NUREG-1175

NRC Safety Research in Support of Regulation

Selected Highlights

U.S. Nuclear Regulatory Commission

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Selected Highlights

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TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF FIGURES

LIST OF TABLES

ACKNOWLEDGMENTS

The staff wish to express their appreciation to all those who assisted in the preparation of this report. Specifically, we wish to thank Marc Goldsmith and Vera Geba of Energy Research Group, Waltham, Massachusetts for their thoughtful and diligent efforts in preparing an initial draft of this report. We also wish to thank Marie Slater (NRC) for her outstanding contributions in gathering, selecting, reviewing, and checking inputs and products and in performing a multitude of other coordination responsibilities for this document. In addition, other Offices of the NRC and the National Laboratories were most helpful in providing materials for and comments on the document.

EXECUTIVE SUMMARY

Over the years, considerable Federal funds have been spent on nuclear safety research. Yet, because the research results have often appeared only as part of a regulation or licensing decision, it can be difficult to reconstruct just what research has contributed to the regulatory effort. This report has been developed to respond to Congressional inquiries about the nature of the research and the ways in which it has contributed to certain key decisions in recent years. The impacts described are representative of the continuing impacts of research expected in future years.

The Office of Nuclear Regulatory Research (Office of Research) of the Nuclear Regulatory Commission (NRC) has, since its inception, provided support to other offices of the NRC. As with any state-of-the-art technology, there have been and continue to be scientific and engineering aspects of nuclear power generation that are not well understood. The task of the Office of Nuclear Regulatory Research is to develop recommendations for, engage in, or contract for research necessary for the performance of NRC's licensing and related regulatory functions.

The research program helps to provide the technical basis for Commission licensing, rulemaking, and inspection and enforcement and assist in the development of regulatory priorities. As new research findings have been developed, NRC requirements have been strengthened or modified to assure safety.

Specific technical objectives in resolving regulatory needs are defined in conjunction with the regulatory staff. Research efforts are contracted for by the Office of Research with National Laboratories, universities, and private contractors. The Office of Research also engages in cooperative research efforts with other U.S. government agencies, U.S. nuclear industry, and foreign organizations. Office of Research staff maintain continual technical oversight to ensure that the research contributes to achieving regulatory objectives.

Safety in nuclear power plants and in nuclear waste management has been greatly improved by enhanced analytical tools, inspection methods, and understanding of safety problems resulting from the NRC's research programs. The capability of the licensing, inspection, and enforcement processes have been significantly enhanced as well. In this respect, research achievements often are not identified separately, but rather as an integral element of major agency decisions. Some research findings have identified previously unanticipated safety problems and have resulted in new or more stringent regulations. Other research findings have shown that existing systems are safer than previously thought and have allowed the reduction or elimination of unnecessarily conservative or even misdirected regulation, with consequent improved plant safety and efficiency, reduced costs to the ratepayer, and more reliable power production.

Some of the major accomplishments of the research program in recent years include:

- The identification of operating limits (temperature and neutron exposure) for pressure vessels of various compositions to reduce the hazard of Pressurized Thermal Shock. This research led to a new NRC rule that improved reactor safety by requiring power plant operator to make a showing that potentially vulnerable conditions will not occur during plant lifetime or take certain actions such as neutron flux reductions.

- The improvement of knowledge on how piping fails, which, taken together with improved nondestructive examination methods, established that current requirements for piping restraints should be modified to improve safety. The research led to a rule change that permitted elimination of requirements for some pipe whip restraints and water jet impingement shields provided certain criteria were met. This has the potential to improve safety by (1) reducing the chances of installation errors that can introduce additional piping stresses, (2) making the piping system more accessible for in-service inspection and (3) reducing radiation exposures for workers.

- The development of an improved understanding of emergency core cooling system performance, specifically, the temperature that fuel can reach during a loss-of-coolant accident. This research led to a revision, now in process, of requirements for emergency core cooling systems that removes unnecessary limitations on operating conditions. The revision will allow improved operating efficiency with no loss of safety.

- The development of better data on sources of hydrogen production in severe accidents, demonstration of the behavior of hydrogen and hydrogen fires in containment, and identification of potentially severe hazards in certain types of containment. This research led to the adoption of two revisions to 10 CFR 50.44 requiring additional hydrogen control equipment and procedures in both BWR Mark IIIs and PWR ice condenser containments. Specifically, the revisions required installation of hydrogen gas control systems to limit hydrogen buildup in containment from an accident and assure equipment survival. Experience at TMI-2 showed that earlier regulations covered severe accident cases incorrectly.

- Better understanding of the behavior of reactor piping systems under seismic conditions, allowing a design better balanced between routinely occurring normal operating loads and infrequently occurring seismic loads. This research work has led to more flexible piping design by permitting the removal of numerous snubbers (which historically have a proven failure rate of over 10%) at approximately thirty nuclear reactors. In addition, the NRC research provided the bases for ASME Code Case N-411 permitting the use of higher damping values. This modification permits more flexible and thus more reliable piping with fewer seismic snubbers. The resulting change produces improved public safety, less occupational radiation exposure, and reduced maintenance costs.

- The development of the technical criteria of the high-level waste disposal regulation 10 CFR 60. These technical criteria set requirements for siting, design, and waste packaging as well as for overall system performance.

Future NRC research will be necessary to continue to improve safety and understanding and to develop better analytic methods to ensure effective regulation. New research will be required to quantify unknowns, ensure that newly discovered phenomena will not reduce safety, and improve levels of nuclear safety wherever possible.

1 THE RESEARCH ROLE OF THE NUCLEAR REGULATORY COMMISSION

The Nuclear Regulatory Commission (NRC) regulates civilian nuclear activities in order to protect the public health and safety, the quality of the environment, and national security. This responsibility requires the development and implementation of effective regulations, licensing processes, and inspection and enforcement activities.

As part of its licensing and regulatory development, the NRC must routinely make policy decisions involving complex technical issues and monitor the implementation of its regulations by industry. The Congress, in the Energy Reorganization Act of 1974, included the Office of Nuclear Regulatory Research as a statutory office of the NRC. In doing so, the Congress recognized the need of the agency to have research resources to provide:

- Independent confirmation of technical issues on which the agency would have to make decisions,

- Independent audit capabilities to assess the adequacy of submittals from various applicants for nuclear facility licensing.

A key purpose of the research program is to assist in the prevention of accidents by providing an understanding of their potential cause and helping to ensure that the regulatory process is effective in reducing the probability of their occurrence. Even when accidents occur, the knowledge attained through research can help those responsible for coping with the accident make the right decisions to mitigate the potentially hazardous consequences.

The research program supports NRC regulatory activities by providing methods and data necessary to achieve a reliable understanding of nuclear-related safety phenomena. The data and methods are generated in forms ranging from quick experimental or computational analyses to comprehensive computer models or experimental programs requiring many years to complete.

Research efforts support many NRC activities in ways that are often not readily apparent. For example, licensing (regulation) involves thorough assessment of operating problems and review of licensee submittals. This process relies on research results and models developed and validated by research. Research efforts also strengthen and improve licensing as well as inspection and enforcement processes and procedures. Thus research achievements often are not identified separately, but rather as an integral element of major agency decisions. Some research findings have identified previously unanticipated safety problems and have resulted in new or more stringent regulations. Other research findings have shown that existing systems are safer than previously thought or that regulations have been misdirected and have allowed the reduction or elimination of unnecessarily conservative or misdirected regulations, with consequent improved plant safety and efficiency, reduced costs to the ratepayer, and more reliable power production.

Over the years, considerable Federal funds have been spent on nuclear safety research. This report has been developed to respond to Congressional inquiries about the nature of the research and the ways in which it has contributed to certain key decisions in recent years. The impacts described are representative of the continuing impacts of research expected in future years.

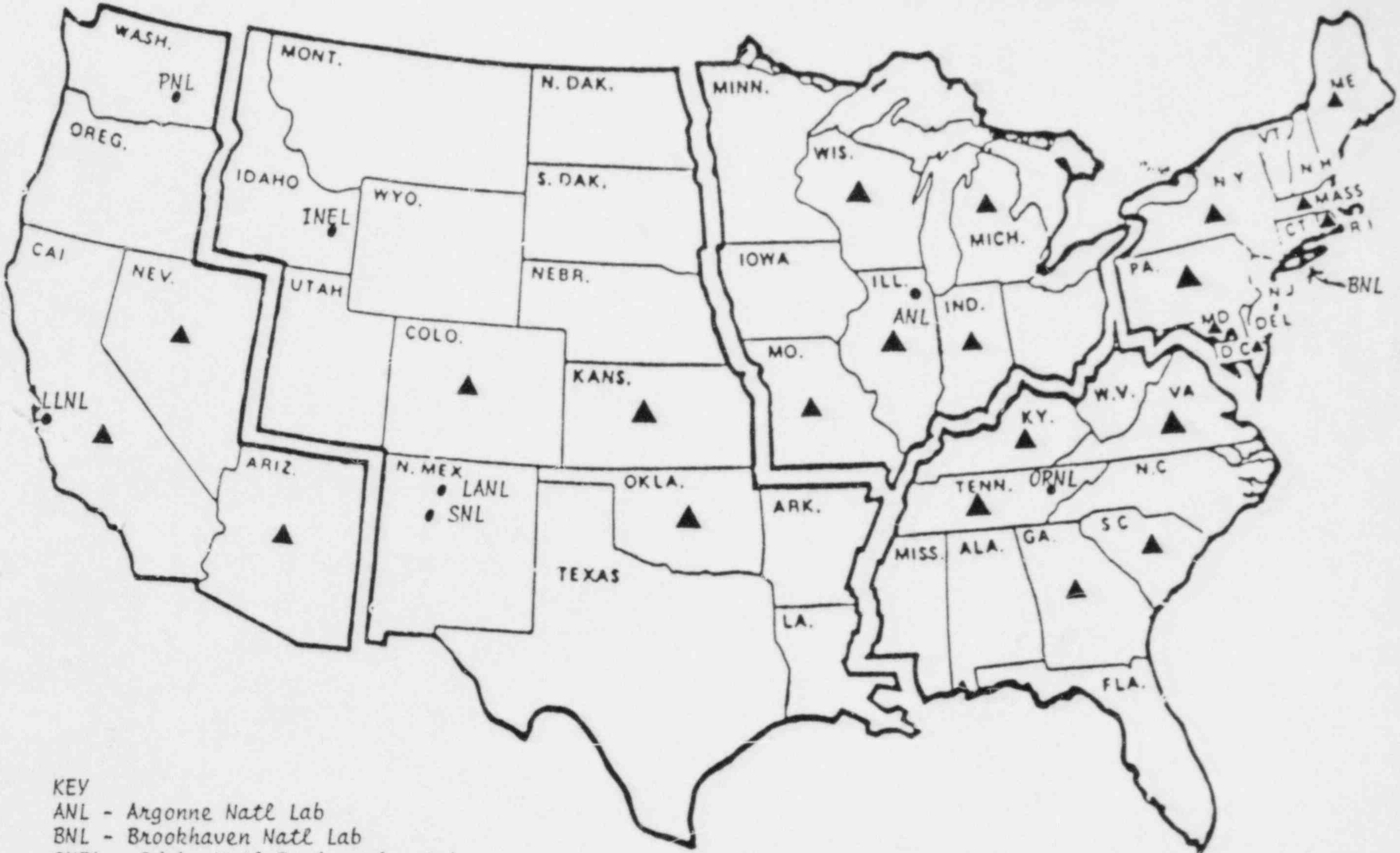
To keep the report of manageable size, only a sample of all the research accomplishments is presented. Hence, only some of the many organizations and facilities which contribute to NRC nuclear safety research can be mentioned.

Current research activities are conducted at many institutions throughout the United States (See Fig. 1.1). The majority of NRC-supported research is conducted at the DOE National Laboratories. NRC also supports work at a number of universities, not-for-profit technical organizations such as Battelle Memorial Institute and the Franklin Institute, professional societies, and private organizations. Some programs are jointly funded with other Federal agencies such as the Department of Energy (DOE), U.S. Geological Survey (USGS), and U.S. Army Corps of Engineers and state governments such as Tennessee and Maine. Additionally, private research partners include the Electric Power Research Institute (EPRI), reactor vendors, and reactor owners groups. Often the research is centered around the joint use of major and expensive test facilities.

