
Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States

Final Report

Main Report

**U.S. Nuclear Regulatory
Commission**



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ABSTRACT

This report was prepared by the Nuclear Regulatory Commission (NRC) staff to assess the implications of the accident at the Chernobyl nuclear power plant as they relate to reactor safety regulation for commercial nuclear power plants in the United States. The facts used in this assessment have been drawn from the U.S. fact-finding report (NUREG-1250) and its sources.

This report consists of two volumes: Volume I, Main Report, and Volume II, Appendix - Public Comments and Their Disposition.

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INTRODUCTION

This report was prepared by the staff of the U.S. Nuclear Regulatory Commission (NRC) to assess the implications of the April 1986 Chernobyl accident in the Soviet Union as they relate to commercial nuclear reactor safety regulation in the United States. Most of the assessment focuses on light-water-reactor power plants. A final chapter addresses graphite-moderated reactors.

With respect to studying the Chernobyl accident, U.S. Government agencies have expended their energies on determining the facts, as well as on assessing those facts in terms of how the accident may affect U.S. policies and practices in the nuclear power field.

The work was divided into two major phases. The first phase, fact finding, was a coordinated effort among several U.S. Government agencies and some private groups; this phase was completed in January 1987 and has been reported in NUREG-1250, "Report on the Accident at the Chernobyl Nuclear Power Station." The second phase, an assessment of the implications of that accident with regard to U.S. policies and practices, is being pursued separately by each organization that participated in NUREG-1250. The present report, as part of this second phase, addresses the safety regulation of commercial nuclear reactors under NRC regulatory jurisdiction. (Department of Energy reactors, not subject to NRC regulation, are not addressed in this NRC study.)

In developing the assessments presented in this report (NUREG-1251), the NRC staff depended on NUREG-1250 and its two major source documents (USSR, 1986; INSAG, 1986) for the facts of the Chernobyl accident. The Soviet document (USSR, 1986) is an official Soviet report to the International Atomic Energy Agency (IAEA) Experts' Meeting held in Vienna August 25-29, 1986; the second (INSAG, 1986) is the report to the IAEA prepared by the International Nuclear Safety Advisory Group at a second meeting in Vienna on August 30 to September 5, 1986.

The assessment of the implications of the Chernobyl accident with regard to commercial nuclear reactor safety regulation in the United States is supported by detailed assessments of a number of particular issues, grouped in six subject areas. The particular issues selected for evaluation were those that are associated with significant factors that led to or exacerbated the consequences of the Chernobyl accident.

A draft of this report was issued for public comment in September 1987. The comments received, together with further work within the NRC, were taken into account in preparing this final version. The passages that have been changed (except for those with minor editorial changes, such as the spelling out of acronyms) are marked by vertical lines in the margin. A separately bound appendix to this report contains the comments received, provides the staff's response to significant issues raised in the comments, and identifies the nature and basis of the resultant changes to the draft report. The changes correct or clarify specific items of information and modify assessments in some areas pertaining to specific issues; they do not substantially change the major aspects of the assessment.

SUMMARY

General Conclusions

A study of the Chernobyl accident has led the NRC staff to the following general conclusions about its effect on safety regulation of commercial nuclear power plants in the United States:

- (1) No immediate changes are needed in the NRC's regulations regarding the design or operation of U.S. commercial nuclear reactors.

Nuclear design, shutdown margin, containment, and operational controls at U.S. reactors protect them against a combination of lapses such as those experienced at Chernobyl. Although the NRC has always acknowledged the possibility of major accidents, its regulatory requirements provide adequate protection against the risks, subject to continuing vigilance for any new information that may suggest particular weaknesses, and also subject to taking measures to secure compliance with the requirements. Assessments in the light of Chernobyl have indicated that the causes of the accident have been largely anticipated and accommodated for commercial U.S. reactor designs.

Yet, the Chernobyl accident has lessons for us. The most important lesson is that it reminds us of the continuing importance of safe design in both concept and implementation, of operational controls, of competence and motivation of plant management and operating staff to operate in strict compliance with controls, and of backup features of defense in depth against potential accidents.

Although a large nuclear power plant accident somewhere in the United States is unlikely because of design and operational features, we cannot relax the care and vigilance that have made it so. Accordingly, further consideration of certain issues is recommended, as discussed.

- (2) Some aspects of requirements and regulations that already exist or are being developed will be reexamined, taking into account the accident at Chernobyl.

Areas that may warrant further study include operator training, emergency planning, and containment performance.

- (3) Study of areas related to certain aspects of the Chernobyl accident will be extended and will provide a basis for confirming or changing existing regulations.

These areas include reactivity accidents, accidents at low power or at zero power (when the reactor is shut down), and characteristics of radionuclide release.

- (4) The Chernobyl experience should remain as part of the background information to be taken into account when dealing with reactor safety issues in the future.

Conclusions About Specific Areas

The accident at Chernobyl suggests that the following specific areas be examined in direct response to that event. (Cross-references in parentheses refer to correspondingly numbered detailed assessments in the body of this report.)

(1) Administrative Controls Over Reactor Operations (Chapter 1)

In general, regulatory provisions at nuclear plants in the United States, if properly implemented, are adequate with respect to administrative controls to ensure that reactor operations are conducted within a safe range of operating conditions. These controls address procedural adequacy and compliance, approval of tests and other unusual operations, bypassing of safety systems, availability of engineered safety features, operating staff attitudes toward safety, management systems, and accident management.

However, the benefits of the following additional provisions should be examined:

- (a) Programs for accident management, including training and the development of procedures for coping with severe core damage and for the effective management of the containment. This provision will be addressed and resolved as part of the implementation of the Commission's Severe Accident Policy.
- (b) The review of administrative controls to seek ways of strengthening technical reviews and the approval of changes, tests, and experiments.
- (c) The review of safety system status displays and the availability of engineered safety features for potential worthwhile improvements.
- (d) The review of current NRC testing requirements for balancing benefits versus risks.
- (e) Measures that might further increase assurance that violations of procedures that could be instrumental in causing an accident or emergency situation or compromising safety margins will not occur.

(2) Reactivity Accidents (Section 2.1)

Positive void reactivity coefficients, which are a characteristic of the RBMK graphite-moderated water-cooled reactors, played a central role in determining the severity of the Chernobyl accident. Commercial reactors in the United States are designed very differently from the RBMK reactor at Chernobyl, and have generally a negative void reactivity coefficient. This provides assurance that the kind of superprompt critical excursion that took place at Chernobyl will not occur. However, the NRC should reconfirm that vulnerabilities and risks from possible accident sequences have been adequately factored into safety analysis reports on which design approvals are based.

(3) Accidents at Low Power and at Zero Power (Shutdown) (Section 2.2)

Regulations for commercial nuclear power plants in the United States require that potential accidents that could occur during all conditions of operation (full, low, and zero power) be considered and provided for in the plant design. Such provisions are considered in safety analyses required in support of licensing. Often, analyses assuming full-power operation are found to be limiting cases--bounding accident risks at low-power operation or when the reactor is shut down. The Chernobyl accident suggests that accident sequences beginning at low power and under shutdown conditions should be reviewed, particularly for situations in which not all engineered safety features are considered necessary to be available.

(4) Multiple-Unit Protection (Section 2.3)

For multiple-unit plants that are operating or are under construction, the Chernobyl experience should be considered in assessing the adequacy of protection of control rooms in the event of an accident at one of the units. This assessment should be performed on the basis of recent research information on radionuclide release.

New multiple-unit plants should not share systems required for shutting down each unit unless designed to enhance the overall level of safety.

(5) Fires (Section 2.4)

Provisions for fighting fires when radiation levels are high should be reviewed to confirm that the current provisions are adequate.

(6) Containment (Chapter 3)

The Chernobyl accident demonstrated the importance of containment performance for mitigation of the risks of nuclear power plant operation. Even before the Chernobyl accident, research programs and regulatory initiatives in the United States addressed the issue of containment performance during severe accidents. A systematic search for plant-specific vulnerabilities (i.e., potential failures that result in unacceptably high risk) is scheduled to begin in 1988, as part of the implementation of the Commission's Severe Accident Policy. This search will include reviews of containment design. The Chernobyl experience should be taken into account in these reviews wherever that experience is relevant.

Filtered venting of containment as a means of limiting offsite consequences of core-melt accidents is being pursued in a number of countries and is being examined in the United States. Anticipated international technical exchanges will enhance U.S. research and evaluation efforts concerning this potential measure.

(7) Emergency Planning (Chapter 4)

Partly because the radionuclide release in the Chernobyl accident is specific to the RBMK design, the size of the 10-mile plume exposure pathway emergency planning zone, which specifically includes the concept of

protective actions outside it if necessary, continues to be viewed as adequate. However, in light of new research information (NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms," and NUREG-1150, "Reactor Risk Reference Document"), the planning bases for relocation and decontamination and for protective measures for the food ingestion pathway are being reexamined in cooperation with the Federal Emergency Management Agency.

(8) Severe-Accident Phenomena (Chapter 5)

The phenomena of the Chernobyl accident were greatly influenced by the design features and materials in the RBMK reactor, which differ in many respects from those of U.S. reactors. The only radionuclide release aspects identified to date that are not currently considered in U.S. analytical models involve two mechanisms of fission-product release from fuel debris. These are mechanical dispersal and chemical stripping (removal of the fuel surface layer, as through chemical change of the uranium oxide). Although it is not clear that these mechanisms will have any effect on accident sequences relevant to U.S. reactors, it is recommended that the need for additional research be assessed.

(9) Graphite-Moderated Reactors (Chapter 6)

The Fort St. Vrain high-temperature gas-cooled reactor (HTGR) is the only licensed and operating commercial graphite-moderated reactor in the United States. A study of the potential for a Chernobyl-type fire and explosion at Fort St. Vrain was initiated immediately after the Chernobyl accident. It should be noted, however, that the licensee for Fort St. Vrain, the Public Service Company of Colorado, has notified the NRC that it will discontinue operations on or before June 30, 1990.

Although the only shared features between the HTGR concept and the Chernobyl design are the use of a graphite moderator and gravity-driven control rods, the 330-MWe Fort St. Vrain HTGR and a proposed modular HTGR concept were reviewed against the Chernobyl candidate issues and the conclusions presented in this document for light-water reactors. This assessment confirms that the concept of the HTGR (because it uses helium coolant in a fully ceramic core, has an overall negative reactivity coefficient, and has completely diverse alternate shutdown and cooling systems) has no direct association with the identified weaknesses of the Chernobyl design. In the areas at issue of operations, design, containment, emergency planning, and severe-accident phenomena, NRC assessments conclude that the implications of the accident at Chernobyl generate no new licensing concerns for HTGRs and both the overall and specific-area conclusions are the same as for light-water reactors. The assessment did not raise any new concerns regarding HTGR severe-accident phenomena. It did reinforce the desirability of undertaking a limited probabilistic risk assessment of Fort St. Vrain. It also suggested consideration of the merits of the possible reinitiation of experiments in graphite thermal stress to enhance confidence in the long-term integrity of the Fort St. Vrain structural graphite. However, no work with respect to Fort St. Vrain is now warranted, in view of the imminent termination of operations.

CHAPTER 1

ADMINISTRATIVE CONTROLS AND OPERATIONAL PRACTICES

In the United States, administrative controls over plant operations include NRC rules and regulations, facility license conditions, Technical Specifications, and plant procedures. The overall administrative control framework requires that safety-related activities at nuclear power plants be conducted in accordance with approved written procedures. These activities include, for example, operations, tests, inspections, calibrations, maintenance, experiments, modifications, safety review and approval functions, and audits. The safety design basis of the plant is based on assumed initial conditions for transients and emergencies. These assumed initial conditions (e.g., temperatures, pressures, control rod positions, and equipment availability) establish a "safe operating envelope." Effective administrative controls are needed to ensure that reactor operations are conducted within this safe operating envelope. Clearly, for administrative controls to be effective they must be technically accurate and complete, they must be understood by those responsible for implementing specific procedures, and management must ensure that they are enforced. A key finding from the Chernobyl accident is that such administrative controls in place at Chernobyl were not effective in maintaining conditions within the safe operating envelope.

In this chapter, the NRC staff reviews the administrative controls over plant operations in the United States to determine if adequate controls are in place to maintain plant conditions within the safe operating envelope. This review includes an assessment of procedural adequacy and compliance, approval of tests, bypassing of safety systems, availability of engineered safety features, operating staff attitudes toward safety, management systems, and accident management. The results of these detailed reviews are reported in the following sections. The staff confirmed that some ongoing activities with a nexus to the Chernobyl accident should continue. In addition, a few new issues requiring staff attention were identified and are presented below.

Emergency operating procedures (EOPs) are intended to ensure safe shutdown and to mitigate the effects of accidents and transients. Facility EOPs are designed for coping with accidents and transients that initiate from within the safe operating envelope. The ability of operators to successfully implement EOPs depends upon plant safety parameters initially being within the safe operating envelope. As a result of the Three Mile Island accident, NRC required that new symptom-based EOPs be developed. These new procedures have not been implemented at all facilities, and NRC audits have identified deficiencies in implementation at several facilities. Thus, licensees must expend significant effort to complete implementation of new EOPs.

Operator training needs to stress fundamentals of reactor safety, how the plant should function, and the underlying danger if plant conditions move outside the safe operating envelope. With adequate training and knowing the possible

consequences, personnel would be less likely to succumb to pressures to speed up, take shortcuts, or defeat safety functions. Operating experience and the Chernobyl event suggest that additional attention to training in the areas of maintenance of safety parameters and plant conditions within the safe operating envelope, emergency operating procedures, and accident management should be considered.

The Chernobyl accident has emphasized the need for contingency planning assuming core damage has occurred to ensure that appropriate controls, training, and planning have prepared the plant staff to manage plant assessment activities, response actions, and emergency actions. Significant effort has been expended to prepare for events involving degraded-core cooling and to upgrade emergency planning. However, more work needs to be done in training and procedure development for coping with severe core damage and for effective management of containment.

Management attention and diligence are required to ensure that plant operations, testing, and maintenance are conducted within the safe operating envelope. Management must focus on ensuring that all of the administrative control systems are effective and enforced. To obtain feedback on the quality of safety activities, the operating staffs must continue to perform audits, internal inspections, and reviews of operating data and events. Qualified and informed individuals must control reviews of changes, tests, and procedures. Experience has shown that some of these reviews have not been of consistently high quality and, in some instances, design changes have been made and testing has been conducted that place the plant outside the safe operating envelope. Industry has acted to improve the review process required by NRC; however, more needs to be done to sharpen the focus on responsibility for safety.

1.1 Administrative Controls To Ensure That Procedures Are Followed and That Procedures Are Adequate

Are controls at U.S. reactors adequate to ensure that operations and other activities at nuclear power plants are performed in accordance with approved written procedures?

When, in order to complete the test, the operators deviated from the approved test procedures and the established administrative procedures, they initiated the Chernobyl accident. Although the test procedure called for the test to be run at 700 to 1000 MWt, the operators could only achieve 200 MWt, but decided to conduct the test anyway. In addition, they violated the fundamental administrative requirement to maintain enough control rods at the proper degree of insertion to be effective in an automatic scram. The operators should not have raised the control rods beyond their administrative limits so that the reserve shutdown reactivity margin limits were violated; they should have terminated the test and shut the reactor down. This violation resulted in the inability to insert enough negative reactivity in the required time by a scram to overcome certain reactivity transients.

The operators violated another administrative procedural limit when they activated and operated two additional main circulating pumps while the other main circulating pumps were running. Such actions (1) violated limits protecting against pump cavitation damage and (2) yielded an abnormally high core flow rate.

The conditions created by running all of the main circulating pumps would also have caused an automatic scram if the operators had not intervened and defeated the scram function. Subsequent operation with the high flow rate resulted in voids being swept from the fuel element channels. This caused a large reactivity loss which was compensated for by control rod withdrawal to an extent that the rods were initially less effective when scrammed.

Other deviations from administrative procedures occurred, such as bypassing safety systems. These are discussed separately. Such deviations and procedures violations are influenced by operator attitudes (also discussed separately). This issue concerns (1) controls by licensees and regulators to ensure that procedures are appropriately written, known to the operators, placed at the work-site, and followed and (2) the adequacy of these controls for some safety functions. Such controls involve plant policies and procedures, industry standards, and regulatory rules and enforcement policy. The specific administrative controls applicable to changes, tests, and experiments are provided in Section 1.2.

1.1.1 Current Regulatory Practice

(1) NRC Requirements and Guidance for Procedure Development and Use

The NRC has a large body of guidance and requirements that includes general and specific measures for development and use of administrative procedures and controls. These controls govern all operating activities at nuclear power plants, and are designed to avoid the types of violations that occurred at Chernobyl. Violations of procedures do occur at licensed plants, but in relation to the number of procedural steps taken at plants, such violations are infrequent, and only rarely do they occur with the knowledge that a violation is being committed. Errors have also been committed because of operator failure to use or refer to procedures. In its program to ensure safety and quality, the NRC has developed and published quality assurance requirements for activities affecting nuclear safety. Criterion V, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50) governing procedures states:

V. Instructions, Procedures and Drawings

Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

This criterion prescribes the general requirement for having procedures and for following them. A second level of administrative controls for procedures is contained in each plant's Technical Specifications, which are a part of the license. Plant Technical Specifications require licensees to establish, implement, and maintain procedures. Both Technical Specifications and Criterion V have the force of law.

Technical Specifications require procedures to be reviewed by the Unit Review Group when initially written and before being changed, except for temporary

changes made on the spot that do not alter the intent. The Unit Review Group is made up of key plant supervisory personnel who are knowledgeable about plant safety. The objective of this review is to ensure that experts from the various technical disciplines review the procedures for operations or changes that could affect safety. This review backs up the technical procedure writer and his/her supervisor's decisions on safety. There is a further screening of procedures and changes to procedures to determine whether or not they may involve an unreviewed safety question or a technical specification, in which case prior NRC approval is required by 10 CFR 50.59. The NRC requires that all of these activities, including compliance with procedures, be periodically audited, and audit results be provided to appropriate management; corrective action is required when deficiencies are found.

(2) Required Procedure Coverage

Technical Specifications require that licensees commit to develop and implement applicable procedures listed in Appendix A to Regulatory Guide 1.33, "Quality Assurance Program Requirements Operation." Licensees make this commitment in their applications. This list of applicable procedures covers essentially all operating and administrative activities (e.g., startup, shutdown, refueling) and requires the development of specific procedures for activities, such as tests and maintenance, at the approximate time but before the test or maintenance activity is performed. Test and administrative procedures undergo the same review as other procedures.

(3) Guidance in Standards

Additional guidance on procedures is provided in American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 3.2-1980, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." The guidelines of this standard provide much more detail than other documents on the measures needed for the development, review, control of changes, and implementation of the procedures. This standard is endorsed by the NRC through Regulatory Guide 1.33, and licensees have committed to comply with Regulatory Guide 1.33 in their license applications.

ANSI/ANS 3.2 requires that procedures be written for all plant safety activities, that they be followed, and that the requirements for use of the procedures be prescribed in writing. It further requires written guidance for operators to contain elements describing when a procedure is to be memorized, when it is to be in hand while the operator is conducting the operation, and when signoffs are required. It identifies situations in which temporary changes can be made and the conditions under which such changes can be made if proper controls are met.

(4) Training on Procedures

Operators must be licensed by the NRC. Since plant operation requires extensive use of procedures, operators are trained in both the technical details of procedures and what is expected of them in terms of using procedures and following procedural provisions. The NRC examines operators in these areas.

(5) NRC Inspection and Enforcement

Important elements in the overall regulation of nuclear power plants are the inspection of licensee activities and the enforcement actions taken when the licensee fails to comply with NRC requirements.

Since a requirement exists in the Technical Specifications that licensees follow procedures, licensed operators must use procedures and must abide by them or face possible disciplinary action from their own management and possible enforcement action by the NRC. Citations and significant fines have been imposed on utilities for such violations of procedures. Licensees' activities are inspected routinely and after each significant event to determine compliance with procedural requirements. These inspections are often done unannounced on backshift and during weekend periods. More severe actions are usually taken for violations of procedures if the act has been willfully performed. Operators are very reluctant to deliberately commit such acts. In an emergency, a licensee is permitted through 10 CFR 50.54(x) to deviate from a procedure or even from a technical specification if the licensed operator determines such deviation is needed to protect the public. When appropriate (e.g., as a result of decreasing Systematic Assessment of Licensee Performance ratings), additional emphasis will be placed on inspectors monitoring the quality and use of procedures.

1.1.2 Work in Progress

(1) Technical Specifications Improvements

The NRC has a priority effort under way to improve Technical Specifications through the Technical Specification Improvement Program. Current Technical Specifications have grown in volume because of lack of guidance on which requirements should be included in them. A Policy Statement defining the scope and purpose of Technical Specifications (52 FR 3788) has been approved by the Commission. Technical Specifications that have been revised in accordance with this Policy Statement will be more closely oriented toward the operator's job and will be rewritten to improve clarity. Bases for requirements will be improved. Technical Specifications that appear in procedures will be easier to understand and to follow.

(2) Symptom/Function-Based Emergency Operating Procedures

One of the lessons learned from the accident at Three Mile Island Unit 2 was the need for symptom/function-based emergency operating procedures (EOPs) for coping with transients and accidents. The NRC has a program in place that is sponsored by vendor owners groups to develop EOPs based on reanalyses of transients and accidents. All licensees are required to implement symptom-based EOPs incorporating good human factors practices. Operators are receiving training on these procedures. The ability of operators to successfully implement these procedures is directly related to their knowledge of whether or not the plant is initially operating within the safe operating envelope.

(3) Refocusing NRC Inspection Activities

The NRC initiated an inspection program to reward good licensee performance by reducing inspections for good performers; below-average performers were inspected more frequently.

In the staff's judgment, a high level of overall compliance and a high level of compliance with procedures go hand in hand. To achieve the coveted high performance rating, licensees will need to have (a) effective administrative controls over procedure development and use as well as (b) good performance in other management and technical areas.

1.1.3 Assessment

Good administrative controls are essential for the safe operation of nuclear power plants. The staff has carefully examined these controls. The assessment of the adequacy of these controls at U.S. reactors is discussed below.

Over the past 15 years, a body of American Nuclear Society standards has been developed and put into place to provide criteria and guidance for procedures and for controls over the procedures. Several key standards have been in use for much of this period; furthermore, these key standards have been revised and refined, becoming effective standards. They address administrative controls, qualifications for nuclear power plant personnel, training, and quality assurance. The NRC has encouraged such standards development, endorsing it through the NRC regulatory guide series. The standards have become the recommended and accepted programs in their respective areas.

These standards contain excellent requirements and guidance for control over administrative and technical procedures. They are geared toward ensuring that technically sound procedures are developed that have been reviewed by a multi-discipline review body, and that have management endorsement and authorization. They also require the use of approved written procedures for essentially all activities at the plants. Required training emphasizes how these procedures are to be used and followed. Management directives and administrative procedures state the philosophy and expectations, i.e., procedures will be written and followed.

The NRC has published guidance and has issued plant-specific Technical Specifications stating requirements in the use of procedures. Although these procedures and specifications allow removal of a single train of redundant systems for test or repair, they prohibit defeating safety systems and prescribe minimum operability requirements for important safety equipment. NRC personnel inspect procedural activities and take enforcement action, when appropriate, against utilities and licensed operators who violate these requirements. The industry-sponsored Institute of Nuclear Power Operations evaluates performance in these same areas and strives for excellence in writing, use, and control of procedures through its evaluation feedback process to management.

Although the staff recognizes that errors and violations will occur, the measures taken by the NRC and industry should keep violations to a minimum. Since Technical Specifications containing the operability requirements for safety equipment are so prominent in operators' and management's minds, the staff believes that operators, because of their concern for safety, will not willingly violate these requirements and put the reactor in jeopardy. It should be recognized, however, that since violations of procedures do nevertheless occur, a study that would characterize the nature, severity, and frequency of violations could be of value. It might provide a firmer basis for a reassuring conclusion or lead to a consideration of additional means of reducing inadvertent violations and deterring willful ones.

Recent audits by the NRC have identified deficiencies in the implementation of the new symptom-based emergency operating procedures (EOPs.) In addition, NRC examinations have identified the need for additional training on the use of these procedures. Therefore, the staff believes work should continue to achieve full implementation of the new EOPs and to provide associated training to operating personnel. Furthermore, the staff believes that the concept of maintaining plant conditions within the safe operating envelope should be emphasized in operator training.

1.1.4 Conclusions and Recommendations

The staff recommends that increased emphasis be placed on implementing symptom-based EOPs and related training. Full implementation of symptom-based EOPs is expected to ensure that procedures are adequate. The staff also recommends an increased emphasis on NRC inspections to determine if those administrative controls needed to ensure that procedures are being followed have been prepared and are in place. Further, the staff recommends initiation of a research program to analyze the frequency, nature, and severity of violations in order to provide a basis for the consideration of measures that might increase assurance that violations that could be instrumental in causing an accident or emergency situation or compromising safety margins will not occur. These measures are intended to reinforce assurance that operations and other safety-related activities will be performed in accordance with approved written procedures.

1.2 Approval of Tests and Other Unusual Operations

Are administrative controls at nuclear power plants adequate to ensure that changes are made safely and that tests and experiments at plants are conducted safely and within the safe operating envelope?

The testing being performed at Chernobyl at the time of the accident was stated to have been prepared by an individual not familiar with the RBMK-1000 type of reactor. Moreover, the Soviet report (USSR, 1986) stated that "the quality of the program was poor and the section on safety measures was drafted in a purely formal way...." In addition to the test program being poorly constructed, its intent was violated in a number of ways. The test power level was chosen to avert control difficulties that would result from changes to the thermal, hydraulic, and nuclear characteristics at low power levels. The test also presumed an automatic trip of the reactor by closing the turbine stop valve when the test was initiated. The trip circuit for this function was defeated by the operators to expedite a retest if the original test failed. An adequately constructed test procedure would establish the prerequisites, including power level, with a warning or caution against lower power levels and would have established in advance any permissible bypasses of safety features.

U.S. standards and administrative control requirements minimize the potential to conduct a test without an adequate safety review. Multiple Federal regulations would have been violated had Chernobyl Unit 4 been a licensed U.S. plant.

In the United States, all changes, tests, and experiments planned to be performed in reactors licensed by the NRC are evaluated against the requirements of 10 CFR 50.59, "Changes, Tests, and Experiments." This regulation establishes which changes, tests, and experiments may be done solely under a licensee's

administrative procedures and which must get prior NRC approval. The NRC staff must review, approve, and authorize any change, test, or experiment that involves an unreviewed safety question or a technical specification.

If the change, test, or experiment does not involve an unreviewed safety question or a technical specification, but does involve reactor safety, it must be done under the administrative control system discussed in Section 1.1 and be submitted to that review and approval process.

The controls to ensure that changes, tests, and experiments are properly dealt with are discussed in this section. These controls are a part of the administrative controls discussed in Section 1.1 and relate to operator attitudes toward safety as discussed in Section 1.5.

