- PSC increase its licensing staff to more expeditiously and adequately address unresolved issues;
- PSC be more thorough in addressing basic safety problems and emphasize resolution of licensing issues so that minimum standards are met or exceeded; and
- PSC emphasize understanding issues that are relevant to all nuclear power reactors, not just those relevant to the design of Fort St. Vrain.

In the functional area of physical security, for which NRC has requirements to prevent sabotage and terrorism, NRC concluded that PSC management had demonstrated a lack of attention and dedication to

[1]On the basis of the SALP board assessment, NRC classifies each functional area in one of three performance categories:
Category 1 - Reduced NRC attention may be appropriate. Licensee management attention and involvement are aggressive and oriented toward nuclear safety; and a high level of performance with respect to operational safety and construction quality is being achieved.
Category 2 - NRC attention should be maintained at normal levels. Licensee management attention and involvement are evident and are concerned with nuclear safety; and a satisfactory performance with respect to operational safety and construction quality is being achieved.
Category 3 - Both NRC and licensee attention should be increased. Licensee management attention or involvement is acceptable and considers nuclear safety; but weaknesses are evident so that minimally satisfactory performance with respect to operational safety and construction quality is being achieved.

promoting an adequate security program, and a lack of support for correcting potentially serious security deficiencies. NRC further concluded that management relied on compensatory measures for an extended period of time instead of quickly correcting basic security deficiencies. For example, because of deficiencies in the systems designed to detect intruders, such as a closed circuit television system and an intrusion detection system, PSC used security personnel to patrol the plant for over a year instead of correcting the deficiencies. In a subsequent letter to PSC, NRC noted that a recently completed security inspection found that PSC had turned around its performance in this area because of increased management commitment to physical security at the plant.

In the functional area of outages, which includes equipment repairs, restorations, modifications, and other activities associated with times when the plant is not operating, NRC found programmatic weaknesses in scheduling, planning, and coordinating outages. For example, PSC has had difficulty in obtaining necessary repair parts in time to support plant work and in providing work instructions in a timely manner.

When properly implemented, quality assurance provides assurance that activities related to safety meet predetermined standards. Because of the importance of this functional area for safety-related systems, NRC expressed concern over the continuing major deficiencies noted in this area for several SALP periods. As a result of weak management direction and support in this area, NRC found that

- internal audits were not being conducted with appropriate expertise,
- quality assurance management was not conducted independently, and
- the quality assurance department was understaffed and poorly directed.

NRC Concludes That Management Deficiencies Have Reduced Safety and Limited Plant Availability

According to NRC, past management deficiencies at Fort St. Vrain have reduced the plant's margin of operational safety and have also contributed to the plant's history of limited availability.

On June 23, 1984, 6 of 37 key safety components—the control rod drives—malfunctioned during an automatic shutdown and failed to completely insert the control rods. Shutdown was accomplished about 20 minutes later by using the remaining control rods and manually inserting the six malfunctioning rods. NRC considered this failure a very serious safety matter and subsequently ordered PSC to shut the plant down immediately until the problem could be resolved. Subsequent investigations into the reasons for this malfunction revealed that lack of preventive maintenance had caused the control rod drive malfunction. According to a PSC official, it is almost certain the control rod drives had become corroded by water that had entered the cooling system on several occasions. The water originated from the water-cooled bearing system on the cooling circulators. PSC subsequently refurbished the control rod drives and received permission to restart the reactor in July 1985, more than a year after the incident.

NRC increased its attention to Fort St. Vrain in the summer of 1984, after the control rod drive failure. This failure prompted an in-depth NRC assessment from July through August 1984, not only of the causes of the equipment malfunction but also of PSC's general conduct of operations at the plant. This special assessment confirmed deficiencies previously identified in NRC reports. Some of the problems reported by the NRC included

- numerous errors attributable to operators and technicians who failed to follow established procedures,
- poor housekeeping at the plant,
- an ineffective preventive maintenance program,
- problems in the spare parts management system, and
- improperly maintained equipment returned to service.

NRC concluded that PSC's operating philosophy subscribed to less formality and less rigid control of operations than was common at other commercial nuclear power plants. It then requested PSC to develop a program to identify the underlying causes of the deficiencies and corrective measures. The scope and schedule for this program was then to be submitted to NRC prior to restarting the plant.

PSC subsequently contracted with the Nuclear United Services Operating Services Corporation to evaluate the management of the nuclear-related activities within PSC. This audit confirmed the basic concerns expressed by NRC in its special assessment and presented recommendations to correct the underlying causes.

Various Causes Identified for Management Weaknesses

The special NRC and NUS assessment reports and subsequent NRC inspections have commented on the underlying reasons for the management deficiencies and operational problems at Fort St. Vrain. NUS found that the majority of Fort St. Vrain's problems stem from the PSC management's belief that because the gas reactor concept is unique to the U.S. commercial nuclear industry, many of the regulations for the operation and maintenance of commercial light-water nuclear power plants should not apply to Fort St. Vrain.