Additionally, from its inception, the NRC research program has actively sought international cooperation in research activities. Table 1.1 lists the foreign countries and organizations with which the NRC cooperates in various areas of research. The benefits of international cooperation include the timely acquisition of knowledge, reduction of the cost of work to all parties involved, and access to a broader range of experience.

FIGURE 1.1

DOE LABORATORIES AND UNIVERSITIES PARTICIPATING IN NRC RESEARCH



KEY
ANL - Argonne Natl Lab
BNL - Brookhaven Natl Lab
INEL - Idaho Natl Engineering Lab
LANL - Los Alamos Natl Lab
LLNL - Lawrence Livermore Natl Lab
ORNL - Oak Ridge Natl Lab
PNL - Pacific Northwest Lab
SNL - Sandia Natl Labs

▲ - Universities

Table 1.1 INTERNATIONAL RESEARCH AGREEMENTS

Country	Area of Agreement*					
	Reactor Safety (General)	Thermal Hydraulic Transients	Risk Analysis	Severe Accidents	Reactor Engineering	Waste Management
Belgium		x			x	
Canada				x		
Commission of European Communities	x					
Finland		x			x	
France	x				x	x
Federal Republic of Germany	x	x		x	x	
Italy		x		x		
Japan	x	x		x		x
Korea				x		
Netherlands				x		
Philippines		x				
Spain	x	x				
Sweden	x	x		x		
Switzerland		x			x	
American Institute in Taiwan			x	x		
United Kingdom	x		x	x	x	
Halden Reactor Project	x					
OECD LOFT		x				

*Specific research efforts are generally the subject of individual tailored agreements under these general agreements.

2 NUCLEAR SAFETY RESEARCH CONTRIBUTIONS

This section focuses primarily on research conducted during the past five years and major regulatory activities applying that research. This time period was selected in order to examine the consequences of research that was substantially the result of the accident at Three Mile Island (TMI-2). As a result of this emphasis, the section omits much important work done prior to TMI-2. Likewise, it gives little attention to research which is only now nearing completion and has not yet been incorporated into NRC regulations or into inspection and licensing procedures. It also does not specifically address every area of research conducted during this period or every regulatory application of research in the areas that are covered.

The research areas selected for inclusion in this report are:

- Pressure Vessel Integrity,
- Piping,
- Small- and Large-Break Loss-of-Coolant Accidents,
- Hydrogen and Containment Loads,
- Source Term Analysis,
- Seismic Hazards,
- High-Level Waste Management.

2.1 Pressure Vessel Integrity

2.1.1 The Problem

The pressure vessel is the only safety component of the primary system for which there is no mitigating system to cope with its failure. Therefore, it is critical that its structural integrity be maintained. Severe cracks or other damage to the pressure vessel can allow coolant to leak out and can, in some cases, lead to severe accidents from overheating of the reactor core, with potential releases of radioactivity to the public.

Nuclear reactor pressure vessels are made of heavy steel sections (8 to 12 inches thick) welded together. These sections and welds are exposed to high pressures, temperatures, and radiation throughout the power plant's life. In PWRs, as pressure vessels are irradiated, resistance to cracking decreases. This decrease in crack resistance is called embrittlement and is of concern in some older plants. The concern arises because stresses that occur in many transients encountered in operation are sufficient to cause the crack to grow. If the material, e.g., a weld, is brittle enough, the crack can propagate to such an extent that the vessel could be described as "shattering like glass."

2.1.2 The Research Program

The pressure vessel research program was designed to identify potential problem areas early enough to take corrective action. One of its major objectives was to determine whether and under what conditions existing cracks can become larger and result in potentially catastrophic leaks.

Like many research programs, the pressure vessel research program has involved a number of complementary efforts, including:

- Development of analytical methods (computer models) to predict how cracks will grow under specified loads,
- Experimental validation of these methods using test samples,
- Measurement of material properties needed as input to the models,
- Dosimetry (measurement of actual radiation exposures).

The modeling and experimental work have been particularly extensive.

The Office of Research has developed computer codes to model material characteristics and the way those charateristics change in reactor environments. Because the radiation received by the vessel differs by location laterally and through the thickness of the vessel, a model was developed that attempted to represent the nonuniform embrittlement of the metal in the vessel. While the metal behavior in the high-embrittlement and low-embrittlement zones can be well represented, the regions of intermediate characteristics are harder to model. Therefore, research is continuing to improve the capability to model vessel behavior in this area.

Extensive experimental work on crack behavior in pressure vessels has been conducted in the Heavy Section Steel Technology (HSST) facilities at Oak Ridge National Laboratory in Tennessee (Figure 2-1). These facilities are large enough to simulate realistic crack depths and sizes and to provide safe areas to test irradiated materials. HSST facilities are used to test sections of pressure vessels as well as weld metals. Some tests are designed to demonstrate whether the computer models predict the behavior of irradiated metals accurately, in particular, cracking. Provisions are available to use low temperatures to make the metal more brittle to simulate the embrittlement induced by long reactor service. Further experimental work on the arrest of cracks in thick section vessels is done by the National Bureau of Standards (NBS) in a cooperative program with the ORNL staff. The research takes advantage of the NBS' six million pound tensile machine, Figure 2-2, for testing of specimens four inches thick and nearly 30-feet long. Important gaps in the understanding of crack initiation, propagation and arrest are being filled by this cooperative research.

2.1.3 Regulatory Impacts of the Research

Improvements in knowledge about pressure vessel behavior developed from NRC research programs have indicated that some regulations are more conservative than necessary, while some are not conservative enough. Therefore, the research has been and is being used to modify regulations as appropriate.

Perhaps the most visible recent application of research in this area is the resolution of the pressurized thermal shock (PTS) problem. Under certain accident sequences, cold water can be injected into a PWR vessel while the reactor is still at high pressure. This creates a condition called pressurized thermal shock. Stresses generated under these conditions may cause preexisting (code-allowed) microscopic cracks to grow--possibly through the pressure vessel or

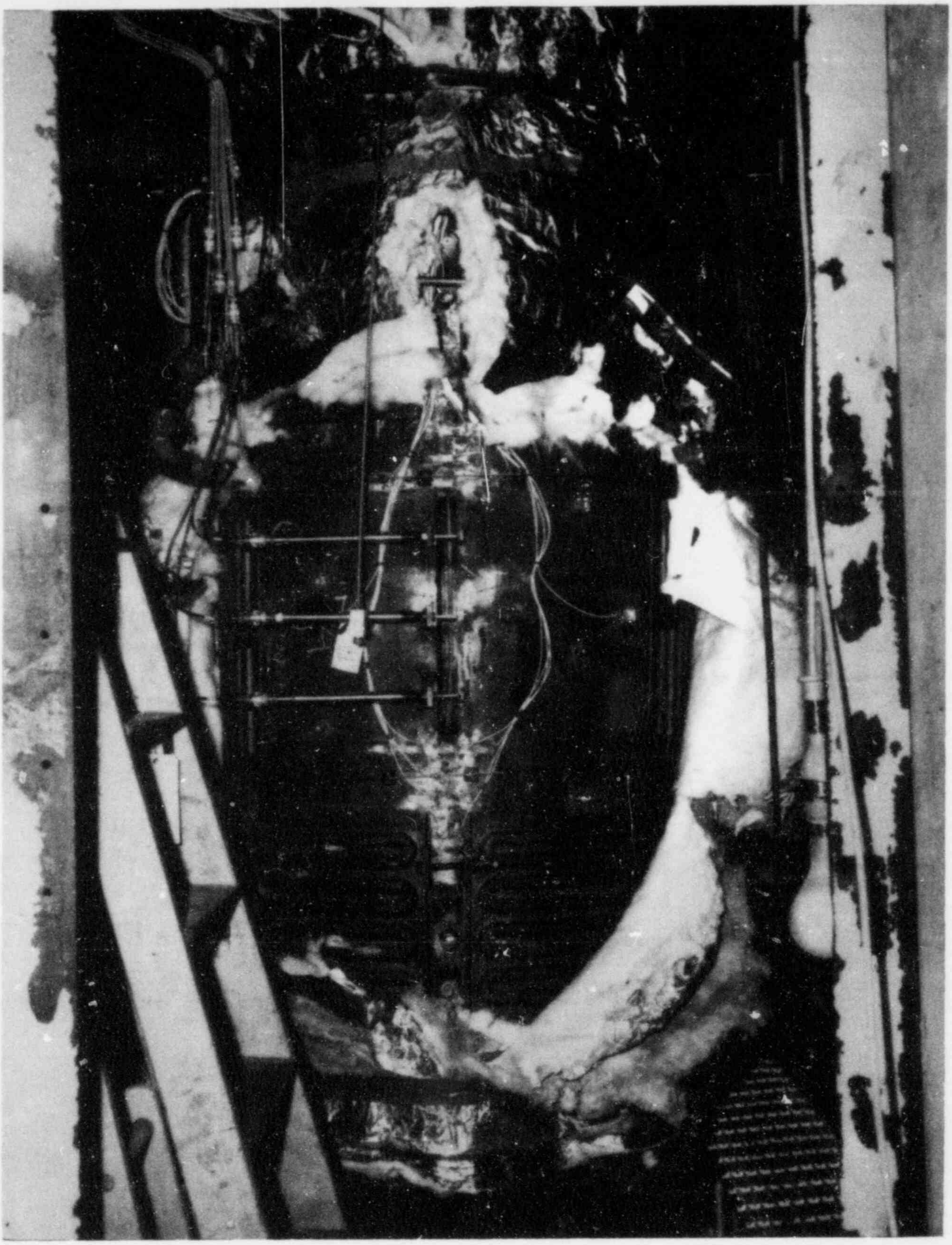


Figure 2-1 The protective insulation has been pulled aside following the testing of a weld-repair portion of a six-inch-thick pressure vessel. A flaw more than five inches deep and 18 inches long was created in the area and was then subjected to pressure overloads more than double the design pressure without disruptive failure.