1.2.1 Current Regulatory Practice

10 CFR 50.59 requires Commission approval for any change to the facility or to procedures described in the Safety Analysis Report and any test or experiment which involves a change to the Technical Specifications or to an unreviewed safety question (USQ). A USQ is defined as a change which increases the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated, creates the possibility of an accident or malfunction of a different type than that previously evaluated, or reduces the margin of safety as defined in the basis of the plant Technical Specifications. The licensee may make the change, which could consist of a new test or experiment, without prior Commission approval if it does not involve a change to the Technical Specifications or a USQ. If such a change, test, or experiment affects nuclear safety, but does not involve a USQ, the change, test, or experiment still must be properly reviewed and approved before implementation. The safety evaluation required by 10 CFR 50.59 is but one of several reviews required either by Technical Specifications or by other plant administrative controls. Figure 1.1 charts the flow of changes, tests, or experiments required to receive proper authorization.

After authorization of the change, test, or experiment has been obtained, the test details have to be converted into a procedure. The process of converting test details into a procedure follows the controls discussed in Section 1.1 for writing, reviewing, approving, and implementing procedures.

NRC personnel inspect selected activities involving changes, tests, or experiments to confirm that 10 CFR 50.59 requirements were satisfied. Resident inspectors at each site stay abreast of licensed activities and periodically confirm that changes, tests, and experiments have been appropriately reviewed. Each plant has an NRC project manager assigned to its main office who also stays abreast of licensed activities. The project manager's role has recently been expanded to include routine review of documentation summaries and selective audits of 10 CFR 50.59 activities.

1.2.2 Work in Progress

Some reviews conducted in accordance with 10 CFR 50.59 have been found inconsistent in depth of review and quality of documentation. On May 27, 1986, NRC management requested that industry develop review criteria and guidelines for

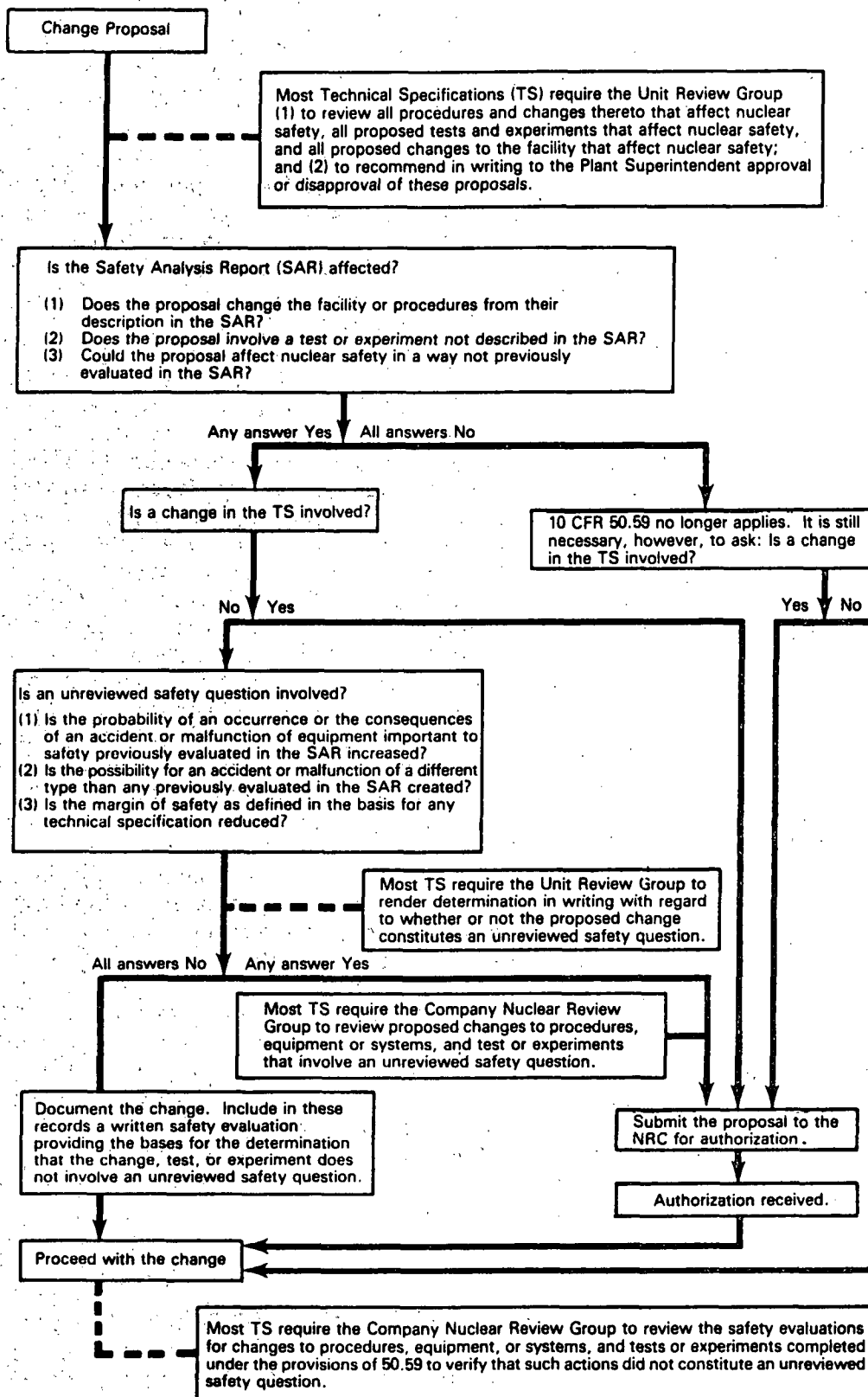


Figure 1.1 Approval of changes, tests, and experiments

licensees conducting reviews of changes, tests, and experiments under the regulatory provisions of 10 CFR 50.59. This work was initiated by the Atomic Industrial Forum (AIF), now a part of the U.S. Council for Energy Awareness, and is now being conducted under the auspices of the Nuclear Management and Resources Council with participation by AIF and the Electric Power Research Institute's Nuclear Safety Analysis Center. This group presented a draft set of criteria and guidelines to NRC management in November 1987. The NRC has reviewed these guidelines and provided comments to the industry. Once these criteria and guidelines are acceptable, they will be used by the NRC--as well as the licensees--to review changes, tests, and experiments by licensees under the provisions of 10 CFR 50.59.

1.2.3 Assessment

Each year licensees conduct thousands of reviews under the provisions of 10 CFR 50.59. Some of the review items should have received prior NRC review, as later determined by inspections and licensee audits. Enforcement penalties have been levied for some of these violations. Nevertheless, considering the large number of changes, tests, and experiments involved, this activity has been mostly successful. The staff has observed some inconsistencies in the level and quality of reviews performed by licensees in making the judgment as to the identification of an unreviewed safety question and thus the involvement of the NRC. Moreover, documentation associated with some of these reviews has sometimes been inconsistent and insufficient.

On occasion, because the unreviewed safety question determination was too narrowly drawn, the licensee determined incorrectly that a unreviewed safety question was not involved. Therefore, the NRC did not review the item. As stated in a memorandum to Commissioner Asselstine (Malsch, 1986), "the Agency's regulatory scheme recognizes that it is neither necessary nor manageable for the Commission to undertake prior review and approval to all subsequent changes to the design or operation of the facility...." It is clear that those items needing prior NRC review should be limited, but the most important items should be reviewed. Also, the resident inspector has access to the lists of all tests for all phases of plant operation to help ensure his/her awareness of tests of potential safety significance.

The fact that the Chernobyl accident was initiated by a test intended to assess equipment capabilities raises a concern about the balance between the benefit of testing and the risks introduced by tests. Although safety reviews are intended to ensure that tests are conducted within the safe operating envelope, equipment and design deficiencies have, in a few instances, led to unacceptable plant conditions (e.g., rapid cooldown during testing at Catawba). However, without such testing, these deficiencies may not have been identified. Therefore, tests should be evaluated to determine the potential risks associated with testing versus the benefit or need for the test.

1.2.4 Conclusions and Recommendations

The NRC should review the results of the joint Nuclear Safety Analysis Center/Atomic Industrial Forum efforts to produce criteria and guidelines for licensee reviews of changes, tests, and experiments to ensure that (1) appropriate depth and quality of reviews will be required, (2) review documentation will be adequately prescribed, and (3) the distinction as to which of these should receive

prior NRC review is appropriately defined. The additional controls thus provided should ensure that operations within the safe operating envelope are maintained. If deficiencies in this review are identified, the NRC should correct them and should publish the criteria and guidelines as the regulatory position on reviews required for changes, tests, and experiments. Also, consideration should be given to an evaluation of whether current NRC testing requirements (e.g., surveillance testing required by Technical Specifications) appropriately balance risks and benefits.

1.3 Bypassing Safety Systems

Multiple safety systems that could prevent or mitigate the consequences of the accident at Chernobyl were intentionally disabled by the plant operators before they initiated a test procedure that ultimately led to the accident. The test procedure apparently called for the bypassing of certain safety systems. It is known that the operators deviated from the test procedure in order to complete the test, and it is suspected that some of the deviations involved the bypassing of additional safety systems. It is apparent that administrative controls governing the availability of safety systems did not exist or were blatantly violated by the operators. Thus, a safe operating envelope was not maintained. In assessing the implications of the Chernobyl event with respect to U.S. commercial reactors, a question raised is whether the ability of operators to override or bypass safety systems, during modes of plant operation in which they should remain operable, is a safety concern. This issue is discussed below. The scope of this discussion is limited to the typical administrative controls and hardware design features used to ensure the availability of sufficient safety systems to respond to transient and accident conditions. The unavailability of safety systems because of sabotage and human error (i.e., unintentionally disabling a safety function versus taking conscious deliberate actions based on poor judgment to override or bypass a safety function) are not within this scope.

Definition of Bypass

The bypass or override of a safety or protection system is typically any action taken by the operator that inhibits or prevents the system or some portion of the system from performing its safety-related protective function(s). In general, two types of bypasses are used at U.S. commercial reactors, both of which are typically initiated manually by the operators in the control room. The first type of bypass is referred to as a "maintenance bypass" and is used to preclude inadvertent or unwanted system actuations when routine testing, maintenance, repair, or calibration activities are being performed during reactor operation. The use of maintenance bypasses allows routine surveillance testing of plant safety systems to detect component failures that may have occurred, and to verify system operability, thus providing assurance that the system will perform as designed when called upon to perform its safety function(s). A maintenance bypass may temporarily reduce the degree of redundancy of equipment, but will not cause the loss of a safety function. The second type of bypass is referred to as an "operating bypass" and is used to permit operational mode changes. An example of an operating bypass is the blocking of an engineered safety features actuation when low reactor coolant system pressure (indicative of a system break during power operation) is detected during a controlled

reactor shutdown, where pressure is intentionally reduced to below the actuation setpoint and safety system actuation is not desirable. Therefore, bypasses are necessary to prevent inadvertent actuations of plant safety systems that might otherwise disrupt plant operation or result in unnecessary challenges to safety systems, and if used correctly, actually contribute to the overall safety of the plant.

1.3.1 Current Regulatory Practice

(1) Technical Specification Restrictions on the Use of Bypasses

The use of bypasses at U.S. commercial reactors is controlled by plant-specific Technical Specifications. These specifications are a part of each reactor operating license, and compliance with them is required. Before granting an operating license, the NRC requires that an analysis be performed to determine the plant response to prescribed bounding design-basis transient and accident events. This is a conservative analysis which assumes the "worst case" initial plant conditions (i.e., the mode of operation, initial parameter values, control system status, etc., that would lead to the most severe design-basis transient or accident) and identifies the safety systems whose successful operation is relied on to prevent or mitigate the consequences of the events so that safety limits are not exceeded. The Technical Specifications require the operability* of safety systems consistent with the transient and accident analysis. They include required actions considered appropriate when a redundant portion (or train) of a safety system is bypassed (or rendered inoperable for any reason) during modes of plant operation for which it is normally required to be operable. These actions require that the bypassed or inoperable portion of the safety system be restored to an operable status within a specified time. This is referred to as "out-of-service time," i.e., an interval of short duration considered sufficient to allow completion of necessary repair activities without unduly restricting reactor operation, and without causing unnecessary risk because part of the system is unavailable for a prolonged time. If the repair cannot be done in the allotted time, the reactor must be shut down or its operation must be restricted to a condition where the system is no longer required to ensure plant safety.

The Technical Specifications for many U.S. commercial reactors include a small number of special test exceptions which permit safety systems to be bypassed by the control room operators in order to perform the tests. These are infrequently performed tests which are carefully staged with significant involvement by the licensee in the control and execution of the tests. They are usually conducted at reduced power with some reactor trip settings lowered. NRC resident inspectors often monitor these tests.

(2) NRC Criteria and Guidance Regarding Bypasses

Requirements for the design of safety systems concerning the use of bypasses are stated in 10 CFR 50.55a(h) and the Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." Two of these requirements, applicable to all U.S. commercial reactors, are summarized below.

*The state of being capable of performing their specified functions.

- Where operating requirements necessitate the use of an operating bypass, the design shall be such that the bypass condition is automatically removed (i.e., system operability automatically restored) when the plant enters a mode of operation for which the safety system is required to be operable in accordance with the Technical Specifications.
- If the protective action of a portion of a safety system has been bypassed or deliberately rendered inoperable for any purpose, this fact shall be continuously indicated in the control room.

The first requirement ensures that a safety system bypassed to permit reactor mode changes will not remain inadvertently bypassed when the plant is returned to a mode of operation for which the system is required to be operable. The second requirement is intended to ensure that sufficient information concerning the inoperable status of safety systems is provided in the control room so that the operators will be continually aware of the status of redundant portions of the protection system. Information on the status of safety systems is typically provided in the control room through a combination of administrative controls (e.g., manually updated status boards and logs) and automatic indication systems (e.g., annunciators and plant computer printouts).

Additional guidance concerning the use of bypasses and the design of bypass circuits is provided in IEEE Standard 338-1975, "IEEE Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems," as supplemented by Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems," Regulatory Guide 1.22, "Periodic Testing of Electric Power and Protection System Actuation Functions," and Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." This guidance emphasizes the importance of providing (a) sufficient redundancy within the safety system so that when a portion of the system is bypassed for maintenance or testing purposes, that capability still exists to accomplish the safety function if required, (b) positive means to prevent a concurrent bypass condition on redundant or diverse safety systems/equipment, (c) automatically actuated continuous indication in the control room of each bypass condition that renders a portion of a safety system inoperable during a mode of plant operation for which the system is required to be operable, and which is expected to occur more than once a year, and (d) measures to ensure that upon completion of work activities which required the bypass condition (e.g., maintenance or testing), the affected systems and equipment are restored to their normal operational status.

1.3.2 Work in Progress

The current effort under way at NRC to revise Regulatory Guide 1.47 was recommended in NUREG/CR-3621, "Safety System Status Monitoring." NUREG/CR-3621 identifies some of the tasks associated with monitoring the status of bypassed safety systems (e.g., updating status boards and determining system status during all modes of operation) which are prone to human errors. These human factors considerations are being reviewed for possible inclusion in Regulatory Guide 1.47.

Another staff effort under way is the implementation of the Maintenance and Surveillance Program Plan (MSPP). The MSPP examines the commercial nuclear

industry work and control processes associated with maintenance and surveillance activities. This includes administrative controls used to ensure the availability of redundant safety systems/equipment.

A related area of activity is work to resolve the generic issue of wrong-unit or wrong-train events (Generic Issue 102). An NRC staff report on this subject (NUREG-1192) was issued in 1986. The report indicated that inadequacies in equipment labeling (absent, illegible, or unclear labels) were the primary contributor to such errors, with deficiencies in training and procedures being additional factors. The effectiveness of voluntary industry efforts, coordinated by the Institute for Nuclear Power Operations, is being evaluated by the NRC staff.

1.3.3 Assessment

(1) Bypass Design Features

In most nuclear power plant designs, the bypass of safety-related equipment is initiated by the plant operators from the control room, or by plant service personnel or instrument technicians from instrument or switchgear cabinets after the bypass has been approved by the control room operators. Before the bypass is effected, procedures require that the operators verify the availability of redundant safety equipment to ensure the bypass will not result in the loss of a safety function. The bypass is typically accomplished by actuation of a bypass or test switch. Operation of the switch will disable a portion of the safety system, and will usually provide inputs to status monitoring points in the control room such as the plant annunciator and computer.

Typically, there are only a few approved methods of effecting safety system bypasses at a given plant. In many cases, the hardware design of the bypass circuitry (for approved methods of bypassing) contains interlocks which make it impossible to bypass redundant portions of safety systems or to bypass a portion of a safety system during a mode of plant operation for which the system is required to be operable. In some designs, trying to bypass redundant portions of a safety system will cause the protective action to occur. These design features make it difficult for the operator to inadvertently or intentionally bypass safety-related functions when the systems are required to be operable. This is especially true for reactor trip systems. The design of bypass circuits varies from plant to plant. In general, it is more difficult to bypass safety functions, either inadvertently or intentionally, at newer plants than at older plants because of improved bypass circuit designs and improved administrative procedures for bypassing.

(2) Intentional Bypass or Override of Safety Systems

Because of the multiple levels of administrative controls governing the use of bypasses at U.S. commercial reactors and hardware design features that physically restrict the misuse or abuse of bypasses, the staff considers the probability of intentionally bypassing safety functions when they are required to be operable to be very remote. However, if an operator is determined to bypass a required safety function, there are many ways in which it could be accomplished. These include installing jumpers, lifting leads, pulling fuses, blocking relays,

and "racking out" breakers* in the safety system logic or actuation circuits, or actions such as closing a normally open local manually operated valve in the safety system process piping. Since there are requirements on the minimum number of control room personnel on duty at a given time, it would be difficult for an individual operator to intentionally bypass a required safety system without just cause. This would take agreement from several control room personnel to deliberately violate safety system operability requirements in the Technical Specifications. Furthermore, plant operation in violation of the Technical Specifications is not taken lightly. The NRC's regulations require staff review and approval before any Technical Specification design or operating requirement can be exceeded. If plant personnel violate Technical Specification requirements that deal with operability of safety systems, these actions can result in penalties and enforcement actions by the NRC; however, licensee attitudes toward compliance with industry and regulatory standards designed to protect public health and safety have been and continue to be very positive.

1.3.4 Conclusions and Recommendations

Even before the accident occurred at Chernobyl, the staff had identified the need to evaluate the implications of bypassing safety systems. The accident simply substantiated that the evaluation needed to be done. Thus, the work under way to revise Regulatory Guide 1.47 should continue as planned. This work includes the development of guidance concerning the reexamination of tests that require the bypassing of safety systems, to ensure that such tests are done only when there is a clear necessity and that the bypassing does not significantly reduce safety margins. Such guidance is supplemented by consideration of measures to reduce the incidence of wrong-train/wrong-unit errors, including clear labeling of equipment, improved verification procedures, and adequate training, in addition to bypass indication.

1.4 Availability of Engineered Safety Features

The operators at Chernobyl bypassed the emergency core cooling system and several reactor protection system setpoints during the test program, which permitted operations outside the safe operating envelope and ultimately led to the accident. U.S. commercial reactors operate according to the requirements contained in their Technical Specifications. These Technical Specifications allow engineered safety features actuation signals and reactor protection system setpoints to be bypassed and engineered safety features to be rendered inoperable during various modes of operation. This is necessary in order to smoothly bring the reactor to power from a shutdown condition, to smoothly shut down the reactor from power operation, to protect equipment from conditions for which it was not designed (e.g., high neutron flux or high pressure), and to test the instrumentation and engineered safety features. Therefore, because it is necessary to bypass certain engineered safety features under a given set of conditions, it is necessary to consider what assurance there is that plant conditions will be maintained in the safe operating envelope and that adequate protection is still provided.

*Physically relocating circuit breakers, thereby opening the circuit.

1.4.1 Current Regulatory Practice

The approach taken in the licensing process to demonstrate that adequate protection is provided by the engineered safety features is to postulate a series of design-basis events. These design-basis events are listed in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," and in the Standard Review Plan (NUREG-0800). The design-basis events are analyzed at the power level, burnup, and other conditions expected to yield the most conservative analysis with respect to the acceptance criteria. They are also analyzed with the assumption that the engineered safety features and reactor protection system functions which are available are consistent with the mode of operation of the reactor (considering the single active failure which would result in the worst consequences from the event). For each of these design-basis events it must be demonstrated that the reactor can be brought to a safe and stable condition and that any radioactive release would be limited to an acceptable level. This is demonstrated by meeting the acceptance criteria of the Standard Review Plan for each event. It follows that this protection must be shown for every mode of reactor operation from full power to refueling conditions. These modes are defined in the Technical Specifications. If an engineered safety feature or a reactor protection system function is not required by the Technical Specifications to be operable during a certain mode, the acceptance criteria must be met without reliance on that equipment or instrumentation. Accordingly, the Technical Specifications identify equipment operability requirements to provide adequate protection. As noted in Section 1.1, administrative controls are established to ensure that the Technical Specifications are followed and, therefore, that appropriate engineered safety features are available.

1.4.2 Work in Progress

A study is currently in progress to address inconsistencies between safety analyses and Technical Specifications. This study is being done for a typical, later-model Westinghouse-designed pressurized-water reactor. It is possible that inconsistencies discovered could have generic applicability. Furthermore, the Technical Specification Improvement Program will result in the development of more operator-oriented Technical Specifications, improved Technical Specification "Bases," and Technical Specifications that identify equipment operability requirements for the appropriate operational modes based on existing analyses.

1.4.3 Assessment

Because of the reliance placed on the Technical Specifications to identify appropriate conditions under which equipment should be operated, the following questions must be addressed for all modes of operation:

- (1) Do the Technical Specifications allow entire engineered safety features to be inoperable during modes of operation when they may be needed?
- (2) Are engineered safety features which may be needed to mitigate design-basis accidents omitted from the Technical Specifications?

(3) Do the Technical Specifications allow an unanalyzed condition?

In general, the response to these questions is "no." However, examples have been identified in which the response may be "yes."

For instance, examples of Technical Specifications that allow equipment to be inoperable in certain modes when it may be needed are:

- The analyses of steam generator tube rupture assume that the operator isolates the affected steam generator by closing the main steam isolation valve on the associated steamline. The Technical Specifications do not require operability of the manual isolation feature in MODE 4 (hot shutdown).
- The auxiliary feedwater system is not required to be operable in MODE 4, but it is permissible to use the steam generators in MODE 4. If the main feedwater system were to fail, makeup water to the steam generator would not be assured.
- The safety injection signal is permitted to be blocked in MODE 3 (hot standby) at less than 1985 psig (for Westinghouse reactors). Hence, safety injection will automatically actuate only on high (level Hi-1) containment pressure. However, for some plants Hi-1 may not actuate the safety injection signal in MODE 3 for certain breaks.
- The number of reactor coolant pumps required to be in operation in MODE 3 may not be consistent with the number assumed operable in the control rod bank withdrawal from subcritical transient.

Some equipment that may perform a safety function, that does not have operability requirements in the Technical Specifications are:

- steam generator relief valves which are required to be safety related by internal NRC guidelines (NUREG-0800)
- auxiliary building filters which are credited with reducing offsite doses

An example of an unanalyzed condition not prohibited by Technical Specifications follows: The Technical Specifications allow both residual heat removal (RHR) pumps to be operating in MODE 4. If a loss-of-coolant accident were to occur and the hot leg containing the RHR suction line became uncovered, both RHR pumps could become inoperable if both were operating as permitted by Technical Specifications.

Furthermore, the time when shutdown cooling must be maintained has been identified as a time when the reactor can be placed in a relatively more vulnerable position than while in operation, since redundancy and availability of other systems may not be present to the same extent as they are during operation at power, because some of the equipment is allowed to be inoperable by the Technical Specifications. For example, an AEOD* study (AEOD, 1985) found that equipment associated with reactor coolant system (RCS) vessel water level monitoring

*Office for Analysis and Evaluation of Operational Data.

during plant shutdown is "frequently inadequate and failure prone." Inadvertent and undetected reductions in RCS inventory were a significant contributor to risk when the RCS was partially drained.

Although licensees typically operate their plants to avoid some of these vulnerabilities, operator recovery actions can generally keep the plant safe in these instances. Under the plant conditions identified where the Technical Specification requirements may not be consistent with the safety analysis, a significantly longer time period (compared to the power operation mode) exists for operator action to restore the safety function before any serious consequences to the plant occur. Thus, these inconsistencies are not expected to represent a serious threat to safety because the operators are generally able to identify such events and take appropriate manual actions to maintain the plant in a safe condition.

However, the above examples indicate that consistency of the Technical Specifications with accident analyses should be considered and improved where necessary, particularly for shutdown conditions, when the availability of equipment assumed for accident analyses was not considered in the licensing process to the same extent as that for power operation. This situation is complicated further because the Technical Specifications allow more equipment to be out of service when the reactor is not in power operation. Since the containment may not be isolated, the consequences may be exacerbated.

1.4.4 Conclusions and Recommendations

Most of the research on the consistency of accident analysis assumptions with equipment availability as defined in the Technical Specifications has been done using several Westinghouse designs. However, because of the differences in design and Technical Specifications, even among Westinghouse-designed reactors, the staff believes that each licensee (in revising its Technical Specification "Bases") should perform a comprehensive review of its specific design (including design-basis-accident analyses) and Technical Specifications to determine if, for each mode of operation defined in the Technical Specifications (1) all equipment required to mitigate the design-basis accidents has corresponding operability requirements and (2) sufficient equipment is available to ensure that safe shutdown cooling can be maintained with redundancy (including reliable flow and level indication) while the reactor is shut down. If the review shows that Technical Specifications require actions that would place the reactor in a less safe mode, the staff should initiate action to change the specifications. It is planned to conduct this review and make any such changes identified through the Technical Specification Improvement Program (TSIP). In addition, in order to ensure that licensees are aware of the need for consistency between Technical Specifications and safety analyses, the staff recommends that future proposed changes to the specifications be accompanied by a justification that the proposed change is consistent with the safety analyses. This could be done with the construction of an adequate basis for the "Bases" section of the Technical Specifications, as is planned in the TSIP.

The above concerns with plant operations in the shutdown condition (MODES 4, 5, and 6) show that events with serious consequences could occur. This area should receive more scrutiny from the NRC and the nuclear power industry. The staff identified these concerns before the Chernobyl accident. The accident reinforced the need to continue this work. The staff therefore recommends that NRC

continue to study this problem at the priority established and recommend ways to improve safety under these conditions, if such improvement is warranted.

1.5 Operating Staff Attitudes Toward Safety

The accident at Chernobyl raised the question whether licensed operators, senior operators, and other staff at nuclear power plants in the United States have and maintain an acceptable level of vigilance toward safety when operating commercial nuclear power plants.

