

# **WATER REACTOR SAFETY RESEARCH PROGRAM**

**A Description of Current and Planned Research**



**Office of Nuclear Regulatory Research  
U. S. Nuclear Regulatory Commission**

Available from  
National Technical Information Service  
Springfield, Virginia 22161  
Price: Printed Copy \$8.00; Microfiche \$3.00

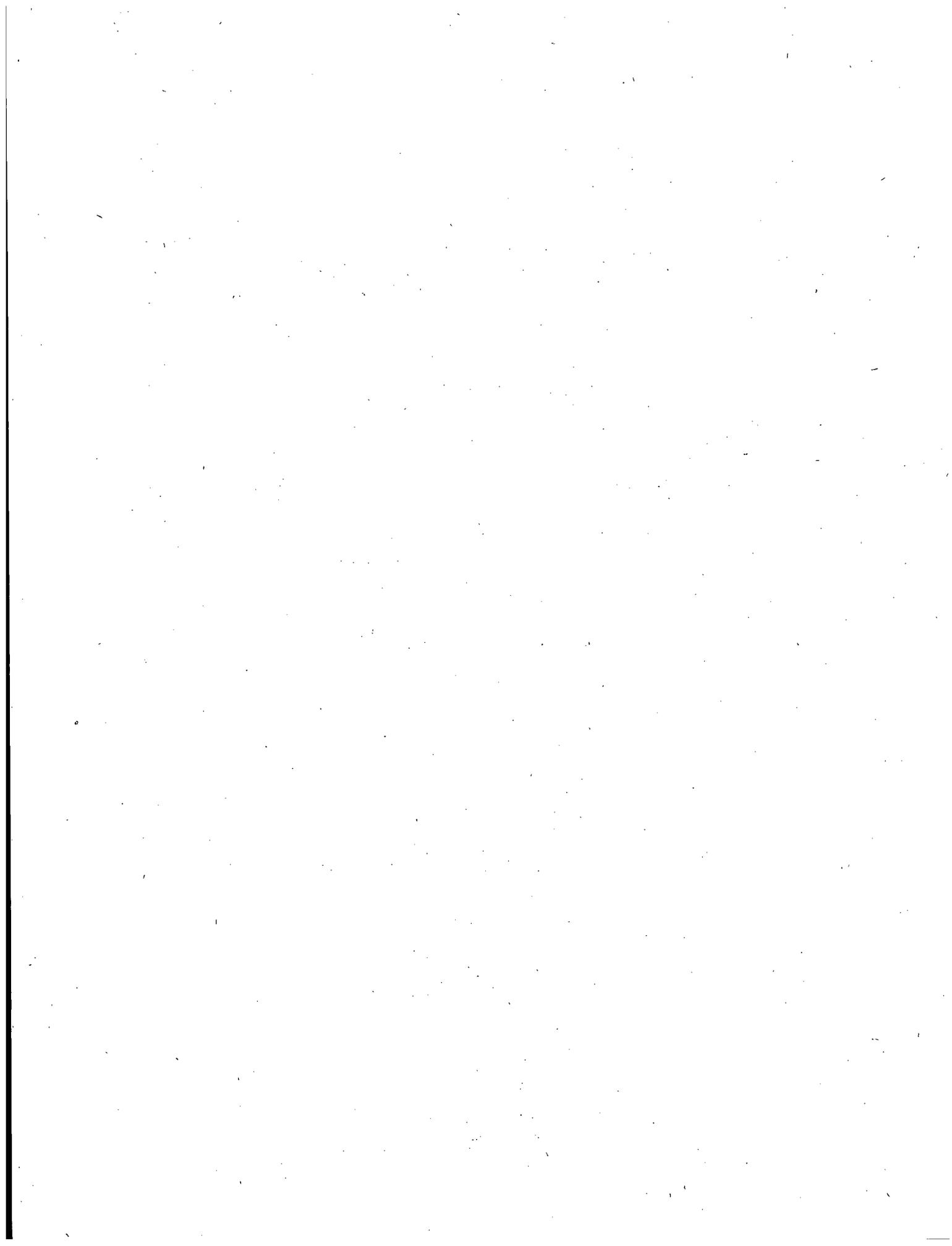
The price of this document for requesters outside  
of the North American Continent can be obtained  
from the National Technical Information Service.

# **WATER REACTOR SAFETY RESEARCH PROGRAM**

## **A Description of Current and Planned Research**

**Manuscript Completed: July 1978**  
**Date Published: February 1979**

**Division of Reactor Safety Research  
Office of Nuclear Regulatory Research  
U. S. Nuclear Regulatory Commission  
Washington, D. C. 20555**



## CONTENTS

	<u>Page</u>
1. INTRODUCTION.....	1
1.2 RESEARCH DEFINITIONS.....	1
1.2.1 Developmental Safety Research.....	1
1.2.2 Confirmatory Safety Research.....	1
1.2.3 Technical Assistance.....	2
1.2.4 Research for Improved Safety.....	2
1.3 REACTOR SAFETY PHILOSOPHY.....	2
1.4 REACTOR SAFETY REGULATORY PROCESS.....	3
1.5 WATER-REACTOR SAFETY-RESEARCH AREAS.....	3
1.5.1 Systems Engineering Branch.....	6
1.5.1.1 Objectives.....	6
1.5.1.2 Background.....	6
1.5.2 Fuel Behavior Research Branch.....	8
1.5.2.1 Objectives.....	8
1.5.2.2 Background.....	8
1.5.3 Analysis Development Branch.....	8
1.5.3.1 Objectives.....	8
1.5.3.2 Background.....	9
1.5.4 Metallurgy and Materials Research Branch.....	9
1.5.4.1 Objectives.....	9
1.5.4.2 Background.....	9
1.5.5 Research Support Branch.....	10
1.5.5.1 Objectives.....	10
1.5.5.2 Background.....	10
1.6 AVAILABILITY OF RESEARCH INFORMATION.....	10
REFERENCES.....	12
2. PROGRAM PLAN FOR THE SYSTEMS ENGINEERING BRANCH.....	13
2.1 INTRODUCTION.....	13
2.2 INTEGRAL SYSTEM STUDIES AND TESTS.....	13
2.2.1 LOFT.....	13
2.2.1.1 Objectives.....	13
2.2.1.2 Present Status.....	14
2.2.1.2.1 The Nonnuclear Experiment Series.....	14
2.2.1.2.2 Results of the First Five Experiments.....	15
2.2.1.2.3 Work Remaining Before Nuclear Experiments Begin...	17

CONTENTS (Cont'd)

	<u>Page</u>
2.2.1.3 Research Program.....	18
2.2.1.3.1 The First Nuclear Series.....	18
2.2.1.3.2 Subsequent Nuclear Experiments.....	18
2.2.1.3.3 LOFT Instrumentation Development.....	18
2.2.2 Semiscale.....	18
2.2.2.1 Objectives.....	19
2.2.2.2 Present Status.....	19
2.2.2.2.1 Blowdown Heat Transfer.....	20
2.2.2.2.2 Reflood Heat Transfer.....	20
2.2.2.2.3 ECC Bypass.....	20
2.2.2.2.4 Alternative Locations for ECC Injection.....	20
2.2.2.2.5 Steam-Generator-Tube Ruptures.....	21
2.2.2.2.6 Foreign Facilities.....	21
2.2.2.3 Research Program.....	22
2.2.2.4 Milestones.....	23
2.2.3 Advanced Instrumentation Development for Two- and Three-Dimensional Tests.....	23
2.2.3.1 Objectives.....	23
2.2.3.2 Present Status.....	23
2.2.3.3 Research Program.....	24
2.2.3.4 Milestones.....	25
2.3 SEPARATE-EFFECTS STUDIES AND TESTS.....	25
2.3.1 Heat Transfer During Blowdown and Reflood.....	25
2.3.1.1 Objectives.....	25
2.3.1.2 Present Status.....	26
2.3.1.2.1 PWR Blowdown Heat Transfer.....	26
2.3.1.2.2 BWR Blowdown and Emergency Cooling Studies.....	26
2.3.1.2.3 Full-Length Emergency Cooling Heat-Transfer Tests (FLECHT) and System Effects and Separate-Effects Test (SEASET).....	27
2.3.1.2.4 Heat-Transfer Correlations and Improvement.....	28
2.3.1.3 Research Program.....	29
2.3.1.3.1 PWR Blowdown Heat-Transfer Program.....	30
2.3.1.3.2 BWR Emergency Cooling Studies.....	31
2.3.1.3.3 FLECHT AND SEASET.....	31
2.3.1.3.4 Heat-Transfer Correlations and Improvements.....	35
2.3.1.4 Milestones.....	36
2.3.2 ECC Bypass and Steam-Water Mixing Research.....	36
2.3.2.1 Objectives.....	36
2.3.2.2 Present Status.....	36
2.3.2.3 Research Program.....	37
2.3.3 Other Programs.....	37
2.3.3.1 Objectives.....	37
2.3.3.2 Present Status.....	38



CONTENTS (Cont'd)

	<u>Page</u>
3.2.3 Fuel-Rod Properties.....	60
3.2.3.1 Present Status.....	60
3.2.3.1.1 Gap Conductance.....	60
3.2.3.1.2 Gas Pressure.....	60
3.2.3.1.3 Fuel-Cladding Mechanical Interactions.....	61
3.2.3.2 Research Program.....	62
3.2.3.2.1 Gap Conductance.....	62
3.2.3.2.2 Commercial Fuel Irradiation Performance.....	62
3.2.3.3 Milestones.....	62
3.3 IN-REACTOR TESTS.....	63
3.3.1 Present Status.....	63
3.3.1.1 Fuel-Rod Performance During Abnormal Operation.....	63
3.3.1.1.1 Power-Cooling Mismatch (PCM).....	63
3.3.1.1.2 Overpower Transients Without Departure from Nucleate Boiling.....	63
3.3.1.1.3 PCM Transients in Pressurized Water Reactors.....	64
3.3.1.1.4 PCM Transients in Boiling Water Reactors.....	64
3.3.1.2 Reactivity-Initiated Accident Tests.....	64
3.3.1.3 LOCA Tests.....	65
3.3.2 Research Program.....	65
3.3.2.1 Integrated Tests of Fuel Models Under Accident Conditions.....	65
3.3.2.1.1 Power Burst Facility.....	66
3.3.2.1.2 In-Reactor Tests at Other Facilities.....	67
3.3.3 Milestones.....	68
3.4 FUEL BEHAVIOR COMPUTER CODES.....	69
3.4.1 Present Status.....	69
3.4.1.1 Steady-State Codes.....	70
3.4.1.2 Codes for the Analysis of Transients.....	71
3.4.2 Research Program.....	72
3.4.2.1 Fuel Behavior Code Development.....	72
3.4.2.2 Material Property Correlations.....	73
3.4.2.3 Steady-State Code Development.....	73
3.4.2.4 Transient Code Development.....	73
3.4.2.5 Fuel Code Assessment.....	74
3.4.3 Milestones.....	75
3.5 FUEL MELTDOWN AND FISSION PRODUCT BEHAVIOR.....	76
3.5.1 Present Status.....	76
3.5.1.1 Fuel Meltdown.....	76
3.5.1.2 Fission-Product Release and Transport.....	77

CONTENTS (Cont'd)

	<u>Page</u>
3.5.2 Research Program.....	78
3.5.2.1 Fuel Meltdown.....	78
3.5.2.1.1 Steam Explosion.....	78
3.5.2.1.2 Molten-Fuel Interactions.....	78
3.5.2.1.3 Heat Transfer in Molten Pools.....	78
3.5.2.1.4 Meltdown Accident Analysis.....	78
3.5.2.2 Fission-Product Release and Transport.....	78
3.5.3 Milestones.....	79
REFERENCES.....	80
4. PROGRAM PLAN FOR THE ANALYSIS DEVELOPMENT BRANCH.....	87
4.1 INTRODUCTION.....	87
4.1.1 Purpose of Analysis Development.....	87
4.1.2 Classification of Computer Codes.....	87
4.1.3 Existing Reactor Coolant System Codes.....	88
4.1.4 Advanced Systems Codes for the Analysis of Reactor Coolant Systems.....	88
4.1.5 Containment Codes.....	90
4.1.6 Component Codes.....	91
4.1.7 Code Assessment.....	92
4.1.8 Sensitivity Studies.....	93
4.1.9 Code Uncertainty Studies.....	93
4.1.10 Supporting Experiments for the Development of Basic Analytical Models.....	93
4.2 SYSTEMS CODES.....	96
4.2.1 Existing Reactor Coolant System Codes.....	96
4.2.1.1 Present Status.....	96
4.2.1.1.1 RELAP-4/MOD 5.....	96
4.2.1.1.2 RELAP-4/MOD 6.....	97
4.2.1.1.3 RELAP-4/MOD 7.....	97
4.2.1.1.4 Scoping Studies with Existing Codes.....	97
4.2.1.1.5 User-Convenient Version of RELAP-4.....	98
4.2.1.2 Research Program.....	98
4.2.1.2.1 RELAP-4/MOD 6.....	98
4.2.1.2.2 RELAP-4/MOD 7.....	98
4.2.1.2.3 WRAP.....	99
4.2.1.3 Milestones.....	99
4.2.2 Advanced Reactor Coolant System Codes.....	99
4.2.2.1 Present Status.....	99
4.2.2.1.1 TRAC.....	99
4.2.2.1.2 THOR.....	100
4.2.2.1.3 RELAP-5.....	101
4.2.2.2 Research Program.....	101
4.2.2.2.1 TRAC.....	101
4.2.2.2.2 THOR.....	102
4.2.2.2.3 RELAP-5.....	102

CONTENTS (Cont'd)

	<u>Page</u>
4.2.2.3 Milestones.....	102
4.2.3 Containment System Codes.....	102
4.2.3.1 Improvements in Existing Codes.....	102
4.2.3.2 Advanced Containment System Codes.....	103
4.2.3.3 BWR Pressure-Suppression Pool Dynamics Analysis.....	103
4.2.3.4 Research Program: Advanced Containment System Codes.....	104
4.2.3.5 Milestones.....	104
4.3 COMPONENT CODES.....	105
4.3.1 Present Status.....	105
4.3.1.1 SCORE-EVET.....	105
4.3.1.2 COBRA-IV.....	105
4.3.1.3 K-TIF.....	106
4.3.1.4 K-FIX.....	106
4.3.1.5 SOLA-FLX.....	106
4.3.2 Research Program.....	106
4.3.2.1 COBRA-TF.....	106
4.3.2.2 K-TIF.....	107
4.3.2.3 K-FIX.....	107
4.3.2.4 SOLA-FLX.....	107
4.3.3 Milestones.....	108
4.4 CODE ASSESSMENT.....	108
4.4.1 Developmental Assessment.....	108
4.4.1.1 RELAP-4/MOD 5.....	108
4.4.1.2 RELAP-4/MOD 6.....	108
4.4.1.3 TRAC.....	108
4.4.1.4 THOR.....	108
4.4.1.5 COBRA-IV and COBRA-TF.....	109
4.4.2 Independent Assessment.....	109
4.5 SUPPORTING EXPERIMENTS FOR THE DEVELOPMENT OF BASIC ANALYTICAL MODELS.....	110
4.5.1 Present Status.....	110
4.5.1.1 Studies of Two-Phase Interactions in Countercurrent Flow.....	110
4.5.1.2 LWR Safety-Phenomena Experiments.....	110
4.5.1.3 Containment Tests.....	110
4.5.1.3.1 Pressure-Suppression Tests and Analysis.....	110
4.5.1.3.2 Studies of Dynamic Loads in Pressure-Suppression Containments.....	110
4.5.1.3.3 Modeling of Pool-Swell Hydrodynamics.....	111
4.5.1.3.4 Marviken Tests.....	111
4.5.2 Research Program.....	111
4.5.2.1 Studies of Two-Phase Interactions in Countercurrent Flow.....	111
4.5.2.2 LWR Safety-Phenomena Experiment.....	111
4.5.2.3 Small-Scale Upper Plenum Deentrainment Tests.....	111
4.5.2.4 Containment Tests.....	112
4.5.2.5 Large-Scale Critical Flow Tests.....	112

CONTENTS (Cont'd)

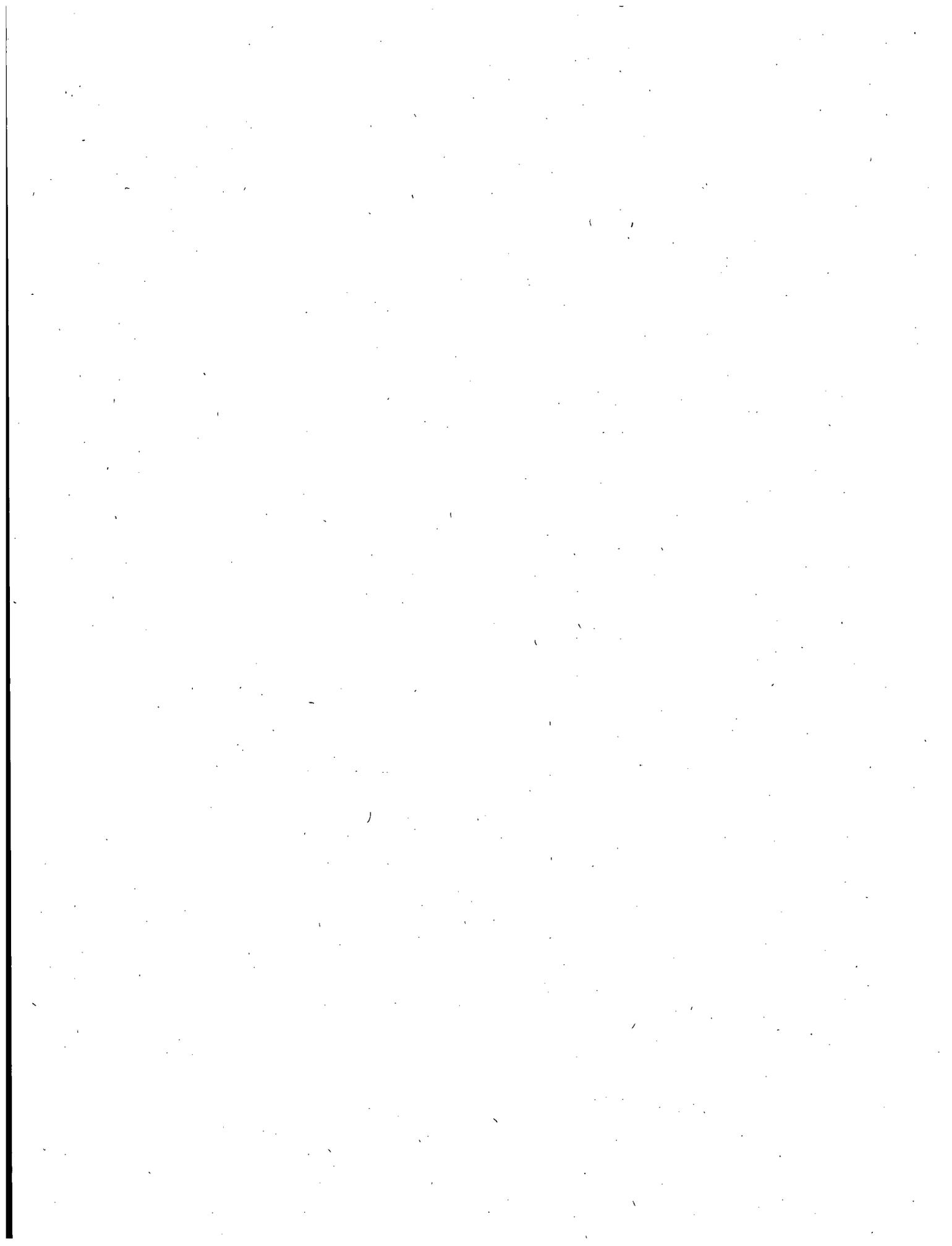
	<u>Page</u>
4.6. CODE UNCERTAINTY STUDIES.....	112
4.7. DEVELOPMENT OF EVALUATION MODEL CODES.....	112
4.7.1 User Needs.....	112
4.7.2 Structure of Evaluation Model Code Packages.....	113
4.7.2.1 BWR LOCA.....	113
4.7.2.2 PWR LOCA.....	113
4.7.3 Schedules.....	113
REFERENCES.....	115
5. PROGRAM PLAN FOR THE METALLURGY AND MATERIALS RESEARCH BRANCH.....	117
5.1 INTRODUCTION.....	117
5.2 FRACTURE AND STRUCTURAL MECHANICS.....	118
5.2.1 Objectives.....	118
5.2.2 Present Status.....	118
5.2.2.1 Vessel Performance.....	119
5.2.2.2 Crack Arrest.....	119
5.2.2.3 Response to Accident Transients.....	120
5.2.2.4 Licensing Criteria.....	120
5.2.3 Research Program.....	120
5.2.3.1 Vessel Performance.....	120
5.2.3.2 Piping.....	120
5.2.3.3 Crack Arrest.....	121
5.2.3.4 Response to Accident Transients.....	121
5.2.3.5 Milestones.....	121
5.3 OPERATIONAL EFFECTS.....	122
5.3.1 Objectives.....	122
5.3.2 Present Status.....	122
5.3.2.1 Irradiation Effects.....	123
5.3.2.2 Annealing.....	123
5.3.2.3 Radiation-Resistant Materials.....	124
5.3.2.4 Cyclic Crack Growth Rate.....	124
5.3.2.5 Steam-Generator Tube Integrity.....	124
5.3.2.6 Stress-Corrosion Cracking.....	124
5.3.2.7 Neutron Dosimetry.....	124
5.3.2.8 Licensing Criteria.....	124
5.3.3 Research Program.....	125
5.3.3.1 Irradiation Effects.....	125
5.3.3.2 Annealing.....	125
5.3.3.3 Radiation-Resistant Materials.....	125
5.3.3.4 Cyclic Crack Growth Rate.....	126
5.3.3.5 Steam-Generator Tube Integrity.....	126
5.3.3.6 Stress-Corrosion Cracking.....	126
5.3.3.7 Neutron Dosimetry.....	126
5.3.3.8 Milestones.....	126

CONTENTS (Cont'd)

	<u>Page</u>
5.4 FLAW DETECTION AND EVALUATION.....	128
5.4.1 Objectives.....	128
5.4.2 Present Status.....	128
5.4.3 Research Program.....	129
5.4.3.1 In-Service Ultrasonic Inspection.....	129
5.4.3.2 Flaw Detection by Acoustic Emission.....	129
5.4.3.3 Advanced Techniques.....	129
5.4.3.4 Milestones.....	130
REFERENCES.....	132
6. PROGRAM PLAN FOR THE RESEARCH SUPPORT BRANCH.....	135
6.1 INTRODUCTION.....	135
6.1.1 Qualification-Testing Evaluation.....	135
6.1.2 Fire Protection Research.....	135
6.1.3 Human Engineering.....	136
6.1.4 Noise Diagnostics.....	136
6.1.5 Inspection and Enforcement Support.....	136
6.1.6 Three-Dimensional LOCA Flow Effects.....	136
6.2 QUALIFICATION-TESTING EVALUATION.....	137
6.2.1 Objectives.....	137
6.2.2 Present Status.....	137
6.2.2.1 Aging.....	137
6.2.2.2 LOCA Testing.....	138
6.2.2.3 Nuclear Source Term Definition.....	138
6.2.3 Research Program.....	138
6.2.3.1 Program Element 1.....	139
6.2.3.2 Program Element 2.....	140
6.2.3.3 Program Element 3.....	141
6.2.4 Milestones.....	141
6.3 FIRE PROTECTION RESEARCH.....	142
6.3.1 Objectives.....	142
6.3.2 Present Status.....	142
6.3.3 Research Program.....	143
6.3.3.1 Program Element 1.....	144
6.3.3.2 Program Element 2.....	144
6.3.3.3 Program Element 3.....	145
6.3.3.4 Program Element 4.....	145
6.3.3.5 Program Element 5.....	146
6.3.3.6 Program Element 6.....	146
6.3.3.7 Program Element 7.....	146
6.3.3.8 Program Element 8.....	146
6.3.4 Milestones.....	147
6.4 HUMAN ENGINEERING.....	147
6.4.1 Objectives.....	147
6.4.2 Present Status.....	147

CONTENTS (Cont'd)

	<u>Page</u>
6.4.3 Research Program.....	148
6.4.4 Milestones.....	149
6.5 NOISE-DIAGNOSTICS RESEARCH.....	149
6.5.1 Objectives.....	149
6.5.2 Present Status.....	149
6.5.3 Research Program.....	150
6.5.4 Milestones.....	150
6.6 INSPECTION AND ENFORCEMENT SUPPORT.....	151
6.6.1 Objectives.....	151
6.6.2 Present Status.....	151
6.6.3 Research Program.....	151
6.7 THREE-DIMENSIONAL LOCA REFLOOD RESEARCH.....	151
6.7.1 Objectives.....	151
6.7.2 Present Status.....	152
6.7.3 Research Program.....	153
6.7.4 Major Milestones.....	153
REFERENCES.....	154
LIST OF ACRONYMS.....	157



## CHAPTER 1

### INTRODUCTION

#### 1.1 OBJECTIVES

The U.S. Nuclear Regulatory Commission (NRC) sponsors confirmatory safety research on light-water reactors in support of the NRC regulatory program. The principal responsibility of the NRC, as implemented through its regulatory program, is to ensure that public health, public safety, and the environment are adequately protected. The NRC performs this function by defining conditions for the use of nuclear power and by ensuring through technical review, audit, and follow-up that these conditions are met. The NRC research program provides technical information, independent of the nuclear industry, to aid in discharging these regulatory responsibilities.<sup>1</sup>

The NRC Nuclear Regulatory Research Program is one of a number of elements that are necessary for a successful, well-regulated nuclear power program in the United States. Although the NRC believes that the nuclear facilities it licenses have an adequate level of safety because of the deliberate conservatisms employed in regulatory safety requirements, confirmatory safety research is required to define with greater precision the safety margins provided in nuclear facilities. The objectives of NRC's research program are the following:

- To maintain a confirmatory research program that supports assurance of public health and safety, and public confidence in the regulatory program.
- To provide objectively evaluated safety data and analytical methods that meet the needs of regulatory activities.
- To provide better quantified estimates of the margins of safety for reactor systems, fuel cycle facilities, and transportation systems.
- To establish a broad and coherent exchange of safety research information with other Federal agencies, industry, and foreign organizations.

The NRC program for confirmatory safety research--in fulfilling its task of providing part of the basis for the assurance of public health and safety, and public confidence in the regulatory program--is largely directed toward defining safety margins. In an oversimplified way, margins imposed in licensing can be said to be divided into (a) an engineering safety margin to allow for normal and abnormal variations in operating parameters plus a desired degree of conservatism and (b) an additional margin to allow for lack of knowledge of the accident processes where such a lack exists. The former must always be retained. It is required not only for the protection of public health and safety but also by codes and regulations imposed by state and Federal governments, including those of the NRC. The NRC research program is providing additional definitions of both types of margins, especially the latter.

#### 1.2 RESEARCH DEFINITIONS

As background for this report, and to aid the reader, four types of research are defined below. This report deals primarily with confirmatory safety research. A separate report<sup>2</sup> has been issued by the NRC on research directed toward improving the safety of nuclear power plants.

##### 1.2.1 DEVELOPMENTAL SAFETY RESEARCH

Developmental safety research is research conducted to evaluate the safety of materials, processes, equipment, etc., that would or might be proposed by an applicant for an NRC license, or a possessor of such a license, in support of an application for a favorable regulatory judgment. Examples of developmental safety research are experiments testing the effectiveness of a specific design for an alternative emergency core cooling system (ECCS).

##### 1.2.2 CONFIRMATORY SAFETY RESEARCH

Confirmatory safety research is research deemed necessary to provide the NRC with a basis for evaluating an application made to it for regulatory judgment, or to provide a basis for a

regulatory requirement or policy, or to provide the NRC with the physical or judgmental capability to regulate the use of nuclear power and materials. If methodologies have to be established, information acquired (data bases built), or analytical approaches assessed, the work involved is research.

Examples of confirmatory safety research are (a) experiments conducted to guide the development of general thermal-hydraulic computer codes for assessing the performance of classes of ECCS systems, (b) experiments performed to assess the validity of these codes, and (c) the development of these codes.

### 1.2.3 TECHNICAL ASSISTANCE

Technical assistance is the NRC use of skilled technical help through contracts and consultation with the intent to use established means (e.g., existing or evolving information or methods) to achieve a well-defined objective of a particular NRC office. An example is the use of existing computer codes to assess the course of postulated accidents in specific reactors. Some adaptation of existing codes may, of course, be necessary.

Another example is the review of existing information to establish the technical basis for a licensing action.

### 1.2.4 RESEARCH FOR IMPROVED SAFETY

Research for improved safety is research on advanced concepts, systems, and processes believed to have potential for improving the safety of nuclear power. Its purpose is to investigate the feasibility, benefits, and costs of the application of these concepts. Research for improved safety can in principle be carried out by industrial and government organizations.

Examples of research for improved safety are (a) the use of more advanced control systems and (b) the application of improved monitoring and accident-diagnostic systems.

## 1.3 REACTOR SAFETY PHILOSOPHY

The principal aim in reactor safety is to protect the public from the release of the radioactive materials (i.e., fission products) contained within nuclear reactors. This aim is accomplished by using three separate barriers to the release of radioactive material: (a) fuel cladding, (b) reactor pressure vessel and piping, and (c) reactor containment.

Figure 1-1 shows the major fuel rod components. Basically, uranium dioxide fuel pellets are contained in a special metal tube called the cladding. Constructed of a zirconium alloy known as Zircaloy, the cladding acts as the first barrier to the release of radioactive material. The Zircaloy tube is designed to retain the fission products during normal operation for the lifetime of the fuel (about 3 years).

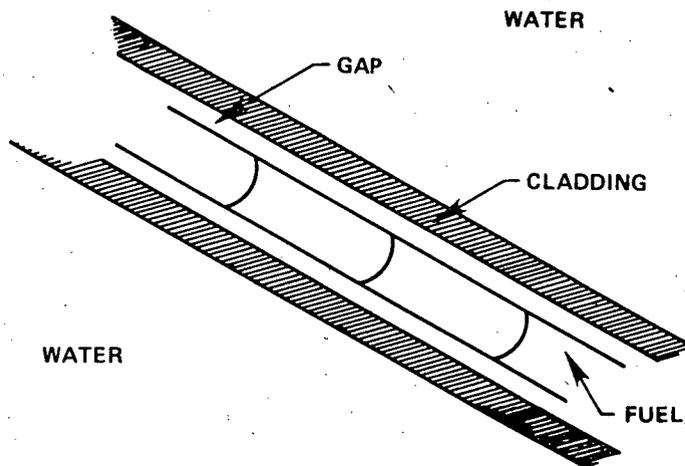


Figure 1-1. Section View of Fuel Rod

Figure 1-2 shows the principal components of a pressurized water reactor (PWR) pressure vessel. The steel walls of the pressure vessel (152 to 254 mm thick) serve as the second barrier to the release of radioactive material. Thus, if fission products were to leak through the fuel-rod cladding, the thick steel pressure vessel and associated coolant piping would contain them.

The third and final barrier to the release of radioactive material is the reactor containment system, a steel-reinforced concrete structure that surrounds the reactor pressure vessel and the coolant piping (Fig. 1-3). It is designed to contain the fission products that might be released as a result of a simultaneous failure of the other two barriers.

The basic approach to ensuring the safety of nuclear plants is to design the plants according to a "defense-in-depth" philosophy, that is, to build in three levels of nuclear safety: (a) design and fabricate the plant for maximum safety, (b) provide protective systems to monitor and correct abnormal conditions, and (c) install engineered safeguard features to mitigate accident consequences. This philosophy is reflected in designing the three barriers against the release of radioactive material.

#### 1.4 REACTOR SAFETY REGULATORY PROCESS<sup>3</sup>

Before a nuclear power plant can be built at a particular site, a construction permit must be obtained from the NRC. A major part of a construction permit application is the preliminary safety analysis report (PSAR). This document describes the design of the proposed plant and presents comprehensive data on the proposed site. It evaluates the range of potential occurrences, including a set of design-basis accidents and the safety features provided to prevent them or, if they should occur, to mitigate their effects on both the public and the facility's employees.

The NRC staff reviews the design of the proposed plant and site to ensure that adequate provisions to protect public health and safety and the environment are included. Design methods and calculation procedures are examined to establish their validity. Checks of actual calculations and other procedures of design and analysis are made by the staff to establish the validity of the applicant's design.

When the NRC staff has completed the first phase of a review, the Advisory Committee on Reactor Safeguards (ACRS), an independent committee established by Congress to advise the NRC on nuclear safety, reviews in public sessions each application for a construction permit and reports in a public letter to the Chairman of the NRC on the acceptability of the plant. The law requires that, before a construction permit is issued for a nuclear power plant, a public hearing be held to allow for full public participation in the NRC decision-making process. The public hearing is conducted by a three-member Atomic Safety and Licensing Board appointed from the NRC's Atomic Safety and Licensing Board Panel. Interested parties opposed to the plant have the right to participate in these hearings. The Board considers all the evidence presented in the hearing, together with proposed findings of fact and conclusions of law filed by the parties, and issues an initial decision.

The Board's initial decision is subject to review by an Atomic Safety and Licensing Appeal Board on its own motion or in response to exceptions filed by any party to the proceeding. The decision may also be reviewed by the Nuclear Regulatory Commissioners. The final NRC decision regarding a licensing action is subject to judicial review in the Federal courts.

When the construction of a nuclear facility has progressed to such a point that final design information and plans for operation are ready, the applicant submits a final safety analysis report (FSAR) in support of an application for an operating license. The FSAR furnishes pertinent details on the final design of the plant and supplies plans for operation, procedures for coping with emergencies, and security provisions for protection against sabotage. This information is also reviewed in detail by the NRC staff and then independently evaluated by the Advisory Committee on Reactor Safeguards in open sessions. The ACRS advice is provided to the Commission by public letter.

Throughout the construction and lifetime of the plant, periodic inspections are conducted to audit safety and compliance with license conditions.

#### 1.5 WATER-REACTOR SAFETY-RESEARCH AREAS

The NRC's confirmatory safety research program on light-water reactors (LWRs) is structured to provide additional and/or independent information to gain confidence that the margins of safety identified in the licensing review are well defined and quantified. In line with the defense-in-depth policy, the principal areas of NRC research in the field of LWR safety may be categorized as follows:

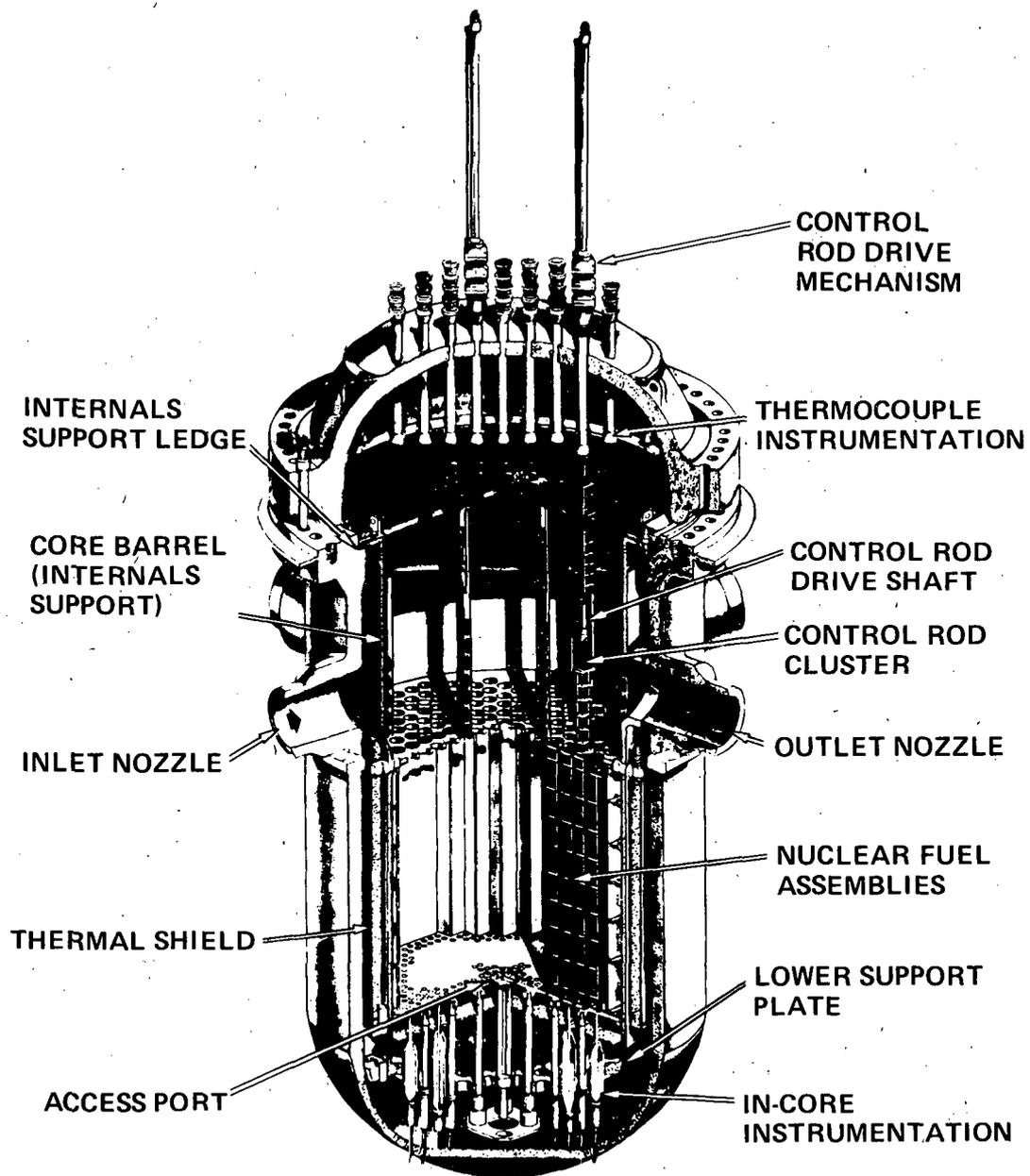


Figure 1-2. PWR Vessel and Fuel Assemblies

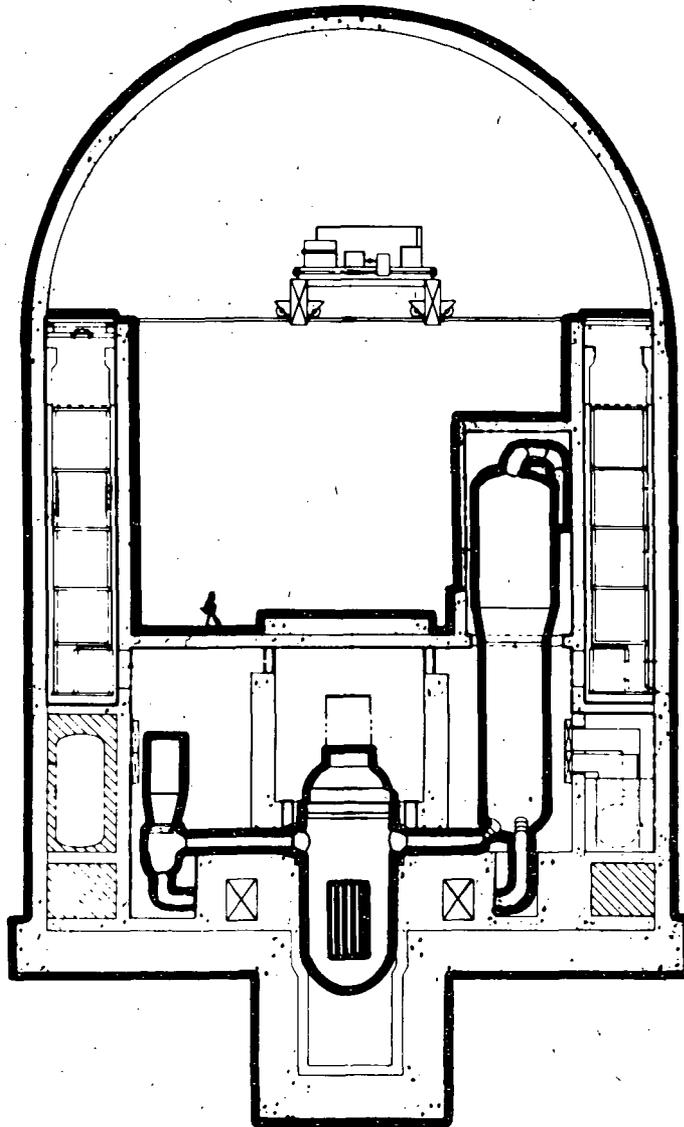


Figure 1-3. Elevation Section of a Typical PWR Containment Vessel

<u>Research Area</u>	<u>Cognizant NRC Research Branch</u>
Thermal-hydraulic tests of postulated accidents* and the effectiveness of the engineered safety features	Systems Engineering Branch
Fuel-rod behavior in postulated accidents and associated failure limits	Fuel Behavior Research Branch
Computer code development for accurately predicting the consequences of postulated reactor accidents	Analysis Development Branch
Safety design and protection of the integrity of the reactor pressure vessel and piping	Metallurgy and Materials Research Branch
Operational safety aspects of nuclear power-plant operation	Research Support Branch

Figure 1-4 shows how the research programs are interrelated to assess thermal-hydraulic and fuel-behavior computer codes, which can be used to quantify the safety margins in the operation of nuclear power plants. Detailed descriptions of the program plans for each of the five categories or elements are provided in the Branch Program Plans, Chapters 2 through 6 of this report. A brief description of the principal tasks and objectives of the confirmatory safety research under each program element is given in the sections that follow.

### 1.5.1 SYSTEMS ENGINEERING BRANCH

#### 1.5.1.1 Objectives

The systems engineering program element is an experimental program designed to provide measured physical data for the development and assessment of computer codes for analyzing the performance of emergency core cooling systems.

#### 1.5.1.2 Background

The loss-of-coolant accident (LOCA) is one of the principal accidents studied by the NRC in assessing the safety of nuclear power plants. Particular attention is focused on the ECCS, which are provided to keep the nuclear fuel cooled in the event of a LOCA. The LOCA can be divided into several phases: blowdown (depressurization), refill (ECC water entering the lower plenum), and reflood (ECC water entering the core). It is necessary to understand the thermal-hydraulic behavior of these phases in order to assess the overall LOCA/ECCS behavior. Specific attention is focused on (a) whether or not the critical heat flux (CHF) is reached during blowdown and (b) the subsequent heat transfer after CHF and during reflood. This is because at CHF there is a marked decrease in the coefficient of heat transfer between the fuel-rod cladding and the cooling water, which can result in an increase in the fuel-rod surface temperature. If the surface temperature of the fuel rod becomes too high, damage to the fuel-rod cladding may occur, and this could lead to the release of radioactive material. Thus, it is important to understand the heat-transfer processes during the various phases of a LOCA.

The systems engineering program element encompasses experimental subelements designed to provide system thermal-hydraulic data under LOCA/ECCS conditions. To address thermal-hydraulics behavior during a LOCA, the systems engineering program consists of two major parts: (a) LOCA separate-effects experiments and (b) LOCA integral tests.

For pressurized water reactors (PWRs), LOCA separate-effects experiments address each of the phases of the LOCA (i.e., blowdown, refill, and reflood, including steam-water mixing and lower plenum filling). The program subelements are:

- The PWR blowdown heat-transfer test (PWR-BDHT).
- ECC bypass research.
- The full-length emergency cooling heat-transfer separate effects and systems effects test (FLECHT-SEASET).

\*The term "postulated accident" is used here to describe a range of design-basis accidents (DBAs) the license applicants must consider in the design of a plant.

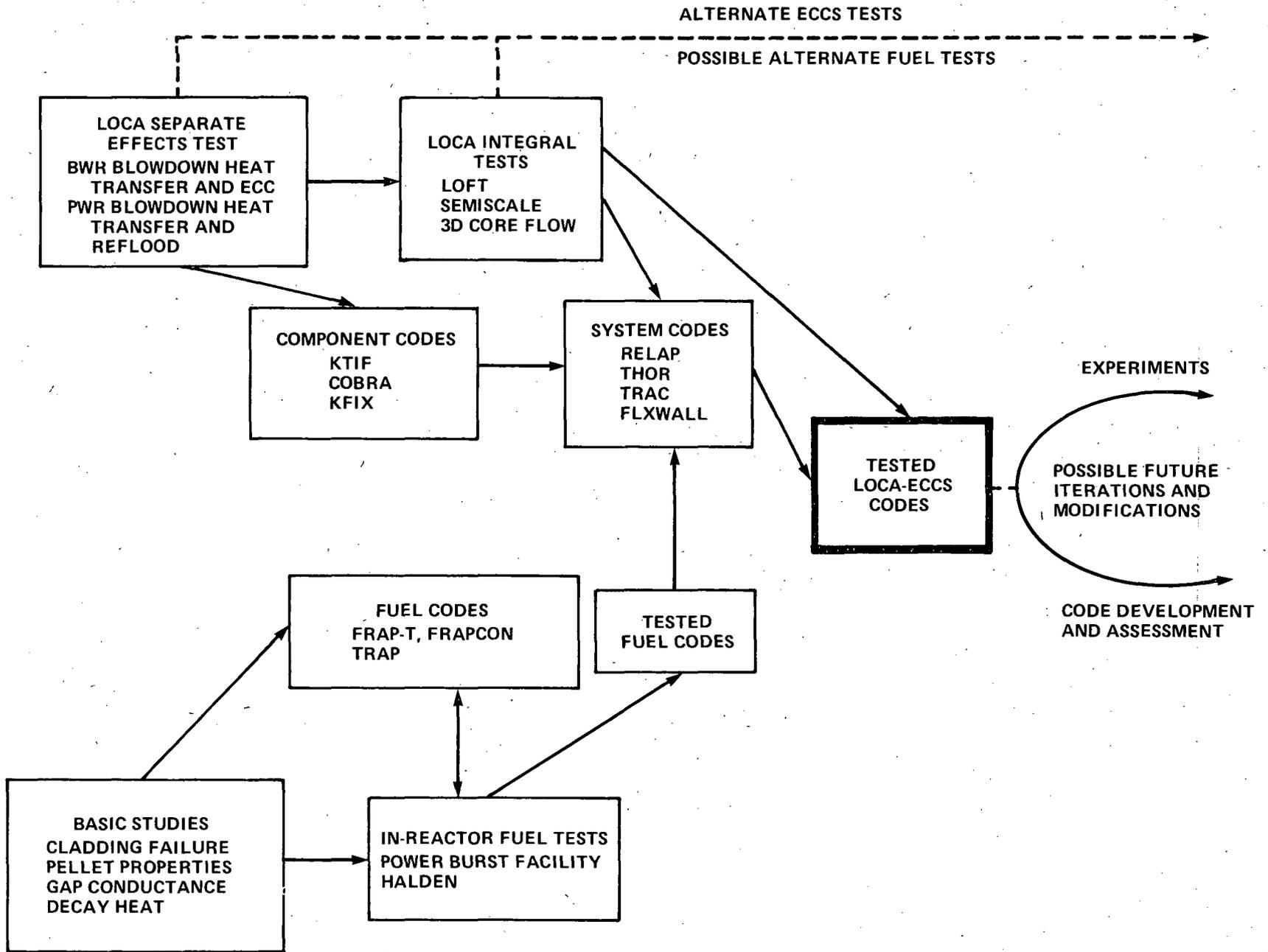


Figure 1-4. Research Plan for Confirmed LWR LOCA Analysis Methods

For boiling water reactors (BWRs), the LOCA separate-effects experiments address primarily BWR blowdown/ECC interactions through tests in the BWR BD/ECC facility. The NRC Office of Nuclear Regulatory Research plans to do research on BWR countercurrent flow limitations and reflood effects in the BD/ECC facility and other separate-effects tests loops.

The LOCA integral tests include:

- Tests in the loss of fluid test (LOFT) facility to study the coupled nuclear and thermal hydraulics of the LOCA during blowdown and reflood.
- Tests in the Semiscale facility to study the thermal-hydraulic aspects of the LOCA in nonnuclear systems.
- Three-dimensional core flow distribution tests to determine the three-dimensional flow distribution effects inside the core and upper plenum of a PWR during the later stages of blowdown and reflood. (See Chapter 6 for more information on these tests.)

## 1.5.2 FUEL BEHAVIOR RESEARCH BRANCH

### 1.5.2.1 Objectives

The fuel behavior program element is an experimental and analytical program designed to provide a more detailed understanding of the response of nuclear fuel assemblies to abnormal or postulated accident conditions.

### 1.5.2.2 Background

The fuel-rod cladding provides the first barrier against the release of radioactive materials in a postulated accident, one of which is the LOCA. Because of the importance of the cladding, it is necessary to understand its coolability and how it can be affected by the course of the accident.

Under certain conditions, the Zircaloy cladding can be affected by the behavior of the fuel pellets enclosed within it. The cladding can also creep and collapse, become embrittled through oxidation in a steam environment, and have its temperature influenced by the heat conductance of the fuel-to-cladding gap and the available stored heat in the fuel. Therefore, in order to assess the behavior of the cladding under abnormal or accident conditions, it is necessary to understand the cladding environment. This makes it necessary to study the internal conditions of the fuel and gas gap as well as such external conditions as the occurrence of the critical heat flux and the ballooning of adjacent fuel rods. If it is postulated that a fuel rod will release radioactive material, then it becomes necessary to obtain information on this material and how it can be transported. Finally, should the postulated accident result in fuel melting, then information is needed on how this molten fuel interacts with and is, in turn, influenced by its environment so that some understanding may be obtained of the containment of the radioactive fuel.

The information generated in the fuel-behavior program element is used to develop physical models for fuel analysis codes and fission-product transport codes. These codes are then validated through integrated in-reactor tests.

The fuel-behavior program element consists of experimental and analytical efforts in four major areas:

- Basic studies of the fuel rod.
- In-reactor tests of fuel rods.
- Development and assessment of fuel codes.
- Experimental and analytical studies of fuel meltdown and fission-product release.

## 1.5.3 ANALYSIS DEVELOPMENT BRANCH

### 1.5.3.1 Objectives

The code development program element is an analytical development program designed to provide better digital computer codes for computing the behavior of full-scale reactor systems under postulated accident conditions. These codes are a key part of the safety assessment of nuclear power plants.

### 1.5.3.2 Background

Most of the present LWR safety code development work is aimed at assessing the consequences of a LOCA and the behavior of the emergency core cooling systems in PWRs and BWRs. However, many of these codes are being developed in a flexible, modular fashion to be easily applicable to other postulated system disturbances, such as anticipated transients without scram and reactivity-initiated accidents. The code development program element is an important part of the reactor safety research program because the computational techniques embodied in the computer codes provide the means of applying the basic experimental and analytical information to commercial nuclear power plants. These computational methods will be used to assess (a) ECCS performance in mitigating the consequences of a postulated LOCA, (b) the influence of the various parts of the reactor system on the course of a LOCA and in preventing fission-product releases, and (c) the response of the reactor system to other postulated accidents.

In modeling an accident like a LOCA, one is dealing with rather complex phenomena. The pressurized water flashes to steam at the break, forcing one to deal with steam-water (two-phase) mixtures, often moving at unequal velocities and not necessarily in thermal equilibrium with each other. The forces unleashed in the LOCA act on the reactor vessel and its supports. The injected ECC water becomes mixed with steam and may be swirled around the downcomer before entering the lower plenum to reflood the core. Steam formed in the core may inhibit the injection of water. Furthermore, all of these events are taking place in more than one dimension and are changing rapidly as a function of time and location. In order to obtain a realistic description of these events, it is necessary to develop analytical and programming techniques beyond those currently available.

The code development program element encompasses analytical subelements designed to model accidents, especially the LOCA, in LWRs. To address the LOCA, the code development program element is divided into three principal subelements: (a) systems codes for the overall analysis of a nuclear power plant, (b) component codes for a more detailed look at specific components, and (c) code evaluation against experimental results to ensure the desired predictive capability. All of these activities have received a top-priority rating from the various user groups.

## 1.5.4 METALLURGY AND MATERIALS RESEARCH BRANCH

### 1.5.4.1 Objectives

The metallurgy and materials research is concerned with the integrity of the primary-system pressure boundary in LWRs. It is an experimental and analytical program designed to upgrade the basis for design, fabrication, operation, and inspection criteria, as well as for the analytical procedures required to evaluate performance under normal, upset, faulted, and accident conditions for the pressure vessel, piping, and associated components of the primary-system pressure boundary of LWRs. Thus, a primary goal is to improve the definition of failure probabilities and failure modes, and to establish ways by which the failure probabilities can be reduced if this is considered necessary.

### 1.5.4.2 Background

Special attention is given to the study of the primary-system pressure boundary in LWRs because of the need to contain the nuclear core materials at all times and thus the need to understand the types of failures that might lead to a breach of this containment. The primary-system pressure boundary of current LWRs includes (a) a steel pressure vessel with a thickness of 6 to 12 inches, (b) steam-generator tubes, and (c) primary piping as much as 4 inches in thickness. These materials have been studied extensively to develop information on trends for mechanical behavior under appropriate conditions of temperature, stress, neutron irradiation, and reactor environment. These studies have necessarily been done with laboratory-scale test specimens because the section thickness and massive size of reactor components, coupled with the necessity of simulating long-term neutron irradiation, make testing of full-scale vessels or components either prohibitively expensive or almost technically unfeasible. Nevertheless, the in-reactor behavior of the full-section-thickness materials and components must be predictable from data obtained largely in small-scale laboratory tests. An important aspect of the work, therefore, is to test materials in a range of thicknesses to validate the analytical prediction techniques. Thus, despite existing knowledge on the properties of primary-system component materials, improvements in information are still sought to round out the basis for judgments affecting continuing reactor safety.

The primary-system integrity program element encompasses those RES experimental and analytical subelements that are designed to provide information on the integrity of the primary-system pressure boundary of LWRs. It consists of three major subelements: (a) fracture mechanics, (b) operational effects, and (c) flaw detection.

The fracture-mechanics work encompasses (a) reactor vessel and piping-system performance under pressure and thermal loading; (b) crack initiation, propagation, and arrest (including static and dynamic studies and the use of irradiated specimens); and (c) response to all operational and postulated conditions. The work on vessel response to transients encompasses thermal shock and steam-line-break accident conditions to assess the effects of abnormal pressures and shock following the injection of cold ECC water after a LOCA.

The operational effects work is directed at obtaining data on (a) irradiation embrittlement, (b) annealing and re-irradiation, (c) residual element effects, (d) cyclic crack growth, (e) steam-generator tube integrity, (f) intergranular stress-corrosion cracking and sensitization, and (g) neutron dosimetry.

The flaw detection and evaluation work covers (a) improved ultrasonic characterization of flaws, (b) acoustic emission studies of flaw growth in piping and pressure vessels and of flaws produced during welding, and (c) advanced nondestructive examination techniques.

In general, the primary-system integrity subelements are geared (a) to produce information to develop and confirm analysis procedures for crack propagation and arrest, steam-line break, thermal shock and pneumatic loading cyclic crack growth, and irradiation embrittlement, all of which help establish the integrity of the primary-system pressure boundary, and (b) to develop basic criteria for testing procedures to ensure accuracy, value, reproducibility, and correlation of results. Ultimately, the results will be incorporated into improved industry code rules and standards for improved LWR safety designs and will help upgrade the NRC basis for decisions on operating reactors. All of the primary-system integrity projects that have been rated have received the highest or next-to-highest priority.

#### 1.5.5. RESEARCH SUPPORT BRANCH

##### 1.5.5.1 Objectives

The Research Support program element is an experimental and analytical program designed (a) to support other NRC offices in the development and confirmation of regulatory standards and guides, and (b) to provide research information on reactor operational safety matters.

##### 1.5.5.2 Background

Aside from the postulated design-basis LOCA that is used by the NRC licensing staff to evaluate the engineered safety features of a nuclear power plant, there are a number of operational safety topics of interest. Many of these topics derive from the day-to-day evaluation of plant operating behavior and are often the subject of specific regulatory guides and standards. In keeping with the topics of interest of the licensing, standards, and inspection staffs, NRC has established a Research Support program element to manage research in support of such specific operational safety matters as fire protection, component qualification-testing evaluation, noise diagnostics, and human engineering studies. In addition, management of the international two- and three-dimensional core reflood flow distribution experiment is in this program element. Future non-LOCA water-reactor safety research performed in support of the recent ACRS report<sup>4</sup> and the NRC plan<sup>2</sup> for improved safety will be included in this program element.

#### 1.6 AVAILABILITY OF RESEARCH INFORMATION

Several references<sup>5-7</sup> have provided technical summaries of the NRC's LWR safety research program. In addition, each of the principal research laboratories or contractors issues formal progress and topical reports. These are available for public viewing or copying at the NRC's Public Document Room (1717 H Street, N.W., Washington, D.C. 20555), and they are also sold to the public through the National Technical Information Service (NTIS), U.S. Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161. Semiannual bibliographic listings of these NRC-sponsored reports are published by the Nuclear Safety Information Center (NSIC) and sold through NTIS. NSIC, in turn, is an NRC/DOE-supported focal point for collecting, storing, evaluating, and disseminating nuclear-safety information. Inquiries about NSIC activities should be sent to Director, Nuclear Safety Information Center, Oak Ridge National Laboratory, P.O. Box Y, Oak Ridge, Tennessee 37830.

NRC-sponsored computer codes, which are developed under the research projects described in this report, are made publicly available at the National Energy Software Center (formerly the Argonne Code Center), Argonne National Laboratory, 9700 South Cass Avenue, Argonne, Illinois 60439.

As a final note, the NRC Press Release No. 76-76<sup>8</sup> states: "Data will now be made directly available to the nuclear industry and the general public before the release of formal reports." This press release elaborates by noting that "the NRC will make available data tapes and operational computer codes on research programs dealing with technical questions associated with postulated loss-of-coolant accidents (LOCA) in light-water-cooled reactors." This policy allows a more rapid dissemination of NRC research information to interested parties who request it. The press release goes on to note that "proprietary information would, of course, not be released" and that "persons requesting the preliminary information must reimburse the NRC contractors for their expenses in preparing copies of the data tapes and the operational computer codes." Requests should be submitted to the Research Support Branch, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

The scheduler and financial status of the principal NRC water reactor safety research projects may be found in the NRC Water Reactor Safety Research Status Summary Report ("Buff Book" Volume 1) available through NTIS and the PDR, as NUREG-0135 series.

When a reasonably complete body of research information is available, a research information letter (RIL) is released summarizing this information and providing additional background references. These research information letters are often in response to specific requests for research from other NRC program offices and they are made available in the NRC Public Document Room. (Note: Most of the new research is initiated in response to formal requests from one of the other NRC program offices: Office of Nuclear Reactor Regulation, Office of Standards Development, Office of Inspection and Enforcement, and Office of Nuclear Materials Safety and Safeguards).

Reference 9 contains an earlier description of the water reactor safety research program.

#### REFERENCES

1. Energy Reorganization Act of 1974, U.S. Public Law 93-348, approved October 11, 1974.
2. U.S. Nuclear Regulatory Commission, Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants, NUREG-0438, April 1978. Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.
3. Reference 2, pp. 3 and 4.
4. Advisory Committee on Reactor Safeguards, Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, USNRC Report NUREG-0392, December 1977. Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.
5. L. S. Tong and G. L. Bennett, "NRC Water-Reactor Safety-Research Program," Nuclear Safety, Vol. 18, No. 1, January-February 1977.
6. T. E. Murley, L. S. Tong, and G. L. Bennett, Summary of LWR Safety Research in the USA, USNRC Report NUREG-0234, May 1977. Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.
7. U.S. Nuclear Regulatory Commission Annual Reports for 1975, 1976, and 1977. Available for purchase from the U.S. Government Printing Office, Washington, D.C. 20402.
8. U.S. Nuclear Regulatory Commission, Press Release No. 76-76, "NRC to Expand Access to Water Reactor Safety Research Data," March 31, 1976.
9. U.S. Nuclear Regulatory Commission, Reactor Safety Research Program, NUREG-75/058, June 1975. Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

CHAPTER 2  
PROGRAM PLAN  
FOR THE SYSTEMS ENGINEERING BRANCH

2.1 INTRODUCTION

The objective of the safety research sponsored by the Systems Engineering Branch is to provide experimental data to establish and confirm analytical methods for assessing the thermal-hydraulic response of the reactor primary coolant system components and of the reactor containment to possible abnormal and accident conditions. Current and planned research areas included in the Program Plan of the Systems Engineering Branch are (a) integral system studies and tests, and (b) separate-effects studies and tests.

2.2 INTEGRAL SYSTEM STUDIES AND TESTS

Integral systems studies and tests provide a means of analyzing and determining experimentally the interrelationships of the various phases of a loss-of-coolant accident (LOCA) by the use of a facility that contains the principal system components. The NRC is currently sponsoring research of this type at the Loss of Fluid Test (LOFT) facility and at the Semiscale facility, both located at the Idaho National Engineering Laboratory (INEL). The primary objectives of the integral system studies and tests are as follows:

- To simulate experimentally the major effects that have been calculated to occur during a LOCA in the reactor coolant system of a pressurized water reactor (PWR).
- To aid in assessing the capability of analytical methods to predict the response of large power reactors to a LOCA, the performance of engineered safety systems, and the margins of safety designed into their performance.
- To provide system tests that might uncover any unexpected events or thresholds that are currently not being considered in the analysis of plant response or in the design of engineered safety systems.
- To explore the use of alternative emergency core cooling (ECC) concepts.
- To provide experiments which will be helpful in assessing the degree to which current analysis methods give a conservative description of the effects of postulated abnormal conditions such as a steam-line break and anticipated transients without scram.

2.2.1 LOFT

2.2.1.1 Objectives

The primary objectives listed above are all addressed in the LOFT program. The LOFT facility is the only safety test facility in the world incorporating a nuclear core and primary and secondary cooling systems. Specific additional objectives to be attained by means of the LOFT facility's unique capabilities include the following:

- To combine information on heat transfer during blowdown, ECC bypass, and reflood into an integrated sequence of events to assess analytical methods.
- To investigate and test alternative emergency core cooling system (ECCS) configurations, such as upper and lower plenum injection, that are expected to enhance plant protection during a LOCA. These concepts should be as insensitive as possible to reactor design parameters and proper functioning of other components (e.g., steam generator or reactor containment), have a sufficiently diverse and abundant flow for their adequacy to be determined without unduly complex evaluation techniques, and be extremely reliable.
- To study anticipated transients with and without scram and other selected transient and accident conditions to which the reactor may be subjected.

### 2.2.1.2 Present Status

LOFT is a 55-MWt PWR facility intended to simulate the behavior of 1000-MWe PWRs in carefully conducted loss-of-coolant experiments (LOCE). The nuclear core, approximately 5.5 feet long and 2 feet in diameter, contains 1300 fuel pins and four control assemblies of typical PWR design. The ratio of primary coolant system volume to core power is similar to that of commercial PWRs. Primary coolant system subvolumes (e.g., cold leg, core region, and hot leg) are also designed with ratios similar to those of PWRs. The unbroken PWR coolant loops are approximately simulated by the single unbroken circulating loop in the LOFT primary system, and the postulated broken PWR loop is simulated by the LOFT blowdown loop with passive components.

While preparations for the installation of a nuclear were proceeding, a series of nonnuclear tests were conducted. During the first four experiments, the core was represented by a device that simulated the core as resistance to coolant flow. The fifth test was conducted with the nuclear core in position but not at power.

#### 2.2.1.2.1 The Nonnuclear Experiment Series

The nonnuclear series consisted of the five experiments listed in Table 2-1. The series was designed to provide operational experience and baseline experimental data helpful to the interpretation of the more complex nuclear experiments.

In every case the primary coolant is pressurized and heated to 540°F, the approximate inlet temperature for nuclear operation. The major differences between the nonnuclear and the nuclear series are that in the nuclear series the coolant is heated as it flows through the core. Because of this extra heat, the coolant condition closely resembles that to be expected in a commercial power reactor during a LOCA.

TABLE 2-1

LOFT NONNUCLEAR EXPERIMENT SERIES L1<sup>a</sup>

Experiment Designation	Break Size	Break Type	Location of ECC Injection	System Pressure (bars)
L1-1	One-half full break area	Hot leg	Cold leg	93
L1-2	Full break area	Cold leg	Cold leg (delayed)	155
L1-3	Full break area	Cold leg	Lower plenum	155
L1-4	Full break area	Cold leg	Cold leg	155
L1-5 <sup>b</sup>	Full break area	Cold leg	Cold leg	155

<sup>a</sup>In all experiments, the break opening time was 17.5 msec.

<sup>b</sup>With nuclear core in place.

The first nonnuclear experiment (L1-1) was a hot-leg break. In this case the coolant flowed up through the reactor vessel and out the break in the blowdown loop, to the suppression tank. Since the break was only 50% of a full break area and the test was initiated at an intermediate pressure, the hydraulic forces on the system were intermediate. The experiment therefore provided a partial check on the system's structural adequacy before it was subjected to more severe loads.

The second experiment (L1-2) was a full-sized cold-leg break at full system pressure, subjecting the system to the most severe anticipated hydraulic loads and using delayed ECC injection to measure the downcomer hot-wall delay. In the third experiment (L1-3), a full-sized cold-leg

break, ECC injection occurred in the lower plenum. This test showed the ideal pressure vessel refill with neither delay nor bypass. The fourth experiment (L1-4) was a full-sized cold-leg break, with the time and location of ECC delivery simulating those in a large PWR. By comparison with L1-2 and L1-3 it was possible to isolate phenomena that tend to impede the movement of ECC water from the cold-leg injection point to the core region.

The final nonnuclear experiment (L1-5) was performed with the nuclear core installed but with the core not generating power. The objective was to obtain the zero-power baseline result for comparison with subsequent nuclear power experiments. LOFT has since achieved criticality and is undergoing power ascension testing.

#### 2.2.1.2.2 Results of the First Five Experiments

The most important single question addressed by the LOFT experiments is, "How well does the measured behavior agree with the behavior predicted by best-estimate and safety computer codes?" (i.e., codes that were developed from separate-effects experiments and experiments with a smaller scale system designed to predict the behavior of large PWRs during a LOCA). The detailed results given in References 1 through 8 show that the agreement is generally very good.