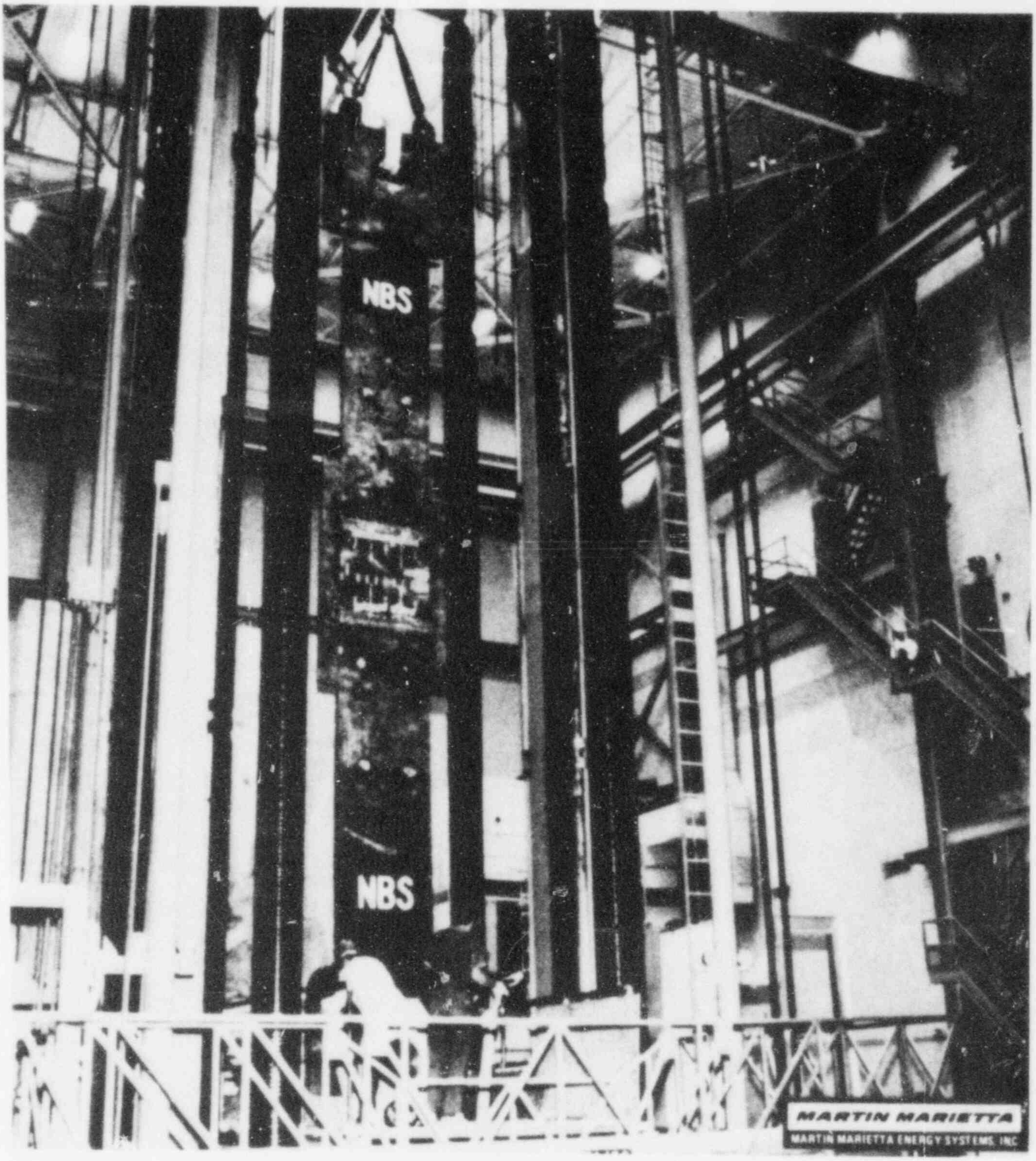


Figure 2-2 HSST Wide--Plate Crack--arrest specimens are vertically mounted in NBS test machine.

its welds if the metal is brittle enough. Older vessels with high copper impurities in the welds are particularly susceptible to failure induced by PTS. NRC research results demonstrated that even the most susceptible vessel has sufficient safety margin that remedial action was not necessary immediately, although such action might be required later. Based on these findings, a shutdown of 8 older PWRs with high-copper-content welds in the pressure vessel was not necessary. This decision has been called "an excellent example of cooperation and support between RES and NRR [Office of Nuclear Reactor Regulation], in striving to accomplish the agency mission."*

The findings also resulted in the addition of a new NRC rule, Section 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." The rule imposes new requirements on operators to ensure pressure vessel integrity. In particular, it requires specific remedial actions such as neutron flux reduction when calculations project that reactor vessel brittleness will exceed certain critical levels during the plant lifetime.

Pressure vessel research has also had application in a number of other areas, including the modification of the ASME Boiler and Pressure Vessel Code, which is referenced by NRC regulation Section 50.55a "Codes and Standards," and the Revision of Appendix G of 10 CFR 50 on fracture toughness requirements.

2.2 Piping

2.2.1 The Problem

Nuclear plant systems use thousands of feet of pipe to carry heated water (in PWRs) or water and steam (in BWRs); leakage of coolant from the piping can lead to severe accidents from overheating of the reactor core, with potential releases of radioactivity to the public in some cases. Although safety systems such as emergency core cooling systems are designed to mitigate the effects of such a failure, the first line of defense is to prevent the accident.

In the absence of more detailed knowledge, it was felt that a bounding case for regulatory purposes would be an assumption that the pipe would break suddenly (a "double-ended guillotine break") and thus allow water to flow freely from both ends of the break. At the high pressures encountered in reactors, such a broken pipe would whip around like an untended garden hose, so the pipe break case leads naturally to consideration of restraints to keep the pipe in place. Thus much research has focused on preventing dangerous cracks from developing or going undetected. Also work has focused on determining if the pipe will indeed tend to break suddenly, or whether it will first develop a leak which can be detected so corrective action can be taken before a sudden break occurs. (Leak before break.)

The issue of the validity of leak before break has been heightened recently by the significant number of cracks detected in stainless steel pipes of all size in Boiling Water Reactors (BWRs). The cracking has occurred because of a unique

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*Memorandum to Robert B. Minogue, Director of the Office of Nuclear Regulatory Research, from Harold R. Denton, Director of the Office of Nuclear Reactor Regulation, "RES Support of the Pressurized Thermal Shock Project," September 25, 1985.

combination of effects from material composition, fabrication procedures and nature of the cooling water in BWRs. Most of the cracks detected so far have not penetrated the whole wall of the pipes, but some have, causing leaks. As those leaks have occurred, however, they have been detected long before the crack could extend to a size critical to safety. Reliable inservice inspection efforts are necessary in order to provide assurance that the cracks will be detected before they grow to leaks.

Ensuring the integrity of critical LWR piping systems requires developing appropriate design criteria, understanding possible mechanisms of piping degradation, defining adequate pre- and in-service inspection programs and techniques, determining acceptability of continued operation or the adequacy of corrective actions when cracks are detected, and establishing the adequacy of proposed fixes for eliminating or reducing the incidence of cracking.

2.2.2 The Research Program

A number of different types of research are conducted on piping systems including:

- Developing and testing in-service inspection techniques to detect and characterize cracks,

- Experimental work on components such as bending tests on pipes to determine resistance to breaking,

- Basic materials studies, including metallurgy and corrosion,

- Development of engineering (e.g., pipe bending) and mechanistic (e.g., corrosion processes in welds) models.

Both field tests and laboratory experiments involving materials from operating reactors have been used to develop improved in-service inspection equipment and procedures. Much of this work has been done in a cooperative effort with the Electric Power Research Institute (EPRI). The results have been taken into account in the ASME Inservice Inspection Code.

The Office of Research participates in an on-going international program--Program for the Inspection of Steel Components (PISC)--to test current and advanced ultrasonic inspection equipment and procedures on a variety of test samples, both with artifically induced flaws and with real ones. The Office of Research is continuing to evaluate an advanced computerized inspection technique, the synthetic aperture focusing technique (SAFT). SAFT provides a pictorial representation of the flaw in real time and has a superior ability to detect and size certain intergranular stress corrosion cracks. Earlier, there was much concern over our ability to detect cracks in the interior of welds in stainless steel piping.

Pipe bending experiments involve large facilities at Battelle-Columbus Laboratories in Ohio that are capable of bending pipes ranging from 4 to 42 inches in diameter. The pipes are heated electrically and filled with water to simulate typical reactor environmental conditions. The experiments are designed to test how much bending pipes with small cracks can withstand before the cracks start to expand and how large the cracks can become before the pipe breaks. There are plans for research to continue with international financial support to examine more extreme conditions of temperature, pressure, and seismic loadings.

The basic material studies use small samples of piping materials. Sophisticated material test facilities at Argonne National Laboratories in Illinois and the David Taylor Naval Ship Research and Development Center in Maryland allow testing for susceptibility to cracking under a wide range of reactor coolant water chemistries, temperatures, and pressures. Material samples removed from operating reactors such as the Georgia Power Company's Hatch Plant are also being tested. The smaller sizes and simpler geometries of material tests allow extensive controlled testing under a variety of conditions. The extrapolation of these results to more realistic geometries can then be done by performing a limited number of more complex tests such as the pipe bending experiments. The tests on actual reactor samples ensure that the artificial techniques used to simulate thermal aging and other long-term environmental effects are realistic.

The extensive experimental work by Battelle, Argonne, and others, both on material characteristics and on piping behavior, was used in the development of models for predicting structural and material behavior in the operating environment. Both mechanistic models, dealing with material behavior, and engineering models, dealing with piping behavior, have been developed. The models are still being modified as additional experimental data become available.

2.2.3 Regulatory Impacts of the Research

In response to conservative assumptions regarding the nature of pipe breaks, the general design criterion applied to piping, GDC-4, in Appendix A to 10 CFR 50, called for consideration of the worst case pipe break, the double-ended guillotine break (DEGB). In turn, this led regulators to develop requirements for component supports and other structural members. In particular, the DEGB criteria resulted in a requirement for massive pipe whip restraints and water jet impingement shields to prevent additional damage to the reactor plant after a DEGB from the unrestrained motions of broken pipes. These provisions made thorough inspection for cracks difficult at best, with significant radiation exposure to workers inspecting that part of the pipe that was accessible. Thus the conservative assumption led to the adoption of design techniques that may have had a negative effect on overall safety.

Bending and breaking experiments have shown that the materials used in piping are sufficiently tough to resist a double-ended guillotine break as postulated. Serious cracks would leak enough to be detected and allow reactor shutdown long before such a break would occur. In addition, probability estimates done by the Office of Research have confirmed that the probability of a double-ended guillotine break is very low.

The consequences of these findings are substantial. They support changing General Design Criterion 4 (GDC-4), "Environmental and Missile Design Bases," which requires that safety-related structures, systems, and components withstand the effects of a large loss-of-coolant accident.

The proposed change would allow the removal of some pipe whip restraints and water jet impingement shields if certain stringent criteria are met. Improper installation of pipe whip restraints and water jet impingement shields could increase piping stresses, reduce the reliability of inservice inspections by restricting inspection access, and increase the radiation exposure associated with in-service inspection and maintenance operations.

Therefore, if the necessary criteria are met, eliminating requirements for these components can improve worker safety and reduce cost. The operational savings could be large enough to justify the removal of this equipment from existing reactors in many cases. The improved inspectability of equipment and reduced installation errors may also reduce the risk to the public.

Research in nondestructive examination (NDE) for stainless steel piping has also resulted in more stringent qualification requirements for inspectors as well as tighter in-service inspection requirements. Proposed modifications to the American Society of Mechanical Engineers (ASME) code will require the utilities to use inspection procedures and equipment with better flaw detection capability than was previously required. Research has also allowed the qualification procedures which test inspection capabilities to be strengthened, both in terms of the number of flaws in a standard test sample that can be detected and in the realism of those flaws (i.e., naturally occurring flaws are used instead of machined flaws).

Research on piping systems has also been applied to a number of other areas of regulatory activity relating to piping, including:

- Development of a model for predicting stress corrosion cracks to ensure continued operational safety (for example, when cracks were found in the large recirculation piping at the Monticello plant in Minnesota, the models showed that the plant could be operated safely until the next regular maintenance period),
- Verification of the value of certain stress reduction techniques proposed by industry (for example, induction heating stress improvement) that will change the residual stresses in a pipe weld to help preclude both initiation and growth of intergranular stress corrosion (IGSC) cracks,
- Verification of industry proposals to use different materials (for example, testing of Type 316NG (nuclear grade) and Type 347 stainless steels for resistance to IGSC to determine if they are acceptable substitutes for existing materials which are cracking),
- Verification of proposals from industry to improve water chemistry control to reduce intergranular stress corrosion.

2.3 Small- and Large-Break Loss-of-Coolant Accidents

2.3.1 The Problem

A loss of coolant in the reactor can, if not stopped or compensated, lead to a sequence of events causing damage or melting in the reactor and possibly culminating in the release of radioactivity to the public. Therefore, an understanding of the consequences of loss of coolant is critical to establishing performance requirements for emergency core cooling systems in order to ensure safety.

Such losses of coolant flow, called loss-of-coolant accidents (LOCA), can result from breaks in reactor coolant, steam, or feedwater piping. Research on piping discussed in the previous section deals with the material characteristics of piping that can lead to cracks or breaks in the piping. Research on LOCAs, discussed in this section, deals with the coolant behavior if a crack or break in piping should occur and thereby threaten a line of defense against release of fission products.

In LOCAs, the movement of water and energy in the vicinity of the nuclear fuel is a major determinant of the integrity of the fuel cladding barrier. The science of fluid and heat movement or transfer is known as thermal hydraulics. Accurate thermal-hydraulic analysis is critical in understanding and predicting the possibility of fuel melting during a LOCA or other coolant transient. A thorough and accurate analysis requires detailed knowledge of the behavior of the complete plant including the nuclear steam supply system and the turbine generator system.