A significant aspect of the Chernobyl accident involved operator decisions and actions that reflected an apparent loss of the sense of vigilance toward safety and ultimately led to operators allowing operations outside the safe operating envelope. The Soviet report (USSR, 1986) identified some potential causes of this unacceptable attitude: (1) pressure on the operators to complete a test during that reactor shutdown as the next opportunity would be more than a year away, (2) test delay may have aggravated operator impatience and contributed to a "mindset" that led to imprudent safety actions, (3) operators being so intent on establishing an acceptable power level for the test that they ignored the unstable state of the reactor, and (4) a clear failure to appreciate the basic reactor physics of the RBMK reactor. Several additional factors, not explicitly noted in the Soviet report, may well have had an important effect: (1) this was the last night shift before a holiday, (2) the night shift itself gives the operator a sense of "needing to get it done on his own," and (3) this was a plant with an excellent reputation for getting the job done. A further contributor could have been a "test" mentality that dismisses violations and lack of precautions because operators rationalize that "it's OK because it's only a test."

1.5.1 Current Regulatory Practice

Regulations do not directly address operator attitudes or sense of vigilance. However, there are regulatory and administrative requirements that address areas which are related to or affect human behavior, attitudes, and preparedness. Regulations under 10 CFR 55 require certification and testing of candidates to ascertain physical and technical acceptability to perform licensed operator duties. Additionally, requalification requirements mandated under this regulation are intended to maintain a level of technical competence through continuing training and performance evaluation that would guard against a failure to appreciate basic reactor physics, systems safety, and administrative constraints.

Routine operational and shutdown requirements demand systematic attention to the status of the power plant equipment. This attention is focused, for example, by structuring the shift turnover process with procedures and signature checkoff sheets. The checkoff sheets include such items as current operational status, identification of out-of-service equipment, status of safety systems and components, surveillance requirements, and limiting conditions for operation to name a few. Additionally, the shift operating personnel monitor and maintain operating logs which document plant status and changes to plant status on an "as occurring" basis to verify that parameters and equipment availability are within the limits of the Technical Specifications.

Regulations under 10 CFR 50.36 require Technical Specifications which, in addition to establishing safety system requirements and limiting conditions for

operation, mandate shift manning levels. Furthermore, in most cases Technical Specifications specify maximum working hours to ensure that rested, qualified operators are available.

To the extent that attitudes are affected by working conditions and environment, the NRC and the industry are involved in control room design and human factors engineering to reduce or eliminate unnecessary stress factors on the job.

Finally, NRC personnel evaluate facility applicants and licensees for compliance with regulations and other requirements governing their operations. When necessary, enforcement actions, such as plant shutdowns and/or fines, are imposed for failure to comply. The presence of onsite resident inspectors provides first-hand observation and allows for immediate feedback on operator vigilance with regard to reactor safety.

1.5.2 Work in Progress

Significant work is currently in progress that should have a direct effect on improving operator performance and abilities to safely control the reactor plant during abnormal and emergency events. One of the cornerstones in this effort is the recent revision to 10 CFR 55. This revision requires licensees to have simulation facilities that conform to either American Nuclear Society Standard 3.5 (1985), "Nuclear Power Plant Simulators for Use in Operator Training," or an acceptable alternative. This will allow training and evaluation of candidates and licensed operators on job performance under simulated normal and abnormal conditions. Additionally, requalification requirements for comprehensive written and operational evaluations have been strengthened in this revision to 10 CFR 55.

The NRC is pilot-testing a modified requalification evaluation process for licensed operators and senior operators with a goal to evaluate licensee requalification programs at half of the nuclear power plants annually. This program was directed by the Commission and will be administered in all five NRC regions beginning in late summer of 1988. The staff anticipates an improvement in the quality and level of operator knowledge and performance and a significantly increased level of facility management attention to operator requalification programs.

Efforts by industry include a major initiative to accredit the training programs of licensed operators and key non-licensed plant personnel (e.g., shift technical advisors, instrumentation and control technicians, technical staff and managers, and non-licensed operators). For a program to be accredited it must contain (1) systematic analysis of jobs to be performed, (2) learning objectives derived from the analysis that describe desired performance after training, (3) training design and implementation based on learning objectives, (4) evaluation of trainee mastery of objectives during training, and (5) evaluation and revision of training based on the performance of trained personnel in the job setting. The NRC participates in licensee training program reviews to evaluate and monitor industry progress in this area during and after accreditation.

Additionally, the NRC has mandated that licensees provide engineering expertise on shift to help operators evaluate and combat abnormal and emergency occurrences, and has required better human-factored, symptom-based emergency operating procedures for their use in coping with emergencies. A shift technical advisor must be on site and available in the control room within 10 minutes, as needed.

Furthermore, as noted in Section 1.6, the NRC has endorsed industry self-improvement initiatives and is developing improved methods of monitoring licensee performance. Industry initiatives to strive for excellence will be monitored by the NRC.

1.5.3 Assessment

The basic issue raises a question of operator vigilance with regard to safety at nuclear power plants. The NRC does not directly evaluate "attitudes." However, the NRC and the nuclear industry do have in place regulations, policies, and programs which require, maintain, and evaluate levels of expertise and professional behavior that could not be judged as satisfactory if vigilance with regard to safety were absent. In assessing the adequacy of an operating staff's attitude toward safety, the NRC must also be satisfied that the nuclear industry's attitude toward safety is uncompromising.

The NRC has no evidence from the Chernobyl accident that would suggest that the accident was caused by individuals affected adversely by working on the midnight shift. The present criteria established for allowable plant staff working hours and shift rotations appear adequate to ensure attentiveness and alertness of individuals working at night.

The firmly established training requirements for operator license candidates, especially as expanded by the lessons learned from the TMI accident (e.g., those pertaining to mitigating core damage, heat transfer, and fluid flow), have significantly raised the operator's appreciation of the physics and thermohydraulic phenomena at work during nuclear power generation. Operator training, administrative controls, and actual plant operations stress compliance with approved directives and regulations which enforce and reinforce the appreciation of and vigilance in all aspects of safety and public health. However, emergency operating procedures and related operator training fall short of the response required to handle severe core damage and to manage the containment under adverse conditions. Furthermore, the nexus between maintenance of the safe operating envelope and severe accidents should be stressed in training programs. Accordingly, emergency operating procedures and operator training should be upgraded in this area and should address the actions required by the NRC Severe Accident Policy Statement.

The requirements and guidance provided by the NRC should create an environment conducive to establishing good attitudes among the operating staff. The possibility of operating staffs at power plants developing unacceptable attitudes toward safety or unacceptable levels of technical competence is not believed to be a serious concern when evaluated in light of the above regulatory actions and industry involvement and commitment. Furthermore, the NRC feels this vigilance with regard to safety extends to all plant personnel because of parallel requirements for training, administrative controls, and procedural compliance, and the NRC evaluations of these items through inspections and measurements of licensee performance.

1.5.4 Conclusions and Recommendations

The staff believes that safeguards against unacceptable operator and plant personnel attitudes toward safety are adequate. This conclusion is based on the significant increase in the quality of training, industry initiatives in accrediting training programs, and regulatory and industry oversight inspections, and

depends on continued vigilance for its validity. The study of the nature and frequency of procedure violations recommended in Section 1.1 may add useful insights about factors influencing attitudes toward safety, including the effect of special circumstances (such as night shift, days before holidays, or complacency based on good plant performance) on worker attitudes and the incidence of violations. When fully implemented, new symptom-based emergency operating procedures should aid operators in coping with accidents and transients. However, more training in their use and in severe-accident response should be provided as knowledge about severe accidents grows. Furthermore, the staff stresses the need for all operators to receive thorough training in the bases for safety features/limits and in basic reactor safety.

1.6 Management Systems

It is important to recognize that the effectiveness of administrative controls depends greatly on the management system supervising the operation of the plant to ensure that operations are maintained within the safe operating envelope. Management oversight at all levels must be effective to ensure that tests, maintenance, and operations are safely conducted and that requirements are enforced. This is also the finding of the international team that investigated the Chernobyl accident. Accordingly, the question is whether reviews should be performed at all U.S. plants to ensure that mechanisms (policies, procedures, decision prerogatives) exist at all levels of management to deal effectively with non-routine operations, emergency planning, and the execution of the types of action required at Chernobyl.

1.6.1 Current Regulatory Practice

As noted in several of the preceding sections, considerable reliance is placed on administrative controls to ensure that plant operations are conducted in accordance with approved procedures and within the desired operating envelope. The management systems required to meet NRC licensing criteria are identified in its Standard Review Plan (NUREG-0800). Typically, the qualifications, experience, and training of key management personnel should comply with the criteria endorsed by NRC Regulatory Guide 1.8, "Personnel Selection and Training."

1.6.2 Work in Progress

Although the NRC has concluded that the management and organization of utilities licensed to operate nuclear power plants are primarily a responsibility of the licensee, the staff does review the organizational structure and qualifications before licensing and periodically afterwards to verify that standards continue to be met. The NRC is developing improved methods of monitoring the performance of licensee management, in order to give early warning of management problems, and to employ its regulations, which are tied to objective performance measures, to initiate evaluation and, where necessary, enforcement mechanisms.

In keeping with this policy, the NRC has terminated work intended to provide the technical basis for formulating new requirements in the field of licensee management and organization and has undertaken (1) the development of licensee performance indicators; (2) improvements in the NRC's program of Systematic Assessment of Licensee Performance and (3) programs to focus attention on particular licensees whose management performance has been found wanting. The NRC

has also endorsed industry self-improvement initiatives in the management area proposed by the Nuclear Management and Resources Council and the Institute of Nuclear Power Operations.

1.6.3 Assessment

It is difficult to assess the effectiveness of U.S. management systems in light of the Chernobyl accident because of the lack of information about Soviet management systems. Also, when one considers how difficult it would be to handle the immediate effects of an accident of the proportions of the one at Chernobyl, the present U.S. method of evaluating management systems (focusing as it does on day-to-day operations) may be inadequate.

Analysis of the Chernobyl accident points out that no one was "in charge of safety." In the United States, safety is everyone's responsibility, but it is a concurrent duty. No single individual can be identified at nuclear power plants as the person who is responsible for safety and has no other duties. It has been suggested that a position for a dedicated high-level, onsite nuclear-safety manager could be established to meet this need. However, it appears that the possibility that the presence of this individual might result in a decrease in the sense of responsibility with regard to safety by other site personnel would outweigh the safety benefits that would be gained by this position. Retention of safety as an integral part of plant management and staff responsibility is judged to be the wiser course.

It is not clear that the management criteria established will ensure that the personnel available to handle emergencies of the type experienced at Chernobyl are available at all times. The NRC requires an emergency management organization for coping with certain emergency situations. Personnel listed on on-call duty rosters are available for assisting plant staff, and management has provided a shift technical advisor to aid the operating staff during transients and accidents. However, the planning, staffing, and mitigative aspects and training provided for emergencies primarily deal with the preventive aspects of and the radiological consequences of emergencies. Planning for the operation of plant controls and systems to cope with severe core damage, and training plant staff to such a task, require additional attention.

1.6.4 Conclusions and Recommendations

The NRC requirements on management systems should be assessed with the following specific points in mind. Licensee management should direct its staffs to proceed diligently toward complete implementation of symptom-based emergency operating procedures. Management should examine the scope of the work needed to cope with severe core damage in order to develop training curricula and procedures on ways of managing the core and containment systems so as to minimize public impact.

1.7 Accident Management

The accident at Chernobyl followed a course determined by performance characteristics of the RBMK reactor that differ greatly from those of U.S. light-water reactors. Nevertheless, the accident at Chernobyl has reconfirmed the need for all--including U.S.--nuclear plants to have an accident management program in place that can effectively cope with the prevention and mitigation of severe core damage events. Plant operators and technical teams at both

Chernobyl and Three Mile Island Unit 2 were confronted with unexpected events for which they were only partially prepared. Their actions contributed significantly to the course of events.

1.7.1 Current Regulatory Practice

Historically, emergency operating procedures and operator training were based on transients and accidents presented in the safety analysis reports and reviewed by NRC as part of the licensing process. Severe accidents were not included.

The accident at Three Mile Island, among other things, focused attention on the importance of severe-accident management. Plant personnel who attempted to control the accident at Three Mile Island had to operate beyond their emergency operating procedures and beyond the principles covered in their training program. In the years following the accident, the NRC developed substantial new requirements to address many of the specific weaknesses that had been identified at Three Mile Island. These new requirements have resulted in more experienced and better trained personnel at the plants, improved procedures for dealing with accident situations, better plant instrumentation and diagnostic tools, and improved emergency planning and response capabilities.

Reactor vendors revised their emergency procedure guidelines. The accident management approach changed from event oriented to symptom oriented, or a combination of event and symptom oriented. The guidelines were reviewed and approved by the NRC before the utilities were authorized to use them in developing the planned plant-specific emergency operating procedures. Most plants have begun implementing the revised guidelines, rewriting their emergency operating procedures, and retraining their operators in the use of those procedures. NRC audits revealed certain deficiencies in translating the approved guidelines into the plant-specific emergency operating procedures (e.g., in areas of procedure formatting, validation, and verification). After staff auditing is completed, it is expected that guidance for taking corrective measures will be developed.

When the deficiencies are corrected, the new emergency operating procedures will be a significant improvement over those used before the accident at Three Mile Island. However, as stated above, these procedures were not designed to fully address severe accidents. Assessment of potential improvements in the prevention and mitigation of severe accidents, including operator actions, is now under way. This assessment will consider the interaction of the emergency operating procedures in implementing the Severe Accident Policy.

1.7.2 Work in Progress

In August 1985, the NRC issued a policy statement on severe accidents (50 FR 32138). The policy statement provides criteria and procedural requirements for the licensing of new plants, and sets goals and a schedule for the systematic examination of existing plants. On the basis of available information, the Commission concluded that existing plants pose no undue risk to the public, and the Commission sees no present basis for immediate action on generic rulemaking or other regulatory changes for these plants because of severe-accident risk. However, the Commission emphasized that systematic examinations of existing plants are needed, encouraged the development of new designs that

might realize safety benefits, and stated that it intends to take all reasonable steps to reduce the chances of occurrence of a severe accident and to mitigate the consequences of such an accident, should one occur.

Implementation of the Commission's Severe Accident Policy is under way. An integral part of completing the implementation is the development of an accident management program for each nuclear plant that would be expanded to address severe accidents. In the case of existing plants, licensees will be requested to systematically examine their plants for severe-accident vulnerabilities and develop the accident management program. The NRC and the nuclear industry under its Industry Degraded Core Rulemaking Program (a program concerned with severe accidents) have already examined four reference plants. Considering leading sequences for a given plant, an accident management program will be established for each plant. Diagnostic instrumentation and safety equipment needed for the execution of accident management actions will be identified. The results of the NRC research are being provided to licensees in the form of guidance for individual plant examinations. Licensees will be requested to systematically examine their plants for severe-accident vulnerabilities and develop an accident management program that will recognize any plant-specific vulnerabilities and take advantage of the particular capabilities of each plant for limiting consequences. However, the accident management program is expected to consider predicted leading severe-accident sequences, along with uncertainties, operator errors, and unanticipated behavior.

The licensee's accident management program is expected to be revised to include consideration of severe accidents as related to the following:

- severe accident management strategies
- organizational structure and responsibilities
- nexus between emergency operating procedures and severe accident management strategies
- training of personnel
- availability and reliability of needed instrumentation and equipment

The staff will review the proposed accident management programs and their implementation according to existing channels of review, audit, and inspection.

With respect to new plant applications, the Commission's Severe Accident Policy calls for the performance of a probabilistic risk assessment (PRA), extending to future applications this additional TMI-related requirement of the Construction Permit Rule [10 CFR 50.34(f)]. For these plants, the PRA results will be used to develop the severe accident management program. Assumptions made in the PRA relative to human actions and performance as well as insights gained from the evaluation of the various severe-accident sequences will be documented in the design stage for future use in the development of emergency operating procedures and training programs for plant operators and emergency teams. Instrumentation and equipment needed to support the accident management effort will be identified together with the conditions these instruments and equipment need to survive.

1.7.3 Assessment

The accident management action taken at Chernobyl to a large extent contained novel approaches dictated by need and were quite successful. However, the Chernobyl operators and technical teams encountered numerous difficulties in their efforts of trying to stabilize and cool the core debris and identify the location and extent of the damage. These difficulties provide insights on evaluating approaches to accident management. Some of the more significant accident management-related events were:

- The reactivity accident progressed very quickly and provided little time for operator interaction.
- Operators were unsuccessful in introducing water into the reactor core in order to cool the core debris.
- Various materials (boron carbide, dolomite, clay, sand, and lead) were dropped into the reactor well from helicopters to mitigate the release of radioactive nuclides.
- To cool the core debris and to provide a blanket against oxygen, a system was installed to feed cold nitrogen to the reactor space.
- Radiological measurements were complicated by the fact that the regular measurement system in the plant had been destroyed and the output of detectors that might have survived was inaccessible.
- Fire and a high radiation field existed in combination, complicating standard firefighting methods.

The Soviet experience at Chernobyl demonstrated the need for preplanning, for developing severe-accident management strategies and methods, and for having the needed tools and materials available. It also focused attention on novel methods. The Soviet experts successfully employed two pioneering methods: the dropping of various materials and gas blanketing.

The ongoing U.S. programs, which had been started before the Chernobyl accident, though concerned with different designs and different specific procedures, are addressing the same basic questions on accident management. Nevertheless, the Chernobyl experience provides additional insight on the development of accident management programs and increases the emphasis on the timely development and implementation of these programs. It also focuses attention on a few specific areas where future research or development work seems to be justified. These areas are

- heat removal from core debris, specifically selection of strategies and materials for heat removal
- development of radiation-hardened diagnostic instrumentation and safety equipment capable of surviving severe-accident environments

Future discussions with the nuclear industry as well as formulation of NRC research programs should address these issues.

Firefighting considerations are discussed in Section 2.4.

1.7.4 Conclusions and Recommendations

The Chernobyl event focused attention on the importance of a systematic approach to develop accident management programs. This experience should enhance previous NRC and industry initiatives to develop and implement accident management strategies at individual plants. Timely execution of the ongoing programs at existing plants and performance of individual plant examinations in response to the NRC's policy statement on severe accidents will provide appropriate assurance that U.S. plants can develop an accident management program to cope with the prevention and mitigation of postulated severe accidents. Insights gained from the Chernobyl event are and should continue to be considered during the implementation of the Severe Accident Policy.

CHAPTER 2

DESIGN

The Chernobyl accident was a superprompt critical reactivity excursion. The accident occurred at Unit 4 because the operators had reduced the power to well below the permissible safe operating level and at the same time neglected following low-power operating procedures. Unit 4 shared a site with three other units (Units 1, 2, and 3) and was contiguous with Unit 3, with which it also shared some common elements. All three of the other units, especially the contiguous one, were exposed to some danger from the accident. Fires aggravated the accident and complicated the management of the accident and its consequences. In this chapter, the staff compares the design features of U.S. reactors with design features of the Chernobyl 4 reactor as it looks for possible regulatory changes implicit in the accident.

The nuclear design of U.S. reactors, notably the absence of positive void coefficients and the presence of control rods that are fast acting and that offer substantial shutdown margins, provides assurance against a Chernobyl-type superprompt critical reactivity excursion. Nevertheless, the staff assessed the possible need for confirmatory reviews of the acceptability of risks from other low-probability reactivity-event sequences.

Accident scenarios that could occur at low-power and zero-power (shutdown) conditions that may not be bounded by analyses for full power are assessed in the light of the Chernobyl accident.

The assessment of the implications for a multiple-unit site includes consideration of the effects of shared shutdown-related systems and the effects of radioactive release on operator safety at the other units.

The adequacy of protection provisions during fires with radiation present is assessed.

2.1 Reactivity Accidents

The reactor physics characteristics of U.S. light-water reactors are very different from those of the graphite-moderated RBMK type of reactor at Chernobyl. Positive void (and moderator temperature) coefficients, which played a central role in aggravating the incipient accident at Chernobyl, are generally absent in U.S. reactors and where present have a limited reactivity insertion potential, which precludes their causing any significant reactivity transient. Substantial required shutdown reactivity margins in conjunction with fast automatic insertion of control rods on signals indicative of unsafe conditions provide protection against the occurrence of reactivity excursions in commercial U.S. reactors.

In the Chernobyl reactor, the primary moderator is graphite. The water in the core is intended primarily as a coolant and its moderation effects are secondary. Its effect on reactivity is largely as a neutron absorber. Thus, a decrease in water density, e.g., from fuel coolant void increase via a power increase or flow decrease, can produce a core reactivity increase as a result of the decreased absorption. The magnitude (and sign) of the void coefficient is a function of the material characteristics of the core and depends on both the core design and operation, varying as a function of fuel burnup, core content, and amount of inserted control poison. It is difficult to calculate the void coefficient in a reactor presenting such a complex core loading and burnup pattern as that which existed at Chernobyl at the time of the reactivity accident. However, on the basis of information from the Soviets and some U.S. calculations, the Chernobyl void coefficient appears to increase (become more positive) over the first year or two of operation, corresponding to the Chernobyl operating history, and also appears to be larger at low void and withdrawn control rod conditions, corresponding to the initial conditions of the accident. It was evidently significantly positive during the accident and possibly did not become significantly negative even at high core void content. The possibility for reactivity insertion was thus apparently maximized by the initial conditions for the event.

In U.S. commercial light-water (power) reactors (LWRs), the coolant water also serves as the neutron moderator. A reduction in water density, therefore, decreases both absorption and neutron moderation. LWR fuel-to-moderator ratios are generally designed to provide an undermoderated core in the power operating range. (That is, a reduction in moderation in the core will tend to reduce reactivity in the core and reduce the power.) There is, therefore, a much stronger tendency (than for RBMK designs) for negative coolant void or temperature reactivity coefficients throughout the range of operating conditions. Boiling-water reactors (BWRs) have a strongly negative void coefficient throughout the power range and, for the most part, a negative temperature coefficient below the power range. Pressurized-water reactors (PWRs) also generally have a negative temperature and a void coefficient in the normal operating range. Furthermore, the LWR coefficients become more strongly negative as the density decreases (high temperature or high void content), thus increasing the tendency to reduce excess core reactivity or power rise in a reactivity transient. However, positive moderator temperature and void coefficients can exist over limited ranges of LWR operation for small time periods of core life. In a BWR at unvoided, lower temperature conditions, the fuel-to-moderator ratio may approach overmoderation (i.e., a reduction in the amount of moderation in the core will add reactivity to the core and tend to increase power) if few control rods are inserted (water replacing rods in the core). A critical reactor generally can only be achieved under these conditions (for some reactors) at zero power, late in a cycle, and the total amount of reactivity that could be inserted via such a coefficient could not produce a significant reactivity transient. In a PWR, boron in the moderator helps control reactivity; a decrease in moderator density results in a decrease in core boron, which, under some conditions, may give rise to a positive moderator coefficient. This can occur only near the beginning of a cycle (when a large boron inventory is needed) and generally only at lower power and minimal xenon and inserted control rod conditions. Within the range of normal extremes of reactor operating conditions, the total integrated reactivity that could be inserted via this positive temperature and void coefficient falls within the range already studied for control rod withdrawal or ejection events.

A quantitative indication of the effect of positive reactivity coefficients in the Chernobyl reactor and in a typical PWR at the beginning of a cycle can be made by comparing the total reactivity that can be inserted by voids in both types of reactor. At Chernobyl this total was estimated by the Soviets to be 2.5% $\Delta k/k$ (USSR, 1986). For a PWR at normally allowed operating conditions, this total at the beginning of core life is limited to about 0.5% $\Delta k/k$, and this limit is, in fact, generally not approached. This difference is significant in that the reactivity insertion was far in excess of prompt criticality at Chernobyl, whereas prompt criticality would not be possible with the maximum positive moderator coefficients on U.S. LWRs.

The Chernobyl reactivity event was also affected by the reactivity characteristics of the control rod scram system. The control rod insertion rate is normally relatively slow, taking about 15 seconds for full insertion. As in any large reactor with rods fully withdrawn, the rods must be inserted a significant distance before effective negative reactivity insertion begins to occur. At Chernobyl more than 5 seconds passed before significant negative reactivity insertion took place. This provided time for a large power increase and excess energy insertion before the control rods could become effective. The delay occurred largely because the operators had withdrawn the control rods beyond the limits allowed. In addition, the Chernobyl operators disconnected several of the signals that would have initiated scram automatically. Scram was initiated by manual operator action.

The U.S. LWRs differ from the Chernobyl reactor in that they have a fast-acting scram system. Full scram insertion occurs within about 2 seconds (PWR) to 4 seconds (BWR) and effective negative reactivity insertion in about half that time. This scram speed, combined with initial fuel Doppler reactivity effects, is sufficient to limit corewide and local energy levels to within conservative values, even for the extremes of reactivity transients normally studied.

It has been suggested by some U.S. observers and calculations that the Chernobyl transient was caused or significantly exacerbated by positive reactivity insertion during the initial control rod scram insertion (via manual pushbutton) as a result of a strong bottom peaked power (and neutron importance) distribution and the physical configuration of the control rods and rod followers. The rods on insertion from the fully withdrawn position remove water (acting as a poison) from the lower reactor region as the followers initially move from the core. The exact nature of the important parameters involved, however, is unclear because of the complexity of the physical and neutronic situation and the lack of available detailed information (e.g., power distributions) on which to base definitive analysis. Thus, some U.S. analyses (e.g., those by Brookhaven National Laboratory) indicate no significant contribution from this effect. Such an effect is not required to produce the observed event, but it is possible that it contributed to it. There is no comparable configuration or significant potential for the positive reactivity effect from a scram in U.S. reactors. (A possible very small effect of reactivity insertion in BWRs, with slow rod insertion, caused by the effects of steam void redistribution is of no significance, particularly at scram speed.) Conservative power distributions are used in standard U.S. transient scram analyses. Extremes of conservative distribution (relating to scram effectiveness) have been studied and have been shown to have minor significance in regard to transient magnitudes (relative to the Chernobyl effects) because of relatively fast, effective scram times.

These basic design differences between Chernobyl and U.S. LWRs preclude a Chernobyl-type positive void coefficient reactivity insertion event from occurring from normal conditions in a U.S. LWR. Normal conditions are those conditions that exist while the reactor is operating within Technical Specification limits. There are, however, other types of reactivity insertion mechanisms which, although they have very low probability, could conceivably have consequences more severe than those already considered in the accident analyses and which perhaps should receive additional consideration. There are also conceivable initial reactor conditions for which the effects of a positive moderator coefficient should possibly be further evaluated. The issue assessed here, therefore, is whether, notwithstanding the major design differences, low-probability accident sequences that could conceivably lead to reactivity excursions should be reassessed to verify previous judgments that their risks are acceptable.

2.1.1 Current Regulatory Practice

Standard NRC practice described in NUREG-0800 includes the review of a large number of events that can be characterized as reactivity transients. These events are primarily driven by changes in reactivity control elements or moderator state parameters. A wide range of relevant parameters and initial conditions is explored. Three-dimensional analyses of the core including moderator coefficients over the whole range of fuel cycles are commonly made. Parameters and initial conditions are chosen to bound conservatively those expected to exist at the limits of permissible design and operating conditions. None of these standard events are significantly autocatalytic in nature (because positive coefficients are not significant) and control rod response is sufficiently rapid [except for anticipated-transient-without-scrum (ATWS) analyses] so that all of these events are satisfactorily terminated by a scram. It is not expected that reasonable exaggeration of transient parameters would dramatically change the event sequences involved.