When Fort St. Vrain began commercial operations in 1979, NRC's regulatory emphasis was almost entirely on safety concerns associated with light-water reactors. This fact, coupled with NRC's confidence in the stable design features and low risk to public health and safety inherent in the design of Fort St. Vrain, resulted in NRC staff giving limited regulatory attention to the plant, according to NRC staff.

In addition, according to NUS, NRC's regulatory emphasis on light-water reactors and its limited budget prevented it from developing extensive expertise in gas reactor technology. NUS believes that this lack of expertise added to the PSC staff's frustration because it continually needed to educate NRC staff in gas reactor technology. PSC officials concurred, stating that this frustration is basic to the problems they have experienced in dealings with NRC.

The NUS report also stated that Fort St. Vrain's unique design resulted in an "inappropriate sense of isolation . . . from the rest of the nuclear industry." Consequently, according to NUS, PSC did not adopt well-established programs and procedures developed by the rest of the nuclear industry to deal with problems similar to those faced at Fort St. Vrain.

NRC officials said that PSC had not actively participated in nuclear industry efforts to improve performance because most of the programs were geared toward light-water reactors. In addition, NRC attributed the management problems at Fort St. Vrain to

- poor inter- and intradepartmental communications,
- weaknesses in corporate oversight that went undetected and uncorrected for a prolonged period, and
- a deteriorating quality assurance department that was not corrected by management.

PSC's Perception of the Causes for Problems at Fort St. Vrain

PSC officials essentially agreed with NRC's assessment that a lack of coordination among the divisions and an ineffectively managed organization were the root causes of its management and performance problems. Some PSC officials believe, however, that Fort St. Vrain's problems are partially attributable to difficulties in dealing with NRC. Specifically, they cited problems in dealing with NRC regulations and NRC's past unresponsiveness to PSC's licensing submittals. These officials stated that NRC regulations are interpreted differently by different NRC staff, thus creating confusion and uncertainty about how to correctly implement certain regulations. In addition, it was difficult to continually have to interpret regulations designed for light-water reactors for their applicability to Fort St. Vrain, a gas-cooled reactor.

These problems and others are exemplified, according to PSC officials, by a problem the plant had with the qualification of the plant's electrical equipment. Beginning in the late 1970s, PSC spent millions of dollars to bring Fort St. Vrain's electrical equipment into what it believed to be compliance with NRC regulations. The current NRC project manager for Fort St. Vrain agreed that NRC had accepted the plant's equipment qualification efforts as sufficient at that time. However, during the time PSC was upgrading this equipment, NRC regulations had become stricter, and PSC efforts to bring its electrical equipment into compliance became unacceptable. PSC received no formal communication from NRC about the inadequacy of its efforts until 1985.

In 1985, NRC officially notified PSC that Fort St. Vrain was not in compliance with NRC regulations. PSC shut down the reactor and has spent approximately an additional $40 million to requalify its equipment to bring it into compliance with updated NRC regulations. NRC officials agreed that they had not communicated effectively to PSC about what specific qualification problems actually needed to be met and had failed to quickly provide information to PSC concerning those areas in which the reactor was inadequate.

Poor staff morale and frustration have also contributed to the problems at Fort St. Vrain, according to PSC officials. In October 1985, PSC hired an outside consultant to evaluate the effectiveness of the Performance Enhancement Program. The consultant found that the intermittent operating history of the plant had resulted in "an environment where employees . . . have become increasingly defensive and frustrated." The report recommended that PSC formalize a human resources plan that would address employees' frustration and defensive attitudes. PSC subsequently developed a project to address its problems in human resource management.

PSC Has Developed a Program to Rectify Deficiencies

In response to NRC's and NUS's management and operational concerns, PSC developed a broad-based Performance Enhancement Program in April 1985. The objective of this program was to assign and implement activities to improve the overall quality, management, and operation of PSC's nuclear organization in a controlled, timely manner. The program originally consisted of six major projects: organizational concerns, planning and scheduling, preventive maintenance, policies and procedures, training, and conduct of operations. Two additional projects have since been incorporated into the program to improve PSC's human resource management and quality assurance programs.

The actions PSC has taken or has underway include:

- consolidating sole responsibility for all nuclear-related operations under a new senior nuclear executive who has an extensive background in nuclear engineering and management of nuclear facilities;
- creating a new division with the sole purpose of handling licensing interactions with NRC; its manager is the central point of contact on licensing issues;
- increasing the quality assurance division staff and appointing a new division manager;
- creating 89 new positions, mostly technical, including licensed personnel with light-water reactor experience, design engineers, training staff, and quality assurance staff;
- revising preventive maintenance procedures, including the addition of post-maintenance testing;
- hiring an outside contractor to assist PSC with human resource management issues (e.g., attitude and motivational problems, and executive management communication).