2.3.2 The Research Program

Research in the thermal-hydraulics area has several components:

- Development of computer models of the different reactor coolant systems, including the steam side of the plant,
- Experiments on scale model tests of the coolant system (integral coolant loops and experiments),
- Experiments on particular phenomena or isolated features of coolant systems (separate-effect experiments).

The Office of Research has developed and maintained several codes to model LOCAs and thermal-hydraulic transients. Different versions of these codes were developed for Pressurized Water Reactor (PWR) systems and for Boiling Water Reactor (BWR) systems. The codes have the ability to simulate the flow of water and heat that occurs in the nuclear power plant. These codes provide NRC with a rapid and independent ability to assess the course of an accident and to analyze accident impacts.

Extensive experimental work involving a number of facilities was performed to develop an understanding of thermal-hydraulic phenomena and to verify the codes. Two basic types of facilities were used in the experimental program: integral-test facilities, so called because they examine the performance of a system with all major components, and separate-effect facilities, so-called because they are used to assess individual thermal-hydraulic phenomena or the behavior of separate components of a system.

A major part of the testing has been done at the Loss-of-Fluid-Test (LOFT) Facility (Figure 2.3) and at the Semiscale facility, both at the Idaho National Engineering Laboratory. LOFT was a 50 megawatt (thermal) pressurized water reactor (PWR) designed as an integral test facility to simulate the major components and responses of a commercial-size PWR during a loss-of-coolant accident. The facility is approximately 1/50 of the volume of a typical commercial plant. Originally operated under an NRC-sponsored program, the program was taken over in 1982 by a 10-nation international consortium, of which the NRC

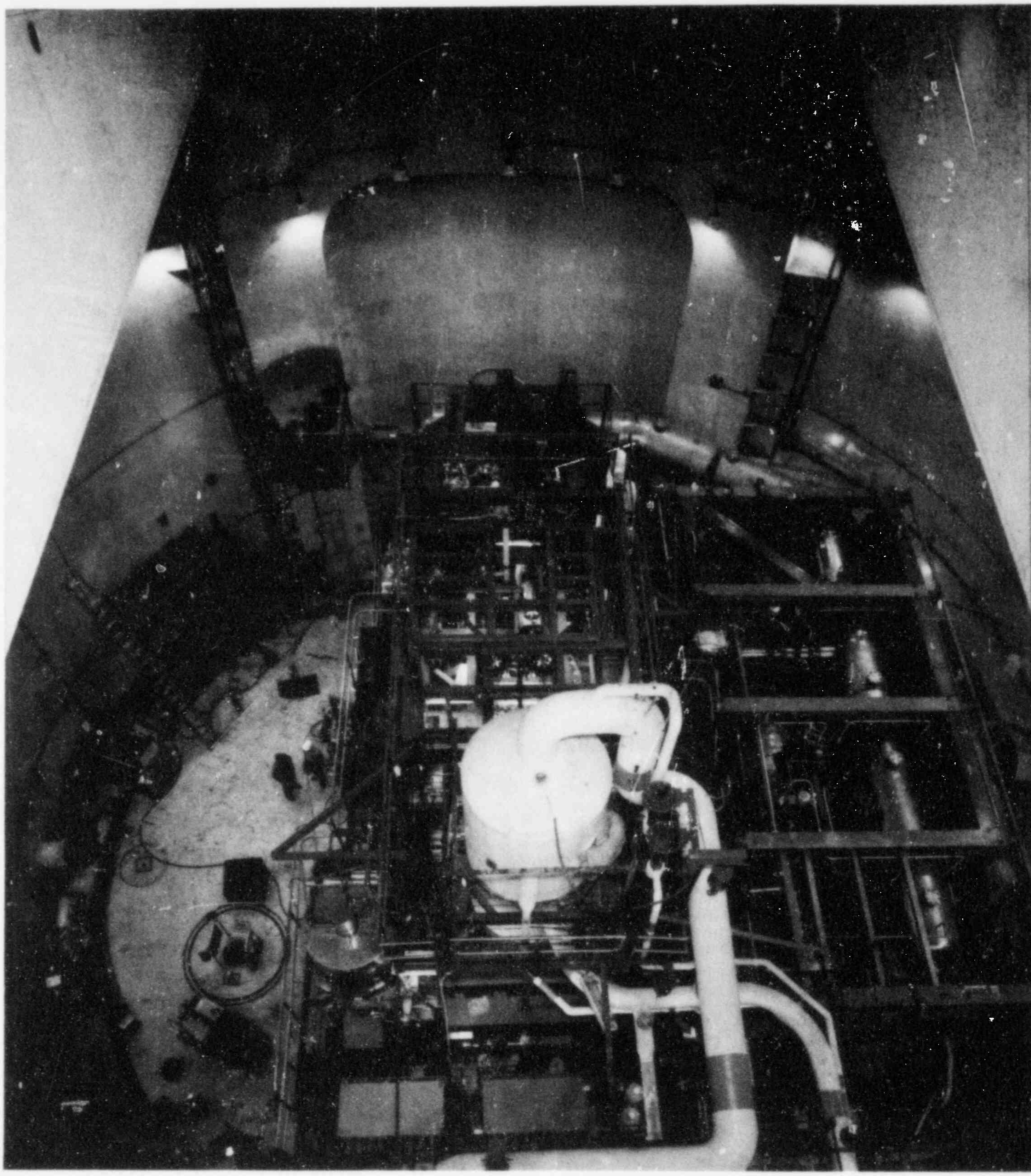

Figure 2.3 LOSS-OF-COOLANT FACILITY (LOFT)

Top view, inside containment. LOFT, at the Idaho National Experimental Laboratory (INEL), has been the site of many experiments simulating various types of reactor accidents. Research results have been used to confirm, assess or improve computer codes used to predict plant behavior.

was a member. About 50 experiments were conducted at the LOFT facility between 1976 and 1985, most of which used the nuclear core to generate the heat for the transient tests. The final test, conducted in 1985, deliberately produced extensive melting in a test fuel assembly in order to study the system behavior and fission product release under severe accident conditions.

Semiscale is another smaller integral test facility. Specifically it is a 1/1500 scale model of a commercial pressurized water reactor that uses a small (25 rod) bundle of full-length electrical heaters to simulate the core. Originally a pretest facility for LOFT, it was later improved and used for additional thermal-hydraulic studies. Ultimately, over one hundred tests were conducted using this facility. Figure 2.4 shows the scale between LOFT, Semiscale, and commercial plants. The planned research programs for both facilities have been completed. LOFT is no longer operating. Semiscale is temporarily shut down, pending a decision regarding its further use or permanent shutdown. Other facilities of a similar nature were operated cooperatively with industry: the Fully Integrated Systems Test (FIST) facility (with G.E. and EPRI) and the Multi-Loop Integrated Systems Test (MIST) facility (with B&W, B&W Owner's Group, and EPRI). FIST is currently shut down, MIST is in operation.

Separate-effect research is conducted at over a dozen facilities in the U.S. and abroad, including universities, DOE National laboratories, and private research organizations. These facilities are used to examine either thermal-hydraulic behavior in specific components and configurations of the reactor or specific phenomena that play a major role in accident analysis. Because of the relatively narrow scope of most of this work, frequent use is made of university faculty and students who are able to produce valuable work without complex, costly test rigs.

2.3.3 Regulatory Impacts of the Research

One of the major recent consequences of research in the thermal-hydraulics area is its impacts on emergency core cooling system performance requirements. An early result was the acceptance of a revised ECCS analysis method for BWRs based on tests at a number of facilities including the joint program at the Full Scale Integral System Test (FIST) facility at General Electric, San Jose. A more fully revised ECCS Rule contained in 10 CFR 50.46 and Appendix K to Part 50, "ECCS Evaluation Models," is being developed. Section 50.46 requires that calculations of ECCS performance be performed using the methods contained in Apprendix K and that these calculations show that the ECCS would prevent the fuel from reaching certain temperatures and other conditions that could result in significant damage to the fuel cladding during a LOCA. The revision, if put into effect, will change the methods the licensee may use to determine the maximum safe operating conditions. The change will permit improved fuel management and operational flexibility in those cases where the new method reduces unnecessary conservatism. According to one industry estimate, these improvements could save electric ratepayers between three and six million dollars per plant per year, with no loss of safety.* If the capacity of plants is uprated based on the ECCs rule changes, the savings could be even greater.

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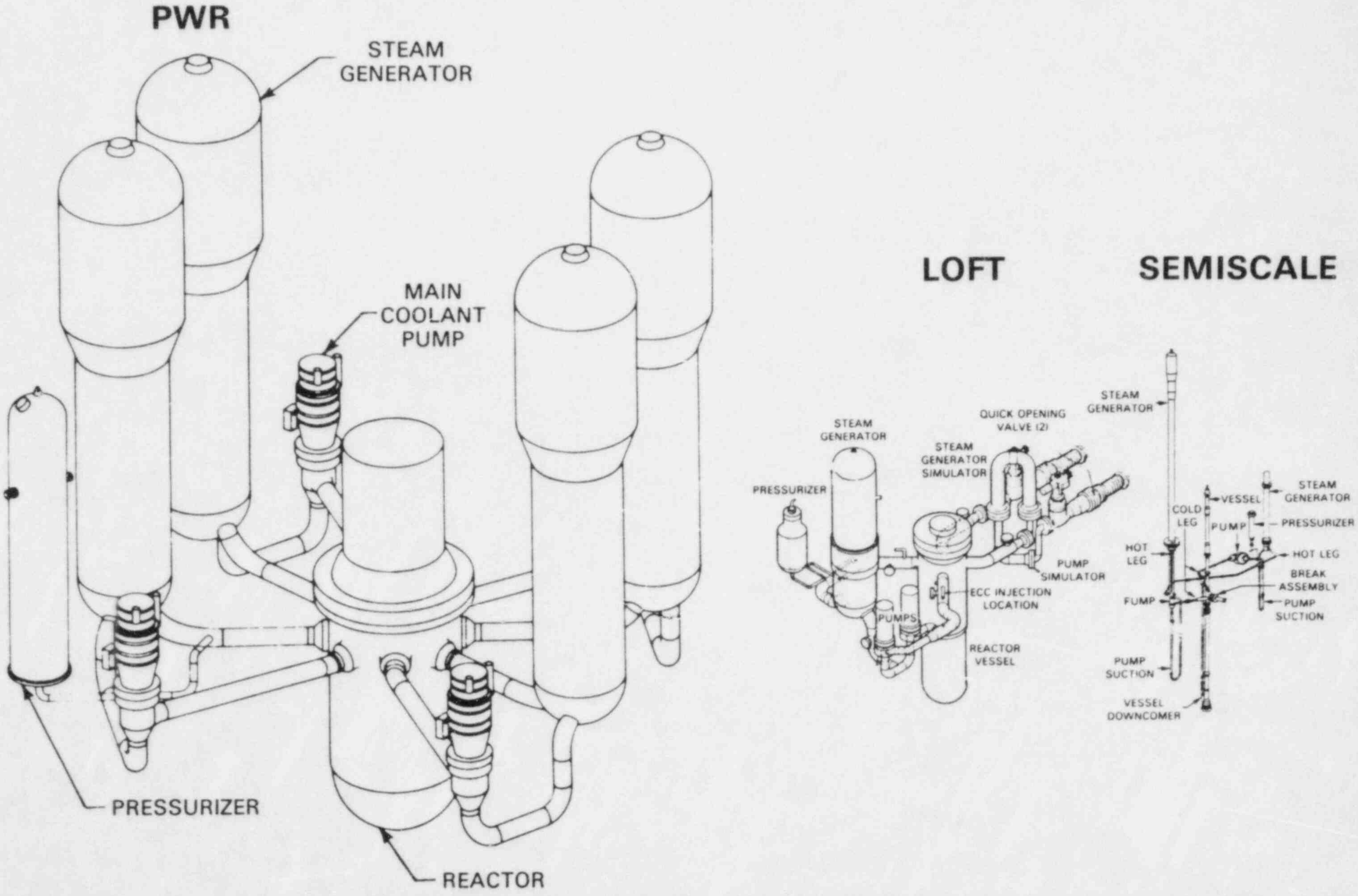
*Letter - E. P. Rahe, Jr., Manager, Nuclear Safety, Westinghouse Electric Company to D. F. Ross, U.S. Nuclear Regulatory Commission, February 8, 1985.