The principal relevant NRC criterion for reactivity insertion events, primarily applied to control rod drop (boiling-water reactor) or ejection (pressurized-water reactor), since as a class they dominate high fuel enthalpy events, is that the peak fuel pellet average enthalpy not exceed 280 cal/g. (UO_2 begins to melt at 265 cal/g and is fully molten at 335 cal/g.) Light-water reactors must be designed and operated so that this limit is not exceeded in analyses using maximum allowed design and operation parameters. In practice, using modern three-dimensional (3D) analysis methods, those maximum events generally do not exceed (and are usually well under) 150 cal/g. Thus, within the standard review area, light-water reactors are far removed from (even local) potentially significant destructive energy levels.

The maximum conceivable reactivity insertion events with multiple failures are not included in these analyses because these extremes are judged to have very low probabilities of occurrence. The selection of events and conditions for analyses is intended to be reasonably exhaustive, and conservative parameters and initial conditions are assumed. However, they do not extend to theoretical extremes if there is a judgment that it would take multiple errors or equipment failures (both of low probability) to attain the extremes. The nature of the extremes is examined in this section. Not only is the positive void coefficient-

type event at Chernobyl examined, but also other potential large reactivity events (e.g., control rod removal) are assessed. Because this issue is based on an assessment of the implications of the Chernobyl accident, the extensions to be examined are not intended to be relatively small perturbations leading, for example, to increased potential for departure from nucleate boiling or to localized damage, but significantly increased and extensive energy depositions with the potential for destroying primary systems.

Significant positive reactivity conceivably may be added to a boiling-water reactor or pressurized-water reactor in three broad categories. These are:

- (1) control absorber removal (control rods or moderator boron)
- (2) moderator state change (reactivity change that results from temperature changes or from a change in steam void content)
- (3) miscellaneous effects (such as xenon loss or fuel cooldown)

The reactivity possibilities of categories 1 and 2 are all appropriately explored in standard safety analyses, with the exploration limited, however, by assumed boundary conditions deemed appropriate (and conservative) on the basis of judgment of probabilities involved and limiting criteria applied. The category 3 effects are generally secondary states occurring (if at all) as a result of preceding events. They (except for fuel cooldown) play no role in analyses of standard events. They generally do not drive events but could conceivably affect terminal phases or, in the case of xenon, initial conditions (they did both at Chernobyl).

Standard practice with respect to these categories of reactivity events includes analyses of event sequences recognized as potentially having a sufficient probability of occurrence to warrant providing appropriate protective measures. The following are significant examples of events considered or analyzed:

(1) Control Rod Ejection in a Boiling-Water Reactor

This event is not analyzed for standard reviews in a boiling-water reactor because the control rod housing support system, provided for safety, has been judged sufficient to reduce the probability of the event to levels below those requiring standard analysis. The control rod drop accident is considered in standard reviews. However, rod ejection has been the subject of some special studies.

(2) Moderator Boron Dilution in a Pressurized-Water Reactor

Analysis for subcritical modes determines that sufficient time exists to halt dilution before criticality is obtained. Transient analyses for these modes are not required. Transients for power modes are bounded by rod withdrawal. Requests for N-1 loop operation have expanded the design basis for dilution events to include possible misoperation of loop stop valves, which results in mixing of dilute water in the loop with borated water in the core. This could result in a reactivity insertion of about 1.5% $\Delta k/k$. Since only a few plants have these stop valves, this event, where it may be pertinent, is being reviewed on a case-by-case basis.

(3) Positive Moderator Reactivity Coefficient in a Pressurized-Water Reactor

Some pressurized-water reactors have Technical Specifications that permit operation with a positive moderator temperature coefficient. Reactors permitted to operate with such a positive temperature (and void) coefficient have all standard events (e.g., loss-of-coolant accident and rod withdrawal) analyzed with appropriate, conservative moderator coefficients. Generally no significant increases in transient effects occur. In most cases, a positive temperature and void coefficient actually exists only near the beginning of the cycle (BOC), at lower power, and under no-xenon, minimum-rod-insertion conditions.

2.1.2 Work in Progress

No work apart from the work discussed in this section is currently being done on any events considered for this issue. The work recommended in Section 2.1.4 has been initiated with NRC-sponsored work at the Brookhaven National Laboratory.

2.1.3 Assessment

The following are significant findings on the Chernobyl event related to reactivity accidents:

- (1) It was evidently driven by reactivity addition.
- (2) The reactivity addition was the result of voiding of the fuel water coolant and the positive void coefficient of reactivity.
- (3) Elements of the RBMK design and initial reactor state conditions both contributed to the event by affecting reactivity characteristics of the coolant and speed of response of the reactivity control system.
 - (a) The RBMK design had an inherent positive void coefficient and operating conditions (e.g., low power, initial high flow, low void) apparently maximized the potential of integral reactivity insertion.
 - (b) The design had slow-moving control rods, and the operating conditions placed them in the least effective location for response.
- (4) Multiple operational errors and departure from prescribed procedures or good practice caused or contributed to the adverse operating conditions. Scrams, including the turbine trip scram, that might have caused timely automatic insertion of the control rods, were disconnected; the scram was accomplished manually, too late to prevent the excursion. It is not known that any mechanical failures contributed to the initial conditions or early transient stages.
- (5) The event was autocatalytic and, once the factors were in place and had initiated the voiding, the positive void reactivity coefficient and power rise interacted to drive the event.
- (6) This made possible a large reactivity addition (with no apparent significant turnaround in the positive coefficient and reactivity insertion) not

compensated by control insertion, resulting in excessive fuel temperature, cladding failure, fuel and coolant interaction, excessive pressure, and destruction of the core.

No close analogies can be drawn between the Chernobyl reactor and U.S. reactors. However, the standard areas of reactivity accident reviews can be extended to events less probable than the standard design-basis events by considering more extended or multiple system failures or errors of operation. Examples of such extended events follow; these expand on the previous standard examples of Section 2.1.1.

(1) Control Rod Ejection in a Boiling-Water Reactor

One rod ejected could be worth as much as 2.5% Δk with an addition rated higher than that for a control rod drop. However, the consequences are not significantly different from those of a rod drop. Generally a double ejection, even if adjacent, would not, because of withdrawal patterns, have significantly greater consequences. Multiple ejection has a large potential worth, and scram might not be effective if there were a multiple ejection of a tight cluster of rods. Since each control rod drive enters the vessel through a different control rod drive housing, the ejection would result from individual failures of these control rod drive housings; no mechanism, however, has been identified that could lead to simultaneous multiple failures of the control rod drive housings during normal operation or as a result of any abnormal condition. In addition, Technical Specifications for boiling-water reactors require control rod drive housing supports to be in place to prevent ejection from the reactor core. Thus, for control rod ejection to take place, this control rod drive housing support would have to be missing.

(2) Moderator Boron Dilution in a Pressurized-Water Reactor

Studies of subcritical mode transients resulting from dilution have indicated that they are not severe if dilution is stopped soon after criticality is reached. Though of low probability, an extreme event would be a loss-of-coolant-accident event (and to a lesser extent, a steamline break) with emergency core cooling system injection with unborated water. Up to approximately 10% Δk could be available for insertion. A large amount could also be available in a maximum dilution carried to completion early in the cycle.

Technical Specifications for pressurized-water reactors call for weekly surveillance of boron concentration and water level of the refueling water storage tank. This provides assurance that should a loss-of-coolant accident or a steamline break occur, the emergency coolant added to the core would contain a sufficient concentration of boron to maintain the core subcritical.

(3) Positive Moderator Reactivity Coefficient in a Pressurized-Water Reactor

In normally allowed critical operating regions (above about 530°F), the maximum integrated moderator reactivity that could be inserted is about 0.5% Δk . Transients resulting from insertion of this amount would be bounded by rod withdrawal or ejection events. Starting a critical event from cold

conditions (erroneously, since this is not a normally allowed critical state) might involve a maximum moderator reactivity insertion potential of about 2% Δk , and could only occur with nearly all rods withdrawn and thus available for scram, at a rate fast enough to counteract the potential reactivity insertion. These numbers are taken from Final Safety Analysis Reports using extremes of allowable operating conditions limited by Technical Specifications.

This would be the closest approach to a direct analogy of the Chernobyl (autocatalytic) conditions. It would, however, lack the automatic voiding condition at Chernobyl; would have to be started by an error or malfunction such as rod withdrawal or cooling failure; would have an effective, fast-acting scram available at low flux levels on the source range monitor or higher level monitor system if needed, and an eventual coefficient sign change; and would probably not differ significantly from an ordinary rod withdrawal at low power. In most cases, positive moderator void effects will be compensated by fuel heatup effects.

The entire range of standard reactivity insertion events has been examined, and extensions such as those in the above examples have been considered. Most of the extensions appear to be the result of assumed additional and arbitrary failures in systems or, to a lesser extent, operations. Many of the extensions do not appear to lead to Chernobyl-type events or approach the consequences of that accident, but lead to mild or to local, limited damage at most.

On the basis of this examination, extensions have been selected which are, based on current judgment, conceivable candidates for further study because they have a potential for serious consequences, while others (e.g., boiling-water-reactor single rod ejection) have been rejected because they do not have that potential or have no apparent mechanism (e.g., boiling-water-reactor multirod withdrawal). The following events appear to be appropriate areas to receive further consideration, which likely would include system and mechanical analysis and probabilistic assessments to better determine if the probability of such events indicates they deserve attention, and/or transient analyses to evaluate event consequences.

Pressurized-Water Reactor

- multiple rod bank withdrawal anticipated transient without scram (ATWS)
- multiple rod ejection (low power)
- unlimited boron dilution
- opening of loop stop valves in a loop containing unborated water
- loss-of-coolant accident or other injection with unborated water
- ATWS with less negative moderator coefficient
- rod withdrawal, heatup, or depressurization from low temperature with positive coefficient

Boiling-Water Reactor

- multiple rod ejection
- boron dilution during ATWS
- rapid boron dilution by emergency core cooling system injection during ATWS
- overpressurization with limited relief
- ATWS with no recirculation pump trip

All of these currently appear to be events of very low probability of occurrence. All involve additional failures or errors, not merely extensions of initial conditions or parameters. They generally involve additional failure mechanisms of a different type from the standard initiators of the class and thus require a diversity of failure. The preliminary judgment, therefore, is that conceivable reactivity accidents are not likely to lead to a Chernobyl-type event. One of the purposes of the analyses above is to estimate the probability of these events so that priorities within NRC can be arranged.

It appears useful to examine these events (and possibly some of the extended areas not currently judged significant), primarily through probability studies and associated systems, structures, and transient-consequence reviews. These could include an examination of the potential for the effect of an operator's failure to comply with administrative controls and how such human error can affect initial conditions, transient parameters, and the operation of mitigating systems. It is recognized that probabilities of potential accident sequences may well be subject to wide uncertainties. This may be particularly true for sequences involving human performance. Recognition of uncertainties in probabilities is an important factor in evaluations. Where potential consequences are severe and probabilities are not well understood, protective measures to be sought need to be commensurate with the risk that may exist. In the course of its studies, the NRC staff will use a probability screening process to prioritize transients. If any events appear to fall within the probability levels of NRC guidelines and involve a significant potential for extensive core damage, they may become a basis for changing design, operational limits (Technical Specifications and administrative controls), or licensing rules (such as the ATWS rule).

2.1.4 Conclusions and Recommendations

Positive void coefficients, which played a central role in the Chernobyl accident, generally do not exist in U.S. reactors; where present, they have a limited reactivity insertion potential that precludes the occurrence of any significant reactivity transient. The nuclear design of U.S. reactors provides assurance against a Chernobyl-type superprompt critical excursion. However, the judgments that determined the identification of possible accident sequences analyzed in Safety Analysis Reports and underlying design approvals should be reviewed to confirm their validity using probabilistic methods. Using NRC ground rules and the NRC review process, either the NRC staff or NRC contractors could do this. This work will be coordinated with the severe accident program, and results will be made available to the industry to improve Technical Specifications if necessary.

2.2 Accidents at Low Power and at Zero Power

One of the unique aspects of the Chernobyl accident is that it occurred at a relatively low power (<7%). This has caused some concern because low-power operation is generally considered to be a safer condition than high- or full-power operation. The principal effect of low power on the Chernobyl accident was related to nuclear/thermohydraulic stability and reactivity insertion. These effects were addressed in Section 2.1. Another important aspect of low-power or zero-power operation is the availability of safety systems.

Sections 1.3 and 1.4 specifically address the subjects of bypassing and availability of safety systems. Different safety systems may be used to provide protection for low-power and shutdown (zero-power) events than are used for high-power events. Technical Specifications prescribe the conditions for bypassing and activating the various systems. The completeness of such Technical Specifications was also addressed in Sections 1.3 and 1.4.

Another issue related to low-power operation is the subject of accident initiators. The Chernobyl accident was initiated because an unusual test was performed at low power. Considering this, the question is posed of whether or not initiators, other than those now assumed, should be considered to ensure that current analyses of design-basis events remain valid. This is also discussed in Section 1.4.

The aspect of low-power operation to be considered here is whether the design-basis events currently are being evaluated at their most limiting power level or whether more attention should be given to these events at low power.

2.2.1 Current Regulatory Practice

10 CFR 50.34 requires applicants to analyze and evaluate the design and performance of structures, systems, and components in order to determine their adequacy for mitigating accidents. The Standard Review Plan (NUREG-0800) requires that this be done for all modes of operation including shutdown modes.

In Preliminary Safety Analysis Reports and Final Safety Analysis Reports, applicants provide an evaluation of each limiting transient to determine the impact of varying reactor power. This evaluation is partly *prima facie* and partly the result of generic or plant-specific sensitivity studies. The results of these evaluations and the numerical results of the worst-case analyses also are presented in Safety Analysis Reports. The results of a sampling of Safety Analysis Reports and topical reports from each reactor vendor for accidents of interest are given in the following sections.

(1) Steamline Break

All three pressurized-water-reactor vendors (Westinghouse, Combustion Engineering, and Babcock & Wilcox) have explored a range of power conditions in generic topical reports on methods of analyzing steamline breaks. A steamline break can challenge containment integrity, specified acceptable fuel design limits, or pressurized thermal shock limits. Recent Westinghouse Final Safety Analysis Reports (FSARs) provide an analysis of the steamline break at both zero and full power. These conditions were generically determined to have the worst potential consequences. The Combustion Engineering (CE) System 80 FSAR provides an analysis of the steamline break at both zero and full power.

The Midland FSAR provides the most comprehensive assessment of steamline breaks in a Babcock & Wilcox reactor. Sensitivity studies therein show that a steamline break at full power presents the worst challenge.

For steamline breaks in boiling-water reactors, the full-power case bounds all the other cases.

(2) Feedline Break

Once again, all three pressurized-water-reactor vendors have explored a range of power conditions in generic topical reports. As would be expected, the Westinghouse and CE reports show that a feedline break at full power presents the worst challenge. Thus, recent CE and Westinghouse FSARs provide an analysis of feedline breaks only at full power. In the Midland FSAR, sensitivity studies show that 55% power is the worst case. This is because secondary-side inventory is adjusted as a function of power level.

For boiling-water reactors, the full-power case bounds all the other cases.

(3) Reactivity Accidents

Increase in feedwater flow, control assembly withdrawals, boron dilution, pressurized-water-reactor rod ejection, and boiling-water-reactor rod drop are all analyzed at zero power. These events are discussed in Section 2.1.

For boiling-water-reactor reactivity events resulting from void collapse, the full-power case bounds all the other cases.

(4) Pump Startup or Pump Trip

Operation at partial power is analyzed for certain pressurized-water reactors. For these cases, less than a full complement of reactor coolant pumps may be operating. Power varies with the number of pumps operating, and inadvertent startup or trip of those pumps is appropriately analyzed.

For boiling-water reactors, power is proportional to flow, and, therefore, the full-power case bounds all the other cases.

(5) Loss-of-Coolant Accident

The loss-of-coolant accident is calculated from full power (plus uncertainties) for both boiling-water reactors and pressurized-water reactors because this maximizes the stored energy in the fuel and the coolant. The staff has been reviewing the question of whether this is bounding with respect to shutdown conditions when portions of the emergency core cooling system may be out of service. This is discussed in Section 1.4.

2.2.2 Work in Progress

Limited NRC work in progress on the subject of regulatory measures to prevent low-power accidents includes consideration of instrumentation and procedural improvements. The consistency of accident analyses and Technical Specifications in the shutdown modes is addressed in Sections 1.3 and 1.4.

2.2.3 Assessment

Recommendations concerning reactivity accidents are presented in Section 2.1. Recommendations concerning bypassing and availability of safety systems during all modes of operation are presented in Sections 1.3 and 1.4. A survey of

accidents required to be analyzed by the Standard Review Plan (NUREG-0800) shows that the steamline break, feedline break, and inadvertent pump startup or shut-down have been adequately studied for an appropriate range of power conditions.

At this time it appears that accident initiators at low power should be systematically studied (as proposed in Section 1.4). Existing probabilistic risk assessments have paid very little attention to low-power conditions or testing in evaluating risk, but may be useful for this task.

2.2.4 Conclusions and Recommendations

Accident initiators at low power should be systematically evaluated as proposed in Section 1.4. This work will be coordinated with the severe accident program, and results will be made available to the industry to help develop improvements in Technical Specifications if necessary.

2.3 Multiple-Unit Protection

The radioactive gas and smoke released during the accident at Chernobyl Unit 4 spread to the other three operating units at the site. The airborne radioactive material was transported to the other units through a shared ventilation system as well as by way of general atmospheric dispersion paths. This raises the question of how accidents at one unit of a multiple-unit site affect the remaining units, and additional questions of how these effects may be compounded when structures, systems, and components are shared between units.

2.3.1 Current Regulatory Practice

The current NRC regulatory practice for protection against radioactive releases can be divided into two parts because the control room has one set of requirements and the rest of the plant has another set of requirements.

General Design Criterion (GDC) 19, "Control Room," of 10 CFR 50, Appendix A, states:

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

The control room protection specified in GDC 19 is implemented through Standard Review Plan (NUREG-0800) Section 6.4 reviews, as required by step 1 of Item III.D.3.4, the NRC Action Plan developed as a result of the Three Mile Island (TMI) Unit 2 accident (NUREG-0660).^{*} As a result of these criteria, control rooms are designed to ensure minimum leakage and generally employ a

^{*}See item 4a on page III.D.3-5, NUREG-0660.

ventilation system to slightly pressurize the control room following an accident. Furthermore, ventilation systems in most control rooms incorporate filtration equipment designed to reduce radioactive particulate and iodine concentrations.

Specific habitability requirements for areas outside the control room are specified in 10 CFR 20.202 and Standard Review Plan Sections 12.3 and 12.4. TMI Task Action Plan Item II.B.2 requires that the dose criteria in GDC 19 be met for vital areas. It does not, however, consider radionuclide transport and release (source terms) greater than those from a design-basis loss-of-coolant accident (specified in Regulatory Guides 1.3 and 1.4) and does not require consideration of significant airborne contamination. Physical separation of equipment may afford some radiation protection.

GDC 5, "Sharing of Structures, Systems, and Components," states:

Structures, systems and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Many multiple-unit plants share a control room and, to ensure a safe shutdown, the NRC review takes into consideration the radiological effect of an accident at one unit on the other. Furthermore, for those plants that have separate control rooms, review under Standard Review Plan Section 6.4 includes an evaluation of how well the control room protects the operator when an accident produces radioactive releases at an adjoining unit.

It is also a relatively common practice at multiple-unit plants to share other safety-related structures and systems, including auxiliary and fuel handling buildings and the associated ventilation systems. Implementation of GDC 5, however, is intended to provide assurance that this sharing will not inhibit the ability to safely shut the plant down after an accident. GDC 5 has resulted in shared systems that have redundancy and isolation suited to maintain required safety functions.

In addition to raising concerns about radiological protection, the Chernobyl accident raised the question of smoke propagating from a burning unit to adjacent units. Current staff guidelines [(1) GDC 3, "Fire Protection," of Appendix A to 10 CFR 50, (2) Appendix R to 10 CFR 50, and (3) Branch Technical Position CMEB 9.5-1 attached to Standard Review Plan Section 9.5.1] provide guidance on containing the spread of a fire beyond a single fire area. There are, however, no specific staff guidelines on smoke propagation. Generally, fire dampers are provided to ensure isolation of fire-affected areas in a short enough time to prevent fire from propagating through shared ventilation systems. The dampers are effective to only a limited degree to control smoke spread. Smoke detectors are employed throughout the plant to provide early warning of fires. Most U.S. plants, particularly newer ones, have incorporated smoke exhaust systems in certain critical areas (e.g., control rooms, electrical equipment rooms). In addition, firefighting plans include the use of portable ventilation equipment to limit the consequences of the spread of smoke.

2.3.2 Work in Progress

In response to concerns expressed by the Advisory Committee on Reactor Safeguards about control room habitability, the NRC initiated a review of the existing design and maintenance of control room ventilation systems. This effort, designated as Generic Issue 83, involves, in part, a survey of 12 operating reactor plant control rooms. This survey is intended to assess whether the actual control room ventilation systems are performing in the manner described by the licensee in response to the criteria of NUREG-0660, Item III.D.3.4, step 1.* The results of the Argonne National Laboratory work on this subject, reported in NUREG/CR-4960 (October 1988), will form a partial basis for determining what further action, if any, is needed.

In addition, step 2 of NUREG-0660, Item III.D.3.4* described the NRC staff's intention to examine control room habitability requirements under degraded-core (severe-accident) conditions. Currently, the NRC has taken no action in this area and no plan has been established to do so because this issue has been assigned a low priority. However, the current NRC study of radionuclide release following severe accidents is continuing and may result in additional criteria for control room habitability design.

As part of its research on fire risk, the NRC staff is investigating the risk significance of smoke control. The staff will determine the need for any additional improvements in fire protection and/or investigations in regard to fire risk at the conclusion of that study.

2.3.3 Assessment

Through the implementation of General Design Criterion 19, control rooms currently provide a degree of protection against radioactive releases from severe accidents by designing them to minimize inleakage, by incorporating pressurization ventilation systems that are intended to maintain the control room at a slight positive pressure after an accident, and by incorporating filtration systems that are effective in reducing the amount of radioactive contamination reaching the control room. Additional efforts currently under way under Generic Issue 83 and with regard to radionuclide releases will provide sufficient additional insight on control room habitability following severe accidents.

The control room is specifically designed to protect personnel from onsite radiation following accidents; other plant areas, however, do not have this protection. In the event of an accident at a multiple-unit site, this practice could keep the plant operator from taking local corrective action should there be unanticipated failure in some equipment, or to eventually take necessary local actions to initiate cold shutdown following a severe accident. Furthermore, the ability to maintain long-term plant shutdown following a severe accident may be affected because an operator will eventually have to enter remote areas of the plant to perform equipment maintenance. However, to minimize the need for immediate local action, automation, control room features, and redundancies are built into the design of shutdown systems. Anticontamination clothing, breathing apparatus, and other protective equipment are available to allow personnel to enter an area in order to take local actions in the longer term.

*See item 4a on page III.D.3-5, NUREG-0660.

In addition, cold shutdown requires only a few simple actions to be taken; thus, access to remote areas is necessary for short periods only. Thus, no further action is needed in regard to access (following a severe accident) to plant areas outside the control room. However, new plants should consider this concern in the design of their ventilation system and postaccident shutdown capability.

Fire protection measures such as fire dampers, smoke detectors, smoke exhaust systems in certain areas, and portable ventilation equipment are considered to be adequate to limit the spread of smoke, particularly in view of the fire protection improvements made in U.S. plants since the implementation of Appendix R to 10 CFR 50. Fire protection at U.S. nuclear power plants is discussed in Section 2.4 of this report.

Finally, although the proper design of shared systems in accordance with General Design Criterion 5 does not compromise safety system functions, and may in fact enhance safety when additional equipment can be employed in place of failed components, sharing may affect the ability to bring a unit not affected by an accident back to normal operation. This could occur if continued sharing is necessary in order to ensure maintaining long-term shutdown of the unit affected by the accident, or if contamination in a unit not affected by the accident is widespread because of shared structures and a shared ventilation system. Therefore, it appears appropriate that the NRC severe-accident policy for new plants, particularly standard plant designs, preclude sharing of systems with such characteristics at multiple-unit sites.

2.3.4 Conclusions and Recommendations

In the event of a severe accident in one unit of a multiple-unit site, the control room operators are adequately protected by design features that will ensure a habitable environment. Control room habitability and radionuclide release studies which will take into account recently developed release descriptions are expected to confirm this conclusion. For areas outside the control room, shutdown system design and control room capability preclude the need for immediate access to remote areas, and measures are available to gain access to take the few longer term actions necessary for accomplishing cold shutdown and performing maintenance. When ventilation and postaccident shutdown systems are being designed for new plants, contamination outside the control room should be considered. Fire protection improvements imposed by Appendix R to 10 CFR 50 will provide effective smoke control. Because sharing may be necessary to ensure shutdown of a unit not affected by the accident and may delay the return of the unit to normal operation, severe-accident policy for new plants should restrict the sharing of systems forming part of the shutdown capability at multiple-unit sites. Systems forming part of the required capability for shutting each unit down should not be shared between units. However, the capability to share systems to provide added defense in depth for safe shutdown remains desirable, subject to appropriate isolation provisions.

2.4 Fire Protection

After the accident at Chernobyl, the Soviets stressed the great importance of firefighting (USSR, 1986).

As a result of the explosion in the Unit 4 reactor, fragments of the core were ejected from the reactor building igniting about 30 fires on roofs and other plant areas. The plant operators determined that the most important action in combating the accident was to extinguish the fires. The immediate threat was that the fire would spread from Unit 4 to Unit 3. Firefighters from neighboring towns responded and within 1 hour contained the fires to Unit 4; all fires were extinguished within 4 hours. Evidently plant personnel began combating the fires and were soon joined by firefighters from neighboring towns.

Of particular concern was the ability to fight fires at locations above grade (i.e., roofs of buildings). The Soviets stressed the need for special equipment to lift firefighting equipment to roofs. Also of concern was the need for protective clothing for firefighters working in radioactive environments. In light of these concerns, a review of firefighting capabilities at nuclear plants and fire protection measures in general is appropriate.

2.4.1 Current Regulatory Practice

There have been many fires in operating U.S. nuclear power plants through September 1986. Of these, the fire on March 22, 1975 at Browns Ferry nuclear plant was the most severe. On the average, a nuclear power plant may experience one or more fires of varying severity during its operating life. Although WASH-1400 concluded that the Browns Ferry fire did not affect the validity of the overall risk assessment, the NRC concluded that cost-effective fire protection measures should be instituted to significantly decrease the frequency and severity of fires and consequently initiated the development of guidelines and rules.