Also of primary importance are the scaling procedures used in relating LOFT results to predictions for a large PWR. The same scaling procedures were used in the design of the smaller (Semiscale) experiment facility. The Semiscale facility is a scale model of LOFT in which the core is simulated by electrical heaters. Its power is one-thirtieth that of LOFT. Each experiment to be run in LOFT is first run in Semiscale. Therefore, a comparison of the experimental results from the two facilities should demonstrate how well the scaling procedures have been developed. Figure 2-1 is a comparison of the transient mass flows in the operating-loop cold leg for duplicate experiments in Semiscale and LOFT. These results are typical and provide strong confidence in the scaling procedures. Figure 2-1 includes a computer code (RELAP-4) prediction for LOFT which supports the earlier conclusion that agreement with the measurements is good.

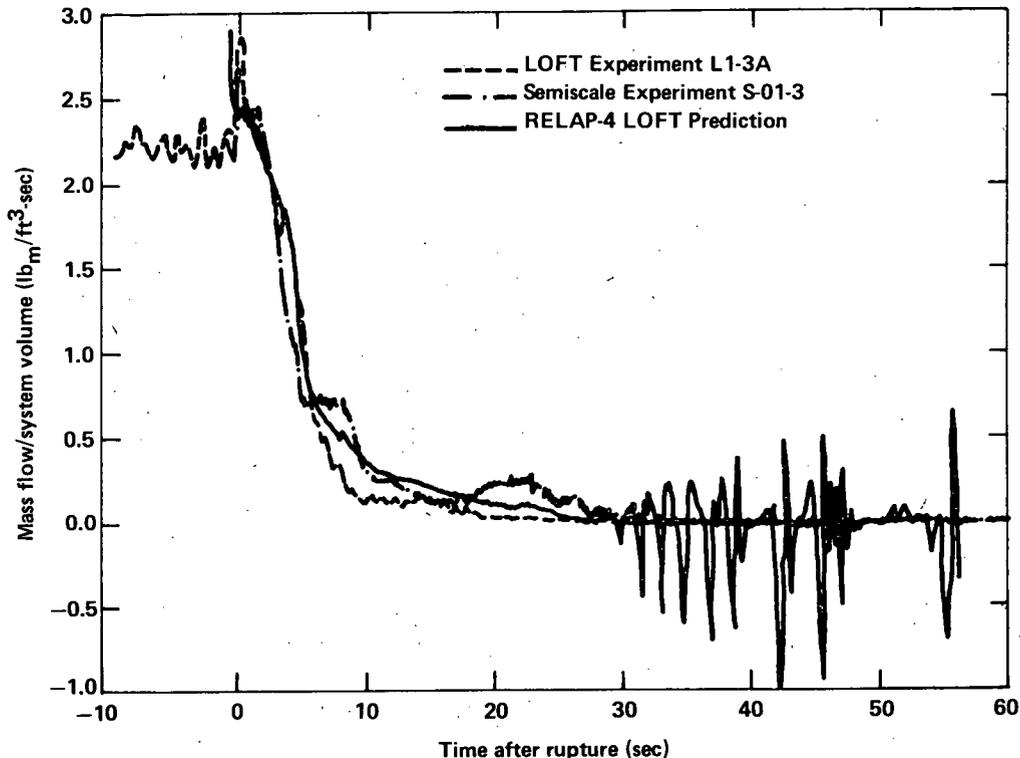


Figure 2-1. Comparison of Transient Mass Flows in the Operating-Loop Cold Leg for Counterpart Experiments in Semiscale and LOFT

The L1-1 Experiment.<sup>2</sup> This was the first blowdown experiment attempted in LOFT and demonstrated that a LOCA experiment could be run as planned. The important conclusions were as follows:

- Measurement of the hydraulic loads showed that the most severe experiments could be performed without fear of structural damage to the facility.
- The bypass flow of ECC water around the top of the downcomer and out the broken loop was less than had been predicted.
- The sweep-out of water from the lower plenum was lower than had been predicted.

The significance of the last two conclusions is that, for a hot-leg break, the RELAP-4 prediction is conservative as to the amount of water remaining in the LOFT reactor vessel at the end of blowdown. This means that reflood in a PWR would probably begin earlier than is predicted by this code.

The L1-2 Experiment.<sup>3</sup> Experiment L1-2 was run with delayed ECC injection. This permitted a measurement of the hot-wall delay without interference from countercurrent flow in the downcomer. The important conclusions were as follows:

- The hot-wall delay was much shorter than predicted.
- Sweep-out of water from the lower plenum for a full-sized break was less than had been predicted.
- Loads in the suppression tank were much lower than had been predicted. This may have beneficial results for BWR safety evaluation.

The L1-3 Experiment.<sup>4,5</sup> A procedural error caused the first attempt at this experiment to be less than successful. The experiment was therefore repeated successfully as L1-3A. The results included data for comparison with L1-2 and L1-4 to separate the effects that impede ECC delivery. The important conclusions were as follows:

- There was no sweep-out of ECC water from the lower plenum.
- System behavior during transients initiated under the same conditions showed excellent repeatability.

The L1-4 Experiment.<sup>7</sup> The L1-4 experiment was run with cold-leg injection during blowdown and was representative of large-reactor performance during the blowdown period for typical ECC injection. Comparison of results with the L1-2 and L1-3 experiments led to the following conclusions:

- ECC delivery to the core inlet was rapid, and hot-wall effects were small.
- The lower plenum was not voided.
- Asymmetric flow was observed in the downcomer.
- ECC bypass behavior was as expected.

The results also confirmed two important and known deficiencies in the RELAP model: a split downcomer model is needed to obtain good agreement between predictions and measurements of reactor vessel inflow and outflow; a nonequilibrium model is required to predict the depressurization rate after ECC injection begins.

The L1-5 Experiment.<sup>8</sup> Experiment L1-5 was run with cold-leg injection during blowdown with the same conditions as the L1-4 experiment but with the core in place and not producing power. This provided a base case against which to compare the future nuclear experiments. The following conclusions were reached:

- The core did not experience significant displacement during the blowdown.
- The reactor vessel voided uniformly in the radial direction, and the core rewet uniformly in the axial direction.

- The depressurization rate was faster than that in the previous experiments due to the smaller system volume resulting from the presence of the core.
- The conclusions of the L1-4 experiment were further substantiated.

### 2.2.1.2.3 Preparations for Nuclear Experiments

The nuclear core has been installed, all nuclear instrumentation and control systems have been tested, and the reactor has reached criticality. After the core was characterized, the L1-5 experiment was run, subjecting the core to the blowdown conditions experienced in the L1-4 experiment. The core was then recharacterized to ensure that no damage had occurred. After a series of power-range tests, to exercise all systems at power, LOFT will be ready to begin loss-of-coolant experiments during nuclear operation. This was scheduled for early 1979, but because of changes in schedule, the nuclear tests are expected to begin toward the end of 1978 (see Table 2-2).

TABLE 2-2  
TENTATIVE LOFT EXPERIMENT SCHEDULE

Designation	Nature of Experiment or Series	Date*
L2-2	200% double-ended cold-leg (DECL) break with core at ~8-kW/ft peak to cause periods of film boiling at modest cladding temperatures (e.g., 1000°F)	December 1978
L4-1	200% DECL break with core at ~8kW/ft peak with lower plenum ECC injection; modest cladding temperatures (e.g., 1000°F)	April 1979
L2-3	200% DECL break with core at 12-kW/ft peak (current licensing value in today's commercial cores) to cause periods of film boiling at higher temperatures (e.g., 1500°F)	September 1979
L2-4	200% DECL break with core at ~16-kW/ft peak (highest design rating in today's commercial cores)	January 1979
L2-5	200% DECL break with core at 12-kW/ft peak, certain hardware assumptions according to Appendix K of 10 CFR 50	May 1980
L2-6	200% DECL break with portion of core fuel pressurized (to be decided whether future experiments should have pressurized fuel or not)	September 1980
L3	Small- and intermediate-size break sensitivity study	1981
L4	Alternative ECCS; e.g., bypass vent-valve, pump-suction injection, hot-leg or upper head injection	1982-1983
L5	Hot-leg breaks	1984
L6	Anticipated transients without scram	1985
L7	Steam-generator tube rupture studies	1985

\*Dates are based on the current best estimate schedule.

### 2.2.1.3 Research Program

#### 2.2.1.3.1 The First Nuclear Series

The goal of this series is to measure the behavior of LOFT systems during a cold-leg break at core heat generation rates up to the current maximum allowable value for large PWRs. The ECC systems will be varied in performance to determine certain conservative constraints imposed by 10 CFR 50, Appendix K.

The power for the first nuclear experiment (L2-2) was selected so as to result in a short period of film boiling on the fuel. The highest cladding temperatures will be in the neighborhood of 1000 K (1340 F). For experiment L2-3, the highest cladding temperatures will be in the neighborhood of 1600 F and may remain there for a few tens of seconds. Between experiments the core's integrity will be determined by measurement and inspection. It may be necessary to replace some fuel assemblies before the final experiment.

The L2-4 experiment will be initiated at a peak linear heat-generation rate that exceeds the nominal ratings and is about equal to the maximum permissible values. The L2-5 experiment will return to the current operating-limit linear heat-generation rate and will study variations in the Appendix K hardware assumptions.

Since the focus of LOFT experiments is on thermal-hydraulic behavior, the first core does not contain pressurized fuel lest it balloon during the high-temperature transients and confound the results. However, an experiment in this series is planned in which some of the fuel rods will be pressurized. This will be run at the same power as the L2-3 and L2-5 experiments (approximately 12 kW/ft) and will serve to determine whether or not future experiments should include pressurized fuel. The schedule for this series is shown in Table 2-2.

#### 2.2.1.3.2 Subsequent Nuclear Experiments

Many other interesting areas of safety research are under consideration for LOFT; a tentative schedule is included in Table 2-2. Experiments with smaller break sizes, other break locations, alternative ECC injection points, and advanced ECC systems are planned to follow the first series. Since the first series will not be completed until 1980, however, any decision is subject to the exigencies of current reactor safety problems and new questions evolving in the licensing area.

#### 2.2.1.3.3 LOFT Instrumentation Development

The LOFT instrumentation development program is part of the total instrumentation effort discussed in Section 2.3.3.3.

### 2.2.2 SEMISCALE

The Semiscale program consists of a continuing series of thermal-hydraulic experiments having as their primary purpose the generation of experimental data that can be applied to the development and assessment of analytical models describing LOCA phenomena in PWR power plants. Emphasis is placed on acquiring system effects data that characterize the most significant thermal-hydraulic phenomena likely to occur in the primary coolant system of a nuclear plant during the depressurization (blowdown) and emergency core cooling phases of a LOCA. The experiments are performed with test systems that simulate the principal physical features of a nuclear plant but are much smaller in volume. Nuclear heating is simulated in the experiments by a "core" comprised of an array of electrically heated rods, each of which has dimensional and heat-flux characteristics similar to those of nuclear fuel rods.

The program covers many facets of LOCA experimental investigation. The Semiscale integral test systems are highly flexible in terms of the geometrical arrangement of primary coolant system piping and are extensively instrumented for the acquisition of thermal-hydraulic data during blowdown and emergency core cooling. The integral tests performed with the systems include introduction of variables to evaluate the relative effects of differing pipe-rupture locations, rupture sizes, flow resistances in selected regions, alternative locations for emergency coolant injection, etc. In addition to integral system effects tests, separate-effects tests are performed to acquire data on two-phase-flow phenomena in specific regions of the primary coolant system.

Semiscale is a nonnuclear one-dimensional representation of a PWR. The 2.0-MW core is 12 feet long and approximately 2.8 inches square; it contains 23 electrically heated pins in a 5x5 array. The primary coolant system is designed with the same system elevations and the same ratio of volume to core power as those of LOFT. System subvolumes (e.g., inlet plenum, core region) are also designed with relative volumes similar to those of LOFT. The unbroken coolant loops are simulated by a single unbroken circulating loop in the Semiscale primary system; the postulated broken loop is simulated by a blowdown loop either with or without active components.

#### 2.2.2.1 Objectives

The primary objectives listed in Section 2.2 will be addressed in the Semiscale program, which serves as the pioneer nonnuclear integrated facility in support of LOFT and other reactor safety program needs. Specific additional objectives to be attained by means of the Semiscale facility's special capabilities include the following:

- To investigate upper head injection phenomena and evaluate analytical capabilities.
- To develop models of component performance under transient conditions that are not otherwise investigated in ongoing separate-effects tests.
- To identify unexpected LOCA phenomena that would not be predicted from separate-effects testing.
- To confirm the adequacy of the computer codes used to predict interactive effects among the components.
- To investigate alternative locations for emergency core coolant injection.
- To ensure the selection of optimum test parameters for LOFT:
  - To assess the reliability of LOFT instrumentation.
  - To evaluate LOFT test results.
  - To address LOFT design compromises.

#### 2.2.2.2 Present Status

Sixty tests have been conducted in Semiscale Mod-1 since it started operation in August 1974. The first test series<sup>9-18</sup> duplicated the current LOFT isothermal test series. These tests were performed to partially confirm scaling effects on LOCA phenomena. The conclusions drawn from comparison of LOFT and Semiscale isothermal results are presented in Section 2.2.1.2.2.

A separate-effects test program was conducted in Semiscale to investigate the blowdown<sup>19-32</sup> and reflood<sup>33-37</sup> characteristics of a 5.5-foot-long core. A baseline integral test series<sup>38-45</sup> was performed to identify any unexpected LOCA phenomena that would not be predicted from separate-effects testing and assess computer code adequacy for predicting interactive effects among components. The Mod-1 configuration also investigated the effects of steam-generator-tube ruptures during a LOCE. In addition, a series of tests was completed to establish the benefit, in Semiscale, of alternative locations for the injection of emergency core coolant.<sup>46-48</sup> The tests have led to a number of conclusions, discussed below.

A major change in program orientation occurred near the end of fiscal year 1977, when conversion of the Semiscale test system from the 1.5-loop Mod-1 configuration to the Mod-3 two-loop configuration was initiated. This conversion, completed in May 1978, represents a shift from experiments related primarily to the LOFT program to those that will provide better representation of the thermal-hydraulic phenomena characterizing commercial PWRs with upper head injection (UHI) of emergency core coolant. New hardware for the Mod-3 system has been so designed as to permit reverting to the 1.5-loop configuration in a relatively short time if additional tests in support of LOFT are needed.

One of the important objectives of converting from a 1.5- to a two-loop configuration is the production of experimental data that will reveal the relative effects on system performance of operating with active versus passive components in the broken (blowdown) loop. Acquisition of knowledge in this area is particularly significant because of the use of inactive broken-loop components (pump and steam-generator resistance simulators) in the LOFT system and in the Semiscale Mod-1 system. The Semiscale Mod-3 system has an operating pump and an active steam generator in the broken loop, both of which are scaled to simulate as closely as possible the performance characteristics of their counterparts in PWRs. Similarly, the Mod-3 system has 12-foot-long fuel-rod simulators so as to acquire core heat-transfer data that will be more

representative of PWRs. Another distinguishing feature of Mod-3 is a vessel downcomer in the form of a cylindrical pipe that is external to the vessel proper. This external downcomer configuration has been adopted to alleviate hot-wall effects, which have caused atypical time delays for the flow of emergency coolant through the concentric downcomer in the Mod-1 vessel. The core orientation in Mod-3 also is different in that the heater rods penetrate the lower vessel head rather than the upper head as they did in Mod-1. The upper plenum-upper head geometry of Mod-3 is designed specifically for upper head injection of emergency core coolant. On completion of UHI experiments a series of experiments will be conducted utilizing a "two-pipe" downcomer configuration to evaluate the effects of the additional downcomer on the reflood characteristics of the core.

The single-pipe downcomer UHI test program will begin in January 1979 and will be completed in May 1979. The two-pipe integral test series will follow these tests and should be completed by November 1979. The basic objective is to determine the effect of upper head emergency coolant injection on core cooling. An additional objective is to determine the sensitivity of system and core response to variations in such UHI parameters as ECC injection rate, upper head fluid temperatures before pipe rupture, and upper head fluid mixing.

#### 2.2.2.2.1 Blowdown Heat Transfer

Variations in the time to critical heat flux (CHF) of 0.3 to 4.0 seconds have been observed in the 5.5-foot heated-length core. These CHF delay times are roughly comparable to those observed in the 12-foot heated-length core at the Oak Ridge National Laboratory (ORNL). Transient CHF in both systems is strongly influenced by local steam qualities and heat fluxes, which in turn are influenced to some extent by the number and location of inactive rods. It was also demonstrated in the Mod-1 tests that precision in calculating core inlet flows (hence local steam qualities) is a direct function of the precision in calculating the critical flow. In addition, it was established that post-CHF heat transfer in Semiscale for a 200% cold-leg break occurs mainly by free convection and by radiation.

#### 2.2.2.2.2 Reflood Heat Transfer

The purpose of these tests was to provide information on the reflood heat-transfer characteristics of the Mod-1 system and to characterize system performance with respect to other configurations of different sizes and performance. The data from the forced flooding tests were compared to the FLECHT data, and the gravity flooding data were compared to the FLECHT-SET data. For the former comparison, the general trend of the cladding temperature rise and quench times in the 5.5-foot heated-length core in the Mod-1 system were in agreement with those observed in the FLECHT facility, which had a 12-foot heated-length core. Comparisons between Mod-1 and FLECHT-SET data showed that the core thermal-hydraulics were somewhat different for the two systems. In addition, it was demonstrated that none of the existing heat-transfer-coefficient correlations produced good overall predictions of the heat-transfer coefficients. Consequently, an improved heat-transfer correlation and heat-transfer logic were developed. The results from this test series produced comparisons with the FLECHT and FLECHT-SET data, which will make the data more useful for model development purposes and will aid the LOFT program in analyzing LOFT core heat-transfer behavior.

#### 2.2.2.2.3 ECC Bypass

It was shown in the Semiscale Mod-1 tests that the degree of steam superheat influences the downcomer countercurrent flow response. This is a synergistic effect that was identified for exploration in separate-effects tests.

It was also found that the accumulator nitrogen influences the reflood behavior in the Semiscale Mod-1 core. It reduces cold-leg steam condensation and thus results in a higher reflood pressure transient. The introduction of accumulator nitrogen pressurizes the downcomer and causes a core-inlet flow surge. The resulting high rates of steam generation, coupled with a reduction in cold-leg condensation, sustain a high system pressure, which promotes improved core heat transfer until the system pressure approaches containment pressure.

#### 2.2.2.2.4 Alternative Locations for ECC Injection

Lower plenum injection provides a substantially earlier core quench than does cold-leg injection (75 seconds for lower plenum versus 200 seconds for cold-leg injection) and reduces the amount of ECC bypass. A vent line between the upper plenum and the upper annulus also gives an earlier core quench than does cold-leg injection (100 versus 200 seconds). The vent line reduces pressure in the upper plenum during reflood and causes an increase in the core reflood rate.

The Mod-1 tests also showed that combined upper plenum and cold-leg injection gives earlier core quench times than does cold-leg injection. However, the core quench mechanism is "from the top down" as opposed to "from the bottom up," which is found in a cold-leg injection test. These results qualitatively resemble those observed in ROSA II.<sup>49,50</sup>

Pump-suction injection is strongly influenced by the performance of the pump during the transient. Depending on the pump speed, pump-suction injection can be more effective than cold-leg injection or vent-line injection (under typical coastdown conditions) and can be worse than strictly cold-leg injection (20% below rated coastdown conditions).

#### 2.2.2.2.5 Steam-Generator-Tube Ruptures

These tests determined the sensitivity of core peak cladding temperature to the magnitude and timing of the flow rate from the secondary side of the steam generator to the primary system. The test results showed that the potential for increased cladding temperature during a LOCA is greatest during the reflood period for more than 12 but less than 50 steam-generator-tube ruptures. The most significant result of this test series was that the computational methods currently used to analyze this transient appear to be extremely conservative. Subsequent tests in the Mod-3 configuration are planned to further substantiate this conclusion.

#### 2.2.2.2.6 Foreign Facilities

There are two foreign facilities comparable to Semiscale that relate to the programs in the Systems Engineering Branch. These are the ROSA facility at the Japan Atomic Energy Research Institute and LOBI at Ispra, Italy. ROSA and LOBI are roughly equivalent to Semiscale but differ in power and volume. LOBI is three times as large as Semiscale and has an active blowdown loop to a scaled containment. The test program shown in Table 2-3 was commenced in 1978. The results should provide important data relative to the scaling issue being addressed by Semiscale and LOFT.

TABLE 2-3  
LOBI EXPERIMENTAL PROGRAM

Number of Experiments	Component or Effect To Be Investigated	Comments
3	Reference experiments	Three rupture positions
3	Evaporator secondary side	Minimum heat sink
9	Pump of intact loop	Three different pump operations during blowdown
3	Pump of broken loop	Pump rotator free during blowdown
1	Evaporator secondary pressure	Heat sink
7	Location of ECC injection	Cold, hot, and combined legs
3	Behavior of downcomer flow	Downcomer heights
11	Rupture size	Controlled pump rundown; varied injection position
6	Small leaks	
4	Rupture of feeding lines	Variation of injection pressure and mass flow rate

### 2.2.2.3 Research Program

The Mod-1 test program has established the characteristics of the Mod-1 core and system response, and will provide a direct baseline for the LOFT heated-core tests. The counterpart tests to the initial LOFT nuclear series were completed in the Spring of 1977. The results have supplemented those of the LOFT nonnuclear experiments. After completion of the LOFT nuclear tests, a final assessment can be made as to:

- The effects of scale on integral behavior under heated core conditions.
- The existence of any size-dependent synergistic effects.
- The existence of any size-dependent phenomena related to the effectiveness of alternative ECC injection concepts tested in Semiscale Mod-1.

As already mentioned, Semiscale has been converted from the Mod-1 to the Mod-3 configuration. The Mod-3 system has a 12-foot core with an active "intact" loop and an active "broken" loop. The Mod-3 upper plenum has been designed to investigate the system effects of the upper head injection (UHI) concept employed in some Westinghouse Electric Corporation PWRs. This series of experiments is considered a high-priority effort by the NRC licensing staff. A one-year test program is planned in this area. The first series of UHI tests will be conducted with a single pipe external to the vessel to model the annular region. The second series of UHI tests will incorporate two external pipes for the downcomer to simulate expected asymmetric flow in the annulus.

On completion of the two-pipe downcomer experiments, the Mod-3 vessel configuration will be altered so as to provide a conventional (non-UHI) upper plenum/upper head. This latter configuration is designated Mod-2. It is the same as Mod-3 except for the upper plenum/upper head region in the vessel and the steam generator in the intact loop. A type II steam generator and associated support structures will replace the type I steam generator in the intact loop, the purpose being to provide improved simulation of PWR performance characteristics in this portion of the Semiscale test system.

The objectives of the Semiscale Mod-2 program are:

- To produce experimental data that will be more representative of PWR performance and will permit comparison of system performance with a conventional upper plenum to that obtained with the Mod-3 UHI configuration.
- To determine the sensitivity of system response and core thermal-hydraulic behavior to pipe breaks of different sizes and types. A small-break sensitivity study may be a part of this study.
- To investigate the influence of steam-generator-tube leaks on reflood system performance. This test series will be similar to the Mod-1 tests but will evaluate the system response for the 12-foot core and active components in the broken loop.
- To investigate the effectiveness of alternative ECC concepts. This test series will aid in determining whether improvements in cooling effectiveness could be made relative to that provided by cold-leg ECC injection. These tests will expand on the previous Mod-1 alternative ECC investigation and a comparison of results will be made.

Data to be obtained from the Mod-3 and Mod-2 tests will be compared with LOFT and Mod-1 data in order to evaluate the effects of physical scale on component and system performance as well as the effects of active versus passive components in the broken loop. The data also will be compared with the separate-effects data provided by the FLECHT and blowdown heat-transfer (BDHT) programs at the Westinghouse Electric Corporation and the Oak Ridge National Laboratory (ORNL), respectively. The Mod-2 experiments are scheduled to begin in early 1980 and will require about 18 months for completion.

The other configurations presently planned in order of priority are:

- Mod-4: LOFT counterpart 1.5-loop system utilizing a 5.5-foot core in a vessel similar to the Mod-2 vessel. (A variation of this configuration will consist of employing active components in the broken loop for performance comparison with the pump and steam-generator resistance simulators that characterize the 1.5-loop concept.)

Mod-5: Same as Mod-2 except for the addition of a second cold leg in the broken loop to simulate the 2 x 4 loop configuration that is used in some PWRs.

Mod-6: Similar to Mod-2 but utilizing a small pitch core (PWR 17 x 17 fuel-rod lattice) and having correspondingly small-diameter (0.375-inch) fuel-rod simulators (heater rods).

#### 2.2.2.4 Milestones

- Mod-3 became operational in January 1978.
- Mod-3 UHI tests to be completed in January 1980.
- Mod-2 non-UHI baseline tests to be completed in April 1980.
- Mod-2 alternative ECCS tests to be completed in December 1980.
- Mod-2 steam-generator-tube rupture tests to be completed in February 1981.
- Mod-2 break sensitivity tests (large and small breaks) to be completed in November 1981.

#### 2.2.3 ADVANCED INSTRUMENTATION DEVELOPMENT FOR TWO- AND THREE-DIMENSIONAL TESTS

##### 2.2.3.1 Objectives

The agreement among the United States, the Federal Republic of Germany, and Japan\* on two- and three-dimensional tests calls for the United States to contribute the TRAC code and advanced instrumentation. The advanced instrumentation is divided into two categories: proved instruments (e.g., mass flow meters, densitometers and liquid-level detectors) and in-core instruments that are in the development stage (e.g., void fraction indicators and thermocouples).

The objectives of this program are to accelerate the development of reliable in-core advanced instrumentation for timely delivery to the two- and three-dimensional test facilities. The objectives can be divided into three levels:

- To develop in-core instruments that will operate in the adverse environments expected of the two- and three-dimensional tests.
- To fabricate such instruments, together with the necessary electronics, for timely delivery to test facilities, with the first shipment to the PKL-I facility in Germany in August 1978.
- To continuously improve the accuracy of measurements through better electronics, calibrations, and modeling.

##### 2.2.3.2 Present Status

In-core instruments are being developed to measure liquid upflow in the core, deentrainment in the upper plenum, and liquid fallback into the core. The development of instruments for measuring countercurrent flow across the upper core support plate is a new task that was added in July 1978. Continuous analytical efforts in support of this work are being made to improve modeling techniques. At present, in-core instruments for measuring film thickness, film velocity, void fraction, and two-phase-flow velocity have been developed and tested, and are in the process of fabrication. The major difficulties are the stringent requirements imposed by the expected in-core environment:

- High temperature (900°C).
- High thermal shock (300°C/sec).
- Highly corrosive environment of the steam-water mixture.
- High electromagnetic noises resulting from the presence of power supplies for heating rods.

\*See Chapter 6 for more details.

Because of this adverse environment, it was necessary to develop new insulating material and new brazing compounds, and to use highly sophisticated electronics. It was also necessary to develop fabrication techniques for inserting eight triaxial cables into 12-foot-long, 0.5-inch-diameter tubes. A steam-water test loop was designed, constructed, and brought into operation. All these tasks were accomplished within the span of 1 year.

### 2.2.3.3 Research Program

The advanced-instrumentation research program for the last half of fiscal year 1978 covers five areas: development, fabrication, qualification and calibration, documentation and onsite installation assistance, and sensor data analysis. The latter involves procurement of a computer system. The instruments will be loaned to Japan and Germany.

Development-related tasks are to:

- Identify probe welding and brazing methods.
- Complete air-water testing of PKL in-core probe configurations.
- Confirm in-core sensor operability in a steam-water environment.
- Complete air-water scoping studies of PKL upper plenum probes.
- Continue void-fraction studies.
- Continue materials development.
- Initiate fundamental studies of probe behavior in two-phase flow (experimental and computer modeling).
- Confirm the electrolytic-potential method of measuring film velocity.

In the fabrication area, three tasks are to be completed in the last half of fiscal year 1978: (a) the design and fabrication of PKL in-core sensors; (b) the fabrication of prototype electronics for steam-water testing; and (c) the design of PKL upper plenum sensors.

Qualification and calibration tasks are to evaluate the testing of (a) actual PKL in-core sensors in a steam-water environment and (b) in-core sensor survival.

The long-range development plan for advanced instrumentation can be summarized as follows:

- Optimization of sensors and electronics
  - Sensor geometry (analytical and experimental)
  - Electromagnetic interference rejection
  - Materials and fabrication methods
  - Reduction of triaxial cable diameter
- Fundamental air-water and steam-water phenomenological studies of impedance and film probes
  - Flow regime effects
  - Relation of noise analysis velocity to liquid velocity
  - Film velocity profile
  - Effect of temperature and other water properties on impedance measurements
  - Probe limitations and error analysis
  - Electrolytic-potential probe
- Investigation of alternative measurement techniques
  - Fiber optics
  - Hot-film anemometry
  - Ultrasonic techniques
- Application of instrumentation technology to other LWR safety research facilities.

#### 2.2.3.4 Milestones

##### Delivery to PKL

- In-core sensors to be delivered in October 1978.
- Upper plenum sensors to be delivered in January 1979.
- Electronics to be delivered in June 1979.

##### Delivery to Slab-Core Test Facility

- Sensors and electronics for core I to be delivered in November 1979 and February 1980, respectively.
- Instruments for core II to be delivered in 1981.
- Electronics for core II to be delivered in 1982.

##### Delivery to Three-Dimensional Upper Plenum Test Facility

- Sensors to be delivered in December 1980.
- Electronics to be delivered in June 1981.

##### Delivery to Cylindrical Core II

- Instruments to be delivered in December 1980.

### 2.3 SEPARATE-EFFECTS STUDIES AND TESTS

Separate-effects studies and tests provide a means of isolating and analyzing the major variables predicted to influence a LOCA. This section discusses the objectives, present status, and research program for the principal categories of studies.

#### 2.3.1 HEAT TRANSFER DURING BLOWDOWN AND REFLOOD

##### 2.3.1.1 Objectives

The objectives of tests investigating heat transfer during the blowdown and reflood phases of a LOCA are the following:

- To determine the estimated time to critical heat flux (CHF) and the heat-transfer rates during the pre- and post-CHF phases of blowdown as influenced by variations in power, system pressure, coolant flow, and break location (PWR geometry).
- To investigate the estimated time to CHF, the hydrodynamics of lower plenum swell, and the post-CHF heat transfer in sufficient detail to evaluate thermal phenomena before and after emergency core cooling (BWR geometry).
- To provide data for use in evaluating analytical predictions of core flow rate as a function of system volume, coolant conditions, and break location and size.
- To determine bottom-flooding flow rates and core temperatures during one-dimensional rod bundle reflood experiments.
- To analyze the relationship between one-dimensional reflood test results and three-dimensional core behavior in PWRs.
- To determine the effect of various system components on the core reflood rate.
- To initiate fundamental investigations that will improve the prediction of transient heat-transfer processes during reflood.
- To develop improved heat-transfer correlations that would be applicable during PWR reflood.
- To provide data on carryover flow rates for use in evaluating analytical predictions.

### 2.3.1.2 Present Status

#### 2.3.1.2.1 PWR Blowdown Heat Transfer

Research programs to study blowdown heat transfer under simulated PWR conditions are being sponsored by the NRC, the Electric Power Research Institute (EPRI), and foreign governments. Although facility design, test procedures, and methods for the electrical simulation of nuclear heat are not the same, all of these programs measure the rate of heat transfer from the rod simulators as the coolant experiences changes in pressure, void distribution, and flow velocity.

The principal NRC effort is being conducted at the Oak Ridge National Laboratory in the Thermal Hydraulic Test Facility (THTF).<sup>51</sup> The THTF is a large nonnuclear pressurized-water loop incorporating a full-length bundle of 49 electrical rods that can be heated and cooled under conditions calculated to correspond to those in a nuclear power reactor. The coolant loop also contains disks that are ruptured in a controlled sequence to simulate various transient flows to and from the heated bundle. The transient flows are selected to approximate flows calculated to occur during a postulated PWR LOCA.

Results obtained at the THTF are being compared with LOCA blowdown assumptions defined in the ECCS Acceptance Criteria<sup>52</sup> and with pretest predictions made with the current best estimate versions of the RELAP computer code.

The results obtained to date are being analyzed. Preliminary evaluations tend to support the conservative assumption of a very short (<1 second) time delay to CHF, as used in the ECCS Acceptance Criteria.

Some of the preliminary conclusions obtained from the ORNL program are listed below.

- With approximately the same test-bundle flow history during the first second after blowdown initiation as is calculated for most PWR cores, the time to CHF for peak power locations is generally between 0.7 and 1.0 second.<sup>53</sup>
- Unpowered rods appear to delay the time to CHF for adjacent powered rods, particularly at locations above the test-bundle centerline.<sup>54</sup>
- Reduced bundle power and peak power density generally increase the time to CHF.<sup>55</sup>
- The RELAP-4 model of THTF closely predicts the peak cladding temperature.<sup>56</sup>

#### 2.3.1.2.2 BWR Blowdown and Emergency Cooling Studies

Studies of heat transfer during blowdown have been completed under the joint sponsorship of NRC, EPRI, and the General Electric Company (GE). The tests were conducted in the Two-Loop Test Apparatus (TLTA) at the GE facility in San Jose, California. Full-length 7 x 7 electrical simulators were scaled to BWR fuel rods. The test program has been extended, also under joint sponsorship, to include blowdown studies and investigations of ECC injection with an 8 x 8 bundle. Initial program planning was completed and testing under the expanded program initiated in late 1976.

The BWR Blowdown Heat Transfer Program Final Report<sup>57</sup> includes the following conclusions, obtained in tests with the 7 x 7 bundle:

1. The two-loop test apparatus (TLTA) provided a very good global simulation of BWR system LOCA conditions for evaluation of BDHT phenomena.
2. The current BWR LOCA evaluation method, when applied to the test apparatus, shows a substantial margin in the prediction of system blowdown performance in the prediction of peak cladding temperature.
3. The system response was observed to be insensitive to large variations in the bundle power (3 to 6.5 MW). This observation is supportive of the approach used in current BWR LOCA evaluation methods of obtaining the nominal core inlet conditions for the bundle heat-up calculations from thermal-hydraulic blowdown calculations using the core average power.
4. Boiling transition (BT) generally occurs after the lower plenum flashing surge due to rod uncovering as the two-phase mixture level is depleted. However, in the peak power bundle tests BT occurred due to exceeding the critical power during the non-typical TLTA core-flow coastdown while the mixture level remained above the bundle.

5. Bundle heat transfer can be generally characterized as nucleate boiling beneath the two-phase mixture level with steam cooling above the mixture level. In the case of the peak power bundle the post-BT heat-transfer mode can be characterized as transition boiling to film boiling. After uncovering the top of the bundle the heat-transfer mode was steam cooling as above.
6. The maximum measured cladding temperature was less than 1400°F (1033 K) for the peak power bundle test, despite the forced BT at about 1 to 2 seconds by the atypically rapid initial core flow coastdown of the TLTA. Maximum temperatures for the average power tests were typically less than 900°F (755 K).
7. Heat transfer to the single-phase steam, which was generated beneath the mixture level both by flashing and by heat transfer from the bundle, is the predominant cooling mechanism in the post-lower plenum flashing period.
8. The hydrodynamics of LP flashing is governed by the system depressurization rate at the time the saturation temperature in the lower plenum (LP) is reached. An overprediction of the depressurization rate, which results from an overprediction of break flow and break quality, can lead to an overestimation of the flow surge during LP flashing. The net effect of this overestimation on the calculated peak cladding temperature is, however, negligible.
9. Some rewetting has been observed during lower plenum flashing. The mechanism of rewetting appears to be governed by the cladding temperature and surface heat flux at the location rewet was observed.
10. Fluid inventory within the bundle in the post-LP flashing period is maintained there due to countercurrent flow-limiting phenomena (CCFL) at the core inlet. Steam updraft from the lower plenum prevents the fluid from draining completely from the core region.
11. In summary, a number of inherent cooling mechanisms were observed for which no credit is taken in the design of LOCA safety systems:
  - a. Bundle cooling due to residual fluid in the bundle (conclusions 5 and 10).
  - b. Steam updraft cooling above the mixture level in the bundle (conclusions 5 and 7).
  - c. Cladding rewetting during lower plenum flashing (conclusion 9).
  - d. Cladding rewetting in the post-lower plenum flashing period due to fluid fallback from the upper plenum.
12. A study of independent parameter variations showed that break area and initial fluid mass had a significant effect on the system thermal-hydraulic blowdown response. These parameters affect the timing of key events (e.g., uncovering of the blowdown line) and hence the bundle heatup response.
13. Those parameters which had much smaller or no discernible effects on global system response over the range investigated were alternate power decay, bundle bypass orificing, initial fluid subcooling, and alternate lower plenum geometry.
14. Steam-line break tests resulted in substantially different system blowdown response as expected. There was also very little bundle heatup in magnitude, extent of the bundle, and in duration.
15. Specific phenomenologically based model refinements have been suggested for break flow, void distribution, and bundle heat transfer.

#### 2.3.1.2.3 Full-Length Emergency Cooling Heat-Transfer Tests (FLECHT) and System Effects and Separate-Effects Test (SEASET)

Interactions among the physical phenomena that determine the rate of heat transfer from heated reactor fuel pins to ECC water during reflod are qualitatively understood. Information from FLECHT tests is available to determine heat-transfer coefficients for various flooding rates, rod temperatures, and system pressures. The FLECHT data (but not correlations) are currently

used in licensing evaluations for all PWR reactors. A limited amount of heat-transfer information is available from several constant-rate-flooding FLECHT tests in which a horizontal plate was used to study "blockage" effects.

Additional FLECHT forced-feed tests, including flooding rates at and below 1 in./sec, were completed in 1976 with a bundle whose axial heat profile was cosine-shaped, in an improved thin circular housing. Similar tests were completed in early 1977 on a bundle whose heat profile was axially skewed and peaked near the top of the bundle. The principal conclusions obtained from the low-flooding-rate cosine-profile tests were as follows:<sup>58</sup>

1. Flooding rates near or below 1.0 in./sec did not affect heat transfer in any unexpected way. No particular significance can be associated with a flooding rate of 1.0 in./sec or below.
2. Trends of heat transfer, temperature rise, and quench times as a function of the various test parameters were essentially the same as in previous forced-flooding tests. One possible exception was the trend of temperature rise with pressure, which was somewhat weaker in the present test series than in previous test series.
3. Droplets entrained near the quench front were found to play a significant role as a heat sink both above and below 1.0 in./sec.
4. An improved best estimate FLECHT heat-transfer correlation has been developed.
5. A semiempirical model for calculating bundle mass effluent fraction has been developed.
6. A method for calculating local qualities and mass flow within the bundle has been developed. The results of this analysis have helped to clarify reflood heat-transfer mechanisms. It is believed the resulting analysis performed on the cosine low-flooding-rate data will be very useful for reflood model development and assessment. The calculation of local qualities, void fractions, flows and the measurement of the vapor temperature will allow effective testing of nonequilibrium post-CHF reflood heat-transfer and entrainment models.

The objectives of the axially skewed heat-profile test series were achieved.<sup>59</sup> The data obtained will be useful for the development and assessment of reflood models. In particular:

- The reflood data base has been extended to include axial power shape effects.
- Flooding rate effects on rod surface for the skewed test series were approximately the same as those observed in the cosine-profile test series.
- The housing effects encountered in the cosine-profile test series were virtually eliminated by using a low-mass housing.
- Additional differential pressure (d/p) cells installed at 12-inch intervals along the length of the bundle have provided more detailed data on mass storage, froth level, quench front, and void fraction--information needed for quench and entrainment model development. Furthermore, the bundle d/p cells in conjunction with the loop instrumentation yielded more accurate overall mass balances in the system.
- Positive and negative bundle flows during gravity-reflood scoping tests were accurately monitored by a bidirectional turboprobe installed in the crossover pipe.

The SEASET program proposed by Westinghouse for joint sponsorship by NRC, EPRI, and Westinghouse was contracted in June 1977. The principal elements of the proposal include added capability to perform separate-effects tests on the effects of blockage, spacer grids, upper plenum geometry, and steam generator. Systems-effects tests will utilize a more sophisticated version of the original FLECHT-SET apparatus.

Efforts have been initiated to provide a closer coupling between reflood experiments and analyses being performed in the United States, Germany, and Japan.

#### 2.3.1.2.4 Heat-Transfer Correlations and Improvement

Ideally, the LOCA-ECC heat-transfer correlations should be formulated by using data for transient heat transfer from bundle geometries. Since local fluid parameter data are not available, the

general approach is to (a) formulate correlations from steady-state data, obtained with heated tubes; (b) check the correlations against the limited amount of transient data that is available for bundles; (c) verify correlations over the entire data base to determine parameter trends and prediction accuracy; and (d) use the correlations in the best estimate code to check against cladding-temperature profiles for transient tests.

Blowdown Heat Transfer. Transient CHF data for bundles are being obtained from Semiscale and THTF, using local hydraulic conditions calculated from best estimate computer codes. In addition, data points available from earlier GE and Westinghouse tests are being used in the correlations whenever possible.

Post-CHF data for bundles are less available than the transient CHF data. Data points are available from 99 steady-state tests. Transient-state data with local hydraulic conditions are now being reduced from Semiscale and THTF tests.

Transient data from tubes are limited, and the only data available in sufficient quantity for use are those from Westinghouse. A large body of film-boiling data from tubes is available. Slow-transient transition-boiling data from tubes are available in limited amounts.

Many steady-state CHF correlations are available. Transient CHF correlations are beginning to emerge, including Hsu's modification<sup>60</sup> of the W-3 equation to account for the void-fraction dryout effect, Griffith's modification<sup>61</sup> of Zuber's pool-boiling CHF to account for the void-fraction dryout effect, and Henry's<sup>62</sup> spontaneous-nucleation hypothesis.

Transition-boiling correlations include the equations of Tong and Young<sup>63</sup> as well as Condie and Bengston.<sup>64</sup> Film-boiling equations include Groeneveld's<sup>65</sup> equations 5.7 and 5.9, Groeneveld's new nonequilibrium correlations,<sup>66</sup> Dougall-Rohsenow's<sup>67</sup> equation for dispersed flow, and Chen's equation<sup>68</sup> for dispersed flow under nonequilibrium conditions.

Reflood. Certain hydraulic effects during reflood are cumulative. Upstream effects and previous flow history influence the rate of cooling. Thus, in determining the whole temperature profile during reflood the entire process must be considered.

Available FLECHT correlations<sup>69</sup> are not phenomenological in nature and are therefore difficult to extrapolate or to apply to other geometries. Under NRC support, Griffith and Kirchner proposed a REFLUX package<sup>70</sup> that is based on a collection of heat-transfer correlations, each applied to a specific heat-transfer regime. The Semiscale group has developed an analytical approach using heat-transfer correlations for each specific regime, but their correlations are more empirical in nature<sup>71</sup> than those of the REFLUX package. The REFLUX package has been incorporated into RELAP-4. Very recently the effects of grid spacers on heat transfer have been studied at MIT.

### 2.3.1.3 Research Program

To satisfy the objectives discussed in Section 2.3.1.1, major programs at ORNL and GE are investigating heat transfer during blowdown under a spectrum of PWR and BWR conditions. The requirements established for each of the major programs are similar in a number of respects:

- The test facilities provide coolant pressure and temperature conditions representative of BWRs and PWRs.
- The length of heater rods, the rod diameters, and coolant channel spacings used in the test facilities are the same as those in BWRs and approximately the same as those in PWRs.
- Sufficient electrical power (5 to 7 MWe) is available at the test facilities to permit the initiation of blowdown tests with 49- or 64-rod bundles operating at power levels corresponding to those in operating reactors.
- The principal interest is to study the thermal transient behavior of the rod bundle during blowdown.
- Each program studies the initial phase of the postulated LOCA, prior to the initiation of ECC flow; the BWR program has been extended to include ECC injection and behavior.

The PWR blowdown heat-transfer (BDHT) program is being conducted at ORNL, and the BWR BDHT program was completed at the GE facilities in San Jose, California. The principal differences between the two programs are as follows:

- PWR systems differ among manufacturers, and no attempt has been made to provide a scaled PWR representation with the testing system used at ORNL. However, this facility is capable of providing a range of test conditions to scope operating conditions.
- For BWR tests, the test facility is scaled to a BWR.

Each of the major BDHT programs contains a companion analytical effort using the RELAP-4 code. By comparing RELAP predictions with results from selected tests and analyzing the reasons for the differences observed, a basis will be established for further refinement of analytical representations of the experiments. In addition to best estimate calculations based on the most realistic assumptions available, licensing model calculations based on assumptions contained in licensing criteria are available. A comparison of results from the two types of calculations provides a measure of the degree of conservatism in the licensing models.

A single-rod pressurized loop is being used at ORNL to provide supporting services to the larger loop program (such as evaluations of electrically heated simulated fuel rods) and as a simpler geometry in which blowdowns can be conducted to test instrumentation techniques and analysis methods.

#### 2.3.1.3.1 PWR Blowdown Heat-Transfer Program

The PWR BDHT experimental program, which uses a 49-rod bundle, is designed to study the LOCA relationships among the principal reactor variables that can alter the rate of blowdown. These include the presence of flow reversal and re-reversal, the time interval before CHF, the rate at which dryout progresses radially and axially along the rods, and similar time-related functions that are important to accident analysis. The ORNL 49-rod experimental program is being conducted within an experimental matrix that explores each of the variables separately over a range of conditions of interest to the NRC in analyzing the blowdown response of PWRs.

The types of tests to be conducted include:

- Steady-state tests to examine systematically the onset of critical conditions as loop and power parameters are varied.
- Blowdown with flow reversal and immediate power decay, using a range of initial power levels.
- Blowdown with flow reversal and delayed power decay, to account for the time delay between inlet pipe rupture and the initiation of a PWR power transient.
- Blowdown from off-design (lower than rated) pressures, to examine the effects of subcooling.
- Blowdown under loop conditions that will permit the study of flow reversal and re-reversal.
- Blowdown using test bundles with cosine and uniform power profiles.

Testing at full bundle power began in mid-1976, and data analysis is under way. One of the early tests of interest was test 105. This full-power (5.973-MW) test was initiated from a test section inlet temperature of 558.3 K (545°F), a test section inlet volumetric flow of 0.0269 m<sup>3</sup>/sec (427 gpm), and a test section outlet pressure of 15.534 MN/m<sup>2</sup> (2253 psig). System decompression was accomplished by introducing a 40% inlet/60% outlet break. The primary coolant pump was tripped coincident with break initiation, but the electric core was operated at full power for 2 seconds into the transient, and thereafter the power was made to decay with a time constant of 0.45 second. A 40% inlet/60% outlet break appeared to provide a reasonable approximation of the desired core flows.

Since the test section outlet subcooling was 13 K (24°F), the outlet break saturated almost instantaneously. The fluid initially between the heat exchangers and the pressurizer strongly affected the test section outlet flow, causing it to remain stagnant for 4.5 seconds into the transient.

Critical heat flux was observed 0.4 second after break initiation. The highest temperature measured with a thermocouple on the cladding was 1011 K (1360°F) at the axial center of the electric core.

Figure 2-2 shows the location of thermocouple planes in test 105. and Figures 2-3 through 2-6 show examples of the data obtained.

Figure 2-3 shows a typical result from a level D thermocouple. Figure 2-4 is typical for a level F, G, or H thermocouple reading, showing again that departure from nucleate boiling occurs in less than 1 second. However, rewetting does not occur as abruptly as it does at level B, probably because this is the highest powered region of the bundle.

Figure 2-5 is typical for a level J thermocouple reading. It shows a shorter temperature spike and a shorter period to rewet than do the thermocouple readings at lower levels.

Figure 2-6 is typical for level K, L, M, and N readings, which are at the lower powered regions. There is no marked temperature spike.

The results of test 105 will be used in planning future tests and in improving the RELAP-4 THTF models.

#### 2.3.1.3.2 BWR Emergency Cooling Studies

BWR core spray systems are designed to inject emergency cooling sprays into the plenum above the nuclear core during a LOCA. Flow of the upper plenum spray water through the core and into the lower plenum, and the subsequent reflood process, can be impeded by the upflow of steam formed in the core which would oppose the downflow of water. A counter-current flow limiting (CCFL) phenomenon occurs when no liquid is allowed to fall. General Electric has conducted its own tests to study CCFL behavior. The results of these tests indicate that the availability of ECC water to the lower plenum through the hotter central regions of the core may be delayed by steam updrafts and that multidimensional effects need to be examined. The current NRC/EPRI/GE program, which will be completed in 1981, does not extend beyond investigations of blowdown and ECC spray interactions. General Electric has submitted a proposal to EPRI and NRC for sharing the cost of an expanded program to study the effects of larger facility dimensions on CCFL and reflood behavior of BWR's. This program, if implemented, would include the following elements: A full scale sector facility to provide a more realistic simulation of the core spray, steam/water mixing in the upper plenum, and draining into the lower plenum. A scaled upper plenum sector experiment with sector angle and radius about one half the full scale experiment. A single bundle system with heated fuel simulators to provide information on how to condition the steam entering each of the 52 unheated fuel bundle simulators in the full scale facility.

#### 2.3.1.3.3 FLECHT and SEASET

The work scope planned jointly by NRC, EPRI, and Westinghouse includes the following tasks:

Unblocked Bundle, Forced Reflood and Gravity Reflood. This task would provide a data base for (a) the development or assessment of computational methods used by others to predict reflood thermal-hydraulic behavior, (b) a baseline for comparison to determine the effect of blockage in rod arrays, and (c) comparison with previous FLECHT 15 x 15 unblocked tests to evaluate bundle geometry.

Twenty-One-Rod Bundle, Flow Blockage and Heat Transfer. This task would obtain, analyze, and evaluate data on the effect of flow blockage on two-phase flow and local heat transfer. The results would be used to assist in planning the blockage and spacer grid tasks and to clarify heat-transfer mechanisms and two-phase-flow characteristics for blocked and unblocked geometries.

Large Blocked Bundle, Forced Reflood and Gravity Reflood. This task would provide a data base for comparison with data from unblocked and existing FLECHT 15 x 15 tests and for assessing the effects of blockage on heat transfer during reflood and entrainment.

Spacer Grid Effects on Heat Transfer During Reflood. This task would use the results of the 21-rod-bundle blockage tests, literature surveys, and discussions to examine spacer grid effects on heat transfer during reflood. Comparable tests would be run on three sets of spacer grids with flow-resistance values that bound those of the current PWR fuel vendors' spacer grids and a FLECHT spacer grid.

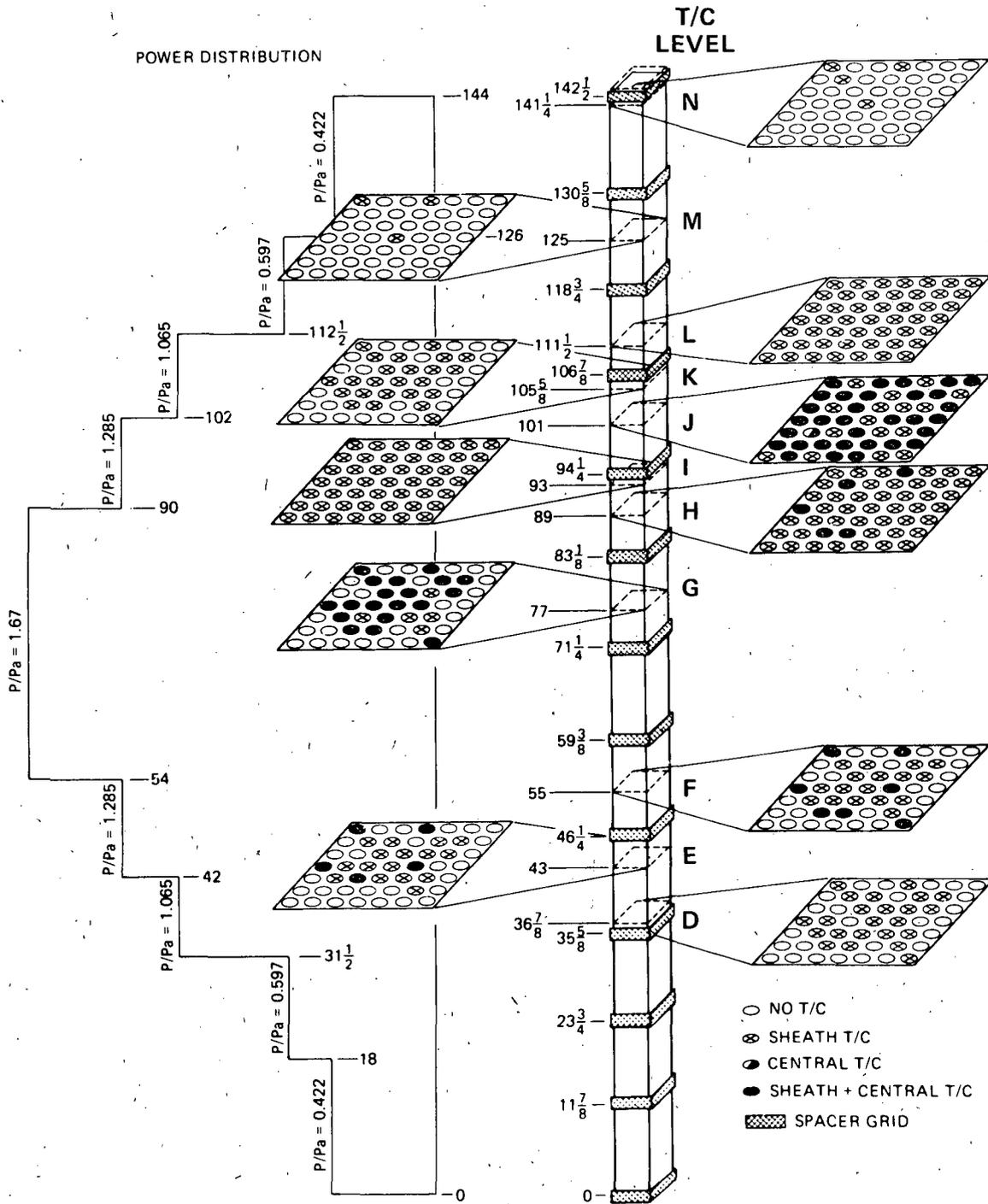


Figure 2-2. Location of Thermocouples in THTF Bundle No. 1

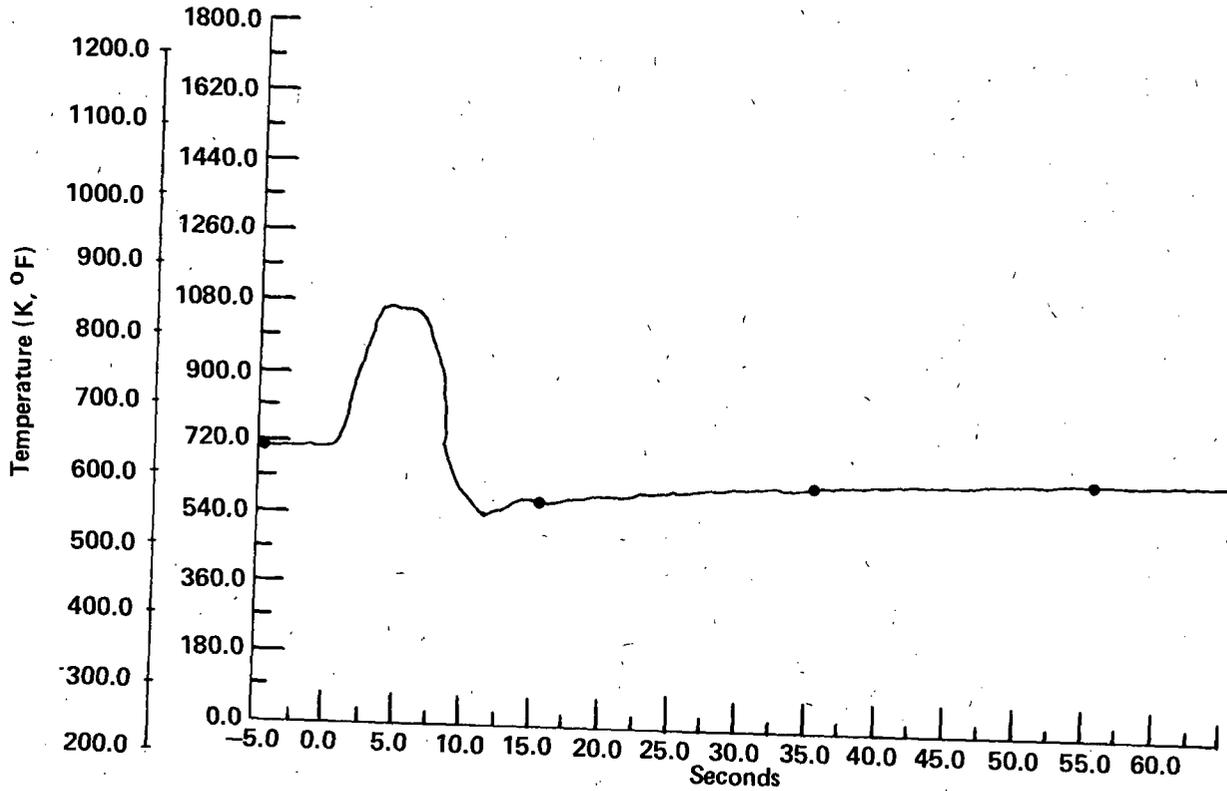


Figure 2-3. Sheath Thermocouple Rod 13, Level D

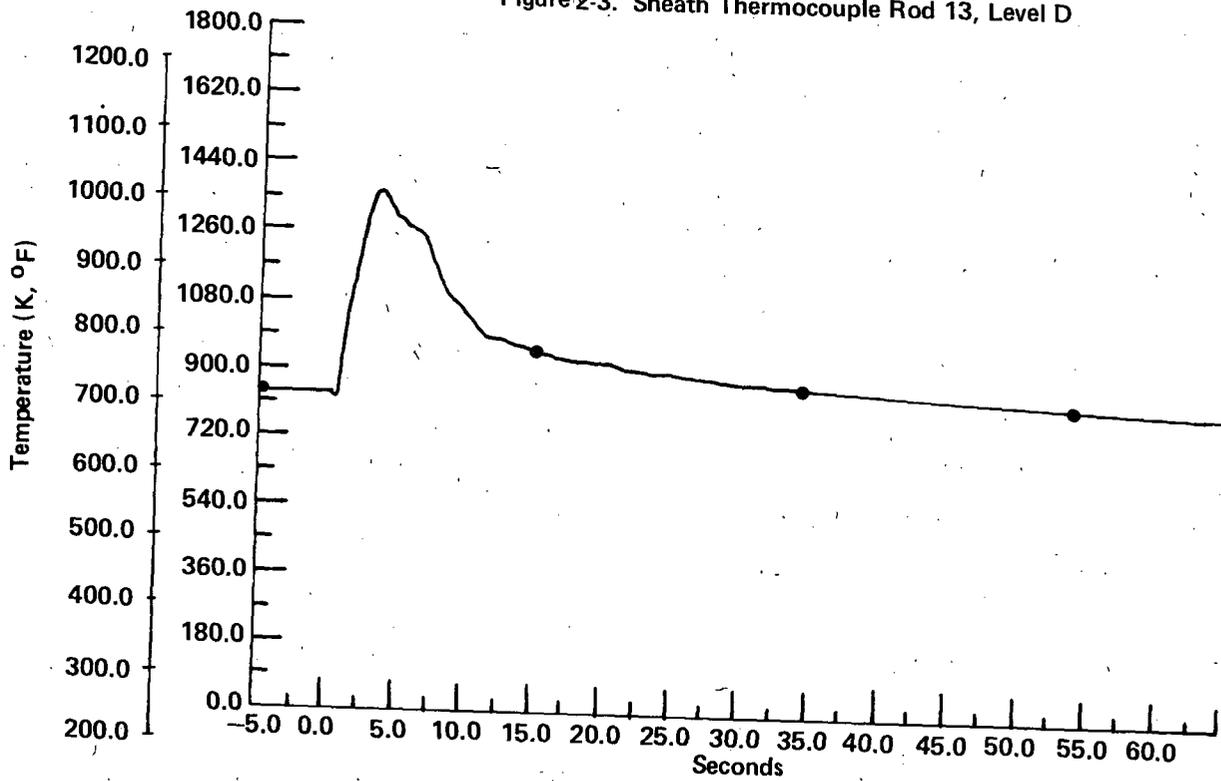


Figure 2-4. Sheath Thermocouple 31, Level G

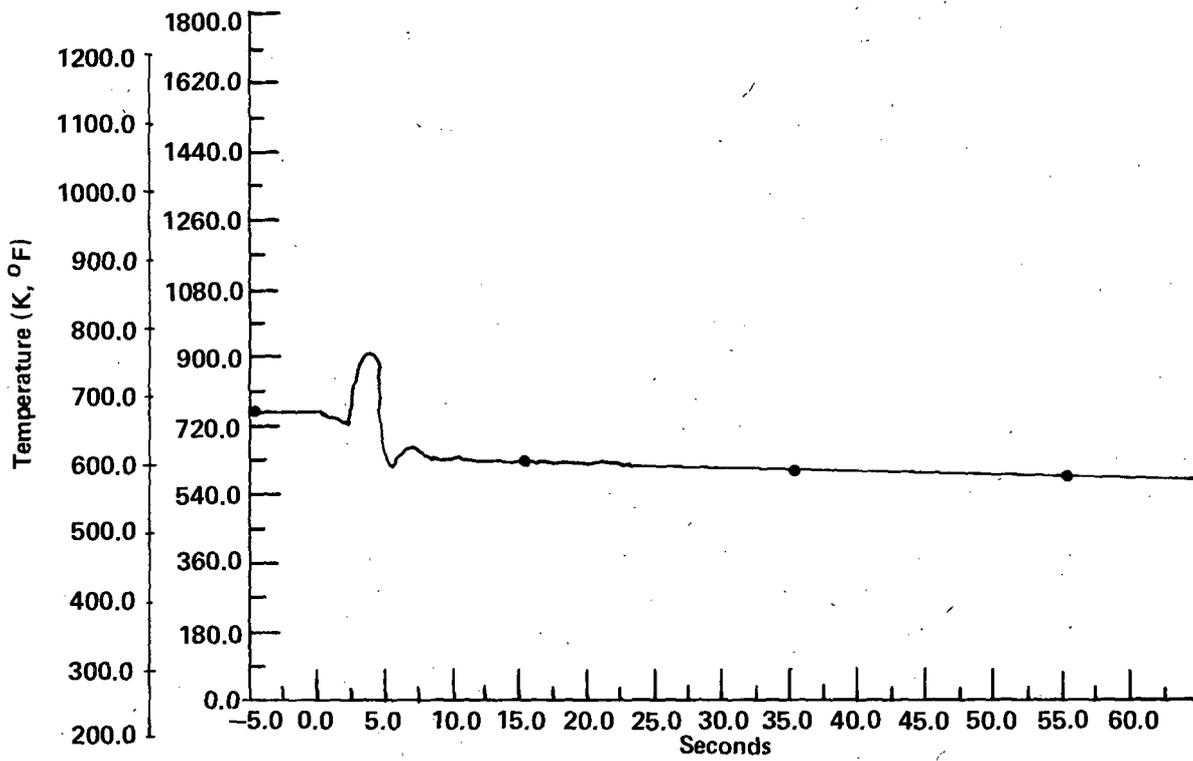


Figure 2-5. Sheath Thermocouple Rod 1, Level J

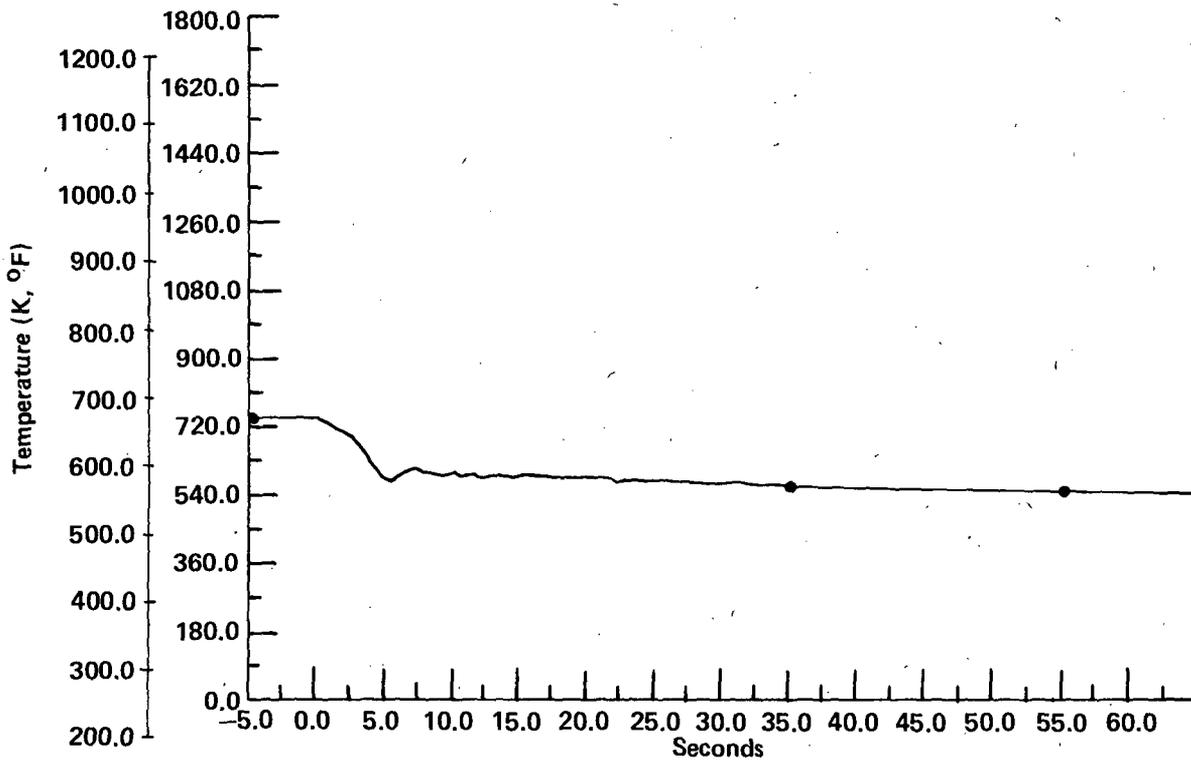


Figure 2-6. Sheath Thermocouple Rod 22, Level K

FLECHT-SEASET Upper Plenum Separate Effects. This task would determine the heat release rate from the larger FLECHT-SEASET steam generator for various known inlet fluid conditions and secondary side conditions.