Figure 2.4

Relative Scale and Configuration of PWR and Major Loss of Coolant Test Facilities



Another major regulatory use of the thermal-hydraulic computer codes was the development of a thermal-hydraulic basis for assessing system performance for the pressurized thermal shock (PTS) issue. The thermal-hydraulic modeling was done in parallel with the materials research described earlier. Since PTS can occur from rapid reactor temperature fluctuations, the specific thermal-hydraulic characteristics needed to be identified. Thermal-hydraulic codes were used to simulate a small-break LOCA that resulted in cold water injection into the pressure vessel. The models showed that adequate mixing occurs to prevent the relatively cold emergency water from creating large temperature decreases in the pressure vessel wall material. This finding, based on EPRI work as well as NRC research, coupled with the materials research that demonstrated adequate fracture resistance to the possible thermal conditions, contributed significantly to resolution of the PTS issue.

2.4 Hydrogen and Containment Loads

2.4.1 The Problem

Risk to the public from the operation of nuclear power plants occurs primarily from those accidents that result in containment failure. Although accidents of such severity constitute a very small fraction of all possible accidents, their potential for public harm makes them important. A key regulatory concern is whether the associated strains and stresses of containment structures and their associated failure modes can be predicted accurately.

One potential mechanism for containment failure involves the generation of hydrogen during a severe accident. After a loss of coolant from the core, the cladding of the nuclear fuel can be overheated and react with the steam left in the reactor core region to form hydrogen. If sufficient quantities of hydrogen are released to reactor containment and combined with oxygen, explosions and fires could result. The sheer bulk of the hydrogen and other gases released into the containment also causes a pressure rise. These combined effects could cause containment leakage and the subsequent release of radioactivity to the environment. Other sources of containment loading are also studied; should the molten core debris escape the vessel and react with the concrete basemat, hydrogen, carbon monoxide, and many other products would be released.

2.4.2 The Research Program

The research program in this area involves the development of computer models and the performance of experiments to improve the understanding of how hydrogen behaves in the reactor containment. The Office of Research has developed several computer codes to model how the hydrogen moves from the core to the containment, the conditions under which it burns or explodes in the containment, and, finally, what temperatures and pressures are generated by the burning.

Code development is supported by a variety of experiments at several facilities:

- The Flame Acceleration Measurements Experiments (FLAME) facility (Figure 2.5) was designed to investigate the effects of the size of the volume where the hydrogen burns (i.e., scale), hydrogen concentration, venting, and obstacles on flame acceleration and detonation. This facility, which is 100 ft long, 8 ft high, and 6 ft wide, is

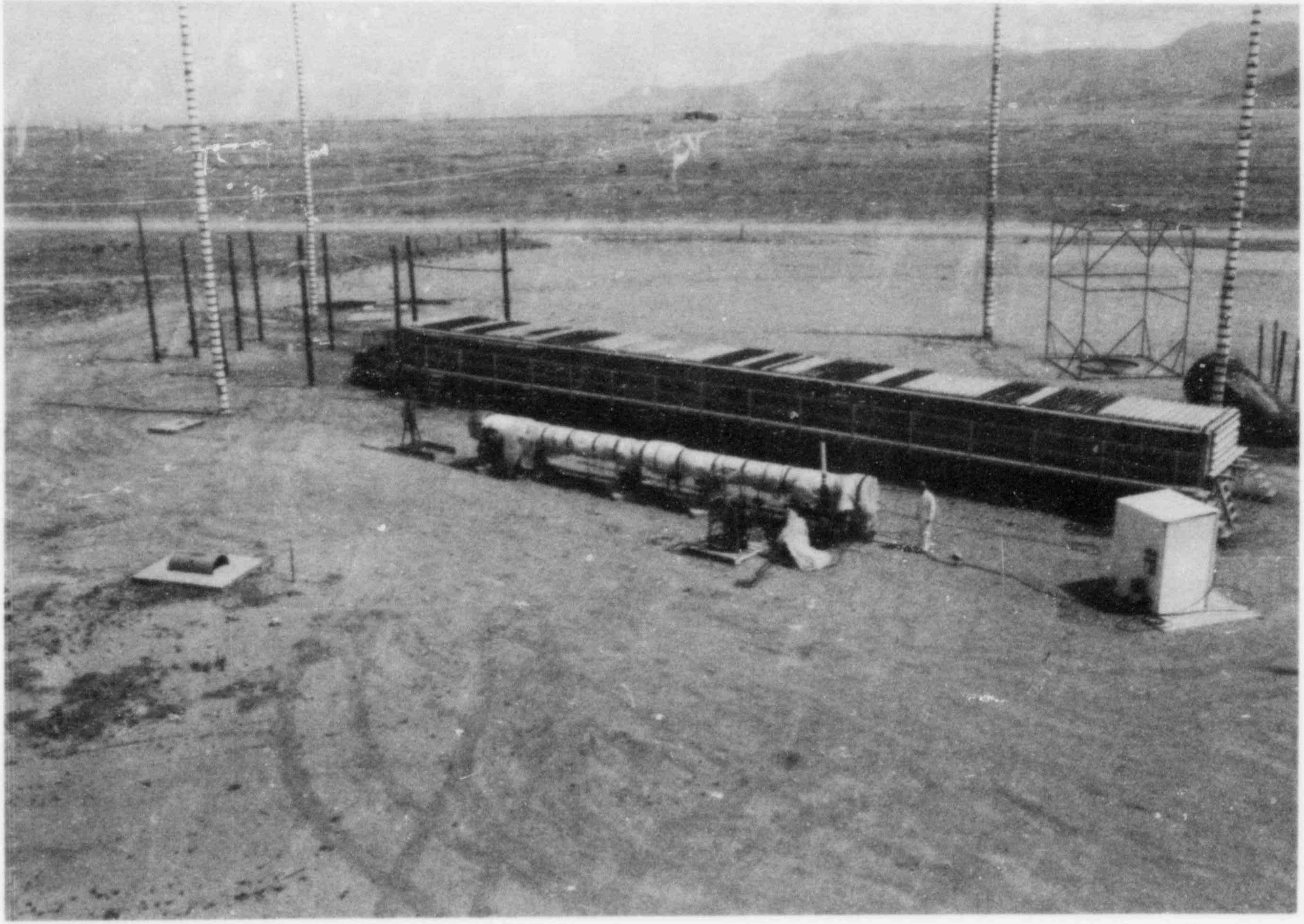


Figure 2.5 THE FLAME AND HDT RESEARCH FACILITIES

These facilities located at Sandia National Laboratories in New Mexico are used to conduct experiments which contributed to the understanding of the behavior of hydrogen in the containment during severe accidents. The FLAME facility is the large rectangular structure and the HDT facility is the smaller cylindrical structure.

approximately a half-scale model of the upper plenum of an ice condenser PWR. Tests are made of how a flame starting at one end of the plenum develops as it travels to the other end.

- The Heated Detonation Tube (HDT) (Figure 2.5) was designed to determine under what conditions a mixture of hydrogen and other gases found in the containment burns (deflagration) and when such a mixture explodes (detonation). The initial concentration of hydrogen and steam and the gas temperature are varied to define those mixtures that detonate and hence could cause severe damage.

- The Nevada Hydrogen Tests (Figure 2.6) were designed to examine the flammability of hydrogen, air, and steam mixtures and igniter performance. (Utilities proposed to use igniters to cause the hydrogen to burn harmlessly before an explosion occurred.) As a secondary byproduct, a limited attempt was made to determine the effects of hydrogen burning on the survival of safety-related equipment. The test program was performed in cooperation with the Electric Power Research Institute (EPRI). Forty separate tests covering a variety of hydrogen and steam concentrations and injection flow rates were performed in a spherical vessel (originally designed to store gas for the nuclear rocket program) approximately 52 feet in diameter. Instruments were added to measure the temperature and pressure during the hydrogen burn.

Several codes have been developed to model a hydrogen burn in different phases at different scales. One code models the overall transport and combustion of hydrogen. Another code focuses on the detailed condition of the local environment of equipment exposed to burning hydrogen. The code development thus far has emphasized primarily the smaller containment types such as those found at the Grand Gulf plant in Mississippi (a BWR Mark III) and the Sequoyah plant in Tennessee (a PWR ice condenser), which were of greatest concern. (An ice-condenser plant uses ice to condense steam in the containment and thus is designed for a lower level of pressure than the more typical containment.) The codes now need to be expanded to incorporate models for other containment types, as well as being extended to a variety of components in different geometries.

2.4.3 Regulatory Impacts of the Research

A major impact of the research has been the identification of the need for additional requirements for hydrogen control equipment in the smaller, weaker containment types. Regulations in 10 CFR 50.44 were amended in 1985 to improve the hydrogen control capability and survivability of safety systems in nuclear power plants that have BWR Mark III or PWR ice condenser containments.

These amendments impose new requirements on owners of BWR Mark III and PWR ice condenser type reactors to:

- Provide each reactor with a system capable of handling hydrogen without loss of containment integrity,

- Demonstrate that safety systems are capable of functioning during and after a hydrogen burn in each reactor that does not use an inerted

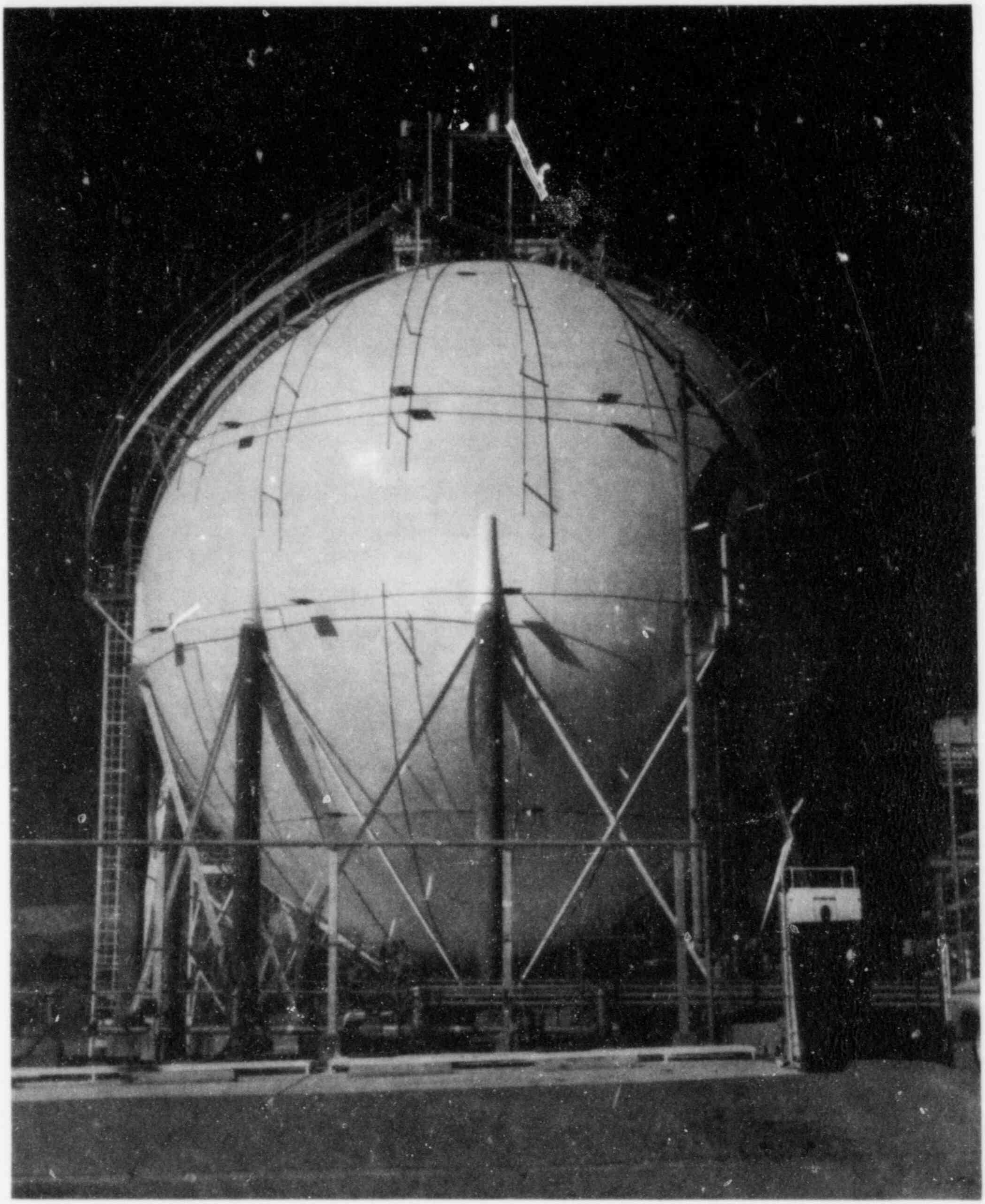


Figure 2.6 Nevada Hydrogen Test Facility

Experiments conducted at this facility on the burning of hydrogen under simulated accident conditions contributed to the technical basis for hydrogen control requirements.

atmosphere (one in which the oxygen is replaced by nitrogen gas) for hydrogen control,

- Perform analyses for each reactor to support the design of the hydrogen control system and to ensure the structural integrity of the containment and the survivability of needed safety systems during a hydrogen burn.