The effort after the Browns Ferry fire resulted in the new rules for fire protection contained in Appendix R to 10 CFR 50. These criteria were intended to amplify the already existing broad guidelines contained in General Design Criterion 3, "Fire Protection," of Appendix A to 10 CFR 50. Additional criteria are contained in Standard Review Plan Section 9.5.1 (NUREG-0800). The fire protection program as promulgated by NRC for nuclear power plants consists of design features, personnel, equipment, and procedures that provide defense-in-depth protection to the public. The primary purposes of the program are to prevent significant fires, to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition, and to minimize radioactive releases to the environment in the event of a significant fire. These guidelines call for management participation in the fire protection program and for design of fire protection features by qualified utility staff. The utility staff is also responsible for fire prevention activities, maintenance of fire protection systems, training, and manual firefighting activities.

The NRC requirements concerning fire brigade and fire brigade training are contained in Sections III.H and III.I of Appendix R to 10 CFR 50 and in Standard Review Plan Section 9.5-1(C.3) (NUREG-0800). A summary of the requirements follows. The aforementioned concerns are addressed specifically within the context of NRC-sponsored requirements and/or guidelines.

Each reactor site is required to have a fire brigade that is trained and equipped for fighting fires. The fire brigade was established to ensure adequate manual firefighting capability for all areas of the plant containing

structures, systems, or components important to safety (this includes roofs, high-radiation areas, and remote plant locations). The fire brigade is typically organized as follows:

- Five members must be on each shift.
- Brigade leader and at least two brigade members must have sufficient training or knowledge of plant safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability.

The shift supervisor is not a member of the fire brigade.

The minimum protective equipment provided for the brigade consists of such items as turnout coats, boots, gloves, hard hats, emergency communications equipment, portable lights, portable ventilation equipment, and portable extinguishers. Self-contained breathing apparatus with full-face, positive-pressure masks is provided for fire brigade, damage control, and control room personnel.

The fire brigade's training program is designed to ensure that the capability to fight potential fires is established and maintained. The program consists of an initial classroom instruction program followed by periodic classroom instruction, firefighting practice, and fire drills. These are detailed below.

(1) Instruction

The initial classroom instruction consists of indoctrination in the plant's firefighting plan and specific identification of each individual's responsibilities, the type and location of fire hazards and associated types of fires that could occur in each plant area, the proper use of available firefighting equipment, and the correct method of fighting each type of fire. The types of fires studied include fires in energized electrical equipment, fires in cables and cable trays, hydrogen fires, fires involving flammable and combustible liquids or hazardous process chemicals, fires resulting from construction or modifications (welding), fires in records, and in hazardous areas in nuclear plants, including roofs.

(2) Practice

Practice sessions are held for the fire brigade on each shift on the proper method of fighting the various types of fires that could occur in a nuclear power plant. These sessions provide brigade members with experience in actually extinguishing fires using emergency breathing apparatus under the strenuous conditions encountered in firefighting.

(3) Drills

Fire brigades drill in the plant so that the fire brigade can practice as a team and at the site it serves.

The drills must be preplanned to establish the training objectives of the drill and are evaluated to determine how well the training objectives have been met. Unannounced drills are planned and evaluated by members of the management staff responsible for plant safety and fire protection.

Performance deficiencies of a fire brigade or of individual fire brigade members are remedied by scheduling additional training for the brigade or members.

Drills must include the following:

- (a) Assessment of fire alarm effectiveness; time required to notify and assemble fire brigade; selection, placement, and use of equipment; and firefighting strategies.
- (b) Assessment of each brigade member's knowledge of his or her role in the firefighting strategy for the area assumed to contain the fire and the brigade member's conformance with established plant firefighting procedures and use of firefighting equipment, including self-contained emergency breathing apparatus, communication equipment, and ventilation equipment, to the extent practicable.
- (c) The simulated use of firefighting equipment required to cope with the situation and type of fire selected for the drill. The area and type of fire chosen for the drill should differ from those used in the previous drill so that brigade members are trained in fighting fires in various plant areas. The situation selected should simulate the size and arrangement of a fire that could reasonably occur in the area selected, allowing for fire development because of the time required to respond, to obtain equipment, and to organize for the fire, assuming loss of automatic suppression capability.
- (d) Assessment of the brigade leader's management of the firefighting effort as to thoroughness, accuracy, and effectiveness.

A major problem at Chernobyl in combating the fires was the inability to get firefighting equipment to the fires, that is, on top of burning roofs that were ignited when hot core material landed on them. NRC guidelines require that roofs be constructed of noncombustible materials that are listed as "acceptable for fire." Such a roof would be difficult to ignite and if ignited, it would inherently retard propagation to other areas. In addition, NRC guidelines require the installation of (1) standpipes and hoses to allow manual firefighting and (2) personnel access and escape routes for each fire area in the plant, thereby addressing the concern of accessibility for firefighting personnel and equipment.

Another problem encountered at Chernobyl was the need to extinguish fires in areas that, as a result of the accident, had become highly radioactive. In the United States, it is currently the practice in nuclear plants that a health physics technician responds to a fire along with the fire brigade, in order to recommend to the brigade leader ways of preventing extensive radiation exposure. Currently, no specific guidelines are provided on fighting fires in a highly radioactive area. Using typical protective equipment (turnout coats, boots, gloves, hard hats, and self-contained breathing apparatus) provides a measure of safety against radioactivity as well as against fire and smoke. Explicit guidelines do exist regarding radiation exposure. If a fire should occur in a high-radiation area, all licensees will follow established utility guidelines and procedures regarding proper attire to protect against radiation in addition to observing guidelines and procedures regarding firefighting apparatus.

At nuclear power plants in the United States, the concept of defense in depth is established to achieve a high degree of safety by using echelons of safety systems. With respect to the fire protection program, the defense-in-depth principle is aimed at achieving an adequate balance in

- preventing fires from starting
- detecting fires quickly, suppressing those fires that occur, putting them out quickly, and limiting their damage
- designing plant safety systems so that a fire that starts in spite of the fire prevention program and burns for a considerable time in spite of fire protection activities will not keep essential plant safety functions from being performed

No one of these echelons can be perfect or complete by itself. Each echelon meets certain minimum requirements; however, strengthening any one can compensate in some measure for weaknesses, known or unknown, in the others.

On November 19, 1980, the Commission published a revised Section 50.48 and a new Appendix R to 10 CFR 50 regarding fire protection features at nuclear power plants. The revised 10 CFR 50.48 and Appendix R became effective on February 17, 1981. Section III of Appendix R contains 15 subsections, lettered A through O, each of which specifies requirements for a particular aspect of the fire protection features at nuclear power plants.

Because it is not possible to predict the specific conditions under which fires may occur and propagate, design-basis protective features rather than the design-basis fire are specified in the rule. Plant-specific features may require protection different from the measures specified in Section III.G (the key section in Appendix R). In such a case, the licensee must demonstrate by means of a detailed fire hazards analysis that existing protection or existing protection in conjunction with proposed modifications will provide a level of safety equivalent to the technical requirements of Section III.G of Appendix R.

In summary, Section III.G deals with fire protection features for ensuring that systems and associated circuits used to achieve and maintain safe shutdown are not damaged by fire. Fire protection configurations must either meet the specific requirements of Section III.G or an alternative fire protection configuration must be justified by a fire hazards analysis. Generally, the staff will accept an alternative fire protection configuration if

- (1) The alternative ensures that one train of equipment necessary to achieve hot shutdown from either the control room or emergency control station(s) is free of fire damage.
- (2) The alternative ensures that fire damage to at least one train of equipment necessary to achieve cold shutdown is limited so that the equipment can be repaired within a reasonable time (for minor repairs, components stored on the site are used).
- (3) Fire-retardant coatings are not used as fire barriers.

- (4) Modifications required to satisfy Section III.G would not enhance fire protection safety levels above that provided by either existing or proposed alternatives.
- (5) Modifications required to meet Section III.G would be detrimental to overall facility safety.

2.4.2 Work in Progress

The NRC is currently working to ensure full implementation of fire protection programs at all U.S. nuclear power plants as follows:

- (1) 95% of all plants licensed before January 1, 1979 have completed fire protection modifications. All plants will be in compliance with 10 CFR 50.48 and Appendix R by 1989.
- (2) All plants licensed after January 1, 1979 have completed significant fire protection modifications and have fully implemented their fire protection programs.

Efforts to date at all plants reflect the consensus that has evolved from the staff's promulgation of rules and generic guidance. NRC, however, continues to evaluate the prudence and effectiveness of its regulations in the area of fire protection via an NRC-sponsored risk-based analysis of fire hazards and effects on safe plant operation/shutdown following an abnormal event.

2.4.3 Assessment

As indicated, the concept of defense in depth is primary in the fire protection program at U.S. nuclear power plants and is implemented by

- (1) preventing fires from starting
- (2) detecting fires quickly, suppressing those fires that occur, and limiting their damage
- (3) designing plant safety systems so that a fire that starts in spite of the fire prevention program and burns for a considerable time in spite of firefighting activities will not keep essential plant safety functions from being performed

The approach indicated above provides a substantial level of protection against fires. The fire brigade's use of typical protective equipment (turncoats, boots, gloves, hard hats, and self-contained breathing apparatus) provides a measure of protection against radioactivity as well as against fires and smoke. Training the fire brigade in the proper use of the protective equipment as stated previously can ensure its effectiveness in protecting personnel from the effects of exposure to radiation. Fire brigade training also includes instruction on the proper use of equipment and firefighting in all plant areas, including roofs. Thus, NRC fire protection practice in general and specific firefighting criteria are considered to be adequate at this time.

Hot fuel and graphite expelled from the core and lack of containment caused major fires at Chernobyl. Since light-water reactors have containments, any fire along with significant radiation, such as that resulting from a core-melt accident, should be confined to the containment building. Nevertheless, from a defense-in-depth perspective, the ability of firefighting teams to fight fires when radiation is present and to prevent the spread of fire to other units is noteworthy and deserves full recognition.

Under its research program, the NRC staff is investigating the risk significance and dominant sources of uncertainty associated with various fire risk issues. In addition to the effectiveness of manual firefighting, risk issues such as smoke control, control system interactions, interactions between seismic effects and fire, ability of equipment to survive under environmental conditions created by fire and fire-suppression activities, adequacy of fire barriers, and adequacy of analytical tools for the study of fire are all being investigated and evaluated for their risk significance. The Chernobyl event has provided the staff with valuable information on manual firefighting activities in high-radiation areas.

2.4.4 Conclusions and Recommendations

As a result of the accident and ensuing fires at Chernobyl, the NRC staff has reviewed the fire programs implemented by NRC rules, regulations, and guidelines and concludes that the programs provide an adequate level of defense in depth for all anticipated events.

Nevertheless, to confirm that the current provisions are adequate, it is recommended that the provisions for firefighting with radiation present be reviewed further. The NRC staff will determine the need for any additional improvements in fire protection and/or investigations in regard to fire risk when the ongoing fire research activities have been concluded. The lessons learned as a result of the accident at Chernobyl will be factored into the information disseminated to the industry as well as other potential actions.

CHAPTER 3

CONTAINMENT

The role of the containment vessel or containment building as a vital barrier to the release of fission products to the environment has been recognized for some time. The public safety record of U.S. nuclear power plants has been enhanced by applying the "defense-in-depth" principle, which relies on a set of independent barriers to fission-product release. The containment is one of these barriers. During the licensing review, applicants must demonstrate that the plant is designed to provide protection for the public in case of accidents up to the design-basis accident. The containment plus certain other engineered safety features, such as spray systems and filters, are designed and relied on to mitigate such events.

The NRC began to give attention to severe accidents even before the accident at Three Mile Island (TMI) and has increased its emphasis in this area since that accident. With regard to containments, one of the first requirements introduced after the TMI accident was intended to reduce the challenge to containment integrity from a hydrogen combustion. Thus, the smaller boiling-water-reactor containments, such as the Mark I and Mark II, were required to be inerted with a nitrogen atmosphere, effectively precluding the possibility of a hydrogen combustion; others were fitted with a hydrogen igniter system designed to burn any hydrogen in a controlled fashion, preventing substantial containment overpressure.

Recently, two groups of experts working on severe accidents assessed containment loading and containment performance. Two reports document that effort. The first report, NUREG-1079, "Estimates of Early Containment Loads From Core Melt Accidents," estimates the magnitude of pressure and temperature loads on containment associated with severe-accident sequences involving significant core damage and presents the results of studies to evaluate the potential effects of such phenomena on containment integrity. The second report, NUREG-1037, "Containment Performance Working Group Report," discusses the leakage rate of containment buildings as a function of increasing internal pressure and temperature.

The Chernobyl accident, with its absence of effective containment, has focused attention on the strengths and performance limits of the substantial containments for U.S. light-water reactors. It has led to added recognition of the significance of ongoing work on the issue of whether U.S. containments that were built using criteria based on design-basis accidents have adequate margins available to prevent the release of large quantities of fission products during severe accidents. Challenges include phenomena such as increased pressures from an uncontrolled hydrogen combustion or release of large quantities of noncondensable gases from core-concrete interactions. Venting the containment in case of certain severe accidents could be an effective way to preserve the long-term containment functional integrity and reduce the uncontrolled release

of radioactive material. The rest of this chapter summarizes the activities already in place in the areas of containment integrity and containment venting and addresses the need for additional work.

3.1 Containment Performance During Severe Accidents

The Chernobyl accident has further focused attention on containments and their performance under severe-accident conditions. This section reviews the highlights of relevant current practice and ongoing work and provides an assessment in relation to the Chernobyl experience.

3.1.1 Current Regulatory Practice

Containment design criteria are based on a set of deterministically derived challenges. Pressure and temperature challenges are based on the so-called design-basis loss-of-coolant accident. Radiation considerations are based on a postulated substantial core-melt accident. Also, external events such as earthquakes, floods, and tornados are considered in the design. The margins of safety provided in U.S. practice have been the subject of considerable research and evaluation, and these studies have indicated the ability of containment systems to survive pressure challenges of 2.5 to 3 times design levels. Severe-accident evaluations and research had progressed to the point that the Commission issued a Severe Accident Policy Statement in August 1985 (50 FR 32138) concluding that existing plants posed no undue risk to the public. However, the Commission pointed out that at each plant there will be systems, components, or procedures that are the most significant contributors to risk. Utilities should identify these contributors and develop appropriate courses of action, if and as needed to ensure acceptable margins of safety. Furthermore, the Commission stated that such examinations "will include specific attention to containment performance in striking a balance between accident prevention and consequence mitigation." Relative to new plant applications, the Commission expressed a desire that new plants should have a higher standard of safety than earlier designs, to cover postulated severe accidents. It also assigned the staff to evaluate the need for new containment performance criteria and, if the need exists, to formulate such criteria.

Both before and since the statement was issued, improvements in containment design have been studied for several plants and designs, including Zion, Indian Point, Limerick, and the GESSAR II standard plant. In addition, research has been conducted on containment challenges and performance (including estimates of uncertainties), and risk outlier searches are planned to be initiated in 1988, through individual plant evaluations by industry.

Improvements aimed at reducing the likelihood of containment challenges through improvements in combustible gas control have been promulgated through revisions to 10 CFR 50.44 ("Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors"); for response to anticipated transients without scram through 10 CFR 50.62 ["Requirements for Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants"]; and for station blackout through proposed rule changes (10 CFR 50.63, "Loss of Alternating Current Power," and General Design Criterion 17, "Electric Power Systems," of Appendix A to 10 CFR 50).

Strategies to enhance containment performance have been considered in developing emergency procedures. One such strategy is controlled containment venting to prevent catastrophic containment failure. Containment venting was considered in NUREG-0956 ("Reassessment of the Technical Bases for Estimating Source Terms") in evaluations of radionuclide releases and was also considered in evaluations supporting the draft NUREG-1150 ("Reactor Risk Reference Document") risk evaluations for plants having such a procedure in place. Containment venting is only one of a number of potential containment performance improvements considered by the staff and industry.

3.1.2 Work in Progress

Work on reassessing risks (draft NUREG-1150), containment performance improvements, and other emerging research is expected to indicate whether changes are warranted in predictions of accident probabilities, and containment system challenges and performance. This research has included substantial experimentation in areas such as containment loads and performance. Combustible gas phenomena, core-concrete interactions, and equipment survivability have also been evaluated. Implementation of the Severe Accident Policy Statement through the individual plant evaluation and containment performance improvement programs, utilizing emerging research, is expected to indicate whether risk outliers exist at specific plants that justify improvements in containment system performance. This implementation is the principal NRC program for identifying plant-specific severe-accident risk outliers and for implementing new requirements. For boiling-water reactors (BWRs) specifically, the staff, the Vermont Yankee licensee, the Pilgrim licensee, and the Boiling Water Reactor Owners Group are all involved in considering initiatives to improve the performance of BWR containments because of a perception that BWR containments designed by the General Electric Co. have a low probability of surviving core-melt accidents. Industry groups such as NUMARC* are involved in assessing containment integrity under severe accident conditions. The staff, through the Severe Accident Program and review of the Industry Degraded Core Rulemaking Program, has similarly evaluated pressurized-water-reactor containments. Activities are also being conducted to develop containment performance criteria for new plants. Foreign initiatives are in progress to improve containment system performance (such as those in Sweden, France, Germany, Finland, and Italy). The initiatives include potential design and procedural changes, one of which is containment venting.

3.1.3 Assessment

Research programs and regulatory initiatives to address the issue of containment performance and potential improvements during severe accidents are currently in progress at NRC. Generic vulnerabilities have been identified for some types of containments, but only a few detailed plant-specific containment assessments have been made. The systematic search for plant-specific problems that is due to begin in 1988 and the NRC's containment performance improvement program are, however, expected to provide more information on containment performance. On the basis of existing research and evaluation programs, new programs or initiatives are not needed as a result of the accident at Chernobyl.

*The Nuclear Management and Resources Council (a confederation of all 55 utilities with nuclear plants either in operation or under construction).

3.1.4 Conclusions and Recommendations

The Chernobyl event has graphically demonstrated the effect of containment performance on the overall risks of nuclear power plant operation. The Chernobyl event should strengthen the commitment by NRC and industry to implement the design and operational improvements needed to provide greater assurance of containment survival in severe accidents. Current programs are adequate for this purpose; new programs or initiatives are not needed.

3.2 Filtered Venting

The Chernobyl accident has focused attention on whether U.S. containments should be provided with filtered venting in order to mitigate the consequences of accidents of the type that occurred at Chernobyl.

3.2.1 Current Regulatory Practice

Venting as a containment strategy has been evaluated and is being incorporated into the emergency operating procedures for some boiling-water reactors. The staff has reviewed the technical bases for such procedures, including both combustible gas and pressure considerations. At the present time there is no regulatory requirement for plant-specific implementation of such a procedure, but it is up to individual utilities to provide for venting as a last resort to prevent gross containment failure on overpressure, and to prevent an excessive buildup of hydrogen in the containment, for accident conditions more severe than were considered in the original containment design. No analogous venting procedures or guidelines exist for pressurized-water reactors.

3.2.2 Work in Progress

Containment venting has been evaluated as part of severe-accident research programs related to radionuclide release assessments (NUREG-0956) and risk assessments (draft NUREG-1150) for reactors that incorporate venting in their emergency procedures, and for generic studies of venting as a mitigation strategy. In addition, evaluations associated with implementation of the Commission's Severe Accident Policy have considered venting. The Advisory Committee on Reactor Safeguards has also expressed interest in venting as a severe-accident mitigation strategy. These evaluations and considerations were all being pursued before the Chernobyl accident occurred.

Filtered venting as a strategy to mitigate the consequences of some severe accidents is being considered and implemented in Sweden, France, and Germany. Venting can preserve containment integrity against excessive containment pressure buildup (slow to moderate) from the loads generated during the severe-accident scenario. At the same time, filtering retains most of the long-lived radioactivity, thus avoiding gross contamination of the areas surrounding the nuclear power plant and minimizing long-term health and economic effects. For example, the Swedish Government now requires venting through newly constructed filtered vents for some reactors, is evaluating others, and has established performance criteria for such systems. The performance criteria provide for automatic actuation and specific filter efficiencies. The transient pressure-relieving capabilities of the improvements vary from early relief for boiling-water reactors (BWRs) to late pressure relief for pressurized-water reactors

(PWRs). The French Government plans to install filtered vents on all PWRs. These vents would be activated manually as a measure to save a threatened containment from late overpressure failure.

For BWRs specifically, filtered venting is one such improvement that, if correctly implemented, could reduce the risk from severe accidents. Preferably, gases should be vented from the wetwell for elevated exhaust through the stack. Existing vents and piping may need upgrading to withstand venting pressures. Non-noble-gas fission products could be removed (i.e., filtered) through the use of sprays and suppression pools. No separate or new filter requirements have been identified. The combination of postaccident fission-product removal and elevated releases would be expected to significantly reduce the offsite consequences of the most-severe-accident sequences. The current strategy is to use existing controlled venting capabilities insofar as possible to mitigate such accidents.

In related activities, a nuclear industry group (the BWR Owners Group) has revised its guidance on emergency procedures for BWRs (BWROG, 1986), and the Electric Power Research Institute (EPRI) will direct an international program that includes experimental research on fission-product attenuation for a number of filtering schemes. The NRC intends to join these efforts. The revised guidance, which the staff is reviewing, includes additional guidance on venting. Individual utilities, however, are not required to implement such guidance and earlier versions of such guidance have not been universally adopted.

3.2.3 Assessment

Filtered venting is an existing strategy currently being evaluated as part of the Severe Accident Policy implementation program in the United States. Anticipated technical exchanges with regulatory and utility representatives in other countries that are occurring as a result of Chernobyl-related initiatives, potential U.S. BWR containment improvements, and the EPRI research are all expected to add to the information available on the effectiveness of filtered venting. Therefore, no new initiatives are needed at this time.

3.2.4 Conclusions and Recommendations

Filtered venting as a means of limiting the offsite consequences of core-melt accidents is being pursued in a number of countries and is being examined in the United States. The U.S. programs of severe-accident research and implementation will, however, be enhanced by the EPRI research and anticipated technical exchanges with representatives from other countries (such as Sweden, France, and Germany) that are implementing filtered venting.

CHAPTER 4

EMERGENCY PLANNING

A number of facts about the Chernobyl accident bear on emergency planning and preparedness around U.S. commercial nuclear power plants. The implications of the Chernobyl accident and the Soviet response for four aspects of U.S. emergency planning, namely: (1) size of the emergency planning zone, (2) medical services, (3) ingestion pathway measures, and (4) decontamination and relocation, are examined in this chapter.

In drawing a nexus between the Soviet response to the Chernobyl accident and emergency planning implications for U.S. plants, contrasts and differences in four areas should be noted. First, there is a substantial difference in the emergency planning base. After the accident at Three Mile Island, large resources were expended to improve emergency planning and response capabilities around U.S. plants. In contrast, although some prior planning appears to have existed in the Soviet Union, perhaps for civil defense, there is little indication that the Soviets have comparable site-specific emergency plans for the general public around their nuclear power plants. Despite this, the Soviets mounted a large and generally effective ad hoc response.

Second, the specifics of the Chernobyl release are unique to the RBMK design. The amounts of radioactive material released from U.S. plants could be as severe, but for many accident sequences, would be considerably less because, among other things, U.S. plants have substantial containments. In addition, although low-probability, fast-moving accident sequences are possible, severe accidents at U.S. plants would, in general, progress more slowly, resulting in longer warning times before release.

Third, some aspects of the Chernobyl evacuation defy comparison with similar aspects at U.S. plants because of economic and societal differences. For example, the Soviets had to assemble 4000 buses and trucks for the Chernobyl evacuation, whereas, in the United States most people have access to private transportation and necessary alternative transportation is preplanned around U.S. nuclear power plants.

Finally, issues such as offsite decontamination and long-term relocation involve matters whose scope and timing typically extend beyond the initial or "emergency" phase of an accident response. Although these matters have a lesser degree of urgency in the early phases of an accident, and hence may not require specific detail in offsite emergency plans for U.S. plants, planners should keep in mind that these are important factors in the long-term response to a significant radiological release.

4.1 Size of the Emergency Planning Zones

The Chernobyl accident has focused attention on the adequacy of the size of emergency planning zones around U.S. commercial nuclear power plants.

The Soviets evacuated a total of about 135,000 people as well as considerable farm livestock from Pripyat, Chernobyl, and other towns and villages within 30 kilometers (18 miles) of the Chernobyl nuclear power plant. This evacuation appears to have taken place in several stages, beginning for the approximately 45,000 residents of Pripyat about 36 hours after the initial release, and extending over several days to a week. The whole-body radiation dose to the majority of individuals did not exceed 25 rem, although about 24,000 persons in the most severely contaminated areas are estimated to have been exposed to whole-body doses in the range of 35-55 rem. The population of Pripyat was initially sheltered as a protective measure and then evacuated when radiation readings increased. In addition to radiation considerations, logistics and contamination control influenced the timing of the evacuation. Despite an apparent lack of site-specific planning, the Soviets mounted a large and generally effective ad hoc response making use of some aspects of civil defense planning. The high initial plume height contributed to relatively low initial dose rates in the immediate vicinity. In addition, efforts by the Soviets to prevent rainfall in the immediate vicinity (by cloud seeding other areas) and the spraying of a chemical polymer on evacuation routes to minimize resuspension of deposited activity were also beneficial. The Soviets took ingestion pathway protective measures within the 30-kilometer zone and well beyond. Ingestion pathway protective measures were also taken in several Soviet bloc countries, in Scandinavia, and in Eastern and Western Europe.

4.1.1 Current Regulatory Practice

Emergency planning is currently required under 10 CFR 50.47 for all U.S. commercial nuclear power plants for two concentric zones having radii of approximately 10 and 50 miles (except for plants with power levels below 250 Mwt and for gas-cooled reactors, which have smaller zones). The inner zone, referred to as the plume exposure pathway emergency planning zone (EPZ), is one in which the principal sources of exposure would come from the radioactive plume and from material deposited on the ground. The outer zone, referred to as the ingestion exposure pathway EPZ, is one in which the principal exposure would come from ingestion of contaminated water or foods such as milk and fresh vegetables. The sizes of these zones were determined from considerations given in NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Plans." These specifically included consideration of the accident risks from the complete spectrum of severe-accident releases given in WASH-1400, "Reactor Safety Study." In addition, a distance of 10 miles was also chosen for the inner zone based on the conclusion that "detailed planning within 10 miles would provide a substantial base for expansion of response efforts in the event that this proved necessary" (NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," page 12).

Requirements for emergency planning in the United States include the capability to alert and notify the population in the plume exposure pathway EPZ within 15 minutes of the decision by responsible public officials that protective actions are necessary. It is the licensee's responsibility to (1) classify the emergency (unusual event, alert, site area emergency, or general emergency) in accordance with the potential risk to the public health and safety; (2) notify offsite authorities; and (3) recommend any public protective actions. The

implementation of the onsite planning requirements is exercised annually by the licensee. It is the offsite authorities' responsibility to (1) determine the proper protective actions, (2) alert and notify the general public within 15 minutes of a decision to recommend protective actions, and (3) assist the general public in carrying out the protective actions. The implementation of the offsite planning requirements is exercised every 2 years by the offsite authorities.