FLECHT-SEASET System Effects. This task would obtain and evaluate data on the effects of bundle, component, and system parameters and alternative ECC injection configurations on thermal-hydraulic behavior during reflood in system-effects tests.

#### 2.3.1.3.4 Heat-Transfer Correlations and Improvements

Two categories of programs are included. The first category consists of programs with the objective of formulating correlations using existing data. The second category supports the first by conducting experiments to obtain data in transient CHF and post-CHF regimes or to observe phenomena to help formulate a model.

Correlation Efforts. A preliminary correlation for transient CHF has been formed by NRC to include two types of CHF (departure from nucleate boiling and dryout) in one equation. ANL is checking the equation against ORNL and Semiscale data.

A film-boiling correlation for post-CHF heat transfer has been developed by Atomic Energy of Canada, Ltd. (AECL). A correlation covering the whole post-CHF regime (both film boiling and transition boiling) has been developed at Lehigh University. The latter has been subjected to correlation comparisons at the INEL. A review of boiling correlations important in BWR accident analyses will be performed at INEL.

A reflood package based on a phenomenological model (REFLUX) has been developed at the Massachusetts Institute of Technology. This package, which predicts better agreement for high flow rates, has recently been improved to provide a more mechanistic description of the transition to dispersed flow. Models for flow oscillation effects in reflood and for natural circulation (or "chimney effects") are being developed at MIT. WRSR staff are correlating FLECHT data for reflood quench front velocity.

Analytical comparisons of nuclear and electrical rods are being made at ANL and also by WRSR staff. Other comparisons are being obtained at the INEL in work supporting the Halden IFA-511 nuclear-electrical rod reflood experiments.

All of the proposed correlations are subject to testing. Much of the past and current work has emphasized PWR correlations. The INEL will perform correlation testing for BWR CHF, reflood, and rewet in 1979, and will update PWR correlations in 1980.

Experimental Programs Supporting Heat-Transfer Correlations. An experimental transient CHF program with Freon in a small loop is being conducted at the Argonne National Laboratory (ANL) to study the spontaneous-nucleation hypothesis. The Laboratory is also analyzing Semiscale data to discern trends. The University of Maryland, under contract with ANL, has recently completed pool boiling experiments to determine the void-fraction effect on CHF.

Atomic Energy of Canada, Ltd., and the University of Ottawa, both under contract with ANL, are obtaining post-CHF data on transition boiling in a low-flow, low-pressure regime under quasi-steady-state conditions. These data will be very useful in assessing Hsu's equation<sup>60</sup> for transition boiling.

Lehigh University is about to complete modification of the test facility to obtain post-CHF film-boiling data in dispersed flow regimes. The main purpose is measurement of nonequilibrium steam superheat to check against existing nonequilibrium quality film-boiling equations.

The Argonne National Laboratory is conducting tests to determine the effect of oscillation on rewetting and heat transfer during reflood. Visual examinations of flow patterns near the quench front will also be made.

The Massachusetts Institute of Technology is conducting tests to determine the effect of power nonuniformity on flow circulation in a bundle.

#### 2.3.1.4 Milestones

- Complete initial time-to-CHF series (using first bundle) in June 1979.
- Install new test bundle (with enhanced capability to measure post-CHF conditions) in October 1979.
- Complete FLECHT-SEASET steam-generator separate-effects tests in December 1978.
- Complete unblocked-bundle forced-reflood and gravity-reflood tests in June 1979.
- Complete 21-rod-bundle flow-blockage and heat-transfer tests in September 1980.
- Complete large-blocked-bundle forced-reflood and gravity-reflood tests in April 1981.
- Complete FLECHT-SEASET upper plenum separate-effects tests in March 1980.
- Complete FLECHT-SEASET system-effects tests in December 1980.
- Void-fraction equation for reflood developed in July 1977.
- Post-CHF equation tested in February 1977.
- Preliminary tests on the effect of controlled oscillations on heat transfer to be completed in August 1978.
- Data on nonequilibrium steam quality during dispersed film boiling to be obtained in 1979.
- Effect of natural circulation on reflood heat transfer to be determined in June 1979.

#### 2.3.2 ECC BYPASS AND STEAM-WATER MIXING RESEARCH

##### 2.3.2.1 Objectives

The objectives of research on ECC bypass and steam-water mixing are as follows:

- Obtain data to related ECC bypass and lower plenum refill phenomena; thereby establishing conditions wherein bypass ends and reflood begins.
- Obtain data for computer code development and model assessment.
- Investigate alternate ECC injection systems.
- Address the question of scale effects on ECC penetration and the refill process.

##### 2.3.2.2 Present Status

The current status is continuance of small scale experiments (1/30, 1/15 and 2/15 scale). Battelle Columbus Laboratories (BCL), Creare, Inc. and Dartmouth College are the principle contractors and an extensive small scale data base has been developed. The outstanding question is extrapolation of small scale results to full scale; this is being addressed through negotiations with the Federal Republic of Germany (FRG) to conduct ECC bypass experiments in conjunction with 3D Upper Plenum experiments at near full scale in the Upper Plenum Test Facility (UPTF). (See also Chapter 6)

FY 1978 activities resulted in the culmination of conceptual design studies for a 1/3 - 1/2 scale ECC Bypass Test Facility (EBTF) and a preliminary evaluation of a large scale Multi-Purpose Test Facility (MTF) which could have conducted ECC bypass and other LOCA related research experiments. Given the outcome of the May 1978 Denver technical meeting and internal NRR staff evaluations, a new large scale facility was not believed to be justified. Therefore, a recommendation was made to the Commission in July 1978, to plan to conduct large scale ECC bypass experiments in the FRG-3D program and obtain a complete data base from the 1/15 - 2/15 scale programs currently underway.

### 2.3.2.3 Research Program

1/15 scale ECC penetration tests have been completed by BCL<sup>72</sup> and Creare<sup>73</sup> in CY 1976, and a comprehensive data base is being obtained in CY 1978 and CY 1979 in the 2/15 scale vessel.

Primary findings to date (based on small scale experiments) are as follows:

1. Condensation is a controlling factor on ECC penetration (i.e., high subcooling of the ECC enhances delivery),
2. Condensation induced transients (e.g., superheated wall effects due to depressurization and system feedback effects) can extend the refill period,<sup>74</sup> and
3. Small scale tests support the  $J^*$  scaling (or flooding theories) approach rather than a  $K^*$  scaling rationale which implies a constant flooding velocity.

As indicated above, the final scaling question will be addressed via tests in the FRG 3D UPTF program. Near term plans call for completion of current ECC penetration and lower plenum refill testing in the BCL and Creare programs in FY 1979-1980. The results of these experiments will be embodied in "correlative" models for use in predicting transient ECC penetration and lower plenum refill phenomena. In addition, plans call for instrumenting the BCL 2/15 scale vessel with Creare type flow regime probes to obtain flow regime data to be used for benchmarking TRAC code downcomer calculations. These small scale experiments will be phased out in FY 1981.

### 2.3.3 OTHER PROGRAMS

#### 2.3.3.1 Objectives

##### Small-Scale Modeling and Experimental Studies

- To obtain experimental data in the following areas as input to model formulation:
  - Condensation.
  - Void generation rate.
  - Nitrogen evolution rate.
  - Steam binding.
  - Upflow and downflow.
- To develop a drift-flux model.

##### Data Bank

- To establish a data bank to store, process, and disseminate data related to WRSR activities.
- To provide an interactive system giving NRC staff and contractors direct access to the existing data in given areas, with capability for retrieval, display, and statistical processing.

## Instrument Development

- To develop devices to measure two-phase flow parameters, including:
  - Flow velocities of each phase.
  - Void fraction of each phase.
  - Quality of each phase.
  - Flow rate and film thickness during film flow.
  - Nonequilibrium enthalpy distribution between phases.
- To develop modeling software to properly translate measurements into meaningful two-phase flow parameters.
- To formulate standards for calibrating two-phase flow measurements.
- To determine what calibration facilities are needed.

## Pump Studies

- To determine transients in two-phase fluid behavior within the pumps under test conditions simulating the blowdown portion of a postulated LOCA in both intact and broken loops. These include tests addressing the potential for pump overspeed induced by the transient. NRC will utilize EPRI/CE and EPRI/B&W test results.
- To use the results of these subscale pump tests to derive pump-transient models for extrapolation to full size and use in LOCA calculations.

### 2.3.3.2 Present Status

#### 2.3.3.2.1 Small-Scale Modeling and Experimental Studies

In many cases, a proper model for phenomenologically based equations cannot be formulated without experimental study of the mechanisms. In this group of studies, each experiment concerns one specific effect. Such studies give better models for use by code developers or better input data for code calculations.

Among the studies listed and discussed in Section 2.3.3.3.1, three are related to the rate of mass transfer between the liquid and the gas phases, and three are related to phase drift under transient conditions. Thus, the common problems are in determining the rate at which an equilibrium state is reached from a nonequilibrium state, whether thermodynamic or hydrodynamic nonequilibrium. The rate of approach to equilibrium is important only in transient processes in which there is not enough time to achieve equilibrium.

Most of these tests have recently been completed and useful results will be forthcoming.

#### 2.3.3.2.2 Data Bank

In the past, research data were usually reported in tabulated form in reports and filed in archives. These data are very difficult to use for the following reasons:

- There is often insufficient information on instrumentation, data acquisition and reduction procedures, and other experimental details to permit evaluation of the uncertainty associated with the data.
- The data are voluminous and are not usually on magnetic tape. Transfer of data to tapes is usually laborious and prone to errors.
- Data are scattered in the literature and difficult to collect.
- Conversion of measured data to local parameters was done with varying degrees of sophistication and care. Various assumptions were used. Comparing local parameters from different sources is very difficult because of this inconsistency.

In order to make data sponsored by WRSR more usable, more consistent, more retrievable, and more informative, an NRC data bank program was started in 1976. The envisioned data bank system will include many topical data bases, one data bank processing system, one measured-data repository, and one local data repository. The functional relationships among the various components are shown in Figure 2-7.

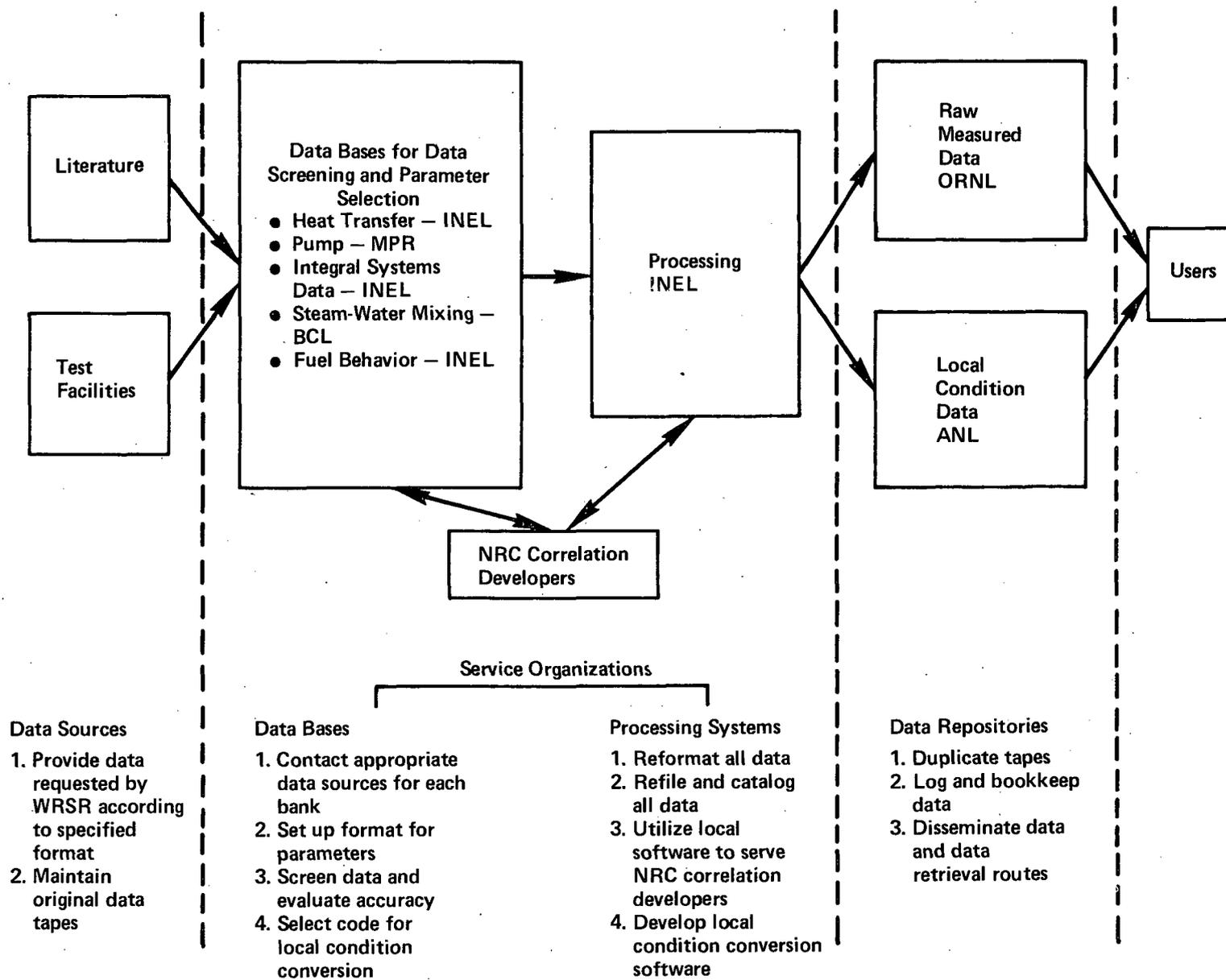


Figure 2-7 Data Bank System

At present, only the data bank processing system, heat-transfer data base, and pump data base are in existence. Data from 35 tests have been processed with the data bank processing system (INEL) for entering into the measured-data repository (ORNL).

#### 2.3.3.2.3 Instrumentation Development

Two-phase instrumentation development is more difficult than single-phase instrumentation development since the phases have different velocities and the void and velocity profiles vary in time and in space, depending on the flow regime. At least three quantities must be determined simultaneously if a two-phase-flow measurement is to be accurate and unambiguous. Three independent measurements are used in three equations to solve for three unknowns. The formulation of the three equations is termed flow modeling.

Until several years ago, most of the measurements were interpreted as homogeneous flows without slip. Translating the measurements into flow parameters could therefore cause substantial errors. Essentially, the effect of flow patterns was not properly considered.

In February 1976 and January 1977, two review group meetings were convened by NRC to review the state of the art of two-phase-flow instrumentation. The basic findings were as follows:

- Existing measurement devices can be improved in sensitivity, response, lifetime, durability, and other qualities, but the return is limited in comparison with the cost.
- Measurement accuracy could be much improved by proper flow modeling.
- New devices should be investigated.
- More refined data analysis algorithms can be developed to extract more information from measurement signals.
- Calibration procedures, methods, and facilities need to be reviewed and defined.

#### 2.3.3.2.4 Pump Studies

Current pump models utilized in LOCA analysis are based on homologous single-phase pump theories and on the assumption that these scaling laws and appropriate two-phase multipliers can be used to calculate transient two-phase pump behavior.

Two-phase pump experiments carried out on the Semiscale pump in fiscal year 1975, in both steady-state and transient modes, yielded head degradation correlations and thus a basis for a transient pump-performance model.<sup>75,76</sup> Data obtained at Babcock & Wilcox (B&W) from steady-state air-water tests with a 1/3-scale pump have been released through an EPRI contract. Both head degradation and torque degradation correlations have been developed. Both the B&W and the Semiscale correlations agree within the seemingly large data scatter. Much of the scatter has recently been eliminated, however, through correlation with the flow coefficient.

#### 2.3.3.3 Research Program

##### 2.3.3.3.1 Small-Scale Modeling and Experimental Studies

Rate of Mass Transfer. Northwestern University has started an experimental program to determine the rate of condensation when a steam jet stream impinges on a water stream. Parameters to be studied include subcooling, impinging angle between jets, gravitational orientation, and flow turbulence. Vertical flow tests are scheduled and a holographic method of two-phase flow measurement will be pursued.

In another experimental program, the Brookhaven National Laboratory will study void generation due to pressurization flashing or wall superheat. Flashing studies will be conducted by directing blowdown through a converging-diverging test section instrumented by a rake of optical probes. Scoping tests are to be conducted in fiscal year 1979.

In a NASA-supported project conducted at NRC's request, the NASA Lewis Center is conducting tests to determine the rate of dissolved-nitrogen evolution during depressurization. Bubble population, growth rate, and void generation rate will be determined.

State University of New York (Stonybrook) is studying droplet size and velocity distribution in two-phase flow by laser-doppler anemometry. The effects of grid spacers, water layers and hot walls on mass transfer are to be studied in fiscal year 1979.

Phase Distribution. In an experimental study of steam binding, the Rensselaer Polytechnic Institute (RPI) is studying parallel channel effects when vapor is rising against liquid in countercurrent flow. Freon is used as the test fluid. The information obtained is useful to BWR and PWR reflood models.

In another series of tests on phase distribution and phase separation, RPI is conducting experiments to determine the migration of voids in channels of various geometries (bundles, wedges, diffusers, etc.). From studies on systems of simple geometry, models can be formulated to account for void distribution in reactor bundles.

In a study of upflow and downflow effects, the Massachusetts Institute of Technology has compared heat-transfer data for similar geometries but with opposite flow directions. Results indicate that downflow is less efficient for heat transfer since drops do not move laterally as much as they do during upflow.

Analysis for Drift-Flux Model. Drift-flux models are being developed at ANL for one-dimensional transient and two-dimensional steady-state and transient applications. The model will be incorporated into the advanced two-dimensional transient code so as to give a better description of two-phase flow. The analyses at ANL and the experiments at RPI are complementary.

#### Milestones

##### Rate of Mass Transfer

- Preliminary data from nitrogen-release tests obtained and analyzed in December 1977.
- Preliminary model for vapor-generation rate during flashing formulated in October 1978.
- Data from condensation tests to be obtained and analyzed by February 1979.

##### Phase Distribution

- Data for upflow and downflow effects analyzed in September 1977.
- Model for parallel-channel flow to be completed in May 1979.
- Models for phase separation and phase distribution to be completed in May 1979.

##### Analysis for Drift Flux Models

- Drift-flux models for one-dimensional transient and two-dimensional steady-state applications to be completed in October 1978.
- Subchannel Analysis for two-fluid drift flux model to be completed in 1979.
- Transport relations for interfacial area concentration to be developed in 1980.

#### 2.3.3.3.2 Data Bank

The current programs for the data bank may be outlined as follows:

Data Bank Processing System (DBPS). The purpose of the DBPS program is to establish a data processing system through which all the NRC data will be cataloged, reformatted, refiled, and submitted to the repository. It will also develop a system for the NRC staff and designated contractors to have direct access to the data in both interactive and batch modes.

Various software programs have been developed in the past year to provide capabilities for generating internal tables and for input/output retrieval and display of data.

Heat-Transfer Data Base (HTDB). The HTDB is a pilot program for the data bank (DB) system before the DB is implemented. The purpose of the HTDB program is to collect, evaluate, and confirm heat-transfer data and to convert the measured data from various sources to local parameters with a consistent set of conversion codes.

Models for local conversion codes have been developed so that local parameters can be interpolated from measured boundary conditions to simultaneously satisfy heat, mass, and momentum balances. After this model was incorporated into the RELAP code, it was tested in a pilot run. The local parameters from the pilot run were input to the Local Data Repository in October 1977.

In addition, a large body (8000 data points) of steady-state heat-transfer data from tubes has been reformatted and input into the Heat-Transfer Data Base.

Initial local condition calculations will be made for tests in THTF, FLECHT, TLTA, and Semiscale, as the initial implementation of the data bank.

Integral Systems Data Base. Expansion of an integral test data base is being planned, and foreign data from Marviken, Studsvik and PKL should be entered into the data base.

Measured-Data Repository. The Measured-Data Repository is located at the RSIC facility at ORNL. When data tapes and related documents are received from the data bank processing system, they are cataloged and entered into the RECON information distribution system. The first pilot run was entered in October 1977. Twenty more runs are being entered in fiscal year 1978.

Local Data Repository (LDR). The LDR is located at the National Energy Software Center at ANL. Local data, together with a computer program for data retrieval, are submitted from the data bank processing system. The programs are written for the IBM 360/575 and the CDC CYBER 173/7600 computer systems. The LDR adapts the programs to other systems. The first pilot run for transient data and the steady-state data was entered in October 1977. Additional transient data will be entered after uncertainties in the calculation of local conditions have been better defined.

#### 2.3.3.3.3 Two-Phase-Flow Instrumentation Development

The NRC's two-phase-flow instrumentation program is summarized in Table 2-4.

For future planning, work will be encouraged in three general areas:

- Development of nonintrusive measurement techniques for application to central regions of heated test bundles and other locations (e.g., piping) where hostile environments and space limitations preclude the acceptance of many direct measurement devices.
- More software refinements to improve the interpretation and resolution of signals and extract additional information from the measured signals. The areas of refinement include two-phase-flow modeling, image analysis, pattern recognition, and spectrum analysis.
- Establishing two-phase flow measurement standards and performing instrumentation calibration.

#### 2.3.3.3.4 Pump Studies

The NRC plans to make use of the results of an EPRI-sponsored program. This program calls for transient and steady-state testing of a 1/5-scale pump in a steam-water mixture (at Combustion Engineering, Inc.), supported by a 1/20-scale test program to study scaling effects (at Creare, Inc.). MIT is under contract with EPRI to analyze the results and to develop phenomenological models to describe two-phase pump performance. Phase I testing at Combustion Engineering, Inc., was completed in May 1977, and the remaining test series was completed in calendar year 1977. Additional tests of 1/4- and 1/5-scale pumps available through Kraftwerk Union in the Federal Republic of Germany are planned at Combustion Engineering, Inc., in calendar year 1978. A pump was tested in the broken loop of Semiscale in January 1978 to further address the overspeed problem.

TABLE 2-4  
INSTRUMENTATION PROJECTS ON TWO-PHASE FLOW

<u>Contractor</u>	<u>Instrumentation</u>
IMPROVEMENT OF EXISTING FLOW DEVICES	
Brookhaven National Laboratory	Local probes to measure flow parameters during void generation
Idaho National Engineering Laboratory	Drag discs, turbine meters, gamma densitometers, pitot tubes, liquid level detectors and spool pieces
Oak Ridge National Laboratory	Pitot tubes and differential pressure devices
NEW FLOW DEVICES	
Argonne National Laboratory	Radioactive tracers (RT)
Idaho National Engineering Laboratory	Pulsed Neutron Activation (PNA), RT and ultrasound
Lehigh University	Film probe for three-dimensional tests
Los Alamos Scientific Laboratory	Stereo lens
Oak Ridge National Laboratory	Impedance probes for three-dimensional test, neutron, gamma, fluidic, and film probes
Rensselaer Polytechnic Institute	Infrared, Radiofrequency, PNA and impedance
State University of New York	Laser-doppler anemometry and calibration
MODELING	
Argonne National Laboratory	Flow split modeling
Brookhaven National Laboratory	Modeling global measurements
Idaho National Engineering Laboratory	Air-water vs steam-water
Lehigh University	Cross-correlations between two film probes
Oak Ridge National Laboratory	Transient cross-correlation
Rensselaer Polytechnic Institute	Flow modeling
ENTHALPY DISTRIBUTION	
Idaho National Engineering Laboratory	Steam superheat probe
Lehigh University	Steam superheat probe
Northwestern University	Probe for condensation rate measurements

#### REFERENCES\*

1. G. D. McPherson, "Non-Nuclear Test in the Loss-of-Fluid Test (LOFT) Facility," Nuclear Safety, May-June 1977. Available in public technical libraries.
2. J. R. Chappel, Experiment Data Report for LOFT Nonnuclear Test L1-1, USNRC Report TREE-NUREG-1025, January 1977.
3. H. C. Robinson, Experiment Data Report for LOFT Nonnuclear Test L1-2, USNRC Report TREE-NUREG-1026, January 1977.
4. G. M. Millar, Experiment Data Report for LOFT Nonnuclear Test L1-3, USNRC Report TREE-NUREG-1065, April 1977.
5. G. M. Millar, Experiment Data Report for LOFT Nonnuclear Test L1-3A, USNRC Report Report TREE-NUREG-1027, December 1976.
6. L. P. Leach, L. J. Ybarrodo, and G. D. McPherson, "Experimental Emergency Core Cooling Results from LOFT Non-Nuclear Tests," Nuclear Technology, 33, 126-149 (mid-April 1977). Available in public technical libraries.
7. D. L. Batt, Experiment Data Report for LOFT Nonnuclear Test L1-4, USNRC Report TREE-NUREG-1084, July 1977.
8. M. S. Jacoby, Experiment Data Report for LOFT Nonnuclear Test L1-5 (Isothermal Test with Core I Installed), USNRC Report TREE-NUREG-1215, June 1978.
9. S. N. Zender, Experiment Data Report for Semiscale Mod-1 Test S-01-2 (Isothermal Blowdown with Core Resistance Simulator), USERDA Report ANCR-1194, February 1975.
10. S. N. Zender, Experiment Data Report for Semiscale Mod-1 Test S-01-3 (Isothermal Blowdown with Core Resistance Simulator), USERDA Report ANCR-1195, March 1975.
11. S. N. Zender, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Tests S-01-4 and S-01-4A (Isothermal Blowdown with Core Resistance Simulator), USERDA Report ANCR-1196, March 1975.
12. S. N. Zender, H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-01-5 (Isothermal Blowdown with Core Resistance Simulator), USERDA Report ANCR-1197, April 1975.
13. S. N. Zender, H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-01-1 (Isothermal Blowdown with Core Resistance Simulator), USERDA Report ANCR-1198, April 1975.
14. H. S. Crapo, M. F. Jensen, K. E. Sackett, and S. N. Zender, Experiment Data Report for Semiscale Mod-1 Test S-01-1B (Isothermal Blowdown with Core Resistance Simulator), USERDA Report ANCR-1199, May 1975.
15. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-01-6 (Isothermal Blowdown with 40-Rod Heater Core), USERDA Report ANCR-1253, September 1975.
16. C. E. Cartmill, Thermal-Hydraulic Response of the Semiscale Mod-1 System - Isothermal Test Series, USERDA Report ANCR-1228, October 1975.
17. R. T. French, An Evaluation of Piping Heat Transfer, Piping Flow Regimes, and Steam Generator Heat Transfer for the Semiscale Mod-1 Isothermal Tests, USERDA Report ANCR-1229, August 1975.

---

\*Unless otherwise indicated, the technical reports listed here are available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

18. D. J. Hanson, R. T. French, and C. J. Shaffer, An Evaluation of ECC Related Phenomena During Semiscale Mod-1 Isothermal Tests, USERDA Report ANCR-1239, November 1975.
19. G. G. Loomis, Intact Loop Pump Performance During the Semiscale Mod-1 Isothermal Test Series, USERDA Report ANCR-1240, October 1975.
20. E. M. Feldman and D. J. Olson, Semiscale Mod-1 Program and System Description for the Blowdown Heat Transfer Tests (Test Series 2), USERDA Report ANCR-1230, August 1975.
21. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experimental Data Report for Semiscale Mod-1 Test S-02-1 (Blowdown Heat Transfer Test), USERDA Report ANCR-1231, July 1975.
22. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-02-2 (Blowdown Heat Transfer Test), USERDA Report ANCR-1232, August 1975.
23. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-02-3 (Blowdown Heat Transfer Test), USERDA Report ANCR-1233, September 1975.
24. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-02-4 (Blowdown Heat Transfer Test), USERDA Report ANCR-1234, November 1975.
25. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-02-5 (Blowdown Heat Transfer Test), USERDA Report ANCR-1235, December 1975.
26. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Tests S-02-9 and S-02-9A (Blowdown Heat Transfer Tests), USERDA Report ANCR-1236, January 1976.
27. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-02-7 (Blowdown Heat Transfer Test), USERDA Report ANCR-1237, November 1975.
28. T. K. Larson, Core Thermal Response During Semiscale Mod-1 Blowdown Heat Transfer Tests, USNRC Report ANCR-NUREG-1285, June 1976.
29. J. M. Cozzuol, Thermal-Hydraulic Analysis of the Semiscale Mod-1 Blowdown Heat Transfer Test Series, USNRC Report ANCR-NUREG-1287, June 1976.
30. H. S. Crapo and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-02-8 (Blowdown Heat Transfer Test), USNRC Report ANCR-NUREG-1238, August 1976.
31. J. M. Cozzuol, R. F. Farman, M. A. Langerman, and T. K. Larson, Semiscale Test S-02-8 Mini-Topical for Standard Problem Five, USNRC Report TREE-NUREG-1015, November 1976.
32. D. G. Hall, A Study of Critical Flow Prediction for Semiscale Mod-1 Loss-of-Coolant Accident Experiments, USNRC Report TREE-NUREG-1006, December 1976.
33. E. M. Feldman, Semiscale Mod-1 Program and System Description for the Reflood Heat Transfer Tests (Test Series 3), USNRC Report ANCR-NUREG-1305, May 1976.
34. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Tests S-03-5, S-03-6, S-03-7, and S-03-8, USNRC Report ANCR-NUREG-1308, June 1976.
35. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Tests S-03-1 Through S-03-4 (Reflood Heat Transfer Tests), USNRC Report ANCR-NUREG-1306, May 1976.
36. G. E. McCreery, G. G. Loomis, H. R. Bruestle, and D. J. Hanson, Thermal-Hydraulic Analysis of Semiscale Mod-1 Reflood Test Series (Gravity Feed Tests), USNRC Report TREE-NUREG-1010, January 1977.
37. A. C. Peterson, G. G. Loomis, and L. L. Chen, Thermal and Hydraulic Response of the Semiscale Mod-1 Core During Forced Feed Reflood Tests, USNRC Report TREE-NUREG-1001, October 1976.
38. H. S. Crapo, M. F. Jensen, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-29-1 (Integral Test with Asymmetrical Break), USNRC Report ANCR-NUREG-1327, July 1976.

39. H. S. Crapo and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-29-2 (Integral Test from Reduced Initial Pressure), USNRC Report ANCR-NUREG-1328, August 1976.
40. H. S. Crapo, B. L. Collins, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-29-3 (Integral Test from Reduced Initial Pressure), USNRC Report ANCR-NUREG-1329, September 1976.
41. H. S. Crapo, B. L. Collins, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-04-1 (Baseline ECC Test), USNRC Report ANCR-NUREG-1330, September 1976.
42. H. S. Crapo, B. L. Collins, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-04-2 (Baseline ECC Test), USNRC Report ANCR-NUREG-1331, September 1976.
43. H. S. Crapo, B. L. Collins, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-04-3 (Baseline ECC Test), USNRC Report TREE-NUREG-1002, October 1976.
44. H. S. Crapo, B. L. Collins, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-04-4 (Baseline ECC Test), USNRC Report TREE-NUREG-1003, October 1976.
45. H. S. Crapo, B. L. Collins, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Tests S-04-5 and S-04-6 (Baseline ECC Tests), USNRC Report TREE-NUREG-1045, January 1977.
46. E. M. Feldman, M. L. Patton, Jr., and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-05-1 (Alternate ECC Injection Test), USNRC Report TREE-NUREG-1050, February 1977.
47. E. M. Feldman, B. L. Collins, and K. E. Sackett, Experiment Data Report for Semiscale Mod-1 Test S-05-2 (Alternate ECC Injection Test), TREE-NUREG-1051, February 1977.
48. B. L. Collins and E. M. Feldman, Experiment Data Report for Semiscale Mod-1 Test S-05-4 (Alternate ECC Injection Test), TREE-NUREG-1053, March 1977.
49. Reactor Safety Laboratory 1, ROSA Group, ROSA-II Test Data Report, 5 (Runs 310, 311, 312, 313, 317), JAERI-M-6709, Japan Atomic Energy Research Institute, September 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
50. Reactor Safety Laboratory 1, ROSA Group, Experimental Program on Performance of Upper Head Injection System with ROSA-II, JAERI-M-6707, Japan Atomic Energy Research Institute, September 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
51. Project Description, ORNL PWR Blowdown Heat Transfer Separate-Effects Program - Thermal-Hydraulic Test Facility (THTF), USNRC Report ORNL/NUREG/TM-2, February 1976.
52. U.S. Atomic Energy Commission, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, Docket No. RM-50-1, December 1973. Available for inspection and copying for a fee in the NRC Public Document Room.
53. W. G. Craddick et al., PWR BDHT Separate-Effects Program Data Evaluation Report - Heat Transfer for THTF Test Series 100, USNRC Report NUREG/CR-0105, September 1978.
54. D. M. Leon, M. D. White, and R. A. Hedrick, PWR BDHT Separate-Effects Program - THTF Experimental Data Report for Test 151, USNRC Report ORNL/NUREG/TM-188, March 1978.
55. D. M. Leon et al., PWR BDHT Separate-Effects Program - THTF Experimental Data Report for Test 153, USNRC Report NUREG/CR-0278, September 1978.
56. RELAP-4-Mod-5 Comparison with PWR-BDHT Test 104 Core Data, PG-R-76-007, December 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
57. BWR Blowdown Heat Transfer Program Final Report, GEAP-21214, General Electric Company, San Jose, Calif., February 1976.
58. PWR FLECHT Cosine Low Flooding Rate Test Series Evaluation Report, Westinghouse Electric Corporation Report WCAP-8838, April 1977.

59. FLECHT Low Flooding Rate Skewed Test Series Data Report, NRC-Westinghouse-EPRI Cooperative Research and Development Report, WCAP-9108, April 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
60. Y. Y. Hsu and W. D. Beckner, A Correlation for the Onset of Transient CHF, report to be published.
61. R. A. Smith and P. Griffith, "A Simple Model for Estimating Time to CHF in a PWR LOCA," paper presented at the National Heat Transfer Conference, ASME paper 76-HT-9, 1976. Available from the American Society of Mechanical Engineers, 345 East 47th Street, New York, New York 10017.
62. R. Henry, "Transient CHF," paper presented at the Fourth WRSR Information Meeting, September 27-30, 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
63. L. S. Tong and J. D. Young, "A Phenomenological Transition and Film Boiling Heat Transfer Correlation," in Heat Transfer 1974, Proceedings of the Fifth International Heat Transfer Conference, Tokyo, Japan, September 3-7, 1974, Vol. IV, pp. 120-124.
64. R. J. Mattson, K. G. Condie, S. J. Bengston, and C. F. Obenchain, "Regression Analysis of Post-CHF Flow Boiling Data," in Heat Transfer 1974, Proceedings of the Fifth International Heat Transfer Conference, Tokyo, Japan, September 3-7, 1974, Vol. IV, pp. 115-119.
65. D. C. Groeneveld, An Investigation of Heat Transfer in the Liquid Deficient Regime, Atomic Energy of Canada, Ltd., Report AECL-3281, 1969. Available for inspection and copying for a fee in the NRC Public Document Room.
66. D. C. Groeneveld and G. G. J. Delorme, "Prediction of Thermal Non-Equilibrium in the Post-Dryout Regime," Nuclear Engineering and Design, 36, 17-26 (January 1976). Available in public technical libraries.
57. R. S. Dougall and W. M. Rohsenow, Film Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities, Massachusetts Institute of Technology, Report No. 9079-26, 1963. Available for inspection and copying for a fee in the NRC Public Document Room.
68. J. C. Chen and R. Sundaram, "Correlation for Post-CHF Heat Transfer," paper presented at the Fourth WRSR Information Meeting, September 27-30, 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
69. F. F. Cadek et al., PWR FLECHT Final Report, Westinghouse Electric Corporation Report WCAP-7665, April 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
70. W. L. Kirchner, Reflood Heat Transfer in a Light Water Reactor, USNRC Report NUREG-0106, 1976.
71. D. J. Hanson "Semiscale Reflood Results," paper presented at the Fourth WRSR Information Meeting, September 27-30, 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
72. R. A. Cudnik et al, Penetration Behavior in a 1/15 Scale Model of a Four-Loop Pressurized Water Reactor, USNRC Report BMI-NUREG-1973, June 1977.
73. C. J. Crowley et al., Downcomer Effects in a 1/15 Scale PWR Geometry - Experimental Data Report, USNRC Report NUREG-0281, May 1977.
74. C. J. Crowley et al., Preliminary Analysis of Condensation - Induced Transients, Quarterly Progress Report, July 1, 1977 - September 30, 1977, Creare Inc. Report TN-271, December 1977.
75. D. J. Olson, Experimental Data Report for Single- and Two-Phase Steady State Tests of the 1-1/2 Loop Semiscale System Pump, ANCR-1150, May 1974.
76. D. J. Olson, Single- and Two-Phase Performance Characteristics of the Mod-1 Semiscale Pump Under Steady State and Transient Fluid Conditions, ANCR-1165, October 1974.

CHAPTER 3  
PROGRAM PLAN  
FOR THE FUEL BEHAVIOR RESEARCH BRANCH

3.1 INTRODUCTION

The program of the Fuel Behavior Research Branch is directed at providing a detailed understanding of the response of nuclear fuel assemblies to abnormal or accident conditions. This understanding is expressed in terms of confirmed analytical models that are incorporated into computer codes; the latter are available to the Regulatory Staff for use in licensing nuclear reactors.

The NRC requires that the safety of the public be ensured through the use of a "defense in depth" in the design and operation of nuclear reactors. Defense in depth includes the use of multiple barriers against any escape of radioactivity. The first barrier is the fuel cladding, which contains not only the fuel pellets but also the fission products generated in the fission process. An understanding of the internal and external conditions that influence the ability of the cladding to retain its integrity during postulated accidents is a prime goal of the fuel behavior program.

Other barriers to the escape of radioactivity are the primary reactor system and the containment building. In order to ensure that consideration of unforeseen events affecting these barriers is not omitted from reactor design and safety analysis, several accident sequences called "design-basis accidents" have been postulated. The analysis of each type of accident must provide assurance that the safety features engineered into the plant are adequate to limit the occurrence and the consequences of any subsequent fission-product release from the fuel, in conformance with the guidelines on radiological doses specified in 10 CFR Part 100.

The loss-of-coolant accident (LOCA) initiated by the rupture of a large primary-coolant pipe has been selected as the design-basis accident for evaluating many of the safety features of LWR power plants. In other postulated accident sequences that would affect fuel-element behavior, the boundary of the reactor-coolant system remains intact, but there is an imbalance between the heat being generated by the fuel and the heat removed by the coolant. The generic term for such accidents is "power-cooling mismatch" (PCM). A PCM would result from a loss of coolant, an overpower transient, a reactivity-initiated accident (RIA) from such causes as control-rod ejection, and an anticipated transient without scram (ATWS).

The condition of the fuel element at the initiation of the accident could greatly influence the course of the accident. The principal initial parameters that must be known for the analysis of a transient are the stored heat and decay heat in the fuel, the gas pressure within the cladding, the extent of contact between the fuel and the cladding, and prior cladding strains. These parameters are interrelated and depend on a number of properties, such as thermal conductivity, thermal expansion, cracking and restructuring of the fuel, the fuel-to-cladding gap width, fission-gas release, and cladding creep. It is therefore necessary to understand the current trends in fuel design and to determine, either by analysis or experiment, their influence on the above-mentioned parameters. Recent examples of such trends are prepressurization of the fuel rods, improved stabilization of pellet density during irradiation, and changes in fuel-rod diameters instituted by all reactor vendors.

The emergency core cooling system (ECCS) is a principal safety feature installed to maintain the integrity and long-term coolability of the fuel during a LOCA. The ECCS Acceptance Criteria<sup>1</sup> are intended to ensure the effectiveness of the ECCS if it should ever be needed in maintaining the structural integrity of the cladding. Two of the criteria supply direct guidance for planning the fuel behavior program:

"Peak Cladding Temperature. The calculated maximum fuel element cladding temperature shall not exceed 1477 K (2200°F).

"Maximum Cladding Oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted

with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at the elevation. For ruptured cladding the circumference does not include the rupture opening."

Subsequent discussion of these criteria in the document identified the need for additional information and improved analysis to narrow the statistical spread of existing data and to relieve excessive conservatism that might have been adopted in setting the criteria. These data needs are related to the following topics:

- Zircaloy oxidation and its embrittling effect.
- The estimated deformation of multirod clusters under postulated accident conditions.
- The stored heat calculated to be in the fuel at the start of the accident.
- The dependence of the fission-product decay heat on time.
- The effect of cladding swelling on the gap heat conductance.

Each of these concerns is addressed in this plan. The Fuel Behavior Research Branch works in close cooperation with the Analysis Development Branch to provide fuel behavior codes that will be compatible with the advanced whole-core codes. In order to develop and test these fuel codes against experiments, the Branch directs an extensive program of out-of-reactor and in-reactor experiments and supporting analyses. The research approach is derived primarily from the specific information needs required for the computation of (a) the steady-state performance of fuel rods as a function of burnup, (b) the response of fuel rods subjected to postulated accidents at any time during their normal useful life, and (c) the release, transport, and deposition of fission products. The program approach is also influenced by the data needs identified in the ECCS Acceptance Criteria and by the NRC's Office of Nuclear Reactor Regulation. Many of the models and correlations in current use are satisfactory, and additional data are not needed. In these cases, the Branch recommends the model or correlation for use in code development, while a research effort is recommended to correct deficiencies in knowledge that significantly influence code predictions.

The Branch sponsors experimental and analytical research on cladding, fuel, and integral fuel-rod properties and performance in areas that are not now modeled with the desired precision. Separate-effects and integral fuel-rod experiments are performed to this end in both out-of-reactor and in-reactor facilities. The out-of-reactor experiments are principally intended for the development (or assessment) of correlations needed by the codes. The in-reactor experiments study integral fuel-rod properties and behavior and include steady-state and transient experiments in such facilities as the Power Burst Facility (PBF), the Loss-of-Fluid Test (LOFT) Facility, and the Halden experimental boiling water reactor in Norway. Short- and long-term steady-state tests examine the properties of integral rods, such as gap heat conductance and fission-gas release, as well as burnup effects on these and other properties. The transient in-reactor tests examine the behavior of single and clustered fuel rods under conditions similar to those postulated to occur during a LOCA, RIA, flow blockage, or other design-basis accidents. The experiments are important to the assessment of the possible consequences of these accidents and in testing the progress of analytical code development. Data generated by these experimental programs are used to derive, improve, or test materials-property correlations and analytical models.

Program review groups consisting of technical experts from various NRC offices have been organized to review the progress of the program, to make recommendations of priorities, and to review the completed correlations before they are released for incorporation into the codes. Consultants from universities and national laboratories are used in the review process.

### 3.2 BASIC STUDIES

This section discusses the current status of, and research program for, basic studies on the properties of cladding and uranium dioxide, and the interaction of cladding and fuel in a fuel rod. Empirical and semiempirical correlations for many of these properties have been published<sup>2</sup> as part of a continuing program sponsored by the Fuel Behavior Research Branch.

### 3.2.1 CLADDING PROPERTIES

#### 3.2.1.1 Present Status

Most LWR fuel is clad with a zirconium alloy because of zirconium's low neutron-absorption cross section and good resistance to water corrosion at high temperatures. Stainless steel cladding is still used in some of the older designs. Zircaloy-2, formed by the addition of small amounts of tin, iron, chromium, and nickel to zirconium, is widely used in BWRs. Zircaloy-4 has physical and mechanical properties that are almost indistinguishable from those of Zircaloy-2. It has a lower hydrogen pickup than does Zircaloy-2 at the temperature, and in the more chemically reducing water environment, of the PWR. The composition and minimum acceptable properties for reactor-grade Zircaloy are specified by ASTM 353-77.<sup>3</sup>

##### 3.2.1.1.1 Texture and Heat Treatment

The importance of texture and heat treatment to the properties of Zircaloys has been recognized in recent years and has been the subject of numerous out-of-reactor tests and a few in-reactor tests.<sup>4,5</sup> The highest strength and best ductility under biaxial loading are associated with a texture in which the basal poles are oriented nearly parallel to the radial direction of the tubing. This texture is best determined by x-ray techniques, but it is most commonly obtained in practice by requiring that the orientation of zirconium hydride platelets formed by adding hydrogen to the tubing be aligned along the circumference of the tubing.<sup>6,7</sup> Each tubing vendor has a proprietary means of producing the desired orientation. The strength and ductility of Zircaloy of a given texture may be varied widely by controlling the degree of cold work and of stress relief. There is no industry-wide consensus of desirable properties, and each vendor selects properties on the basis of desired performance. In practice, irradiation effects tend to bring the properties toward common values, and there is no definitive evidence that one set is superior to another in an accident situation.

At present, the only way to describe the major features of the texture of a given Zircaloy specimen is by the use of a pole figure determined by x-ray diffraction. The pole figure is a two-dimensional representation of a three-dimensional map of the areal concentration of chosen crystallographic directions in a coordinate system formed by the three orthogonal fabrication directions. Although this method does not permit the texture to be stated simply, it is the only available source of information that can be directly related to the anisotropy of mechanical properties. Two things are needed: (a) a method of describing the major features of the pole figure by, for example, a simple three- or four-symbol code so that purchase specifications can be stated accurately and simply, the pole figure can be described with sufficient accuracy for message transmittal or correlation with properties in tabulated data, or for statements usable in computer codes such as FRAP; and (b) a correlation that mathematically relates the pole figure to the anisotropy of properties. With these available, the anisotropy of mechanical properties of all types would not have to be determined directly on each fabricated lot of Zircaloy tubing to allow cladding behavior to be accurately predicted for postulated accidents. Instead, only the x-ray diffraction pole figure would have to be determined, a relatively inexpensive procedure.

##### 3.2.1.1.2 Mechanical Properties of Unirradiated Zircaloy

Important mechanical properties of fuel cladding include yield stress, ultimate stress, uniform and total elongations, reduction of area and thickness at fracture, burst stresses, burst strains, stress rupture, collapse pressure, creep rates (both internal and external pressurization), and low-cycle fatigue. It is necessary to know most of these properties as functions of temperature, oxygen content and distribution, state of stress (uniaxial, biaxial, severe restraint, etc.), and strain rate. Some also should be known as a function of preferred orientation and of the testing environment (steam, air, vacuum, or argon).

A satisfactory data base now exists for most of these properties in steam, vacuum, or argon environments at temperatures from 300 to above 1270 K (75 to 1800°F) for material in the following conditions: (a) as-received (cold-worked and stress-relieved), (b) homogeneously distributed oxygen at several contents, and (c) as-oxidized to produce oxygen gradients.<sup>8</sup> More than 40 single-rod burst tests using internal electric heaters have been conducted in steam, and 3 tests have been conducted in argon.<sup>9</sup> The burst-pressure/burst-temperature data determined confirm literature values, but the burst strains in steam are consistently and significantly lower, being never more than 30% of the total circumferential strain at any temperature except for a very narrow temperature range between about 1300 and 1350 K (1875 and 1970°F) within which the strains observed are as great as 80%.

Creep rates for internally pressurized tubing have been determined in both out-of-reactor<sup>10</sup> and in-reactor<sup>11</sup> studies. Studies are now under way for both out-of-reactor<sup>12,13</sup> and in-reactor<sup>14</sup> measurements using external pressurization, the condition existing in operating power reactors. Since Zircaloy is highly anisotropic in its mechanical properties, the creep rates determined by internal pressure may be significantly higher than those typical of external pressure. The creep rates for external pressurization are needed for accurate prediction of gap closure during operation and the physical dimensions of the fuel rod at the start of a postulated accident.

Stress-corrosion cracking in out-of-reactor tests has been reported for Zircaloy exposed to the chemical action of iodine,<sup>15</sup> cesium,<sup>16,17</sup> cadmium,<sup>18</sup> and cerium,<sup>17</sup> but the extrapolation of the data to explain fuel-rod failures observed in reactors is somewhat dubious since the evidence is mainly circumstantial and the cracking rates observed in fuel rods failing in reactors are much higher than the maximum rates observed in out-of-reactor tests. Furthermore, none of the out-of-reactor tests has been performed on irradiated Zircaloy tubing. A recent paper<sup>19</sup> has reported that brittle cracking can occur in unirradiated Zircaloy at temperatures near 625 K (650°F) in very low strain rate tensile tests in air, even though the material has never been exposed to a known corrodant or fission product. Thus it is possible that unirradiated Zircaloy can fail by the stress-rupture mechanism known to occur in high-strength steels and in some titanium alloys. Since stress-rupture failures can occur in a few minutes to an hour or more in these materials, at crack growth rates similar to those required for in-reactor fuel-rod failures, the stress-rupture mechanism may be the cause of brittle cracking in fuel-rod cladding stressed by an expanding fuel pellet during a power increase.

Tests conducted at room temperature show that hydride formation causes considerable loss of ductility and makes the material sensitive to shock loading during handling. Above about 475 K (395°F) the embrittling effect of hydride precipitates is reduced, and at reactor coolant temperatures uniformly distributed hydrides are not considered deleterious.<sup>20</sup> Localized hydride formation (so-called sunburst failures), caused either by a large internal source of hydrogen or by moisture entry through a defect, results in blister formation. Ultimately, under operating stresses, the hydride cracks lead to wall penetration.

#### 3.2.1.1.3 Mechanical Properties of Irradiated Zircaloy

Information obtained from in-reactor tests and from post-irradiation tests is available for most of the properties of interest. For irradiated materials the scatter is generally greater, and since tests are more expensive, there are fewer data points for statistical evaluation. Until recently, most information was obtained at or below the operating temperatures of BWRs.

Irradiation results in a reduction in ductility, a reduction in impact strength, and an increase in strength at and below normal reactor operating temperatures. The yield strength of annealed Zircaloy-2 doubles and remains essentially constant above a neutron fluence of approximately  $5 \times 10^{20}$  n/cm<sup>2</sup>. The yield strength and ultimate strength are effectively the same.<sup>21</sup> The consensus is that strength changes tend to saturate between  $10^{21}$  and  $10^{22}$  n/cm<sup>2</sup> (>1 MeV) but that the effect on ductility saturates between  $3 \times 10^{19}$  and  $10^{20}$  n/cm<sup>2</sup> (>1 MeV).<sup>22</sup> The strain at which plastic instability sets in is reduced with increasing irradiation.

Most of the work performed in this area has involved relatively low integrated fast fluxes (< $10^{20}$  n/cm<sup>2</sup>; E > 1 MeV) and temperatures ranging from 575 to 875 K (575 to 1110°F). There are some data taken at irradiation temperatures as high as 1025 K (1380°F). In all of the low-fluence studies made to date, the authors conclude that higher irradiation temperatures result in reduced radiation damage. However, there is evidence that, though this is true at low fluences, there is a crossover of properties in the range of 1 to  $2 \times 10^{20}$  n/cm<sup>2</sup>, above which the yield strength is greater for the higher irradiation temperature.<sup>22</sup> Furthermore, the damage at higher irradiation temperatures persists to higher out-of-reactor annealing temperatures. This type of behavior has been found by others to be common to several other structural materials, most notably the stainless steels.

Preliminary indications are that since recovery increases significantly beyond 575 K (575°F), the temperature of maximum radiation damage for exposures in the range of  $10^{20}$  to  $10^{22}$  n/cm<sup>2</sup> is somewhere near and below this temperature. However, the recovery annealings are conducted out of reactor, and hence the extension of these results to in-reactor applications may not be warranted.

A significant body of data on the as-irradiated mechanical properties and on the kinetics of irradiation-damage annealing has been obtained<sup>23</sup> on one lot of Zircaloy tubing removed from spent commercial PWR fuel with a burnup greater than 30,000 MWD/MT UO<sub>2</sub>. The properties examined were uniaxial tensile strengths and elongations in tubing at temperatures from 300 to 975 K (80 to 1290°F) and burst tests at 645 K (700°F). Specimens were tested in the as-received

condition, after isothermal annealing at temperatures of up to 975 K (1290°F) and after annealing by transient heating to temperatures of up to 1275 K (1830°F) at rates from 0.5 to 25 K/sec. Similar tests are now under way on a second lot of cladding from spent commercial fuel with a burnup of less than 20,000 MWd/MT UO<sub>2</sub>. Burst tests have been conducted during transient heating to temperatures of up to about 1275 K (1830°F) at several heating rates. The kinetics of irradiation-damage annealing appear to vary with the evaluation method. Yield, tensile, and burst strengths can be fully recovered at some temperatures, while elongation decreases significantly below that observed in the as-received condition, with all tests conducted at 645 K (700°F). Thus there is a "strain-aging" or "aging" phenomenon that affects elongation but not strength properties. More data are needed for cladding at lower burnup, so that the "saturation exposure" for the several properties can be determined. Transient heating burst tests have shown that the properties at burst temperatures of 980 K (1300°F) and higher are essentially the same in both irradiated and unirradiated Zircaloy tubing.

It has been known for some time that irradiation increases the uniaxial tensile creep rate of sheet-type specimens of Zircaloy during in-reactor testing,<sup>24</sup> and some data have been obtained<sup>25</sup> in biaxial tension at about 575 K (575°F). However, there are few observations available on either in-reactor or out-of-reactor creep of Zircaloy tubing under external pressure.<sup>14</sup> No such data have been determined continuously during a test, a necessary condition if primary creep rates are to be determined.

One limitation on the mechanical performance of the cladding is the lack of ductility under the multiaxial stress conditions that the cladding experiences as a result of interaction with the fuel. This is not well understood under steady-state conditions and has been studied only a few times under overpower transients.<sup>26</sup> Additional data are needed to determine the amount of conservatism in the currently used strain-to-failure value of 1%.

Fuel rods are known to have failed even at low burnup during increases in power in operating commercial reactors<sup>27,28</sup> and in deliberate overpower tests in test reactors.<sup>26,29,30</sup> Evidence from postfailure examination indicates that the fuel pellet was expanded by the power increase and stressed the cladding in circumferential tension. Some of the failures were detected during the power increase, some a few minutes after the new power level was reached and held.<sup>31</sup> In all cases, the initial cracking seems to have been by an intergranular-transgranular cracking mechanism, giving the appearance of a "brittle" failure mode. The final part of the crack growth to failure was always by the typical ductile-dimpling mode observed in shear failures. While there is little or no doubt that the ultimate cause of the failure was stressing of the cladding in circumferential tension (with the severe restraint that no axial strain was permitted because of the friction between the cladding and the fuel-pellet column), the mechanism by which the crack initiates and grows in an apparently brittle manner is in doubt. Conclusive experiments have not been performed. The mechanism has been ascribed to stress-corrosion cracking,<sup>27,30</sup> but the evidence is only circumstantial. Stress-corrosion cracking can be produced in unirradiated Zircaloy by the chemical action of the fission products iodine,<sup>15</sup> cesium,<sup>16,17</sup> and cadmium,<sup>18</sup> but only at crack growth rates that are orders of magnitude smaller than those that must exist for the very short failure times observed in real fuel rods. At these strain rates, the unirradiated tubes exposed to the corrodants fail in a ductile manner. The recent observation of a stress-rupture mechanism operating to produce "brittle" crack formation in axial tension in air in unirradiated Zircaloy<sup>19</sup> at strain rates of the magnitude of those required for in-reactor failures allows another mechanism to be postulated. Irradiated tubing must be tested to determine whether the stress-rupture mechanism can be made to operate in both fueled and nonfueled rods.

The effects of the temperature gradient across the cladding on the distribution of radiation damage have not been investigated. In the hottest zone of a peak fuel rod in a PWR, the temperature difference between the inner and outer cladding surfaces is typically 60 K (110°F). Variations in the tensile properties across the cladding may possibly affect the failure mode. Irradiation does not appear to affect high-cycle, low-strain fatigue. Low-cycle, high-strain fatigue endurance seems to be strongly affected by irradiation owing to the decreased ductility, but the data are quite sparse. No data are available on thermal fatigue or stress relaxation, both of which may be important mechanisms in causing accelerated cladding creepdown and cladding collapse under irradiation and changing power loads.

#### 3.2.1.1.4 Zircaloy Oxidation

The oxidation of Zircaloy in steam is an important phenomenon in accident analysis since (a) hydrogen is stoichiometrically generated by the reaction between the metal and the steam, (b) the heat of reaction is high and must be removed to prevent overheating and possible autoignition of the cladding, and (c) the oxygen consumed forms two brittle layers that reduce the wall thickness capable of carrying tensile stresses. The two brittle layers are zirconium oxide

and an oxygen-stabilized alpha phase. The oxygen also dissolves in the remaining beta phase and causes it to be embrittled. For Zircaloy oxidation in steam, the Baker-Just rate-constant equation is currently used for all calculations related to accident analyses. Many objections have been raised to the use of this equation since it does not agree with experimental data in the temperature range of interest. The oxidation of Zircaloy has therefore been the subject of several investigations.<sup>32-35</sup> The new findings are in general agreement as to the physical characteristics of the oxidation process, the mathematical form of the rate curves [parabolic at temperatures of 1270 K (1800°F) and above], and the exponential form of the rate-constant equation. The rate constants in all other equations are significantly lower than that of the Baker-Just equation<sup>36</sup> and fall into two groups, having activation energies of about 34,000 cal/mole<sup>32-37</sup> and of about 40,000 cal/mole.<sup>34,35,38,39</sup> The newer experimental data fall within a small scatter band between the actual data curves reported by Cathcart<sup>39</sup> and by Lemmon,<sup>38</sup> with an activation energy of approximately 38,000 to 39,000 cal/mole. The Baker-Just rate-constant equation falls within the scatter band of the newer data at temperatures of 1370 to 1270 K (2000 to 1832°F) and becomes increasingly conservative as the temperature increases above 1370 K (2000°F).

From data reported by Cathcart,<sup>39</sup> the rate constant,  $\delta^2/2$ , at 1477 K (2200°F) is only 58% of that of the Baker-Just equation, and thus only 76% of the oxidation predicted by the Baker-Just equation is actually observed. Calculation of peak cladding temperatures during a given postulated LOCA showed that the maximum temperature would be approximately 56 K (100°F) lower according to the new rate equation than according to the Baker-Just equation.<sup>40</sup>

In addition to the equations for the amount of oxygen consumed in the reaction of steam with Zircaloy, equations are needed for the rate of growth of the thickness of the oxide layer, the thickness of the oxygen-stabilized alpha layer, and for their combined thicknesses. Furthermore, an equation is needed for the rate of oxygen diffusion in beta-phase Zircaloy. These equations are needed for calculating the degree of embrittlement of Zircaloy fuel cladding that has undergone given time-temperature histories in postulated accident scenarios. Two sets of equations for the rates of growth of the oxide, the alpha, and the combined layer thicknesses have been reported.<sup>32,39</sup> A new determination<sup>41</sup> of the rate of oxygen diffusion in beta-phase Zircaloy has indicated that the rate is approximately half that previously reported.<sup>42</sup> Since the new data for three independent methods of determination were found to be in agreement, it is thought that the new determination is correct.

Metallographic examination of the transverse cross section of the rupture zone of a transiently heated Zircaloy tube burst in steam at about 1350 K (1975°F) has shown<sup>43</sup> that the oxide layer formed during heating is cracked in the thickness direction during burst deformation, new metal is exposed by rapid deformation of the underlying wall at the crack (causing the wall to thin there), the newly exposed metal is rapidly oxidized, and the local deformation ceases. It is not yet known whether the deformation itself accelerates the local oxidation or whether the thickness of the oxide on the newly exposed metal is typical of that formed on bare metal in the time permitted at the temperature range of crack formation. More study is required before the modeling of the entire oxidation process in a fuel rod burst during a postulated LOCA can be completed.

### 3.2.1.2 Research Program

#### 3.2.1.2.1 Cladding Properties at High Temperatures

Better understanding is needed of the progress and consequences of the oxidation and deformation of Zircaloy cladding during a LOCA (or other reactor event that would result in temperature increases in the cladding). The goal is to improve evaluation of the conservatism of the ECCS Acceptance Criteria and to improve the statistical basis for the criteria. The objectives of the experiments to be conducted in this area are to determine (a) the effects of oxidation, rapid heating, and irradiation on the strength and ductility of Zircaloy in the beta phase; (b) the consequences of those effects on the properties of the Zircaloy cladding after the oxygen-contaminated beta phase has transformed to the alpha phase on cooling to the temperatures that would be expected during a LOCA reflood; and (c) the potential of methods or modifications that may be proposed for eliminating or alleviating the worst of the effects and/or consequences. The data determined should also allow the formulation of better cladding failure criteria than those now used and should improve the data base for models used in the codes for steady-state and transient conditions.

### 3.2.1.2.2 Cladding Properties at Reactor Operating Temperatures

More information is needed on the mechanical properties of Zircaloy tubing at and near reactor normal operating temperatures. The physical condition of the fuel rod must be described at the start of a postulated design-basis accident, and the behavior of the fuel rod must be predicted during accidents such as ATWS and PCM and in power ramps within operating procedures. Zircaloy is a highly anisotropic material. Its yield strength in compression can be twice that in tension along the same stress axis. No presently available theory of the stress-strain behavior of ductile materials in three-dimensional stress space can be used to analyze or predict accurately the behavior of a material like Zircaloy.

In-Reactor Creepdown of Zircaloy. At present, the creepdown of Zircaloy tubing on the fuel pellet is analyzed in terms of the creep rates determined either in axial tension or in circumferential tension produced by internal pressure. Neither condition can accurately predict the behavior of Zircaloy tubing in the condition of circumferential compression, which is the condition that actually exists in a fuel rod in an operating reactor. The measurements must be made directly to allow the needed improvement in cladding creepdown models and a better understanding of how the "bamboo structure" is developed in operating fuel rods. These measurements are also needed for a better prediction of gap closure during operation and for defining the state of the fuel rod at the start of a postulated accident. Only very limited data have been obtained under conditions such as those encountered in an operating commercial reactor (i.e., external pressure, small gaps between cladding and fuel or mandrel, and an LWR flux spectrum). At 20 points, continuous measurements will be made of the displacement of the surface of test specimens having controlled cladding-to-mandrel gaps and "pellet-to-pellet" gaps under known external pressures and cladding temperatures. Particular emphasis will be placed on measurements in the primary stage of creep, since this is the stage of major importance for base-load changes (or for load following) after the cladding has first encountered the fuel pellet during operation. Measurements will be made both in reactor and out of reactor.<sup>12</sup> An agreement has been made with ECN-Petten to conduct the in-reactor creepdown experiments in the high-flux reactor at Petten, the Netherlands, starting in February 1978.

Plastic Stress-Strain Behavior of Zircaloy. The plastic strain behavior of Zircaloy under complex stress systems cannot be analyzed by available theories and analytical techniques used for the more conventional isotropic materials such as steels.

Before a suitable theory can be developed, it will be necessary to obtain a considerable body of experimental data on the plastic strain behavior of Zircaloy under known and controlled biaxial stress conditions<sup>44</sup> for a variety of textures and over the temperature range from about 300 to 675 K (70 to 750°F).

For lack of better information, conventional stress-strain properties of Zircaloy, determined by the standard tensile test, are used in the design of cladding and for the analysis of its behavior during postulated accidents. However, the computer codes (e.g., FRAP and SSYST) used for the analysis of postulated accidents need information on the behavior of the material in true-stress/true-strain/constant-true-strain rate testing for the accurate prediction of stress-strain behavior at strains greater than about 2%.

Zircaloy is one of the strain-rate-sensitive materials whose performance cannot be accurately predicted from the load-elongation curve obtained in the conventional constant-head-rate (or constant-gage-rate) tensile tests used for such structural materials as steels, and aluminum alloys. A new method of tensile testing is under development of allow collection directly of true-stress/true-strain/constant-true-strain rate data for strains from yielding to fracture. This test method will be modified to allow uniaxial tension and compression testing of anisotropic Zircaloy and then to allow testing in biaxial stress. A test of the feasibility of the new strain-rate-control method using stainless steel (which is not very sensitive to the strain rate) has already shown that the plastic flow stress at a true strain of 0.8 is 20% higher in tests conducted with constant-true-rate control than in tests conducted with constant-head-rate control.

### 3.2.1.2.3 Cladding Deformation During Single-Rod and Multirod Burst Testing of Unirradiated Pressurized Zircaloy Tubing

The deformation and extent of flow blockage of the coolant channels of a fuel assembly require further study to reduce the statistical uncertainty of the present data and to evaluate the potential of methods or modifications that may be proposed during the study to eliminate or alleviate the worst of the effects or consequences. Experiments are being performed with electrically heated rods to give flattened temperature gradients comparable to those in PWRs and BWRs, with internal pressures from 100 to 1800 psi and heating rates of up to 55 K/sec (100°F/sec). Single rods and clusters of 16 and 64 rods with typical PWR grid spacings will be

studied. The rods are approximately 2 meters (6.5 feet) long with a heated length of 0.92 meter (3 feet), and the grid spacings are about 0.61 meter (2 feet). The data from the single-rod tests<sup>9</sup> will be compared with the data from the cluster tests, and the development of a correlation attempted. The correlation method should allow the prediction of multirod performance from single-rod tests and should greatly decrease the cost of evaluating various cluster configurations and cladding modifications.

#### 3.2.1.2.4 Effects of Temperature on the Strength and Ductility of Irradiated Zircaloy

Spent fuel rods have been obtained from several operating commercial PWRs as experimental materials. The spent rods have been characterized by profilometry, gamma-scanning, and burnup analyses. Specimens have been cut and the fuel removed by pushing or shaking. Uniaxial tensile tests have been conducted on material in the as-received condition at 300 to 1255 K (70 to 1800°F). Specimens annealed isothermally at various temperatures and times, and others annealed by transient heating at various rates to temperatures from 670 to 1255 K (750 to 1800°F), have been tested at 640 K (700°F) by uniaxial tensile tests and by pressure-ramped burst tests.<sup>23</sup> Transient heated, closed-system burst tests on specimens in both as-received and annealed conditions will be conducted shortly. Bend tests and expanding-mandrel tests will be conducted on specimens in both conditions at 640 K (700°F). The same tests are being conducted on unirradiated cladding in the same equipment to permit correlation with similar tests conducted out of the irradiation cells. The spent fuels are selected to allow study of a range of burnups, cladding manufacturers, reactor designs, and operating conditions.

The kinetics of irradiation-damage annealing have been found to depend very strongly on the method of evaluation. The recovery in yield strength and ultimate strength occurs much quicker and at lower temperatures than does the recovery of ductility.