2.5 Severe Accident and Source Term Research

2.5.1 The Problem

The quantity, timing, and characteristics of radioactive materials (called "source terms") released in a severe reactor accident determine the potential hazard of the accident to the public. Because source terms used in current regulation do not accurately reflect actual radioactivity releases, they cannot reliably be used to determine how best to prevent severe accidents or mitigate their effects. Nor are they adequate for risk assessments. Experience during the TMI-2 accident showed that regulatory models developed for siting regulations were not representative of what happened.

2.5.2 The Research Program

As part of the post-TMI efforts to improve the understanding of severe accidents (those that are beyond the normal design basis) in order to prevent their occurrence or mitigate their consequences, the Office of Research embarked on a Severe Accident Research Program which included detailed studies of source terms.

Research on severe accidents and the resulting source terms has involved experiments to collect data on the consequences of a variety of reactor accidents and the development of better models of the complex chemical and physical processes involved. This research had to address the detailed behavior of a large number of phenomena that can occur during the course of an accident, including:

- Fission product release from the fuel,
- Reactor coolant behavior during a severe transient,
- Core melting geometries and characteristics,
- Interactions between the melting core and concrete basemat,
- Aerosol behavior within the containment,
- Fission product transport within the reactor coolant system.
- Effects of hydrogen burning and detonation.

The interrelationships between these phenomena and the extent of the research program can be illustrated by the computer codes used (see Figure 2.7). Each of these computer codes required experimental studies to develop a basic understanding of the physical and chemical phenomena involved and to validate the models. Experiments have included actual melting of fuel rods, pouring molten core materials into test concrete vessels, and other tests.

About the time this research program was started, a group was formed under industry auspices to make the best possible use of existing technology to produce a best estimate of the source term under the same conditions that the NRC

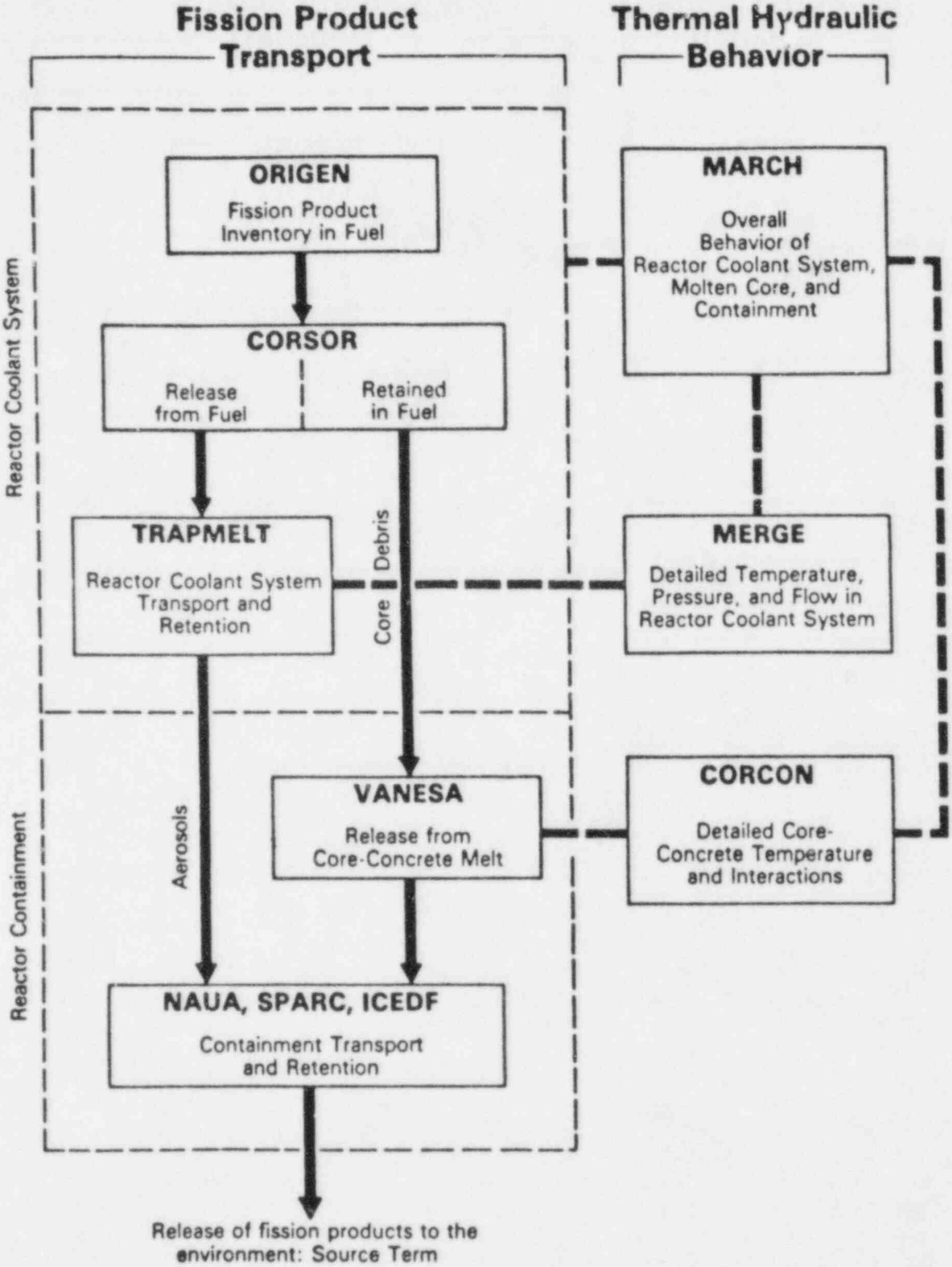


Figure 2.7 Source Term Codes

was investigating. The aim was to determine at an early date the areas of technical agreement, anticipating that any new regulations coming from severe accident research would involve a complex technical base. That research program is known as the Industry Degraded Core Rulemaking program (IDCOR).

2.5.3 Regulatory Impacts of the Research

Severe accident and source term research has identified "new" phenomena and errors in previously used source term values. For example, research identified the possibility that, during severe accidents initiated by a small break loss-of-coolant or a total loss of electric power (station blackout), high pressure ejection of molten materials from the core and consequent immediate overheating of the containment could occur. These findings and others have been used in developing a plan ("Implementation Plan for the Severe Accident Policy Statement and the Regulatory Use of New Source Term Information," February 28, 1986, SECY-86-76) that provides for the resolution of severe accident issues through the regulatory utilization (including both rule changes and regulatory practices) of improved information on source terms. Major area addressed in this plan include consideration of:

- Revised treatment of severe accidents in near-term Environmental Impact Statement;
- Removal of spray additives in PWRs (Standard Review Plan Section 6.5.2);
- Credit for fission product scrubbing in BWR suppression pools (Regulatory Guide 1.3, 10 CFR 100, and Standard Review Plan);
- Changes in emergency planning requirements;
- Allowable containment leak rates (10 CFR 50 Appendix J);
- Modifications of control room habitability requirement (Regulatory Guide 1.52 and Standard Review Plan Section 6.4)
- Environmental qualification of equipment (10 CFR 50.49 and Regulatory Guide 1.99);
- Prioritization of safety issues;
- Revision of siting criteria (10 CFR 100);
- Revisions of accident monitoring and management guidance (Regulatory Guide 1.97);
- Development of containment performance criteria; and
- Resolution of generic vulnerabilities.

Progress has been reviewed and information exchanged regularly with the industry program, IDCOR. Many areas of agreement have been established, and, just as important, areas of technical difference made precise. In addition, the technical work is undergoing a number of reviews by independent experts, and extensive public comment has been solicited and received. The sum of these efforts is to ensure that the technical basis for any rule changes is, and is widely perceived to be, on firm grounds. Such an achievement is considered a requirement for any such far-reaching change in rules, whether that change increases or decreases industry burden.

2.6 Seismic Hazards

2.6.1 The Problem

Earthquakes simultaneously challenge all safety systems. The sudden and severe ground motion that occurs during an earthquake may cause failures related to a design or construction error. Also, multiple failures or systems interactions may defeat the required redundancy and diversity in plant design.

The NRC has imposed increasingly stringent seismic design requirements on nuclear power plant structures and safety equipment. These requirements are well beyond those used in conventional earthquake construction, primarily because various mechanical and electrical systems, including piping, must function during and after the earthquake. Current regulatory criteria have encouraged the designers of nuclear power plants to construct very stiff piping systems with many pipe supports and snubbers, to resist seismic loads. These very stiff pipe systems, along with the unreliability of snubbers may, in fact, diminish the overall safety of the plant. More flexible piping systems with fewer snubbers would result in an improved balance in safety between operating and accident conditions and reduce radiation exposure to workers performing maintenance and inservice inspection.

Additionally, the recent position taken by the USGS that the Charleston, South Carolina earthquake may be considered to occur in a wider area than originally thought may require the reevaluation of plants for earthquakes larger than their original design basis. Thus it is important to estimate the reserve capacity or design margin of nuclear power plants to provide the technical basis for regulatory decisionmaking.