Although PSC's original schedule called for essential program completion by the end of 1986, some slippage has occurred, according to PSC officials, because licensing activity has increased and the overall scope of the Performance Enhancement Program has broadened. While hoping to complete most projects in 1987, PSC officials believe the program is open-ended and will be a continuing effort to improve management and operations at the plant. PSC provides semiannual briefings to NRC on the program's progress.

Performance Enhancement Program Has Contributed to Significant Improvements in Fort St. Vrain Operations

Although the May 1986 SALP report was critical and indicated weaknesses in the management and operations of Fort St. Vrain, NRC also recognized improvement in a number of areas as a result of PSC's Performance Enhancement Program. In particular, NRC noted improvements in the areas of plant operations, licensing, and maintenance, as well as increasing management involvement in resolving problems. NRC concluded, however, that "continued high level attention would be necessary to ensure lasting improvement."

On October 17, 1986, PSC and NRC officials responsible for the plant briefed the NRC commissioners on the status and readiness of the plant to resume power operations in the future. At that time, the plant was shut down in order to complete equipment modifications that would bring it into compliance with NRC regulations. NRC staff from headquarters and the regional office expressed their confidence that the substantial improvements being made in the management, technical, and plant improvement areas did support restart. Recent special inspections conducted at Fort St. Vrain had confirmed that the improvement trends noted in the last SALP report were continuing and that major strides had been made, particularly in the licensing and security areas.

Despite NRC staff support and confidence in the improvements being made at Fort St. Vrain, the NRC commissioners were not confident that all the improvements currently planned by PSC would be completed. Prior to granting restart approval, they wanted to see concrete evidence that corrective actions were taking place. Therefore, they requested the NRC staff to closely monitor the progress at the plant until the pending modifications were completed.

NRC's monitoring continued to show that improvements were being made at Fort St. Vrain. Specifically, an inspection of quality assurance activities noted improvements in the division since the new senior nuclear executive came on board. In November 1986, NRC conducted another inspection to assess the effectiveness of the management initiatives to improve the performance of the Fort St. Vrain staff. The inspectors noted that "management initiatives to improve and to change the performance at Fort St. Vrain were having an effect throughout the organization. The commitment to quality and safety was universal." NRC staff further concluded that "the attitude of the [PSC] personnel and the effectiveness of management have shown considerable improvement and should not impose any constraint on plant restart."

On February 26, 1987, PSC and NRC again briefed the NRC commissioners on the status of the plant and requested permission to start up the plant. As a result of the briefing, the plant was restarted on April 17, 1987.

On July 1, 1987, NRC released its latest overall assessment of performance covering the period from May 1986 through April 1987. This assessment showed significant improvement in five of six functional areas previously rated deficient. The functional areas of maintenance, security, outages, quality assurance, and licensing activities were all elevated from minimally satisfactory to satisfactory. The one remaining area, emergency preparedness, which was rated minimally satisfactory in 1986 and again in the most recent rating, was noted by NRC as showing a positive improvement trend.

Conclusions

According to NRC's operational safety assessments, Fort St. Vrain's performance has been declining since 1983. The annual assessment for the period ending May 1986 found performance to be "minimally satisfactory" in 6 of 11 functional areas. NRC officials believe that management problems have led to this declining performance. In response, PSC has developed and is implementing a program to rectify its management problems. As a result of PSC's efforts, the most recent NRC assessment showed significant improvements, with only one functional area rated as minimally satisfactory.

Emergency Preparedness Program for Fort St. Vrain Is Minimally Satisfactory

PSC and NRC are responsible for emergency preparedness within the Fort St. Vrain reactor facility's boundaries. In addition, PSC, NRC, Colorado, and Weld County are required to develop emergency preparedness plans and procedures to protect the public and the environment in the event that radioactive materials are released beyond the facility's boundaries. FEMA provides guidance and assistance to the state in preparing for and responding to a nuclear emergency at Fort St. Vrain.

According to NRC's most recent assessment, PSC's capabilities in the emergency preparedness area are minimally satisfactory for protecting the public health and safety; and, improvements are needed in the management commitment to emergency preparedness and in the training of employees responsible for emergency preparedness. According to PSC officials, the past lack of management emphasis on and dedication toward emergency preparedness stemmed from management's belief that the Fort St. Vrain plant was inherently safe and thus should not be subject to the same standards as other nuclear plants. In addition, recent management initiatives at Fort St. Vrain, including the development of an action plan to improve PSC's emergency preparedness program, indicate that PSC management has dedicated itself to improving its emergency preparedness program.