4.1.2 Work in Progress

Regarding EPZ size, a major NRC research effort began about 1981 to obtain a better understanding of radionuclide transport and release under severe-accident conditions. This research on radionuclide release includes the development and application of new computer codes for core-melt phenomena and containment performance. It also includes an extensive review effort by peer reviewers and industry groups, as well as independent assessment under the auspices of the American Physical Society. The report explaining and detailing this revised methodology to calculate accident radionuclide release was published in July 1986 as NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms." Revised risk profiles which apply this methodology were published for comment as draft NUREG-1150, "Risk Reference Document," in February 1987. This document attempts to use the new source term information to provide insights about (1) how offsite doses would be expected to vary with distance for the plants analyzed and (2) the relative effectiveness of different offsite protective actions at various distances from the plants.

4.1.3 Assessment

One difficulty in assessing the implications of emergency actions taken at Chernobyl for U.S. commercial nuclear power plants is the vast difference in the emergency planning base between the United States and the Soviet Union. After the accident at Three Mile Island, large resources were expended in the United States to improve site-specific and generic emergency planning capabilities. Utility, State, local, and Federal emergency plans were developed, reviewed, and exercised. Alert and notification systems have been designed, installed, and tested within the plume exposure pathway EPZs (10-mile radius) for all operating U.S. plants. The populations within the plume exposure pathway for U.S. plants are annually provided with informational materials that are to be used in the event of an emergency. These materials contain protective actions that will be taken and include telephone numbers for public inquiries.

In contrast, there is little indication that the Soviets have comparable site-specific emergency plans for the general public around their nuclear power plants. Although some prior planning existed, perhaps for civil defense, Soviet authorities indicated that many of the protective actions taken were ad hoc measures. Although a severe accident in the United States could require some ad hoc measures to be taken, a detailed planning base exists to facilitate implementation of any necessary protective actions.

Another difficulty in assessing the implications of the Chernobyl accident is that the specifics of the Chernobyl release are unique to the RBMK design. The amounts of radioactive material released from U.S. plants could be as severe but would for many accident sequences be considerably less because, among other things, U.S. plants have substantial containments. In addition, although

low-probability, fast-moving accident sequences are possible, severe accidents at U.S. commercial nuclear plants would generally progress at a slower pace, resulting in longer warning times before release.

With regard to the issue of EPZ size, the Soviets evacuated the population out to 18 miles, or roughly twice the distance for which an evacuation capability is required to be demonstrated in the United States. Similarly, measures were taken to prevent ingestion of foodstuffs, milk, and water at distances considerably greater than the 50-mile ingestion exposure pathway in the United States. This might imply that the U.S. EPZs are too small. However, examination of the background leading to the U.S. requirements leads to a different conclusion.

The sizes of the EPZs were derived from accident considerations, including the severe accidents studied in the Reactor Safety Study (WASH-1400). The most severe and most unlikely accidents studied in WASH-1400 involve releases of radioactivity that are comparable in magnitude to that which was actually released at Chernobyl. WASH-1400 estimated that most core-melt accidents would result in significantly lower releases. The 10-mile and 50-mile EPZs were chosen as a planning basis to demonstrate a capability and to provide emergency plans with the flexibility of dealing with a broad range of accident releases, rather than being based solely on a single highly unlikely event, such as the worst case. It was recognized that protective actions might need to be taken beyond these planning zone distances for the most severe releases. NUREG-0654 clearly notes:

The choice of the size of the Emergency Planning Zones represents a judgement on the extent of detailed planning which must be performed to assure an adequate response base. In a particular emergency, protective actions might well be restricted to a small part of the planning zones. On the other hand, for worst possible accidents, protective actions would need to be taken outside the planning zones.

Consequently, a release magnitude similar to the one associated with Chernobyl and the possibility that ad hoc actions beyond the planning zone boundaries might be needed for very unlikely events were considered and have been factored into the development of U.S. requirements, including the sizes of the EPZs.

4.1.4 Conclusions and Recommendations

The Chernobyl accident and the Soviet response do not reveal any apparent deficiency in U.S. plans and preparedness, including the 10-mile plume exposure pathway EPZ size and the 50-mile ingestion exposure pathway EPZ size. These zones provide an adequate basis to plan and carry out the full range of protective actions for the populations within these zones, as well as beyond them, if the need should arise. Any changes in EPZ sizes should be based on revised insights coming from current U.S. research on severe-accident releases.

4.2 Medical Services

The Chernobyl accident has focused attention on (1) the adequacy of the U.S. Government's policy on potassium iodide (KI) and (2) the adequacy of medical services around U.S. nuclear power plants.

At Chernobyl, KI was distributed to school children within about 6 hours of the accident, and to the entire population of Pripjat the morning of the following day; ultimately it was given to the population in the 30-kilometer zone and other areas. The Soviets report no serious adverse reactions to KI. Polish authorities also distributed KI to the population in parts of eastern Poland.

Two hundred and three plant and response personnel suffered acute radiation sickness; many of these people also had other injuries, primarily burns. By the end of July, 29 had died. A specialized medical team, which arrived within 12 hours of the accident, examined 350 persons and performed about 1000 blood analyses. To provide medical care for the evacuees during the first few days after the accident, 450 brigades (made up of 1240 physicians, 920 nurses, and 360 physicians' assistants) and more than 3400 other personnel were mobilized. These medical personnel were not all at the same location. No persons from the general public were found to be victims of acute radiation sickness.

4.2.1 Current Regulatory Practice

The U.S. Government policy on distribution of KI around nuclear power sites for use as a thyroid-blocking agent was published in the Federal Register (50 FR 30258) on July 24, 1985. Although the stockpiling and use of KI are recommended for emergency workers and institutionalized individuals, the U.S. Government's position with regard to the predistribution or stockpiling of KI for use by the general public is that it should not be required. The policy statement elaborates:

While valid arguments may be made for the use of KI, the preponderance of information indicates that a nationwide requirement for the predistribution or stockpiling for use by the general public would not be worthwhile. This is based on the ability to evacuate the general population and the cost effectiveness of a nationwide program which has been analyzed by the NRC and DOE National Laboratories (NUREG/CR-1433).^{*} While the use of KI can clearly provide additional protection in certain circumstances, the assessment of the effectiveness of KI and other protective actions and their implementation problems indicates that the decision to use KI (and/or other protective actions) should be made by the states and, if appropriate, local authorities on a site specific basis.

With regard to medical services around nuclear power plants, NRC licensees are required to provide an onsite first-aid capability for onsite personnel and emergency workers and to arrange for local (primary) and backup hospitals that have the capability for evaluating radiation exposure and uptake, including assurance that persons providing these services are adequately prepared to handle contaminated individuals.

^{*}"Examination of the Use of Potassium Iodide (KI) as an Emergency Protection Measure for Nuclear Reactor Accidents," October 1980.

For offsite members of the general public, Commission policy is contained in a statement published in the Federal Register on September 17, 1986 (51 FR 32904) titled, "Emergency Planning - Medical Services." The Commission stated that its regulation required that preaccident arrangements be made for medical services for individuals who might be severely exposed to dangerous levels of off-site radiation following an accident at a nuclear power plant. Such arrangements would include (1) identification of the capacities, special capabilities, or other unique characteristics of the listed medical facilities; (2) a good-faith reasonable effort by licensees or local or State governments to facilitate or obtain written agreements with the listed medical facilities and transportation providers; (3) provision for making available necessary training for emergency response personnel to identify, transport, and provide emergency first aid to severely exposed individuals; and (4) a good-faith reasonable effort by licensees or State or local governments to see that appropriate drills and exercises are conducted which include simulated severely exposed individuals. The Federal Emergency Management Agency (FEMA) and NRC staff have prepared guidance for implementation of this policy (FEMA, 1986). The guidance should be implemented within about 1½ years.

4.2.2 Work in Progress

As mentioned above, State and local governments and licensees will be implementing the Commission's new policy statement on medical services. In addition, the medical community is addressing implications of the Chernobyl accident on the area of medical response through its traditional mechanisms; for example, the American Medical Association (AMA) held an international conference on nonmilitary radiation emergencies in November 1986. In addition, the AMA's Radiation Advisory Committee met with representatives of selected Federal and private agencies in Washington, D.C., on January 28, 1988 to discuss methods for increasing physicians' knowledge about, and involvement in, activities of Federal agencies that are involved in planning for and responding to radiation emergencies.

4.2.3 Assessment

The Soviets credited the use of KI by the Pripjat population with keeping thyroid exposures within the permissible limits (stated as less than 30 rad) for 97% of the 206 evacuees tested at one relocation center. They said there were no serious adverse reactions from the use of KI (USSR, 1986). In Poland, the use of KI was reported to have reduced the potential thyroid dose to children by factors of 6 to 10. However, one Polish survey of 46 persons also showed that administration of KI had no effect in reducing the thyroid dose for about one-third of this group (DOE, 1987).

The policy statement of the U.S. Government acknowledges the effectiveness of using KI for certain individuals. These are principally those individuals, such as emergency workers or institutionalized individuals, who may be exposed to the release over an extended period. For members of the general public, however, these conditions generally are not applicable, because evacuation is generally feasible and, when carried out, is more effective in dose reduction than administration of KI, since it can reduce the dose for all body organs and not merely the thyroid gland. Because of these considerations, the policy statement concludes that a nationwide requirement for the predistribution or stockpiling for use by the general public would not be worthwhile. It further

concludes that the decision to use KI should be made by the States and, if appropriate, by local authorities on a site-specific basis. (Tennessee has predistributed KI, and Alabama has stockpiled it.) The apparently successful use of KI at Pripyat does not alter the validity of guidance that recognizes that evacuation of the general public in the affected area could result in a greater overall dose reduction.

The Soviets mounted an impressive and effective medical response to the Chernobyl accident. Fortunately, the United States has not had to respond to a radiation medical emergency of that magnitude, although the U.S. medical community has responded to other sizeable medical emergencies at home and abroad.

In the United States, the present and future medical response capabilities in the regions around commercial nuclear power plants were described in Section 4.2.1 above. The accident at Chernobyl emphasizes the prudent nature of such measures. A national response to a Chernobyl-type accident would be coordinated through the Federal Radiological Emergency Response Plan (50 FR 46542), which has the resources of the Radiation Emergency Assistance Center/Training Site (REAC/TS) at Oak Ridge, Tennessee and the National Disaster Medical System (NDMS) headquartered in Rockville, Maryland.

The REAC/TS has its own response team to a radiation emergency, trains emergency medical personnel, and maintains a computerized registry of approximately 1650 people who have been trained at its center; the registry includes 650 physicians. The NDMS has four medical assistance teams (MATs) that can respond to radiological emergencies; these teams are augmented by health physicists from the Food and Drug Administration, the Department of Energy, and other sources. Currently, the NDMS has enrolled in its program 76,478 hospital beds in 965 non-Federal medical institutions. Its goal is to have 100,000 non-Federal beds and 150 MATs enrolled in its program. The NDMS also has a goal to train all of its teams in the handling of patients exposed to radiological, biological, or chemical contaminants.

4.2.4 Conclusions and Recommendations

The apparently successful use of potassium iodide by the Soviets does not alter the validity of U.S. Government policy that predistributing or stockpiling potassium iodide for use by the general public should not be required. Rather, this decision should be made by individual States and by local authorities.

Further, the staff concludes that the present arrangements and future plans for medical services around U.S. commercial nuclear power plants are adequate. The national capability is both substantial and growing. Also, the international offers of medical support to the Soviet Union following the Chernobyl accident demonstrate that the U.S. regional and national medical response can be augmented, if necessary, by a response from the international medical community.

4.3 Ingestion Pathway Measures

The Chernobyl accident focuses attention on (1) the adequacy of U.S. standards for the ingestion of radioactive materials in food and water (and the mechanisms for adapting those standards to changing conditions) and (2) the adequacy of U.S. plans and preparedness for taking measures to protect the public from the ingestion of hazardous levels of radioactive materials in food and water.

Soviet authorities initiated measures to protect the public from receiving unacceptably high levels of radiation through consumption of radioactively contaminated food. These measures were taken in two stages. Immediately after the accident, standards were promulgated governing the permissible content of iodine-131 (I-131) in milk and milk products. Cows were placed on stored feed. Similar standards were introduced governing the I-131 content in meat, poultry, eggs, berries, and raw materials used for medical purposes. On May 30, 1986, standards for cesium-134, cesium-137, and rare earth isotopes were issued to reflect the changes in the composition of the radiation contamination at that time. The permissible whole-body and internal organ dose in these standards was 5 rem committed dose.

Derived intervention levels were adopted in the United States for imported foods and in most European countries for imported and domestic foods. These levels differed from one country to another and may have contributed not only to public confusion and misunderstanding but also to possible international economic and trade problems. On an international basis there is a need to set scientific criteria for consistency in radiation dose levels at which intervention may be recommended.

4.3.1 Current Regulatory Practice

The Food and Drug Administration has published action levels (47 FR 47073) to provide State and local agencies with recommendations for taking protective action if an incident should cause contamination of human food and animal feeds. These can be used to determine whether levels of radiation encountered in food after a radiological incident warrant protective action and to suggest appropriate action that may be taken if action is warranted. In the United States, the State and local governments have primary responsibility for taking protective actions to protect the public from the ingestion of contaminated food.

The U.S. response to a radiological emergency is governed by the Federal Radiological Emergency Response Plan (FRERP) (50 FR 46542). The FRERP establishes a mechanism for a coordinated Federal assessment of the consequences of a nuclear accident occurring within the United States. It also specifies authorities and responsibilities of each Federal agency that may play a significant role during a radiological emergency. The FRERP includes the Federal Radiological Monitoring and Assessment Plan for use by Federal agencies with radiological monitoring and assessment capabilities.

The FRERP recognizes that State or local governments have primary responsibility for determining and implementing any measure to protect the public. Therefore, one of the principal areas in which the Federal Government assists State and local governments is in advising them on protective action recommendations for the public.

4.3.2 Work in Progress

The Federal Emergency Management Agency, in conjunction with a working group of the Federal Radiological Preparedness Coordinating Committee, with representatives from the NRC, the Food and Drug Administration, the U.S. Environmental Protection Agency, and the U.S. Department of Agriculture, has developed guidance on plans and exercises pertaining to the ingestion pathway. The guidance

provides planning considerations for protecting the human food chain, including animal feed and water, which may be contaminated following a radioactive release resulting from an accident at a commercial nuclear power plant. The document, Guidance Memorandum IN-I, "The Ingestion Pathway," was issued on February 26, 1988, and is directed to State and local government emergency planners with responsibilities pertaining to ingestion exposure pathways (within a 50-mile radius of a nuclear power plant) as well as Federal evaluators. The guidance should be reflected in State and local radiological emergency plans by the end of calendar year 1988 and demonstrated in the next exercise that includes testing of ingestion-related measures.

4.3.3 Assessment

The Protective Action Guides for human food and animal feed are minimum considerations to apply during and after an accident, although Federal, State, and local governments can modify them. The Federal mechanism for providing recommendations to State and local governments is the Federal Radiological Emergency Response Plan. The adequacy of the Federal guidance will be reviewed through evaluations provided by U.S. research on radionuclide release and dispersion, as well as through other coordination activities by appropriate Federal agencies. To date, it appears that the existing Federal guidance will provide adequate protection for members of the general public from contaminated food.

The U.S. Government does have plans and preparedness measures in place to protect the public from ingesting hazardous levels of radioactive materials in food. The adequacy of the ingestion planning distance (50 miles) should be reexamined, taking into consideration U.S. source term research.

4.3.4 Conclusions and Recommendations

The guidance, planning, and preparedness around U.S. nuclear power plants for taking protective measures that deal with the ingestion pathway appear to be adequate. Ingestion pathway measures should be reexamined, in cooperation with the Federal Emergency Management Agency, the U.S. Department of Agriculture, and the Department of Health and Human Services, as part of the application of U.S. source term research. It should be noted, however, that past and current research results indicate that the interdiction of foodstuffs at large distances (beyond 50 miles) may be necessary for very large, low-probability source terms. This was recognized in NUREG-0396 and is the reason that the 50-mile ingestion exposure emergency planning zone is recognized as a planning base that can be expanded if the need arises.

4.4 Decontamination and Relocation

The Chernobyl accident focuses attention on the adequacy of U.S. plans and preparedness to mount large-scale decontamination and relocation efforts.

The Soviets evacuated and relocated 135,000 people and 19,000 cattle from an area within a 30-kilometer radius of the Chernobyl nuclear power plant. Apparently, some of these people have been permanently relocated.

The 30-kilometer area was subdivided into three zones of 0-3, 3-10, and 10-30 kilometers. All transport was strictly monitored for radioactivity, and decontamination points were established. At the boundary of each zone, workers were

transferred from one vehicle to another to reduce transmission of radioactive materials. The Soviets are decontaminating large areas of cropland, forest, orchard, etc., and are also taking measures to prevent or minimize contamination of the watershed and the Pripyat River.

4.4.1 Current Regulatory Practice

In the United States, onsite decontamination is the responsibility of the utility. Offsite decontamination would be conducted subject to the Environmental Protection Agency (EPA) operational guidelines for external exposure and food pathways. To enable re-entry, EPA is preparing proposed formal guidance for Federal agencies. To date, large-scale environmental decontaminations have been associated with weapons tests and have been handled on an ad hoc basis by the Department of Energy (DOE).

4.4.2 Work in Progress

The Federal Emergency Management Agency is responsible for coordinating the Federal aspects of relocation efforts with the States, through the Federal Radiological Emergency Response Plan (FRERP) mentioned previously. The March 1984 Federal field exercise (FFE-1) at the St. Lucie Plant in Florida tested the FRERP up to the re-entry and recovery phase of emergency consequences management. A report on the 1985 Federal "table-top" exercise focused on decontamination and recovery. The 1987 Federal field exercise (FFE-2) at the Zion plant simulated a contaminated area off site resulting from an accidental release, and exercised the measurement and assessment of this area, as well as interdiction of such foods as milk and fresh vegetables. Decontamination and relocation planning are concerns for continuing exercise activity in the U.S. Radiological Emergency Preparedness program. This is an activity that should receive further attention by appropriate U.S. agencies if, and when, additional information becomes available from the Soviets about their experience with the consequences of the accident at Chernobyl.

Research on large-scale environmental decontamination efforts is currently being conducted in the Pacific in conjunction with the rehabilitation efforts for Eniwetok Atoll, by Lawrence Livermore National Laboratory, under contract with DOE. Several efforts and reports focus on decontamination limits, but no criteria have been established.

4.4.3 Assessment

Decontamination techniques employed by the Soviets, including decontamination of personnel, appear to be similar to those used in the United States in support of the nuclear weapons testing program, the accident at Three Mile Island, and interdiction related to chemical spills. Desert areas and coral atolls have been decontaminated, but the United States has little experience in the large-scale decontamination of forests and orchards or croplands with the purpose of restoring viability and productivity to the land. The Soviets are using special agrotechnical and decontamination measures designed to enable contaminated lands to be used one day. These methods include changing the system of soil cultivation, the use of special polymer dust-suppression compounds, and changing harvesting and crop-processing methods.

Again, as before, strict application of the Soviet experience with decontamination from the Chernobyl accident to the United States is not possible. Decontamination capabilities in the United States will have to be examined in light of U.S. radionuclide source terms. However, from the Soviet experience, there will be much to learn about the technology of decontamination. This information will transfer over an extended period of time as the Soviets become able to assess the effectiveness of the measures they have taken.

Similarly, the Soviet relocation effort will have to be viewed in the light of U.S. source term research.

4.4.4 Conclusions and Recommendations

The effectiveness of Soviet decontamination and relocation efforts should be examined as the data become available. The U.S. capabilities should be examined within parameters provided by U.S. source term research.

CHAPTER 5

SEVERE-ACCIDENT PHENOMENA

The highly energetic reactivity excursion accident at Chernobyl mechanically disrupted the core, rapidly vaporized the water coolant with which the fragmented fuel came into contact, and generated combustible hydrogen by chemical reaction of core materials (notably zirconium) and water at the high temperatures reached in the accident. Because of basic design differences between the RBMK reactor of Chernobyl and U.S. light-water reactors, the specific accident mechanisms involved at Chernobyl have no exact parallel in U.S. reactors. However, these Chernobyl phenomena are assessed for implications by analogy for radionuclide releases, steam explosions, and combustible gas generation and deflagration control in U.S. reactors.

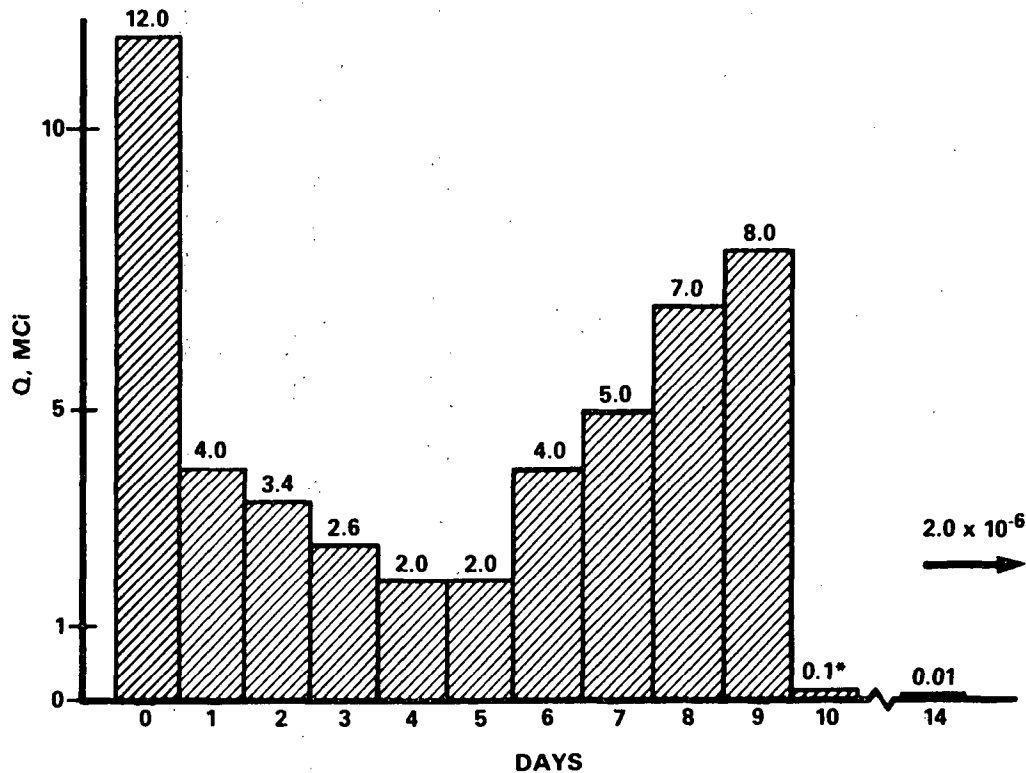
To assess the implications of radionuclide release one must examine the possibility that the United States may need to extend its research on such release--for example, to enhance understanding of mechanical disruption mechanisms, in addition to accident processes dominated by core melting, on which the current U.S. research effort on radionuclide release focuses. Questions that the Chernobyl accident may raise about the adequacy of U.S. safety measures dealing with steam explosion and combustible gas control are assessed.

5.1 Source Term

The Chernobyl accident led to a large, energetic release of radionuclides to the environment over a period of 10 days. Early Soviet estimates indicated that essentially all of the noble gases, about 10 to 20% of the volatile elements (iodine, cesium, and tellurium), and about 3 to 6% of the remaining elements in the reactor core were released. A later study sponsored by the Department of Energy concluded that the releases of volatile elements were larger than the early Soviet estimates (30 to 50% for cesium and tellurium and 40 to 60% for iodine) and that the initial energetic release was a smaller fraction of the total release (Warman, 1987).

The release that took place on the first day of the accident (April 26, 1986) occurred as a highly energetic release (with an initial plume height of about 1000 to 2000 meters) without any warning. As reported by the Soviets (USSR, 1986) (see Figure 5.1), the first day's releases were approximately 25% of the total release. In the days that followed, the daily release rate fell steadily to the end of the sixth day, then rose to the end of the tenth day. After the tenth day the release rate suddenly dropped, because of the actions taken at the damaged reactor, to less than 1% of its initial average value and continued to decline thereafter.

Although the Chernobyl reactor had significantly different design and operational characteristics than those of the U.S. commercial light-water reactors, the characteristics of the Chernobyl source term (timing, energy, magnitude, and



*recalculated for May 6, 1986.

Figure 5.1 Daily radionuclide release into the atmosphere from the damaged unit (not including noble gases)

Source: Soviet experts at the Vienna meeting (USSR, 1986)

other characteristics of radionuclide release) as described above raise several issues related to the state-of-the-art understanding of severe reactor accident source terms. Broadly, the issues are:

- (1) Do the magnitude and other characteristics of the Chernobyl source term confirm or contradict those that would be predicted for U.S. light-water reactors, considering current NRC methods?
- (2) Are there radionuclide release and in-plant transport mechanisms identified in the Chernobyl accident that may not have been considered in staff evaluations?

5.1.1 Current Regulatory Practice

Radionuclide releases to the environment from reactor accidents ("source terms") are deeply embedded in the regulatory policy and practices of the NRC. Consideration of source terms entered the regulatory process via the evaluation of postulated accidents (so-called design-basis accidents) in the safety review to

assess (1) plant mitigation features and (2) the suitability of a site. The Code of Federal Regulations (10 CFR 100) requires that an accidental fission-product release be postulated to occur within containment and that its radiological consequences be evaluated assuming the containment to be leaking at its maximum permissible rate. The release into the containment is derived from the 1962 Atomic Energy Commission (AEC) report TID-14844 ("Calculation of Distance Factors for Power and Test Reactor Sites"), and consists of 100% of the noble gases, 50% of the iodines (half of which are assumed to deposit on interior surfaces very rapidly), and 1% of the remainder of the core. With regard to this release, a footnote to 10 CFR 100 states that it "would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

Use of the TID-14844 release has not been confined to a determination of plant and site suitability alone. The regulatory applications of this release cover a wide range, including the basis for (1) the radiation accident environment for which safety-related equipment should be qualified, (2) postaccident habitability requirements for the control room, (3) performance of important fission-product cleanup systems such as sprays and filters, and (4) post-accident sampling systems and accessibility.

The first systematic evaluation of the probabilities and consequences of severe accidents, including their source terms, was given in a 1975 AEC report, WASH-1400,* "Reactor Safety Study: An Assessment of Risk in U.S. Commercial Nuclear Power Plants." The spectrum of releases given there includes releases resulting from core melt and containment failure. The most severe release categories from WASH-1400 entail releases of volatile fission products of comparable or greater magnitudes than were released at Chernobyl, although the releases of low-volatility species were higher for Chernobyl.