#### 3.2.1.2.5 Zircaloy Oxidation

Measurements have been made of the rate of Zircaloy oxidation in steam at temperatures between 1090 and 1700 K (1500 and 2600°F) (from about the start of beta-phase formation from the alpha phase to the probable maximum temperature of interest). The program includes isothermal oxidation and oxygen-diffusion studies in steam environments representative of LOCA behavior. The data have been used to predict the oxidation and oxygen penetration of beta-phase Zircaloy during thermal transients that bracket LOCA conditions. The predictions were checked by steam oxidation tests conducted under transient conditions with a variety of heating rates. All of the data determined have been reported in various quarterly progress reports,<sup>33</sup> and a final report has been issued.<sup>39</sup> A computer code has been written and tested for calculating oxygen distributions and temperature profiles in Zircaloy during oxidation in steam.<sup>45</sup> Scoping tests of the effects of steam pressure<sup>46</sup> on the oxidation rate of Zircaloy have shown no effects up to pressures of 1500 psi at 1375 K (2010°F) and increasing rates with increasing pressures up to 2000 psi at 1175 K (1650°F), though the maximum rate was less than that predicted by extrapolation of the Baker-Just equation<sup>36</sup> to 1175 K (1650°F).

#### 3.2.1.2.6 Mechanical Properties of Zircaloy Containing Oxygen

The mechanical properties of Zircaloy have been determined as functions of oxygen distribution and content, strain rate, biaxial stressing, microstructure, texture, and temperature over the range between 423 and 1700 K (300 and 2600°F). The strength and ductility of Zircaloy cladding at any temperature are strongly dependent on such factors and are important in producing and controlling cladding deformation during LOCA and PCM accidents. The effects of quenching stresses will be determined on the properties of oxidized Zircaloy tubing. Most of the experimental data have been reported in quarterly progress reports,<sup>8</sup> and several topical reports are in draft form.

#### 3.2.1.3 Milestones

##### Cladding Properties

- Data collection and analysis have been completed and reported on the out-of-reactor collapse of Zircaloy tubes under external pressure at 590 to 700 K (600 to 800°F).
- Collection of data on the uniaxial properties of Zircaloy in tension and compression tested with constant-true-strain-rate control should be completed by September 1978 and on the biaxial properties by September 1979.

- The rates of creep in Zircaloy tubing under external pressure have been determined out of reactor; in-reactor determinations should be completed by December 1979. A confirmed correlation should be available by April 1980.

#### Cladding Deformation During Single-Rod and Multirod Burst Testing of Unirradiated Pressurized Zircaloy Tubing

- Results of more than 40 single-rod tests conducted in steam (showing significantly lower burst strains) have been reported.
- Tests on two clusters of 16 rods were completed in October 1977, and Quick-Look Reports have been issued.<sup>47, 48</sup>
- Data from four 16-rod clusters and one 64-rod cluster should be available by March 1980.
- A preliminary correlation including rod interaction, scaling factors, flow blockage, heating rate, initial and burst pressures, and burst strains should be available by June 1980.
- Comparison of the experimental results and correlation with predictions made with data from other studies will be completed by December 1980.

#### Temperature Effects on Irradiated Zircaloy Cladding

- Isothermal uniaxial tensile and pressure-ramped burst tests have been completed on one lot of spent fuel with a nominal burnup of 35,000 Mwd/MT UO<sub>2</sub>. Burst tests during transient heating have been completed.
- A second lot of spent commercial PWR fuel with a nominal burnup of 12,000 Mwd/MT UO<sub>2</sub> has been obtained and is being characterized. Isothermal tensile and burst tests have been completed, and transient heating burst tests have begun.
- Tests on both lots of spent fuel were completed by June 1978. Tests on a third lot (second lot with a higher burnup) will be completed by April 1979.
- A confirmed model and correlation on the mechanical properties of irradiated Zircaloy for LOCA analyses should be available by June 1979.

#### Zircaloy Oxidation

- A confirmed correlation on the isothermal and transient oxidation of Zircaloy in steam has been developed and reported.<sup>39</sup>
- Effects of impurities in the steam (hydrogen, oxygen, nitrogen) and variations in the composition of Zircaloy (within normal manufacturing tolerances) have been found to be negligible.<sup>39</sup>
- A computer code has been written and tested for calculating oxygen distributions and temperature profiles in Zircaloy during oxidation in steam.<sup>45</sup>
- The diffusion rate of oxygen in beta-phase Zircaloy has been redetermined and found to be half the rate previously reported.<sup>41</sup>
- The uniaxial mechanical properties of homogeneous oxygen-Zircaloy alloys have been determined and reported. A final report has been issued.<sup>49</sup>
- The biaxial mechanical properties of homogeneous oxygen-Zircaloy alloys have been determined and reported. Similar measurements have been made in biaxial stress on specimens oxidized to various total oxygen contents inhomogeneously distributed. A final report has been issued.<sup>50</sup>
- The superplastic properties of as-received and oxygen-containing Zircaloy have been determined at 975 to 1575 K (1290 to 2375°F) for strain rates from  $1 \times 10^{-5}$  to  $5 \times 10^{-1} \text{ sec}^{-1}$  in both controlled-strain-rate tests and transient heating burst tests.<sup>51</sup>
- An instability criterion has been developed for determining the circumferential plastic strain at which the deformation becomes localized to produce the final burst.<sup>52, 53</sup>

- An instrumented pseudo-Charpy impact test has been developed to evaluate the embrittlement of Zircaloy by oxygen.
- A cladding embrittlement criterion and a test method for evaluating it will be submitted for critical review in September 1978.

### 3.2.2 PROPERTIES OF URANIUM DIOXIDE

#### 3.2.2.1 Present Status

Uranium dioxide ( $UO_2$ ), the basic fuel material for the nuclear power industry, is used in the form of cylindrical sintered pellets with as-fabrication densities of 92 to 97% of theoretical. Most current usage, because of former concern over densification, is at the high-density end of the range. Uranium dioxide is selected over other potential fuel material because of its excellent chemical stability, compatibility with Zircaloy, dimensional stability during irradiation, and high melting point. The properties of uranium dioxide have been studied for over 20 years, and extensive irradiation experience has been accumulated.

##### 3.2.2.1.1 Thermal Conductivity

The principal disadvantage of using uranium dioxide is its low thermal conductivity, which limits the power density attainable without developing unacceptable internal temperatures and results in steep temperature gradients within the fuel pellet. An accurate knowledge of uranium dioxide thermal conductivity is important since this property determines the temperature at each position within the fuel pellet, and the temperature, in turn, is the principal variable governing all other physical and mechanical properties and all dynamic processes occurring in the uranium dioxide. The thermal conductivity is almost universally expressed as the integrated conductivity of a solid cylindrical rod of uranium dioxide, 95% of theoretical density between 273 K (32°F) and the melting point. The accurate measurement of the thermal conductivity either in or out of reactor is a complex undertaking and has been the subject of many investigations and much controversy, particularly for the temperature range from 1775 K (2735°F) to the melting temperature, where thermometry becomes difficult.

A best estimate evaluation<sup>2</sup> of available data has indicated a mean value of 97 W/cm. The NRR staff has chosen a conservative value of 93 W/cm for use in its calculations involving the thermal conductivity of uranium dioxide. Further work in determining the effects of irradiation and microstructure would yield only minor incremental improvements and is therefore accorded low priority.

##### 3.2.2.1.2 Fission-Product Release from Fuel Pellets

The release of the gaseous fission products krypton and xenon from uranium dioxide during irradiation has been studied for many years. A quantitative understanding of gas release has evolved slowly, both because several mechanisms for describing the movement of the gas are involved and because the detailed operating history of the fuel rod that was the source of the experimental data was not often available. Fuel temperature is the principal determinant of gas release. The small amount (<3%) of gas released during normal operation is knocked out of the fuel pellet's free surface. As fuel temperatures exceed 1725 K (2592°F), migration of gas bubbles can be significant, and releases approaching 100% can be observed for molten fuel. Burnup has no discernible effect on gas release below 15 Gwd/MTM. There is growing evidence for the enhancement of release<sup>54</sup> above this burnup, and particularly above 30 Gwd/MTM. An improved knowledge of the inventory of short-lived (half-life < 10 days) isotopes in the fuel-rod void volume is desirable from the standpoint of calculating the radiological consequences of accidents. The time delay that these isotopes experience in migrating to the gap can be calculated, but experimental confirmation of these predictions is desirable.

Mechanistic models<sup>55</sup> based on the growth and migration of intergranular bubbles are emerging as most generally applicable in describing gas release during steady-state and transient conditions. However, they are often uneconomical to use routinely in computer calculations. In their place semiempirical correlations<sup>56</sup> based on available operating data and post-irradiation measurements of gas release are used in evaluations for licensing.

Nongaseous fission products are of interest not only because of their contribution to the radiological consequences of accidents, but in some cases, notably iodine and cesium, also because of the roles they may play in determining fuel performance. It is a commonly held opinion that trace amounts of these two elements may attack the cladding chemically and thus make it more susceptible to breaching. Although the migration mechanisms for iodine, cesium,

and the less volatile fission products are not well characterized, it is known that essentially all of them are retained in the fuel pellet during normal operation.

#### 3.2.2.1.3 Fuel Swelling

The generation of solid and gaseous fission products causes the uranium dioxide matrix to expand, and this expansion must be allowed for if the fuel rods are to attain high burnups. The volume expansion due to fuel swelling is accommodated by fabricating the pellets to less than theoretical density and by providing additional width for the gap between the pellet and the cladding. Under normal irradiation conditions, fuel swelling is not particularly important and nominal values of 0.3 to 1.0%  $\Delta V/V$  per 1% burnup are used,<sup>2</sup> depending on the fuel temperature. Since fuel swelling accompanies gas release, most of the mechanisms proposed to describe gas release also include a provision for describing the accompanying swelling.<sup>55</sup> The need for additional data applicable to steady-state operation is not great except as an adjunct to better understanding of the mechanical responses (cracking, thermal expansion, and swelling) that normally occur in a pellet. There is, however, little information on the magnitude of the swelling during a temperature transient and on the viscoelastic properties of pellets when contained by cladding.

#### 3.2.2.1.4 Fuel Densification

Irradiation-induced densification of sintered uranium dioxide pellet fuel can result in shortened fuel columns and increased gap widths. A thorough study<sup>57</sup> of densification has shown that fuel densifies because of the annihilation of small (<1- $\mu\text{m}$  diameter) pores by fission fragments. Dependencies on fission rate, temperature, burnup, initial grain size, initial density, and pore size distribution were elucidated. It was also demonstrated that densification can be controlled by tailoring the microstructure of the as-fabricated fuel. An out-of-reactor test for uranium dioxide pellet stability that can be correlated with in-reactor behavior has been proposed and shown to be reliable.<sup>58</sup> Through these programs the densification of uranium dioxide has become well enough understood to make additional work unnecessary except for possible investigations of advanced fuel compositions.

#### 3.2.2.1.5 Decay Heat

The decay of fission products is a secondary source of heat in the fuel until fission ceases. At that point, the decay heat is the principal driving force for increases in fuel temperature. Until recently, decay heat was represented by the proposed American Nuclear Society Standard ANS 5.1 (1973),<sup>59</sup> which was based on data from the late 1950s. The uncertainty in those data was judged to approach 15%, especially at cooling times of less than 100 seconds. The NRC has prescribed that a conservative value of 1.2 times the 1973 value be used in evaluating the effectiveness of ECCS. More recent calculations and measurements of decay heat in irradiated uranium-235 have demonstrated that the 1973 standard is itself conservative during the first seconds after shutdown. Furthermore, the uncertainties in the new data are nominally less than 5% at short cooling times and decrease as cooling time increases. On the basis of these data, a new standard is being developed and proposed by the American Nuclear Society. Subsequent research related to decay heat need only confirm the calculations for other fissile isotopes of potential interest (e.g., Pu-239, U-238, Pu-241).

#### 3.2.2.1.6 Other Properties of Uranium Dioxide

Creep rates of uranium dioxide have been measured as functions of stress and temperature to 2775 K (4535°F).<sup>63</sup> Microstructural examination revealed that extensive grain growth and reorientation occur during the process. Creep rates compare well with values extrapolated from lower temperature work. Such measurements in an irradiation environment at low temperature may still be necessary.

Besides temperature and irradiation, which affect the thermal conductivity of uranium dioxide, other factors are either of secondary importance or are so well known that more information is not necessary at present. These factors include density, melting point, change of stoichiometry, burnup coefficient of thermal expansion, heat capacity, and certain mechanical properties such as Young's modulus and Poisson's ratio.

#### 3.2.2.2 Research Program

Experiments on fuel properties are providing additional experimental information on changes in fuel pellets in steady-state and transient operation. Emphasis has been placed initially on obtaining information on fission-gas release during temperature transients. The information has been used to improve gas-release models in computer codes for steady-state operation and for transients.

### 3.2.2.2.1 Transient Release of Fission Gas

An out-of-reactor experiment using direct electrical heating to provide temperature gradients and transients similar to those in a fuel pellet in a reactor has been developed. The rate and amount of gas released and its distribution within the pellet are being measured as functions of heating rate, temperature profile, burnup (to 30 GWd/MTM), and radial restraint. Correlation of results with those of integral in-reactor transient tests is being attempted. A mechanistic model (GRASS) to describe steady-state and transient gas release has also been developed.

### 3.2.2.2.2 In-Reactor Transient Gas Release

Two instrumented experiments have been planned for irradiation in the experimental BWR in Halden, Norway. Information is being obtained on the absorption of helium and the release of fission gas during overpower transients following steady-state operation. Data on transient gas flow and fuel centerline temperature are to be obtained for burnups of up to 20,000 MWD/MTM. The first assembly (IFA-429) contains 18 fuel rods, each about 27 cm (11 inches) long, and was loaded into the reactor in May 1975. The second assembly (IFA-430) contains four rods. Two of the rods have pressure taps along the length for measuring the gas flow rate.

### 3.2.2.2.3 Decay Heat

Experimental and analytical programs have been implemented to reduce the uncertainty associated with decay heat from thermal fission in LWR fuel. The program elements include:

- Experimental determination of decay heat in irradiated U-235 by direct calorimetric measurement<sup>61</sup> and by beta and gamma spectroscopy.<sup>62</sup>
- Analysis of decay schemes, statistical treatment of data on decay, and computerized capability to predict integral decay heat for various reactor conditions using summation calculations.<sup>60</sup>
- A continuation of the program to measure the fission-product decay heat from plutonium.

### 3.2.2.3 Milestones

- Preliminary steady-state and transient versions of GRASS for FRAPCON and FRAP-T have been completed.
- Basic experiments to establish the effects of heating rate, temperature profile, and burnup on transient gas release will be completed in the Summer of 1978.
- Experiments to establish the effects of radial restraint will be completed in December 1978.
- Improved and confirmed models of gas release will be available in December 1979.
- Assembly IFA-430 will be installed in the Halden BWR in the Fall of 1978.
- Additional rods from assembly IFA-429 were removed for post-irradiation examination in September 1977.
- Interim summary of data on fission-gas release from assembly IFA-429 in the Halden BWR was issued in 1977.
- Confirmed correlation for transient gas release will be completed in 1979.
- A data report for the first irradiation cycle of assembly IFA-430 will be issued in mid-1979.
- Spectroscopic and calorimetric measurements of U-235 have been completed and final reports have been issued.<sup>61-62</sup>
- Spectroscopic and calorimetric measurements of Pu-239 have been completed; final reports will be issued in December 1978.

### 3.2.3 FUEL-ROD PROPERTIES

#### 3.2.3.1 Present Status

A nuclear fuel rod, made up of uranium dioxide pellets enclosed in a Zircaloy tube, has some properties that cannot be derived separately from the intrinsic properties of the uranium dioxide and the Zircaloy. For example, calculation of the stored heat in the fuel rod requires knowledge of the gap conductance between the fuel and the cladding in addition to knowledge of the thermal conductivity of uranium dioxide and Zircaloy and of the heat-generation rate. Other such properties are the stresses on the cladding from mechanical interaction with the fuel, the gas pressure in the rod, and the chemical effects of the fission products on the cladding.

##### 3.2.3.1.1 Gap Conductance

The thermal behavior of an LWR fuel rod is complex. The mechanisms that have been postulated to influence the thermal behavior of an LWR fuel rod include the following:

- Changes in the dimensions of the fuel-to-cladding gap from pellet cracking, pellet relocation, fuel densification and swelling, thermal expansion, and cladding creep-down.
- Changes in the fuel thermal conductivity from pellet cracking (nonradial cracks) and restructuring.
- Changes in the composition of the gas in the gap or in fuel cracks from impurity-gas release, fission-gas release, and fill-gas absorption.

With pressed and sintered pellets there is usually an appreciable resistance to heat transfer between the pellet surface and the cladding. The interfacial resistance may be the result of a gas-filled gap or uranium dioxide in actual contact with the cladding. Data on fuel centerline temperature and rod internal pressure tend to support the contention that the thermal response of an LWR fuel rod is strongly influenced by stochastic pellet cracking and pellet fragment relocation mechanisms. As fuel burnup progresses, pellet cracking and relocation, pellet swelling, thermal expansion, and cladding creepdown combine to close the gap. The rate of gap closure has been shown to depend on such operating variables as the rate of power increase, number of power cycles, and power level.

Several in-reactor experiments to obtain values for gap conductance have been made, and a large number of investigators have attempted to infer gap conductance from the examination of fuel rods that were irradiated for other purposes. Gap conductance is not measured directly but is derived from measurements of fuel and/or cladding temperatures. Most of the reliable experiments utilized small ( $\leq 200 \mu\text{m}$ ) diametral gaps. There is very little well-characterized data for thermal reactor fuel with larger diametral gaps, especially in the 33- to 50-kW/m (10- to 15-kW/ft) operating power range. Experiments sponsored by NRC have been reported in References 64, 65, and 66.

Reference 64 is the first of a series of data reports that will present test data from the EG&G Idaho-Halden experiment IFA-429. The IFA-429 is an 18-rod test assembly designed to study fission-gas release and fill-gas absorption in prepressurized (2.58-MPa helium) PWR-type fuel rods. This data report presents assembly power history and individual fuel-rod power, temperature, pressure, and burnup data for the Halden irradiation period beginning in June 1975 and ending in February 1976. During this period average burnup for the accumulated assembly reached approximately 5000 Mwd/MTM. Reported fuel-rod heat ratings cover a range from 17 to 300 kW/m with measured fuel centerline temperatures of 1375 to 1475 K (2015 to 2195°F) for the highest power. Measured fuel-rod pressures showed no appreciable change during the period covered.

Results from BPNL-Halden experiment IFA-431 were reported in Reference 65. For one of the rods in this experiment, the average gap conductance uncertainty over the range of measurement was +19%. The uncertainty in the gap-conductance measurement changed as a function of linear heat rating. The experiment includes two fill gases (pure xenon and pure helium) and three pellet-to-cladding gaps. The absolute error in determining the temperature drop across the gap is less than 100°C for any of the combinations of gap diameter and fill gas used.

##### 3.2.3.1.2 Gas Pressure

Knowledge of the gas pressure in the fuel rod is necessary for calculating the stress on the fuel cladding. The internal pressure is expected to increase with burnup. At the end of

fuel-rod life, depending on the degree of prepressurization and internal void volume, it may approach or slightly exceed the pressure of the coolant. The change in internal pressure depends, for example, on fission-gas release, helium absorption, pellet swelling, cladding creepdown, ratio of cold plenum volume to gap volume, and differences in the thermal expansion coefficients of the fuel and the cladding.

Although time-dependent data at burnups to 30,000 MWd/MTU have been obtained<sup>67</sup> in test reactors, the internal pressure of fuel rods during reactor operation is more often obtained by calculations normalized to pressures measured on rods during post-irradiation examination. The uncertainty in the calculation is believed to be about 25%. (See Reference 68 and pages 140-148 of Reference 2.) More recent in-reactor experiments are equipped with improved pressure transducers to provide more reliable data. The current uncertainty leaves too much unreliability in predicting the direction and mode of cladding deformation during postulated power-cooling-mismatch or reactivity-initiated accidents because of the expectation that the pressure differential would remain small.

In a loss-of-coolant accident, substantial positive pressure differentials across the cladding would be expected. The ballooning behavior of Zircaloy is sensitive to combinations of pressure differential and cladding temperature. In addition, the nature of axial flow within a fuel rod may have an important bearing on fuel-rod ballooning and failure characteristics.<sup>69</sup> If gas flow from the plenum is sufficiently delayed, the cladding temperature may drop before the ballooning region can be pressurized, so that the ballooning is stopped before failure.

### 3.2.3.1.3 Fuel-Cladding Mechanical Interactions

Mechanical interactions between fuel and cladding resulting in elastic and plastic deformation have been measured directly in reactor as a function of fuel-rod power level, ramp rate, and power cycling, for moderate burnups. These interactions result in changes in both rod length and diameter. A number of post-irradiation measurements of plastic deformation of commercial fuel rods resulting from fuel-cladding interaction have also been made. The radial deformation results principally from the gap closure caused by pellet relocation and, at high fuel-rod powers, from thermal expansion of the pellet against the cladding. A number of failures of fuel-rod cladding have been traced to this interaction, and the shape of most fuel pellets used in current reactors has been altered to reduce the incidence of failure. Elongation of the fuel rod during power cycling can also be caused by this interaction. Although the contact is usually mechanical, instances of apparent bonding of the pellet to the cladding have been reported.

The bulk of the available data on dimensional changes in cladding and in fuel as functions of power has come from tests in which the fuel operated at high heat ratings and at low burnups. It is necessary to repeat these measurements at power ratings more typical of current operation and as a function of burnup.

In commercial power reactors there have been recent fuel-rod failures that were ascribed to interactions between the pellet and the cladding. In one case, the reactor was deliberately ramped in power just before normal shutdown for removal of a portion of the fuel assemblies.<sup>28</sup> In the second case, the reactor was being operated for base power load, and fuel-rod failures were observed after what were thought to be mild changes in load over a relatively long time interval.<sup>27</sup> Both incidents resulted in fission-gas release to the primary system. In the latter case, some of the failed rods were examined destructively. The evidence showed that the fuel pellets had expanded to stress the cladding in circumferential tension, producing a highly localized stress state in the cladding. At the positions of failure, a crack had initiated and propagated in the radial direction in a brittle mode until a ductile shear-type deformation fracture completed the failure. The failure was ascribed to stress-corrosion cracking by fission products liberated in the failed rods by fuel densification. The evidence is, however, circumstantial, and stress rupture cannot be eliminated as a possible mode of failure in these fuel rods.

Whether the failure mechanism of "brittle cracking" in fuel rods stressed by expanding fuel pellets is due to stress-corrosion cracking or to stress rupture, there appears to be a "critical stress" below which the cladding does not fail. The solution to the problem will depend partly on the mechanism of failure (a barrier layer could prevent crack initiation by stress-corrosion cracking but not by stress rupture) and partly on the "critical stress" level required. An understanding of the actual mechanism of failure is required, since the solution can affect the analyses of fuel-rod behavior during postulated accidents.

### 3.2.3.2 Research Program

Experiments on fuel-rod properties have as their objectives (a) improvement of models for calculating gap conductance in a fuel rod, (b) determination of the extent to which fuel-pellet expansion and bonding to the cladding affect the axial flow of gas within a fuel rod, and (c) the post-irradiation examination of commercial fuel rods to study the mechanism of failures resulting from fuel-cladding interaction.

#### 3.2.3.2.1 Gap Conductance

Knowledge of gap conductance is a major factor in calculating the stored heat in the fuel pellet at the onset of a LOCA. The heat transfer across small gaps is apparently higher than that predicted by current analytical models. Out-of-reactor experiments include extension of the Ross-Stoute data on the contact conductance of Zircaloy-uranium dioxide interfaces to higher interfacial temperatures and gas pressures. The parameters to be varied are gas composition and the characteristics of the fuel-cladding interface.

In-reactor instrumented experiments are under way to provide data on initial gap closure, gap conductance, and fuel temperature. The effect of fuel-pellet eccentricity on gap conductance will also be studied. Six rods have been irradiated to about 5000 Mwd/MTM. A long-term irradiation of six rods (assembly IFA-432) has been initiated in the Halden experimental BWR to measure fuel temperature, rod pressure, and cladding elongation. This experiment will provide a life history of well-characterized rods out to approximately 25,000 Mwd/MTM. The rods will have operated at a nominal level of 49 kW/m (15 kW/ft) with periodic operation at 33 to 40 kW/m (10 to 12 kW/ft). Post-irradiation examination of the rods will provide additional data for gas-release and fuel-swelling correlations. The experiment should serve to confirm codes used to calculate the properties of fuel in steady-state operation. A series of brief irradiation experiments will be conducted at the Power Burst Facility. A parametric determination will be made of the effects of pellet density, gap dimension, fill-gas composition and pressure, and power level. These experiments will provide additional data for confirming gap-conductance models over a wider range of variables than is available from the high-burnup test.

#### 3.2.3.2.2 Commercial Fuel Irradiation Performance

Data describing the physical characteristics of fuel rods that failed as a result of interaction between fuel and cladding are limited to the immediate surroundings of a detected rupture or incipient failure in the few fuel rods destructively examined to date. Attempts to discover incipient failures (cracks that have not penetrated to the outer surface) by nondestructive examination (eddy-current and ultrasonic) have had little or no success.<sup>70</sup> Little profilometry of failed rods has been done, and then only in the region of the "bamboo structure" at a pellet-pellet interface where a rupture had been detected or anticipated. An extensive program of nondestructive and destructive examination will be conducted on fuel rods removed from failed fuel bundles from commercial power reactors. Initially, two BWR 7 x 7 failed fuel bundles (with a burnup of about 8000 Mwd/MT UO<sub>2</sub> and power ramped during operation to a power level exceeding vendor specifications) will be examined visually and by pulsed eddy-current scanning, profilometry, and gamma-scanning methods. A limited number of selected rods will be examined by neutron radiography. Both failed and intact rods will be sectioned for metallographic, scanning electron microscope (SEM), and microprobe studies, and some of the sectionings and examinations will be carried out entirely in an inert atmosphere to prevent any reaction with air or moisture or with any of the possible stress corrodants that might be present. Both SEM fractography and microprobe analyses will be performed on the fracture surfaces of cracks prepared and opened up under the inert atmosphere. Some autoradiography and burnup measurements will be made on selected specimens. Failed PWR bundles will be added to the study when they become available. The data obtained will provide a greatly expanded base for analyzing failures resulting from fuel-cladding interactions and for planning studies aimed at conclusively identifying the mechanism of failure.

#### 3.2.3.3 Milestones

##### Gap Conductance

- Measurements of ex-reactor conductance at low gas pressures were completed in July 1978.
- Revised correlation for contact conductance should be available in May 1979.

- High-burnup irradiation tests of assembly IFA-432 will be completed in the Winter of 1978.
- Oscillation gap-conductance experiments at the Power Burst Facility will be completed in October 1978.

#### Commercial Fuel Irradiation Performance

- Two failed fuel bundles from a commercial BWR have been obtained and delivered.
- Profilometry and pulsed-eddy-current measurements on selected BWR rods will be completed by September 1978.
- Destructive examination of selected failed BWR rod specimens has been completed.
- A report on the post-irradiation examination of PWR (H. B. Robinson station) fuel rods will be issued in 1978.
- A report describing the results of post-irradiation examination with emphasis on evidence of fuel-cladding interaction and fuel structure will be issued in 1979.

### 3.3 IN-REACTOR TESTS

#### 3.3.1 PRESENT STATUS

##### 3.3.1.1 Fuel-Rod Performance During Abnormal Operation

To date, most information in this area has been obtained from small-scale experiments in test reactors and from out-of-reactor tests in which separate properties such as cladding deformation or fission-gas release were measured.

##### 3.3.1.1.1 Power-Cooling Mismatch (PCM)

One class of postulated accidents involves a mismatch between the power generated by a fuel rod and the power carried off by the coolant, i.e., increases in rod power and/or decreases in coolant flow. Power increases could be caused by control-rod withdrawal. Flow decreases could be caused by the failure of a primary pump or by the blockage of a fuel-rod cooling channel. PCM accidents are among those being investigated at the Power Burst Facility (PBF) in Idaho.

##### 3.3.1.1.2 Overpower Transients Without Departure from Nucleate Boiling (DNB)

Anticipated transients without scram (ATWS) include various events that might happen during the operation of an LWR and are usually equivalent to mild PCMs (with no critical heat flux). In a PWR, an ATWS could occur in response to the loss of feedwater flow, leading to an increase in primary coolant temperature and pressure. In a BWR, it could occur in response to a primary coolant system pressure increase, leading to an increase in moderation and power. These incidents are generally assumed to produce little damage to the fuel or to the nuclear system process barrier. The primary controlling parameters for the pre-critical-heat-flux PCM are power ramp rate, incremental increase in power, burnup, diametral gap, and fuel density.

The principal cause of fuel-rod failure during power ramps in which departure from nucleate boiling is avoided appears to be stress due to fuel-cladding interactions. Preliminary calculations indicate that the probability of failure is directly related to cladding hoop stress during this type of accident.<sup>71</sup>

A few tests of cladding deformation after a rapid (5-minute) power increase above steady-state operation have been conducted on large-diameter rods.<sup>26</sup> The important parameters are cladding ductility and degree of overstraining. Power increases of 20% did not cause failure of rods with a burnup of 17,000 MWd/MTM. An additional 30% increase in power to 65.6 kW/m (20 kW/ft) caused failure within 5 minutes. The critical heat flux was not exceeded. Rods containing high-density pellets with small gaps are most susceptible to this type of failure. Typically, cracks in cladding are oriented axially and start at ridge locations (pellet interfaces) on the inner cladding surface. The ridges were not present before the overpower transient.<sup>26</sup>

This type of failure, though it results in extensive hairline cracking and the release of volatile fission products, does not seem to cause fuel exposure to the coolant.

### 3.3.1.1.3 PCM Transients in Pressurized Water Reactors

In recent tests at the Power Burst Facility (PWR conditions) with both fresh and preirradiated fuel rods, single fuel rods were taken into film boiling at peak linear heat-generation rates ranging from 56 to 76 kW/m (17 to 23 kW/ft).<sup>72</sup> Rods were held in film boiling for total times as long as 11 minutes. Peak cladding temperatures reached as high as 1640 K (2490°F) in some of the tests. Thirteen tests were carried out on a total of 37 fuel rods. These tests were conducted to determine the modes, mechanisms, and consequences of fuel-cladding failure caused by higher than average local power levels or reduced cooling capability at high local heat fluxes. The following conclusions have been reached:

- Departure from nucleate boiling and subsequent film boiling generally did not result in rod failure. None of the rods failed at power.
- There were no significant differences between the failure characteristics of irradiated and unirradiated fuel rods.
- There was no melting of fuel in the rod during conditions of film boiling at rod powers typical of commercial power reactors.
- Under higher power conditions, when molten fuel approached the cladding, it solidified and there was no significant reaction of molten fuel with the cladding.
- Rods prepressurized to simulate normal conditions in commercial fuel rods, under all but extreme high burnup conditions, were observed to collapse onto the fuel due to the higher pressure of the reactor coolant outside the cladding.

In a companion test, a rod that was overpressurized internally to simulate very high burnup (extreme end of life) rod plenum pressures failed (at normal power but with very low flow) with a limited amount of ballooning, as predicted.<sup>73</sup>

### 3.3.1.1.4 PCM Transients in Boiling Water Reactors

In-reactor tests under BWR conditions during which coolant flow was reduced while the rod power was maintained have been reported by at least two laboratories. In the reported experiments,<sup>74</sup> the rods were brought to dryout by valving off the coolant flow. The cladding temperature rose to between 870 and 1070 K (1105 and 1465°F) and remained there for up to 5 minutes without causing rod failure. In later tests, different rods were subjected to 3, 33, and 60 burnout excursions over periods of several years. They remained in the reactor core without failure to exposures as high as 23,000 Mwd/MTM. In more recent experiments, a cluster of 36 fuel rods, 1.22 meters (4 feet) long, was subjected to dryout conditions. The fuel cluster experienced more than 120 such onset-of-dryout events at ratings of up to 90 kW/m (28 kW/ft) as well as nine post-dryout transients at ratings of up to 78 kW/m (24 kW/ft). The maximum surface temperature recorded was 875 K (1115°F), and the integrated period of time under post-dryout conditions was not less than 15 minutes. The cluster was irradiated to 900 Mwd/MTM before discharge. The most serious damage to the rods found during post-irradiation examination was a reduction in the strength of annealed rods.

### 3.3.1.2 Reactivity-Initiated Accident Tests

The possibility that a reactivity transient might be induced in a nuclear reactor by the sudden removal of a control rod was probably the first safety concern to be identified. The TREAT reactor and a series of SPERT reactors were built to study this possibility. Experiments with these reactors led to a very full understanding of the inherent stabilizing features of LWRs. Tests on Zircaloy-clad fuel rods with uranium dioxide pellets have been conducted in both TREAT and SPERT to determine the failure thresholds for such rods and the magnitude of any metal-water or fuel-water reactions that could result from a sudden increase in power generation. These tests provided initial conditions of cold startup and zero flow.

Although the reactor periods for the SPERT and TREAT tests were very different (less than 10 msec and 39 to 238 msec, respectively), both gave similar results relative to energy deposition in the fuel at the point of failure. In single-rod tests with unirradiated rods, the failure threshold was found to be at an energy deposition of about 225 cal/g UO<sub>2</sub> whereas prompt dispersal of the fuel began at energy depositions of about 340 cal/g. In transient tests on a small number of irradiated rods with burnups of 1000 to 32,000 Mwd/MTM, the data exhibited considerable scatter but indicated that the threshold for failure was lowered somewhat. Transient pressures considerably larger than those encountered in tests with single unirradiated rods were also

observed. The SPERT tests also indicate that the failure threshold for fuel-rod clusters may be lower than that for single rods. The energy-conversion ratio for a seven-rod cluster was about 1.6 times that of a single rod, but it was still low.<sup>75</sup>

Based primarily on the data from these transient tests, threshold energy limits have been established for incipient fuel-rod failure and prompt fuel dispersal. The current criteria for satisfactory LWR performance for a rapid reactivity-initiated accident set an energy deposition of 280 cal/g as an upper limit to prevent prompt dispersal of the fuel.

A new experimental program on reactivity-initiated accidents was recently initiated in the Japanese Nuclear Safety Research Reactor (NSRR). The tests have all been performed in a water-filled instrumented capsule at a low ambient temperature and pressure. To date, PWR-type fuel rods have been tested at energy depositions of 40 to 334 cal/g UO<sub>2</sub>.<sup>76</sup>

The initial test series consisted of scoping tests with 2.6 and 10% enriched rods. The results were generally consistent with those of the TREAT and SPERT tests. The results of these and subsequent scoping tests were classified into the following significant stages of change:

Stage	Typical Energy Input (cal/g)
0. No visible change	<100
1. Mere discoloration (oxidation)	120
2. Distortion	150-220
3. Failure by crack in the cladding	255
4. Loss of structural integrity	240-310
5. Fragmentation	>310

The second test series employed reduced-diameter fuel pellets and standard-inside-diameter cladding to determine the effect of gap width on the behavior of cladding temperature and on the mode of fuel failure during transients. At the initial energy deposition of about 160 cal/g UO<sub>2</sub>, there were significant reductions in peak cladding temperatures, as compared with fuel rods with standard gaps. Above the 160 cal/g UO<sub>2</sub> base, the effect of cold gap size in reducing the peak cladding temperature became negligible.

The third test series employed waterlogged rods. In the test where both the annulus and the upper plenum were completely filled with water and the fill holes welded shut, the rod failed at an energy input of about 100 cal/g UO<sub>2</sub>, and, as the energy input continued to rise to the 141 cal/g UO<sub>2</sub> level, the fuel was powdered and dispersed within the capsule.

With each successive generation of LWRs, power generation and fuel burnup have increased, and the margin between the design limits and failure thresholds has potentially decreased. Additional testing with irradiated rods and clusters is desirable to improve the statistical basis of the criteria and to include the simulation of transients from hot-standby and power-operating conditions.

### 3.3.1.3 LOCA Tests

Two small-cluster tests have been conducted in the TREAT reactor to study cladding deformation and oxidation.<sup>77</sup> The results of these tests are not considered representative because of the nonuniform circumferential temperatures and the sharp axial power peaking. Current heatup tests in Germany and the LOCA blowdown tests at the Power Burst Facility are showing that cladding deformation and oxidation in reactor are moderate and consistent with results from out-of-reactor tests using electrical heating. Results from experiments on larger clusters or longer fuel rods in the PHEBUS (French), LOFT (United States), NRU (Canada), and ESSOR/SARA (Italy) programs will provide additional information on rod-to-rod interactions, three-dimensional effects, improved statistics, and commercial fuel experience.

## 3.3.2 RESEARCH PROGRAM

### 3.3.2.1 Integrated Tests of Fuel Models Under Accident Conditions

Tests are being conducted in the Power Burst Facility to improve understanding of the response of fuel rods under experimental conditions in which fuel is expected to fail. Similar tests will be performed at other facilities, such as the NRU reactor in Canada, the ESSOR/SARA test loop in Italy, and LOFT at the Idaho National Engineering Laboratory. These tests will permit a more detailed examination than does the current store of information of the mechanisms of

fuel-rod failure and possible failure propagation from one rod to another. With a more realistic view of the accident sequence, some of the present conservative assumptions in the analysis may be replaced by others that are closer to best estimates. The features of experiments that will be done on fuel rods in these reactors are not similar in all respects to those done in large reactors.

The use of test reactors is necessary because of the impracticality of exposing the whole core of a large reactor to the spectrum of experiments to be conducted and also because of the lack of provisions for making multiple measurements on the fuel rods. Each of the test reactors for which experiments are planned, or from which information is expected through international cooperation, has unique capabilities, but none can supply all of the needed information. The test reactors vary in the neutron spectrum (light or heavy-water moderation), enrichment requirements, bundle size (1 to 37 rods), and bundle length (0.9 to 4 meters).

#### 3.3.2.1.1 Power Burst Facility

In-Reactor Tests. The Power Burst Facility (PBF) is a water-cooled and water-moderated reactor, contained in an open-top steel vessel. It is operated for the U.S. Department of Energy and the NRC by EG&G Idaho, Inc.

The reactor core is designed for both steady-state and pulsed-mode operation. One to twenty-five test fuel elements with an active length no greater than 91 cm are fitted into a test train together with the necessary test instrumentation. The assembled test train is then fitted into a pressurizable heavy-walled metal cylinder 15.5 cm in diameter (the IPT). The IPT is mounted vertically and concentric to the vertical axis of the reactor core and the containing vessel.

The in-reactor tube has six to eight openings, permitting the use of up to 100 pairs of instrumentation test leads. Typical test instrumentation includes inlet and/or exit flow meters (up to five per test); absolute- and differential-pressure transducers for monitoring fluid and fuel-element plenum pressures; surface and internal thermocouples for monitoring fuel, cladding, plenum, and coolant temperatures; ultrasonic thermometers; linear variable differential transformers (deflection indicators); radiation-flux monitor wires and foils; and self-powered neutron detectors. Suitable instrumentation, signal-conditioning equipment, and data-accumulation and data-reduction equipment and services are available.

The PBF test program for the period through 1981 includes tests in each of the following areas: (a) power-cooling mismatch, with both unirradiated and pre-irradiated fuel rods (16 tests); (b) LOCA with both unirradiated and pre-irradiated fuel rods (10 tests); (c) flow blockage, with previously unirradiated fuel rods (3 tests); (d) reactivity-initiated accident, with both fresh and pre-irradiated fuel rods (12 tests); and (e) gap conductance (stored energy--7 tests).

The PBF test series may be described as follows:

- a. Power-Cooling-Mismatch Tests. These tests study the critical-heat-flux (CHF) and post-CHF behavior of single rods (four at a time) and nine-rod clusters under a variety of power and cooling conditions, in which CHF is achieved either by increasing the fuel-rod power at a steady coolant flow, or by decreasing the coolant flow at a steady fuel-rod power, or by simultaneously decreasing the coolant flow and increasing the fuel-rod power. (To date, only the specific combinations of final fuel-rod power level and final flow rate appear to be important.) These tests also study the effects of irradiation and burnup on the thermal-mechanical properties of fuel-rod components (particularly claddings).

Coolant flow, stored energy, and test-termination temperatures and post-CHF cladding deformation are among the test variables measured.

- b. LOCA Tests. These tests will study fuel-rod behavior, e.g., cladding deformation and oxidation of single-rod (four at a time) assemblies under blowdown conditions. Parameters to be varied include irradiation history and cold internal pressures. Sixteen-rod clusters will be tested under heatup conditions. Results will be correlated with those of out-of reactor tests.
- c. Flow-Blockage Tests. These tests will study fuel-rod behavior, e.g., cladding temperatures and geometric profiles of multiple-rod assemblies (25 rods) under flow blockages of 80 to 98%.

- d. Reactivity-Initiated Accident Tests. These tests will study the behavior of irradiated and unirradiated fuel rods under rod-drop and rod-ejection conditions. Independent rod tests, cluster tests, and model development/evaluation tests will be performed. The effects of irradiation, cluster size, coolant flow, and initial power level will be studied.
- e. Gap Conductance (Stored Energy) Tests. These tests study the gap conductance and stored energy of irradiated and unirradiated rods. Parameters measured include irradiation history, gap size, fill-gas pressure, and pellet densities. Power oscillation (transfer function) and integral K dt gap-conductance measurement methods are being compared.

Facility Modifications. The Power Burst Facility was originally designed to perform burst (RIA) tests. The requirements of the Fuel Behavior Program extended beyond burst testing into the area of the closely controlled thermal and hydraulic conditions required by the power-cooling mismatch, gap-conductance, irradiation effects, and flow-blockage experimental programs. The in-reactor loop has been modified and will be further modified to provide the improved performance demanded by these tests.

A major modification has just been completed to provide the coolant conditions expected in a LOCA. This modification provides fast-acting valves that permit the simulation of the depressurization and flow conditions of a LOCA.

Another major modification just completed is the addition of a new data acquisition and reduction system (DARS). This computer-controlled data system considerably extends the number of data channels that can be recorded, provides for higher data-sampling rates, and provides reduced data much sooner than did the previous system.

#### 3.3.2.1.2 In-Reactor Tests at Other Facilities

The Power Burst Facility is particularly useful for the in-reactor testing and performance evaluation of LWR fuels under abnormal operating conditions, but many in-reactor tests of LWR fuel behavior under abnormal operating conditions can be performed to good advantage in other reactor safety test facilities. Accordingly, the Fuel Behavior Research Branch has developed its in-reactor testing programs so that they either are or will be complementary to, or participate in, the reactor safety fuel behavior programs performed at a number of other facilities, including most of the following:

- The LOFT Facility at the Idaho National Engineering Laboratory
- Nuclear Safety Research Reactor (NSRR) in Japan
- FR-2 in the Federal Republic of Germany
- The Halden experimental boiling water reactor in Norway
- The ESSOR/SARA test loop in Italy
- PHEBUS in France
- The NRU reactor in Canada
- The BR-2 reactor in Belgium

Nuclear Safety Research Reactor (NSRR) RIA Tests. The NSRR reactor in Japan is fueled with uranium-zirconium hydride, moderated with zirconium hydride and water, and cooled with light water; it is an annular-core pulse reactor contained in an open-top swimming pool. The NSRR has an experimental hole 23 cm in diameter in the center of the core to accommodate experiments, either a capsule or a loop. The NSRR test programs through 1979 will concentrate on RIA tests with short (12- to 15-cm) fuel rods. Initial tests will be performed with the fuel rods placed in partially water filled capsules at ambient pressure. In the near future the water in the capsules will be heated and thus pressurized. A pressurized test loop is planned for use in the final phases of the NSRR RIA test program.

It is presently planned that the NSRR test program will concentrate on the parametric testing of short previously unirradiated single rods or very small clusters, in contrast with the Fuel Behavior Research Branch intent that later tests in the Power Burst Facility include a range of

cluster sizes of preirradiated fuel elements nominally 91 cm long. RIA capsule tests can be performed very rapidly in the NSRR, and it is anticipated that several hundred short rods will be tested.

FR-2 LOCA Fission-Heat Tests. The DK loop of the FR-2 reactor at Karlsruhe, in the Federal Republic of Germany, was designed for in-reactor simulated LOCA heatup testing of single fuel elements in superheated steam. The active core length is about 0.5 meter. Currently it is anticipated that by 1980 at least 40 fuel elements, both preirradiated and fresh, will have been tested. Test parameters include gap size, internal pressure, preirradiation to 35 Gwd/MTM, and heatup rate. In-reactor tests will include electrically heated rods and will be closely coordinated with an out-of-reactor test program. (In contrast with the FR-2 single-rod LOCA heatup tests, it is anticipated that the LOCA single-rod and 16-rod tests in the Power Burst Facility will include blowdown and refill.)

Halden Gap Conductance (Stored Energy), LOCA, and Dual-Purpose Tests. The Halden experimental facility in Norway is a heavy-water-moderated BWR. Many different fuel-behavior tests have been performed in the Halden reactor, including the gas-release and gap-conductance tests discussed elsewhere in this section. A new dual-purpose fuel-element test will be initiated<sup>92</sup> by the Halden group in 1978. A set of instrumented fuel rods will be irradiated in the Halden reactor for several months, during which time repeated gap-conductance measurements will be made. The instrumented fuel rods will then be transported to the Power Burst Facility. The original instrument sensors will be reconnected to signal-conditioning equipment and used to determine the gap-conductance values for the fuel rods when the latter are operated under equivalent test conditions in the Power Burst Facility. After completion of the gap-conductance measurements, the fuel rods will be used in RIA or LOCA tests at the Power Burst Facility.

NRU Fuel-Cluster Tests. The NRU reactor, located at the Atomic Energy Commission of Canada (AECL) facility at Chalk River, Ontario, is a 125-MW heavy-water-moderated, light-water-cooled test reactor. The NRC is negotiating a contract with AECL to conduct a series of LOCA heatup tests in the U-2 test loop. The facility will be capable of supporting a test program to study LOCA refill and reflood on clusters of 32 full-length commercially irradiated fuel rods. A program of five tests is being planned, starting in the Spring of 1979, with the series completed in the Fall of 1980.

ESSOR/SARA Tests. Since the ESSOR/SARA test loop in Italy has two-phase inlet capability, axial temperature profiles of coolant and fuel for long and short rods can be compared for single-facility cross correlation of length scaling relationships. When compared with PBF single-rod test results and 204-rod-bundle LOFT test results, the small-cluster (e.g., SARA) test results will provide an intermediate test of bundle-size scaling relationships. Although the cost for the test train will be higher than that for the PBF single-rod test trains, it should be appreciably lower than the cost of an instrumented LOFT fuel assembly. Thus this test train will generally represent a cost-effective compromise with respect to the quality of test data and cost per data point, although it certainly cannot totally supplant the larger cluster size of LOFT.

It is anticipated that the Fuel Behavior Research Branch will actively participate in the ESSOR/SARA test program.

LOFT LOCA Fuel Behavior Tests. The LOFT facility is a 55-MW pressurized water reactor at the Idaho National Engineering Laboratory. The typical LOFT central fuel assembly contains 204 fuel rods with an active length of 1.68 meters. The central fuel assemblies used in the initial tests will contain unpressurized fuel elements and will have limited fuel-element instrumentation, consisting primarily of selectively placed cladding thermocouples on a few fuel elements. It is expected that many of the later central fuel assemblies will contain a significant fraction of well-instrumented pressurized fuel elements. Instrumentation will include LVDTs (fuel-element elongation monitors), fuel centerline thermocouples, fuel-element plenum thermocouples, cladding thermocouples, and fuel-element gas plenum pressure transducers. Incorporation of test loop fission-product monitoring is also under consideration. Parameters to be evaluated include fuel and cladding temperatures, internal gas pressure, and axial and radial distortion of cladding.

### 3.3.3 Milestones

#### Power Burst Facility

- The data acquisition and reduction system has been installed and was started in August 1977.
- The LOCA modification was completed in December 1977.

- The first LOCA blowdown test and initial assessment of pretest prediction models were completed in January 1978.
- The first power-cooling-mismatch small-cluster test and initial assessment of pretest prediction models were completed in March 1978.
- The first reactivity-initiated accident single-rod test and initial assessment of pretest prediction models will be completed in August 1978.
- The first RIA failure-propagation cluster test and initial assessment of pretest prediction models will be completed in June 1981.
- The first flow-blockage, single-phase-inlet, small-cluster test and initial assessment of pretest prediction models will be completed in October 1981.
- Initial 4-year 40-test series and initial model cross-correlation studies will be completed in December 1981.

#### ESSOR/SARA Test Program

- The first in-reactor long-rod fuel-bundle flow-blockage tests and comparison of test results with pretest predictions will be completed in October 1980.
- The first in-reactor fuel-bundle flow-blockage test with two-phase inlet flow and short fuel rods, comparison of fuel-enthalpy and coolant-enthalpy profiles and fuel-rod length scaling relationships with those of the previous long-rod test, and assessment of scaling models will be completed in July 1981.
- The first in-reactor LOCA blowdown test with a long-rod fuel bundle and comparison of test results with pretest predictions will be completed in March 1981.
- The first in-reactor LOCA blowdown test with a preirradiated long-rod fuel bundle and comparison of test results with pretest predictions will be completed in December 1981.
- Assessment of models for the propagation of fuel-rod failure in LOCA and flow-blockage accidents, including beginning-of-life and end-of-life (preirradiated) conditions, will be completed in October 1982.

#### NRU Reactor

- The first in-reactor LOCA fission-heat test with a long-rod bundle and comparison of test results with pretest predictions will be completed in June 1980.
- The first in-reactor LOCA fission-heat test with long preirradiated fuel rods and comparison of test results with pretest predictions will be completed in December 1980.

### 3.4 FUEL BEHAVIOR COMPUTER CODES

#### 3.4.1 PRESENT STATUS

At the time that the Fuel Behavior Research program was beginning, available computer codes indicated substantial deficiencies in many areas with regard to both steady-state and transient phenomena. Since that time, significant new data from out-of-reactor and in-reactor research programs have contributed greatly to code development. A recent review<sup>78</sup> of six primarily steady-state codes has indeed revealed a much improved situation (as discussed below), but further development is needed, and the transient codes are not yet adequate because of the greater complexity of phenomena occurring during postulated accidents.

Fuel behavior codes must analyze the thermal, mechanical, and internal gas response of fuel-rod components with the goal of predicting rod condition and integrity. Modeling of thermal behavior during normal and accident conditions must include the surface heat transfer, heat transfer across the fuel-to-cladding gap, the thermal conductivity of fuel and cladding, the power generation distribution in the fuel, and the solution of the conduction equation. These aspects of the thermal calculations are listed approximately in order of their importance. Unfortunately, a listing of current ability to treat them accurately would be in the inverse order.

Modeling the mechanical response of fuel rods (or fuel and cladding structure) involves consideration of fuel-cladding mechanical interactions (FCMI); cladding creep, ballooning, and failure;

and fuel thermal expansion, swelling, densification, and creep. The phenomena that are important in steady-state operation (creep, swelling, etc.) significantly affect behavior during transients. The transient codes must therefore in some manner consider these phenomena, either by direct calculation or by linkage to a steady-state code. These response phenomena are, of course, coupled to one another as well as to the thermal behavior factors. One very important parameter that is difficult to calculate because of this strong coupling is the fuel-to-cladding gap width.

Modeling of the internal gas response is important for determining the loading that it applies to the cladding and for determining heat transfer across the fuel-to-cladding gap. The key modeling areas associated with these effects are axial gas flow, fission-gas release, plenum gas temperature, and voids and void temperature.

#### 3.4.1.1 Steady-State Codes

The principal publicly available steady-state fuel behavior codes and their current versions are FRAP-S3,<sup>79</sup> COMETHE-IIIJ,<sup>80</sup> BEHAVE-4,<sup>81</sup> LIFE-THERMAL-1,<sup>82</sup> GAPCON-THERMAL-3,<sup>83</sup> CYGRO-3,<sup>84</sup> and FMODEL.<sup>85</sup> FRAP-S and its transient counterpart FRAP-T<sup>86</sup> are being developed at the Idaho National Engineering Laboratory under the direction of the Fuel Behavior Research Branch.\* The other codes have various foreign and U.S. origins. A recent evaluation of these latter codes for potential use by the electric utility industry has concluded that COMETHE-IIIJ is the most versatile code for both thermal and structural analyses of Zircaloy-clad uranium dioxide fuel rods.<sup>78</sup> CYGRO-3 and FMODEL are reported to be much less suitable for use by the utility industry. FRAP-S2<sup>87</sup> was not included in this study, but independent assessment work<sup>88</sup> indicates that it compares favorably with the best of these codes.

The NRC recently decided to combine its development efforts on GAPCON-THERMAL and FRAP-S into a single steady-state code, FRAPCON, which is discussed in more detail in Section 3.4.2.

Information from the Halden IFA-431 experiment indicates that the uncertainty in temperature difference across the fuel-to-cladding gap remains relatively constant and less than 100 K in spite of the gap-conductance value.<sup>89</sup> Modeling of these rods with GAPCON-THERMAL has kept pace with this reduction of experimental uncertainty since predictions of fuel centerline temperature are within 10% of the data.

The ability of the computer codes to model the mechanical response of fuel rods varies much more than does their ability to model thermal response. In general, the deformation models are superior in the response regime prior to the onset of fuel-cladding mechanical interactions and at temperatures below the onset of fuel plasticity, 1800 to 2000 K (2780 to 3140°F). Comparisons between code and theory show uneven results, however. Some cases have been studied in which small positive permanent strains occurred by the end of life due to fuel-cladding contact, but in which the codes predicted significant negative tangential strains due to creep effects.<sup>78</sup> Other cases have been examined in which code predictions agree well with post-irradiation examination data.<sup>90</sup> An assessment study of FRAP-S2<sup>88</sup> found that the calculated axial fuel expansion is within 50% of the data but that stack shortening for rods accumulating burnup is underestimated by 20 to 80% for cases exhibiting fuel-cladding mechanical interactions in addition to fuel densification. A principal cause of these problems is that fuel plasticity is treated indirectly or not at all in these codes. The models affecting gap closure and mechanical interactions are also in need of improvement.

The various codes also differ in their ability to model internal gas response. Several of the codes (FRAP-S2 and BEHAVE-4) significantly overestimate the amount of gas release, typically by a factor of 2 or 3 in the range of highest release sensitivity, 1700 to 2700 K (2600 to 4400°F).<sup>78,88</sup> LIFE-THERMAL-1 has been found to give excellent gas-release predictions (within 10 to 50%), but its model will not permit a fission-gas burst.<sup>78</sup> COMETHE-IIIJ has had intermediate success in fission-gas-release prediction, although it has the most sophisticated gas-release model, accounting for the effects of grain size, porosity, bubble migration, and restructuring. The temperature-dependent-only gas-release models, such as those in FRAP, GAPCON, and BEHAVE, are best applied at higher burnups (25 Gwd/MTU), which are associated with the build-up of interconnected porosity. If a significant amount of fission-gas is released, internal pressure can be predicted to within approximately 20% of the data.<sup>88,90</sup>

COMETHE-IIIJ, being developed by BELGONUCLEAIRE and S. M. Stoller Corp., is relatively easy to use and provides good predictions of steady-state fuel temperature, equiaxed grain growth, cladding strain, and fission-gas release. Recent improvements in COMETHE include models for

\*FRAP-S = Fuel Rod Analysis Program--Steady State; FRAP-T = Fuel Rod Analysis Program--Transient.

Zircaloy anisotropy, fission-gas release, fuel grain growth, and ridging analysis. Fuel properties not treated include creep, melting, stored energy, and volumetric swelling and fuel relocation at temperatures below about 1675 K (2555°F). Cladding phenomena not treated include plasticity, corrosion, failure, and axial friction forces from fuel-cladding interactions.

BEHAVE-4 is being developed by Science Applications, Inc. and LIFE-THERMAL-1 by O'Donnell and Associates, Inc. Of the codes examined, BEHAVE-4 contains the most complete structural models, since it uses constitutive equations to describe the deformation mechanics of Zircaloy. One weakness of the code is its present inability to account for significant permanent cladding strains during power ramps with fuel-cladding interaction present. It does, however, rigorously solve a set of axial force-displacement equations and allows loads to act on the ends of the pellet stack. Both this code and LIFE-THERMAL-1 (which is a conversion of the fast-breeder-reactor code LIFE-III for LWR use) consider friction forces between fuel and cladding. The LIFE-THERMAL-1 code excelled the others in its fission-gas-release model, but its structural analysis includes only an isotropic stress-strain model and does not allow permanent cladding strain due to fuel-cladding interactions; furthermore, its gap-closure rate is considered to be too high.

Of the codes studied, GAPCON-THERMAL-2, developed at Battelle Pacific Northwest Laboratories, requires a relatively small amount of computer time. The code was largely restricted to thermal analyses of fuel rods; however, an elastic-plastic mechanical model has recently been added. An evaluation of its performance has not yet been completed. This code is unique among those considered in that it provides two sets of behavioral models, one designated "best estimate" and the other "conservative." Both sets of models presently provide conservatively high temperatures, possibly due to the methods used in the original code calibration. Again, the organization of the code allows for reasonably simple improvement.

The steady-state code FRAP-S was developed for use at the Idaho National Engineering Laboratory both as a normal-operation analysis tool and as the generator of burnup-dependent initial conditions required for the FRAP-T (transient) code. As stated earlier, it is being replaced by FRAPCON. FRAP-S/FRAPCON seeks to model all of the important phenomena involved in nonaccident situations during the life of LWR fuel rods. It iteratively calculates the interrelated effects of fuel and cladding temperature, rod internal pressure, fuel and cladding elastic and plastic deformation, release of fission-product gases, fuel swelling, cladding growth resulting from irradiation, cladding corrosion, and crud deposition, all as a function of time and specific power.

FRAP-S3 included a number of features generally not found in other codes, including failure prediction models, more sophisticated fuel-cladding interaction models, and a larger package of material properties. Frequent and independent assessment studies are also performed and published as a guide to code users and code developers. The most recent study<sup>91</sup> found that improvements incorporated since the previous version (including a new treatment of fuel pellet-relocation and related fuel thermal conductivity effects) have resulted in a more realistic description of fuel behavior under moderate operating conditions. The thermal and mechanical model development that is used generally relates to the response regimes associated with highest power rods as opposed to core-average rods. In addition to the modeling problems discussed above, FRAP-S3 now yields a standard fuel-temperature error of 198 K for unpressurized rods and 254 K for pressurized rods. These discrepancies approach the present experimental uncertainties, however, and are of the same magnitude as those experienced by the other codes. Research work described in Section 3.4.2 is intended to reduce these experimental uncertainties and to create a corresponding improvement in modeling capability.

#### 3.4.1.2 Codes for the Analysis of Transients

There are fewer LWR fuel behavior codes for accident analysis than there are codes for use in studies of normal steady-state operation. A transient code must be able to calculate the temperature increases and the accompanying time- and temperature-dependent processes expected during postulated occurrences such as LOCAs, power-cooling-mismatch accidents, reactivity-initiated accidents, and inlet flow blockages, as well as the processes occurring during normal operation. The code must also be general enough to treat expected asymmetries and all of the phenomena occurring up to and including fuel melt.

To meet these demands, the models for FRAP-T have been developed primarily from first-principles formulations so that they will apply over a wide range of response conditions and will not be limited by the range of available data. FRAP-T predicts the time dependence of many coupled variables at an arbitrary number of axial positions for any transient power history. Calculated are the fuel-rod temperature distribution, gap conductance, internal pressure, cladding strain,

time and location of cladding failure, cladding surface temperature (including surface heat transfer), and coolant conditions (including temperature, enthalpy, and quality). The primary input data required are the following: power history, descriptions of the fuel-rod cold state, the time-dependent conditions of the coolant surrounding the rod, the axial power profile, and code running requirements, including the mesh size, time step, and convergence criteria for pressures and temperatures. The results of either a steady-state or an earlier transient calculation may, of course, be stored on tape and read by FRAP-T to satisfy these input requirements. FRAP-S/FRAPCON or FRAP-T may be used for this purpose, and transient coolant conditions calculated<sup>93</sup> with RELAP-4 may be read from a tape. The output may optionally include plots of up to 20 variables as a function of time.

Assessment work indicates that the thermal models and predictions of FRAP-T3 are quite good.<sup>94</sup> Steady-state data comparisons indicate that predictions are within the data uncertainty and within 10% of the indicated temperatures for rods with a low fission-gas content. Agreement between measured and predicted flow and power conditions at the initiation of critical heat flux is also quite good for bundle geometries with uniform flow and small cold-wall effects.

Probabilities of low-temperature cladding failure during power increases are well represented by FRAP-T's FRAIL<sup>71</sup> subcode for hard gap-closure conditions that are not accompanied by a significant bulk fuel plasticity effect. Comparison of predicted transient cladding temperatures with data from the TREAT-LOCA tests shows excellent agreement, generally within 50 K.

The FRAP-T3 deformation and cladding-failure models are able to predict available out-of-reactor tube rupture data for isothermal tests at 550 to 700 K (530 to 800°F) to within the approximately 20% scatter in burst pressures. For heatup tests the burst temperature is underestimated by about 30%, with corresponding overestimates of rupture strain (since many tubes in the data sample burst in the low-strain-at-failure region associated with the alpha-beta phase transition). Deformation and failure models, which have been recently improved or developed but not yet entirely integrated into FRAP-T, should improve these predictions. The fuel-rod failure subcode FRAIL<sup>71</sup> now also contains failure models for cladding melt, oxide-layer wall thinning, eutectic melting, collapse, overstress failure, cumulative stress damage, and overstrain failure. Small-deformation behavior of rods should be better calculated by the recently improved FRACAS subcode,<sup>95</sup> which considers the simultaneous deformation of an elastic-plastic fuel pellet and an elastic-plastic cladding based on a finite-difference scheme. Large deformations are determined by the BALLOON subcode, which depends intimately on cladding material properties. Cladding material properties are continuously being reviewed.

The FRAP-T3 assessment study<sup>94</sup> shows that the models in greatest need of improvement relate to calculations of rod pressure, nonuniform gap closure, fuel deformation, and high-temperature cladding deformation. The calculation of fuel-rod pressure may be one of the final models to be successfully confirmed because of its dependence on knowledge of transient gas release and void volume. Nonuniform gap closure is another especially difficult modeling area. Some help in resolving these problems is being obtained by examinations of SSYST, a transient code being developed in Germany.<sup>96</sup> The primary differences between these two codes involve calculations of open-gap heat transfer, fuel and cladding deformations, and material properties. Although these codes agree in their predictions of the beginning and end of ballooning during a LOCA, the magnitudes of predicted deformations and potential flow blockages are in some disagreement and must be resolved.

### 3.4.2 RESEARCH PROGRAM

#### 3.4.2.1 Fuel Behavior Code Development

The development effort for fuel behavior codes involves concurrent work in three parallel and closely related areas: correlations for material properties, development of the steady-state fuel behavior code FRAP-S/FRAPCON, and development of the transient fuel behavior code FRAP-T. The material property correlations are used by both codes, and the two codes are linked so that the predictions of FRAP-S/FRAPCON can be used as initial conditions for a transient fuel behavior analysis. In developing these computer programs, the Branch has followed, as much as possible, the policy of modeling all of the phenomena that the completed code is expected to describe, even though many of the models are initially very simple. This makes it possible to conduct sensitivity analyses of the code predictions in relation to the constituent models and provides guidance as to the amount of development that a given model requires.

The magnitude of the code development and assessment task directed by the Branch may be appreciated by considering that fuel behavior codes must involve more than 40 important parameters in more than 20 key models that describe aspects of fuel and fuel-rod behavior. Substantial uncertainties may be associated with some of these models and parameters, and many of the models are

coupled to one another in a complex way. Added to this is the requirement that the code must describe a wide range of accident conditions on the basis of a limited number of tests. It is firmly believed, however, that by an iterative development-assessment process employing the experiments and analyses described in Section 3.3.2 the primary code assessment goals can be achieved.

#### 3.4.2.2 MATERIAL PROPERTY CORRELATIONS

The Branch has directed the development of a handbook of material properties for use in the analysis of LWR fuel-rod behavior.<sup>2</sup> This handbook is to be updated at yearly intervals with new and improved correlations for fuel, cladding, gas, and fuel-rod material properties. These correlations are programmed in a modular FORTRAN package named MATPRO for use in the current versions of steady-state and transient codes.

The approaches taken in this work have ranged from (a) a least-squares fit to available data using a polynomial or other function with little or no theoretical basis to (b) a semiempirical correlation employing an analytical expression suggested by theory with constants determined by comparison with data. The Branch has directed that the latter approach be followed in current and future work whenever possible. Estimates of the uncertainties in the correlations are now also being made to allow sensitivity studies with FRAP-S/FRAPCON and FRAP-T. Future modeling efforts will be concentrated largely in the areas of cladding deformation and oxidation, fuel swelling, densification, restructuring, and gas release and in the upgrading of correlations to include additional irradiation and burnup effects. The success of these tasks will depend principally on the data to be generated in the Fuel Behavior Experiment Program.

#### 3.4.2.3 Steady-State Code Development

As stated earlier, the NRC has decided to combine the steady-state codes FRAP-S and GAPCON into a new code, FRAPCON. This new code is being developed jointly by the Idaho National Engineering Laboratory (INEL) and Battelle Pacific Northwest Laboratories (BNPL) and will contain, as user options, both best estimate and evaluation models. It is designed to be more user-oriented through the use of such features as variable mesh dimensioning, dynamic storage allocation, and increased modularity.

FRAPCON will be released with three-part documentation: a description of the analytical models (including numerical solution procedures and user instructions), a description of the related material property models (on which the code heavily relies), and a description of the code assessment work (including comparisons of predictions with available data). An increasing emphasis is to be placed on the code assessment work and on sensitivity studies in order to judge the progress of code development and to evaluate the need for further data.

The analytical models<sup>79,87</sup> of the most recently developed steady-state code, FRAP-S3, have been documented and evaluated and have been sent to the National Energy Software Center. FRAP-S2 has been fully documented and is also at the National Energy Software Center.

#### 3.4.2.4 Transient Code Development

The transient code FRAP-T must describe the condition of the cladding--the first barrier to the release of fission products--during and after a postulated accident. Its primary purpose is to predict whether this barrier has been breached as well as the nature, location, and time of the failure and subsequent rod deformation. As may be imagined, more processes can take place during severe transients than during normal steady-state operation due to the possibilities of large temperature increases in relatively short periods of time and due to complex rod-to-rod interactions.