The federal, state, and local agencies responsible for emergency preparedness are ready for an emergency that results in releases of radiation beyond the facility's boundaries.

Emergency Preparedness Responsibilities at Fort St. Vrain

Adequate on-site emergency planning and response are necessary to protect workers, the public, and the environment from the release of radioactive materials in the event of an accident at a nuclear power plant. The radiological emergency response plan for Fort St. Vrain was prepared by PSC and was approved by NRC in 1980. The plan includes the following five main areas:

- description of the organization that manages emergencies,
- definition and assignment of responsibilities for emergency response actions,
- classification of emergencies according to the severity of consequences,[1]
- courses of action and protective measures to mitigate the consequences of an accident and protect workers and the public, and

[1] Emergencies are classified into four categories of increasing severity that require a higher level of response: notification of unusual event, alert, site area emergency, and general emergency.

- a plan and organizational structure to restore the plant to normal operations after the emergency has been resolved.

In addition, PSC has prepared detailed procedures to implement the plan and provide specific guidance during an emergency.

PSC Responses During a Radiological Emergency

PSC's emergency response organizations operate from three centers within the facility's boundaries and three centers beyond the facility's boundaries. The control room, the technical support center, and the personnel control center are located within the facility's boundaries, while the forward command post, the executive command post, and the state emergency operations center are located beyond the Fort St. Vrain boundaries. When an accident occurs, personnel are notified of the emergency and go to the center to which they are assigned. When the centers are staffed and operational, officials at the center notify appropriate officials (detailed in the emergency preparedness plan) and begin their assigned duties. Personnel at these centers have emergency preparedness responsibilities that can change depending upon (1) the order in which the centers become fully operational and (2) how severe the emergency is.

In the control room, the shift supervisor is responsible for bringing an accident under control. The shift supervisor is also initially responsible for directing and coordinating PSC's emergency response operations, until relieved by the control room director or the technical support center director.

When fully staffed and operational, the technical support center is responsible for collecting and analyzing the information necessary for assessing plant operations, monitoring and assessing the consequences of radiological releases, and providing technical support to the control room. If the technical support center becomes fully operational before the forward command post, the technical support center director is also responsible for all emergency responses until relieved by the Corporate Emergency Director at the forward command post, located at Fort Lupton, Colorado (about 12 miles from Fort St. Vrain).

The primary and secondary locations for the personnel control center are within the facility's boundaries at the training center and the nuclear licensing operations complex, respectively. Staff at the personnel control center are responsible for (1) maintaining accountability of personnel; (2) search and rescue efforts such as first aid, medical transportation, and personnel decontamination; (3) emergency maintenance such as mechanical and electrical repair and damage control; and (4) extinguishing fires or eliminating dangerous gases. In addition, first aid and decontamination equipment, protective clothing, portable lighting, and protective breathing apparatus are stored at the personnel control center.

When the forward command post is fully staffed and operational, the PSC Vice President for Nuclear Operations becomes the Corporate Emergency Director—primary decisionmaker for PSC throughout the emergency. The Vice President directs all PSC responses throughout the emergency, such as classifying the emergency and making protective action recommendations, notifying and coordinating with state and local agencies, and mitigating the results of the emergency.

The executive command post is located in the PSC headquarters building in downtown Denver (about 35 miles from Fort St. Vrain). The post is headed by PSC's Chairman and Chief Executive Officer and is staffed by senior PSC management. It provides back-up support to the other emergency centers but has no decision-making authority for emergency preparedness at Fort St. Vrain.

The state emergency operations center is located at Golden, Colorado (about 40 miles from Fort St. Vrain). This center is the primary decision-making site for Colorado in the event of a nuclear accident. This site is also the primary media relations and government affairs center for PSC. PSC personnel assigned to this center answer questions from the public and the media and prepare press releases on any emergency at Fort St. Vrain. These personnel include the Assistant Vice President for Government Affairs, the Manager of Corporate Communications, and the Media Relations Director.

NRC Maintains Oversight of Emergency Preparedness Program

Oversight of PSC's emergency preparedness program for protecting the radiological health and safety of the public, environment, and workers is primarily the responsibility of NRC Region IV's Investigation and Enforcement Branch. It maintains oversight of emergency preparedness activities primarily by conducting unannounced emergency preparedness inspections at Fort St. Vrain and observing and participating in the facility's annual emergency preparedness exercises. In addition, the region's Reactor Projects Branch has two inspectors permanently stationed at Fort St. Vrain who provide additional oversight of emergency preparedness activities on an as-needed basis.