2.6.2 The Research Program

Research in this area includes these components:

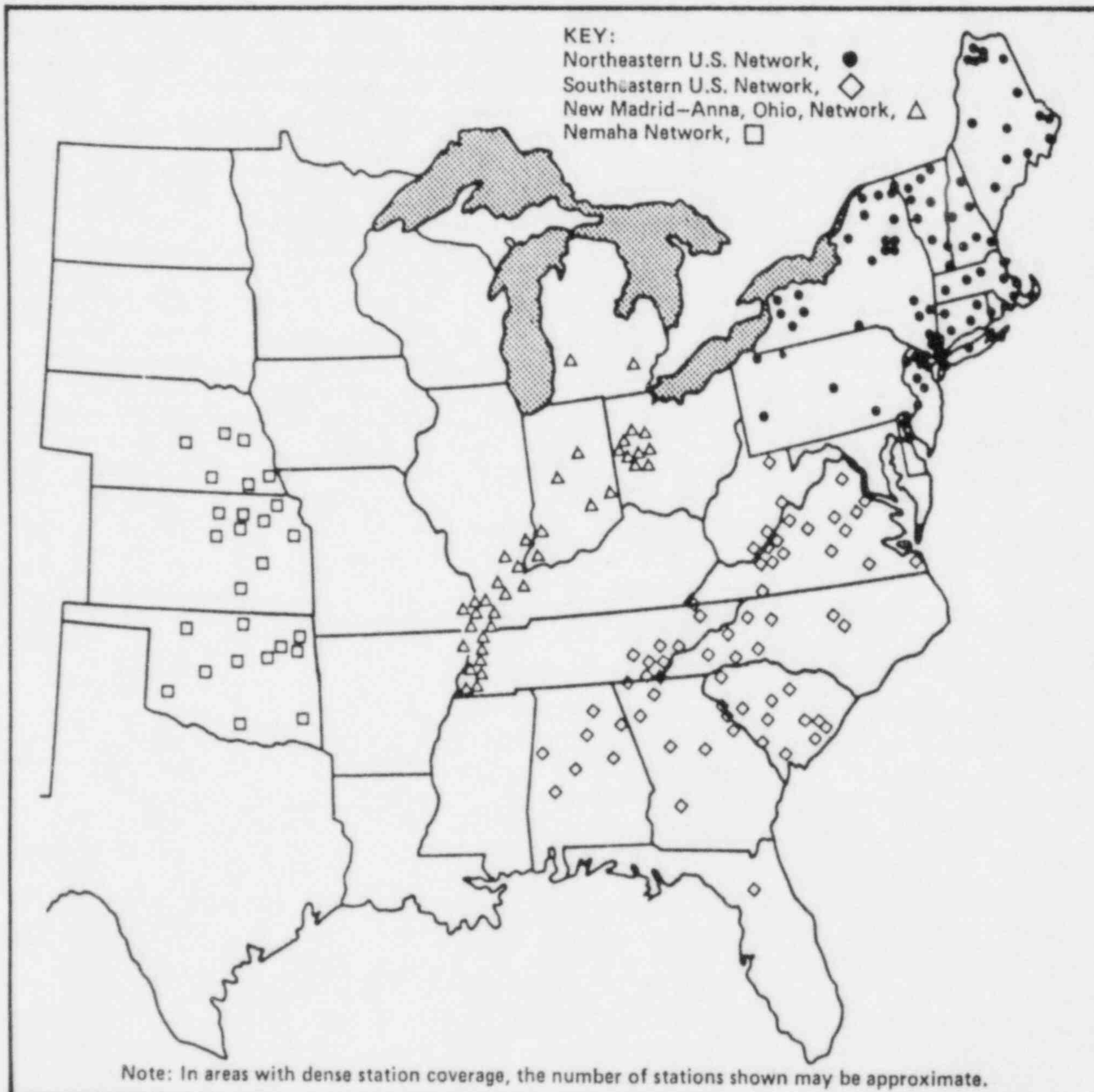
- Collection of data on actual earthquakes.
- Development of methodology to predict the response of structures and components to earthquakes,
- Testing, data collection and analysis to validate methodology,
- Testing of structures and components to determine their fragilities (failure levels).

In the Eastern and Central U.S., where large earthquakes are infrequent compared to the West, the understanding of earthquakes has been especially poor. In particular, the extent to which major earthquakes have been localized on a specific geologic structure has been questioned.

Thus, to determine whether the bases for some Eastern nuclear power plants were too low, the Office of Research has been collecting data about earthquake recurrence rate, energy, ground motion, and location through four eastern regional seismic networks. Operated in cooperation with universities and State and Federal agencies, these four networks include 240 stations and cover much of the Eastern and Central U.S. (See Figure 2.8.)

The other arm of the program is to develop a methodology to predict the response of safety structures, systems, and components at nuclear power plants to earthquakes, to validate that methodology, and to another establish at what levels important safety structures and components will fail. Much of this work requires use of large test facilities, called shaker tables, which induce loads similar to those generated by earthquakes. The latest effort in this regard is a cooperative program with Japan to use the large shaker table at their Todatsu facility to test our analysis methods.

This research will be applied to decisionmaking at two levels - first, to make a determination that seismic safety margins do or do not exist, and second (later in time), to better quantify those margins.



Regional earthquake monitoring networks in the Eastern and Central United States currently supported by the Nuclear Regulator Commission.

Figure 2.8 NRC Regional Seismic Networks

The NRC supports seismic research networks in specific regions where definition of seismic hazard needs improvement. The networks pinpoint earthquake locations, magnitudes, and other characteristics. This helps the regulator to set earthquake design standards.

2.6.3 Regulatory Impacts of the Research

One of the most important impacts of recent seismic research is the finding that certain early seismic design methods and assumptions thought to be conservative were actually decreasing safety. In particular, experiments on pipe systems have shown that realistic damping values are considerably higher than the values used in existing designs. Fracture mechanics and reliability models developed by the Office of Research demonstrate that flexible piping is usually more reliable than the currently required rigid or stiff piping. These results will, when validated, allow the use of more flexible piping and reduce the use of unreliable snubbers. These changes will actually increase seismic safety margins in some cases and permit considerable savings in maintenance costs to utility ratepayers while increasing public safety.

In addition, research on seismic risks has been used in a variety of licensing decisions and in developing regulatory guidance. In particular, it has been used in licensing decisions on the type and level of seismic resistance necessary for both new plants and for older plants being reviewed under the NRC's Systematic Evaluation Program (SEP), e.g., Diablo Canyon Seismic Reassessment, Byron/Braidwood Soil-Structure Interaction Study. In addition, seismic data collected from two recent earthquakes indicate that Regulatory Guide 1.60, "Seismic Design Spectra," which is based primarily on Western U.S. data may require some refinement for application to Eastern U.S. sites. The earthquakes appeared to have "higher frequencies" whose effects need further evaluation.

2.7 High-Level Waste Management

2.7.1 The Problem

The High-Level Waste (HLW) disposal policy for the U.S. is defined by the Nuclear Waste Policy Act, signed into law in 1983, which provides for the development of geologic repositories for the permanent disposal of high-level radioactive waste. Among its many provisions, the Act established schedules and funding mechanisms for repository development. Although the majority of actions required under the Act are to be performed by the Department of Energy (DOE), the Act provided the NRC with the responsibility of licensing these facilities. DOE must identify and characterize three sites by 1990. At that time, DOE is scheduled to submit a license application for construction authorization to the NRC.

A high-level waste repository poses unique considerations and uncertainties related to waste emplacement, monitoring, and performance characteristics. NRC must have an independent capability to evaluate DOE safety analyses and predict whether long-term releases will be in compliance with established regulations. The NRC research program is structured to provide this necessary technical capability to evaluate DOE's site characterization activities and to assess DOE's license application when it is submitted.

The NRC research program has provided the technical foundation for staff development of a set of regulations for the review and licensing of high-level waste repositories. This framework for NRC review has allowed the formal licensing activities and the supporting research to be focused on significant issues. The major components of the regulatory framework are Procedural Requirements and Technical Criteria for Licensing the Disposal of High-Level Waste.

2.7.2 The Research Program

The overall system performance standard issued by the Environmental Protection Agency for long-term (10,000 years) releases to the environment has required the development and application of predictive models of repository performance. Application of these models to various system components requires research on technical questions that are in the forefront of technology. The HLW research program has been developing an understanding of basic phenomena and processes of HLW geologic repositories in order to form a technical basis for independently assessing DOE license submittals for construction and operation.

HLW research includes research on the overall waste package that contains the radioactive materials and on the characteristics of the geologic repository media in which HLW is to be buried. Currently, major research and development efforts are centered on spent reactor fuel as the "waste form" and on tuff and salt as the repository media.

Model development, waste package component testing, and environmental studies performed for HLW disposal include:

- Experiments to determine the major ways in which waste packages fail,
- Evaluation of the ways in which rock materials affect movement of radioactive materials,
- Investigations of how long-term climatic conditions may affect the repository,
- Performance testing of methods of sealing access to the repository,
- Measurements of groundwater flow in fractured rock.

2.7.3 Regulatory Impacts of the Research

The major accomplishment of NRC research efforts to date has been the development of the technical criteria of the HLW disposal regulation in 10 CFR Part 60. That rule is now final, having first been subject to thorough public comment. These technical criteria set requirements for siting, design, and waste packaging as well as for overall system performance.

The High-Level Waste research program has also provided specific information for developing a regulatory assessment of the technical issues associated with licensing a HLW repository. These activities facilitate ongoing prelicensing consultations between NRC and DOE aimed at early identification and resolution of technical issues important to licensing. Examples are:

- Research results from the University of Arizona on a method to evaluate the water flow characteristics of proposed repository sites. This work contributed to the licensing generic technical position on hydrologic testing.
- Extensive research information from Lawrence Berkeley Labs, Oak Ridge National Laboratory, and the Australian Atomic Energy Commission on

the way in which the mineral composition and rock structure at various existing sites below the water table contributed to the licensing generic technical positon on how such rocks hold on to the waste.

Research on transport in the soil above the water table by the University of Arizona provided a special technical support document, "Disposal of High-Level Radioactive Wastes in the Unsaturated Zone: Technical Consideration and Response to Comments," NUREG-1046. This report provided the technical support for the amendment to 10 CFR Part 60 dealing with disposal of HLW in the soil above the water table.

3 CURRENT AND FUTURE RESEARCH DIRECTIONS IN SUPPORT OF REGULATION

3.1 Overall Research Directions

As the knowledge of nuclear power plant safety increases, new research is required to quantify safety margins, ensure that newly discovered phenomena do not reduce safety, and establish increased levels of safety. Research emphasis will change based on new research findings and industry initiatives and experience gained in the operation of nuclear power plants.

As this change occurs, more focus is expected to be given to applying knowledge gained from research programs to date to solve operational difficulties as they arise. Current analyses show that unexpected problems continue to occur at a fairly constant rate. In addition, because there have been no new nuclear power reactors ordered in the U.S. since 1978, there has been a deemphasis on research directed at the licensing of new reactors and a focus on research directed at regulatory and safety issues related to operating reactors.

The potential problems of aging plants and their implications for life extension of these plants are expected to become a growing research concern. As the first nation to have nuclear power, the U.S. also has some of the world's oldest operating reactors. With a continually increasing demand for electricity and the economic benefits from avoiding new construction, the NRC anticipates that utilities will be motivated to keep existing plants operating beyond their original design life. Therefore, the problems of aging and life extension will become significant research areas in the near future.

The research program will continue to be shaped by the lack of standardization in reactor designs in the U.S. In most cases, reactor designs from each vendor and, often, a number of different configurations of each reactor type, must be examined separately by the NRC in addressing safety concerns. The consequences of the lack of standardization have been particularly severe in thermal-hydraulic and severe-accident analyses, where the results are strongly dependent on plant design. Such analyses are an important and integral part of probabilistic risk assessments. Future work will need to continue to address safety concerns for different reactor types and configurations. This underscores the need to maintain a continuing U.S. experimental capability for analysis of operational events.

The general pressure to reduce Federal expenditures, which has led to budget reductions in the last few years, will increase the incentive for international collaboration efforts. From its inception, the NRC has participated in international research agreements. This cooperation has improved the quality of research by providing for shared insights, timely access to a broad range of knowledge and experience, and the cross fertilization needed to resolve the complex problems of nuclear safety. It has also reduced research costs to the U.S. Increasing efforts to share facilities, personnel, and perspectives with other countries are expected to allow the NRC to continue to respond to the questions of the future in a timely manner and at a lower cost to the taxpayer. International collaboration can thus partially offset the negative impacts of reduced budgets, but the ability of NRC to participate in international programs still requires that the U.S. maintain the ability to share expenses and to exchange research results from its own programs.

3.2 Specific Research

The Office of Research efforts are focused in six major areas where the NRC has current and continuing needs for research results. These areas are:

- Component Aging and Degradation. As U.S. plants have become more mature, a number of phenomena of corrosion, radiation embrittlement, fatigue, and other effects have been encountered that have raised serious questions about the integrity of the primary pressure boundary and hence about the continued safety and viability of the plants. These questions have led to a perception of a need to improve techniques of nondestructive examination and to improve the inspection of steam generators. Other problems of this kind may arise as plants age but have not yet been identified. Further, current indications are that utilities will wish to continue to operate existing plants beyond their design life to continue to meet demands for electric power. In order to regulate such plants, the NRC will need to decide how to test for degradation and overall performance in older plants through inspections and on-going surveillance programs and to ensure continuing safety regarding the replacement of selected old components and modifications in operations. Research is needed to ensure that, as they become older, operating nuclear plants continue to meet health and safety requirements.