The final FRAP-T code will closely couple all thermal, mechanical, and hydraulic phenomena that may be involved in postulated accidents. Both single-rod and multirod behavior will be modeled in very general terms for all states of reactor life. The code will calculate fuel and cladding temperatures, rod deformations, and cladding-fuel-coolant reactions to the point where significant rod disintegration or failure propagation has occurred.

Future modeling efforts will increasingly focus on the problems of developing an efficient fuel code/thermal hydraulic code link. In addition, improved material properties for transient conditions must be incorporated as they become available. Coupling of FRAP-T to a multichannel code and the addition of failure-propagation models will be necessary for more realistic application to postulated LWR accidents. Rod failure or nonfailed-rod distortion may disturb core flow or deform adjacent rods in such a way that the failure or distortion mode is propagated across

the fuel bundle or core. Although the occurrence of such fuel failure propagation during a design-basis accident is only speculative, models are required in order to determine whether and when failure propagation might occur. The development of the multirod transient code will be substantially complicated by the asymmetries in fuel-rod behavior that can develop during accident excursions. Much of the cylindrical and circumferential symmetry that tends to apply to steady-state behavior is lost in severe transient situations. Under these conditions, rods tend to balloon or bow to one side, zirconium-water reactions start in patches, eutectic formation occurs at grid contact points, and so on.

FRAP-T3, the most recently completed version of the transient code, has been fully documented at the National Energy Software Center, including a description of the analytical models,<sup>86</sup> the material properties used,<sup>2,97</sup> and the code assessment study.<sup>94</sup> FRAP-T3 is passively linked to RELAP-4/ MOD-5<sup>93</sup> for optional utilization of RELAP-generated coolant conditions and may obtain initial fuel-rod conditions by reading the burnup-dependent results calculated by FRAP-S3.

#### 3.4.2.5 Fuel Code Assessment

The assessment of any code is necessarily linked to the development of models and experimental data generation. Because of the scarcity of data in some areas and the lack of accurate models in others, code assessment usually follows an iterative process; that is, as a result of comparisons between a body of data and a given version of a code, weaknesses may become apparent in certain models and new areas of experimental need may become clear. This results in new experiments being designed and conducted and new models being developed to replace those that lack sufficient sophistication. In order to accomplish a workable iterative development-assessment process, the following time sequence of steps has been established:

- a. Identification of all variables that may influence fuel-rod behavior and all phenomena that require modeling. Modeling requirements for each important time phase of postulated accidents should be tentatively identified, and outer bounds should be placed on accident ranges.
- b. Use of current computer codes and analytical techniques to perform sensitivity analyses. This task should clarify the relative influence of particular input variables, the relative importance of individual models, and the accident time periods during which each model is most influential; it should also help determine the measurement accuracies required. A comparison of code predictions with out-of-reactor and in-reactor data, and with available closed-form analytical solutions for simplified cases, will provide confidence that the code used gives physically reasonable results.
- c. Development of a test program to examine (1) significant phenomena over appropriate ranges as an aid in the development of models and (2) integrated systems for assessment purposes. Previous experimental work and the results of sensitivity analyses will guide the separate-effects test development. Experimental design should provide for study of the extreme ranges postulated for model applicability wherever possible.
- d. Determination of (1) the data uncertainty due to both random and systematic errors and (2) the model prediction uncertainty. The contribution of individual models to any disagreements must be ascertained, and areas needing improvement must be identified.
- e. Iteration. The assessment process requires models to guide experiment design and experiments to guide model development. Key data needs that are identified in step (d) should be satisfied with out-of-reactor tests when possible and otherwise in newly designed reactor experiments.
- f. Monitoring of the differences between experiment and code predictions as a function of development time to evaluate the development and assessment process.

The comparison of data with corresponding code predictions is the key step in model evaluation, bearing directly on the need for, and form of, further testing. There are two equally important aspects of this activity: (a) the choice of the most appropriate experiment and data time period for the relevant goal, and (b) the use of acceptable statistical criteria for agreement. The importance of the former point cannot be overemphasized. Various coupled models will be involved in different combinations and with differing degrees of importance in every time period of every experiment. Evaluation of an individual model will require particular reliance on the data sets during which that model is most influential, plus iterative-comparison calculations with other data.

The objective of the fuel code assessment process is to provide statistically stated uncertainty bounds to the code predictions of the response of key fuel-rod parameters.

An extensive data base is necessary to accomplish these goals. The sources of this data are the integral test programs described in Section 3.3.2.1 and in benchmark experiments conducted by other organizations and foreign countries. An extensive effort has been undertaken to identify and follow those research programs that are expected to supply the needed data, by means of personnel assignments and information exchanges.

### 3.4.3 Milestones

#### Material Property Correlations

- The MATPRO-9 handbook describing 12 correlations for fuel properties, 17 for cladding properties, and 4 for gas and fuel-rod material properties was published in January 1977.
- The programming of MATPRO-10 for FRAP-T4, including (a) a model for high-temperature oxidation and its effect on cladding deformation, (b) an improved cladding-deformation model that incorporates high-strain-rate and anisotropy effects, and (c) ductile and low-strain failure criteria was completed and published in April 1978.
- The MATPRO-11 handbook will be published in February 1979.
- The programming of MATPRO-11 for FRAP-T5, including (a) cladding embrittlement criteria for LOCA reflood and (b) a model for the effect of high-temperature annealing (of cold work and irradiation damage) on cladding plastic deformation properties, was completed in June 1978.
- The programming of MATPRO-12 for FRAPCON-2, including (a) an improved cladding creep model, (b) a fill-gas absorption model, and (c) an improved gap-conductance model, will be completed in October 1979.
- The MATPRO-12 handbook will be published in February 1980.
- The programming of MATPRO for FRAP-T6, including (a) the final oxygen embrittlement model including effects during PCM loading, (b) the final model for irradiation annealing and transient temperature bursts from out-of-reactor data, and (c) a uranium dioxide-Zircaloy reaction model, will be completed in 1980.

#### Steady-State Codes

- FRAP-S2 code and documentation were sent to the National Energy Software Center in December 1976. Code includes models for fuel-cladding mechanical interaction and material properties from MATPRO-7.
- FRAP-S3 code and documentation were sent to the National Energy Software Center in July 1978. Code includes (a) preliminary uncertainty-analysis procedures; (b) failure models for overstress, overstrain, oxidation, ballooning, collapse, melting, and flow blockage; and (c) MATPRO material properties.
- FRAPCON-1 code and documentation will be sent to the National Energy Software Center in September 1978. Code to include (a) dynamic storage allocation and variable mesh, (b) deformable-pellet model with creep, (c) rod-bowing model, (d) advanced GRASS\* code for fuel swelling, (e) final uncertainty-analysis procedures, (f) a failure model for chemical and irradiation effects, and (g) MATPRO-11 material properties.
- FRAPCON-2 code and documentation will be sent to the National Energy Software Center in October 1979. Code to include (a) optimized fuel-cladding interaction models, (b) final GRASS code for fuel swelling, (c) a failure model for fatigue, and (d) MATPRO-12 material properties.

#### Transient Codes

- FRAP-T3 code and documentation were sent to the National Energy Software Center in April 1977. Code includes (a) a two-dimensional heat-conduction model; (b) improved gas-flow and cladding-ballooning models; (c) failure models for overstress, overstrain, eutectic melting, and oxidation; and (d) MATPRO-8 material properties.

---

\*Developed by the Argonne National Laboratory.

- FRAP-T4 code and documentation were sent to the National Energy Software Center in July 1978. Code includes (a) a preliminary multirod thermal-hydraulic calculation; (b) a model with a central void extending for part of its length; (c) failure models for ballooning, collapse, melting, and flow blockage; (d) preliminary uncertainty-analysis procedures; and (e) MATPRO-10 material properties.
- FRAP-T5 code and documentation will be sent to the National Energy Software Center in February 1979. Code to include (a) an improved multirod capability by an active link to a multirod thermal-hydraulic code, (b) thermal models for core reflood, (c) an improved gas-flow model based on Halden and German data, (d) failure models for chemical and irradiation effects, (e) linkage capability with RELAP-4/MOD-6, and (f) MATPRO-12 material properties.
- FRAP-T6 code and documentation will be sent to the National Energy Software Center in December 1979. Code to include (a) an active link to an improved multirod thermal-hydraulic code, (b) the final FRAP-T/ GRASS code integration, (c) optimization of subcodes for failure prediction and for fuel-cladding mechanical interaction, (d) improved bundle-deformation models from ORNL data, and (e) MATPRO-14 material properties.

#### Independent Code Assessment

- Independent assessment of FRAP-S3 and FRAP-T3 was completed in 1977.
- First report on data bank for code assessment was issued in January 1978.
- Independent assessment of FRAP-T4 was completed in July 1978.
- Independent assessment of FRAPCON-1 to be completed in February 1979.
- Report on data bank to be issued in January 1979.
- First report on uncertainty of code application to commercial reactors to be issued in December 1980.
- Independent assessment of FRAP-T5 and T6 to be completed in July 1979 and 1980, respectively.
- Independent assessment of FRAPCON-2 and FRAPCON-3 (if necessary) to be completed by February 1980 and 1981, respectively.

### 3.5 FUEL MELTDOWN AND FISSION-PRODUCT BEHAVIOR

#### 3.5.1 PRESENT STATUS

##### 3.5.1.1 Fuel Meltdown

There is a very small probability of failure of the engineered safety features provided to prevent the melting of the fuel assemblies in the event of a LOCA or a severe transient. The Reactor Safety Study<sup>98</sup> has reported that the probability of a core meltdown in present-generation reactors is somewhat greater than was previously believed ( $\sim 10^{-5}$  per reactor year) but that its consequences would not be as great. In order to arrive at this conclusion, it was necessary to evaluate the physical processes that might occur during meltdown and to use engineering judgment in establishing realistic assumptions necessary for the analyses. Uncertainties in some of these assumptions made it desirable to reexamine key physical phenomena more closely. A review of available experimental data<sup>99</sup> concluded that in general a sizable body of useful data exists, but the experimental conditions are usually such that the results are not always directly applicable to the meltdown case. The report also suggested that sensitivity studies using state-of-the-art models of meltdown should be conducted to determine the importance of physical phenomena in relation to the overall consequences of the postulated meltdown accident.

As a result, programs have been instituted to examine fission-product behavior during meltdown, natural convection in molten pools, interactions between molten core materials and concrete, steam explosions, and the effects of these phenomena on meltdown probabilities and consequences. Close cooperation with a similarly oriented program in the Federal Republic of Germany has also been instituted. Some of the key findings to date include the following:

- Elevated system pressures are effective in reducing the probability of vapor explosion.<sup>100</sup>
- A relatively long time is required to establish steady-state single-phase convection in molten pools at high Rayleigh numbers.<sup>101</sup>
- Concretes with high and low carbonate contents behave in a qualitatively similar manner on contact with molten core materials.<sup>102</sup>
- Concrete penetration is thermally dominated (1 cm/min is a nominal value).<sup>102</sup>
- Steel in the melt will chemically reduce decomposition gases to carbon monoxide and hydrogen, which will then burn on contact with air.<sup>102</sup>
- Single-phase natural convection is not a dominant heat-transfer mechanism during core-concrete interactions.<sup>102</sup>
- A mechanistic model of core-concrete interactions has been developed.<sup>103</sup>

Significant progress has been made in identifying and characterizing these phenomena, but several key issues remain to be resolved. These include:

- Directional partitioning of heat flux during interactions between concrete and molten core materials.
- Probabilities associated with steam explosions.
- Necessary scale of experiments.
- Prediction of containment failure modes.
- Leakage of radioactivity from containment.

### 3.5.1.2 Fission-Product Release and Transport

It is necessary to know the radiological consequences of meltdown and other less severe accidents. Accurate estimates of fission-product inventory in a reactor core can be made from knowledge of the operating history. However, the amount and form of radionuclides released during a postulated accident can only be inferred by extrapolation of limited experimental data and application of simplified analytical models. Experimental investigations have been conducted primarily out of reactor on small samples (1 to 100 g) of relatively low burnup fuel (trace to 4 Gwd/MTM) with rare instances of higher burnup (up to 20 Gwd/MTM). Current practice is to make what are judged to be conservative assumptions<sup>104,105</sup> regarding fission-product release from the fuel and transport through the various barriers without detailed consideration of the mechanisms of release and transport. Greater emphasis has recently been placed on instrumenting commercial and test reactors so that releases of activity from the fuel can be monitored with greater reliability and accuracy. Out-of-reactor experiments<sup>106,107</sup> using irradiated fuel under more controlled conditions are elucidating the mechanisms of release and transport. In-reactor experiments monitoring releases of activity during normal operating conditions as functions of power and of defect size for rods intentionally made defective are under way in Japan and France. The development of mechanistic models to trace the path of fission products once they leave the fuel is also in progress.<sup>108</sup>

Considerable effort has been expended to examine the behavior of airborne fission products in the containment. Scaled facilities having up to slightly over 1% of the volume of the containment in a typical pressurized water reactor have been employed to examine atmospheric conditions, airborne concentrations, physical and chemical states of various species, mechanisms of fission-product removal from the airborne state, and thermal effects. Particular attention has been paid to iodine, and one outcome of these investigations has been the establishment of Regulatory Guides<sup>104</sup> specifying the partition of available iodine in the containment into discrete physical forms. Elemental vapor and chemically active particulate forms of iodine are readily removed from the air by chemical sprays and filtering systems. Methyl iodide, other organic iodides, and possibly hypoiodous acid have been identified as persistent airborne species but are conservatively believed to make up no more than 4% of the total iodine released into the containment. Fission-product and fuel aerosols are effectively removed from the containment space by agglomeration and gravitational settling as well as engineered safety features. Analytical models have been developed to predict the airborne concentration of fission products as a function of time for a given set of input data, including fission-product concentration, particle-size distribution, containment atmospheric conditions, and geometry.<sup>109</sup>

### 3.5.2 RESEARCH PROGRAM

The objective of this program is to provide the information needed for a mechanistic assessment of the consequences of a broad range of postulated accidents, including fuel meltdown.

#### 3.5.2.1 Fuel Meltdown

The overall objective of the meltdown research is to identify, characterize, and quantify physical phenomena postulated to occur during meltdown accidents. The primary goal will be accomplished when all significant safety-related phenomena are understood well enough to be modeled reasonably within conservative bounds.

Topics of current investigations include steam explosions, molten-fuel interactions, heat transfer, fission-product behavior, and accident analysis. These efforts are coordinated with a similar program in the Federal Republic of Germany to optimize the use of research resources.

##### 3.5.2.1.1 Steam Explosion

It is likely that, during a meltdown, molten core materials would come in contact with water. To assess the outcome of such an interaction, a probability of a steam explosion is assigned, based on the body of existing experimental data and industrial experience, most of which involves materials other than those of interest here. Experiments are being conducted to define more clearly the physical conditions required to initiate and to propagate explosive interactions between water and various molten LWR core materials and to estimate the energy release associated with such interactions. A concurrent analytical effort is directed at predicting more accurately the consequences of such explosions in terms of the potentially destructive pressure pulses generated.

##### 3.5.2.1.2 Molten-Fuel Interactions

Uncertainties in the physical and chemical behavior of the molten core have made it difficult to formulate detailed analytical models of the meltdown process. In particular, the nature of the interaction between molten core materials and the concrete in the base of the containment is not well understood. Experiments are being conducted with typical molten materials at high temperatures and various forms of concrete in order to define the key physical and chemical phenomena involved in the interaction. Scaling effects will be investigated.

Additional examinations of the behavior of the core in the early stages of meltdown may also be necessary to complement studies being performed in the Federal Republic of Germany.

##### 3.5.2.1.3 Heat Transfer in Molten Pools

Experiments are being conducted to generate correlations describing the partitioning of heat fluxes from molten pools that generate their own heat. Various geometries and thermal boundary conditions are investigated.

##### 3.5.2.1.4 Meltdown Accident Analysis

Analyzing an event as complex as a core meltdown accident requires the linkage of several codes, each of which describes a particular phenomenon or phase of the accident. The overall code can then be used to analyze the interdependencies of phenomena and to determine how sensitive the consequences of the accident are to uncertainties in those phenomena. A program has been implemented to perform this task, its objective being to establish priorities for further research.

#### 3.5.2.2 Fission-Product Release and Transport

An experimental program aimed at providing information useful in safety analyses has been initiated. Emphasis is placed on quantifying the release of iodine, cesium, and other semi-volatile fission products from irradiated rods under simulated accident conditions. A supportive analytical program dealing with the development and testing of models for fission-product migration from a failed fuel rod into the containment vessel is also under way. The improved model should include provisions for various mass flow paths; fission-product source terms, chemical states, and particle-size distributions; and physical mechanisms for the natural removal of airborne activity. Plans are under consideration for the construction of a test facility to evaluate the transport model.

### 3.5.3 Milestones

#### Steam Explosion

- Small-scale triggering experiments and definition of the effects of scaling completed in September 1978.
- Initial criteria for temperature and system pressure limits necessary to inhibit explosions confirmed by September 1978.
- Larger scale experiments have been initiated.
- Initial analysis of containment failure probability to be completed in March 1979.
- Model of thermal to mechanical energy conversion to be confirmed in December 1979.

#### Molten-Fuel Interactions

- Large-scale steel-concrete scoping tests completed in July 1976.
- Preliminary model of core-concrete interaction developed in October 1976.
- Small-scale transient-heating experiments completed and effects of scaling defined in October 1977.
- Small- and large-scale sustained-heating experiments to be completed in December 1978.
- Improved model of core-concrete interaction to be completed in November 1978.
- Assessment of improved model to be completed in March 1979.

#### Heat Transfer in Molten Pools

- Correlations for steady-state and transient natural convection in a horizontal layer completed in June 1976.
- Correlations for steady-state and transient natural convection in a hemispherical segment completed in August 1977.
- Complete correlations for steady-state and transient two-fluid horizontal layer completed in July 1978.
- Complete correlations for steady-state gas-driven convection to be completed in September 1978.

#### Meltdown Accident Analysis

- Linkage of phenomenological models into overall meltdown accident code completed in September 1977.
- First round of parametric analyses completed in March 1978.
- Justification for research priorities to be completed in December 1978.

#### Fission-Product Release and Transport

- Experiments to determine fission-product release from simulated irradiated rods completed.
- Preliminary fission-product transport model for controlled LOCA completed.
- Correlation of fission-product release from irradiated fuel during controlled LOCA completed in November 1977.
- Fission-product transport model (TRAP) for meltdown accident completed in April 1978.
- Correlation of fission-product source term during meltdown to be completed in September 1979.

#### REFERENCES\*

1. U.S. Atomic Energy Commission, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, Docket No. RM-50-1, December 1973. Available for inspection and copying for a fee in the NRC Public Document Room.
2. P. E. MacDonald and L. B. Thompson, eds., MATPRO-Version 09, A Handbook of Materials Properties for Use in the Analysis of Light Water Fuel Rod Behavior, USNRC Report TREE-NUREG-1005, December 1976.
3. American Society for Testing and Materials, "Standard Specification for Wrought Zirconium and Zirconium Alloy Seamless and Welded Tubes for Nuclear Service," ASTM B 353-77, December 1977. Available in public technical libraries.
4. H. Maki, M. Ooyama, T. Okubo, and Y. Mishima, "Effects of Texture and Annealing Temperature on Plastic Deformation Behavior of Zircaloy-2 Fuel Cladding Tube," British Nuclear Engineering Society, International Conference on Nuclear Fuel Performance, October 1973, paper 62.
5. G. Dressler, K.-H. Matucha, and P. Wincierz, "Experimental and Theoretical Analysis of Anisotropic Mechanical Properties of Zircaloy Fuel Element Claddings," Second International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, 10-14 September 1973, Vol. 1, Part C, paper C2/2.
6. R. H. Chapman, Characterization of Zircaloy-4 Tubing Procured for Fuel Cladding Research Programs, USNRC Report ORNL/NUREG/TM-29, July 1976, p. 39.
7. American Society for Testing and Materials, "Recommended Procedure for Determination of Hydride Orientation in Zirconium Alloy Tubing," Annex A2 to "Standard Specification for Wrought Zirconium and Zirconium Alloy Seamless and Welded Tubes for Nuclear Service," ASTM B 353-77, March 1977. Available in public technical libraries.
8. H. M. Chung, A. M. Garde, and T. F. Kassner, "Mechanical Properties of Zircaloy Containing Oxygen," Light-Water-Reactor Safety Research Program: Quarterly Progress Report, July-September 1976, USNRC Report ANL-76-121, December 1976.
9. R. H. Chapman, Multirod Burst Test Program, Quarterly Progress Report for October-December 1976, USNRC Report ORNL/NUREG/TM-95, April 1977.
10. J. M. Frenkel and M. Weisz, "Effect of the Annealing Temperature on the Creep Strength of Cold-Worked Zircaloy-4 Cladding," in Symposium on Zirconium in Nuclear Applications, American Society for Testing and Materials, Philadelphia, Pa., ASTM STP 551, August 1974, pp. 140144. Available in public technical libraries.
11. D. S. Wood, "High Deformation Creep Behavior of 0.6-in.-Diameter Zirconium Alloy Tubes Under Irradiation," in Symposium on Zirconium in Nuclear Applications, American Society for Testing and Materials, Philadelphia, Pa., ASTM STP 551, August 1974, pp. 274-291. Available in public technical libraries.
12. D. O. Hobson, Quarterly Progress Report on the Creepdown and Collapse of Zircaloy Fuel Cladding, Program Sponsored by the NRC Division of Reactor Safety Research for April-June 1976, USNRC Report ORNL/NUREG/TM-52, October 1976.
13. D. O. Hobson, Zircaloy Fuel Cladding Collapse Test Plan, USNRC Report ORNL/NUREG/TM-45, August 1976.
14. M. Ichikawa, K. Joon, and K. D. Knudsen, "In-Pile Measurements of Fuel Rod Deformation, and Verification of the Finite Element Model FEMAXI," in Proceedings of Specialist Meeting on the Behaviour of Water Reactor Fuel Elements under Accident Conditions, Spätind, Norway, 13-16 September 1976, Part 2, CSNI Report No. 13, Committee on the Safety of Nuclear Installations, OECD Nuclear Energy Agency. Available in the NRC Public Document Room in file for Report NUREG-0006.

\*Unless otherwise indicated, USNRC technical reports and reports by the national laboratories are available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

15. H. S. Rosenbaum, "The Interaction of Iodine with Zircaloy-2," Electrochem. Technol., 4, 153-156 (1966). Available in public technical libraries.
16. D. Cubicciotti et al., The Nature of Fission-Product Deposits Inside Light-Water-Reactor Fuel Rods, EPRI Project No. RP 455-1, Stanford Research Institute, Menlo Park, California, November 1976. Available from the Electric Power Research Institute, Research Reports Center, P.O. Box 10090, Palo Alto, California 94303.
17. J. C. Wood, B. A. Surette, I. M. London, and J. Baird, "Environmentally Induced Fracture of Zircaloy by Iodine and Cesium: The Effects of Strain Rate, Localized Stresses, and Temperature," J. Nucl. Materials, 57, 155-179 (1975). Available in public technical libraries.
18. J. H. Davies et al., Irradiation Tests To Characterize the PCI Failure Mechanism, General Electric Company, NEDO-21551, February 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
19. G. P. Sabol and S. G. McDonald, "Strain-Rate Dependent Mechanical Response of Zircaloy-4," J. Nucl. Materials, 64, 223-227 (1977). Available in public technical libraries.
20. W. Evans and G. W. Parry, "The Deformation Behavior of Zircaloy-2 Containing Directionally Oriented Zirconium Hydride Precipitates," in Symposium on Zirconium and Its Alloys, the Electrochemical Society, Inc., 1966.
21. J. E. Irvin, "Effects of Irradiation and Environment on the Mechanical Properties and Hydrogen Pickup of Zircaloy," in Zirconium and Its Alloys, the Electrochemical Society, Inc., 1966.
22. A. L. Bement, in Radiation Effects, Metal Society Conference, Gordon and Breach, 1965, pp. 671-725.
23. A. A. Bauer et al., Evaluating Strength and Ductility of Irradiated Zircaloy, Quarterly Progress Reports for April-June 1976, USNRC Report BMI-NUREG-1956; July-September 1976, BMI-NUREG-1961; October-December 1976, BMI-NUREG-1967; January-March 1977, BMI-NUREG-1971; April-June 1977, BMI-NUREG-1976.
24. C. C. Dollins and F. A. Nichols, "Mechanisms of Irradiation Creep in Zirconium-Base Alloys," in Zirconium in Nuclear Applications, American Society for Testing and Materials, Philadelphia, Pa., ASTM STP 551, 1974 pp. 229-248. Available in public technical libraries.
25. E. F. Ibrahim, "In-Reactor Creep of Zr-2.5Nb Tubes at 570 K," in Zirconium in Nuclear Applications, American Society for Testing and Materials, Philadelphia, Pa., ASTM STP 551, 1974, pp. 249-262. Available in public technical libraries.
26. E. Rolstad and K. Svanholm, "Overpower-to-Failure Experiment on a High Burnup Fuel Rod," British Nuclear Engineering Society, International Conference on Nuclear Fuel Performance, October 1973, paper 77.
27. N. Fuhrman et al., Evaluation of Fuel Rod Performance in Maine Yankee Core I, -- Final Report, Task C, EPRI NP-218, November 1976. Available from the Electric Power Research Institute, Research Reports Center, P.O. Box 10090, Palo Alto, California 94303.
28. J. T. A. Roberts, "The In-Core Power Ramp Experiment in Oskarshamn 1," EPRI Project No. RP-829, presented at the EPRI Fuel Performance Program Contractor's Overview Meeting, Dallas, Texas, April 5-6, 1977. Available in the NRC Public Document Room in file for Report NUREG-0006.
29. G. R. Thomas, "Studsвик Inter-ramp Project," EPRI Project No. RP-507, presented at the EPRI Fuel Performance Program Contractor's Overview Meeting, Dallas, Texas, April 5-6, 1977. Available in the NRC Public Document Room in file for Report NUREG-0006.
30. S. Aas, K. D. Olshausen, and K. Videm, "Fuel Failures Caused by Overpower Ramps," British Nuclear Engineering Society, International Conference on Nuclear Fuel Performance, October 1973, paper 55. Available in the NRC Public Document Room in file for Report NUREG-0006.
31. P. Knudsen and N. Kjaer-Pedersen, "Performing Analysis of PWR Power Ramp Tests," ASME Winter Annual Meeting, Houston, Texas, November 30-December 4, 1975. Available in the NRC Public Document Room in file for Report NUREG-0006.

32. R. R. Biederman, R. G. Ballinger, and W. G. Dobson, A Study of Zircaloy 4-Steam Oxidation Reaction Kinetics, Final Report, EPRI NP-225, September 1976. Available from the Electric Power Research Institute, Research Reports Center, P.O. Box 10090, Palo Alto, California 94303.
33. J. V. Cathcart, Quarterly Progress Report on the Zirconium Metal-Water Oxidation Kinetics Program Sponsored by the NRC Division of Reactor Safety Research for October-December 1976, USNRC Report ORNL/NUREG/TM-87, February 1977.
34. S. Kawasaki, T. Furuta, and M. Hashimoto, Reaction of Zircaloy Cladding with Steam under Simulated Loss-of-Coolant Accident Conditions, Japan Atomic Energy Research Institute, JAERI-M-6181, July 1975. Available in the NRC Public Document Room in file for Report NUREG-0006.
35. S. Leistikow, H. V. Berg, and D. Jennert, "Comparative Studies of Zircaloy 4 High Temperature Steam Oxidation under Isothermal and Temperature-Transient Conditions," in Proceedings of Specialist Meeting on the Behaviour of Water Reactor Fuel Elements Under Accident Conditions, Spätind, Norway, September 13-16, 1976, Part 1, CSNI Report No. 13, Committee on the Safety of Nuclear Installations, OECD Nuclear Energy Agency. Available in the NRC Public Document Room in file for Report NUREG-0006.
36. L. Baker and L. C. Just, Studies of Metal-Water Reactions at High Temperatures; III. Experimental and Theoretical Studies of the Zirconium-Water Reaction, Argonne National Laboratory, ANL-6548, May 1962.
37. A. W. Lemmon, Jr., Studies Relating to the Reaction Between Zirconium and Water at High Temperatures, Battelle Columbus Laboratories, BMI-1154, January 1957.
38. H. H. Klepfer, reported in G. J. Scatina, Fuel Cladding Embrittlement during a Loss-of-Coolant Accident, General Electric Company, NEDO-10674, October 1972. Available for inspection and copying for a fee in the NRC Public Document Room.
39. J. V. Cathcart et al., Zirconium Metal-Water Oxidation Kinetics IV: Reaction Rate Studies, USNRC Report ORNL/NUREG-17, August 1977.
40. S. Levine, Research Information Letter 9, "High Temperature Oxidation of Zircaloy Fuel Cladding in Steam," Memorandum to B. C. Rusche, Director ONRR/NRC, March 14, 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
41. R. A. Perkins, Zirconium Metal-Water Oxidation Kinetics. II. Oxygen-18 Diffusion in  $\beta$ -Zircaloy, USNRC Report ORNL/NUREG/TM-19, July 1976.
42. M. W. Mallett, W. M. Albrecht, and P. R. Wilson, "The Diffusion of Oxygen in Alpha and Beta Zircaloy-2 and Zircaloy-3 at High Temperatures," J. Electrochem. Soc., 106, 181-184 (1959). Available in public technical libraries.
43. J. V. Cathcart and E. T. Rose, "Metallographic Examination of PS-18," in R. H. Chapman, Multirod Burst Test Program Quarterly Progress Report for April-June 1976, USNRC Report ORNL/NUREG/TM-74, January 1977, pp. 30-43.
44. M. L. Picklesimer, "Deformation, Creep, and Fracture in Alpha-Zirconium Alloys," in Symposium on Zirconium and Its Alloys, the Electrochemical Society, Inc., 1966, pp. 289-300.
45. S. Malang, SIMTRAN I - A Computer Code for the Simultaneous Calculation of Oxygen Distributions and Temperature Profiles in Zircaloy During Exposure to High-Temperature Oxidizing Environments, Oak Ridge National Laboratory, ORNL-5083, November 1975.
46. R. E. Pawel, J. V. Cathcart, J. J. Campbell, and S. J. Jury, Zirconium Metal-Water Oxidation Kinetics V: Oxidation of Zircaloy in High Pressure Steam, USNRC Report ORNL/NUREG-31, December 1977.
47. R. H. Chapman, Multirod Burst Test Program, Quick-Look Report, MRBT No. 1 4 x 4 Bundle Burst Test, Oak Ridge National Laboratory, ORNL/MRBT-2, September 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
48. R. H. Chapman, Multirod Burst Test Program, Quick-Look Report, MRBT No. 2 4 x 4 Bundle Burst Test, Oak Ridge National Laboratory, ORNL/MRBT-3, November 1977. Available for inspection and copying for a fee in the NRC Public Document Room.

49. A. Garde, H. Chung, and T. F. Kassner, Uniaxial Tensile Properties of Zircaloy Containing Oxygen: Summary Report, USNRC Report, ANL-77-30, June 1977.
50. H. Chung, A. Garde, and T. F. Kassner, Deformation Characteristics of Zircaloy Cladding in Steam and Vacuum under Transient Heating Conditions: Summary Report, ANL-77-31.
51. A. Garde, H. Chung, and T. F. Kassner, "Micrograin Superplasticity in Zircaloy," Acta Metallurgica, 26, 153-166 (1978). Available in public technical libraries.
52. E. I. H. Lin, "Analysis of Ballooning Deformation of an Internally Pressurized Thin Wall Tube During Fast Thermal Transients," in Proc. SMIRT Conference, San Francisco, Calif., August 1977.
53. E. I. H. Lin, "Plastic Instability Criteria for Necking of Bars and Ballooning of Tubes," in Proc. SMIRT Conference, San Francisco, Calif., August 1977.
54. R. G. Bellamy and J. B. Rich, "Grain Boundary Gas Release and Swelling in High-Burnup Uranium Dioxide," J. Nucl. Materials, 33, 64-76 (1969). Available in public technical libraries.
55. J. Rest, M. G. Seitz, L. R. Kelman, and S. M. Gehl, "Development and Experimental Verification of SST: A Steady-State and Transient Fuel Response and Fission Product Release Code," in Proceedings of Specialist Meeting on the Behaviour of Water Reactor Fuel Elements under Accident Conditions, Spätind, Norway, 13-16, September 13-16, 1976, Part 1, CSNI Report No. 13, Committee on the Safety of Nuclear Installations, OECD Nuclear Energy Agency. Available in the NRC Public Document Room in file for Report NUREG-0006.
56. U.S. Nuclear Regulatory Commission, The Role of Fission Gas Release in Reactor Licensing, USNRC Report NUREG-75/077, November 1975.
57. D. W. Brite et al., EEI/EPRI Fuel Densification Project, Final Report, EPRI-131, June 1975. Available from the Electric Power Research Institute, Research Report Center, P.O. Box 10090, Palo Alto, California 94303.
58. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.126, "An Acceptable Model and Related Statistical Methods for Analysis of Fuel Densification", March 1977.
59. American Nuclear Society, ANS Standard 5.1, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors", Hinsdale, Ill., October 1971 (revised 1973).
60. B. I. Spinrad, Evaluation of Fission Product Afterheat, Annual Report July 1, 1975 - September 20, 1976, Oregon State University, USNRC Report NUREG-0018-4, January 1977.
61. J. L. Yarnell and P. J. Bendt, Decay Heat from Products of U-235 Thermal Fission by Fast Response Boil-Off Calorimetry, USNRC Report LA-NUREG-6713, September 1977.
62. J. K. Dickens et al., Fission-Product Energy Release for Times Following Thermal-Neutron Fission of  $^{235}\text{U}$  Between 2 and 14000 Seconds, USNRC Report ORNL/NUREG-14, October 1977.
63. O. D. Slagle, "Creep of  $\text{UO}_2$  at 2500°C," presented at the Annual Meeting of the American Ceramic Society, Chicago, Ill., April 1977. Available in the NRC Public Document Room in file for Report NUREG-0006.
64. W. F. Domenico and J. A. Ramsthaler, IFA-429 Data Report No. 1, USNRC Report TREE-NUREG-1009, December 1976.
65. R. K. Marshall et al., "Gap Conductance Uncertainties Deduced from Measured Fuel Centerline Temperatures," Trans. Amer. Nucl. Soc., 24, 155-156 (November 1976). Available in public technical libraries.
66. R. W. Garner, et al., Gap Conductance Test Series Test GCI-3 Test Results Report and Summary of Piggyback Tests, EG&G Idaho, Inc. TFBP-TR-186, March 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
67. E. T. Laats, P. E. MacDonald, and W. J. Quapp, USNRC-OECD Halden Project Fuel Behavior Test Program--Experiment Data Report for Test Assemblies IFA-226 and IFA-239, USERDA Report ANCR-1270, December 1975.

68. C. E. Beyer and C. R. Hann, Prediction of Fission Gas Release from UO<sub>2</sub> Fuel, Battelle Pacific Northwest Laboratories, BNWL-1875, November 1974.
69. S. Dagbjartsson and B. Murdock, Axial Gas Flow in Irradiated PWR Fuel Rods, EG&G Idaho, Inc. TREE-NUREG-1158, September 1977.
70. N. Fuhrman and V. Pasupathi, Evaluation of Fuel Rod Performance in Maine Yankee Core I, EPRI NP-218, November 1976. Available from the Electric Power Research Institute, Research Reports Center, P.O. Box 10090, Palo Alto, California 94303.
71. J. D. Kerrigan, "Status of the Fuel Rod Failure Subcode, FRAIL-3," Quarterly Report on Programs Sponsored by Division of Reactor Safety Research October-December 1976, TREE-NUREG-1070, April 1977, pp. 111-125.
72. W. J. Quapp, C. M. Allison, and L. C. Farrar, "PWR Fuel Rod Behavior Under Film Boiling Conditions," Trans. Amer. Nucl. Soc., 23, 300-302 (1976). Available in public technical libraries.
73. D. W. Croucher et al., Irradiation Effects Test Series, Test IE-5, Test Results Report, EG&G Idaho, Inc., TREE-NUREG-1130, January 1978.
74. G. Kjaerheim, "Heat Transfer as a Limiting Factor in Water Cooled Nuclear Reactors," Nucl. Eng. and Design, 21, 279-302 (1972). Available in public technical libraries.
75. L. B. Thompson, D. L. Hargman, and P. E. MacDonald, Fuel Behavior Program Description: RIA Fuel Behavior Experiment Requirements, Idaho National Engineering Laboratory, RE-S-76-187, October 1976, pp. 5-28. Available for inspection and copying for a fee in the NRC Public Document Room.
76. M. Ishikawa and K. Tomii, Quarterly Progress Report on the NSRR Experiments, Combined October 1975-March 1976, Japan Atomic Energy Research Institute, JAERI-6635, June 1976.
77. R. A. Lorenz, D. O. Hobson, and G. W. Parker, "Fuel Rod Failure Under Loss of Coolant Conditions in TREAT," Nucl. Technol., 11(4), 502-520 (1970). Available in public technical libraries.
78. H. R. Freeburn et al., Light Water Reactor Fuel Rod Modeling Code Evaluation, Final Report, EPRI NP-369, March 1977. Available from the Electric Power Research Institute, Research Reports Center, P.O. Box 10090, Palo Alto, California 94303.
79. J. A. Dearien, M. Bohn, G. A. Berna, J. D. Kerrigan, and D. R. Coleman, FRAP-S3 Analytical Models and Input Manual: FRAP-S3, A Computer Code for the Steady-State Analysis of Oxide Fuel Rods, Volume 1, EG&G, Idaho, Inc., TFBP-TR 164, October 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
80. J. T. A. Roberts et al., Planning Support Document for the EPRI LWR Fuel Performance Program, EPRI NP-737-SR, January 1978, pp. 48-52. Available from the Electric Power Research Institute, Research Reports Center, P.O. Box 10090, Palo Alto, California 94303.
81. E. T. Rumble and R. G. Stuart, BEHAVE-4 LWR Fuel Rod Analysis Code, EPRI NP-892, August 1978. Available from the Electric Power Research Institute, Research Reports Center, P.O. Box 10090, Palo Alto, California 94303.
82. H. B. Meieran and E. Estermann, LIFE-THERMAL Code Description and User's Manual, ODAI-RP-397-2(26), February 1977. Available from the Electric Power Research Institute, Research Reports Center, P.O. Box 10090, Palo Alto, California 94303.
83. D. D. Lanning, C. L. Mohr, F. E. Panisko, and K. B. Stewart, GAPCON-THERMAL-3 Code Description, USNRC Report PNL-2434, January 1978.
84. E. Duncombe, C. M. Fredrich, and J. K. Fischer, CYGRO-3--A Computer Program to Determine Temperatures, Stresses, and Deformations in Oxide Fuel Rods (LWBR Development Program), WAPD-TM-961, March 1970.
85. F. J. Homan, W. J. Lackey, and C. M. Cox, FMODEL--A Fortran-IV Computer Program to Predict In-Reactor Behavior of LMFBF Fuel Pins, Oak Ridge National Laboratory, ORNL-4285, January 1973.

86. J. A. Dearien, L. J. Siefken, and M. P. Bohn, FRAP-T3: A Computer Code for the Transient Analysis of Oxide Fuel Rods, Vol. 1, "FRAP-T3 Analytical Models and Input Manual," EG&G Idaho, Inc., TFBP-TR-194, August 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
87. J. A. Dearien, G. A. Berna, M. P. Bohn, D. R. Coleman, and P. E. MacDonald, FRAP-S2 Analytical Models and Input Manual: FRAP-S2, A Computer Code for the Steady-State Analysis of Oxide Fuel Rods, Vol. 1, EG&G Idaho, Inc., TREE-NUREG-1107, July 1977.
88. D. R. Coleman and E. T. Laats, Model Verification Report: FRAP-S2 - A Computer Code for Steady-State Analysis of Oxide Fuel Rods, EG&G, Idaho, Inc., TREE-NUREG-1107, July 1977.
89. C. R. Hann et al., Application of Experimental Data to the Reduction of Stored Energy Uncertainty, paper presented at the Fourth WRSR Information Meeting, September 27-30, 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
90. J. M. Broughton et al., Comparison of Results from the Steady-State Fuel Behavior Codes GAPCON-THERMAL, LIFE-LWR, and FRAP-S, paper presented at the Fourth WRSR Information Meeting, September 27-30, 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
91. D. R. Coleman, E. T. Laats, and N. R. Scofield, FRAP-S3: A Computer Code for Steady-State Analysis of Oxide Fuel Rods, Vol. 2, "Model Verification Report," EG&G Idaho, Inc., TFBP-TR-228, October 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
92. W. J. Bailey, ed., Quarterly Progress Report to NRC Fuel Behavior Research Branch for Period Ending March 1977, Battelle Pacific Northwest Laboratories, BNWL-2279, April 1977, pp 154-156.
93. RELAP4/MOD5, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems, Vols. I-III, USNRC Report ANCR-NUREG-1335, September 1976.
94. D. R. Coleman and E. T. Laats, FRAP-T3: A Computer Code for Transient Analysis of Oxide Fuel Rods, Vol. 2, "Model Verification Report," EG&G Idaho, Inc., TFBP-TR-194, August 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
95. M. P. Bohn, Status Report on the Development of FRAGAS-II, A Subcode for the Analysis of Fuel and Cladding Deformation, EG&G Idaho, Inc., TFBP-TR-151, December 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
96. S. J. Dagbjartsson, FRAP-T2 and SSYST-1: A Comparative Study of the Basic Models and Computational Results, EG&G Idaho, Inc., TFBP-TR-167, January 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
97. P. E. MacDonald and L. B. Thompson, eds., MATPRO--A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, Aerojet Nuclear Company, ANCR-1263, February 1976.
98. U.S. Nuclear Regulatory Commission, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400 (NUREG-75/014), October 1975.
99. W. B. Murfin, ed., Core-Meltdown Experimental Review, Sandia Laboratories, SAND74-0382, August 1975.
100. R. E. Henry, Light Water Reactor Safety Research Program: Quarterly Progress Report, July-September 1976, USNRC Report ANL-76-121, December 1976.
101. F. A. Kulacki and A. A. Emara, Transient Natural Convection in an Internally Heated Fluid Layer, Ohio State University, USNRC Report NUREG-0078, August 1976.
102. D. A. Dahlgren, Light Water Reactor Safety Research Program, Quarterly Report, January-March 1976, Sandia Laboratories, SAND 76-0369, September 1976.
103. W. B. Murfin, A Preliminary Model of Core/Concrete Interactions, Sandia Laboratories, SAND 77-0370, August 1977.

104. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.3 (1.4), "Assumptions Use Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors (Pressurized Water Reactors)," June 1974.
105. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.
106. R. A. Lorenz, J. L. Collins, and O. L. Kirkland, Quarterly Progress Report of Fission Product Release from LWR Fuel for the Period October-December 1976, USNRC Report ORNL/NUREG/TM-88, March 1977.
107. S. M. Gehl et al., "Transient Heating Tests," in Light-Water-Reactor Safety Research Program: Quarterly Progress Report, January-March 1977, USNRC Report ANL-77-34, June 1977, pp. 60-63; April-June 1977, USNRC Report ANL-77-59, not dated, pp. 88-94; July-September 1977, USNRC Report ANL-78-3, January 1978, pp. 35-40; October-December 1977, USNRC Report NUREG/CR-0089, May 1978, pp. 16-18.
108. J. A. Gieseke et al., Analysis of Fission Product Transport under Terminated LOCA Conditions, USNRC Report BMI-NUREG-1990, December 1977.
109. A. K. Postma and W. F. Pasedag, A Review of Mathematical Models for Predicting Spray Removal of Fission Products in Reactor Containment Vessel, USAEC Report WASH-1329, June 1974.

CHAPTER 4  
PROGRAM PLAN  
FOR  
THE ANALYSIS DEVELOPMENT BRANCH

4.1 INTRODUCTION

4.1.1 PURPOSE OF ANALYSIS DEVELOPMENT

In analyzing the safety margins built into a nuclear power plant, the NRC pays particular attention to determining the response of the plant to postulated design-basis accidents, such as the loss-of-coolant accident (LOCA) and the reactivity-initiated accident (RIA). Estimates of the consequences of these postulated accidents for full-scale reactors are necessarily based on computer code calculations.

Most of the present LWR safety code development work is aimed at assessing the consequences of a LOCA and the behavior of the emergency core cooling systems (ECCS) in pressurized water reactors (PWRs) and boiling water reactors (BWRs). However, many of these codes are being developed in a flexible, modular fashion to be easily applicable to other operating disturbances, such as anticipated transients without scram (ATWS) and reactivity-initiated accidents. Code development is an important part of the reactor safety research program because the computer codes provide the computational means of applying basic experimental and analytical information to commercial nuclear power plants. These computational methods will be used to assess (a) ECCS performance in mitigating the consequences of a postulated LOCA, (b) the influence of various parts of the reactor system on the course of a LOCA and in preventing fuel failures, (c) the response of the reactor system to other postulated accidents, and (d) the degree of conservatism implicit in licensing-type analyses.

A LOCA involves rather complex phenomena. For example, in a pressurized water reactor, the water flashes to steam at the pipe break. The analysis must therefore deal with two-phase (steam-water) mixtures, the two phases often moving at unequal velocities and not necessarily in thermal equilibrium with each other. The forces occurring during a LOCA act on the reactor vessel and its supports. The injected emergency core cooling water becomes mixed with steam and may be swirled around the downcomer before entering the lower plenum to reflood the core. Furthermore, these events occur in three dimensions and change rapidly with time and location. In order to obtain a more detailed description of these events, it is necessary to improve current analytical and programming techniques.

4.1.2 CLASSIFICATION OF COMPUTER CODES

The codes being developed under the sponsorship of the Office of Nuclear Regulatory Research (RES) are divided into two broad categories: (a) systems codes and (b) component codes. The systems codes must be capable of analyzing thermal-hydraulic transients in PWR and BWR power plants. The component codes model in greater detail the behavior of the various individual components of a reactor coolant system.

The systems codes are further subdivided into reactor coolant system (RCS) codes and containment system codes, and there are two versions of RCS codes: The evaluation model (EM) version employs conservative assumptions about the physical phenomena occurring in postulated accidents; such assumptions are defined in the Commission's ECCS Acceptance Criteria for LOCA analysis.<sup>1</sup> The best estimate (BE) versions employ the currently available state of the art in realistic modeling of the physical processes. Their main purpose is to evaluate the degree of conservatism implicit in the licensing (EM) calculations. Only in the case of the BE codes can modeling details be confirmed through comparison of code calculations with available test data. In addition, the BE codes are used in the design of test facilities and in the interpretation of test results. The increased knowledge gained in the overall research effort and the improved state of the art are continuously reflected in the improved versions of the BE codes.

The containment system codes must be able to consider various containment types, as discussed in Section 4.1.5.

Some systems codes were in existence before the Reactor Safety Research Division was established. Review of these codes led to a decision (a) to improve the existing codes to the limit of their

capabilities and (b) to develop advanced versions. The decision to do both resulted from the anticipated lengthy development schedules for the advanced codes and the need for a calculational capability in the meantime. The limitations of the existing codes and their correction in the advanced codes are explained in Sections 4.1.3 and 4.1.4.

The component codes now being developed under the sponsorship of Water Reactor Safety Research (WRSR) are all of the advanced type. Their details are described in Section 4.1.6.

A summary of the WRSR computer code classification is shown in Figure 4-1.

#### 4.1.3 EXISTING REACTOR COOLANT SYSTEM CODES

The existing reactor coolant system codes are based on the following fundamental assumptions and simplifications: The liquid and its vapor are in thermal and mechanical equilibrium; that is, their temperatures are equal, and both flow with the same velocity. In fact, these two fluids are considered to act as a homogeneous mixture, a pseudofluid whose properties can be defined by the steam tables. The assumptions of thermal and mechanical equilibrium allow the thermal-hydraulic behavior to be represented with only three field equations (conservation of the mass, momentum, and energy of the mixture) supplemented by the equation of state (steam tables) and the constitutive equations for wall shear and wall heat transfer. The numerical solution of these equations is based on the lumped-parameter approximation, in which the system is represented by a collection of control volumes (nodes), interconnected by "junctions." Although the nodes could be connected in arbitrary ways, implying multidimensional flow, the conservation of momentum is, in fact, written and solved as a one-dimensional case for each flow "junction." Further details can be found in Reference 2. The present "existing" code, RELAP-4, including its numerical solution method, is based on the Bettis Atomic Laboratory's FLASH-4 code. (The earlier versions of RELAP were based on earlier versions of FLASH.)

This type of code was found to be very suitable for bounding calculations (determination of upper limits), which do not demand attention to details in the modeling of individual physical processes. The RELAP-4/MOD 3 code version was adapted in 1974 to comply with the ECCS Acceptance Criteria and was incorporated into the WREM (Water Reactor Evaluation Model) package; the WREM package included other codes, designed for analyzing the PWR reflood regime (FLOOD code) as well as PWR and BWR hot channels (TOODEE and MOXY, respectively).

The next effort in improving RELAP-4 concentrated on introducing features necessary for describing the consequences of a small break. The primary new feature allowed for the separation of liquid and vapor by gravity. To this end the conservation-of-energy equation was modified to account for the separate convective fluxes of liquid and vapor. Specification of the individual fluxes was accomplished through the addition of another constitutive equation that defines the magnitude of the relative velocity (the difference between the velocities of the two phases) in terms of the local void fraction. Moreover, it was found possible to use the Wallis flooding correlation<sup>3</sup> to define the limits on the relative velocity in countercurrent flow situations. Models were also introduced to allow consideration of the separated cocurrent or countercurrent flow in horizontal pipes. These separate-flow models are yet to be checked out. The RELAP-4/MOD 5 version, which was released to the public through the National Energy Software Center (formerly the Argonne Code Center) in the Summer of 1976, incorporated all of these improvements plus some improvements in correlations for heat transfer during blowdown, treatment of choked flow, and general cleanup of coding. The MOD 5 became the first LOCA best estimate code, although still limited to the LWR blowdown regime only. Additional features were necessary to extend its applicability to the reflood regime in PWR and BWR plants (see Section 4.2.1). These improvements in the RELAP-4 code allow some calculations of the best estimate type, but it is known that they are of limited value.

The plans for an intermediate update of the licensing (EM) code package are described in Section 4.7.

#### 4.1.4 ADVANCED SYSTEMS CODES FOR THE ANALYSIS OF REACTOR COOLANT SYSTEMS

The ultimate goal in developing an advanced systems analysis code is the ability to model, as realistically as possible and in a feasible manner, the thermodynamic and flow processes of the coolant undergoing blowdown or other transients, in all flow regimes. The system code does not permit modeling of the fine structure of the fluid, such as turbulence, growth and interactions of vapor bubbles, details of interfacial phenomena (e.g., wavelet and subsequent droplet formation), vortices, and details of the temperature profiles near phase interfaces. A macroscopic view of the fluid that does not oversimplify the processes to the point of omitting essential phenomena is, however, within reach. This view is limited to the size of a computational cell, in which averaged properties of each fluid phase are used.

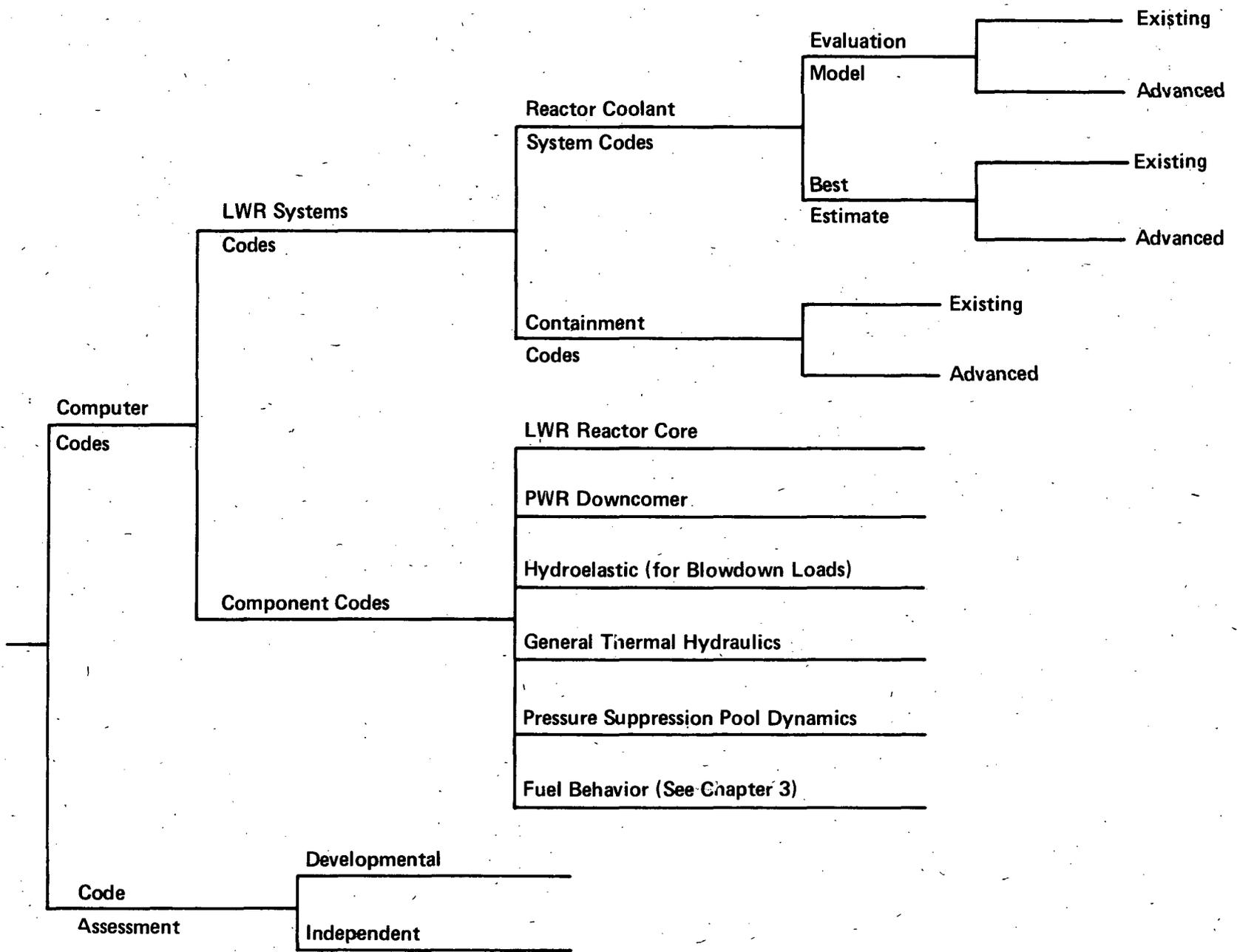


Figure 4-1. Computer Code Classification

In the development of the advanced codes, the Reactor Safety Research (RSR) Division is pursuing a policy of "development in parallel paths." According to this policy, different approaches to best estimate modeling of the transient two-phase flow processes are being explored simultaneously in order to establish which approach is most suitable. After a while it became clear that the approach adopted by the Los Alamos Scientific Laboratory (LASL) in its TRAC code is acceptable for the best estimate advanced code for detailed LWR systems analysis. The TRAC approach eliminates various limitations of the existing code. Details are described in Section 4.2.2.

The TRAC code employs multidimensional modeling for all fluid regions within the reactor vessel and one-dimensional modeling for loop piping, the steam generator, and the pressurizer. The total number of computational cells may exceed 1000, and the computer running time may exceed 10 hours on a CDC-7600 computer (for a 60-second LOCA transient). It is obvious, therefore, that a three-dimensional version of TRAC could not be used for a variety of sensitivity studies involving parametric evaluation of the effects of such items as the spectrum of break sizes, break locations, initial power profiles, and various ECC hardware malfunctions. Such studies demand codes that are much faster running but are free of the important restrictions present in RELAP-4.

The THOR code, being developed at the Brookhaven National Laboratory (BNL), is structured along these lines. It is designed to handle thermal and mechanical nonequilibrium conditions and to account for spatial distribution effects by integrating fluid property profiles over the control volumes, arriving at the volume-averaged quantities that are employed by the conservation equations. The THOR code will also have the ability to track density "shocks" and the movement of the flow and heat-transfer regimes. It is expected to be fast running and therefore suitable for sensitivity studies. It will first be developed as a simplified best estimate code to allow testing against experimental results and later will be converted into the evaluation model form suitable for licensing applications.

A number of novel modeling and numerical solution concepts are being employed in THOR development and may lead to significant problems whose resolution could strongly affect the completion schedules. For this reason RSR has decided to continue with a parallel-path philosophy in this area of the fast-running, simpler advanced codes, suitable as a next-generation licensing tool. To this end the Idaho National Engineering Laboratory, under RSR sponsorship, is examining the RELAP-5 code in terms of a somewhat different approach, containing fewer new concepts. Furthermore, noding flexibility will be introduced into the TRAC code, with the objective of reducing the number of computational cells. This may lead to a simplified version of TRAC, suitable for fast, repetitive calculations that do not demand great accuracy.

The parallel development of THOR, RELAP-5, and simplified TRAC will be reconsidered after one of these has resulted in a simplified best estimate code that meets NRC needs concerning (a) the ability to predict, reasonably well, the results of a variety of integral and separate-effects tests; (b) computational speed and user convenience; and (c) ability to predict both PWR and BWR transients, spanning all LOCA regimes, in a single calculation. In addition, NRC must be convinced that the code can perform plant and model sensitivity studies in a user-convenient manner and can be easily converted into the evaluation model and also into a non-LOCA code.

#### 4.1.5 CONTAINMENT CODES

The reactor containment building provides the third barrier between radioactive isotopes in the fuel and the external environment. The integrity of each barrier, be it the fuel cladding (first barrier), the reactor vessel and piping walls (the second barrier), or the containment building (third barrier), is subjected to a careful study. The purpose of containment analysis is to predict the time histories of pressure and temperature for the air-steam-water mixture along the containment building wall. These, in turn, are to be used to predict the structural response of the wall and to assess the safety of its design.

Most PWR containment designs feature the dry containment concept, which provides adequate containment volume for steam expansion to prevent the peak pressure from exceeding a prescribed value during blowdown, with no dependence on pressure suppression. These containment structures may be of a single-compartment or of a multicompartment design. The next most widely used concept (mainly for BWRs) is based on pressure suppression by steam condensation rather than expansion. The steam is ducted into a large pool of water that condenses it. Another less widely used containment concept employs an array of ice columns, instead of a pool of water, to condense the steam and thereby reduce the required containment volume. The development of containment codes addresses itself first to improved descriptions of the flow of steam-water-air mixtures in multiple dry compartments; this capability is generic to the analysis of all three containment concepts. To it will be added the following: (a) the ability to describe the wetwell and the venting process, (b) development associated with heat and mass transfer in the ice condenser, and (c) the effects of delay in the opening of ice-condenser doors.

In the case of dry (full-pressure) multicompartment containments, the LOCA-induced transients that require multidimensional models for best estimate analysis involve the reactor cavity, the break compartment, and, possibly, the adjacent compartments that communicate with the break compartment via large flow passages.

If a pipe break occurred very close to the reactor vessel, the reactor cavity would experience an asymmetric pressure-wave propagation, causing significant lateral loading on the reactor vessel supports.

Multidimensional treatment of fluid dynamics in the break compartment is needed in the best estimate analysis to describe the flashing jet load, stripping of liquid droplets from the gas stream, possible air-pocket formation, and gradual mixing of air with the vapor-liquid mixture. Should choking occur in the communicating flow passages, the discharge flow into the adjacent compartments, and therefore the pressure loading on the interior walls and barriers, would be influenced by the local composition of the flowing stream. These reactor-cavity loads can also occur in the pressure-suppression (PS) containments.

Of much greater interest for PS containments are the wetwell loads caused by the process of venting air and steam. Air venting into the wetwell can produce large up and down loads, especially in the torus (Mark I) containment configuration, and water impact loads on the distribution header or other structural internals as the pool swells.

The subsequent steam-venting phase can cause large condensation-induced loads after the air has been purged from the drywell. These loads can occur laterally on the downcomers (in Mark I and II containments) and radially on the wetwell walls as the collapsing steam bubbles at the downcomer exits create local pressure spikes. In some designs hydroelastic effects need to be considered in calculating the wetwell loads. Activation of the pressure-relief valves in PS containments can also cause significant wetwell loads.

The effects of postulated burning or detonation of hydrogen-air mixtures are to be evaluated by separate calculations and the resulting overpressures superimposed on those caused by the reactor coolant discharge. Mechanical loads caused by the impact of solid objects (missiles) and by seismic events are handled by structural dynamics codes.

The current status of WRSR-sponsored containment-code development and assessments, and plans for future work are described in Section 4.2.3.

#### 4.1.6 COMPONENT CODES

A systems code should be as complex as necessary to describe the dominant effects. The degree of complexity needed cannot be determined in arbitrary ways. The component codes allow various system components to be nodalized and modeled in as complex a way as is desired (consistent with the available state of the art). The effects of gradual simplifications in nodalization (or degrees of freedom) as well as in modeling can then be studied. One can thus establish, for any given system component, the simplest model that retains the features significantly affecting the behavior of the whole system, particularly the peak cladding temperature. These simplified models of the components are then adopted in the systems code.

In some instances it may be possible to examine the behavior of some system components under conditions such that hydraulic coupling with adjacent components can be ignored. The alternative is to make various "separate-effects" experiments, each dealing with a single component (core, downcomer, pump). In all such instances the component codes offer great advantages.

During the subcooled portion of blowdown, structural/hydraulic coupling can affect the instantaneous pressures. This coupling affects the hydraulic loads on the reactor vessel and/or steam-generator internals, which, in turn, are either flexible in comparison with the vessel wall or can be momentarily displaced. It can be argued that in most cases of interest the rigid-wall assumption results in conservative loads. In certain cases such conservatism is excessive and imposes severe design limitations. There is a need, therefore, for improved analytical techniques that include hydroelastic coupling in multidimensional flow passages. Such techniques would result in a "best estimate" code applicable to the subcooled and transition-blowdown regimes, during which the blowdown loads experience maxima. The hydroelastic code is to complement the advanced systems code, since the systems code emphasizes the saturated blowdown, refill, and reflood regimes.

Such a hydroelastic code belongs to the family of component codes because, in its present format, it models only the hydroelastic processes within the reactor vessel. Similar hydroelastic

effects could also take place in the wetwell of the pressure-suppression (BWR) containment during the steam-venting phase.

Component codes (other than the fuel behavior codes) now under development are described in Section 4.3; fuel behavior codes are described in Chapter 3.

#### 4.1.7 CODE ASSESSMENT

Code assessment is performed on two levels: the first level, developmental assessment, is conducted by the code developers, during the code development stage. The scope of developmental assessment includes (a) checkout of individual models or correlations with test data obtained in basic or "model development" experiments, (b) comparisons of code modules or of component codes with data obtained in separate-effects experiments, and (c) limited comparisons of systems codes with data obtained in integral tests. Some test data sources and the distinction between the above-mentioned types of experiment are described in Reference 4. While performing these comparisons the code developers are free to modify and improve the code.

Code developers must also assess the code's ability to address thermal-hydraulic phenomena in full-size LWR geometries by determining whether the models and correlations used contain the necessary scaling information. During the developmental assessment, sensitivity studies are also made concerning system nodding, time-step effects, and the consequences of various assumptions and simplifications.

The second level of code assessment, independent assessment, is performed by personnel who were not involved in the development of the code. The purpose of independent assessment is to (a) provide an independent audit of the code's capability; (b) perform further comparisons with test data, emphasizing "blind" pretest predictions; and (c) evaluate the code error.

The code error is obtained by running a statistically significant number of comparisons between code predictions and test data. The best estimates are used for all input parameters, system nodalization, time-step control, and test measurements (e.g., of the peak cladding temperature). The code error accounts for the combined effects attributable to:

- Modeling simplifications and time-space averaging.
- Numerical solution (deletion of higher order terms, truncation errors, loose convergence criteria, spatial resolution, linearization).
- Simplifications in descriptions of material properties and equations of state.
- Simplifications in constitutive equations concerning exchange of mass, momentum, and energy across liquid-vapor interfaces.
- Simplifications in expressions for wall friction and heat transfer.
- Inability to handle stochastic phenomena.

It does not account for uncertainties in the code input and in the embedded coefficients used in various adopted best estimate correlations since these particular effects are explored in the code uncertainty studies.

Since the independent assessment is performed only for codes that have already been released to the public (via the National Energy Software Center), no further adjustments to that particular version of the code are allowed during that stage (i.e., the code is "frozen"). Justification of the code's applicability to full-scale LWR geometries is to be provided by the code developers. During the independent assessment, these justifications are reviewed and checked. The application of the code to the evaluation of LWR accident consequences is the responsibility of NRC. Code uncertainty studies during that stage are executed by NRC with the contractor's help.

A code must be evaluated to quantify the error in the prediction of the peak cladding temperature that is associated with intrinsic modeling deficiencies rather than code input uncertainties. Codes found to contain large errors will be either improved or abandoned. Judgments of a code's adequacy must take into account the sensitivity of the peak cladding temperature to some discrepancy between test data and a code-calculated parameter (e.g., some local pressure, or flow, or density history). Quantification of the actual code error can be made if enough independent comparisons have been made between test data and code predictions. The essence of this paragraph is in pointing out that the assessment process is not used to prove the veracity of the code (i.e., its ability to perfectly simulate nature). Instead, it is a process in which the code's

strengths and weaknesses are assessed to see whether the code is acceptable for its intended missions.

#### 4.1.8 SENSITIVITY STUDIES

Code sensitivity studies are performed during the code development stage, primarily to explore the sensitivity of the peak cladding temperature to variations in (a) the physical models, (b) the adopted degrees of freedom and system nodalization, and (c) time-step control. These studies enable code developers to identify modeling parameters whose uncertainty can significantly affect important code results. Sensitivity studies usually involve varying one parameter at a time. An important by-product of these studies is a list of the important parameters that need to be considered later during code uncertainty studies.

#### 4.1.9 CODE UNCERTAINTY STUDIES

The application of a code to studies of the consequences of postulated accidents calls for a study of code uncertainty. The purpose of such a study is to explore the manner in which the combined effects of simultaneously varying all the important parameters (within their known uncertainty range) and of the known code error determine the uncertainty in the calculated peak cladding temperature.

Since the code uncertainty study involves an LWR plant, this activity belongs to the realm of code application (by the NRC) rather than code development.