NRC notifies licensees if they are not in compliance with its regulations or policies through notices of violations and through such administrative actions as notices of deficiencies. Violations occur when a licensee fails to meet a federal regulatory requirement and can result in enforcement actions such as civil penalties and/or fines. Administrative actions, such as notices of deficiencies, are less severe than violations and are not part of NRC's regulatory or enforcement policy. NRC normally cites deficiencies during a licensee's annual exercise to point out areas where improvements are needed.[2]

NRC's Region IV subjectively ranks its licensees' emergency preparedness programs through its SALP reports. The rankings are based upon the results of exercises, inspections, and the region's personal knowledge of a facility. These reports pinpoint areas for concentration in future emergency preparedness inspections and serve notice to a licensee that improvements should be made (see ch. 3).

Problems in Emergency Preparedness Program at Fort St. Vrain Attributed to Management Deficiencies

NRC has identified several weaknesses in the emergency preparedness program at Fort St. Vrain. For the past several years, the emergency preparedness program at the plant has been only minimally satisfactory in ensuring the health and safety of the workers, the public, and the environment. From June 1985 to August 1986, NRC conducted four emergency preparedness inspections at Fort St. Vrain—two unannounced inspections and two inspections of the licensee's annual site-wide exercise—and an overall evaluation of the licensee's management performance.[3] These inspections and evaluations indicated two problems. Emergency preparedness employees were not adequately trained in

[2]NRC officials said they developed this mechanism to encourage candor by licensees during the annual exercises. By not issuing a formal violation during an observed exercise, NRC officials expect the licensee to be open and candid about potential problems in the emergency preparedness area.

[3]Licensees are required to test their emergency preparedness capabilities annually through exercises; and state and local agencies are required to conduct exercises biennially. These exercises are conducted under simulated accident conditions during which the licensees' employees, NRC, FEMA, other federal, and state and local officials act as role players. The discussions in this chapter on the responses during the exercises are role-playing responses to simulated accidents and simulated accident conditions. None of these accidents actually took place.

their roles and responsibilities during an emergency, and PSC management was not dedicated to correcting the identified deficiencies. NRC officials said that these weaknesses, either individually or taken as a whole, do not jeopardize the health and safety of the public, workers, or the environment. NRC's observations indicated that PSC management's commitment to a quality program was not demonstrated.

In response, PSC officials stated that improvements have been made in the emergency preparedness area in the past year and will continue; however, many of these improvements cannot be measured at this time. They will be evaluated during the next annual exercise, scheduled for late 1987. Beyond specific improvements, the major improvement in emergency preparedness at Fort St. Vrain, according to NRC officials, is management's commitment to improvement and its commitment to work with NRC in improving the emergency preparedness program.

Corporate Emergency Director Inadequately Trained

The Corporate Emergency Director is responsible for the licensee's overall responses to an emergency. However, during the last two annual exercises, NRC's inspection reports noted that he did not adequately assume control for the overall direction of PSC's responses to the emergency. Throughout the exercises the Corporate Emergency Director was indecisive in making recommendations either to state agencies or to his own staff. As a result of his indecisiveness, other employees took decisive actions as events required.

In the 1986 exercise, the Corporate Emergency Director did not activate the forward command post within the required 90 minutes, even though it was fully operational and staffed, nor did he notify the technical support center that responsibility should be transferred to the forward command post. Consequently, the technical support center director declared a general emergency and provided protective response recommendations without consulting the Corporate Emergency Director. Throughout the exercise, according to NRC inspectors, another PSC employee briefed the staff and appeared to be the decisionmaker for PSC.

During the 1985 exercise, the Corporate Emergency Director did not make a decision on the severity of the accident, even though deteriorating plant conditions required a decision. The control room director assumed the authority of the Corporate Emergency Director and declared a general emergency.

Because the Corporate Emergency Director failed to demonstrate to NRC inspectors that he had received the necessary training to adequately manage the overall direction of the licensee's operations during an emergency, NRC officials interviewed the Corporate Emergency Director to determine his qualifications as overall director of the licensee's emergency responses. The interview indicated that the Corporate Emergency Director had not been adequately trained in his role and responsibilities during an emergency. As a result, NRC officials attempted to review his training records but found no record of his initial training or retraining, as required by the licensee's training manual. PSC officials said that the Corporate Emergency Director had not received the training because of an oversight by PSC's training department. PSC officials stated that after discovery of the oversight the Corporate Emergency Director received the necessary training, which should provide him with stronger leadership skills.

In commenting on a draft of this report, PSC officials informed us that a new Corporate Emergency Director has been hired who is fully qualified to perform the required duties.

Other Emergency Preparedness Employees Inadequately Trained

NRC's inspection reports also noted that PSC had not established training requirements or provided training to other emergency preparedness employees at Fort St. Vrain. In the last two annual exercises, the team responsible for caring for and decontaminating accident victims failed to remove the victims' contaminated protective clothing when the victims were delivered to the decontamination facility. In one of the exercises, the same team failed to provide first aid to a victim who had a broken leg. The team members did not put a splint on the victim's leg even though splints were available in their trauma kit.