Source terms from severe accidents (beyond-design-basis accidents) entered into regulatory usage over the next several years, accelerated by the Three Mile Island accident and its aftermath. Current regulatory applications of severe-accident source terms rely to a large extent on the insights of WASH-1400, and include (1) the basis for the sizes of emergency planning zones for all plants, (2) the basis for staff assessments of severe-accident risk given in plant environmental impact statements, and (3) the basis for staff prioritization of generic safety issues, unresolved safety issues, and other regulatory analyses. Source term assessments based on WASH-1400 methodology appear in many probabilistic risk assessment studies performed to date.

Hence, any insight gained with regard to source terms has the potential for affecting a broad spectrum of regulatory applications.

5.1.2 Work in Progress.

Source term estimates under accident conditions began to be of great interest shortly after the Three Mile Island accident when it was observed that only

*WASH-1400 was designated as NUREG-75/014 by the NRC when it succeeded the Atomic Energy Commission.

relatively small amounts of iodine were released compared with the amount of noble gases. This led a number of observers to claim that severe-accident releases were much lower than previously estimated.

A major NRC research effort began about 1981 and has been under way since then to obtain a better understanding of fission-product transport and release mechanisms under severe-accident conditions. Severe-accident research has also been sponsored by the Electric Power Research Institute and under the industry's Industry Degraded Core Rulemaking Program. This research has included a very large and extensive staff and contractor effort, involving a number of national laboratories and private companies, and has resulted in the development and application of several new computer codes to examine core-melt phenomena and containment loadings. Work by the NRC staff has also included significant review efforts by peer reviewers, foreign partners in NRC research programs, industry groups, and the general public. An independent evaluation of the NRC results was also performed under the auspices of the American Physical Society. An NRC report assessing and detailing this revised methodology to calculate accident source terms was published in July 1986 as NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms." The staff is revising risk profiles for five operating U.S. light-water reactors which will utilize the new methodology. This effort has been issued for comment as draft NUREG-1150, "Reactor Risk Reference Document."

Ten regulatory areas affected by knowledge about source terms have been identified in SECY 86-76 (February 28, 1986), which describes the staff's plans for implementing the Commission's Severe Accident Policy Statement (50 FR 32138) as well as the staff's intended use (in regulatory applications) of information about source terms.

5.1.3 Assessment

A comparison of the characteristics of the Chernobyl release with regard to quantities released, timing, duration, and release energy with those predicted for U.S. light-water reactors is useful.

The total quantity of fission products released from Chernobyl was large and is considered to be comparable with the quantities predicted to be released for the worst cases (those involving core melt with early containment failure or containment bypass) studied for U.S. light-water reactors using WASH-1400 as well as the most recent source term methodology. Many core-melt sequences for U.S. plants are predicted to result in considerably lower amounts of fission products released to the environment, chiefly because of the mitigating effects of the containment and other fission-product cleanup systems. In this regard, the report by the International Nuclear Safety Advisory Group (INSAG, 1986) has noted that the Chernobyl release represents a near worst case in terms of the risks of nuclear energy.

The Chernobyl release occurred with essentially no warning. This is considered unique to the RBMK design and is a consequence of its sensitivity to large reactivity-initiated accidents. Accident-sequence progression for U.S. reactors is estimated to occur more slowly. Although a small number of severe-accident sequences could progress rapidly, resulting in releases within a fraction of an hour from the onset of discernible off-normal conditions, the progression for most accidents is considered to take hours.

The energy of the Chernobyl release was unusual. Approximately 25% of the total release was in the initial plume, which also had sufficient energy to result in an initial plume height of about 1000 to 2000 meters. This is considerably in excess of the plume heights predicted for energetic severe sequences in U.S. plants, which are estimated to be about a few hundred meters. A release duration of 10 days is large compared with the predictions of WASH-1400. This may have been due to the exigencies of the WASH-1400 consequence model, however, which did not adequately model releases of greater duration. The newer source term methodology predicts longer duration releases, principally from interactions between core and concrete. Releases from U.S. plants are usually predicted to gradually decline as the core debris gradually cools. The release rate during the Chernobyl accident decreased rapidly in the first few days and then increased in the last few days, presumably because the materials deposited on the degraded core (as a part of the actions taken to manage the accident) acted initially to filter radionuclide releases and later as insulation that allowed core debris to heat up before cold nitrogen was used to cool the core permanently.

(1) Mechanisms Involved in Mechanical Releases

The Chernobyl accident was a reactivity-initiated accident (RIA). With an estimated average energy insertion in excess of 300 cal/g, the skewed power distribution in the core would have led to local regions in the core with much higher fuel enthalpies--perhaps 400 to 600 cal/g. On the basis of a relatively good understanding of RIAs (see MacDonald, 1980), an explosive core disassembly such as took place in Chernobyl would have been expected. Figure 5.2, for example, shows that UO_2 fuel with 400 to 600-cal/g energy deposition would be fully molten and at least partially vaporized because of fundamental properties of the material. Figure 5.3, from in-reactor RIA tests, shows that debris recovered from tests in excess of 300-cal/g energy deposition is indeed pulverized, facilitating rapid heat transfer associated with the generation of high pressures.

Although RIAs are relatively well understood, they have not been included in recent source term assessments because such large RIAs have been "designed out" of U.S. light-water reactors. However, mechanical releases related to the power excursion and to the mechanical core disassembly during the Chernobyl accident amounted to between 3 and 6% of the fuel material and the fission products contained therein. Because of the serious consequences of the Chernobyl accident, the potential for RIAs is being reexamined, and the mechanical processes involved in the dispersion of fuel material should be further investigated.

Although consideration of RIAs like the one that occurred at Chernobyl does not seem to be warranted for U.S. light-water reactors, other energetic events are possible in light-water reactors that might lead to mechanical releases. These events are high-pressure melt ejection, steam explosions, and hydrogen detonation. Although all of these events are being studied with regard to their likelihood of occurrence and their consequences, associated mechanical releases of fission products have not been quantified in current source term models, and the study of such releases has only just begun to receive attention. Because some of these phenomena appear to have played a dominant role in the releases at Chernobyl, it is very important to understand these phenomena more completely and to improve the modeling in NRC source term assessment codes.

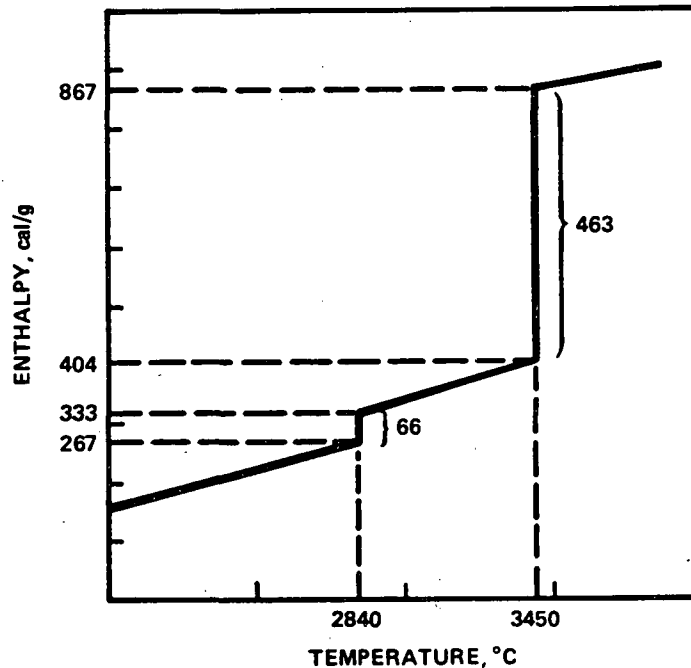


Figure 5.2 Enthalpy of UO₂

(2) Mechanisms Involved in the Late Enhanced Release

At this time, the staff does not completely understand mechanisms associated with the increased rate of release at Chernobyl which began about 6 days after the initial release and peaked on the 10th day (to almost the initial value). However, one or more of the following explanations may apply:

- (a) Increasing temperatures may have vaporized fuel and fission products in the debris. The increasing temperatures were probably caused by decay heat and the insulating effects of materials deposited on the debris (boron carbide, dolomite, clay, sand, and lead) and also by the graphite fire.
- (b) Enhanced gas flow from the hot debris may have resuspended particulate debris that had settled back into the core region after the initial release.
- (c) Enhanced oxidation (conversion of UO₂ to U₃O₈) or other chemical reactions involving carbon may have produced small particles of fuel material and fission products that were transported as aerosols.

Releases resulting from vaporization (item a) and resuspension (item b) have been considered in source term evaluations for U.S. light-water reactors, but release mechanisms involving chemical reactions (item c) are not included as models in current source term analytical methods, except in connection with interactions between molten core debris and concrete.

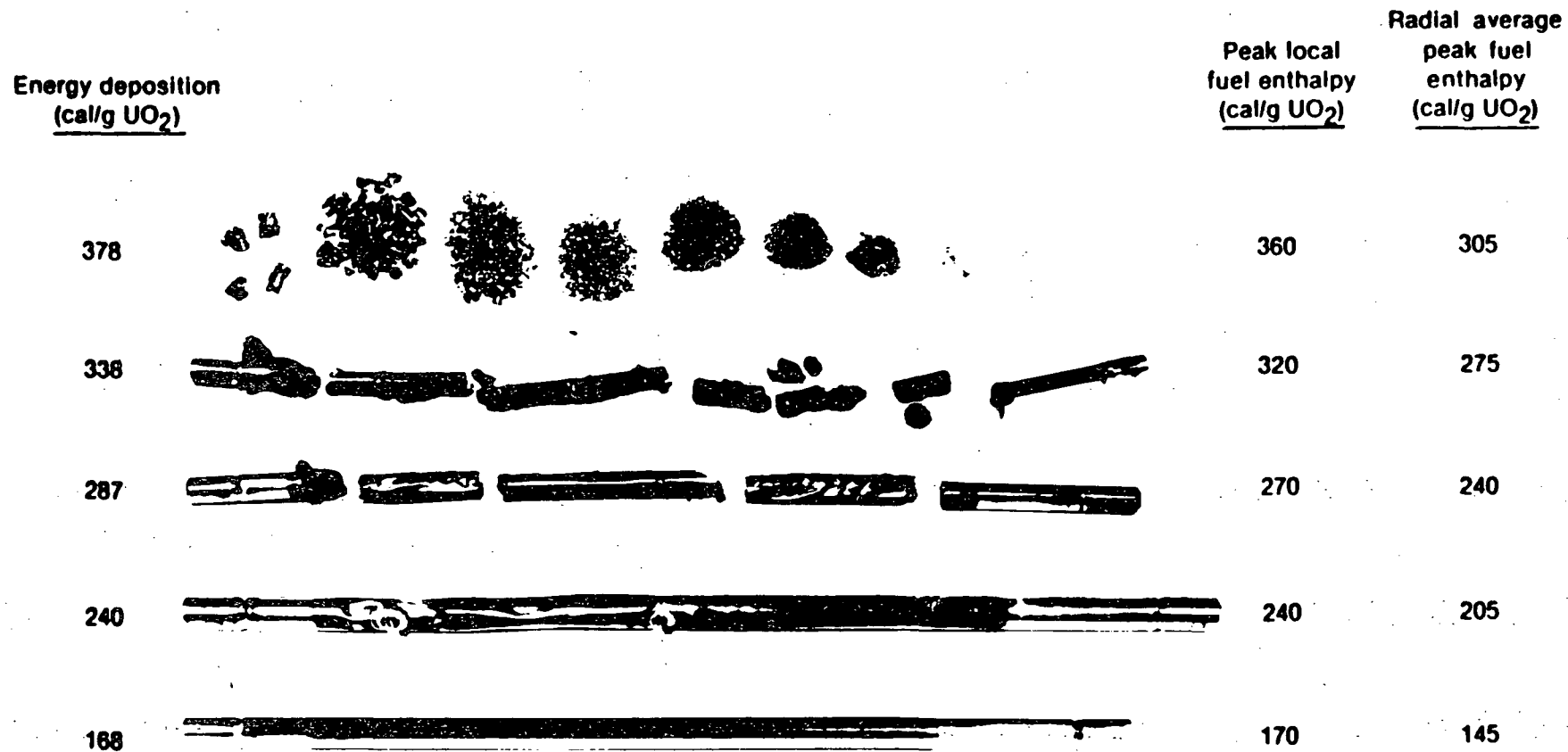


Figure 5.3 Fuel rod damage for various energy depositions in the SPXM rods tested in the CDC
Source: MacDonald, 1980

At even relatively low temperatures (e.g., around 1000°C), UO_2 can be further oxidized in the presence of oxygen to form U_3O_8 . When UO_2 in the form of solid fuel pellets or fragments oxidizes to U_3O_8 , a loose, powdery material is produced on the surface because U_3O_8 is 20% less dense than UO_2 . This powdery material would also contain fission products previously retained in the solid UO_2 , provided that the preceding temperature history had not produced complete release.

These fission products would be "stripped" off the surface in proportion to the amount of UO_2 converted to U_3O_8 . It is postulated that at Chernobyl the powdery U_3O_8 containing fission products was entrained in some increased gas flow, thus contributing to this second release. Air samples collected by aircraft were found to contain U_3O_8 particles and seem to support this hypothesis. However, the presence of U_3O_8 does not confirm this simple oxidation process because other chemical reactions involving carbon might have produced the same result and because UO_2 released to the atmosphere might have oxidized after its release.

Shortly before the accident at Chernobyl, NRC-supported research at Battelle Columbus Laboratories had shown that fission-product stripping could take place by thermal mechanisms. In that research it was found that fission products were released in proportion to the amount of UO_2 that was vaporized when temperatures were high enough (above 2000°C) to produce copious UO_2 vaporization. That is, fission-product releases were no longer proportional to their own vapor pressures but rather to the vapor pressure of UO_2 .

The stripping of fission products by the removal of the uranium oxide surface layers, whether by chemical or thermal mechanisms, is not currently modeled in NRC source term codes, and the Chernobyl accident underscores the importance of accounting for this mechanism. It should be kept in mind, however, that the transport of such released fission products as aerosols depends strongly on particle size and carrier gas flow and such fission products would not necessarily be carried into the atmosphere in other accident sequences.

(3) Other Observations

A number of other observations related to the source terms have been made as a result of the accident at Chernobyl. These are described below.

(a) Sudden Drop of Enhanced Release Rate

No confirmed explanation for the sudden drop in release rate after May 6 has been offered. However, three hypotheses have been offered.

- Nitrogen gas introduced under pressure (May 4 or 5) beneath the core succeeded in cooling the core and prevented further oxidation reactions.
- Graphite slumped about this time, and the slumping may have enhanced cooling or brought cool materials into the hot debris.
- During the phase of enhanced release (before May 6), parts of the core debris reheated as a result of residual decay and may have liquefied because of reduced heat loss through the molten cover provided by the

deposited materials. The liquefied debris eventually fell into lower pipe runs where it froze. Continued cooling flow of gas into the pipe runs may have prevented any further release from the quenched debris.

- According to the Soviet experts (USSR, 1986) the materials dropped on the core (sand, clay, dolomite, boron carbide, and lead) interacted with radionuclides to produce nonvolatile and more refractory chemical forms.

(b) Effects of Materials Deposited on the Core

The Soviet strategy of depositing on the core materials such as sand, clay, dolomite, boron carbide, and lead seems to have been effective in initially reducing and subsequently terminating the radionuclide release. This strategy was augmented later by introducing cold nitrogen into the reactor vault, which is believed to have assisted in the sudden drop of the release.

It would be possible to study filtering effects, chemical reactions, and temperature effects of materials such as those used at Chernobyl to determine their effectiveness in mitigating fission-product release, and such studies will at least be considered as part of the NRC's developing severe accident management program. However, priority in this program will be given to measures to prevent or halt the progression of a severe accident.

(c) Mechanisms for Release of Single Elements

There are reports that aerosol particles containing pure cerium, cesium, or ruthenium were found in the Chernobyl release. This is a surprising result that is not explained by the present technology. It is not clear whether this observation has any significance, but there is some interest in investigating this matter to determine whether it would shed further light on U.S. source term technology, particularly on understanding the influence of chemistry on the release.

(d) Hydrogen Generation From Dispersal of Fragmented Debris

During the reactivity-initiated accident, the oxidation of fragmented and dispersed core materials led to the production of hydrogen. There is some speculation that this hydrogen may have been involved in a second explosion. It is not clear that the hydrogen could have become mixed with oxygen via air ingress rapidly enough to be involved in that second explosion. Nevertheless, the generation of hydrogen from the dispersed fragmented debris is probably an important process. This process has already been identified in the NRC's research program on source term release, and is being studied currently.

(e) Physical and Chemical Forms of Iodine

Physical and chemical forms of the fission-product iodine were a subject of debate before the Chernobyl accident. Some believe that cesium iodide, an aerosol particulate form, will dominate; others believe that molecular iodine, hydrogen iodide, organic iodide, or some other vapor-phase form may be prevalent.

There is no information about the initial chemical form of iodine in the Chernobyl release. There were reports from Sweden that gaseous forms of iodine reached that country. Cesium iodide, which may have been the initial form of iodine in the release, was, however, exposed to the atmospheric conditions for extended periods of time en route to Sweden; during this time iodine would be expected to become converted to gaseous forms. The Swedish observation is probably inconclusive. The chemical forms of fission-product iodine are currently being investigated in NRC research programs, and no new insights have been gained from the Chernobyl accident that would influence this investigation.

5.1.4 Conclusions and Recommendations

Many differences exist between the RBMK design and the design of U.S. light-water reactors (LWRs) and between the Chernobyl accident and those hypothesized for U.S. LWRs. There are, however, similarities in physical processes that may occur in both reactor types. The magnitude of the Chernobyl source term is comparable to the worst-case releases studied for U.S. LWRs. Many severe-accident scenarios in U.S. reactors would be expected to result in considerably less amounts of released radionuclides. However, the lack of any warning before the impending initial release and the composition of the radionuclide release appear to be unique to the reactivity-initiated accident of the Chernobyl type. After the initial release, the subsequent course of the radionuclide release appears to have been strongly influenced by the accident-management strategies followed to control the release and cool the reactor.

Little is seen in the Chernobyl event that would provide new insights on or suggest inadequacies in current U.S. source term technology. The major areas affected that have been identified to date and that are not currently modeled in U.S. source term analytical methods involve two mechanisms of fission-product release from fuel debris, namely, mechanical dispersal and chemical stripping. Although it is not clear that these mechanisms will have any effect on accident sequences relevant to U.S. reactors, it is recommended that the need for additional research be assessed. This research would be conducted to understand these mechanisms better and to incorporate such phenomena into the NRC's analytical models of source term evaluation, as appropriate.

5.2 Steam Explosions

The term "steam explosion" refers to a phenomenon in which molten fuel rapidly fragments into very fine particles and is dispersed within the coolant, to which it transfers much of its energy, resulting in steam generation, shock waves, and possible mechanical damage. If such events were to take place on a large enough scale within the reactor pressure vessel, missiles could be generated which might penetrate the containment and allow early release of fission products. In the Reactor Safety Study (WASH-1400), this mode of containment failure was denoted by the symbol alpha (α) and is often referred to as α -mode containment failure or simply α -mode failure. Another potential loading from an in-vessel steam explosion is direct shock of the reactor vessel. Both explosive and nonexplosive interactions between molten fuel and coolant significantly affect in-vessel core-melt progression, including hydrogen generation, in a core uncover accident.

According to current information offered at the Vienna meeting of August 1986 (USSR, 1986), the Soviets have attributed the mechanical destruction observed during the Chernobyl accident to a steam explosion. In this regard, the basic observations about this event are:

- (1) A reactivity-initiated accident occurred because of boiling with a positive coolant void reactivity. According to U.S. initial approximate estimates, the void effect could yield a strong overpower condition. According to Soviet results, the overpower was strong enough to produce more than 300 cal/g in the fuel within a few seconds (USSR, 1986; INSAG, 1986).
- (2) The Soviets assumed that at this energy level fuel rod destruction occurred on a large scale yielding rapid vapor generation, augmented core voiding, and a very strong power pulse. They presented the results of a calculation for this runaway condition, but they indicated the uncertainties in their analysis. Still, the Soviets currently believe that the reactor was shut down by a mechanical disassembly involving homogenization of fuel with the moderator and relocation of graphite. Within this general context of a very strong power pulse, the Soviets could visualize an intense fuel/coolant interaction, i.e., a steam explosion.
- (3) A good portion of the roof of the reactor building appears to have been blown off, and many graphite blocks are seen to lie outside in the immediate vicinity of the plant. It is not clear whether this mechanical damage was a direct consequence of the power pulse or of a subsequent explosion that was heard 2 to 3 seconds later.

Whether the mechanical damage was the result of a classical steam explosion from a molten fuel-coolant interaction (FCI) or the result of a non-FCI steam blowdown may never be known for sure. To determine this would require very precise analyses utilizing three-dimensional space-dependent neutronic-thermal-hydraulic and hydrodynamic (molten fuel dynamics) computer codes to determine the relative locations of molten fuel and water within the core channels at the time of the "explosion." Molten fuel and water must co-exist for a classical steam explosion to occur. Such analyses would also require detailed information on the prior operating history and configuration of the reactor before the accident. It is not likely that a dependable evaluation could be produced (NUREG-1250, page 4-10).

The SL-1 accident (which involved a low-power reactor with plate-type metallic fuel being tested at the Atomic Energy Commission's National Reactor Testing Station in Idaho) and the BORAX-1 intentional destructive test (Boiling Reactor Experiment, with an open-tank reactor using plate-type metallic fuel, also conducted at the Idaho testing station) also represent destructive events. In these cases, large rapid energy insertions produced partial fuel vaporization, driving solid and liquid fuel debris into the water resulting in conditions conducive to steam explosions.

In the Chernobyl accident, core-average energies of about 300 cal/g are calculated for the power burst and it is inferred that local energies were considerably higher. For reference, melting of UO_2 begins at about 270 cal/g and vaporization begins at about 400 cal/g (see Figure 5.2). Given local energies in

the range of 400 to 600 cal/g as are believed to have occurred, a destructive steam explosion would be possible.

On the basis of a large number of power burst tests in the power burst facility and SPERT-CDC (capsule driven core) reactors (United States) and in the NSRR reactor (Japan), it is known that steam explosions (or even significant pressure pulses) do not occur when peak radially averaged energies are less than 300 cal/g. For example, a summary of 27 typical SPERT-CDC test results shows only two with substantial pressure generation (MacDonald, 1980). The maximum events recorded were a 26-bar pressure pulse for test CDC-474 at 275 cal/g UO_2 and a 162-bar pressure pulse for test CDC-569 at 282 cal/g UO_2 . Some tests in the SPERT-CDC test series were run with higher energy depositions. Energy depositions greater than 300 cal/g UO_2 have produced significant pressure pulses (about 100 to 150 bar) in some but not all tests. The more recent power burst facility tests show that idealized conditions are difficult to achieve. Here no major energetic events were seen for power pulses producing up to 285 cal/g.

Only test RIA-ST-4 run at 350 cal/g yielded a 350-bar pressure pulse (MacDonald, 1980). Licensing requirements in the United States are designed to limit reactivity-initiated accidents to peak radially averaged enthalpies of less than 280 cal/g, thus precluding major fuel disruption and related steam explosions. The major reactivity-initiated accidents analyzed for reactor licensing considerations are the control rod drop accident (boiling-water reactor) and the control rod ejection accident (pressurized-water reactor); realistic estimates for these accidents give maximum local energies of about only 100 cal/g or less. Because of the negative void coefficients of reactivity and other characteristics in U.S. reactors, large corewide superprompt critical reactivity-initiated accidents, like the event at Chernobyl, are thought to be highly unlikely. (See Section 2.1.)

Notwithstanding the uncertainties and qualifications regarding the likelihood for highly energetic steam explosion events as a consequence of reactivity-initiated accidents, the potential consequences of such an event are fully appreciated not only for Chernobyl but also for U.S. power reactors. The issue then is whether this potential can be physically realized, given the particular designs and operating constraints of the U.S. reactors.

5.2.1 Current Regulatory Practice

One of the early safety precautions in U.S. power reactors was the implementation of design features to limit reactivity-initiated accidents to events yielding specific energy depositions in the fuel to values less than 280 cal/g. Current U.S. regulations limit reactivity-initiated accidents to those that yield peak radial average fuel enthalpy of less than 280 cal/g. Ample experimental evidence exists that such events cannot lead to energetic interactions between fuel and coolant.

Energetic steam explosions are considered in the United States in the context of risk assessment and mitigation studies. Such studies, beginning with WASH-1400, are continually evolving both in depth and sophistication, particularly since the accident at Three Mile Island. However, the phenomenology involves an initially separated configuration of molten core material and water

which is widely perceived as having a vastly reduced potential for highly energetic behavior. The principal reasoning is related to limitations on pour rates during contacting, and on associated rates of coarse mixing (called pre-mixing). The detailed quantitative aspects of such assessments are rather involved but have been continuously improving. Currently, subjective estimates of an upper bound for the α -failure probability (conditional on core melt) are placed at a value of 0.01 (NUREG-1116, "A Review of the Current Understanding of the Potential for Containment Failure From an In-Vessel Steam Explosion"). By definition a best-estimate value would be below this number, although it is difficult to quantify with confidence.

5.2.2 Work in Progress

No research is under way or planned on the reactivity-initiated accident (RIA) prompt burst steam explosions with fuel-vapor-driven fragmentation and mixing of the molten fuel and water that are relevant to the Chernobyl accident. Further work for U.S. reactors on RIA steam explosions, if found to be needed, would be performed as part of an overall investigation of RIAs and their consequences rather than as broad-based research.

Work is under way to assess the effect of in-vessel steam explosions on severe-accident risk in U.S. light-water reactors. Much of this work is being performed under the Severe Accident Risk Reduction Program (SARRP) at Sandia National Laboratories. In the SARRP analysis, the α -mode containment failure probability from an in-vessel steam explosion as assessed in draft NUREG-1150 was found not to have a substantial effect on either the severe-accident risk or the uncertainty in risk (NUREG/CR-4551).

Current research on steam explosions consists primarily of the development and assessment of the semimechanistic integrated fuel coolant interaction (IFCI) computer model of explosive and nonexplosive fuel interactions, including hydrogen generation, for integration into the MELPROG mechanistic in-vessel melt progression code. IFCI provides a mechanistic treatment of both the preexplosion mixing phase and the explosion phase (if conditions permit), but IFCI does require a parametric input trigger for the explosion. Work is also continuing on using existing experimental data for modeling the nonexplosive mixing phase of the interaction.

Another major effort involves the development of an integrated probabilistic treatment of the potential for α -mode failure given a core-melt accident. This work has been reported in Nuclear Science and Engineering (Theofanous, 1987). It also has been published as NUREG/CR-5030, "An Assessment of Steam-Explosion-Induced Containment Failure." The work focuses on providing a well-defined methodology (including uncertainty analysis) for an integrated evaluation of the conditional likelihood for α -mode failure given an assumed core-melt condition. Results from peer reviews of this work are included in NUREG/CR-5030.