The first step in the uncertainty study is a selection, by the NRC, of the case to be examined, including not only the plant type and the pipe-break size and location but also assumptions concerning (a) the plant life and the associated fuel condition (burnup, axial and radial power distribution, fission-gas release, gap conductance), (b) the operating condition of the ECCS train, and (c) the availability of offsite power.

The second step is to identify the ranges and the probability distributions of the uncertainties in the selected (important) input parameters. When the input parameters are related to some correlations (e.g., for heat transfer, flooding, liquid entrainment, or pressure drop), test-data scatter around the nominal values can provide information on the uncertainty range and probabilities. In other instances assessments must be based on less direct experimental or analytical evidence. The parameters to be varied in the uncertainty study are selected by the NRC on the basis of (a) the code sensitivity studies during code development and (b) the results of experimental measurements of parametric variations.

The third step is to generate a response surface for the selected code inputs. Particular care must be devoted to choosing the sampling method (among the defined ranges of each selected code input) that minimizes the number of computer runs required.

The fourth and the last step is to perform a Monte Carlo analysis of the response surface, using the probability distributions of uncertainties in the input parameter. The outcome of this step is a probability distribution function for the peak cladding temperature (PCT). The code error (i.e., the error in the PCT not attributable to input uncertainties) must then be convoluted into the PCT probability distribution function. Figure 4-2 illustrates these steps.

Knowledge of the probability distribution function is essential in defining the margin of safety contained in the licensing (evaluation model) calculations. However, this knowledge must be supplemented by experiments to identify how the assumed plant condition (at the beginning of the uncertainty study) affects the course of the postulated accident.

Criteria defining the margin of safety have not yet been firmly established. One possibility is to define the difference between the most probable peak cladding temperature and the current licensing (evaluation model) value in terms of the number of standard deviations.

Figure 4-3 illustrates the division between the independent assessment of the code and application of the code to evaluation of the margin of safety, including various major steps in each activity.

#### 4.1.10 SUPPORTING EXPERIMENTS FOR THE DEVELOPMENT OF BASIC ANALYTICAL MODELS

The developers of advanced codes need information on basic physical phenomena in greater detail than is available from integral systems tests or even from separate-effects tests. Examples are (a) the transport of mass, momentum, and energy across vapor-liquid interfaces, for various flow regimes relevant to LWR transients; (b) the spatial distribution of liquid and vapor during

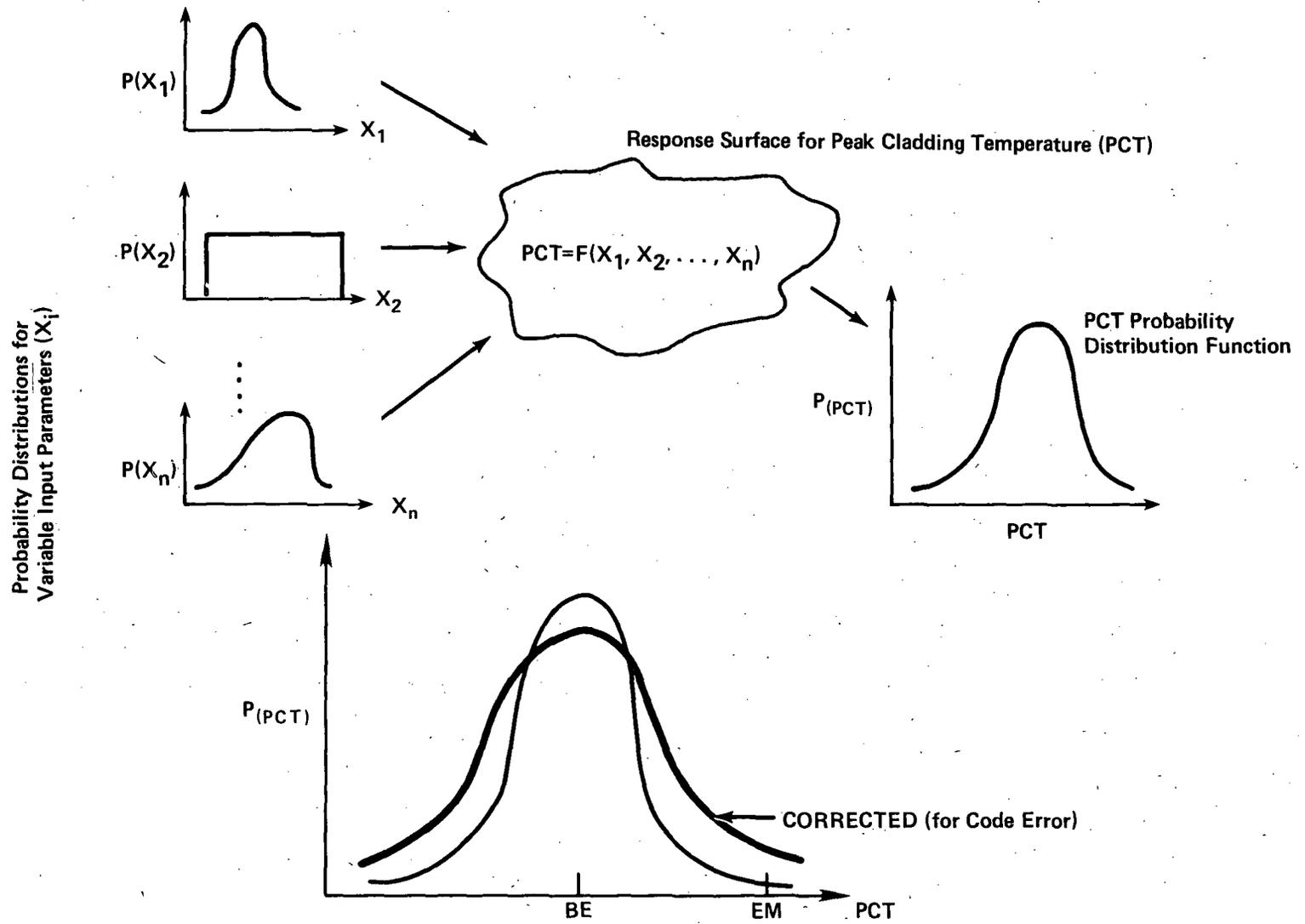


Figure 4-2. Uncertainty Study for Calculated Peak Cladding Temperature

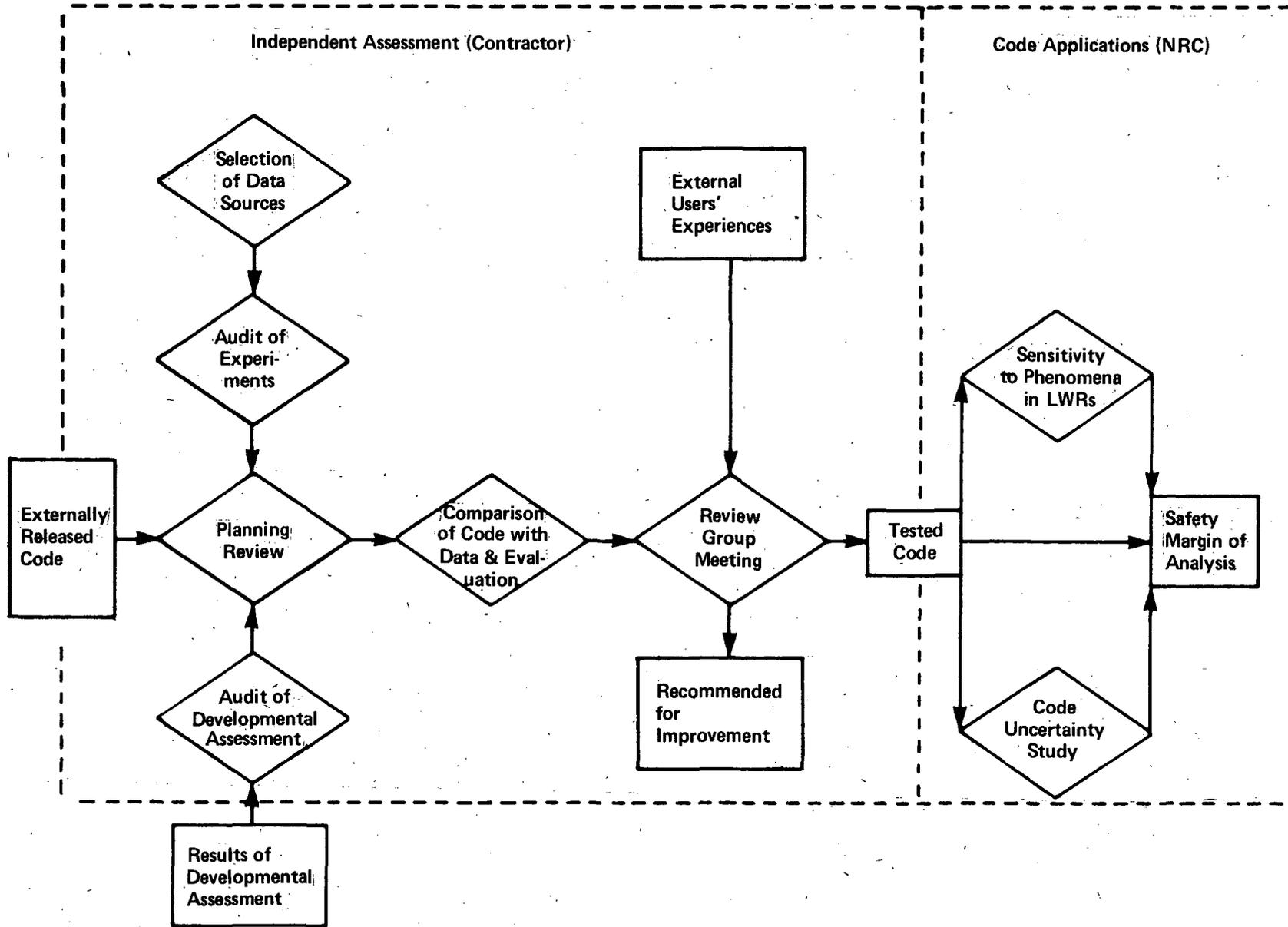


Figure 4-3

steady and transient flows; and (c) the rates of momentum and energy transfer between the wetted walls (or wetted internal structures) and both the liquid and the vapor phases, for various flow and heat-transfer regimes.

The advanced best estimate codes employ multidimensional as well as one-dimensional modeling in which all of the above-listed phenomena are space-averaged over the computational cell and time-averaged over periods that are short in comparison with time constants for fluid dynamics and heat transport, yet long in comparison with unimportant thermal-fluid dynamics periods such as the transport times of individual bubbles or droplets.

Our knowledge of two-phase flow regimes is based solely on studies of steady flow in long pipes. This knowledge has limited applicability to LWR coolant systems. We need to study phase redistribution during transients, in complex multidimensional geometries. This requires understanding of the internal forces responsible for vapor migration, drop coalescence, drop breakup, and liquid entrainment. This basic understanding must then be used to identify the dominant phenomena and to describe them in a simplified manner.

There is a strong interdependence among interfacial transport (of mass, momentum, and energy), the flow regime, and the volumetric (space-averaged and time-averaged) concentration of the vapor-liquid interfacial area. The coupling among these various effects must be studied and modeled. Such studies require sophisticated measurement techniques and great ingenuity in defining the experiments to isolate and quantify the individual mechanisms and effects.

The Analysis Development Branch (ADB) staff is heavily involved in the selection, monitoring, and technical direction of the experiments. In addition to these basic tests, the ADB staff directs the WRSR effort in the area of containment tests, such as the 1/5-scale Mark I pressure-suppression-pool dynamics test at the Lawrence Livermore Laboratory (LLL) and the laboratory-scale containment-related tests and scaling studies at the University of California at Los Angeles and the Massachusetts Institute of Technology. Furthermore, the ADB staff monitors, for WRSR, foreign tests in the area of critical flow and of LWR containments, provides technical consultation to foreign research review groups (e.g., the NORHAV Program of the Nordic Group), and monitors various foreign tests as well as foreign code developments.

The supporting experiments for which the Analysis Development Branch is responsible are discussed in Section 4.5. Other tests are described in Chapter 2.

## 4.2 SYSTEMS CODES

### 4.2.1 EXISTING REACTOR COOLANT SYSTEM CODES

#### 4.2.1.1 Present Status

##### 4.2.1.1.1 RELAP-4/MOD 5

The RELAP-4/MOD 5 code, developed by the Idaho National Engineering Laboratory, is applicable to analyses of LWR LOCA blowdown regimes only. Version 1 was released to NRC at the beginning of January 1976. Since then, various errors have been found and corrected, resulting in Version 2, which was documented and released to the public (National Energy Software Center) in the Summer of 1976. This code can now be used:

- a. Without frequent "water packing" problems.
- b. Without "water stacking" in vertically arrayed control volumes.
- c. With the relative motion between vapor and liquid included in the energy equation, to allow treatment of inhomogeneous and countercurrent flow.
- d. With improved, composite models for best estimate critical flow calculations.
- e. With the "flooding" model for ECCS bypass and with the lower plenum entrainment model.
- f. With ability to consider air-steam-liquid mixtures (for containment studies only).

Complete documentation of RELAP-4/MOD 5 is available.<sup>5</sup> Emphasis is on the best estimate version, although various options can be selected for conservative analysis.

#### 4.2.1.1.2 RELAP-4/MOD 6

The purpose of RELAP-4/MOD 6 is to extend the applicability of RELAP-4 to the PWR reflood regime. The first version of MOD 6 was internally released (to the NRC) in January 1977. It features

- a. Improved (best estimate) heat-transfer coefficients and their selection logic.
- b. Moving mesh for heat-conduction calculations, to allow description of the axial cladding temperature details in the vicinity of a moving quench front.
- c. Models for liquid entrainment in the core above the quench front.
- d. Models for liquid entrainment from the pool in the upper plenum.
- e. Flooding at the upper core-support plate.
- f. Up to four parallel core channels, without crossflow, each channel having its own moving mesh for heat conduction.
- g. Deentrainment of liquid droplets in the upper plenum.
- h. Consideration of thermal nonequilibrium in the core channels (in the regime of dispersed liquid droplets). In this region heat transfer from the fuel rods is apportioned between the vapor (which can be superheated) and the saturated liquid droplets.

The code is not capable of a continuous PWR LOCA analysis (blowdown through reflood), and the models referred to in items c, d, e, g, and h above are fairly rudimentary.

Update 3 of MOD 6 was externally released in January 1978. A best estimate preprocessor is being incorporated to allow automatic selection of the best estimate options.

#### 4.2.1.1.3 RELAP-4/MOD 7

The original purpose of MOD 7 was to extend the applicability of RELAP-4 to the BWR reflood regime. Work during fiscal year 1977 was aimed at

- a. Improving the descriptions of jet-pump behavior during all phases of LOCA.
- b. Improving the descriptions of the steam-water separator.
- c. Incorporating the description of ECC sprays within the upper plenum.
- d. Incorporating the description of quench propagation due to the falling film and/or droplets originating from the upper plenum sprays. This is in addition to the existing (MOD 6) capability to consider quenching due to bottom reflood.

These specific features were incorporated into Version 1 of RELAP-4/MOD 7. Since RELAP-4/MOD 6 was found to be lacking in user-convenient features necessary for repetitive PWR LOCA analyses, the mission of RELAP-4/MOD 7 was changed from a BWR LOCA emphasis to a user-convenient PWR code, described in Section 4.2.1.2.2.

#### 4.2.1.1.4 Scoping Studies with Existing Codes

Toward the end of 1976 a comparison study was conducted, utilizing RELAP-4/MOD 5, to explore the differences in the peak cladding temperatures calculated by the best estimate and the conservative (EM) code versions. These predictions were made for the Westinghouse (PWR) Zion plant.<sup>6</sup> This study, named BE/EM, showed that the BE-calculated cladding temperature during the blowdown regime is significantly lower than that calculated in licensing evaluations.

The BE/EM study was extended during the Spring of 1977 into the reflood regime. This study explored the consequences of sequentially changing the evaluation model as follows:

- a. Replacing the EM model for the metal-water reaction by the BE model.
- b. Decreasing the EM values for decay heat by 20% (which represents the upper bound for BE analysis).
- c. Allowing the ECC liquid injected by the accumulators in the intact loops to penetrate into the lower plenum.

- d. Considering complete deentrainment of liquid in the upper plenum.
- e. Allowing complete fallback of liquid (without flooding) into the core during reflood.

The purpose of this study was to evaluate the relative merits of these various modifications. The Zion plant was again used as a basis, and the RELAP-4/MOD 6 code was employed for reflood analysis. The results obtained are considered to be very preliminary since the assessment of MOD 6 had not been completed before this study.

Under NRC sponsorship, Sandia Laboratories is using the appropriately modified RELAP-4/MOD 5 code to evaluate an alternative ECC concept: upper head injection of ECC water.

#### 4.2.1.1.5 User-Convenient Version of RELAP-4

The RELAP-4/MOD 5 code, developed at the Idaho National Engineering Laboratory, was cast into the user-convenient version at the Savannah River Laboratory (SRL), utilizing SRL's data-management system named JOSHUA. This activity, completed toward the end of fiscal year 1977, resulted in the code package WRAP, which allows

- a. Automatic initialization (calculation of steady-state conditions).
- b. Renodalization on restart.
- c. Ease of system geometry description through the use of the existing library of LWR system components.

The WRAP package was designed to allow the sequential running of various codes needed for licensing analysis, including the codes for blowdown, reflood, and hot-channel analysis, for both PWR and BWR plants. Current plans for this code package are described in Section 4.7.2.

#### 4.2.1.2 Research Program

Original plans called for improvements in the existing RELAP-4 best estimate code, in accordance with specifications prepared jointly in 1975 by RSR and DTR (now NRR/DSS) personnel and agreed on by RSR and DTR management. As already mentioned, RELAP-4/MOD 5 was the first best estimate code version that was applicable to LWR LOCA blowdown. It was issued to the public during the Summer of 1976.

##### 4.2.1.2.1 RELAP-4/MOD 6

The MOD 6 version was expected to extend the applicability of RELAP-4 to best estimate descriptions of the reflood phase of a LOCA in pressurized water reactors. Update 3 of MOD 6 was released internally in September 1977. Additional work was performed before the January 1978 public release of this code to (a) resolve the issue of oscillatory quench-front propagation, (b) make more comparisons with test data, (c) define the best estimate nodalization for PWRs and for existing reflood test facilities, and (d) define the best estimate values of various input parameters, together with the best estimate modeling options.

Sandia Laboratories will use RELAP-4/MOD 6 Update 3 for PWR LOCA code uncertainty studies during 1978.

##### 4.2.1.2.2 RELAP-4/MOD 7

The MOD 7 code was originally planned as an extension of MOD 6 into the BWR reflood regime. Any weaknesses found during the assessment of MOD 6 and 7 were supposed to be corrected in MOD 8. Further improvements, especially with respect to user convenience (self-initialization and renodalization on restart), were to be added in this "end of the line" version of RELAP-4.

However, MOD 6 cannot provide a calculational capability for uninterrupted analysis of a complete PWR LOCA (blowdown through reflood). In addition, difficulties arise in the use of MOD 6 to calculate the refill process. Currently this is provided by a separate (hand) calculation. Since the system nodalization required by MOD 6 for reflood analysis differs substantially from that normally used for blowdown analyses, considerable effort is needed to define code input for reflood calculations.

To correct these weaknesses, the Idaho National Engineering Laboratory has recommended that (a) MOD 6 be released in its present state, so that the uncertainty studies (by Sandia) and the independent assessment of MOD 6 can begin; and (b) plans for MOD 7 be restructured to eliminate the above-mentioned weaknesses of MOD 6. According to these plans, the user-convenient features

originally scoped for MOD 8 would also be added to MOD 7. Consequently, RELAP-4/MOD 7 would be essentially a PWR LOCA code with a limited capability for BWR reflood analysis. The latter was introduced during fiscal year 1977. The release to the public of MOD 7 capable of an integrated analysis of a whole PWR LOCA is now scheduled for June 1979. This code will terminate the RELAP-4 series.

#### 4.2.1.2.3 WRAP

As already explained, the initial WRAP package is the user-convenient version of RELAP-4/MOD 5. This package will be expanded to include fuel-behavior codes, hot-channel codes, the PWR-reflood code, and the NORCOOL code; the latter is a BWR-reflood code developed for NRC use at the RISØ Institute in Denmark as part of the information exchange agreement between the NRC and the Nordic group (NORHAV Program). Further details are given in Section 4.7.2.

#### 4.2.1.3 Milestones

- RELAP-4/MOD 5 completed and externally released (to the National Energy Software Center) in the Summer of 1976.
- RELAP-4/MOD 6, Update 3, externally released at the end of January 1978.
- RELAP-4/MOD 7, Update 3, to be completed at the end of September 1979.
- WRAP (BWR LOCA) to be completed in December 1978.
- WRAP (PWR LOCA) to be internally released (to NRC) in December 1978.

### 4.2.2 ADVANCED REACTOR COOLANT SYSTEM CODES

#### 4.2.2.1 Present Status

##### 4.2.2.1.1 TRAC

The TRAC code, being developed at the Los Alamos Scientific Laboratory, is the NRC's most advanced best estimate systems code. Among its features is the capability for multidimensional (r,  $\theta$ , z) representation of the reactor (PWR and BWR) vessel interior, employing the two-fluid model. This model uses separate conservation equations (for mass, momentum, and energy) for the liquid and the vapor, including the interfacial balance equations. Thus, fully nonequilibrium conditions, both thermal and mechanical, are handled.

Another feature of the TRAC code is one-dimensional representation of the reactor coolant loops and components (e.g., pressurizer, steam generator, accumulators, and pumps). The drift-flux model employed in this representation consists of the following five conservation equations:

- a. Mixture mass balance
- b. Vapor mass balance
- c. Mixture momentum balance
- d. Mixture energy balance
- e. Vapor energy balance

Therefore the model allows fully nonequilibrium thermal and partially nonequilibrium mechanical conditions. The balance equations are complemented by the constitutive equations for the local vapor-generation rate, the relative velocities of vapor and liquid, wall friction, heat transfer from the walls to the individual phases (liquid and vapor), and heat transfer between vapor and liquid.

Other characteristics and capabilities of TRAC can be summarized as follows:

- a. All phases of a LOCA (e.g., blowdown, refill, and reflood) are treated in a consistent and continuous manner.
- b. A separate module is provided for each component. These modules can be coupled in any desired arrangement to analyze any reactor system or experiment. Individual modules can be readily modified without disturbing the rest of the code.

- c. The code is also modular with respect to function; that is, individual capabilities (e.g., fluid dynamics, heat transfer, equations of state) are contained in individual modules, which can easily be changed on an individual basis.
- d. A combination of semiimplicit and fully implicit techniques has been developed to provide an efficient numerical solution of the set of equations.
- e. A sophisticated set of flow-regime-dependent constitutive equations has been implemented.
- f. Wall and fuel-rod heat transfer, including quench-front propagation, are modeled. Each axial channel (a vertical stack of mesh cells with the same  $r-\theta$  boundaries) can contain an average fuel rod and a hot rod. The average rod couples directly with the thermal-hydraulic behavior, whereas the hot rods are decoupled.
- g. The two-phase vessel module treats such key phenomena as ECC bypass in the downcomer, multidimensional flows in the core and plena, pool formation and fallback at the upper core-support plate, liquid-level tracking, entrainment and deentrainment, and both rising and falling quench fronts.
- h. Automatic self-initialization is provided.
- i. Sophisticated graphic output, including color code movies, has been developed.

Because of its detailed multidimensional thermal-hydraulic capability, TRAC can address a broad range of problems. In some applications, the relatively long running times will tend to limit the use of the three-dimensional version for sensitivity studies.

The PWR LOCA version of TRAC (TRAC-P1) was completed in March 1978. A faster running version (TRAC-PIA) was documented and released to the National Energy Software Center in July 1978. Extension to BWR LOCA analysis will be completed and released by the end of December 1978. The developmental assessment, to be completed before the code is externally released, is described in Section 4.4.1.3. Further details are described in various LASL Quarterly Progress Reports (see, for example, Reference 7).

#### 4.2.2.1.2 THOR

The THOR code, being developed at the Brookhaven National Laboratory, is an advanced LWR systems code intended for repetitive best estimate analyses, for sensitivity studies, and for the next-generation licensing (EM) code. It is important, therefore, that this code be economical to run, in spite of the introduction of significantly advanced modeling in comparison to RELAP-4. The following features are planned for THOR:

- a. One-dimensional and lumped-parameter geometry representation.
- b. Partial thermal and mechanical nonequilibrium capability through the four-conservation-equation drift-flux model wherever required. The dispersed phase (vapor bubbles or liquid droplets) is considered to be under saturation conditions. The four conservation equations are mixture mass balance, vapor mass balance, mixture momentum, and mixture energy balance. These are complemented with the constitutive equations for the local vapor-generation rate, wall friction, wall heat transfer, and the relationship between the enthalpy of the mixture and the enthalpy of the continuous phase, the dispersed phase being in thermal equilibrium.
- c. Spatial profiles of the dependent variables that are assumed to be linear and spatially integrated to arrive at the volume-averaged properties (for "lumps") and the relation to their entry and exit values. This technique requires a simultaneous solution of a set of ordinary differential equations (for volume-averaged properties), together with a set of algebraic equations that relate the volume-averaged properties with their end (boundary) values.
- d. One-dimensional neutron kinetics (in contrast to the point kinetics in RELAP-4). This feature becomes important for the analysis of anticipated transients without scram and the small-break LOCA.
- e. Self-initialization.
- f. Modular code structure, which maintains the identity of the various LWR system components.

- g. Ability to track liquid levels and moving boundaries of flow regimes and heat-transfer regimes.

To achieve the above objectives in an efficient and logical manner, sequential stages of increasing model complexity have been planned:

- a. Computational feasibility will be shown through sequential stages for the increased geometric complexity (e.g., the ability to calculate (1) a pipe/pump loop transient and (2) a two-loop PWR).
- b. The LWR system will be "hard-wired" (i.e., a specific geometrical arrangement of the system components is to be considered). Generalization to arbitrary geometries, if proved possible, will be considered in 1978 or beyond.

The first licensing (EM) version of THOR is expected to be completed during calendar year 1979.

#### 4.2.2.1.3 RELAP-5

The RELAP-5 code is being developed by the Idaho National Engineering Laboratory (INEL) as a "parallel path" effort to ensure that a fast-running advanced systems code will be available on a timely basis (for advanced licensing calculations), as indicated in Section 4.1.4. For this code, therefore, the intent is to employ the proved, existing techniques for the implicit, one-dimensional solution of the four-conservation-equation model. The model is presently being explored in a "pilot code." Ability to solve the following conservation equations has been demonstrated for a single channel (pipe): mixture mass balance, mixture momentum balance, equation for the difference between the momentum balances of the two phases, and mixture energy equation. In addition, a partial thermal nonequilibrium version is also considered, through the addition of the vapor mass balance. It will be noted that full mechanical nonequilibrium conditions are allowed, due to two momentum equations. Current plans call for demonstrating the ability to consider choked flow, sudden flow-area changes (contractions, expansions, orifices), and tees (branches) before proceeding with the development of the RELAP-5 systems code.

#### 4.2.2.2 Research Program

##### 4.2.2.2.1 TRAC

TRAC represents the state of the art in the development of the advanced, multidimensional codes applicable to the best estimate analysis of LWR coolant systems.

The first PWR version of TRAC was completed at the end of December 1977 and was internally released in March 1978.

In 1978, work was initiated on (a) extending TRAC to a BWR LOCA; and (b) a version of TRAC capable of significant simplifications in the description of system geometry. The aim is to produce a fast-running version of TRAC, suitable for the repetitive calculations needed for uncertainty studies and licensing evaluations. In addition, further comparisons will be made with the existing test data base and TRAC sensitivity studies will be performed. The completion of a first BWR LOCA version of TRAC (TRAC-B1) is now scheduled for the end of December 1978.

An important role is planned for TRAC in (a) providing information needed for the design of the German upper plenum deentrainment test and Japan's 2000-rod reflood test facilities\* (for both a cylindrical and a "slab" core); (b) providing the calculated initial and boundary conditions for these test facilities; and (c) performing pretest predictions and posttest comparisons with test data. Through these tasks TRAC will undergo extensive assessment in these large-scale test facilities. In combination with assessment at other test facilities (LOFT, Semiscale, Two-Loop Test Assembly, and PKL), this is expected to establish the credibility of TRAC's calculations of the consequences of LOCAs in light-water reactors.

Further long-term tasks include extending TRAC to best estimate analyses of non-LOCA accidents such as anticipated transients without scram and reactivity-initiated accidents. TRAC will also be adapted to the future designs of LWR hardware involving alternative emergency core cooling concepts. The full three-dimensional version of TRAC will serve as a yardstick against which simpler systems codes can be normalized.

\*See Section 6.7 for a discussion of the international three-dimensional reflood test program, which includes the German and Japanese test facilities.

#### 4.2.2.2.2 THOR

THOR is being developed as an advanced one-dimensional systems code for fast repetitive studies of the consequences of postulated accidents.

A best estimate version of THOR applicable to a fixed LWR geometry is expected toward the end of 1978, and an evaluation model version one year later. Thereafter THOR will be expanded to handle anticipated transients without scram and reactivity-initiated accidents for licensing-audit (evaluation model) calculations. The development of a generalized, rather than fixed, geometrical arrangement for THOR will proceed simultaneously, to allow the simulation of various geometries for integral tests.

#### 4.2.2.2.3 RELAP-5

The development goals for the RELAP-5, simplified TRAC, and THOR codes are, in this parallel-path approach, identical: to provide NRC with a fast-running version of an advanced systems code, suitable for a next-generation evaluation model and capable of economical repetitive calculations in both the best estimate and the evaluation model modes.

A pilot code to investigate the feasibility of RELAP-5 modeling techniques was completed at the end of 1977. This effort is being followed by the development of the actual RELAP-5 systems code.

The purpose and scope of RELAP-5 development were outlined in Section 4.2.2.1.3. The development of RELAP-5 will emphasize user convenience and applicability to the repetitive licensing-audit calculations for both LOCA and non-LOCA transients in light-water reactors. The first public release of RELAP-5 (LOCA version) is scheduled for 1980. However, the reader is reminded of the criteria stated at the end of Section 4.1.4, which call for a reexamination and reconsideration of the need to continue the parallel-path concept.

#### 4.2.2.3 Milestones

- TRAC-P1 (PWR LOCA) completed in March 1978 and internally released.
- TRAC-P1A (PWR LOCA) to be released to the public in December 1978.
- TRAC-B1 (BWR LOCA) to be internally released by May 1979.
- TRAC/ATWS (LWR ATWS) to be internally released by December 31, 1979.
- TRAC/RIA (LWR RIA) to be internally released by December 31, 1980.
- Simplified TRAC for PWR LOCA may be internally released by April 1979 (schedule being reviewed).
- THOR-BE to be internally released by December 31, 1978.
- THOR-EM to be internally released by December 31, 1979.
- RELAP-5 to be internally released by December 31, 1979.

### 4.2.3 CONTAINMENT SYSTEM CODES

#### 4.2.3.1 Improvements in Existing Codes

The codes described in this section are being developed under the NRR Technical Assistance Program. The assessment of CONTEMPT codes is sponsored by WRSR.

CONTEMPT-LT/MOD 26 was developed at INEL for the conservative analysis of long-term transients with a very limited subcompartment analysis capability. A BWR pressure-suppression option is included. This code was released to the public in September 1976. Development of the MOD 2 version is presently under way.<sup>8</sup>

CONTEMPT-4 was developed at INEL for analyses of multicompartment dry containments. An ice-condenser option has been incorporated, including the simulation of long-term heat removal. This code was released to the public in February 1978.

The COMPARE code is being developed at the Los Alamos Scientific Laboratory. Its mission is similar to that of CONTEMPT-4. The MOD 1 version of this code was completed in September 1977.

#### 4.2.3.2 Advanced Containment System Codes

Under WRSR sponsorship, INEL is developing a multidimensional system code called BEACON (Best Estimate Advanced Containment Code). BEACON/MOD 1 was internally released in December 1976.

BEACON/MOD 1 is capable of describing the mixed geometry in which two-dimensional regions can be joined with one-dimensional regions, while the latter can be coupled with the lumped-parameter regions. This is in contrast to the existing codes, which are of the lumped-parameter type.

In addition, BEACON/MOD 1 employs a multifield model based on the KACHINA code developed by the Los Alamos Scientific Laboratory (LASL). In MOD 1 the transport of mass between phases is ignored. However, the motion of liquid droplets is tracked separately from that of steam and air. Air and steam are assumed to be at the same temperature and moving with equal velocities. Various boundary conditions have been introduced to allow the simulation of fairly complex geometries.

Since LASL's K-FIX code resulted in significant improvements over its predecessor (KACHINA), INEL has based the BEACON/MOD 2 version on K-FIX. In order to make K-FIX even more applicable to analyses of containment problems, LASL added the mass-conservation equation for the noncondensable gas (air) before transmitting K-FIX to INEL.

BEACON/MOD 2, which was released to the public at the end of December 1977, is highly suitable for the analysis of hydraulic loads in the reactor cavity, in single- or multiple-compartment dry containments, and in the drywell and the vent pipes of pressure-suppression containments. These hydraulic loads are highest during the first few seconds of blowdown and are caused by pressure-wave propagation (shocks), jet impingement, and intercompartment pressure differentials.

#### 4.2.3.3 BWR Pressure-Suppression Pool Dynamics Analysis

The Lawrence Livermore Laboratory is developing the CHAMP series of codes. These codes combine the Lagrangian and the Eulerian reference frames and are therefore suitable for tracking gas-liquid interfaces (level swell) and for considering the effects of pool-wall motion (hydroelastic effects).

There are two distinct stages of LOCA that affect pressure-suppression pool dynamics. In the early stage, steam is pushing air out of the drywell and into the wetwell pool. During the vent clearing process the hydraulic forces load the pool bottom. When air is released into the pool, it causes pool swelling and compresses the gas region above the pool, causing upward-directed hydraulic loads. In addition, the upward-accelerating pool surface could impact some wetwell structures.

The second stage of interest takes place when the air has been purged from the drywell and the steam discharge flow through the ruptured pipe has diminished in intensity. During this stage, the steam issuing from the downcomers condenses rapidly as it contacts the cold liquid, leading to an oscillatory, or "chugging," flow and pressure field. Concern has been expressed--as a result of the Marviken I test series in Sweden--that these chugging loads on the wetwell walls might increase in amplitude if resonance were to occur. On the other hand, experimental evidence obtained in Germany has shown that wetwell wall flexibility plays a major role in determining the maximum amplitudes of the pressure oscillations.

These two stages clearly require different modeling emphasis. Consequently, the first part of the WRSR-sponsored effort at the Lawrence Livermore Laboratory is to obtain the air-venting calculational capability, including

- a. Air venting with single downcomers in axisymmetric geometries.
- b. Effects of neighboring downcomers.
- c. Generation of a simpler (lumped-parameter) module for air venting, for addition to INEL's BEACON code.

The second part is to incorporate two-phase-flow capability, thus permitting analyses of the second stage (steam venting). The same three steps mentioned above would be followed. Finally, the consequences of adding hydroelastic effects are to be analyzed.

Progress has been slow so far, mainly because of the very long computer running times required for the explicit-finite-difference technique in the current version of CHAMP. The Lawrence Livermore Laboratory is presently investigating the use of incompressible-fluid codes and the introduction of an implicit numerical scheme.

#### 4.2.3.4 Research Program: Advanced Containment System Codes

BEACON/MOD 2, which has state-of-the-art capability for investigating hydraulic transients in multicompartment dry containments, was released to the public at the end of December 1977. That code version is designed primarily for the evaluation of jet impingement and reactor cavity loads. During 1978 routines will be added for the calculation of liquid deentrainment and film formation on compartment walls, and also for wall heat transfer. The BEACON-2A version, to be internally released at the end of December 1978, will therefore be applicable to best estimate analysis of intercompartment pressure and temperature differentials. The addition of the code modules needed for describing BWR (pressure-suppression) containment systems is scheduled for the BEACON/MOD 3 version. For example, the Mark I and Mark II containment wetwell modules and the ice-condenser module will be added in calendar year 1979. The work on the Mark I and II pool-dynamics modules will draw extensively on the knowledge gained at the Lawrence Livermore Laboratory, where the wetwell (pool dynamics) transients are studied in detail by means of multidimensional component codes, such as CHAMP. The wetwell modules to be added to the BEACON/MOD 3 containment systems code will, of necessity, be less detailed but will retain the important features that affect the overall system behavior.

The Mark III (horizontal venting) pool-dynamics module will be added to BEACON/MOD 3 during 1980. Complete documentation of BEACON/MOD 3 and its release to the public are scheduled for the end of fiscal year 1980.

Analyses related to non-design-basis LOCA-induced transients are scheduled for fiscal year 1981. These will include hydrogen explosion and the effects of a large release of noncondensable gas (resulting from the interaction between molten metal and concrete). These and further analyses (in fiscal year 1981 and later) will respond to future needs defined by the Division of Nuclear Reactor Regulation.

Pool-dynamics analyses at the Lawrence Livermore Laboratory are performed with detailed component codes such as PELE-IC. They are discussed in this section, rather than together with other component codes (Section 4.3), for continuity of subject matter. For background information the reader should refer to Section 4.2.3.3.

Present plans call for the merger of PELE-IC, which is a multidimensional, incompressible, semi-implicit Eulerian fluid code, with a structural-dynamics finite-element code. PELE-IC is intended for the analysis of air venting and of steam condensation effects on pool dynamics.

Problems of long running time would be aggravated by the inclusion of two-phase capability. It is therefore necessary to emphasize only the key features of the steam-venting process that the scaled-down or even the full-scale (segment) experiments could not resolve.

One such key feature is the effect of wall flexibility on condensation-induced pressure spikes. It may be possible to specify the pulsating flow of liquid at the bottom of the downcomer(s), as induced by the condensation process, and study the pressure field in the single-phase (liquid) pool in the presence of flexible walls. The pulsating flow could be specified on the basis of experimental data on the growth and collapse of steam bubbles. This technique would bypass the need for the difficult two-phase-flow calculations. The amplitude and frequency of pulsating flow could be obtained as a function of the number of active downcomers, their size, immersion depth, the pool liquid temperature, and the steam flow rate.

Whichever technique for analyzing the details of pressure-suppression pool dynamics becomes feasible and operational, the byproduct of studies using the component codes must be a highly simplified, lumped-parameter module, suitable for incorporation into the BEACON/MOD 3 containment systems code. A separate module will be developed for the air-venting and steam-venting phases, for each representative wetwell type.

#### 4.2.3.5 Milestones

- BEACON-MOD 2A to be released to the public (NESC) in December 1978.
- BEACON-MOD 3 to be released to the public in December 1979.
- PELE-IC to be released to the public in September 1979.

### 4.3 COMPONENT CODES

Component codes are designed to provide thermal-hydraulic details for individual system components. The objective is to study component behavior and to learn to what degree models could be simplified for inclusion into a systems code.

#### 4.3.1 PRESENT STATUS

##### 4.3.1.1 SCORE-EVET

SCORE-EVET is a time-dependent multidimensional (x,y,z) code developed at INEL for describing reactor-core transients. It assumes that there is thermal and mechanical equilibrium, that liquid and vapor travel with equal velocities in the same direction, and that liquid and vapor have the same (saturation) temperature. Consequently, only three mixture-conservation equations need be considered. The code was completed during 1976 and documented in 1977 for public release. No plans exist for future improvements of this code because of the current capabilities of TRAC, COBRA-4, and K-FIX.

##### 4.3.1.2 COBRA-IV

The COBRA code was originally developed by Battelle Pacific Northwest Laboratories for reactor-core subchannel analysis. Since 1967, there have been five stages in its development: COBRA-I, II, III, III-C, and IV-1. The last indicated version (IV-1) was sponsored by WRSR and released to the public in mid-1976.

In addition to its subchannel analysis capability, COBRA-IV-1 can be applied to fuel assemblies and to core-wide analysis. Its general features can be summarized as follows:

- a. Homogeneous equilibrium model of two-phase transients.
- b. Geometric representation is quasi-three-dimensional in that there are only two momentum equations for the mixture: one equation for the axial (vertical) direction, and the second for the lateral direction, irrespective of the azimuthal direction of flow.
- c. The mixture density is a function of the local enthalpy only, evaluated at a reference pressure. This reference pressure is independent of location within the core. Sonic propagation effects are therefore ignored.
- d. Turbulent mixing is considered, for single-phase and for two-phase applications.
- e. The mixture-energy equation is solved implicitly in the ACE (Advanced Continuous Eulerian) scheme, which allows steep density gradients to be tracked.
- f. Heat transfer within the fuel rod is solved by the transient orthogonal collocation method. All heat-transfer regimes covered by the "boiling curve" are represented.

Since the public release of COBRA-IV-I,<sup>8</sup> work has begun on extending the code's capability to handle thermal and mechanical nonequilibrium. A new version, COBRA-TF, is presently in the checkout stage. This version allows the user to select (a) a five-conservation-equation drift-flux version, (b) a six-conservation-equation two-fluid version, or (c) a three-conservation-equation drift-flux model. The latter considers the mixture balance equations for mass, momentum, and energy, including terms for vapor drift with respect to the center of gravity of the mixture. Both phases are in thermal equilibrium.

In the five-conservation-equation drift-flux model, the balances include liquid mass, vapor mass, mixture momentum, vapor energy, and liquid energy. An additional (algebraic) equation is utilized for prescribing the drift velocity,  $V_{gj}$ , in both the three- and in five-equation models. In the two-fluid model this is replaced by an additional momentum equation.

All options allow a quasi-three-dimensional representation as described in item (a) above. It should be noted that the five-equation drift-flux model allows for unconstrained thermal nonequilibrium conditions and constrained mechanical nonequilibrium conditions.

The COBRA-TF version is currently being used for scoping analyses on the behavior of upper head injection (an alternative ECC configuration) during blowdown and reflood.

#### 4.3.1.3 K-TIF

K-TIF is a descendant of the Los Alamos Scientific Laboratory's KACHINA code (hence the prefix K). The letters TIF stand for "two incompressible fluids" to identify the modeling assumption. The code is being used for detailed numerical studies of PWR downcomer flow dynamics during the ECC water injection period. It considers a two-dimensional (x,y) geometry, with provisions to account for nozzle flows perpendicular to the arrangement of the computational cells. In this arrangement the downcomer is "unwrapped" so that the annular fluid-gap region is represented by a "slab" whose two vertical sides are numerically coupled.

The mathematical representation of fluid dynamics uses mass, momentum, and energy balance for the liquid phase but only mass and momentum balance for the vapor phase. The vapor is assumed to be at saturation. Constitutive equations are used for local condensation (or vapor generation) rates, for the interphase transports of momentum and of energy, and for the wall shear. Since the downcomer flow occurs at velocities well below sonic, the incompressible-fluid assumption is reasonable.

One of the main purposes of K-TIF is to explore various schemes for modeling the interphase transport processes and to evaluate them by comparison with data from separate-effects (downcomer) experiments. The results of this effort are being used in the TRAC code.

So far K-TIF has been used for simulating steady-state ECC bypass experiments (data from Battelle Columbus Laboratories, the Semiscale facility, and Creare, Inc.). The code description and comparison results are given in Reference 9. Recently K-TIF has been used to simulate ECC bypass transients; results were compared with test data obtained by Creare, Inc. The first version of K-TIF was internally released in December 1977.

#### 4.3.1.4 K-FIX

Developed at the Los Alamos Scientific Laboratory, K-FIX is an updated version of the KACHINA code, featuring a fully implicit exchange numerical technique. It represents the state of the art in the numerical simulation of multidimensional two-phase transients. The version<sup>10</sup> released to the public in 1977 is based on a full two-fluid model (unconstrained thermal and mechanical nonequilibrium conditions) for two-dimensional, axisymmetric geometries.

So far, K-FIX has been used for numerical simulation studies of choked flow and has been transmitted to INEL to serve as a basis for the advanced containment code BEACON/MOD 2. During the latter part of 1977 a three-dimensional capability was added to allow the modeling of two-phase transients within plena.

The main objective of K-FIX is to explore various concepts for modeling interphase transport phenomena. Sensitivity studies and comparisons with data from basic (or model development) experiments are used, and the results are incorporated into TRAC, THCR, and other systems codes.

#### 4.3.1.5 SOLA-FLX

The purpose of the SOLA-FLX code, being developed at the Los Alamos Scientific Laboratory, is to provide numerical simulation of hydroelastic phenomena occurring in the interior of the reactor vessel during the early portion of a LOCA. The final aim is ability to calculate the hydraulic forces acting on the reactor vessel and its internals (e.g., the core barrel) and the resulting deflections and stresses in the internals.

SOLA-FLX couples multidimensional hydraulics calculations (through the SOLA-DF code) with structural dynamics calculations (through the FLX code). SOLA-DF employs a transient drift-flux model that accounts for a constrained thermal and mechanical nonequilibrium in two dimensions. Both single- and two-phase fluids can be considered, including the choked flow at the break.

During 1976 the code was restricted to axisymmetric geometries only and was compared with experimental data obtained by the Systems, Science, Software Company. Work during 1977 was aimed at extending capabilities to handle nonsymmetric transients, both fluid and structural.

### 4.3.2 RESEARCH PROGRAM

#### 4.3.2.1 COBRA-TF

Current work in the development and application of the COBRA code is in response to NRR Research Request RSR 76-6, dated November 8, 1976. COBRA was originally developed for core subchannel analyses and more recently has been expanded to whole-core and even whole-vessel analyses.

Modeling of upper head injection (UHI) and drain for emergency core cooling water, testing against UHI drain tests by Westinghouse, and modeling of UHI reflood in pressurized water reactors were completed during calendar year 1977. Code testing against Westinghouse G-2 heat-transfer tests was performed at the beginning of 1978.

The analysis of an entire LOCA (blowdown, refill, reflood) for a PWR vessel fitted with upper head injection, code sensitivity study, and testing against Semiscale UHI test data are scheduled for completion by December 1978.

Code testing against data obtained at several test facilities (TLTA, PKL, and LOFT) and pretest predictions for three-dimensional reflood tests (upper plenum and slab core) are planned for fiscal year 1979.

#### 4.3.2.2 K-TIF

This code is being used for detailed studies of flow within a PWR downcomer, during the accumulator injection process. The following activities were completed during calendar year 1977:

- a. Two-dimensional modeling of the lower plenum was added, to remove uncertainties in the specification of boundary conditions along the lower edge of the downcomer.
- b. Capability was introduced to handle hot-wall effects.
- c. Further improvements were introduced in the description of the interphase transport of mass, momentum, and energy, and in establishing flow patterns.
- d. Comparisons were made with some 1/15-scale downcomer penetration test data (Battelle Columbus Laboratories and Creare, Inc.), both steady-state and transient.

The first version of K-TIF was completed at the end of December 1977 and was released to the public in April 1978. During fiscal year 1978 this code has been extensively tested against existing test data.

#### 4.3.2.3 K-FIX

Future plans for this general purpose multidimensional code that is capable of handling thermal and mechanical nonequilibrium conditions include:

- a. Further improvements in the modeling of interphase mass, momentum, and energy transfer; in the modeling of momentum and energy transfer from wetted walls (including the reactor vessel internals); and in the description of flow patterns.
- b. Pretest and posttest predictions for full-scale upper plenum tests.
- c. Modeling of heat transfer from fuel rods to allow prediction of the full-scale slab-core reflood test.
- d. Further studies on the numerical simulation of choked flow (for various break geometries).
- e. Continuous transfer of technology to TRAC, THOR, BEACON, and RELAP-5.

#### 4.3.2.4 SOLA-FLX

SOLA-FLX accounts for hydroelastic effects. It is being applied to studies of PWR core-barrel loads during the subcooled phase of blowdown. The code's nonsymmetric hydroelastic calculation capability for PWR core-barrel loads was demonstrated<sup>11</sup> during 1977. Future plans can be summarized as follows:

- a. Pretest and posttest predictions will be performed for HDR blowdown tests. These tests will start during 1979 and will feature a near-full-scale vessel containing a highly flexible core-support barrel. The HDR tests are sponsored by the Federal Republic of Germany.
- b. The code will be documented and released to public (National Energy Software Center) after testing against HDR tests.

- c. The code's capability will be extended to deal with other hydroelastic load situations (e.g., steam-generator divider plate and baffles during a steam-line break).
- d. Technology will be transferred to the TRAC systems code.

The development of SOLA-FLX was started by RSR in anticipation of NRR needs, which were subsequently expressed in various NRR memos.

#### 4.3.3 Milestones

- COBRA-TF (UHI) released internally (to NRC) in 1978.
- K-FIX (two-dimensional) released to the public (National Energy Software Center) in 1977.
- K-TIF released internally in December 1977.
- K-FIX (three-dimensional) to be released internally in September 1978 and externally in April 1979.
- SOLA-FLX released internally in December 1977. Release to the public is scheduled for February 1979.

#### 4.4 CODE ASSESSMENT

##### 4.4.1 DEVELOPMENTAL ASSESSMENT

###### 4.4.1.1 RELAP-4/MOD 5

The developmental assessment of RELAP-4/MOD 5 is reported in Volume III of Reference 5. Calculated results were compared with data for Semiscale test S-02-9, representing a full double-ended cold-leg-break blowdown in a commercial pressurized water reactor. Besides those reported in Reference 5, comparisons were made with data obtained at the Two-Loop Test Assembly (TLTA - a BWR blowdown heat transfer test facility at the General Electric Company) for test No. 22, which simulated a steam-line break. In addition, several comparisons were made with data on the Semiscale downcomer and lower plenum behavior (tests S-01-4A and S-02-4, respectively).

Checkout analyses without data comparisons were reported for the blowdown portions of 6, 25, and 200% cold-leg breaks in representative pressurized water reactors, 6 and 200% recirculation line breaks in a representative boiling water reactor, and a 200% cold-leg break in the Loss-of-Fluid Test (LOFT) Facility.

###### 4.4.1.2 RELAP-4/MOD 6

To date, code results have been compared with data for Semiscale tests S-02-6, S-02-9, and S-03-2; and PWR Full-Length Emergency Cooling Heat Transfer (FLECHT) tests 0085, 7934, 2928, 4444 (low and high flooding rates), 5239, 6638 (high and low pressures), and 5636 and 4930 (different peak power). Comparisons have also been made with data from Semiscale test S-03-6 and PWR FLECHT separate-effects test 3105, both of which are gravity-fed bottom-reflood tests. Finally, one comparison each was performed for LOFT (L1-4) and for TLTA tests.

###### 4.4.1.3 TRAC

Comparisons were made of code results with (a) analytical solutions for U-tube manometer oscillations, (b) RSR Standard Problem 2 (CISE one-dimensional blowdown test with flow reversal and heat addition), (c) NRC Standard Problems 1 and 2 (Edwards' pipe blowdown<sup>12</sup> and Semiscale isothermal blowdown, respectively), (d) a Semiscale blowdown test featuring a heated core, and (e) two 1/5-scale steady-state downcomer-penetration tests (small and large subcooling) performed by Creare, Inc. In addition, checks with the multidimensional modules of TRAC were made to determine its capability to calculate the downcomer and core refill process. The following additional comparisons are to be made before TRAC is released to the public: two FLECHT tests (3920 and 4930), Semiscale test S-04-1 (complete LOCA), a Semiscale small-pipe-break test, and LOFT test L1-4.

###### 4.4.1.4 THOR

Comparisons with test data were reported in Brookhaven National Laboratory's Quarterly Progress Reports (see, for example, References 13 through 17) on NRC Standard Problem 1 and on the CISE experiment referred to in Section 4.4.1.3 above.

#### 4.4.1.5 COBRA-IV and COBRA-TF

Code comparisons with test data are reported<sup>8</sup> for the following:

- a. Nine-rod fuel bundle experiment (General Electric Company).
- b. Inlet jetting (Combustion Engineering, Inc.).
- c. Flow blockage (tests at the Westinghouse Electric Corporation and at Battelle Pacific Northwest Laboratories).
- d. Wire-wrapped bundle.
- e. Application for the analysis of departure from nucleate boiling (DNB).
- f. Core blowdown at the Semiscale facility.
- g. Turbulent plane jet.

More recently, COBRA-TF results were compared with General Electric test data on (a) a sinusoidally varying power input in a heated tube with a constant inlet flow of Freon, (b) exponential power decay at a constant inlet flow, and (c) a flow decay at constant power. COBRA-TF results were also compared with data from Creare, Inc., flat-downcomer tests. Reports on these comparisons have not yet been released to the public.

Finally, COBRA-IV was used in several multidimensional scoping calculations for PWR core blowdown and reflood hydraulics.

#### 4.4.2 INDEPENDENT ASSESSMENT

As already mentioned, only codes that have been released to the public are to be independently assessed. A limited amount of independent assessment has been performed for the RELAP-4/MOD 5 code. This involved running International Standard Problems 3 (CISE test), 5, and 6. The latter featured a small (6%) cold-leg break in the Semiscale facility (test S-02-6). Pretest predictions were also made for NRC Standard Problem 7 (a 200% cold-leg-break isothermal blowdown test in LOFT).

In response to a request from NRC's Office of Nuclear Reactor Regulation (NRR), WRSR has also funded the independent assessment of the WHAM code, which was released to the public by Kaiser Engineers in 1967. It has been widely applied in the analysis of blowdown loads. The Idaho National Engineering Laboratory has used various modeling options, according to NRR specifications, and has compared WHAM results with test data (containment system experiment) obtained at Battelle Pacific Northwest Laboratories.

Independent assessment of RELAP-4/MOD 6 is presently under way. During fiscal year 1979 and beyond, independent assessment will be performed for TRAC (PWR and BWR versions), RELAP-4/MOD 7, THOR, and RELAP-5. To the extent that adequate data sources are available, the advanced systems codes will also receive developmental and independent assessment for anticipated transients without scram and reactivity-initiated accidents.

Whenever possible, independent assessment will stress comparison of "blind" pretest analyses with experimental data, using pretest analyses available from formal assessment tasks, from the NRC Standard Problem Program, and from analyses performed by the experimenters.

Steps have been taken to ensure the availability of pretest analyses of key experiments (made as part of normal test predictions by the experimenters) using the current publicly released versions of codes.

Only systems codes will be independently assessed, emphasizing comparisons with test data obtained at integral test facilities (e.g., Semiscale, LOBI, LOFT, PKL, TLTA, and the Japanese 2000-rod reflood test facility). In addition, code results will be compared with those obtained in some of the important separate-effects tests, such as the downcomer tests at Battelle Columbus Laboratories, FLECHT separate-effects tests, pump loop tests at Combustion Engineering, Inc., the upper plenum test at Kraftwerk Union AG (Germany), full-scale slab-core tests in Japan, blowdown tests at the Power Burst Facility, and the Swedish tests on BWR bundle reflood.

Developmental and independent assessment of these codes will result in recommendations for future work, including improved analysis models, additional measurement requirements, or additional experiments. The results of the independent assessments will be documented for each code in an assessment topical report.

#### 4.5 SUPPORTING EXPERIMENTS FOR THE DEVELOPMENT OF BASIC ANALYTICAL MODELS

##### 4.5.1 PRESENT STATUS

##### 4.5.1.1 Studies of Two-Phase Interactions in Countercurrent Flow

At the University of Houston, analytical models and tests are being developed on interactions between two phases in cocurrent and countercurrent flow, through flooding.

As reported in Reference 18, tests on countercurrent flow of air-water mixtures indicate that the onset of flooding is related to the onset of entrainment.

##### 4.5.1.2 LWR Safety-Phenomena Experiments

The objective of this program at the Los Alamos Scientific Laboratory is to provide data from bench-scale experiments on two-phase flow, particularly countercurrent flow data for an "unwrapped" downcomer in an experimental simulation of the reactor-vessel upper plenum. Two-phase flow instrumentation will also be developed as required.

A bench-scale test system representing an unwrapped PWR downcomer has been constructed, and testing has begun on countercurrent air-water flow. Direct photographic observation with colored water and also flash x-ray photography are being used in the initial tests to establish flow patterns for a variety of geometries and fluid-flow conditions. Proof testing and calibration of hot film sensors have been completed.

The Laboratory has also subcontracted with the Systems, Science, and Software Company to perform small-scale hydroelastic experiments featuring axisymmetric blowdown of a vessel containing a core barrel.

##### 4.5.1.3 Containment Tests

##### 4.5.1.3.1 Pressure-Suppression Tests and Analysis

This program at the Lawrence Livermore Laboratory was initiated in response to NRR Research Request RSR 76-1, to obtain experimental data for (a) scale-model confirmation and licensing of the Mark I pressure-suppression containment design and (b) for code assessment.

Two 1/5-scale test chambers were designed, manufactured, installed, and instrumented. One section represents a 7.5-degree sector of the Mark I torus in a circumferential direction; the second (main) section represents a 90-degree sector. The 7.5-degree sector contains one pair of downcomers; the 90-degree sector has 12 pairs. The chamber in the former is a counterpart of the General Electric Company's 1/12-scale tests. The 90-degree sector was selected to explore multidimensional effects.

The air-venting phase of these experiments has been completed, and a final report has been written. The facility has been "mothballed" pending (a) review of the final report to determine whether additional air-venting tests may be required and (b) the outcome of the steam-venting scaling studies, to determine whether useful information on steam venting could be obtained in this 1/5-scale test facility.

##### 4.5.1.3.2 Studies of Dynamic Loads in Pressure-Suppression Containments

Conducted under WRSR sponsorship at the University of California at Los Angeles, this program is to provide basic experimental data on the transient thermal-hydraulics of air and steam injection into water, to establish scaling relationships for steam venting, and to elucidate free-surface mass transfer phenomena in BWR pressure-suppression pools.

Extensive tests have been performed on the mechanism of vent clearing with air in a laboratory-scale single-vent pressure-suppression model. Hydrodynamic instability of the liquid surface and of air-bubble expansion has been observed and quantified. Studies of the scaling laws for air venting have been completed, and a similar study for steam venting has been initiated.

#### 4.5.1.3.3 Modeling of Pool-Swell Hydrodynamics

In accordance with NRR Research Request RSR 76-4, the objective of this program at the Massachusetts Institute of Technology is to develop scaling laws for LOCA-induced air-carryover pool swelling in a BWR pressure-suppression containment and to confirm these and the scaling laws for uploads and downloads during air venting in laboratory-scale experiments. Tests have been conducted in three laboratory scales. The results show that pool swell, vent clearing, and the resultant hydraulic loads on containment wetwell walls can be properly scaled by the approach proposed by F. J. Moody of the General Electric Company.

A final report was issued in December 1977.

#### 4.5.1.3.4 Marviken Tests

The Marviken power plant (AB Atomenergi) in Sweden was originally built as a boiling heavy-water reactor with natural recirculation. However, for a number of reasons, it was never operated as a nuclear power station. An oil-fired boiler was added, leaving the reactor and reactor building available for testing.

A series of full-scale pressure-suppression containment tests (designated Marviken I) was undertaken in 1972, under multinational sponsorship, including that of the U.S. Atomic Energy Commission. Sixteen tests were run through May 1973, and the results were reported in November 1974.

Because of observed pressure oscillations in the wetwell, the multinational participants (including NRC) agreed to sponsor a second series of tests. In this Marviken II test program emphasis was placed on measurements and studies of system dynamics to determine the cause of the pressure oscillations. The second series of tests was completed during 1976, and the final report was issued toward the end of 1977.

The pressure oscillations originate from steam condensation in the suppression pool. Condensation causes steam bubble collapse and imploding liquid water hammer. Investigations were made of the dependence of such oscillations on various parameters such as mass flow of steam, pool liquid temperature, noncondensable-gas effect, downcomer immersion depth, and downcomer and vent flow areas.

### 4.5.2 RESEARCH PROGRAM

#### 4.5.2.1 Studies of Two-Phase Interactions in Countercurrent Flow

Experiments on cocurrent and countercurrent flows will continue at the University of Houston. The objective is to obtain data on interfacial wave structure, film thickness, and mass and momentum transfer between phases. Experiments will also be performed to determine the axial and radial motion of droplets in the annular mist flow regime, and basic models will be developed.

In fiscal year 1978, NRC has also sponsored, at the University of Houston, a basic study of two-phase flow regimes during steady flows and transients, in the geometries of interest in LWR safety work. This experimental and analytical work is to provide basic information needed in the development of advanced LOCA codes.

#### 4.5.2.2 LWR Safety-Phenomena Experiments

In this experimental program conducted at the Los Alamos Scientific Laboratory, testing of countercurrent two-phase flow with the "unwrapped" PWR downcomer model will be completed. Tests will be conducted on upper plenum deentrainment phenomena, with emphasis on investigating the use of the Storz lens for visual determination of local fluid-velocity vectors, droplet number density, and representative droplet sizes. Computer software will be developed for quantifying the observed phenomena and automatically recording digitized information as functions of location and time on computer tapes.

Also planned are tests on liquid entrainment by steam jets flowing through liquid pools, originating from the simulated upper core-support plate. Countercurrent flow and flooding at the upper core-support plate will also be studied.

#### 4.5.2.3 Small-Scale Upper Plenum Deentrainment Tests

Plans for this program at Harwell, United Kingdom, include a study of liquid-droplet reentrainment from an established falling film on a simulated support column and a control-rod tube, in

the presence of air or steam flow directed at various angles of incidence. Also studied will be the formation of this falling liquid film by the impingement of liquid droplets carried by a gas (air or steam) stream directed at various angles of incidence. The study of liquid-droplet deentrainment and reentrainment will be expanded to (a) a single row of simulated support columns and control-rod guide tubes, and (b) multiple rows.

Analytical models will be developed for the above-described individual and combined processes. The desired end product is a simplified model of the volumetric "sink" for the liquid fraction carried by steam in the PWR upper plenum and the rate at which liquid fallback replenishes the liquid accumulating above the upper core-support plate. These simplified models will be used in the TRAC systems code.

#### 4.5.2.4 Containment Tests

The pressure-suppression tests and analyses described in Section 4.5.1.3 will be completed. The MIT study has been expanded to cover studies of scaling laws for steam venting. In addition, detailed studies of air and steam venting through horizontal vents typical of the Mark III pressure-suppression containment will be initiated.

#### 4.5.2.5 Large-Scale Critical Flow Tests

Contractual arrangements have been made among NRC, the Electric Power Research Institute, France, and the Nordic countries to measure critical flow through large (up to 500 mm I.D.) nozzles at the Marviken test facility. These Swedish tests, named MX-III-CFL (or Marviken III), will emphasize the critical flow of subcooled and very low quality fluid, and will use a variety of independent measurements. Preparation, including two shakedown tests conducted in December 1977, was completed during 1977. Test completion is expected during 1979.

### 4.6 CODE UNCERTAINTY STUDIES

The approach is based on a response-surface representation of the output of a code, for a particular LWR system. The effect of uncertainties in input variables on a key output variable (peak cladding temperature) is then obtained from the response surface. The ability of the response-surface approach to provide a good representation of code output has been demonstrated at the Idaho National Engineering Laboratory and at Sandia Laboratories by analyses with a simplified model of a PWR system blowdown. The objective was to determine the capability of the response-surface approach and the ability to select code input uncertainties.

A similar (parallel-path) effort is sponsored by WRSR at the Los Alamos Scientific Laboratory. Reference 19 presents two types of sampling plans as alternatives to random sampling. These plans are shown to be improvements over random sampling with respect to the variance of a class of estimators including the sample mean and the sample cumulative distribution function. The partial rank correlation coefficient is presented as a measure of sensitivity.

At present WRSR is sponsoring at Sandia Laboratories an evaluation of the uncertainty in the best estimate prediction of the peak cladding temperature in a typical PWR, using the existing LOCA code (RELAP-4/MOD 6, Version 3).

### 4.7 DEVELOPMENT OF EVALUATION MODEL CODES

#### 4.7.1 USER NEEDS

By prior (1974) agreement between the regulatory and the research offices of NRC, responsibility for developing present-generation evaluation model codes was retained by the regulatory office. The research office assumed responsibility for developing best estimate codes and the advanced code candidates for future evaluation model codes. The bulk of this Branch Program Plan reflects that policy, which was still in existence at the time this plan was drafted.

In June 1977 the Office of Nuclear Reactor Regulation requested the Office of Nuclear Regulatory Research to also sponsor work on an updated package of evaluation model codes suitable for audit calculations of LOCA consequences in both PWR and BWR plants. The goal is to provide NRR, as quickly as possible, with a code package that would be "frozen" for a few years, pending completion of the advanced evaluation model code. The user (NRR) stated that there was a high-priority need for the BWR LOCA evaluation model package.

## 4.7.2 STRUCTURE OF EVALUATION MODEL CODE PACKAGES

### 4.7.2.1 BWR LOCA

The structure of the evaluation model code package for BWR LOCA analysis is shown in Table 4-1.

All codes listed in the second column of Table 4-1 will be modules in the evaluation model package called WRAP (Water Reactor Analysis Package), now under development at the Savannah River Laboratory (SRL). As explained in Sections 4.2.1.1.5 and 4.2.1.2.3, WRAP is ideally suited for linking various codes together since it is based on a powerful SRL-development data management system named JOSHUA. The RELAP-4/MOD 5 code developed by EG&G has already been restructured at SRL and incorporated into WRAP in a user-convenient format requiring minimal input preparation.

The NRC does not at present have a computer code for the reflood phase of a BWR LOCA. On the other hand, under the Information Exchange Agreement between NRC and the Nordic countries (known as the NORHAV Agreement), the RISØ Research Institute in Denmark has agreed to develop a BWR reflood code (NORCOOL) to NRC specifications. The first version, NORCOOL-1, was made available to NRC in December 1977. This best estimate code will be converted to the required evaluation model format at SRL.

TABLE 4-1  
STRUCTURE OF EVALUATION MODEL PACKAGE FOR BWR ANALYSIS

<u>Phenomenon</u>	<u>Code (Developer)</u>	<u>Input from</u>	<u>Output to</u>
Initialization of fuel thermal condition	GAPCON-2 (BPNL)	User	RELAP-4/MOD 5, MOXY
System blowdown	RELAP-4/MOD 5 (EG&G and SRL)	GAPCON-2 user	NORCOOL-1, MOXY
System reflood	NORCOOL-1 (RISØ, Denmark)	RELAP-4/ MOD 5, user	MOXY, user
Fuel heatup (hot channel)	MOXY (EG&G and NRR)	GAPCON, RELAP, NORCOLL, user	User

### 4.7.2.2 PWR LOCA

The structure of the evaluation model code package for PWR LOCA analysis is shown in Table 4-2.