In addition, during interviews and observations of seven health physics technicians, NRC inspectors found that these employees could not determine the habitability of the control room or the iodine content in a simulated radioactive cloud.[4] In reaching this conclusion, NRC inspectors found, for example, that

[4]A major concern during a radiological emergency at a nuclear facility is that a radioactive material, Iodine-131, could be released into the atmosphere and then absorbed by the thyroid glands of individuals exposed to the radioactive cloud.

- six of the seven health physics technicians were not knowledgeable about existing procedures to determine the habitability conditions in the control room;
- three did not realize that one of their major responsibilities was the protection of emergency workers;
- five said they did not know what actions to take or how to prioritize them;
- six could not explain the technique necessary to determine if a radioactive plume contained iodine, nor could they properly interpret the results; and
- several technicians acknowledged that they needed more training for their emergency response duties.

PSC's response to the NRC inspection reports acknowledged that further specialized training should be provided to these health physics technicians. Further, PSC said it had prepared a new lesson plan for training technicians in the areas of habitability determinations, plume tracking, airborne sampling, and good health physics practices during an emergency.

Improvements in Emergency Preparedness Require Management Commitment

NRC has found that Fort St. Vrain's management has not been strongly committed to improving its emergency preparedness program. Rather than simply meeting minimal NRC safety standards on regulatory requirements, NRC maintains that management should be constantly concerned with (1) seeking ways to maintain and improve its existing program, (2) taking corrective actions on violations identified by NRC, and (3) responding to improvements suggested by NRC. According to NRC and PSC officials, this lack of a strong management commitment can be attributed to the management's assumption that because Fort St. Vrain employs a unique and inherently safer reactor design, emergency preparedness is not as crucial as at other commercial reactors.

Management Has Not Complied With Recommendations Made by NRC

Fort St. Vrain management has not complied with NRC recommendations to correct emergency preparedness deficiencies identified by NRC during inspections and annual exercises. In the 1985 annual exercise at Fort St. Vrain, NRC inspectors identified 11 deficiencies that required corrective action by the licensee. Management agreed to correct these deficiencies; however, during the 1986 annual exercise, NRC identified 5 deficiencies from the prior exercise that had not been corrected. NRC issued Fort St. Vrain a notice of violation for failure to correct these past deficiencies.

Although NRC officials said they do not usually issue notices of violation during exercises because they want to promote candor in the role players, they took this step because of PSC management's lack of commitment to taking corrective actions in the emergency preparedness area. The same deficiencies cited in 1985 occurred in 1986 because management had not provided sufficient training to the personnel responsible for emergency preparedness. In addition, during an unannounced inspection 3 months prior to the 1986 exercise, NRC inspectors also noted that the PSC official responsible for overall direction of the emergency preparedness program had still not received adequate training in his role as Corporate Emergency Director. NRC officials believed they had given PSC management sufficient notice to correct its emergency preparedness program and that failure to take corrective action indicated that management lacked the commitment to do so.

Further, in the 1986 SALP, PSC's emergency preparedness program received a minimally satisfactory rating. NRC was especially concerned with management's lack of commitment to this area; it concluded that performance had declined significantly from the previous evaluation. In particular, NRC was concerned with PSC management's lack of responsiveness to NRC-identified deficiencies and violations. In addition, the number and type of deficiencies and violations indicated that emergency preparedness personnel were not receiving adequate training. In some cases, employees were not receiving even the minimum training required.

Management Lacked Motivation to Improve Its Emergency Preparedness Program Because of the Inherent Safety of Its Reactor

PSC management considers the Fort St. Vrain reactor to be inherently safer than light-water reactors because it permits more time to respond to an accident before potential releases of radioactive elements into the environment can occur.

According to NRC officials, the management at Fort St. Vrain has developed a "Titanic Syndrome." That is, management is convinced that an accident involving releases of radiation into the environment is much less likely at Fort St. Vrain than at other commercial reactors. Therefore, PSC management believes it should not be subject to the same emergency preparedness standards as operators of light-water reactors. PSC management agreed that its past actions were based on the belief the Fort St. Vrain plant was inherently safe.

NRC officials, however, believe that management at Fort St. Vrain must prove to NRC that its reactor should not be subject to the same standards. According to NRC officials, PSC management has not done that, and until it does, Fort St. Vrain should conform to the same standards as other commercial reactors in the United States.