5.2.3 Assessment

The steam explosion phenomena associated with core dryout are quite different from those associated with the strong overpower conditions generated in reactivity-initiated accidents. As a result of the high power and associated

fuel melting, vaporization, and cladding failure, the fuel and coolant are forced together coherently (the prompt-critical power pulse in water reactors imposes millisecond-scale coherence) which ensures a violent thermal interaction. For these reasons, the vapor-driven fragmentation and mixing of the interspersed fuel and coolant in prompt-burst power excursions have been strongly contrasted in the past to the pouring mode of contact found in the slow meltdown situations relevant for current U.S. commercial reactors. Hence the Chernobyl accident has little relevance to the staff's current treatment of steam explosions and α -mode containment failure.

5.2.4 Conclusions and Recommendations

Considering the experience of the Chernobyl accident, it may be worthwhile to reexamine reactivity-initiated accidents in a broader context, consistent with modern probabilistic risk assessment approaches, in order to obtain a more comprehensive picture of the risk due to reactivity-initiated accidents; that is, without arbitrary limits on what is presumed as a credible event, but rather by considering the likelihood of all possible events. Within such efforts it may become necessary to quantify the severity of interactions between fuel and coolant within a phenomenological context outside the realm of present (or past) assessments. The extent of new efforts in such areas should be dictated by the likelihood of corresponding initiating events. Steam explosions of lesser direct mechanical consequences could have some effect on safety systems and/or functions that affect containment integrity. The contribution to risk from such events is believed to be small. However, it would be helpful to assess whether there is a need to expend further efforts in this area. Both explosive and nonexplosive molten-fuel-coolant interactions may significantly affect in-vessel core-melt progression and are now being included in melt-progression analysis.

5.3 Combustible Gas

The Soviet RBMK design utilizes large amounts of zirconium and graphite in the reactor core, both of which may oxidize under certain conditions resulting in the generation of large quantities of combustible gases, principally hydrogen and carbon monoxide. The generation of large quantities of combustible gases was not apparently considered as part of the Soviet containment design. The Chernobyl accident produced reactor core conditions that may have led to the generation of large quantities of combustible gases which, in turn, may have influenced the evolution and consequences of the accident.

The need to deal with the generation of combustible gas, principally hydrogen, as a consequence of reactor accidents has been recognized in the United States since the early days of light-water reactors. The burning and/or detonation of combustible gases are of concern in reactor safety for several reasons. First, a large enough energy release might threaten the integrity of the containment. Second, even if the containment survived, important safety equipment might be irreparably damaged, thus increasing the severity of the accident. Furthermore, since significant amounts of hydrogen can be generated early in the evolution of a severe reactor accident (i.e., before the reactor vessel fails), combustion can result in containment failure before expulsion of the molten core, leading to the largest radioactivity releases to the environs.

In addition to the generation of hydrogen within the reactor vessel, principally by the oxidation of hot zirconium, combustible gas is generated outside the vessel as a result of interactions between the molten core and concrete if the vessel fails. This occurs as gases from the decomposing concrete (largely steam and carbon dioxide) pass through the debris pool and react chemically with the liquid metals to form hydrogen and carbon monoxide.

5.3.1/5.3.2 Current Regulatory Practice/Work in Progress

To better understand the rationale behind the various NRC requirements dealing with hydrogen control, it is useful to consider three classes of reactor accidents. They include the design-basis accidents (DBAs), the degraded-core accidents, and the core-melt accidents.

Design-basis accidents (e.g., loss-of-coolant accidents and main steamline break accidents) are those accidents that must be thoroughly analyzed for plant design and licensing purposes. An institutional framework has been established in regard to such matters as safety margins, redundancy of equipment, seismic design capability, and quality assurance.

Requirements for combustible gas control capability for DBAs were developed in the 1960s and were codified as regulations in 10 CFR 50.44 in 1978. These requirements initially addressed the hydrogen generation associated with DBAs, including (1) limited metal and water reaction involving the fuel element cladding; (2) the radiolytic decomposition of the water in the reactor core and the containment sump; (3) the corrosion of certain metals in the containment because of the action of spray solutions; and (4) possible synergistic effects of chemical, thermal, and radiation byproducts of accident sequences on protective coatings and electric cable insulation.

Degraded-core accidents have been identified as a discrete set of accidents since the accident at Three Mile Island in March 1979. They are intended to include those accidents that are more severe than the DBAs (i.e., oxidation of more than about 5% of the fuel cladding), but which are successfully terminated short of core melt. Analyses to date of this class of accidents have postulated the oxidation of as much as 75% of the active cladding that surrounds the fuel. Requirements for safety margins in analyses of degraded-core accidents are substantially reduced relative to those for DBAs. Several licensing requirements have already been issued for dealing with hydrogen control during postulated degraded-core accidents. Moreover, the requirements for dealing with hydrogen releases during degraded-core accidents are interim requirements, pending completion of longer term efforts for dealing with core-melt accidents.

The interim requirements related to the Mark I and II boiling-water reactor (BWR) containments were issued in the form of a final rule on December 1, 1981 (46 FR 58484). The requirement to inert the smaller pressure-suppression containments was instituted because of the limited ability of these designs to tolerate the range of consequences stemming from hydrogen combustion, coupled with the knowledge that for a number of years some Mark I and II containments had successfully operated inerted.

The interim requirements for those ice condenser and Mark III BWR plants for which a construction permit was issued before March 28, 1979 were published as

a rule on January 25, 1985 (50 FR 3498). The rule requires that these plants be provided with a hydrogen control system capable of handling an amount of hydrogen equivalent to that generated from a 75% fuel cladding and water reaction without loss of containment structural integrity.

The deliberate ignition concept has been the subject of NRC review since the Tennessee Valley Authority initially proposed such a system in mid-1980. At present, after investigating alternative approaches, all ice condenser PWR and Mark III BWR utility owners have chosen deliberate ignition as the solution to the hydrogen control issue for degraded-core accidents.

In order to gain a better understanding of hydrogen generation and control in reactor accidents, both the NRC and the nuclear power industry have sponsored extensive analytical and experimental work over the last several years. This research provided (1) the technical insights to support licensing of the ice condenser and Mark III containments and (2) the technical background information to support the development of additional requirements for hydrogen control for core-melt accidents. Various research programs have investigated relevant phenomena, such as hydrogen generation, transport and spatial distribution within containment, detection, combustion modes (including deflagrations, diffusion flames, flame acceleration, and detonations). Other programs have investigated mitigation schemes, equipment survivability, and the effects of combustion on fission-product releases. Hydrogen control has also been identified as an Unresolved Safety Issue, USI A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment."

Currently there are no requirements for degraded-core-accident hydrogen control related to dry containments for operating reactors or near-term operating license applications. At the time rulemaking occurred for the ice condenser and Mark III containments, it was the staff's judgment that additional requirements were unnecessary. However, the staff committed to continue to investigate the merits of additional hydrogen control for dry containments and report the findings to the Commission. It is anticipated that the staff will report to the Commission on this matter by early 1989.

Core-melt accidents are those accidents that involve sufficient reconfiguration of the core as to make it uncoolable. They involve a failure of the reactor vessel and a relocation of the core materials to the containment floor. Substantial analyses and experiments have been in progress to develop a better understanding of the various phenomena associated with core-melt accidents. A separate set of requirements was issued as a final rule on January 15, 1982 (47 FR 2286) to address hydrogen control requirements for pending construction permit and manufacturing license applications. Some of these requirements go beyond those needed for dealing with degraded-core accidents. They were imposed with the anticipation that future efforts will require them and because the effect of their imposition on initiation of construction was minimal. In this regard, the principal requirements are:

- (1) A dedicated penetration is to be provided for possible use with a filtered venting system.
- (2) Structural integrity of the containment is ensured for internal pressures of at least 45 psig.

- (3) Alternative hydrogen control systems are to be analyzed assuming 100% reaction between fuel cladding and water, and the resultant uniform hydrogen concentration in the containment must be less than 10% by volume or the mixture must be rendered nonflammable.
- (4) Essential systems must be shown to survive the environments associated with the accidents considered.

Any additional requirements for hydrogen control during postulated core-melt accidents will be deferred, pending completion of work now under way on the general subject of severe accidents.

5.3.3 Assessment

In the early stages of the Chernobyl accident, a very high power excursion, followed by what apparently was a steam explosion, caused severe damage to the core, disrupted the cooling system, and damaged the shroud surrounding the core. The subsequent large accumulation of thermal energy eventually caused a partial meltdown of the core. Zirconium-niobium alloy from the pressure tubes and from the fuel cladding reacted with steam, causing the generation of hydrogen. Another important source of combustible gases was the graphite used as a neutron moderator in the core, which, when reacting with water and/or steam from the disrupted cooling system, produced hydrogen and carbon monoxide. It is also possible that some methane gas was produced. Later in the accident, air was admitted through the damaged core shroud and the graphite burned, emitting more carbon monoxide. It was estimated that about 10% of the graphite burned [approximately 120 metric tons (tonnes)]. In the analyses of degraded-core accidents in U.S. reactors, the zirconium and water reaction is considered to be the principal mechanism by which hydrogen is generated. Thus, this source of hydrogen and its consequences have been studied extensively. In the Chernobyl plant, larger amounts of metal and higher temperatures were responsible for generating more hydrogen and for increased generation rates, but basic mechanisms were probably similar to those postulated in the degraded-core analyses. The real difference was the presence of graphite. In the United States, only one commercial reactor (Fort St. Vrain Nuclear Generating Station) uses a graphite moderator. This reactor, however, of a completely different design than the Chernobyl plant, uses helium as a primary coolant. There is no analogy, therefore, between what happened at Chernobyl and the consequences of postulated accidents in this reactor. Concomitant generation of hydrogen and carbon monoxide has been considered in severe-accident analyses of U.S. reactors when molten core material reacts with concrete.

The possibility that the hydrogen generated in the Chernobyl core caused a detonation or highly accelerated flame (explosion) cannot be ruled out. A detonation could be possible under the Chernobyl accident conditions for the following reasons:

- (1) The hydrogen-air mixture could have been locally very sensitive.
- (2) The very hot core materials could provide a strong ignition source for direct initiation of a detonation.

- (3) The turbulence generated in the vicinity of the fuel-loading machine and the adjacent walls could lead to deflagration-to-detonation transition.

Very significant differences in plant characteristics and in the types of containment buildings, along with the dearth of data on the Chernobyl accident, make it practically impossible to draw any analogies between hydrogen transport and combustion in the Chernobyl plant and U.S. reactors.

5.3.4 Conclusions and Recommendations

In summary, although the conditions that existed during the Chernobyl accident may have caused large amounts of combustible gases to generate, it cannot be concluded from the available data that these gases were generated by some new or different mechanisms or produced consequences not previously investigated as part of severe-accident analyses for U.S. reactors. It is difficult to apply observations from the Chernobyl accident to U.S. plants because of significant design differences between the RBMK and nuclear power reactors in the United States; furthermore, the NRC staff still lacks detailed accident data. Considering the preliminary evaluation, it does not appear that any additional work is warranted solely on the basis of the Chernobyl event. The staff concludes that its current and proposed research program on combustible gas phenomena in conjunction with the study of severe accidents would be adequate for addressing this issue in U.S. reactors.

CHAPTER 6

GRAPHITE-MODERATED REACTORS

The Fort St. Vrain high-temperature gas-cooled reactor (HTGR) in Weld County, Colorado, and the Department of Energy's (DOE's) N-reactor at the Hanford Reservation in Washington State are the only graphite-moderated power reactors operating in the United States. Because the N-reactor is not licensed by the NRC and is under the authority of DOE, the implications of the Chernobyl accident for the N-reactor were assessed separately by DOE and others.

Since the original preparation of this report, the licensee for Fort St. Vrain, the Public Service Company of Colorado, has notified the NRC that it will discontinue operation on or before June 30, 1990 (Williams, 1988).

In addition to licensing reactors that generate electric power, the NRC also licenses non-power reactors (those used for testing, research, and the production of radioactive isotopes); some of these are moderated by graphite or use graphite for neutron reflectors and for other purposes. These reactors have comparatively low fission-product inventories, and the risk to the public is not comparable to the risk from power reactors. The NRC staff has reviewed a petition for rulemaking from The Committee To Bridge the Gap with respect to probabilities and consequences posed by graphite fires that might be caused by stored energy release in non-power reactors and in the Fort St. Vrain reactor. The NRC staff denied the petition (52 FR 37321) on the basis that existing emergency plans for non-power reactors are adequate and that existing information is adequate for a safety evaluation of the effect of stored energy on the potential for graphite burning and the associated danger to the health and safety of the public, and for the reasons discussed herein for the Fort St. Vrain reactor. A review of the literature and existing data by Brookhaven National Laboratory (BNL) has been published separately (NUREG/CR-4981, "A Safety Assessment of the Use of Graphite in Nuclear Reactors Licensed by the U.S. NRC"). The BNL report concluded: "There is no new evidence associated with either the Windscale accident or the Chernobyl accident that indicates a credible potential for a graphite burning accident in any of the reactors considered in this review." Therefore, the NRC will not discuss further in this document the role of graphite in the consideration of accidents at non-power reactors.

The HTGR type of reactor has been under development in the United States and West Germany since the late 1950s. Fort St. Vrain is the only operating HTGR in the United States, although additional HTGR experience and technology have been gained through operation of the Peach Bottom Unit 1 HTGR from 1967 through 1974 and through development and licensing programs for advanced HTGRs. Two HTGRs are operating in West Germany and have contributed to the HTGR technology base in the United States. Currently, HTGR development efforts in the United States are being concentrated on the modular HTGR (MHTGR) concept that uses available HTGR technology in combination with inherent and passive safety features. The MHTGR concept is being proposed by DOE in conformance with the Commission's recently published "Statement of Policy for the Regulation of

Advanced Nuclear Power Plants" (51 FR 24643). Thus, assessment of the Chernobyl implications and candidate issues has value both for Fort St. Vrain and the MHTGR. Although the discussion that follows largely centers on Fort St. Vrain, because of its operating status, it also addresses the MHTGR to the extent appropriate in supporting the staff's current review of the MHTGR concept.

The HTGR concept, with emphasis on Fort St. Vrain, is assessed here against the issues raised by the Chernobyl accident: issues of operations, design, containment, emergency planning, and severe-accident phenomena. The general conclusions and those pertaining to the principal specific areas for light-water reactors presented in this document are also assessed from the HTGR perspective. The discussions that follow illustrate how the unique features of the HTGR concept were considered in forming these assessments and also how certain specific assessments for HTGRs were derived.

6.1 The Fort St. Vrain Reactor and the Modular High-Temperature Gas-Cooled Reactor

The only features that the 330-MWe Fort St. Vrain reactor, the MHTGR, and the Chernobyl design have in common are the use of a graphite moderator and the use of gravity-driven control rods. At Fort St. Vrain a helium coolant is used which is pressurized to 700 psi and which flows downward through 1/2-inch-diameter holes in a fully ceramic (graphite) core. Thorium and uranium dicarbide-coated fuel particles are dispersed in hexagonal graphite blocks 31.2 inches long and 14.2 inches measured across the flat sides of the hexagon. The coating of each fuel particle (from inside to outside) consists of a porous carbon buffer layer, a layer of dense isotropic carbon, a silicon carbide layer, and an outer coating of dense isotropic carbon. The reactor core and the entire primary coolant system, including steam generators and helium circulators, are enclosed in a prestressed concrete reactor vessel which, through use of inner and outer penetration seals and in conjunction with a filtered and vented confinement building, satisfies the NRC's general design criteria for reactor containment.

The MHTGR concept uses a fuel and reactor design derived from the Fort St. Vrain reactor. However, the reactor will be contained in a steel pressure vessel and the helium circulator and steam generator in a connected second steel vessel rather than full enclosure of the primary system in a single prestressed concrete reactor vessel. Its safety approach is based on an inherent negative power coefficient and selection of the reactor power density and vessel size so that decay heat can be removed passively from the exterior wall of the vessel for postulated accidents. Decay heat would be removed by natural convection airflows that are adequate to preclude fission-product release from the fuel, or unacceptable damage to the reactor vessel or to other vital reactor systems. The reference plant would consist of four such modules and would produce total electric power of 550 MWe.

6.2 Assessment

6.2.1 Operations

Administrative control and operational practices at Fort St. Vrain, although generally similar to those of light-water reactors, originally contained some

differences believed to reflect the unique features of the HTGR concept. In recent years, however, changes have been made to bring plant operations much closer to those of light-water reactors. A program to upgrade the Technical Specifications is currently under way that will result in administrative controls that are comparable to those of light-water reactors. Furthermore, although regulations do not require that American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code procedures be followed, inservice inspection and testing requirements are nevertheless being restructured into a format that utilizes ASME Code requirements "as applicable." The Fort St. Vrain reactor also must meet the same or equivalent requirements as those for light-water reactors with respect to quality assurance, equipment qualification, external events, physical security, fire protection, radiation protection, and operator training and qualification.

Two important differences between HTGRs and light-water reactors with respect to operational safety are the slower response of HTGRs to plant transients, because of low-power density, and their increased margin to fuel failure, because of the fully ceramic core. These differences formed the basis for permitting less prescription in some administrative procedures and are considered to enhance overall safety.

The designers of the MHTGR are proposing a design that utilizes inherent and passive safety features and fully automated plant control systems that tend to minimize the need for operator action to ensure safety, thus reducing the importance of the man-machine interface to reactor safety. The staff is reviewing this approach and will include its findings in a safety evaluation report on the modular high-temperature gas-cooled reactor; the report is scheduled to be issued in 1989.

6.2.2 Design

(1) Reactivity Accidents

Unlike the Chernobyl reactor, the Fort St. Vrain reactor (HTGR) and the MHTGR have overall negative power and temperature reactivity coefficients, and reactivity additions can be terminated by diverse, redundant shutdown devices--gravity-operated control rods and boron carbide pellets in hoppers above the core. The helium coolant has essentially no reactivity effect.

Like light-water reactors, however, very large reactivity insertion accidents must be precluded by both the reactor protection system and by structural designs. At Fort St. Vrain, control rod ejection is precluded by two separate and diverse structural systems. Other potential mechanisms for reactivity insertions of a more extensive but less probable nature are water ingress, loss of control rod integrity by overheating, and the downward displacement of the core from the top suspended control rods caused by failure of the core support structure. These mechanisms are highly unlikely for both Fort St. Vrain and advanced HTGRs, but probabilistic risk assessment studies to further ensure their low probabilities are being considered for Fort St. Vrain and will be reviewed for the MHTGR.

(2) Accidents at Low Power and at Zero Power

Except for a prompt critical event, the HTGR's characteristically slow thermal response to transients and large margin to fuel failure make accidents at low power or at zero power of lesser concern than for light-water reactors. Use of probabilistic risk assessment is being considered to explore this operating regime further both for Fort St. Vrain and for advanced HTGRs.

(3) Multiple-Unit Protection

This issue is not of direct concern for Fort St. Vrain because it is a single-unit plant and there are no plans to construct an additional nuclear unit. However, certain low-probability accidents might result in habitability concerns for manual actions that might have to be performed in the reactor building outside the control room. Probabilistic risk assessment is being considered as the appropriate means for evaluating this concern.

The issue is different for the MHTGR because operators for four reactors would be stationed in a single control room. The designers are proposing that the operators serve primarily to monitor reactor operations and that individual reactor safety functions be automatically and locally controlled. The staff is reviewing this proposed approach and will report its findings as part of the MHTGR review.

(4) Fires and Explosions

A study of the potential for a Chernobyl-type fire and for explosions derived from hydrogen and carbon monoxide "water gas" at Fort St. Vrain was initiated immediately after the Chernobyl event (Brey, 1986). The staff has reached the conclusion that the use of a helium coolant, the overall negative reactivity coefficient, completely diverse alternative shutdown and cooling systems, and the protection offered by the prestressed concrete reactor vessel against reactor fires, internal postulated explosions, and fission-product release to the environs remove Fort St. Vrain from any vulnerability characteristic of the Chernobyl design. In assessing the potential for a graphite fire, the licensee was asked to consider the highly improbable simultaneous failures of penetrations both at the top and bottom of the prestressed concrete reactor vessel, which would cause a chimney effect for sustained air ingress. Although the staff believes that the occurrence of such an event is extremely improbable, it agrees with the licensee that if the need arose, the reactor building could be flooded with water to a level sufficient to defeat the chimney effect and subsequently terminate the fire.

The potential for and consequences of fires in the MHTGR are being considered in the staff's review of that reactor and will be assessed in a safety evaluation report on the MHTGR scheduled to be issued in 1989. Work is under way to address the subject of chimney fires and may show that, even with chimney-type geometry, graphite fires could not occur in either the MHTGR or Fort St. Vrain.

(5) Containment

As indicated in Section 6.1, the Fort St. Vrain reactor and primary coolant system are both enclosed in a prestressed concrete reactor vessel. This technology, originally developed for European gas-cooled reactors, is considered in England and Scotland as providing appropriate containment for the many carbon-dioxide-cooled, graphite-moderated power reactors in their utility systems. At Fort St. Vrain, additional containment protection was gained by using double penetration closures, a prestressed concrete reactor vessel liner cooling system for diverse emergency decay heat removal, and a building enclosure that provides for immediate pressure relief by venting, followed by a controlled filtered release of the building's atmosphere. This design was found to meet the general design criteria in effect at the time the construction permit was issued, and it was again found acceptable following the Chernobyl event. No further consideration of the Fort St. Vrain containment system as a Chernobyl candidate safety issue is considered necessary.

In the MHTGR, steel vessels rather than a prestressed concrete reactor vessel will be used, and containment credit will not be taken for its surrounding structures because of the inherent and passive safety characteristics of the fuel design and decay heat removal system. The staff's review of the MHTGR will address the adequacy of this approach.

(6) Emergency Planning

Following the accident at Three Mile Island Unit 2, emergency planning needs for Fort St. Vrain were reviewed and it was concluded that an emergency planning zone with a radius of 5 miles would be sufficient, rather than the 10-mile radius required for light-water reactors. This decision acknowledged the longer time needed for an accident to progress to the fission-product release stage and the lower fission-product inventory associated with a reactor of Fort St. Vrain's power level.

The MHTGR designers claim that the design's inherent safety characteristics will simplify offsite emergency planning and will permit reduction of the emergency planning zone to the site boundary. The staff is considering this claim in its review of the MHTGR concept.

(7) Severe-Accident Phenomena

Severe-accident phenomena for Fort St. Vrain in terms of graphite fires and combustible gas explosions have been studied as described above in "Fires and Explosions." Loss of forced convective cooling and helium depressurization accidents are considered as design-basis accidents, and although core temperatures become elevated, fission-product release to the environs meets the guidelines of 10 CFR 100. Severe accidents beyond the design-basis accidents, other than those discussed in "Fires and Explosions," which could include combined loss of forced cooling and helium depressurization or large reactivity insertions, are now being studied in accordance with the NRC's Severe Accident Policy. A limited probabilistic risk assessment (PRA) study is in progress, utilizing past severe-accident and risk studies pertaining to a large and advanced HTGR concept (Reilly, 1984), PRA studies already performed for Fort St. Vrain's design-basis accidents, and component experience from other operating gas-cooled

reactors as appropriate (e.g., prestressed concrete reactor vessel performance, gravity control rod performance, and primary system components). The staff will consider uncertainties in this analysis that result from such factors as a limited technology base in assessing the needs and benefits derived from such a PRA.

It is not expected that a PRA study for Fort St. Vrain would result in any modifications or additions to plant equipment. Rather, the study is expected to identify or confirm areas of high risk and to provide information useful for plant operations. In particular, the PRA results would support the program now in progress to upgrade the Technical Specifications and would address the subject of component reliability needs. It would be useful to know more about component reliability in assessing current programs and programs being developed in inservice inspection and testing and in maintenance.

Although this PRA study would not be directly applicable to the MHTGR concept, indirect benefits would include improved bases for selecting MHTGR components, systems, and structures and helping develop the capabilities for the study of severe accidents at MHTGRs.

It should be noted that completion of this Fort St. Vrain PRA as considered at the time of the original writing of this report is no longer warranted in view of the imminent termination of the operation of Fort St. Vrain (Williams, 1988).

One of the major objectives of the MHTGR review is the identification of those accidents (including severe accidents) that must be considered in the design as well as for emergency planning purposes. In regard to these accidents, the reactor will be analyzed for such events as (1) total withdrawal of all control rods, (2) 36-hour station blackout, (3) failure of the principal safety-grade heat removal system, (4) rapid depressurization as caused by vessel failure, and (5) severe seismic events consistent with those analyzed for light-water reactors. Furthermore, the residual hazard of a graphite fire will be fully explored. Because of the MHTGR's passive safety features, the expectation is that should these events occur, large consequences would not result.

The staff has considered reinitiating an experimental program to investigate how graphite performs under thermal stress, which would relate to the integrity of the graphite support structure for the Fort St. Vrain core. This included consideration of a potential mechanism of reactivity insertion by downward displacement of the core from top suspended control rods as a result of the failure of the graphite core support structure. A determination was made that the issue raised no immediate concerns, that no immediate action at Fort St. Vrain is warranted on account of this issue, that the experiments need not be reinitiated solely on the basis of the Chernobyl accident, and that funds were not available to support this effort. At the time of the original writing of this report, the NRC staff was considering further the question of whether this work should be undertaken to enhance confidence in the long-term integrity of the Fort St. Vrain structural graphite, an issue mooted by current plans to terminate Fort St. Vrain operation by June 30, 1990.

6.3 Conclusions and Recommendations

The HTGR at Fort St. Vrain and the MHTGR now being developed and reviewed have been assessed against issues raised by the Chernobyl accident in a manner similar to the assessment performed for light-water reactors. Except for the use of a graphite moderator and gravity-driven control rods, these HTGRs and the Chernobyl reactor have no other features in common. The staff assessed the other areas at issue--operations, design, containment, emergency planning, and severe-accident phenomena--and found that the implications of the Chernobyl accident have generated no new licensing concerns for HTGRs and that general conclusions and those pertaining to specific areas are the same as those for light-water reactors. In performing these assessments, the staff reviewed the existing information related to these areas and concludes that programs under way or being considered adequately satisfy any concerns that could be generated because of the Chernobyl accident. A limited Fort St. Vrain probabilistic risk assessment and further experiments with structural graphite were being considered before the Chernobyl accident (Denton, April 2, 1982). While the Chernobyl events supported the need for such work, the imminent termination of the operation of Fort St. Vrain has removed that need. The issues raised by the Chernobyl accident have not caused any new concerns about HTGR severe-accident phenomena but rather enter into the NRC plans described.

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This report was prepared by the Nuclear Regulatory Commission (NRC) staff to assess the implications of the accident at the Chernobyl nuclear power plant as they relate to reactor safety regulation for commercial nuclear power plants in the United States. The facts used in this assessment have been drawn from the U.S. fact-finding report (NUREG-1250) and its sources. The report consists of two volumes: Volume I, Main Report and Volume II, Appendix -- Public Comments and Their Disposition.

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