- Transient Analysis. The continuing incidence of new and unexpected transients indicates an on-going need for regulatory actions. In order to be equipped to respond, the staff needs continued verified code improvements for applicability to different plant configurations and transients and for improving the capability to perform transient analyses more rapidly and easily. This is especially important because of the lack of standardization in designs among U.S. reactors. Each unexpected transient that occurs in one plant must be evaluated to determine how it may apply to other reactors that differ to various degrees in design and configuration. Thermal-hydraulic research, as well as research in several other areas, will also be needed to assess proposals from industry to improve capacity factors.

- Severe Accident Consequences. To reduce the large uncertainties in estimates of risk arising from severe accidents and to define sound methods of preventing or mitigating such accidents, research is needed in several areas. These include better definition of the release of fission products when the molten core reacts with the concrete basemat and the subsequent pressure and temperature loads on the containment. This knowledge will be most useful in defining proper and effective emergency response to a severe accident. To do this work successfully, it is necessary to know how the core melts down to and through the pressure vessel. Such knowledge should help develop accident management techniques that will keep an accident from progressing to the point that containment is endangered.

- Probabilistic Risk Analysis. There is a need for research to improve PRAs if they are to be fully useful. Needed research includes improvements in the data base to include component and human reliability, assessment of maintenance experience and operational feedback from operating plants, and incorporation of the insights and understandings that have developed from other NRC programs such as the thermal-hydraulic and source term reassessment programs. PRAs also need methodological improvement to incorporate failures that might arise from such events as earthquakes, fires, and internal and external flooding. These factors have been found to be major contributors to risk in some cases, so their incorporation into PRAs is important. Another much needed improvement in PRAs is to provide ways to reduce uncertainty in risk assessments or, at the least, to better understand the range and chief sources of uncertainty.

- Seismic Research. With the improvement in the understanding of geology and seismology in recent years, indications are that a number of the older plants in the Eastern and Central U.S. may not have been designed for high enough seismic hazard levels. Furthermore, probabilistic risk assessment studies have shown that seismic risk is significant. Thus, although there exists a considerable seismic safety margin in present designs, research is needed to develop improved, well-validated seismic analysis methods to reassess currently operating plants to ensure that adequate margins exist and to quantify these margins to some degree.

  Waste Disposal. Disposal of both low- and high-level waste involves issues related to waste form, waste package integrity, and the long-term retention of radionuclides in the disposal facility environment that are at the forefront of technology. Research is required to establish regulatory criteria to permit sound evaluation of proposals for disposal facilities and to ensure that all regulatory requirements, particularly those on radionuclide release limits, will be met.

Table 3-1 lists principal areas where changes might be expected to result from research efforts currently inplace or planned for 1986 and beyond.

TABLE 3-1

PROJECTED AREAS OF REGULATION AFFECTED BY CURRENTLY PLANNED RESEARCH, 1986-89, AND SOME KEY PRODUCTS

Pressure Vessel Integrity:

Pressurized Thermal Shock

Validated fracture analysis methodology
Fracture toughness and crack arrest of irradiated vessel steel and weld metal

Piping

Validated methodology for load capacity of flawed and degraded piping
Validation of leak-before-break concept
Data on actual failure modes
Evaluation of aging and degradation in LWR materials

Emergency Core Cooling Systems

Modifications to allow accident fuel clad temperatures to be realistically calculated

Hydrogen and Containment

Completed technical basis

Severe Accidents

Emergency planning regulation, graded offsite response
Control room habitability
Containment performance requirement
Accident management and monitoring

Seismic Hazards

Rule for Eastern U.S. seismic zones
Information base for developing site-specific spectra
Methods for handling uncertainties in risk assessment from seismic hazard
Validation of engineering analysis techniques used in assessment of seismic risk

Equipment Qualification

Evaluation of methods for qualifying equipment for design basis events and for assessing performance beyond the design basis

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GLOSSARY

GLOSSARY

AEC	Atomic Energy Commission. The independent civilian agency of the federal government which had statutory responsibility for atomic energy matters prior to the creation of the Nuclear Regulatory Commission.
Aerosol	A suspension of particles contained in a gas. When the particles are very small, they can remain suspended for a long time (hours).
Anticipated Transient Without Scram (ATWS)	An improbable nuclear power plant occurrence that assumes complete failure of the plant to scram (see Scram) when scram is called for, and proceeds until other action is taken either automatically or by an operator.
ASME	American Society of Mechanical Engineers. A professional society which develops standards in many engineering areas, including the nuclear area.
BWR	Boiling water reactor. A nuclear reactor in which water is used as both a coolant and a moderator. It is allowed to boil in the core, and the resulting steam is used directly to drive a turbine and then condensed and returned to the reactor vessel.
10 CFR	Title 10 to the Code of Federal Regulations. Title 10, "Energy" is composed of four volumes. The first volume, Parts 0-199, contains the regulations of the Nuclear Regulatory Commission.
Containment	An enclosure around a nuclear reactor to retain radioactive materials that otherwise might be released to the atmosphere in the event of an accident.
Control Rod	A rod, plate, or tube containing a strong neutron absorbing material (hafnium, boron, cadmium, etc.) used to establish or eliminate nuclear criticality. A control rod absorbs neutrons, preventing them from causing further fissions.
Coolant	A fluid that is circulated through a nuclear reactor to remove heat. Water is used in most U.S. power reactors.
Core	The central portion of a nuclear reactor containing the fuel elements, moderator, neutron poisons, and support structures. The volume where fission occurs, producing heat.

Core Melt	This term applies to the overheating of a reactor core as a result of the failure of reactor shutdown or cooling systems leading to substantial melting of the radioactive fuel and the structures that hold the fuel in place.
Decommissioning	Steps taken after permanent shutdown of a reactor to ensure continued safety and noncontamination of the environment.
Defense in depth	An engineering practice involving multiple types of protection against accidents. It includes quality assurance and control in plant design, construction, and operation; backup systems; engineered safety features to confine the consequences of accidents; siting; and emergency planning.
Deflagration	A combustion wave that is traveling at a speed that is subsonic relative to the unburned gas. Most fires are of this type.
Design Basis	A requirement that limits the design of a facility, system, or item of equipment.
Design Basis Accident	The most serious reactor accident for which the system is designed.
Detonation	A combustion wave that is traveling at a speed that is supersonic relative to the unburned gas. Associated with an explosion.
ECCS	Emergency core cooling system. A backup system designed to keep the nuclear fuel cooled in the event that normal coolant is lost.
Equipment Qualification	Process by which the capability of equipment to survive and function during design basis events is demonstrated.
EPRI	Electric Power Research Institute. A research organization owned and operated by a consortium of U.S. electric power utilities. EPRI conducts nuclear and non-nuclear research.
Failure	The inability of a system or component to perform its intended function.
Fission	The splitting of a heavy nucleus into two or more parts (which are nuclei of lighter elements) accompanied by the release of a relatively large amount of energy and frequently two or more neutrons. Fission can occur spontaneously, but usually it is caused by the absorption of neutrons, gamma rays, or other particles.

Fission Products	Nuclei formed by the fission of heavy elements. Almost all are radioactive.
Fuel	Fissionable material used, or usable, to produce energy in a reactor. The term is also applied to a mixture such as natural uranium, in which only part of the atoms are readily fissionable, provided the mixture can be made to sustain a chain reaction.
Fuel Cladding	The outer jacket of nuclear fuel elements. It provides structural support of the fuel and prevents the release of fission products into the coolant. Aluminum, stainless steel, and zirconium alloys are typical cladding materials. Zirconium is the currently preferred material.
HLW	High-Level Waste (see radioactive waste).
IDCOR	Industry degraded core rulemaking program.
IGSCC	Intergranular stress-corrosion cracking. Corrosion cracking in BWR piping caused by high local stress, sensitization of material, and high oxygen content in water. The corrosion occurs along the boundaries between different grains.
LLW	Low-Level Waste (see radioactive waste).
LOCA	Loss-of-coolant accident. A LOCA can result from breaks in pipes carrying the coolant or leaks from valves that control its flow.
LWR	Light Water Reactor. A nuclear reactor in which ordinary water is used as the coolant. PWRs and BWRs are both LWRs.
NRC	Nuclear Regulatory Commission. The Federal Agency responsible for the regulating, licensing, and inspecting of nuclear facilities and materials and for conducting the research necessary to perform these functions.
NRR	(Office of) Nuclear Reactor Regulation, NRC. The entity within the NRC responsible for licensing nuclear reactors.
Nuclide	A species of atom that can be distinguished by its atomic weight, atomic number, and energy state. A radionuclide is a radioactive nuclide.
Pipe Whip	Severe movement of broken piping arising from reaction to water flow changes, similar to an uncontrolled garden hose whipping around.

Pressure Vessel	A strong-walled container that houses the core of most power reactors.
Pressurized Thermal Shock	A phenomenon of cracking under conditions of rapid cooling at high pressure (which may occur in some accident sequences); it may occur in certain pressure vessels embrittled from years of neutron radiation, depending on the material composition.
Pressurized Water Reactor (PWR)	A power reactor in which heat is transferred from the core to a boiler by water kept under high pressure to achieve high temperature and prevent boiling in the core. Steam is generated in the boiler by water separated by tubes from the water leaving the core.
Probabilistic Risk Assessment or Probabilistic Risk Analysis (PRA)	The art of mathematically quantifying an average risk based on observed and calculated component and human failure rates and the anticipated consequences associated with these failures, which may occur either singly or in combination. Probabilistic risk assessment typically involves the use of event trees and fault trees, although these are not the only tools available for such assessments.
Radiation	The propagation of energy through matter or space in the form of waves. Includes particles (alpha and beta rays, free neutrons, etc.), and electromagnetic radiation (Gamma and X-rays).
Radiation Dosimetry	The measurement of the amount of radiation delivered to or absorbed at a specific place.
Radioactive Waste	Equipment and materials (from nuclear operations) that are radioactive and that are being discarded. Wastes are generally referred to as high level, having radioactivity concentrations of hundreds to thousands of curies per gallon or cubic foot such as spent nuclear fuel or low level, in the range of 1 microcurie per gallon or cubic foot, which can include contaminated clothing or medical wastes.
Regulatory Guide	An NRC publication that is used to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, and otherwise to provide guidance to applicants. Regulatory Guides are not NRC requirements.
RES	(Office of) Nuclear Regulatory Research, NRC. The entity within NRC responsible for conducting research needed to support the regulatory process.

Risk
: The product of the probability of occurrence of an accident and the magnitude of the consequences given that occurrence

Scram
: The sudden shutdown of a nuclear reactor, usually by rapid insertion of safety rods and control rods. Emergencies or deviations from normal reactor operation cause reactor operator or automatic control equipment to trip the reactor.

Seismic
: Pertaining to an earthquake.

Seismicity
: Relationship of the frequency and distribution of earthquakes.

SEP
: Systematic Evaluation Program. A program in which 8 of the older operating plants were evaluated against the "intent" of the current licensing criteria for selected issues.

Severe Accident
: An incident in a nuclear reactor which continues to the point that cladding is damaged and fission products are released from the fuel rods. Radioactivity may or may not be released outside the reactor.

Snubber
: A shock absorber attached to piping to lessen the effects of vibration or movement.

Thermal-Hydraulic
: Pertaining to fluid movement and heat transfer within the reactor system.

Transient
: A malfunction or deviation from normal reactor operation.

Waste Form
: The radioactive waste material and any encapsulating or stabilizing matrix (such as spent reactor UO_2 fuel and borosilicate glass).

Waste Package
: The waste form and any other containers, shielding, packing, overpacking, and other absorbent materials immediately surrounding an individual waste container.

Water Hammer
: A physical phenomenon, usually a loud banging noise, that occurs when water flows are suddenly interrupted. Water hammers often exert large forces on pipes.

Water Impingement
: The impact of water or steam under high pressure striking another piece of equipment following a pipe break.

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