Because it believes that NRC should relax its standards regarding the Fort St. Vrain reactor, PSC management has been reluctant to comply with NRC recommendations for improving its emergency preparedness program. According to NRC officials, PSC management has constantly required NRC to prove to management that its actions actually violated regulatory requirements. If there was no actual violation, then management felt little or no action was necessary to improve the emergency preparedness program. For example, NRC has recently developed a collocation plan that directly involves NRC officials in annual exercises as participants rather than as merely observers who critique the exercises. Under this plan, as many as 25 NRC officials would work directly with a comparable number of PSC employees. NRC sent an emergency preparedness team to Fort St. Vrain to explain to management and employees how the plan would work and what involvement would be needed by PSC. Only two officials from Fort St. Vrain attended the meeting. An NRC official said that at other commercial reactors in the region all of the employees who would be involved in the plan attended the meeting.

Improvements in Emergency Preparedness Identified Since Last Exercise

Since the last annual exercise in August 1986, NRC officials told us that they have seen a definite improvement in PSC management's commitment to improving its emergency preparedness program. In July 1986, PSC hired a new Vice President for Nuclear Operations who said he is convinced that the program must be improved. According to NRC officials, the Vice President for Nuclear Operations has told them that PSC management will no longer be satisfied with just meeting the minimum regulatory requirements and safety standards. Futhermore, the Vice President has established an emergency preparedness action plan for improving the program, comprising nine projects that began in late 1986 and are due for completion in mid-1988.

The action plan identifies several areas in emergency preparedness where improvements are needed. In particular, the plan calls for upgrading the emergency preparedness training program at Fort St. Vrain by revising lesson plans, developing comprehensive drill schedules, developing computerized training records to identify when training is needed, and implementing new training schedules. PSC has hired additional emergency preparedness training staff and has started design of a computerized tracking system to include NRC-identified deficiencies and ideas for improvement, as well as PSC's own ideas for improvement. The tracking system will better permit management to identify what corrective actions have been or need to be taken to comply with NRC's and its own recommendations to improve its emergency preparedness program. To launch this plan, emergency preparedness personnel have visited more highly rated commercial facilities to determine what they have done to improve their programs. They have also hired a nationally recognized consulting firm to help them rewrite their emergency preparedness plan. This plan should be rewritten by early 1988.

In November 1986, NRC officials interviewed 101 PSC employees to determine if (1) employees were aware of PSC's initiatives to improve quality and overall performance and (2) management's commitment had enhanced the performance and attitudes of these employees. These interviews indicated that the employees were aware of PSC's initiatives in the area of quality and performance, and that employees' morale and commitment had improved because of management's initiatives.

NRC officials told us they are cautiously optimistic about PSC's efforts to improve its emergency preparedness program. They have seen an increased responsiveness to the recommendations made in its inspection reports, but they still consider the program to be minimally satisfactory. They will not be able to fully evaluate the effect of PSC's recent efforts until the next annual exercise, scheduled for late 1987.

Emergency Preparedness Responsibilities Beyond the Fort St. Vrain Boundaries

Adequate emergency planning and response are necessary to protect public safety in the event that radioactive materials are released into the environment as the result of an accident at nuclear plants. Emergency plans for areas beyond a nuclear facility's boundaries are prepared by the affected state and local governments with the assistance of FEMA. These plans are then submitted to FEMA for its review. FEMA provides NRC with its determination as to the adequacy of the plans, and NRC uses this information in making its decision on whether to license a nuclear power plant.

According to FEMA officials, the emergency preparedness exercise held in 1985 indicated that state and local agencies responsible for emergency preparedness are prepared to protect the public in the event of a nuclear accident at Fort St. Vrain. State officials attribute their preparedness to the fact that they test and drill for other nuclear emergencies besides Fort St. Vrain, such as emergencies at a Department of Energy nuclear facility and emergencies involving nuclear transportation accidents and nuclear attacks.

State Emergency Preparedness Plan for Fort St. Vrain Approved by NRC

The purposes of an emergency response plan are to prescribe actions, define responsibilities, and provide for coordination of activities to protect the public and the environment in the event of an accident involving the release of radioactive materials. Some of the more important aspects of emergency response actions are discovering and tracking the release; notifying responsible officials and the public of the release and its possible consequences; mitigating the consequences of the release of radioactive materials; evacuating the public from the area (if necessary); and reentering the area after the effects of the emergency have been mitigated.

Colorado's emergency response plan for Fort St. Vrain is one part of its overall three-part state plan to respond to nuclear emergencies. The other two parts of the plan deal with the state's responses to a nuclear accident at the Department of Energy's Rocky Flats nuclear weapons facility or to a nuclear emergency such as a transportation accident or nuclear attack. Colorado's emergency response plan was in place prior to the opening of Fort St. Vrain. The state added to this plan a part dealing with emergencies at Fort St. Vrain and submitted it to FEMA for its review. FEMA completed its review in 1980; NRC approved the plan in 1980.

Weld County is the only county in Colorado whose boundaries encompass the 5-mile plume exposure pathway emergency planning zone around Fort St. Vrain.[5] Weld County uses the state plan as its emergency preparedness plan; however, the county does have its own operating procedures to implement its responsibilities under the state plan.