All of the codes listed in the second column of Table 4-2 will form an integral part of the WRAP package being developed at SRL. The existing RELAP-4/MOD 5 code cannot calculate the refill phenomena. The main reason lies in the basis assumption of thermal equilibrium, which causes drastic depressurization when cold emergency core cooling water and steam are present in the same control volume. Consequently, the bridge between the PWR blowdown and reflood analyses was always provided by hand calculation. The recipe for this particular calculation will be provided by EG&G. The Savannah River Laboratory will incorporate this side calculation into its WRAP code package, eliminating the need for human intervention (hand calculation).

The fuel behavior code FRAP-T is presently being modified, through a cooperative effort by EG&G and Battelle Pacific Northwest Laboratories, to reflect the evaluation model requirements. The product will be named FRAP-TEM.

### 4.7.3 SCHEDULES

Present schedules specify the release of the complete BWR LOCA package by December 1978. The Office of Nuclear Reactor Regulation is responsible for supplying the Savannah River Laboratory with the GAPCON-2 and MOXY codes; EG&G is responsible for supplying various subroutines that must be added to RELAP-4/MOD 5 to meet NRR requirements.

The timetable for completing the PWR LOCA evaluation model package is not yet definite. The bulk of the work will start immediately on completion of the BWR LOCA package described in Section 4.7.2.1 above. It should be noted that the WRAP modules for PWR system self-initialization and renoding (on restart) have already been completed and demonstrated, as part of RSR-sponsored research at the Savannah River Laboratory.

TABLE 4-2

STRUCTURE OF EVALUATION MODEL PACKAGE FOR PWR ANALYSIS

<u>Phenomenon</u>	<u>Code (Developer)</u>	<u>Input from</u>	<u>Output to</u>
Initialization of thermal condition	GAPCON-2 (BPNL)	User	RELAP-4/MOD 5, FRAP-TEM
System blowdown	RELAP-4/MOD 5	GAPCON-2 User	Refill module, FRAP-TEM
System refill	Refill module in WRAP (EG&G, SRL)	RELAP-4/MOD 5, User	Reflood code, FRAP-TEM (see below)
System reflood	Two options: FLOOD/MOD 1 and RELAP-4/FLOOD (EG&G)	Refill module, user	FRAP-TEM, user
Hot channel (fuel behavior)	FRAP-TEM (EG&G, BPNL)	User, GAPCON-2, RELAP-4/MOD 5, refill module, reflood code	User

#### REFERENCES\*

1. U.S. Atomic Energy Commission, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, Docket No. RM-50-1, December 1973. Available for inspection and copying for a fee in the NRC Public Document Room.
2. S. Fabric, "Review of Existing Codes for Loss-of-Coolant Accident Analysis", Advances in Nuclear Science and Technology, 10, 365-404 (1977).
3. G. B. Wallis, One-Dimensional Two-Phase Flow, McGraw-Hill Book Company, New York, 1969.
4. S. Fabric, "Data Sources for LOCA Code Verification," Nuclear Safety, 17, No. 6, 671-685 (November-December 1976).
5. RELAP4/MOD5: A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems; User's Manual, Volumes I, II, and III, USNRC Report ANCR-NUREG-1335, September 1976.
6. G. W. Johnsen, F. W. Childs, and J. M. Broughton, A Comparison of "Best-Estimate" and "Evaluation Model" LOCA Calculations, the BE/EM Study, EG&G Idaho, Inc., Report PG-R-76-009, 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
7. K. D. Lathrop, comp., Reactor Safety and Technology, Quarterly Progress Report, October 1-December 31, 1976, USNRC Report LA-NUREG-6698-PR, February 1977.
8. C. W. Stewart et al., COBRA-IV: The Model and The Method, BNWL-2214, July 1977.
9. A. A. Amsden, T. D. Butler, and F. H. Harlow, Numerical Study of Downcomer Flow Dynamics, USNRC Report LA-NUREG-6797-SR, May 1977.
10. W. C. Rivard and M. D. Torrey, K-FIX: A Computer Program for Transient, Two-Dimensional, Two-Fluid Flow, USNRC Report LA-NUREG-6623, April 1977.
11. J. K. Dienes, C. W. Hirt, and L. R. Stein, Computer Simulation of the Hydroelastic Response of a Pressurized Water Reactor to a Sudden Depressurization, USNRC Report LA-NUREG-6772-MS, April 1977.
12. A. R. Edwards and T. P. O'Brien, "Studies of Phenomena Connected with the Depressurization of Water Reactors," Journal of the British Nuclear Energy Society, 9, No. 2, 125-135 (1970).
13. P. G. Kroeger, G. S. Srikantiah, and W. Wulff, "THOR Code Development," in Reactor Safety Research Programs, Quarterly Progress Report, January 1-March 31, 1977, USNRC Report BNL-NUREG-50661, May 1977, pp. 159-191.
14. G. S. Srikantiah, P. G. Kroeger, and W. Wulff, "THOR Code Development," in Reactor Safety Research Programs, Quarterly Progress Report, April 1-June 30, 1977, USNRC Report BNL-NUREG-50683, August 1977, pp. 169-192.
15. P. G. Kroeger, G. S. Srikantiah, and W. Wulff, "THOR Code Development," in Reactor Safety Research Programs, Quarterly Progress Report, July 1-September 31, 1977, USNRC Report BNL-NUREG-50747, December 1977, pp. 193-207.
16. W. Wulff, P. G. Kroeger, and G. S. Srikantiah, "THOR Code Development," in Reactor Safety Research Programs, Quarterly Progress Report, October 1-December 1, 1977, USNRC Report BNL-NUREG-50785, February 1978, pp. 186-203.
17. W. Wulff and P. Saha, "THOR Code Development," in Reactor Safety Research Programs, Quarterly Progress Report, January 1-March 31, 1978, USNRC Report BNL-NUREG-50820, May 1978, pp. 136-150.

---

\*The USNRC reports listed here are available for purchase from the National Technical Information Service, Springfield, Va. 22161.

18. A. E. Dukler and L. Smith, Two-Phase Interactions in Countercurrent Flow Studies of the Flooding Mechanism: Progress Report for November 1, 1975-September 30, 1976, USNRC Report NUREG-0214, June 1977.
19. M. D. McKay, W. J. Conover, and D. E. Whiteman, Report on the Application of Statistical Techniques to the Analysis of Computer Codes, USNRC Report LA-NUREG-6526-MS, September 1976.

CHAPTER 5  
PROGRAM PLAN  
FOR THE METALLURGY AND MATERIALS RESEARCH BRANCH

5.1 INTRODUCTION

The Metallurgy and Materials Research Branch program is concerned with the integrity of the primary-system pressure boundary in light-water reactors (LWRs). It is an experimental and analytical program designed to upgrade the bases for design, fabrication, operation, and inspection criteria, as well as for the analytical procedures required to evaluate performance under normal, upset, accident, and faulted conditions. Thus a primary goal is to improve the definition of failure probabilities and failure modes, and to establish ways by which the failure probabilities can be reduced if this is considered necessary.

The primary system integrity program consists of three major areas of research: (a) fracture and structural mechanics, (b) operational effects, and (c) flaw detection and evaluation.

The fracture and structural mechanics work encompasses (a) reactor vessel and piping-system performance under pressure and thermal loading; (b) crack initiation, propagation, and arrest (including static and dynamic studies and the use of irradiated specimens); and (c) response to all operational and postulated conditions. In particular, the work on vessel response to postulated conditions includes thermal shock and steam-line-break accident conditions to assess the effects of abnormal pressures and thermal shock following the injection of cold emergency core cooling (ECC) water after various loss-of-coolant accidents (LOCAs) or steam-line breaks.

The operational effects work is directed at obtaining data on (a) irradiation embrittlement, (b) annealing and re-irradiation, (c) residual element effects, (d) cyclic crack growth, (e) steam-generator tube integrity, (f) intergranular stress-corrosion cracking and sensitization, and (g) neutron dosimetry.

The flaw detection and evaluation work covers (a) improved ultrasonic characterization of flaws, (b) acoustic emission studies of flaw growth in piping and pressure vessels and of flaws produced during welding, (c) improved eddy-current inspection of steam-generator tubing, and (d) advanced nondestructive examination techniques.

Special attention is given to the study of the primary-system pressure boundary in LWRs because of the need to contain the nuclear core materials at all times and thus the need to understand the types of failures that might lead to a breach of this containment. The primary-system pressure boundary of current LWRs includes (a) a steel pressure vessel with a thickness of 6 to 12 inches, (b) steam-generator tubes in the case of pressurized water reactors (PWRs), and (c) primary piping as much as 4 inches in thickness and including pumps and valves. Typical materials for reactor pressure vessels are A533B Class 1 plate and A508 Class 2 steel forgings; typical materials for piping are A106B, A516 Grade 70, and type 304 stainless steel. These materials have been studied extensively to develop information on mechanical behavior under appropriate conditions of temperature, stress, neutron irradiation, and reactor environment.<sup>1</sup> The studies have been typically performed on laboratory-scale specimens because the section thickness and massive size of reactor components, coupled with the necessity of simulating long-term neutron irradiation, make testing of full-scale vessels or components either prohibitively expensive or almost technically unfeasible. Nevertheless, the in-reactor behavior of full-section-thickness materials and components must be predictable from data obtained largely in laboratory-scale tests. An important aspect of the work, therefore, is to test materials in a range of thicknesses to validate the analytical prediction techniques. Thus, despite existing knowledge on the properties of primary-system component materials, improvements in information are still sought to round out the basis for judgments affecting continuing reactor safety.

The approach used for the Branch plan is to identify the areas of greatest importance to LWR safety and to initiate high-priority studies on problems in those areas. This identification is mainly accomplished by being responsive to the needs of the Office of Nuclear Reactor Regulation (NRR). The research programs are then formed to develop analysis criteria, or testing procedures, or new materials, with all results ultimately being incorporated into new or improved NRR positions as well as Federal and industry code rules and standards.

## 5.2 FRACTURE AND STRUCTURAL MECHANICS

### 5.2.1 OBJECTIVES

The objectives of the fracture and structural mechanics work are as follows:

- To develop fracture-analysis procedures and design criteria for predicting the stress levels and flaw sizes required for crack initiation and subsequent propagation and/or arrest in LWR pressure vessels and primary piping under elastic, elastic-plastic, and fully plastic conditions.
- To test and validate experimentally procedures, criteria, and design curves [e.g., the reference fracture-toughness ( $K_{IR}$ ) curve], using a range of thin (less than 1 inch) to thick (up to 12 inches) specimens and model vessel tests.
- To show that slow-load fracture toughness, rapid-load fracture toughness, and crack-arrest toughness results obtained with small laboratory specimens are comparable to similar results obtained with thick-section ( $\leq 4$  inches) laboratory test and structural prototype specimens, both unirradiated and irradiated. Investigations are planned to permit the results to be incorporated into fracture-toughness data libraries for unirradiated and irradiated materials.
- To provide further experimental justification for the analytical methods used in predicting the extent of crack propagation that could occur in a hot reactor vessel subjected to the injection of cold ECC water following a large or small LOCA or a steam-line break.
- To examine and confirm the adequacy of current structural design criteria for nuclear piping systems and isolated and closely spaced pressure vessel nozzles.

### 5.2.2 PRESENT STATUS

Fracture toughness and crack arrest in LWR vessel and piping materials have been studied extensively by many organizations over the years. Much of the data was obtained with relatively small laboratory specimens, 1 inch or less in thickness.<sup>1-5</sup> However, some very useful correlations have been made between small-specimen data and data obtained with 8- to 12-inch-thick compact specimens and 12-inch-thick dynamic tear specimens.<sup>6,7</sup> Pipe-rupture studies have been conducted in the ductile regime, using piping generally less than 2 inches thick.

At present, the fracture-toughness criterion is governed by the reference fracture-toughness ( $K_{IR}$ ) curve of the ASME Boiler and Pressure Vessel Code. This curve defines the most conservative limits, or the so-called lower bound, of all available valid data on slow-load fracture toughness, rapid-load fracture toughness, and crack-arrest toughness for specimens up to 12 inches thick. Because only a few heats of steel were used to obtain the data for this  $K_{IR}$  curve, additional data from more heats and more product forms of material are being obtained.

Crack initiation is governed by slow-load fracture toughness or rapid-load fracture toughness, whereas crack arrest is presently defined by the limits of the  $K_{IR}$  curve. Crack arrest has been actively studied in recent years.<sup>8</sup> The methodology for predicting when, where, and under what conditions a running crack will stop has now been established for specimen geometries,<sup>8</sup> and experimental validation is under way.

Piping systems are complex, beginning at the reactor vessel as nozzles, and include tees, reducers, elbows, junctions, and straight runs of various thicknesses, diameters, and materials. Because the potential accident conditions are so varied, many supports, hangers, snubbers, and restraints are used. This complexity lends itself to the inclusion of more, rather than fewer, hangers, restraints, etc., which ultimately could lead to a significant lessening of piping-system flexibility and thus reduce its ability to safely respond to accident-type loadings.

With regard to the requirement for maintaining pressure vessel integrity following the injection of ECC water after a LOCA, a fracture-analysis methodology has been developed and validated in test cylinders to permit a rational integration of all pertinent factors, including the applicability of linear elastic fracture mechanics (LEFM). This includes crack initiation and arrest, thermal stress gradients, and the alteration of these parameters as a function of time, crack depth and length, vessel material properties, and temperature range. An application of this analogy is reported in Reference 9, both for test cylinders and for a reference calculational model of a reactor pressure vessel.

### 5.2.2.1 Vessel Performance

Intensive research over the past few years has resulted in the development and validation of linear elastic fracture mechanics as a fracture-analysis methodology for pressure vessels and piping with cracks having sufficient constraint, under elastic stresses, and at or below the transition temperature.<sup>10,11</sup> The criterion based on this method is embodied in ASME Code Section III, Appendix G, and has been validated for slow-load fracture toughness with up to 12-inch specimens (steel plate O2 from the Heavy Section Steel Technology Program) and for rapid-load fracture toughness with up to 8-inch specimens. Criteria for fracture-safe operations under elastic-plastic and fully plastic conditions are now of interest. This is because the arrest of a crack initiating in a local brittle region, for example, may be dependent on the increased toughness of the surrounding material. To quantify the transition from brittleness to increasing toughness, additional information is needed on material responses (e.g., possible crack initiation, propagation, and arrest) under appropriate test conditions of temperature, stress, and radiation-induced changes, and their gradients. Research is under way at many laboratories and universities to develop an elastic-plastic fracture-analysis criterion based on both the J-integral<sup>12</sup> and the equivalent energy concepts,<sup>13</sup> as well as to predict the elastic-plastic stress state at the tip of a crack by means of a three-dimensional finite-element computer code. These techniques must be studied carefully from a theoretical standpoint for both pressure vessels and piping. The results must then be validated by experiments on a spectrum of test configurations, ranging from small to large specimens and to vessels. Unirradiated materials must be used to validate the criterion, but it must subsequently be validated with irradiated or simulated-irradiated materials as well. Vessels and piping must be tested under both hydraulic and pneumatic (or sustained) loadings.

Fracture-toughness and crack-arrest criteria developed from tests with small and large specimens should be substantiated with tests on suitably thick pressure vessel configurations under loadings at elevated and normal temperatures. Such tests will provide the experimental data required for comparison with predictions of flaw sizes and stress levels leading to failure, as well as failure mode and effect on safety.

Pressure tests of 6-inch-thick pressure vessels have been carried out with carefully sized flaws placed in walls and in nozzle regions, under a series of temperature and stress conditions.<sup>14</sup> The flaw regions were formed either from brittle electron-beam welds to produce a "natural" crack as a result of a dynamic crack pop-in or were fatigue precracked. The tests were designed to produce very long ( $\sim 12$  inches) and deep (one-half the thickness) cracks for evaluating elastic and elastic-plastic loading criteria and methodologies. The results will aid in establishing the relationship between the fracture-toughness and crack-stability criteria, and the details of the ultimate fracture, and hence will aid in establishing the design margin of safety against failure.

Sustained-load testing of a vessel has been conducted to investigate the extent of fracture that would occur if a reactor vessel were to fail in a fully pressurized mode. Specifically, the investigation considered the progress and extent of crack propagation that could result from the stored energy available in a high-temperature system (essentially loading resulting from system blowdown). The test showed that even though a through-the-wall-crack occurred at ductile shelf toughness temperatures, no further propagation occurred as a result of the sustained load.

Design of nuclear piping systems requires an accurate structural knowledge of the entire system, both static and dynamic. This entails knowing the stress indices and flexibility factors for such elements as tees, elbows, reducers, and joints. The design and relative placement of nozzles are critical factors in ensuring a safe structure.

### 5.2.2.2 Crack Arrest

Standard test methods and specimen geometries were submitted to ASTM Committee E-24 in March 1977 to be considered as a tentative standard. An international cooperative testing program has been organized with 29 participants. The objective is to test the applicability of the proposed methods for measuring crack-arrest toughness in the range of practical interest and to define the clarifications and refinements that are found to be necessary. Two-dimensional, dynamic, finite-element analyses are developed for cylinder geometries that will be applicable to test cylinders as well as reactor vessels. They have been used to analyze the thermal shock experiments and will be used to analyze reactor vessels subjected to ECCS operation following a LOCA. Thick specimens will be tested to determine the relationship between crack-arrest toughness measurement capacity and specimen thickness. Irradiations are beginning.

### 5.2.2.3 Response to Accident Transients

Four thermal shock tests have been performed on cracked steel cylinders with degraded toughness properties somewhat simulating irradiation embrittlement. The results confirmed predictions of crack noninitiation under severe thermal shock loadings as well as crack initiation and estimated crack-arrest positions. It has been shown that linear elastic fracture mechanics does characterize the thermal shock methodology and that operational pressure-temperature transients can be accurately analyzed. The beneficial effects of warm prestressing are currently being studied in more detail because this type of treatment is expected to severely limit crack initiation, and thereby penetration, in the thickness direction.

### 5.2.2.4 Licensing Criteria

Linear elastic fracture mechanics is the analytical procedure presently used by the NRC for evaluating the relationships among stress, flaw size, and fracture toughness for the onset of rapid fracture in the nonductile range.<sup>15</sup>

Reactor piping is designed by architect/engineers using various code and evaluation procedures with overall guidance from the ASME Boiler and Pressure Vessel Code and a series of NRC regulatory guides on such matters as pipe whip and pipe restraints.

A crack-arrest criterion for a running crack can now be defined as the minimum in the curve of the imposed stress intensity factor,  $K_{IDmin}$  or  $K_{Ia}$ , versus crack velocity. This increases confidence in the use of the  $K_{IR}$  curve for crack arrest, since all data show the  $K_{IR}$  curve to be conservative.<sup>8</sup> The currently imposed criteria for fracture toughness and crack arrest in LWR materials appear in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G ("Protection Against Nonductile Fracture") and Section XI, Appendix A ("Evaluation"), and in 10 CFR Part 50, Appendix A ("General Design Criterion 31--Fracture Prevention of Reactor Coolant Pressure Boundary") and Appendix G ("Fracture Toughness Requirements").

Regulatory Guide 1.2,<sup>16</sup> "Thermal Shock to Reactor Pressure Vessels," describes a suitable way to implement General Design Criterion 31. The essence of Criterion 31 and the consequent requirement for thermal shock is that "the reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized."

## 5.2.3 RESEARCH PROGRAM

### 5.2.3.1 Vessel Performance

The objective of this research program is to develop the theory and methodology for elastic-plastic and fully plastic fracture analyses of reactor pressure vessels and primary piping. The analysis procedure must permit quantitative prediction of allowable flaw sizes and stress levels using appropriate toughness values under all operating conditions (normal, upset, emergency, and faulted). It must consider the onset of crack extension under slow (static) and rapid (ATWS pressure spike) loading conditions. Concepts to be used include (but are not limited to) the J-integral and equivalent energy methods. The fracture-toughness measurement method and standard specimens must be developed and submitted for adoption as standards by such organizations as ASTM and ASME. Furthermore, specimens and test methods must be evaluated for adequacy in reactor surveillance. Toughness data pertaining to the onset of slow, stable tearing, the onset of rapid crack extension (slow and rapid load), and crack arrest ( $K_{IDmin}$  and  $K_{Ia}$ ) must be obtained by standard methods from multiple heats of appropriate specimens for vessels and piping. In the unirradiated condition, this should include plate, forging, cast, weld, and weld heat-affected-zone material. When small specimens are employed for this purpose, it must be shown that the results conform with large-specimen results. Data and results must be applicable to the validation or modification of the  $K_{IR}$  curve. Trends should be developed with small test specimens. However, the theory, analytical procedures, and testing methods must be validated with large specimens.

Validation of vessel repair-weld procedures is being pursued by pressure testing of intermediate-size vessels containing cracks in or near the repair welds.

### 5.2.3.2 Piping

To study the effects of interacting nozzles, finite-element computer codes are being developed that will evaluate first two and then three interacting nozzles. The ASME Code defines "interacting" as a case in which the perturbation in the stress field between the nozzles is greater

than 10%. The inclusion of a safety injection nozzle into the reactor pressure vessel near the main nozzles is the primary reason for studying three interacting nozzles. In addition, piping-system failures are being characterized to determine the "weak link" locations and to evaluate the load-carrying ability of each part of a piping system, considering thermal, pressure, fit-up, and dynamic loadings. These data will then be used to evaluate the structural integrity of piping systems and to identify those parts of the system that should be modified. In particular, the requirements for hangers, snubbers, and pipe-whip restraints will be critically examined to ensure that they do result in a lower probability of failure; if not, the requirements will be modified.

#### 5.2.3.3 Crack Arrest

Theory and methodology are being developed for the analysis of crack arrest in reactor pressure vessels for situations such as thermal shock (resulting from ECCS operation) and embrittled region crack pop-in. The analysis procedure will permit prediction not only of whether or not a crack will be arrested but also of the conditions for crack arrest (crack size in a given wall thickness, etc.), in terms of the nature of the event, the initial operating conditions, the structural configuration and the initial crack configuration, and the appropriate toughness criterion. The relation between rapid-load fracture toughness and change in crack length as a function of temperature and irradiation is being used as input to the structural analysis of crack arrest. The analysis considers the influence of the kinetic energy of the structure (during crack extension) on crack arrest by evaluating the "path dependency" of the process, including multiple events of onset of crack extension and arrest. The initial analysis will be based on linear elastic material behavior; the need for developing an elastic-plastic analysis will be evaluated from (and based on) the elastic analysis results. The crack-arrest specimens and test method and analysis must lead to the adoption of standards by such organizations as ASTM and ASME. The specimens and test methods must be evaluated for adequacy in reactor surveillance. Validation is required for the initiation of a running crack in simulated-embrittled material in a pressure vessel configuration. The objective is to confirm the capability of the crack-arrest methodology to predict the arrest of a running crack before it reaches a critical size.

#### 5.2.3.4 Response to Accident Transients

Tests and analyses are in progress to establish the feasibility of performing a thermal shock test on a degraded cylinder to demonstrate the beneficial effects of warm prestressing. Consideration is being given both to the use of a liquid-nitrogen spray with a special coating on the metal surface to enhance the thermal shock and to the modification of an existing high-temperature gas test facility for cryogenic capability.

#### 5.2.3.5 Milestones (by Fiscal Year)

##### Vessel Performance

- 1977 Sustained loading analysis confirmed.  
Stress intensity factor vs pressure confirmed for nozzle-corner cracks.  
Confirmatory research on stress interactions between nozzles.  
Piping system research/design practice quantified.  
Weld repair toughness tested in ITVs.
- 1978 Dynamic bend beam test practice submitted to ASTM.  
Code recommendations for multiple interacting nozzles submitted.  
Structural integrity of weld-repaired pressure vessels established.  
Correlations of elastic-plastic and slow-load fracture toughness to Charpy V-notch results extended to irradiated materials.
- 1979 Elastic-plastic analysis methodology confirmed for structural models.
- 1980 Data on initial fracture toughness and cyclic crack growth rate developed for higher strength steels (unirradiated).  
Research on multiple pipe interactions and snubbers confirmed.
- 1981 Data on fracture toughness and cyclic crack growth rate confirmed for higher strength steels.

##### Crack Arrest

- 1977 Standard test specimen and practice for crack-arrest toughness recommended.  
Two-dimensional analysis and dynamic evaluation of ASTM standard specimens completed.  
Weld-line effects on crack propagation and arrest established.

- Data on initial crack-arrest toughness developed for the  $K_{IR}$  curve of unirradiated specimens.
- 1978 Crack-arrest methodology validated.  
Data on initial crack-arrest toughness developed for the  $K_{IR}$  curve of irradiated specimens.  
Combined thermal/pressure stress confirmed for crack arrest.
- 1979 Standard crack-arrest surveillance specimen developed.  
Crack-arrest methodology confirmed in model structures.  
 $K_{IR}$  curve incorporating crack arrest updated or confirmed.  
Crack penetration from combined thermal and pressure loads established.
- 1980 Procedures recommended for analysis and design to ensure crack arrest in LWRs.  
Crack-arrest behavior confirmed for new steels.

### Response to Accident Transients

- 1977 Thermal shock analysis confirmed.  
Dynamic arrest analysis of thermal shock confirmed by results from the Heavy Section Steel Technology Program.  
Warm-prestressing benefits quantified.
- 1978 Thermal shock, small-break, and steam-line-break analysis developed.  
Sensitivity analysis completed on factors causing crack penetration due to thermal shock.
- 1979 Warm-prestressing benefits confirmed on structures.  
Safety analysis methodology validated for steam-line-break accidents.

## 5.3 OPERATIONAL EFFECTS

### 5.3.1 OBJECTIVES

The objectives of the operational effects work are as follows:

- To extend the data and techniques applied in evaluating fracture toughness and crack arrest in unirradiated materials to neutron-irradiated materials, especially base and weld metal with potentially limiting properties in the irradiated condition.
- To develop methods for mitigating radiation-induced embrittlement through post-irradiation heat treatment (annealing), along with a thorough understanding of re-irradiation effects and subsequent annealings and re-irradiations.
- To develop an understanding of the causes of radiation-induced embrittlement and show how radiation-resistant materials can be produced.
- To establish the magnitude and characteristics of crack initiation and growth rate from cyclic loading in LWR vessels and piping.
- To confirm steam-generator tube burst strength and integrity.
- To develop a predictive capability for stress-corrosion cracking in steam-generator tubing.
- To evaluate susceptibility to intergranular stress-corrosion cracking in the welds of stainless steel piping.
- To develop and validate methods for neutron dosimetry and applications to reactor vessel surveillance irradiations.

### 5.3.2 PRESENT STATUS

The important effects of neutron bombardment on reactor vessel beltline structural materials include an upward shift in the reference nil-ductility transition temperature by several hundred degrees Fahrenheit, a reduction in ductile shelf-level energy absorption strength, and a reduction in tensile ductility. All of these factors combine to reduce fracture toughness and the potential for crack arrest. Because of the problems inherent in research on irradiated materials--including space limitations in reactors, shipping-cask sizes, and radiation limits

for hot cells--most research on irradiation effects on vessel materials has been conducted with test specimens 1 inch or less in thickness. However, initial correlations have been made between results from small specimens and those from 2- and 4-inch-thick compact specimens tested in 1975.<sup>17</sup> Systematic studies of post-irradiation heat treatment at temperatures above the operating temperature of irradiated steel have shown that a significant portion of the pre-irradiation toughness can be recovered in this way to extend the useful life of reactor vessels with renewed fracture-toughness capability.<sup>1</sup>

Neutron-induced embrittlement in ferritic pressure vessel steels has been studied extensively.<sup>1,18</sup> It has been shown that embrittlement can be significantly reduced simply by complying with the draft ASTM recommendation for upper limits of 0.10 wt% copper and 0.012 wt% phosphorus in the chemical composition of the steel. Furthermore, a mechanism by which copper affects neutron embrittlement in steels has been proposed.<sup>19</sup>

A high mean load has been observed to cause increases in the cyclic crack growth rate of steels.<sup>20</sup> Thus the growing base of fatigue and cyclic crack growth rate data is being continuously reviewed to ensure that it is adequately conservative in light of these and other findings.

Environmental effects in steam generators have caused wastage, cracking, and denting of tubing. Denting is particularly insidious because the large eddy-current signal from the dent itself precludes detection of any other degradation in the dented region. The large strains in the dented regions have caused stress-corrosion cracking from the primary side, and the cracking has been detected only after tube failure. The present concepts of the influence of crack size, wastage, and denting on tube integrity during both primary and secondary system overpressures are being validated.

Stress-assisted intergranular corrosion cracking in the BWR coolant environment continues to occur in seamless small- and intermediate-diameter austenitic steel piping; it has not been observed in large-diameter piping. The primary factors causing this phenomenon are known. They include oxygen in the coolant, high stresses, and sensitization of the stainless steel. The exact combination of factors that actually produces cracking has not yet been conclusively established.

Present procedures for determining susceptibility to intergranular attack are given by ASTM Standard A262-70.<sup>21</sup> The tests described in this standard detect susceptibility in austenitic stainless steels to intergranular attack by highly corrosive chemical solutions in the absence of stress and have been found unreliable for application to BWRs. The required improvement is a test that will detect susceptibility to intergranular stress-corrosion cracking in the heat-affected zones of austenitic stainless steel welds for BWR service.

Procedures currently exist for determining the neutron flux and fluence from surveillance irradiations in both experimental test reactors and power reactors. However, the variance associated with the calculations and application/correlation procedures is greater than uncertainties in the associated mechanical property measurements. Techniques and procedures developed for fast-reactor neutron dosimetry analysis are now being applied to LWR neutron dosimetry.

#### 5.3.2.1 Irradiation Effects

One series of irradiated compact specimens <4 inches thick has been tested. The results showed that toughness does increase rapidly at higher temperatures and that the shift in transition is reasonably well predicted by data from small specimens.<sup>17</sup> Second and third irradiations of low-ductile-shelf weld-metal compact specimens up to 4 inches thick have been completed, and work is under way to validate methods for the specimen tests.

#### 5.3.2.2 Annealing

Radiation-induced embrittlement of reactor vessel steels can be mitigated by post-irradiation annealing, and trends have been established for the required time at temperature. This work, done largely with Charpy V-notch specimens, has developed preliminary data on the recovery of both the transition temperature and the ductile shelf toughness. Very little data is available on fracture toughness in terms of fracture mechanics.

### 5.3.2.3 Radiation-Resistant Materials

Research on radiation-resistant materials has shown that increasing amounts of copper and phosphorus have increasing influence on radiation embrittlement in pressure vessel steels.<sup>18</sup> Based on these empirical results, limits have been set for the copper and phosphorus content of reactor pressure vessel steels. A mechanism for the embrittling effect of copper has been proposed<sup>19</sup> but has not, as yet, been confirmed.

### 5.3.2.4 Cyclic Crack Growth Rate

It is recognized that small flaws, material defects, and inhomogeneities will always exist to some extent in materials to be used in nuclear service. Although such irregularities will initially be below the established limits that would require repairs, they can grow as a result of cyclic loading during normal operation. The potential for cyclic crack growth in reactor structural materials should be experimentally assessed to gain confidence that flaws cannot grow to a "critical" size. Therefore, it is important to extend the data base and to improve the understanding of cyclic crack growth in reactor structural materials, especially for the environment and cyclic loading rates that represent realistic reactor operating service.

Some crack growth rate data have been established for irradiated materials under reactor service. Much of the work on unirradiated steels was conducted at relatively rapid cyclic frequencies. More recently, however, slower cyclic rates (of 1 cpm and less) have shown significant increases in crack growth rates from cycling in a water environment.<sup>22</sup> Because slow cycling more nearly approaches the realistic service performance of an operating reactor, future research should emphasize the use of varied loading times and long hold times in the development of data on cyclic crack growth rates.

### 5.3.2.5 Steam-Generator Tube Integrity

Tubing for PWR steam generators is subject to wastage, cracking, and denting at support-plate locations. As already mentioned, denting is particularly insidious because it precludes meaningful eddy-current inspection. The large accumulated strain in the dented region causes primary-side intergranular stress-corrosion cracking, which remains undetected until leakage. Large safety margins are established for steam-generator tubing, so that large in-service degradation (40 to 60%) of the tube wall can be tolerated.

### 5.3.2.6 Stress-Corrosion Cracking

A rapid and accurate test is being developed to determine the susceptibility of type 304 stainless steel to intergranular stress-corrosion cracking (SCC). Susceptibility to intergranular SCC can arise, for example, from slow cooling after overheating during welding in the field. It is therefore mandatory that stainless steel welds be inspected before service to ensure that all areas of excessive sensitization are identified for prompt and easy repair or for enhanced in-service inspection. Detection should be as quantitative as possible and must be nondestructive. The currently used copper-copper sulfate-sulfuric acid test and the oxalic acid etch test give poor, if any, correlation for BWR service and are destructive. Furthermore, these tests are not useful in the field.

### 5.3.2.7 Neutron Dosimetry

Neutron dosimetry--the measurement of the damaging neutrons causing embrittlement of pressure vessel steel--is the only method for relating steel embrittlement from surveillance irradiations to embrittlement in the power reactor vessel wall to assess safe reactor life. At present, mechanical property measurements are more accurate than neutron dosimetry. Laboratories and LWR vendors use ASTM procedures for neutron dosimetry in test and power reactor surveillance, but each uses its own set of experiments to verify that calculations and extrapolations are consistent with measurements. Furthermore, an energy threshold of  $E > 1$  MeV is currently being used as the criterion for neutron flux and fluence, even though research data<sup>23</sup> show that neutrons with energies between 0.1 and 1 MeV can contribute as much as 40% of the embrittlement attributed to neutrons with energies higher than 1 MeV.

### 5.3.2.8 Licensing Criteria

The present licensing criteria for irradiation effects on ferritic pressure vessel materials are contained in Appendix H, "Reactor Vessel Material Surveillance Requirements," of 10 CFR Part 50. In summary, the fracture toughness of an irradiated vessel must be assessed by (a) establishing the neutron-induced increase in the reference nil-ductility transition temperature, as measured by Charpy V-notch specimens, and (b) translating from pre- to post-irradiation

behavior along the 50-ft-lb energy level without permitting the upper shelf energy level to drop below 50 ft-lb and without permitting the specimens to exhibit a lateral expansion of less than 35 mils.

With respect to annealing (post-irradiation heat treatment of irradiated vessel steel), only the U.S. Army's SM-1A reactor vessel has been subjected to annealing.<sup>24</sup> Guidance can be taken from this action, as well as from the draft ASTM standard on pressure vessel annealing now in the final stages of ballot-approval procedures.

Control of residual elements is important for limiting radiation-induced embrittlement. The applicable criteria are contained in Appendix H of 10 CFR Part 50.

At present, rules for fatigue design are given in ASME Code Section III, for stresses over 35<sup>m</sup> and temperatures <700°F. These rules, which are invoked for licensing purposes, apply to unirradiated material prior to service. The research needed will update the present code rules and extend them to irradiated materials. However, new criteria based on crack propagation under cyclic loading may be desirable in the future.

Procedures for ensuring the integrity of steam-generator tubing, based on inspection results, are given in Regulatory Guide 1.121,<sup>25</sup> Regulatory Guide 1.83,<sup>26</sup> ASME Code Section III, Appendix F ("Rules for Evaluation of Faulted Conditions"), and several internal licensing draft procedures.

Upper limits are set on the allowable oxygen concentration for normal operation and for short abnormal transients. Calculated stresses must not exceed ASME Code design levels. However, the ASME Code does not take into account weld residual stresses, which can be very high. Requirements for the control of stainless steel sensitization, either by furnace heat treatment or welding, are given in Regulatory Guide 1.44.<sup>27</sup> Rules for proper fabrication to preclude sensitization to intergranular stress-corrosion cracking in reactor system piping welds are given in Section III of the ASME Boiler and Pressure Vessel Code. These rules have been adopted as the licensing criteria and have been extended in Regulatory Guide 1.44 with respect to required preheat and postweld heat-treatment procedures.

Standard ASTM procedures are used to evaluate neutron dosimetry in surveillance irradiations. The correlation between neutron fluence (integrated exposure) and mechanical property changes is given in Regulatory Guide 1.99.<sup>28</sup>

### 5.3.3 RESEARCH PROGRAM

#### 5.3.3.1 Irradiation Effects

The data base for validating the fracture-toughness behavior of irradiated material must be established from results obtained with the validated specimens developed in other parts of the program and must include multiple heats of representative pressure vessel plate, forgings, welds, and weld heat-affected-zone material. Valid data at high toughness levels are required for subsequent application to operating reactor pressure vessels. Specifically, about 10 different weld metal-flux-fabrication combinations of pressure vessel welds are being irradiated or prepared for irradiation in compact specimens through 4 inches thick. The primary goal is to investigate the toughness level, in fracture-mechanics terms, of low ductile shelf material after irradiation and on the ductile shelf. Transition-temperature studies will be performed on a second-priority basis. The J-R curve method is being developed for this testing.

#### 5.3.3.2 Annealing

Three different weld metal-flux-fabrication combinations of reactor vessel welds are being studied in a program consisting of irradiation, annealing, re-irradiation, reannealing, and subsequent re-irradiation. The time and temperature of post-irradiation annealing are important variables in this program, and the combination affects the recovery of both the ductile shelf fracture toughness and the transition temperature. The results will be made directly applicable to reactor vessel annealing.

#### 5.3.3.3 Radiation-Resistant Materials

The primary compositional and metallurgical variables governing irradiation embrittlement and recovery are being systematically studied and isolated. Materials include different heats of A533B and A5058 Class 2 pressure vessel steel and welds, specially prepared laboratory heats of steel, and small heats of newer higher strength steels.

#### 5.3.3.4 Cyclic Crack Growth Rate

A systematic study is to be conducted of the effect of coolant environment, neutron irradiation, and temperature on fatigue crack initiation and cyclic crack growth rates. The effects of interest include cyclic rate (especially very slow,  $\leq 0.1$  cpm), tension hold time, mean applied stress  $K$  (at both large and small  $\Delta K$  amplitudes), the scatterband resulting from heat-to-heat variations, and studies at the low range of  $\Delta K$  to establish the threshold stress intensity factor more accurately. The full range of pressure vessel and piping materials and product forms is to be studied. Pressure vessel materials are being irradiated and tested in environments simulating those of PWRs and BWRs. Piping materials must be tested under PWR/BWR coolant environments. Results obtained with laboratory test specimens are to be correlated with the behavior of similar material under field conditions. A comparison is needed of in-reactor cyclic response against results obtained out of reactor.

#### 5.3.3.5 Steam-Generator Tube Integrity

Tubing typical of several major designs of PWR steam generators is being tested in both the burst and the collapse modes. Artificial cracks with depths ranging from about 25% of the wall thickness to through-wall cracks and areas of wastage ranging in depth from 25 to 90% of wall thickness will be introduced in the tubes to determine their effect on both burst and collapse pressure. Leak rates from various size flaws will also be determined. Tubing with simulated natural flaws and flawed tubing removed from operating steam generators will be included if possible. Margins of safety against burst and collapse will be established from the test results, and empirical predictive models for steam-generator tube performance will be developed.

#### 5.3.3.6 Stress-Corrosion Cracking

Research is under way in three primary areas of interest for intergranular stress-corrosion cracking. A field test is being developed for detecting and measuring the degree of sensitization and susceptibility to SCC of welded stainless steel piping. Laboratory feasibility studies are being extended to field confirmation. Adoption of the test as a standard acceptance and inspection procedure is anticipated.

The residual stress level resulting from welding in piping is being studied to develop an analytical method for predicting such residual stress levels on the basis of fabrication parameters. Evaluation of the analytical procedure is proceeding on a variety of sizes and types of nuclear grade weldments, and studies are in progress to optimize the fabrication parameters for minimizing residual stresses.

The SCC characteristics of steam-generator tube material (Inconel 600) are being investigated as a function of stress, strain, temperature, metallurgical condition, water chemistry, and other important parameters. The aim is to establish a predictive capability for stress-corrosion cracking in steam-generator tubing during the service life of a steam generator.

#### 5.3.3.7 Neutron Dosimetry

The neutron flux and the spectrum of neutrons by energy level are being both calculated and measured in a wide variety of test and power reactors. The objective is to confirm procedures for calculating and extrapolating the neutron flux in reactor surveillance irradiations. For the experimental irradiations, as many as 20 different flux-monitor materials will be included to cover the entire neutron energy spectrum. In selected instances, spectrometry will be used to establish the spectrum in the important energy range between 0.1 and 1.0 MeV. Confirmation procedures will use a simulated pressure vessel wall wherein neutron-flux monitors and mechanical property specimens can be irradiated for comparison with the pretest calculations. Primary mechanical property characterizations will be in fracture-mechanics terms.

#### 5.3.3.8 Milestones

##### Irradiation Effects

- 1977 Data developed on initial irradiated low ductile shelf weld metal toughness. Initial dynamic toughness and size-effect correlation established.
- 1978 Irradiated low ductile shelf toughness initial evaluations in fracture-mechanics terms. Single-specimen J-R curve test method developed for static and dynamic loadings.

- 1979  $K_{TD}$  curve updated to incorporate dynamic and ductile shelf toughness. Work on the fracture toughness of low ductile shelf weld metals continued and extended into the transition region. Elastic-plastic analysis validated for irradiated steels.
- 1980 Initial embrittlement data developed on newer, higher strength steels.
- 1982 Toughness test procedures established for irradiated new steels.

#### Annealing

- 1977 Specimens irradiated for the irradiation, annealing, re-irradiation sequence.
- 1978 Test and analysis of "annealing" completed.
- 1979 Procedures for reactor vessel annealing validated and recommended. Application of annealing to reactor pressure vessel safety.

#### Radiation-Resistant Materials

- 1977 Completion of cooperative program for establishing residual element limits in production vessel plate and welds.
- 1978 Isolation of primary compositional and metallurgical factors influencing radiation embrittlement in current reactor pressure vessel steels.
- 1979 Updating of code rules for compositional and metallurgical controls to minimize radiation embrittlement in current reactor pressure vessels steels.
- 1980 Study of compositional and metallurgical factors influencing radiation embrittlement in newer reactor pressure vessel steels.

#### Cyclic Crack Growth Rate

- 1977 BWR crack-initiation and growth rate data developed for the ASME Code. Primary variables established for PWR crack growth rate studies.
- 1978 Data on initial crack growth rate developed for irradiated steels. Crack growth rate trends at high  $\Delta K$  obtained with 4-inch-thick specimens.
- 1979 Code rules for irradiated and unirradiated crack growth rate updated.
- 1980 Initial crack growth rate data obtained on new steels.
- 1981 Irradiated crack growth rates obtained on new steels.

#### Steam-Generator Tubing Integrity

- 1977 Data collection and preliminary evaluation of steam-generator tube bursting initiated.
- 1978 Test results correlated for machine and chemically produced simulated flaws. Model for establishing steam-generator tube integrity from inspection results completed.
- 1979 Specimens of service-degraded steam-generator tubing obtained. Margin to failure of laboratory-degraded steam-generator tubes determined.
- 1980 Steam-generator tube integrity/inspection model validated.
- 1981/1982 Model developed for predicting the remaining life of steam-generator tubing and for establishing the optimum inspection period.
- 1983 Methods and materials for mitigating steam-generator tube degradation evaluated.

#### Intergranular Stress-Corrosion Cracking (IGSCC)

- 1978 Residual stress calculations for welds completed. Residual stresses in HSST-ITV weld repairs calculated.

Procedures developed for evaluating IGSCC susceptibility of BWR piping internal surfaces. Electrochemical IGSCC techniques validated in the field. Residual stress assessment criteria validated.

- 1979 Field method for detecting IGSCC susceptibility of BWR piping validated, and time-temperature-stress relationships for SCC of steam-generator tube material established.
- 1980 Capability developed for predicting steam-generator-tube stress-corrosion cracking from environmental and materials considerations.
- 1981- Improved methods, materials, and chemistry controls qualified for mitigating or eliminating steam generator tube degradation and stress-corrosion cracking.
- 1983

#### Neutron Dosimetry

- 1977 Flux monitors selected and fabricated for benchmark reactor facilities. LWR surveillance exposure application procedure drafted.
- 1978 Initial analysis of benchmark facility flux-spectrum irradiations. Simulated vessel wall damage facility designed.
- 1979 Simulated vessel wall damage facility constructed. Vessel surveillance analysis procedure updated.
- 1980 Flux and damage from simulated vessel wall experiment tested and analyzed.
- 1981 Standards and licensing procedures for LWR dosimetry established.

### 5.4 FLAW DETECTION AND EVALUATION

#### 5.4.1 OBJECTIVES

The purpose of this task is to develop current and advanced testing procedures for preservice inspection and in-service inspection and detection/monitoring of flaws and defects and of flaw growth in LWR vessels, piping, and steam-generator tubing.

#### 5.4.2 PRESENT STATUS

Inspection of nuclear reactor components by ultrasonic techniques is required both prior to service and during shutdown for periodic in-service inspections. Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for In-Service Inspection of Nuclear Power Plant Components," defines inspection criteria and allowable flaw sizes, based on linear elastic fracture mechanics, for various locations within reactor components. In the present inspection procedure, the pulse-echo amplitude and search-unit position are evaluated as a basis for flaw detection and sizing. Although ultrasonic testing is the presently accepted and most useful volumetric inspection technique, its reliability for flaw detection and sizing (using the Code procedure) is questionable and often inadequate. Significant advances have been made recently in the signal processing of pulse-echo data to form a synthetic aperture focused image of high resolution for greatly enhanced flaw characterization.<sup>29-31</sup>

Continuous on-line surveillance represents a goal because feasibility is yet to be demonstrated. The technique to be employed is acoustic emission, and while important advances in instrumentation have been recently realized, acoustic emission data analysis and extrapolation to real structures still require development and final proof testing in operating nuclear systems.<sup>32</sup> Furthermore, it is noted that acoustic emission data on crack growth may need validation by ultrasonic testing during a shutdown-period inspection.

Continuous acoustic emission inspection during welding, for the detection of microcracking during weld solidification and cooling, has been established in nonnuclear applications<sup>33</sup> and is being carried forward in the nuclear application to demonstrate the proof of principle for the detection of cracks and other types of rejectable flaws.<sup>34-36</sup>

The presently used ASME Code eddy-current inspection techniques are fast, but they can produce unreliable inspection results because of the many independent variables that affect the signals. For example, the detection of flaws in a tube dented region surrounded by corrosion products and the steam-generator support plate is extremely difficult. Existing mathematical models will be

used to develop computer programs for designing optimum probes, instrumentation, and techniques for improved eddy-current inspection of steam-generator tubing.

#### 5.4.3 RESEARCH PROGRAM

##### 5.4.3.1 In-Service Ultrasonic Inspection

It is necessary to improve the ultrasonic inspection capability for carbon and stainless steel plate and piping in response to the stringent flaw-size/flaw evaluation requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Currently, ultrasonic inspection becomes less reliable at greater search depths, especially in stainless steel base material and welds. This is due to the high signal attenuation and grain-boundary scattering resulting from the large grain size. Because of inaccuracies in calibration and the changing properties of ultrasonic testing transducers and associated electronics, reinspection results are not as reproducible as desired for optimum comparison of in-service inspection results from one inspection to another.

The upgrading of ultrasonic inspection is focusing on developing more aspects of the information resulting from a pulse-echo test, including phase, frequency, amplitude, and search-unit position. Sensitivity of the results to the specific operator, a specific calibration test, or a specific transducer is also being reduced or eliminated. Greater detection sensitivity is being developed, and flaws are being characterized with much improved resolution. Means for the storage and ready retrieval of the information for meaningful reevaluation are also being developed, along with automated comparison of inspection results for the detection of flaw growth between inspection periods. The importance of ultrasonic inspection records is expected to increase, with reference being made to past records for comparison with current information. For such comparisons to be most accurate, flaw and sensor locations must be accurately determined and recorded, and the inspection results must be made independent of changing transducers and electronics properties. The difficulty of ultrasonically inspecting austenitic stainless steel base metal, weld deposits, and the interface is being minimized by processing the data to greatly decrease electronic and grain-boundary scattering noise and to obtain focused images independent of signal amplitude above a detection level. These procedures greatly increase the sensitivity and resolution of ultrasonic inspection and flaw evaluation in stainless steel. The laboratory test procedures are being validated on realistic plate and piping samples, in addition to being adapted for typical in-service inspection procedures.

##### 5.4.3.2 Flaw Detection by Acoustic Emission

Techniques are being developed for the continuous on-line monitoring of flaws and defects. Detection of flaws, and the monitoring of flaw or defect growth during service, is one of the most powerful means of preventing unexpected failure of large primary-system components during service. Both acoustic emission and ultrasonic testing have been shown to have the capability to detect flaws and thereby monitor flaw growth, but it is desirable to continually upgrade the capabilities of both techniques for qualitative and quantitative information on flaw size, shape, location and orientation--information on which positive safety judgments can be based.

Acoustic emission is being developed for on-line flaw monitoring. Acoustic emission probes eventually will be placed on vessels and piping to monitor signals emitted during operation; improved nondestructive examination methods would then be used during shutdown periods for further characterization of the flaws detected. At present, relationships are being drawn between acoustic emission signals and mechanical property effects obtained during the testing of fracture and fatigue-type specimens of both plate and weldments. Acoustic emission signals from other sources that may be present during reactor monitoring are also being evaluated. A laboratory program will then be conducted on fully characterized natural flaws to validate both the detection and the quantification abilities of the techniques developed. The techniques will then be extended to the full range of conditions prevalent in reactor operation, and procedures will be established for distinguishing among acoustic emission signals from various sources such as flaws, strained regions, reactor operations, etc. With such a baseline library available, it will be possible to begin actual structure and reactor vessel monitoring so that signal detection and quantification will be much more meaningful.

##### 5.4.3.3 Improved Eddy-Current Inspection for Steam-Generator Tubing

Improved eddy-current techniques will be developed to separate the effects of diameter variations, probe wobble, tube supports, and conductivity variations from defect-size, defect-depth, and wall-thickness variations. Mathematical models and computer codes for eddy-current tests will be used to computer-design optimized probes, instrumentation, and techniques for multifrequency, multiproperty examinations. The program will develop at least two optimized designs: one for the general inspection of steam-generator tubing and another for special conditions such

as denting. Optimized probes and instrumentation will subsequently be built, evaluated in the laboratory, and finally validated by in-service inspections of steam-generator tubing.

#### 5.4.3.4 Advanced Techniques

Numerous new techniques for nondestructive examination are being developed. Such techniques are continually reviewed and, if seen to be especially promising, are funded in carefully controlled assessment studies. A program was recently initiated to study the feasibility of internal friction monitoring techniques for the prediction of incipient intergranular stress-corrosion cracking in welded stainless steel BWR piping.

#### 5.4.3.5 Milestones (by Fiscal Year)

##### In-Service Ultrasonic Inspection

- 1977 Three-dimensional ultrasonic testing data analyzed for imaging of natural flaws.
- 1978 Ultrasonic testing imaging improved for thick sections, and three-dimensional display of ultrasonic testing data enhanced. Deeper penetration and higher sensitivity ultrasonic testing equipment developed for flaw evaluation. Adaptation of synthetic aperture processing to field inspection begun.
- 1979 Three-dimensional ultrasonic testing imaging and display validated
- 1980 Synthetic aperture focusing technique for ultrasonic flaw characterization validated in the field for ASME Code acceptance.
- 1981- Improved ultrasonic technique for fast and reliable flaw detection developed.
- 1983 Flaw-detection probability and fracture-mechanics models for reactor vessel integrity assessment developed and evaluated.

##### Acoustic Emission Flaw Detection

- 1977 Piping and vessel flaw detection monitors demonstrated in weld shops.
- 1978 Acoustic emission/material property/flaw severity model developed. Acoustic emission models for distinguishing different defects from noises developed. Acoustic emission/flaw type/flaw size correlation developed from monitoring during welding.
- 1979 Acoustic emission/material property/flaw size and defect/noise discrimination models validated in laboratory for application to reactor monitoring. Quantified weld flaw monitors validated in the field.
- 1980 Reactor instrumented for acoustic emission monitoring; system and models for vessel integrity evaluation optimized.
- 1981 In-service acoustic emission flaw detection and monitoring of vessels and piping instituted and validated.
- 1982 Optimization and validation begun for in-service acoustic emission flaw detection.
- 1983 Validation of acoustic emission flaw detection and monitoring of reactor vessels and piping completed and recommendations submitted to ASME Code.

##### Improved Eddy-Current Inspection for Steam-Generator Tubing

- 1978 Computer design of improved eddy-current instrumentation for steam-generator-tube in-service inspection developed.
- 1979 Improved eddy-current probes, instrumentation, and test technique evaluated in the laboratory.
- 1980- Improved eddy-current in-service inspection of steam-generator tubing validated in the field and accepted by the ASME Code.
- 1981

### Advanced Techniques

- 1977 Feasibility of alternative nondestructive examination techniques established.
- 1978 Techniques established for improved nondestructive examination systems shown to be feasible in fiscal year 1977.
- 1979 Internal friction monitoring methodology developed for early prediction of incipient intergranular stress-corrosion cracking in piping.  
Flaw detection and evaluation capability of new nondestructive examination methods evaluated in the laboratory.
- 1980 Newer nondestructive examination methods initially confirmed in field applications.
- 1981 Field studies of new nondestructive examination methods for current LWRs completed.  
Internal friction monitoring methodology for early detection of piping susceptibility to intergranular stress-corrosion cracking validated in the field.
- 1982- New nondestructive examination methods for current LWRs validated.
- 1984 Internal friction monitoring of pressure vessels for early warning of possible degradation developed, evaluated, and validated.

## REFERENCES

1. L. E. Steele, Neutron Irradiation Embrittlement of Reactor Pressure Vessel Steels, Technical Report Series No. 163, International Atomic Energy Agency, Vienna, 1975 (STI/DOC/10/163).
2. W. S. Pellini et al., Review of Concepts and Status of Procedures for Fracture-Safe Design of Complex Welded Structures Involving Metals of Low to Ultra-High Strength Levels, NRL-6300, Naval Research Laboratory, Washington, D.C., June 1965.
3. F. J. Loss and W. S. Pellini, Coupling of Fracture Mechanics and Transition Temperature Approaches to Fracture-Safe Design, NRL-6913, Naval Research Laboratory, Washington, D.C., April 14, 1969.
4. Fracture Toughness Testing and Its Applications, American Society for Testing and Materials, Philadelphia, Pa., ASTM STP 381, 1965.
5. W. L. Server, J. W. Sheckherd, and R. A. Wulleart, Fracture Toughness Data for Ferritic Nuclear Pressure Vessel Materials, EPRI NP-119, prepared for the Electric Power Research Institute by Effects Technology, Inc., Santa Barbara, Calif. April 1976. Available from the Electric Power Research Institute, Research Reports Center, P.O. Box 10090, Palo Alto, Calif. 94303.
6. F. J. Loss, Dynamic Tear Test Investigations of the Fracture Toughness of Thick-Section Steel, NRL-7056, Naval Research Laboratory, Washington, D.C., May 1970.
7. W. O. Shabbits, Dynamic Fracture Toughness Properties of Heavy Section A533 Grade B, Class 1 Steel Plate, WCAP-7623, Westinghouse Electric Corporation, Pittsburgh, Pa., December 1970.
8. G. T. Hahn and M. F. Kanninen, eds., Fast Fracture and Crack Arrest, American Society for Testing and Materials, Philadelphia, Pa., ASTM STP 627, 1977.
9. R. D. Cheverton, Pressure Vessel Fracture Studies Pertaining to A PWR LOCA-ECC Thermal Shock: Experiments TSE-1 and TSE-2, ORNL/NUREG/TM-31, September 1976. Available for purchase from the National Technical Information Service, Springfield, Va. 22161.
10. F. M. Burdekin and M. G. Dawes, "Practical Use of Linear Elastic and Yielding Fracture Mechanics with Particular Reference to Pressure Vessels," in Proceedings of Conference on Practical Application of Fracture Mechanics to Pressure Vessel Technology, London, May 3-5, 1971, Institute of Mechanical Engineers, London, 1971, pp. 28-37.
11. F. J. Witt and T. R. Mager, A Procedure for Determining Bounding Values on Fracture Toughness  $K_{Ic}$  at Any Temperature, ORNL/TM-3894, October 1972.
12. P. C. Paris, "Fracture Mechanics in the Elastic-Plastic Regime," in Flaw Growth and Structure, American Society for Testing and Materials, Philadelphia, Pa., ASTM STP 631, 1977, pp. 3-27.
13. F. J. Witt, "The Equivalent Energy Method for Calculating Elastic-Plastic Fracture," presented at the 4th National Symposium on Fracture Mechanics, Pittsburgh, Pa., August 24-26, 1970. Available for inspection and copying for a fee in the NRC Public Document Room.
14. J. G. Merkle, G. D. Whitman, and R. H. Bryan, An Evaluation of The HSST Program Intermediate Pressure Vessel Tests in Terms of Light-Water-Reactor Pressure Vessel Safety, ORNL/TM-5090, November 1975. Available for purchase from the National Technical Information Service, Springfield, VA. 22161.
15. G. R. Irwin, Handbuch der Physik, Vol. 6, Springer-Verlag, Berlin, 1958, pp. 551-590.
16. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.2, "Thermal Shock to Reactor Pressure Vessels," Washington, D.C. Available for inspection and copying for a fee in the NRC Public Document Room.

17. J. A. Davidson, L. J. Ceschini, R. P. Shogan, and G. V. Rao, The Irradiated Dynamic Fracture Toughness of ASTM A533, Grade B, Class 1 Steel Plate and Submerged Arc Weldment, Heavy Steel Technology Program, Technical Report No. 41, WCAP-8775, Westinghouse Electric Corporation, Pittsburgh, Pa., October 1976.
18. J. R. Hawthorne, "Demonstration of Improved Radiation Embrittlement Resistance of A533-B Steel Through Control of Selected Residual Elements," in Proceedings of Symposium on Radiation Effects on Structural Alloys for Nuclear Reactor Applications, Niagara Falls, Canada, June 29-July 1, 1970, American Society for Testing and Materials, Philadelphia, Pa., ASTM STP 481, 1970, pp. 96-127.
19. F. A. Smidt and J. A. Sprague, "Property Changes Resulting from Impurity-Defect Interactions in Iron and Pressure Vessel Alloys," in Effects of Radiation on Substructure and Mechanical Properties of Metals and Alloys, American Society for Testing and Materials, Philadelphia, Pa., ASTM STP 529, 1973, p. 78.
20. G. D. Whitman, Quarterly Progress Report on the Heavy Section Steel Technology Program for October-December 1975, USNRC Report ORNL/NUREG/TM-3, April 1976. Available for purchase from the National Technical Information Service, Springfield, Va. 22161.
21. ASTM Standard A 262-70, "Standard Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels," American Society for Testing and Materials, Philadelphia, Pa., 1970.
22. T. R. Mager, J. D. Landes, D. M. Moon, and V. J. McLaughlin, The Effect of Low Frequencies on the Fatigue Crack Growth Characteristics of A533 Grade B Class 1 Plate in an Environment of High-Temperature Primary Grade Nuclear Reactor Water, Technical Report No. 35, WCAP-8256, Westinghouse Electric Corporation, December 1973.
23. C. Z. Serpan, Jr., "Damage-Function Analysis of Neutron-Induced Embrittlement in A302-B Steel at 550 F (288 C)," in Effects of Radiation on Substructure and Mechanical Properties of Metals and Alloys, American Society for Testing and Materials, Philadelphia, Pa., ASTM STP 529, 1973, pp. 92-106.
24. U. Potapovs, G. M. Knighton, and A. S. Denton, "Critique of In-Place Annealing of SM-1A Nuclear Reactor Vessel," Nuclear Engineering and Design, 8, 39-57 (1968). Available in public technical libraries.
25. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.121, "Bases for Plugging Steam Generator Tubes," August 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
26. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.83, "In-Service Inspection of Pressurized Water Reactor Steam Generator Tubes," June 1975. Available for inspection and copying for a fee in the NRC Public Document Room.
27. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," May 1973. Available for inspection and copying for a fee in the NRC Public Document Room.
28. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials" April 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
29. J. R. Frederick, J. A. Seydel, and R. C. Fairchild, Improved Ultrasonic Nondestructive Testing of Pressure Vessels, Yearly Progress Report, August 1, 1974-July 31, 1975, USNRC Report NUREG-0007-1, January 1976. Available for purchase from the National Technical Information Service, Springfield, Va. 22161.
30. J. R. Frederick, R. C. Fairchild, and B. H. Anderson, Improved Ultrasonic Testing of Pressure Vessels, Annual Progress Report, August 1, 1975-July 31, 1976, USNRC Report NUREG-0007-2, September 1977. Available for purchase from the National Technical Information Service, Springfield, Va. 22161.
31. J. R. Frederick et al., Improved Ultrasonic Nondestructive Testing of Pressure Vessels USNRC Report NUREG/CR-0135, May 1978. Available for purchase from the National Technical Information Service, Springfield, Va. 22161.

32. E. R. Reinhart, comp., In-Service Inspection Program for Nuclear Reactor Vessels, Project 17-2440, Progress Report No. 8, March 1, 1972-March 1, 1973, Final Technical Report to Steering Committee, EEI Project RP 79 (Edison Electric Institute/Tennessee Valley Authority), Southwest Research Institute, May 21, 1973. Available for purchase from the National Technical Information Service, Springfield, Va. 22161.
33. D. W. Prine and R. N. Clark, Inspection of Nuclear Reactor Welding by Acoustic Emission, First Interim Report, May 1975. Available in the NRC Public Document Room in file for Report NUREG-0006.
34. D. W. Prine and R. N. Clark, Inspection of Nuclear Reactor Welding by Acoustic Emission, Final Report, Phase I, USNRC Report NUREG-0035-1, February 1976. Available for purchase from the National Technical Information Service, Springfield, Va. 22161.
35. D. W. Prine, Inspection of Nuclear Reactor Welding by Acoustic Emission, Data Report, November 1975-May 1976, USNRC Report NUREG-0035-2, September 1976. Available for purchase from the National Technical Information Service, Springfield, Va. 22161.
36. D. W. Prine, Inspection of Nuclear Reactor Welding by Acoustic Emission, May 1976-March 1977, USNRC Report NUREG-0035-3, June 1977. Available for purchase from the National Technical Information Service, Springfield, Va. 22161.

CHAPTER 6  
PROGRAM PLAN  
FOR THE RESEARCH SUPPORT BRANCH

## 6.1 INTRODUCTION

The programs of the Research Support Branch are directed at providing research information in support of NRC standards and guides, inspection and enforcement, and nuclear reactor regulation. In general, the Research Support Branch sponsors a category of research termed "reactor operational safety," that is, research aimed at providing direct assistance to NRC officials concerned with the operational and operational-safety aspects of nuclear power plants. In addition, the Research Support Branch coordinates an international thermal-hydraulics program related to loss-of-coolant accidents (LOCAs) in pressurized water reactors.

The NRC requires a defense-in-depth design philosophy to ensure the safety of nuclear power plants. Essentially, this means that three levels of safety are incorporated: (a) the plant is designed and fabricated for maximum safety, (b) protective systems are provided to monitor and correct abnormal conditions, and (c) engineered safety features are installed to mitigate the consequences of accidents.\*

Criteria for the defense-in-depth concept are presented in Chapter I of Title 10 ("Energy") of the U.S. Code of Federal Regulations, in particular, Appendix A ("General Design Criteria for Nuclear Power Plants") and Appendix B ("Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants") of Part 50 of Chapter I. The Research Support Branch programs are related primarily to research in support of NRC implementation of the criteria contained in these two appendices and the related NRC standards, guides, and branch technical positions. The topics currently addressed include qualification-testing evaluation, fire protection, human engineering, noise diagnostics, systems and equipment for inspection and enforcement, and three-dimensional LOCA flow effects. The principal safety topics covered by the Research Support Branch are described briefly below and in more detail in Sections 6.2. through 6.7.

### 6.1.1 QUALIFICATION-TESTING EVALUATION

The qualification-testing evaluation program is focused on obtaining the data needed to answer certain questions about the testing of safety class equipment to assess performance during and after postulated accident conditions. The near-term qualification test program is being conducted to answer questions about assessing conformance with IEEE Std 323-1974,<sup>2</sup> "Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations," and NRC Regulatory Guide 1.89,<sup>3</sup> "Qualification of Class IE Equipment for Nuclear Power Plants," which endorses IEEE Std 323-1974 with certain changes. The end products are data, criteria, and procedures that will enable the applicant and the NRC to ascertain that an acceptable qualification-testing program has been conducted.

The specific questions considered include aging; nuclear source term definition; synergisms; the performance indicators that must be monitored; failure definition; allowable thermal and nuclear-radiation-flux gradients; test sample preparation, quality control, mounting, and connections; chemical and steam flow rates; degree of mixing required; degree of impingement; and vibration.

### 6.1.2 FIRE PROTECTION RESEARCH

The fire protection program emphasizes the collection of confirmatory data needed in support of current design standards and regulatory guides for fire protection and control in LWR nuclear power plants and the collection of confirmatory data that will provide an improved technical base for modifying these standards and guides where appropriate. The objectives of the research are to

- Decrease the vulnerability of the plant to fire.

\*An excellent short discussion of the defense-in-depth concept is contained in U.S. Nuclear Regulatory Commission Annual Report 1976.<sup>1</sup>

- Provide for better control of fires.
- Mitigate the effects of fires on plant safety systems.
- Remove unnecessary design restrictions.

The fire protection research plan is based on the general recommendations of NUREG-0050,<sup>4</sup> with specific details reflecting the requirements of the following NRC offices: Nuclear Reactor Regulation (NRR), Standards Development, and Inspection and Enforcement.

#### 6.1.3 HUMAN ENGINEERING

The human-engineering research program is concerned with assessing the role of human errors in reactor operational safety. It includes specific studies in support of human error investigations and the development of associated training programs by the NRC Office of Inspection and Enforcement, the study of safety-related operator actions in support of the development of guides and standards by the Office of Standards Development (OSD), and a continuing review of the application of ergonometics in the design of nuclear power plants.