Agencies' Responsibilities for Emergency Preparedness

Responsibility for emergency preparedness around Fort St. Vrain is divided between Colorado and Weld County. Two state agencies have primary emergency preparedness responsibilities—the Division of Disaster Emergency Services (DODES) within the Colorado Department of

[5] An emergency planning zone is an area around a nuclear facility that is used to facilitate off-site emergency planning and to develop a base of responses. The plume exposure pathway is that area where the population could experience direct exposure to radioactive materials.

Safety, and the Division of Radiation Control within the Colorado Department of Health. Locally, the Weld County Civil Defense Agency, Department of Emergency Services, is responsible for emergency preparedness around Fort St. Vrain. This agency is assisted primarily by the Weld County Sheriff's Office.

When an accident occurs at Fort St. Vrain, PSC notifies the Weld County Communications Center of the event. After verifying the authenticity of the event by return call to PSC, these officials notify the Weld County Sheriff's Department, the state DODES, the Department of Health, the State Patrol, and other county officials listed in the emergency preparedness plan. DODES then notifies the governor and other state officials listed in the plan. The Department of Health also completes its notification of all other health agencies, including the Division of Radiological Health.

The Colorado Department of Health is responsible for assessing the severity of the accident—in terms of potential danger to the off-site population—and for recommending actions to protect the public from the dangers of radiation exposure. It provides this information to the governor and to DODES, which exercises overall coordination and control of emergency response activities beyond the boundaries of Fort St. Vrain.

Under the direction of DODES, the Weld County Sheriff's Department is responsible for evacuating the public during an emergency at Fort St. Vrain. The Sheriff's Department is assisted by other state agencies.

Testing of Emergency Preparedness Capabilities Beyond the Fort St. Vrain Boundaries

To determine if state and local agencies can adequately respond to a radiological emergency, the agencies with emergency preparedness responsibilities and the utility that operates the reactor test their capabilities to respond to an emergency at least every 2 years. These agencies conduct biennial exercises with the utility that operates the commercial nuclear reactor.

Colorado conducted exercises with PSC in June 1983 and June 1985. Both exercises were observed by FEMA and NRC officials. (Another exercise is scheduled for late 1987.) In both exercises the state demonstrated to FEMA and NRC that it can adequately safeguard the health and safety of the public in the event of a release of radioactive materials from the Fort St. Vrain nuclear reactor. State and local agencies actions conformed with the procedures in their emergency preparedness plan.

The exercises did identify areas that should be improved prior to the next exercise. The area most needing improvement was the field-monitoring activities of the Colorado Department of Health. Because it lacked standard operating procedures for issuing equipment, the field-monitoring team did not receive all the equipment it needed to adequately assess a radioactive plume released at Fort St. Vrain. For example, the team did not receive environmental sampling equipment, air samples, or protective clothing. In addition, all but one of the radiation instruments that were issued had expired calibration dates. As a result, the accuracy of the readings from this equipment would be questionable.

The system used to alert and notify the public within the 5-mile emergency planning zone around Fort St. Vrain also needed improvement. At the time of the 1985 exercise, the alert and notification system consisted of warning messages broadcast over the Emergency Broadcast System and the tone alert radios that PSC had installed in the residences within the 5-mile planning zone. This exercise and one conducted in 1984 indicated that less than 70 percent of the population were receiving the alert. Many people had turned off the radios and did not hear the tone alert or the broadcast. As a result, FEMA had recommended that a siren system be built in Platteville (the only town within the 5-mile planning zone).

A FEMA official told us that both of these areas have been improved since the June 1985 exercise. The Colorado Department of Health has written standard operating procedures for the issuance of equipment to the field-monitoring team; the radiation instruments have been recalibrated; and procedures have been instituted to ensure that equipment is calibrated on a regular basis. In addition, PSC installed a siren in Platteville in 1986. FEMA tested the siren and surveyed the population in August 1986. That survey, the most recent, indicated that about 80 percent of the population within the 5-mile planning zone heard the siren.

The state agencies with emergency preparedness responsibilities at Fort St. Vrain are also responsible for emergency preparedness at the Department of Energy's Rocky Flats nuclear facility and accidents involving the transportation of nuclear materials. Similar planning and implementing procedures are used whether the accident occurs at Fort St. Vrain, Rocky Flats, or on a state highway. State emergency preparedness officials said that they test their plans and procedures for responding to each of these emergencies and that this redundancy in testing of their emergency preparedness capabilities assists them in their ability to respond to an emergency at Fort St. Vrain.

Major Contributors to This Report

Resources, Community, and Ecomonic Development Division, Washington, D.C.

Keith O. Fultz, Associate Director, (202) 275-1441
Thomas E. Melloy, Group Director

Denver Regional Office

Debra J. Carr, Evaluator

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