#### 6.1.4 NOISE DIAGNOSTICS

The noise-diagnostics research program (conducted in conjunction with NRR and OSD) has supported licensing activities by the use of noise-diagnostics techniques in independent assessments of core-barrel motion in operating pressurized water reactors and in-core instrument-tube vibrations in operating boiling water reactors of the BWR-4 type. More recent noise-diagnostics studies jointly supported by RES, NRR, and OSD have been concerned with assessing the performance of existing loose-parts-monitoring systems in operating reactors. The current RES program includes methods development studies, laboratory research on loose-parts monitoring, and assessment of the use of noise-diagnostics techniques to determine reactor stability. The noise-diagnostics studies support the development of guides and standards such as Regulatory Guide 1.133,<sup>5</sup> "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," and proposed regulatory guides on core-barrel motion monitoring and pipe vibration. This research program supports NRR efforts as needed for independent evaluation of the cause and correction of various vibration-induced malfunctions at operating nuclear power plants.

#### 6.1.5 INSPECTION AND ENFORCEMENT SUPPORT

The Research Support Branch program includes activities in support of the NRC Office of Inspection and Enforcement. One of these is assessment of the contribution of human errors to licensee non-routine events at operating nuclear power plants, mentioned above under "Human Engineering." Other support activities under discussion with the Office of Inspection and Enforcement are concerned with methods for improving the NRC assessment of licensee performance.

#### 6.1.6 THREE-DIMENSIONAL LOCA FLOW EFFECTS

An international research program involving the participation of the Federal Republic of Germany (FRG), Japan, and the NRC is being coordinated by the Research Support Branch. Its purpose is to measure three-dimensional flow effects during the reflood phase of a LOCA. Test facilities being built in both Germany and Japan will use advanced experimental measurement instrumentation loaned by the NRC. The NRC-sponsored computer code TRAC, under development at the Los Alamos Scientific Laboratory (LASL), is to be used for analyzing the experimental data and for extrapolation to full-scale operating nuclear power plants. The test facilities in Germany consist of a 340-rod electrical PWR core simulation LOCA system test and a 180° full-scale PWR upper plenum test. The Japanese facilities will have a 2000-rod electrical PWR cylindrical LOCA system test and a full-PWR-radius 2000-rod slab-core test. The slab-core test facility will support the measurement and assessment of three-dimensional flow effects within the core of a pressurized water reactor during the reflood phase of a LOCA. The upper plenum test facility will be used to measure entrainment and deentrainment effects during reflood. The TRAC code will be used to match boundary and initial test conditions among these separate-effects tests and other analyses.

This research program addresses the extent of steam binding during the reflood phase of a postulated PWR LOCA and any effects that the steam binding may have on the entrance of the emergency core coolant into the reactor core. These experiments will permit computer codes that have been confirmed by the essentially one-dimensional LOFT and Semiscale tests (see Chapter 2) to be scaled to a more exact three-dimensional representation.

## 6.2 QUALIFICATION-TESTING EVALUATION

Programs were started in fiscal year 1975 to study the significance of synergisms in LOCA testing. In fiscal year 1976, programs were started to determine the effects of aging and to determine the composition of the nuclear source term as defined by NRC Regulatory Guide 1.89.<sup>3</sup> In the latter half of fiscal year 1976, these studies were included in a broader program to evaluate the qualification testing of safety-related equipment and to resolve specific anomalies and uncertainties associated with the qualification testing outlined in IEEE Std 323-1974.<sup>2</sup>

### 6.2.1 OBJECTIVES

The objectives of qualification-testing evaluation are to obtain data to further improve the technical basis for current standards and regulatory guides for Class IE safety-related equipment\* and to obtain data for any modification of these standards and guides where appropriate. Specific objectives are to assess:

1. Current LOCA and main-steam-line break (MSLB) testing procedures and data to support modifications of these procedures.
2. Other qualification-testing procedures that are intended to eliminate nonrandom multiple failures.

### 6.2.2 PRESENT STATUS

At present, IEEE Std 323-1974<sup>2</sup> is used as the basis for establishing equipment qualification tests. The purpose of the standard and of the qualification program required by NRC is to confirm that safety-related equipment will perform as required under anticipated accident and operational conditions. Although there is no question as to the intent of IEEE Std 323-1974 or the NRC qualification requirements, specific issues have been raised about the actual testing and analysis that comprise the qualification program. The NRC research program is directed toward resolving these issues with regard to the qualification of Class IE components for operation under LOCA, post-LOCA, and MSLB conditions. Specifically, the following issues are being addressed:

- What is an adequate accelerated-aging methodology that can be used in preparing test samples for LOCA qualification testing?
- What are the specifications for a nuclear radiation simulator that will result in a test meeting the requirements of Regulatory Guide 1.89?
- What are the synergistic effects of a simultaneous steam-and-radiation LOCA environment?

#### 6.2.2.1 Aging

Considerations of aging in the qualification test program are important because of the potential for some aging mechanism, not detected through routine periodic testing, to create a weakened condition in a safety-related component. Such a weakened condition could result in common-mode failures in redundant safety-related equipment subjected to overstress conditions resulting from an accident condition such as a LOCA. Qualification testing is difficult because true aged condition should ideally be simulated by exposing the test component to the same low levels of stress over the same relatively long periods of use as would be experienced in operation. Any practical aging qualification test must therefore be based on an accelerated-aging methodology. At present, thermal aging is simulated by using the Arrhenius equation as a basis for testing accelerated aging. The Arrhenius equation, which is used to extrapolate from high thermal stress applied for a short time to lower stress applied for a longer time, is based on the assumptions that the chemical reaction rate is dependent only on temperature and that over the range of consideration it is constant.

For many safety-related materials in use today, there is some doubt as to the validity of using this assumption to extrapolate from times of less than 1 year to 40 years. The NRC research program is centered around this issue, and its goal is to develop and prove techniques that will adequately simulate long periods of aging at low levels of stress.

---

\*Safety classification of the electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal or are otherwise essential in preventing significant releases of radioactive material to the environment.

Current industrial practice is to consider only thermal aging. Other aging mechanisms (e.g., radiation, humidity, oxygen, and electrical stresses) and the synergisms of these aging mechanisms with thermal stress and normal operating mechanical stresses caused by cycling and vibration are not generally considered. Although these considerations may not be important, further evaluation is required to demonstrate that the important aging mechanisms are included in the current qualification tests.

#### 6.2.2.2 LOCA Testing

IEEE Std 323-1974<sup>2</sup> permits the qualification of safety-related equipment for LOCA conditions by means of a sequential test mode. Specifically, the first portion of the sequence calls for thermal aging, followed by LOCA radiation and vibration. After this portion of the test, a reference ambient-environment operation is established and is followed by LOCA steam and chemical spray conditions at the anticipated LOCA transient rate. After 3 hours, reference ambient conditions are again established and are immediately followed by a return to the LOCA steam and chemical spray conditions at the LOCA transient rate. After 3 additional hours at LOCA steam conditions, the steam and chemical spray conditions are reduced to simulate the post-LOCA period.

Questions have been raised as to the validity of conducting the LOCA radiation portion of the test before applying the LOCA steam and chemical spray conditions. The standard was written in its present form because no synergism between steam and radiation was anticipated and testing under the combined conditions of radiation and steam is difficult. To date there is insufficient evidence that the sequential test sequence recommended in IEEE Std 323-1974 adequately simulates the combined radiation-and-steam environment that the safety-related equipment would be exposed to in an actual LOCA. Preliminary evaluation of tests at Sandia Laboratories did not indicate a significant functional synergism for electrical cables but additional work is required to evaluate other safety-related components.

Current practice is to omit vibration testing from the LOCA qualification test. Furthermore, the test environment used was intended to apply to both BWRs and PWRs. This combined environment may be too severe in some aspects and not severe enough with regard to peak temperatures and exposure to superheat conditions.

#### 6.2.2.3 Nuclear Source Term Definition

One of the exceptions taken to IEEE Std 323-1974 by the NRC is the radiation environment specified for use in LOCA qualification testing. The nuclear radiation environment established by an applicant for type testing is to be based on the LOCA radiation releases defined in Regulatory Guide 1.89.<sup>3</sup> The accident conditions are defined in terms of the percentage of halogens and solid fission products contained in the coolant, the percentage of noble gases and halogens released to the containment atmosphere, and the percentage of halogens plated out on surfaces inside the containment. In qualification tests for a specific safety-related component, the nuclear source must be defined in terms of the spectra, doses, and dose rates that would result from the radiation releases, defined in Regulatory Guide 1.89. This problem requires consideration of several factors, such as containment size, internal containment design and shielding, location of the safety-related component within the containment, and location of the primary-system break. The adequacy of currently used nuclear sources in simulating the accident radiation environment requires additional evaluation and this work is underway.

#### 6.2.3 RESEARCH PROGRAM

The current NRC qualification-testing evaluation program is directed toward assessing current LOCA testing procedures (objective 1) and includes three specific research areas previously identified by NRC. The specific goals and scope of this program have been discussed and reviewed, through the Qualification Testing Evaluation Review Group and by all NRC organizations with responsibility for qualification testing. The portion of the program intended to assess other qualification-testing procedures for Class IE systems (objective 2) is currently in the planning stage. An overall qualification-testing evaluation program plan, including work toward both program objectives and covering the long-range plans for this area of research, will be prepared by RSR. The program plan will be updated on a continuing basis with guidance from the review group. Changes will be made in the research program to reflect current NRC needs and to ensure that research areas are given the correct priority.

The following guidelines will be utilized by the Qualification Testing Evaluation Review Group in considering the technical planning of the qualification-testing evaluation program:

- Is the research specifically related to stated NRC needs? Specific licensing questions and support for regulatory guides must be considered in assessing proposed test plans.
- Is there a high probability that the research will produce usable data? Each research project will be examined on a continuing basis to ensure that the value of the data in verifying overall plant safety is commensurate with the degree of uncertainty in completing the work as planned.

The following specific program elements are included in the current RSR qualification-testing evaluation program:

1. An assessment of LOCA and MSLB testing procedures, including potential synergistic effects.
2. Development of accelerated-aging techniques for qualification testing of safety-related equipment.
3. Definition of nuclear source terms based on Regulatory Guide 1.89 accident assumptions and evaluation of the adequacy of radiation simulators.

#### 6.2.3.1 Program Element 1

This program element, initiated in fiscal year 1975, covers the assessment of LOCA and MSLB testing procedures, including synergisms among the stresses applied. Progress to date has been the development of test equipment and methods to compare sequential and simultaneous tests on identical test samples of safety-related material. The current effort consists of four tasks.

##### Task 1

LOCA qualification tests have been conducted (a) sequentially, as recommended in IEEE Std 323-1974, with radiation exposure preceding the steam and chemical spray environment, and (b) simultaneously, with radiation and steam and chemical environments imposed together. The same experimental test chamber and identical test samples are used in both cases. The specimens from both tests are subjected to a qualitative comparison of performance, using online measurements and post-test evaluation. Tests have been conducted on electrical cables, paints, connector assemblies, cable splices, and lubricants. The radiation source utilized was cobalt-60, and the usable test chamber size was approximately 4 by 8 by 20 inches. The test profile was a composite of the PWR/BWR profile recommended in the appendix to IEEE Std 323-1974. The current series of tests was completed in 1977,<sup>6-8</sup> and a report on synergistic effects in LOCA qualification testing has been prepared.<sup>9</sup>

##### Task 2

A contract was signed with a commercial testing laboratory to compile and evaluate its LOCA qualification test data for electrical cables on a generic basis. It was intended that these data be used to augment the synergistic test data being obtained by Sandia Laboratories and also as a data base for later qualification-testing studies. The generic-basis data, for which the laboratory has obtained a release from its clients, were used along with the Sandia test data in an evaluation of the synergistic effects associated with LOCA testing. This contractor is also being used in an advisory capacity for the design of a new test facility at Sandia Laboratories and in formulating a long-range test program at Sandia.

##### Task 3

The test facility being used by Sandia Laboratories for synergistic effects testing is considered adequate only for relative comparisons on a qualitative basis such as relative differences between sequential and simultaneous testing. The facility, because of its limited size and because the fixed geometrical relationship between radiation source and test chamber results in radiation gradients across test samples, cannot be used for quantitative evaluation of LOCA testing. Quantitative studies will require an improved facility, for which Sandia has prepared a design proposal.<sup>10</sup> The following are being considered in conjunction with the improved facility:

- Existing facilities at other locations.
- Parametric evaluation of radiation source size and test chamber size.
- Test chamber capabilities, such as the ability to perform dynamic testing on safety-related components and required test profiles.

- Source and/or test chamber positioning, instrumentation, and data handling.
- Large test chamber for testing under steam conditions alone.

#### Task 4

A test plan will be prepared by Sandia Laboratories, defining the components and materials, the types of test to be conducted, and the performance parameters to be monitored and evaluated. The basis for this test plan is an evaluation of the LOCA-sensitivity of safety-related equipment. These data are being obtained by subcontract with a reactor plant design organization using a pressurized water reactor currently under design. The specific data that will be obtained is a list of safety-related equipment along with the functional requirements for normal service and for accident conditions, plant location, and normal-service and accident environmental conditions. In addition to the items that are currently being tested (electrical cable, cable connectors, and splices) it is anticipated that the test plan will include such items as electric motor and solenoid valve operators, limit switches, electronic components, junction boxes, pressure transmitters, neutron detectors, electrical signal cable for instruments, resistance temperature detectors, and materials and components utilized in hydrogen recombiners, decay heat removal equipment, and containment penetrations. Specific tests will probably include aging, thermal radiation, LOCA, and main-steam-line break.

In order to ensure that all qualification tests are conducted on materials and components that meet current NRC requirements for quality assurance, a comprehensive review of all potential test specimens will be conducted before conducting any methodology tests. Test specimens will be chosen from generic categories identified by a survey of manufacturers supplying the material or component to be tested. A pretest review of the specific test item will be made to identify factors that could impact the methodology assessment. When test specimens have been judged to have been designed, fabricated, and procured in accordance with NRC quality assurance requirements and this judgment has been confirmed by a pretest inspection and/or test, methodology-oriented qualification tests will be conducted for the purpose of confirming existing test procedures and to provide data that can be used to modify these procedures where required.

#### 6.2.3.2 Program Element 2

This program element covers the development of techniques that can be utilized to simulate the natural aging process for safety-related materials on an accelerated basis. The program was initiated in fiscal year 1976. Progress to date has consisted of a study (completed in fiscal year 1976) to determine the state-of-the-art in material aging. A research program was initiated covering problem areas identified in the study. The current effort consists of the six tasks described below.

#### Task 1

Single-environment aging tests are being conducted to obtain data on the separate effects of radiation, temperature, and humidity.<sup>11</sup> The tests at present are limited to polymeric electrical cable materials utilized with safety-related systems but may be expanded to include other safety-related materials at a later date. The testing is based on the assumption that the important failure mode of electrical cable will be mechanical damage of the insulating or jacket material caused by embrittlement. Data obtained as part of the state-of-the-art report and scoping tests have confirmed this assumption. The parameter used as a measure of this damage is relative elongation of the insulating and jacket material with the conductor removed. From these tests, accelerated damage at the higher stress levels (acceleration functions) in a single environment will be obtained, using a test period of about 1 year. Elongation is being used as a relative damage indicator to evaluate the aging-simulation techniques and for comparison with naturally aged cable samples.

#### Task 2

Combined-environment aging tests will be conducted to obtain data on the synergistic aging effect of temperature and radiation. Synergism with other aging parameters will be evaluated later in the program. As in the case of the single-environment aging tests, relatively low stress levels and long test cycles will be used. Temperatures from 90 to 150°C and radiation dose rates from  $10^3$  to  $10^5$  rad/hr are planned for this testing.

#### Task 3

Tests to determine rate effects are under way. Of particular concern are the rate effects associated with oxygen diffusion and with radiation. These tests will determine the minimum accelerated aging period from which extrapolation to the required material life can be made.

#### Task 4

A study is in progress to identify alternative damage indicators that could be utilized in addition to the material elongation criterion currently used as the reference aging-damage indicator for electrical cable. Although this study is at present limited in scope to the selection of a relative damage indicator to confirm the aging-simulation techniques for electrical cable insulation, the general question of damage criteria will have to be addressed in conjunction with other aspects of the LOCA qualification test. The current qualification-testing standards require that the component under test continue to carry out its prescribed function following the LOCA. It is usually not practical to demonstrate this in any test that can be designed and used. A more basic measure of damage will have to be developed and used for important safety-related equipment.

#### Task 5

Naturally aged samples are being collected so that the aging-simulation techniques being developed in tasks 1 through 4 can be checked with naturally aged material.

#### Task 6

As a backup to the aging-simulation techniques being developed in tasks 1 through 4, an alternative method is being evaluated in which resistance to aging degradation for short periods of time would be assessed and requalification tests used to extend the acceptable lifetime in short time increments through the use of duplicate sacrificial samples.

#### 6.2.3.3 Program Element 3

This program element, initiated in fiscal year 1970, covers the calculation of the nuclear source terms for the accident assumptions made in Regulatory Guide 1.89. Progress to date has consisted of analysis to determine the time relationships following a LOCA for radiation doses, dose rates, energy spectra, and particle types. These data show that current industry practice with regard to radiation simulation testing may differ significantly in terms of dose rate, spectra, and particle type from that required by Regulatory Guide 1.89. The ongoing work in this area is aimed at determining the importance of these differences in terms of potential damage to safety-related equipment.<sup>12-14</sup> The current effort consists of three tasks.

#### Task 1

Additional source-term calculations will be made on the basis of Regulatory Guide 1.89 assumptions, taking into account new codes and test data developed in other programs. Also performed will be calculations based on the proposed revision to Regulatory Guide 1.89, which allows for reduced release assumptions for certain classes of safety-related equipment. In addition, source-term calculations will be made with best estimate LOCA release assumptions as required.

#### Task 2

An evaluation is being made of the adequacy of radiation simulators currently used to duplicate the hypothetical environment following the radioactive release postulated in Regulatory Guide 1.89. An initial assessment will be made by comparing the dose rates and energy spectra resulting from the conservative accident assumptions in Regulatory Guide 1.89 with the dose rates and energy spectra obtainable from practical simulators. Final simulator evaluation will take into account the practical significance of these differences as evaluated in task 3 of program element 3. Cobalt-60, cesium-137, spent-fuel elements, and beta-particle machines will be included in this assessment.

#### Task 3

The damage to safety-related equipment materials will be determined as a function of the gamma and beta dose rates. A determination will also be made of how closely the dose-rate profiles resulting from the Regulatory Guide 1.89 assumptions must be simulated during qualification testing. Materials studies will be conducted using available radiation damage data, and additional experimental data will be obtained as needed. Also of concern is the damage of safety-related equipment materials by beta particles as a function of depth-dose profiles. Materials studies will be conducted and supplemented by experimentation if needed.

#### 6.2.4 MILESTONES

- Evaluation of synergistic effects in the combined radiation and steam environment was completed in July 1977.

- Evaluation of nuclear source simulators based on Regulatory Guide 1.89 will be completed in March 1979.
- Qualification-testing evaluation facility is scheduled for completion in January 1979.
- Development of single-environment aging model will be completed in July 1979.
- Development of combined-environment aging model will be completed in September 1979.

### 6.3 FIRE PROTECTION RESEARCH

A limited-scope cable-tray-separation verification program to obtain data in support of some of the provisions of Regulatory Guide 1.75,<sup>15</sup> "Physical Independence of Electric Systems," was initiated in fiscal year 1975. As a result of the Browns Ferry fire and the recommendations made in NUREG-0050,<sup>4</sup> the Office of Nuclear Regulatory Research, with the support of an NRC review group, established a fire protection research program to augment the ongoing cable-separation studies and establish other areas of research. The new research covers other portions of Regulatory Guide 1.75 as well as portions of Regulatory Guide 1.120,<sup>16</sup> "Fire Protection Guidelines for Nuclear Power Plants."

#### 6.3.1 OBJECTIVES

The overall objective of the fire protection research program is to obtain data in support of current regulatory guides and standards for fire protection and control in LWR power plants and to establish an improved technical basis for modifying these guides and standards where appropriate. Although the research program is being conducted mainly in support of regulatory guides and standards activities, it is expected that the data obtained will aid licensing decisions on design changes and proposed new designs. Specific objectives of the research are to

1. Provide data needed to confirm or modify NRC fire protection requirements.
2. Determine the adequacy with regard to fire protection of generic designs and materials utilized in nuclear power plant safety systems.

#### 6.3.2 PRESENT STATUS

The NRC is interested in determining the performance of safety-related equipment during and after a postulated fire. This subject is particularly important because of the many differences in systems design and postulated fires within each plant and among plants. Furthermore, materials for cable insulation, jackets, and coatings are under constant development, and new products are constantly being introduced. The capability of these materials, when used in Class IE equipment and systems, to prevent the propagation of fire requires additional quantification.

The performance of materials in use today is evaluated primarily by separate-effects testing oriented to single-component evaluation. The relative fire-retardancy of electrical cables is determined by tests specified in IEEE Std 383-1974.<sup>17</sup> However, the extrapolation of test results to typical plant cable-tray configurations and postulated fires would be improved through the addition of more experimental data.

The possible effects of aging on the fire retardancy of electrical cables have not been fully addressed, primarily because there has not been sufficient time to accumulate data on naturally aged materials. Accelerated aging of electrical cable is used in LOCA qualification testing, but not for IEEE Std 383-1974 testing. If aging is shown to degrade the fire retardancy of the materials, then these data will be used in establishing the design standards and guides for new systems and will influence decisions on methods of upgrading the fire retardancy of materials already in use.

Currently applied regulatory guides cover redundant safety equipment, separation criteria, and fire protection criteria. Regulatory Guide 1.75 is based on the assumption that fire hazards are limited to failures or faults internal to the electrical equipment or connecting cables. The guide requires a minimum separation between redundant safety divisions and is based on the assumption that electrical cables can pass an IEEE Std 383-1974 test. Since there is no guarantee that a fire will not propagate to a redundant safety division, particularly if there is an exposure fire, proposed Regulatory Guide 1.120<sup>16</sup> was prepared by NRC to establish criteria to mitigate,

detect, and protect against exposure fires. If issued as proposed, this regulatory guide will require a 3-hour fire barrier between redundant safety divisions in most plant areas, and, at least for new plants, the correlation between flammability (IEEE Std 383-1974) and separation distance (Regulatory Guide 1.75) will be less important. Appendix A to Branch Technical Position APCSB 9.5-11<sup>8</sup> is currently used for new plant reviews and is almost identical with the proposed Regulatory Guide 1.120.

Existing LWR power plants utilize cable conduits, fire barriers, and penetration firestops. Currently, there are no accepted standards or guides that specifically address these cable system components, and test data are required so that definitive guidance can be formulated on their use. These data could be utilized to modify Regulatory Guides 1.120 and 1.75 as appropriate or as a basis for new regulatory guides. Other topics requiring additional data are the effectiveness of smoke and fire detection systems, and of fire-extinguishing systems.

### 6.3.3 RESEARCH PROGRAM<sup>19-28</sup>

The fire protection research program is based on research areas identified in NUREG-0050<sup>4</sup> and by review of current design standards and guidelines. The program is aimed at providing confirmatory data relating to the operation of Class I systems under the conditions they would encounter during the appropriate design-basis fires. The specific objectives and scope were detailed in a program plan prepared by the Research Support Branch. The program plan was reviewed by the NRC offices of Nuclear Reactor Regulation, Standards Development, and Inspection and Enforcement through the Fire Protection Review Group. It is updated on a continuing basis with guidance from the review group, and changes are made in the research program to reflect current NRC needs and to ensure that research areas are given the correct priority.

The following guidelines are used by the Fire Protection Review Group in considering the technical planning of the fire protection research program:

1. Is the research specifically related to stated NRC needs? The NRC fire protection research program is based on NUREG-0050.<sup>4</sup> Specific licensing questions and support for regulatory guides must be considered in assessing proposed test plans.
2. Is there a high probability that the research will produce usable data? Each research project will be examined on a continuing basis to ensure that the value of the data in confirming overall plant safety is commensurate with the degree of uncertainty in completing the work as planned.

The following specific program elements are included in the fire protection research program:

1. Obtain data on the effectiveness of cable-tray separation criteria in ensuring the functional integrity of redundant safety systems.
2. Obtain data on the effectiveness of conduits, fire barriers, and penetration firestops.
3. Obtain data on the effectiveness of coating materials.
4. Obtain data on the fire retardancy of aged materials.
5. Obtain data on IEEE Std 383-1974 and development of improved small-scale cable-system qualification tests.
6. Obtain data on the effectiveness of safety-related equipment (other than cable) when subjected to exposure fires.
7. Obtain data on fire detection system performance.
8. Obtain data on the effectiveness of water and other fire-extinguishing agents.

The first program element includes an ongoing effort at Sandia Laboratories to confirm cable-tray-separation design practice. The scope of this effort has been expanded, and program elements 2 through 6 have been added to cover the effectiveness of additional cable-tray configurations and components and an evaluation of separation criteria utilized for other safety-related equipment. Program elements 7 and 8 cover specific research areas required for licensing decisions and will be implemented in fiscal year 1978. These elements will also be used in support of Regulatory Guide 1.120.

#### 6.3.3.1 Program Element 1

The objective of this program element is to provide data in support of certain portions of Regulatory Guide 1.75. The initial effort is aimed at producing data on the effectiveness of cable-tray separation in preventing the spread of a fire to redundant trays. Horizontal cable trays simulating an open-space configuration have been tested to date.

The program was initiated in fiscal year 1975. Progress to date has consisted of screening tests for cables currently used in nuclear power plants to identify the cable systems most susceptible to fire and capable of passing the IEEE Std 383-1974 flame-retardancy test. The first series of full-scale tests was aimed at determining the adequacy of separation distances of 5 feet for vertical cable trays and 3 feet for horizontal cable trays. The test fires were started by overloading selected wires in cable trays with 60 and 40% of the volume filled with a single size and type of conductor.

Tests were conducted at varying separation distances to determine the minimum separation necessary for cables most susceptible to fire. Vertical separation distances from 5 feet down to 10.5 inches and horizontal separation distances from 3 feet down to 8 inches were tested. For electrically initiated fires in a horizontal open-space configuration, it was determined that a fire will not propagate from the ignited tray to adjacent trays. These tests were conducted with 12-gage single-conductor and 12-gage triplex wire, utilizing both uniform and random-pattern cable packing.

Tests were also conducted with an experimental exposure fire. The objective was to determine whether cable-tray separation alone is sufficient to prevent fire propagation between trays and between redundant safety divisions.

The type and size of the worst-case exposure fire that must be considered for licensing are based on a fire-hazard analysis and will vary from plant to plant; they will also differ among different locations within the plant. Accordingly, no attempt was made to define a design-basis fire for the exposure-fire tests. Single-tray tests were conducted to find a reasonable set of conditions that would result in a fully developed cable-tray fire. The experimental exposure fire was then used in all full-scale cable-tray exposure-fire tests. Propane burners were used to start an exposure fire in one tray, with a barrier placed between it and the tray above. When a fully developed fire was obtained in the first tray, the burners were turned off and the barrier was removed. This method allows experimental study of fire propagation from tray to tray under specific conditions.

A series of tests was conducted on arrays of cable trays, with both electrical and exposure-fire initiation. An array of 14 closely spaced cable trays was used to simulate a single safety division. Simulated redundant safety divisions were separated by the required 5-foot vertical and 3-foot horizontal distance. The results of these tests were summarized in a Research Information Letter.<sup>29</sup> The principal conclusion was that a fully developed fire in the bottom cable tray of a stacked array may propagate to a redundant safety division without fire suppression systems (as expected). On the other hand, electrically initiated fires do not propagate.

In order to determine the characteristics of a cable-tray fire in cable tunnels or in areas where structural walls are close enough to the tray to influence the fire, some of the tests will be repeated to include the effect of walls and ceilings. The first tests will be conducted with a 3-foot vertical and 1-foot horizontal separation. Other separation distances, representative fire loadings, and closed (dead-ended) tests will be included in the testing program.

In typical plant installations, cable trays are oriented vertically at some locations and in others are oriented both vertically and horizontally. Vertical cable trays will be tested in both the open-space configuration and with walls and ceilings close enough to affect the fire.

#### 6.3.3.2 Program Element 2

The objective of program element 2 is to provide information on the effectiveness of the cable conduits, fire barriers, and penetration firestops currently used in nuclear power plants. The initial effort will be directed at producing data that can be used to validate current design practices and their effectiveness in preventing fire propagation and cable damage.

Fire barriers will be tested initially in a horizontal open-space configuration. Testing with vertical cable trays will be conducted at a later date.

Current plans call for exposure-fire testing with (a) single open-ladder cable trays covered with thermal board and fiberboard at the top, (b) open-ladder cable trays with a solid metal cover, (c) open-ladder cable trays with a metal vented cover, (d) cable trays with a solid metal bottom, (e) cable trays with a solid metal cover and bottom, and (f) open-ladder trays with a fiber-blanket cover. Most of these tests have been completed. The results show no fire propagation for the barriers (shields) tested. The exposure-fire test conditions described under program element 1 will be utilized, and burn time will be increased to determine the time required for a fully developed fire in cables capable of passing the IEEE Std 383-1974 fire-retardancy test as well as in cables that cannot pass the test. Some multiple-tray tests will be conducted to study tray-to-tray fire propagation in at least worst-case barrier designs.

The need to test cable conduits and fire barriers in an enclosed configuration to determine the effects of structural walls and ceilings will be determined after completion of the enclosed-area tests on cable separation. If the wall effects are significant with cable trays alone, further testing will be conducted with conduits and barriers.

Penetration firestops will be evaluated with respect to technical issues that are not resolved in industry standards under development and by industry tests currently in progress. The plan is to conduct a limited number of small-scale tests and at least one large-scale test aimed specifically at the areas of concern. These technical issues will probably include the simulation of realistic pressure differences across the penetration, simulation of realistic cable-tray loads, and measurement of cold-side temperatures.

#### 6.3.3.3 Program Element 3

The objective of this program element is to provide information on the effectiveness of fire-retardant coating materials when used in typical cable-tray installations.

A survey of coating materials available for use in cable trays was initiated in August 1976. Generic types were chosen for testing and evaluation in small- and large-scale cable systems tests. Small-scale tests on basic coating properties have been conducted. Full-scale tests have been completed with single and two-tray horizontal configurations. Preliminary results show that all coatings offer a measure of additional protection. However the relative effectiveness of the different fire retardant coatings tested varied widely. These full-scale tests correlated well with the small-scale tests.

Coating application methods may be evaluated to identify problems and performance variations to be anticipated in use with typical cable systems. The factors considered may include storage, age, environmental conditions, and shrinkage. The evaluation of application methods may include the use of the coating material as a fire break.

#### 6.3.3.4 Program Element 4

The objective of program element 4 is to provide information on the fire retardancy of aged materials. As presently planned, the program is aimed at cable and coating materials used in the first three program elements but may be expanded at a later date to include materials from other safety-related equipment.

Both accelerated and natural aging will be considered in the program. The results of an ongoing material-aging study at Sandia Laboratories will be applied. Some naturally aged cable and coating samples will be tested and the fire retardancy compared with that of cable aged by accelerated methods. The physical properties of cable material will be evaluated only to the extent that they affect the fire retardancy of the material. Cracking and spalling of protective coating material applied to a cable tray and changes in the bond between coating and cable and between coating and tray will be examined after simulated long-term aging.

The present plan calls for a survey of generic cables to identify the fire retardants in current use. A screening test will be developed for determining the figure of merit for stability as a function of age and ambient temperatures. Small-scale tests will be conducted on different generic cable insulations and fire-retardant additives to determine the worst-case combinations. Aging tests will be performed to determine aging-acceleration functions using the methods being developed in the qualification-testing evaluation program at Sandia Laboratories. Concurrently, cables will be aged by currently used methods, and full-tray electrical initiation and/or exposure-fire tests will be conducted.

#### 6.3.3.5 Program Element 5

This program element is designed to establish how well defined and repeatable is the cable flame-retardancy test of IEEE Std 383-1974 and to what extent total cable-system performance can be predicted from this small-scale test.

Certain test features not specifically addressed or defined in the current standard will be studied to determine their significance. Included in this category will be:

- Test cell size and configuration
- Air flow requirements
- Cable-tray design and orientation to the flame source
- Flame source energy rate
- Extent to which tray is filled with cable (percentage of volume)
- Cable size and material
- Mixing cable sizes and materials
- Cable-tray orientation (vertical and/or horizontal)

An augmented version of IEEE Std 383-1974 will be prepared and tests conducted on cables used in Program Element 1 to determine the degree to which the small-scale IEEE Std 383-1974 test can predict large-system performance.

A study will be conducted of test parameters and procedures that can be added to IEEE Std 383-1974 to extend its scope to cable conduits, fire barriers, and penetration firestops.

#### 6.3.3.6 Program Element 6

The objective of this program element is to evaluate the effects of exposure fires on safety-related equipment. Only safety-related equipment that is not totally isolated from redundant equipment will be considered.

The scope of this program element will be established by other NRC studies on the probability of redundant safety-related equipment being affected by the same exposure fire and the risk assessment for this event. The risk assessment will be made by first identifying those areas in the plant in which it is difficult to totally isolate redundant safety-related equipment (e.g., the cable spreading room and the control room) and then examining the effects of a plant-specific design-basis exposure fire on all safety-related equipment in that area.

#### 6.3.3.7 Program Element 7

This program element is concerned with testing the performance of fire-detection systems. It will provide data that can be used as the basis for a guide or standard for the design, installation, and utilization of systems for detecting fire and products of combustion. A survey and performance evaluation of currently used fire and smoke detection systems will be conducted. The evaluation will take into account constraints and requirements imposed by typical plant installations, including air flow and local air flow stratification. Detection-system sensitivities will be established and evaluated for adequacy under conditions prevailing in typical design-basis fires; this should help determine detector location and response-time requirements. Development of in-place test methods will be considered for detection systems.

#### 6.3.3.8 Program Element 8

The objective of this program element is to test the effectiveness of water and other fire-extinguishing agents and their potential damage to safety-related equipment. An improved technical base will be provided for establishing criteria on when and where water and other fire-extinguishing agents can be used. This effort is planned not as a complete study on fire-extinguishing systems, but rather as a study of areas identified as requiring evaluation.

Current plans call for determining the minimum concentration and soaking time required for Halon and carbon dioxide to extinguish a fully developed cable-tray fire. The testing will be conducted in a facility large enough to simulate a full-scale cable fire and typical plant enclosure. Halon 1301, carbon dioxide, and water will be included in the tests. Data on smoke and fire detection and actuation systems will also be obtained.

#### 6.3.4 MILESTONES

- Open-space tests on the effectiveness of cable-tray separation in preventing the tray-to-tray propagation of electrically initiated fires (horizontal) were completed in May 1977.
- Open-space tests on the effectiveness of cable-tray separation in preventing the tray-to-tray propagation of exposure fires (horizontal tray orientation) were completed in July 1977.
- Horizontal open-space tests for coatings were completed in January 1978.
- Horizontal open-space tests for barriers were completed in April 1978.
- Enclosed-area cable testing is scheduled to be completed in December 1978.
- Evaluation of penetration firestops is to be completed in January 1980.
- Cable-aging evaluation is to be completed in September 1980.

#### 6.4 HUMAN ENGINEERING

The research effort on human engineering is devoted to a general assessment of the role of human errors in reactor safety and a study of the means for potentially reducing human errors in the operation of nuclear power plants.

##### 6.4.1 OBJECTIVES

The specific objectives of current research efforts are:

- Assessment of accident investigation techniques developed by the U.S. Department of Energy (DOE) (e.g., critical-incident method) and their use by NRC in investigating abnormal events.
- Development of programs for training NRC inspectors and licensee reviewers in the principles and problems of ergonometics, investigations of accidents involving significant human-error contributors, and the review of licensee designs and operating practices for optimum use of ergonometics.
- Evaluation of human errors based on NRC licensee event reports<sup>30</sup> and DOE unusual occurrence reports.
- Study of safety-related operator actions to provide data for licensing criteria and the development of guides and standards governing manual versus automatic actuation of safety systems.

##### 6.4.2 PRESENT STATUS

Human errors have been a significant cause of abnormal occurrences in nuclear power plants.<sup>1</sup> The contribution of human errors to the unavailability of safety systems and components was noted in the Reactor Safety Study.<sup>31</sup> The latter stated that "an actuarial data base for human error rates in nuclear power plants does not exist" and "in general, the design of controls and displays and their arrangements on operator panels in the nuclear plants studied in this analysis deviate from human engineering standards specified for the design of man-machine systems and accepted as standard practice for military systems." The report<sup>32</sup> to the American Physical Society by the Study Group on Light-Water-Reactor Safety recommended that "human engineering of reactor controls, which might significantly reduce the chance of operator errors, should be improved. We also encourage the automation of more control functions and increased operator training with simulators, especially in the accident-simulation mode." Human errors and the design of control rooms were also identified as a major area of concern by former General Electric employees, in testimony before the Joint Committee on Atomic Energy.<sup>33</sup>

In an effort to determine whether improvements could be made, NRC contracted in 1976 with the Aerospace Corporation for a study of control-room displays and operator performance; the final report<sup>34</sup> was issued in March 1977. This study identified a number of instances in which the design of control rooms in operating nuclear plants was not based on optimum human-engineering principles. It did conclude that control-room designs and associated operating procedures, utilizing current training and licensing practices for nuclear operators, are sufficient to.

ensure safe operation. Improvements in control-room design, operator training, and operating practices to increase the margins of safety were recommended. Other recommendations for improved human engineering in control centers were presented.

The NRC participates in the development of industrial guides and standards. An industrial standard (ANS 51.4, ANSI N660) has been drafted and issued for trial use in evaluating safety-related operator actions.<sup>35</sup> As noted in this standard, "there are now no generally accepted criteria for safety related operator actions." The standard proposes criteria for determining the time to be allowed for manually initiating the operation of safety systems. However, a firm technical base is lacking for the criteria proposed, and the need for research in this area is high-lighted. Other industrial standards are being written for the design of display and control facilities<sup>36</sup> and control rooms.<sup>37</sup>

Advanced control-room designs utilizing graphic displays (CRTs) in conjunction with computers are being proposed for the next generation of nuclear plants. Typical examples are the General Electric Company's NUCLINET<sup>38</sup> and the advanced control-room design developed by Combustion Engineering, Inc. These control rooms provide for more informative displays for system diagnosis and hence should contribute to safer and more efficient plant operation. Safety criteria and guides for assessing advanced control-room designs will have to be prepared by NRC after the completion of a detailed technical review of the requirements for human engineering.

The NRC is a member of the Halden Program\* Group, which is conducting a number of studies on automatic process supervision and control in nuclear power plants. The use of advanced control-room displays, computer operation, and diagnostics is being studied and demonstrated experimentally at the Halden reactor facility.

Training programs in MORT (Management Oversight and Risk Tree), RSOS (Reported Significant Observation Studies), risk management, and accident investigation techniques have been given to DOE contractors and employees for several years. The NRC currently does not have an ongoing program for training inspectors and licensee reviewers in the principles of ergonometics. Many of the techniques and principles presented in manuals and training programs for the above-mentioned topics would, with minor changes, be applicable to nuclear power plant safety studies and investigations.

#### 6.4.3 RESEARCH PROGRAM

The NRC analysis of present-generation control-room displays was concluded with the Aerospace Study in 1977. The Electric Power Research Institute (EPRI), however, has active programs in the human engineering of control rooms; NRC access to the results of this research is being provided through the NRC Human Engineering Review Group. The Reactor Safety Research Division also retains Dr. Alan Swain of Sandia Laboratories (an authority on human engineering) to provide an ongoing review of human-engineering research in the United States and abroad.

Computers and CRT displays are expected to be extensively used in the next generation of nuclear power plants. The application of computers to safety systems is currently under review by the Office of Nuclear Reactor Regulation; an example is the core protection calculator proposed by Combustion Engineering, Inc., for Arkansas Unit 2. The Branch research plans include future studies to support the licensing review of advanced control-room designs and computer application to safety-system operation; they also include support activities for the development of criteria, guides, and standards.

The Offices of Standards Development and NRR have been participating in the development of the nuclear standard (ANSI N660) for safety-related operator actions.<sup>35</sup> The Research Support Branch has been supporting this effort by developing quantitative values and criteria for determining the time required for activating safety systems during normal, upset, and abnormal conditions.

The Research Support Branch has initiated a study of the application of DOE system safety programs (MORT, etc.) to NRC inspection and license review practices. A plan for producing training programs and manuals for accident investigations and ergonomic reviews is to be developed. Depending on the Office of Inspection and Enforcement interest, these plans would be implemented in follow-on research activities.

A general review of the role of human errors as reported in NRC licensee events reports and DOE abnormal occurrence reports is being made. This study is expected to categorize the role of human errors and identify potential future research that would contribute to reducing the potential for human errors in the operation of nuclear power plants.

\*An experimental program conducted by the Organization for Economic Cooperation and Development at the Halden BWR facility in Norway.

#### 6.4.4 MILESTONES

- The development of a plan for adopting DOE training programs and manuals for system safety practices and investigations was completed in July 1978.
- The phase A collection and analysis of historical data on safety-related operator actions are scheduled for completion by January 1979.

#### 6.5 NOISE-DIAGNOSTICS RESEARCH

The research effort on noise diagnostics supports NRC on an as-needed basis in independently evaluating hydraulically induced failures and subsequent corrective actions in operating nuclear power plants. In addition, it provides for the development of methods and techniques, criteria for system performance, and guides and standards.

##### 6.5.1 OBJECTIVES

The specific objectives of current research efforts are as follows:

- To develop methods and techniques for using noise diagnostics to detect and quantify flow-induced vibration problems with reactor internals fuel channel plugging.
- To develop criteria for the performance of loose-parts monitoring systems.
- To evaluate the use of noise diagnostics for measuring BWR reactor stability.
- To support the development of standards and guides such as those for monitoring PWR core-barrel motion and loose-parts detection.
- To provide, where requested by the Office of Nuclear Reactor Regulation, noise-diagnostic measurements at operating reactors in order to assess flow-induced failures and evaluate licensee correction.
- To provide for NRC coordinated review of surveillance and diagnostic monitoring methods and systems.

##### 6.5.2 PRESENT STATUS

Noise diagnostics has been used in the past to identify and monitor system malfunctions in a number of operating nuclear power plants; for example, it was used by NRC to analyze and quantify excessive core-barrel motion in the Palisades nuclear power plant.<sup>39</sup> The problem was studied with the aid of a Fourier analyzer, using onsite measurements of signals from in-core and out-of-core neutron-flux chambers, primary coolant temperature variations, and out-of-vessel accelerometers to compute the power spectral density. Technical specifications were later issued by NRC to provide guidance on monitoring for excessive core-barrel motion in operating PWRs.<sup>40</sup> More recently, noise analysis was used by NRC to assist in determining the cause, and monitoring the magnitude, of instrument-tube vibrations in BWR-4 cores.<sup>41</sup>

Noise analysis is a powerful tool for diagnosing and identifying the source of malfunctions that cause the vibration of reactor internals and flow anomalies. The ultimate use of this technique would be in conjunction with an automated surveillance and diagnostic system to provide an alarm in the control room on the detection of an impending component failure or system malfunction.

Monitoring systems for loose parts are being installed in nuclear stations currently starting operation. They use signals from accelerometers and sonic detectors mounted on primary-system piping and the reactor pressure vessel to locate and identify loose metallic parts in the primary system. Loose parts could result in primary-system degradation by blocking fuel coolant channels, impairing valve functioning, or damaging pumps. Early detection and identification of the size and location of loose parts are thus important to safe operation. The performance and use of existing systems were evaluated in 1977.<sup>42</sup>

Reactor kinetics and stability measurements were made last year at several BWRs, using standard transfer-function techniques and analytical calculations of reactor systems kinetics. Several months are required to obtain stability results by this technique. In addition, the result is

dependent on an exact knowledge of many reactor coefficients and the model employed. Noise diagnostics has been proposed by Japanese researchers as a means of measuring reactor stability directly.

### 6.5.3 RESEARCH PROGRAM

A program on noise diagnostics was initiated at the Oak Ridge National Laboratory in fiscal year 1977.<sup>43-51</sup> It provides for developing the technical base for NRC guides and standards, evaluating operating licensed reactors, and reviewing licensee-proposed noise diagnostics, loose-parts monitoring, and automated surveillance systems.

Noise-diagnostic techniques capable of detecting core internals vibration in addition to PWR core-barrel motion and BWR in-core instrument-tube vibration are being developed. Advanced calculational methods (combination of perturbation theory and adjoint methods) are being used to determine whether noise diagnostics can detect and quantify the movement or vibrations of fuel elements, control rods, and other components of the reactor core. If these methods prove successful, noise measurements will be made at operating reactors to confirm their capability. The analytical phase is to be completed in 1979. Field tests to confirm the methodology will be conducted in 1980.

Bandwidth-related errors in PWR barrel-motion detection methods have been resolved. Design changes (i.e., changes in bypass flow) made to BWR-4 fuel elements by the licensee and vendor eliminated instrument-tube vibrations. Neutron and accelerometer signal measurements at all three Browns Ferry units in January-February 1977 confirmed the adequacy of the correction. Regulatory guides are being developed for both the PWR core-barrel motion and BWR instrument-tube vibration measurements.

Criteria for the performance of loose-parts monitoring systems are to be developed in fiscal year 1978. The research program will include a study of sensor-mounting schemes, calibration methods, and the information content of sensor signals. The results will be tested in a laboratory flow loop in fiscal year 1979. Demonstration at an operating nuclear facility is being considered for fiscal year 1980.

The application of an automated plant-anomaly detection and diagnostics system to the surveillance of reactor safety systems has been studied under a DOE contract. Techniques for statistical pattern recognition were examined. Further demonstration of a computer-based automated surveillance system may be studied by DOE in fiscal year 1979 with NRC participation.

A feasibility study will be conducted in fiscal year 1978 to assess whether noise diagnostics can be used to measure BWR stability. A peak in the power spectrum density at about 0.5 Hz may be an indicator of corewide BWR behavior. The LAPUR kinetics code will be used to study the sensitivity of the 0.5-Hz peak. Partial-coherence methods will be used to analyze spectral data taken in 1976 and 1977 at several BWRs in an effort to determine the source of the 0.5 Hz-peak. If these experiments show that noise diagnostics can measure reactor stability, additional measurements will be proposed for fiscal year 1979.

A file of signature measurements for typical BWR and PWR core spectral behavior would be useful in possible future NRC noise-diagnostics investigations at nuclear power stations experiencing abnormal operation or malfunctions. A study in fiscal year 1979 will search for similarities in signals from different examples of a given vendor's PWR design. It would be impractical to acquire measurements at each and every station. A proposal for signature measurements at several typical PWRs will be prepared if universality of behavior can be proved. Measurements would then be taken and analyzed during fiscal years 1979 and 1980.

Support of NRC license review groups in evaluating vibration problems at licensed facilities will continue throughout the program on a high-priority basis. Assistance will also be rendered to Standards Development (OSD) in developing the technical basis for guides and standards.

### 6.5.4 MILESTONES

- A technical report will be prepared in September 1979 for the assessment of advanced calculation methods.
- The bench-top laboratory investigation of loose-parts monitoring criteria have been completed and the loop tests will be completed in November 1979.
- PWR signature measurements will be made at several operating nuclear power stations starting in 1979.

- The feasibility of using noise-diagnostics measurements for determining BWR stability will be assessed by June 1979.

## 6.6 INSPECTION AND ENFORCEMENT SUPPORT

The current research effort in support of the Office of Inspection and Enforcement (I&E) has been described and discussed in Section 6.4, "Human Engineering." Other studies under consideration are described below.

### 6.6.1 OBJECTIVES

The objectives for possible future research in support of Inspection and Enforcement are:

- To develop methods for improving the NRC assessment of licensee performance.
- To assess devices, instrumentation, and telemetry required for measuring process variables and sampling effluent streams at licensed nuclear facilities in order to confirm compliance with NRC regulations.
- To assess portable instrumentation for use by NRC inspectors, including x-ray machines, nondestructive testing devices, health physics equipment, instruments for detecting special nuclear materials, and mobile laboratories.
- To design, develop, and test a prototype instrument package for use by I&E in achieving the above measurement objectives.

### 6.6.2 PRESENT STATUS

The NRC Office of Inspection and Enforcement is responsible for ascertaining that nuclear power plants are built and operated in conformance with NRC-approved technical specifications. At present, NRC inspectors have to rely on data and measurements provided by the licensee.

### 6.6.3 RESEARCH PROGRAM

The study of field measurement systems for I&E use is to begin in fiscal year 1980 if I&E internal management studies in progress define a user requirement. The initial phase of this study would be directed at identifying such requirements as type of measurement to be made, accuracy, and equipment portability. This would be followed by a design effort in which appropriate instrumentation would be selected, design drawings and specifications would be prepared, and packaging details would be developed. The fabrication of a demonstration unit would follow.

## 6.7 THREE-DIMENSIONAL LOCA REFLOOD RESEARCH

Recent analytical investigations indicate that an improved understanding of reflood behavior in a PWR during LOCA could result in a calculated cladding temperature as much as 540°F (300°C) below that calculated with currently prescribed licensing assumptions and evaluation models. The importance of the reflood portion of a LOCA in determining peak cladding temperature is recognized the world over; facilities are being constructed and operated and research programs are active in many countries, notably the Federal Republic of Germany, Japan, and the United States. However, each of the countries conducting research has recognized that the construction of a large-scale integral-test facility is very expensive. Accordingly, interest has developed in defining an international cooperative study to use the specific expertise and facilities available in individual countries. The study would be structured for maximum benefit and minimum time and cost to each participating country. This interest has led to a cooperative program in Germany, Japan, and the United States to study PWR reflood phenomena.

### 6.7.1 OBJECTIVES

The objectives and expected results from this cooperative program are:

1. To obtain experimental information on the reflood phase of a LOCA in full-scale geometry, including:
  - a. Deentrainment, reentrainment, and fallback of water from the upper plenum into the core.
  - b. Axial and radial flow distribution in a two-dimensional full-width, full-height core.

- c. Possibility and nature of flow oscillations between the core and the downcomer.
2. To confirm the physical models in the multidimensional system code TRAC, including the following:
  - a. Hydrodynamic effects in the system by using the data from existing small-scale integral-system tests and the 2000-rod cylindrical core integral test.
  - b. Scaling effects by using the data from existing small separate-effects and integral system tests, the planned full-scale three-dimensional upper plenum test, and the full-scale two-dimensional slab-core reflood test.
3. To quantify, by means of the TRAC code, the margin of conservatism in current assumptions about the effect on heat transfer during PWR reflood of (a) core flow redistribution and (b) the fraction of liquid carried over and vaporized in the steam generator.

#### 6.7.2 PRESENT STATUS

A preliminary study was performed for the NRC to scope a three-dimensional program using an integral test facility (one-fourth LPWR sector) and a program using two separate-effects facilities (core and upper plenum). The results of these indicated a cost of about \$300 million for the integral program and \$150 million (\$60 and \$90 million, respectively) for the separate-effects program, with tests scheduled to begin in the mid-1980s. At about the same time, and as a result of the continuing research coordination between the NRC and foreign countries, it was learned that Japan was planning an upper plenum separate-effects program. Subsequent discussions resulted in informal agreement among the NRC, Japan, and the Federal Republic of Germany to cooperate in a joint investigation of three-dimensional flow distribution phenomena. The specifics of the proposed cooperative arrangement are described below.

The Federal Republic of Germany is planning to build in Erlangen a 180° sector of a full-scale PWR upper plenum, including internals, for the experimental study of deentrainment and water fallback into a PWR core during reflood. Steam-water mixtures will be injected at the bottom of the upper core plate to simulate reflood carryover conditions. The NRC's TRAC code will be used to predict reflood conditions and water deentrainment from the liquid carryover during the tests. The NRC will also loan advanced two-phase-flow instrumentation to improve the measurement of key parameters. This instrumentation is being developed and fabricated at the Oak Ridge National Laboratory, Los Alamos Scientific Laboratory, and the Idaho National Engineering Laboratory under NRC research contracts.

The Japanese Atomic Energy Research Institute is planning, as another part of the three-dimensional cooperative program, to construct a slab-core facility in Tokai to study the redistribution of steam and water flow within the core during reflood. The basic design of this facility has been established through the design efforts of the NRC in consultation with Japan and Germany. The facility is being designed to provide boundary conditions at the upper core plate compatible with those to be used in the German upper plenum facility. A coordinating group from the three countries has agreed on the key features of the design. The NRC's TRAC code will be used to predict the reflood core flow behavior in experiments to be conducted in this facility. The NRC is also providing advanced two-phase-flow instrumentation on loan.

By cooperation among the three countries, the United States will thus obtain experimental information on steam and water behavior in the upper plenum and core of a typical PWR during the reflood phase of a LOCA. The TRAC code will be confirmed and can then be used to predict the reflood behavior of full-size U.S. PWRs. By sharing in the research with Japan and Germany (the U.S. contribution consists largely of developing and fabricating the advanced instrumentation, TRAC code development and application calculations, technical support and coordination, and U.S. upper plenum structures), the United States will be able to realize a significant savings in cost.

An international agreement among Germany, Japan, and the United States for cooperation on three-dimensional reflood research is expected to be signed by each country by December 1978. Design work on the upper plenum test facility in Germany and on the slab-core test facility in Japan has been started. The development of advanced instrumentation to be supplied on loan by the NRC has also been initiated. The initial TRAC code model has been completed. A training program for international groups interested in the use of TRAC was conducted in March 1978.

### 6.7.3 RESEARCH PROGRAM

The TRAC code is being developed by the Los Alamos Scientific Laboratory. The development code models will be gradually expanded to provide for three-dimensional geometry and two-phase-flow representation. The code will be initially confirmed against data from existing one-dimensional small-scale separate-effects tests and later by data from one-dimensional systems tests such as those at the Semiscale facility.

TRAC post-test calculations will be made for experiments in the electrically heated 340-rod core-simulation PKL test facility in Germany. Pretest predictions will be made for the electrically heated 2000-rod cylindrical-core systems test facility now under construction in Japan. The TRAC code will be used in the design of, and pretest predictions for, the upper plenum separate-effects test facility to be built in Germany and the slab-core separate-effects test facility to be built in Japan. After confirmation with experimental data from the above facilities, TRAC will be used to predict the performance of full-scale PWRs during reflood.

Advanced instrumentation, consisting of various configurations of film and impedance probes, is under development at Lehigh University and the Oak Ridge National Laboratory (ORNL). The design and fabrication of probes that the NRC has proposed to loan for installation in the PKL core II and the 2000-rod cylindrical core test have also been assigned to ORNL. In addition, ORNL is to provide air-water and steam-water test facilities for calibrating these probes over a range of two-phase-flow conditions. Additional film and impedance probes are to be loaned for the upper plenum and slab-core separate-effects tests.

Additional instrumentation, proved by past use, is to be provided on loan for use in the above-mentioned German and Japanese experimental test facilities. It consists of spool pieces, liquid-level detectors, and drag disks to be designed and fabricated by the Idaho National Engineering Laboratory, and a Storz lens system to be provided by the Los Alamos Scientific Laboratory.

The NRC is also directing the coordination of support activities and the exchange of data and information among the three parties--Germany, Japan, and the United States. The overall technical direction of research is provided by a coordinating committee consisting of a representative from each country supported by designated program managers in each country. The United States proposes to locate site engineers at the major German and Japanese test facilities and in turn would provide for foreign representation at principal U.S. reflood research facilities.

### 6.7.4 MAJOR MILESTONES

- The construction of the 2000-rod cylindrical-core test facility is expected to be finished in March 1979.
- The construction of the upper plenum separate-effects test facility is expected to be finished in January 1982.
- The construction of the 2000-rod slab-core test facility is expected to be finished in September 1980.
- The initial TRAC code development was completed in January 1978.
- Completion of the first series of tests in the 2000-rod cylindrical core test facility is scheduled for July 1981.
- Completion of the first series of tests in the 2000-rod slab-core separate-effects test facility is scheduled for July 1982.
- Completion of the first series of tests in the upper plenum separate-effects test facility is scheduled for December 1983.
- On the basis of the above schedule, assessment of the TRAC code for predicting PWR reflood behavior should be completed by January 1984.

#### REFERENCES\*

1. U.S. Nuclear Regulatory Commission Annual Report 1976, Washington, D.C., 1977. Available from the U.S. Government Printing Office, Washington, D.C.
2. Institute of Electrical and Electronics Engineers, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations," Std 323-1974, New York, February 1974.
3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants," Washington, D.C., November 1974.
4. U.S. Nuclear Regulatory Commission, Recommendations Related to Browns Ferry Fire, NUREG-0050, Washington, D.C., February 1976. Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.
5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," Washington, D.C., September 1977.
6. L. L. Bonzon, Quick Look Report on Testing to Evaluate Synergistic Effects from LOCA Environments, Test VI, Simultaneous Mode; Cables, Connectors, and Lubricants, Sandia Laboratories, January 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
7. L. L. Bonzon, Quick Look Report on Testing to Evaluate Synergistic Effects from LOCA Environments, Test VII, Sequential Mode; Cables, Connectors, and Lubricants, Sandia Laboratories, February 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
8. L. L. Bonzon, Quick Look Report on Testing to Evaluate Synergistic Effects from LOCA Environments, Test VIII, Sequential Mode; Cables, Connectors, Splices, and Electrical Insulation Samples, Sandia Laboratories, July 1977. Available for inspection and copying for a fee in the Public Document Room.
9. F. V. Thome, Preliminary Data Report, Testing to Evaluate Synergistic Effects from LOCA Environments, USNRC Report SAND 78-0718, April 1978.
10. Qualification Testing Evaluation Program, Light Water Reactor Safety Research, Quarterly Report, October-December 1977, USNRC Report NUREG-CR-0180, April 1978.
11. K. T. Gillen, E. A. Salazar, and C. W. Frank, Proposed Research on Class I Components to Test a General Approach to Accelerated Aging Under Combined Stress Environments, USNRC Report NUREG-0237, April 1977.
12. L. L. Bonzon, Radiation Signature Following the Hypothesized LOCA, USNRC Report NUREG 76-6521, September 1977, revised October 1977.
13. Definition of Loss of Coolant Accident Radiation Source, USNRC Report SAND 78-0090, February 1978.
14. L. L. Bonzon, N. A. Lurie, D. H. Houston, and J. A. Naber, Definition of Loss-of-Coolant Accident Radiation Source: Summary and Conclusions, USNRC Report NUREG/CR-0183, May 1978.
15. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.75, Physical Independence of Electric Systems," Revision 1, Washington, D.C., January 1975.
16. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants," Washington, D.C., June 1976.
17. Institute of Electrical and Electronics Engineers, "IEEE Standard for Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations," Std 383-1974, New York, April 1974.

\* Unless otherwise indicated, the USNRC reports listed here are available for purchase from the National Technical Services, Springfield, Va. 22161.

18. U. S. Nuclear Regulatory Commission, Branch Technical Position APCS 9.5-1, referenced in Standard Review Plan Section 9.5.1, "Fire Protection Systems" (made available on May 3, 1976). Available for inspection and copying for a fee in the NRC Public Document Room.
19. L. J. Klamerus, Quick Look Report on Effect of Cable Tray Separation in the Event of an Electrical Fire, Sandia Laboratories, July 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
20. L. J. Klamerus, Quick Look Report on Effect of Cable Tray Separation in the Event of an Electrical Fire, Sandia Laboratories, August 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
21. L. J. Klamerus, Quick Look Report on Fire Protection Research, Sandia Laboratories, November 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
22. L. J. Klamerus, Quick Look Report on Fire Protection Research, Sandia Laboratories, December 1976. Available for inspection and copying for a fee in the NRC Public Document Room.
23. L. J. Klamerus, Quick Look Report on Fire Protection Research, Sandia Laboratories, February 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
24. L. J. Klamerus, Quick Look Report on Fire Protection Research, Sandia Laboratories, March 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
25. L. J. Klamerus, Quick Look Report on Fire Protection Research, Sandia Laboratories, SAND 77-1424, July 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
26. L. J. Klamerus, A Preliminary Report on Fire Protection Research Program (July 6, 1977 Test), USNRC Report SAND 77-1424, October 1977.
27. L. J. Klamerus, A Preliminary Report on Fire Protection Research Program, Fire Retardant Coatings Tests (December 7, 1977-January 31, 1978), USNRC Report SAND 78-0518, March 1978.
28. L. Przybyla and W. J. Christian, Development and Verification of Fire Tests for Cable Systems and System Components, USNRC Report NUREG/CR-0152, June 1978.
29. U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Research Information Letter 14, "Physical Separation Criteria for Electrical Cable Trays (Horizontal Open Space Configuration)," November 9, 1977. Available for inspection and copying for a fee in the NRC Public Document Room.
30. U. S. Nuclear Regulatory Commission, Office of Management and Program Analysis, Licensee Event Report File. Available for inspection and copying for a fee in the NRC Public Document Room.
31. U. S. Nuclear Regulatory Commission, Reactor Safety Study, An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants, WASH-1400, Washington, D. C., October 1975, Appendix III.
32. "Report to the APS by the Study Group on Light-Water Reactor Safety," Reviews of Modern Physics, 47, Supplement No. 1, Summer 1975. Available in public technical libraries.
33. Investigation of Charges Relating to Nuclear Reactor Safety, Hearings before the Joint Committee on Atomic Energy, February 18, 23 and 24, and March 2 and 4, 1976; Vol. 1, "Hearings and Appendices 1-11," 1976. Available from the U. S. Government Printing Office, Washington, D. C.
34. F. C. Finlayson, T. A. Hussman, Jr., K. R. Smith, Jr., R. L. Crolius, and W. W. Willis, Human Engineering of Nuclear Power Plant Control Rooms and Its Effects on Operator Performance, ATR-77-(2815)-1, Aerospace Corp., February 1977. Available for inspection and copying for a fee in the NRC Public Document Room.

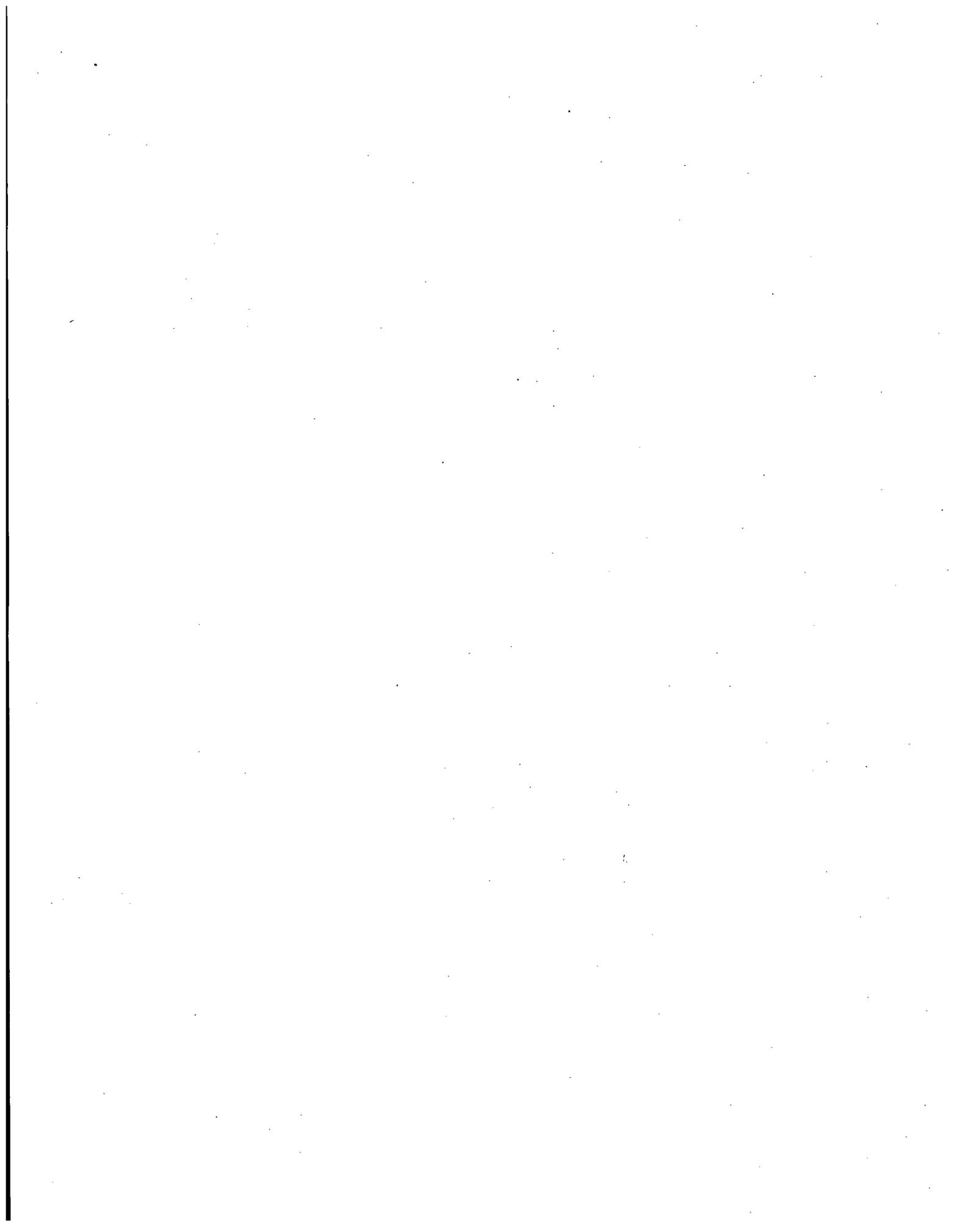
35. American Nuclear Society, "Proposed American National Standard Criteria for Safety-Related Operator Actions," ANS 51.4 (ANSI N660), Hinsdale, Ill., January 1977.
36. Institute of Electrical and Electronics Engineers, "IEEE Recommended Practice for the Design of Display and Control Facilities for Central Control Rooms of Nuclear Power Generating Stations," Std 566-1977, New York, July 1977.
37. Institute of Electrical and Electronics Engineers, "Standard for the Design of Control Rooms for Nuclear Power Generating Stations," Std P567, Draft 3, New York, December 1975.
38. J. N. Shukla and R. H. Wong, "NUCLENET Control Complex," IEEE Control Room Design, Specialists Meeting on, 1975.
39. D. N. Fry, R. C. Kryter, and J. C. Robinson, "Analysis of Neutron-Density Oscillations Resulting from Core Barrel Motion in a PWR Nuclear Power Plant," Annals of Nuclear Energy, 2, 341 (1975). Available in public technical libraries.
40. "Inservice Monitoring of Core and Core Support Structure Motion via Neutron-Flux Measurement," Regulatory Guide Issuance and Availability, Federal Register, 40(126), 27508 (June 30, 1975). Available in public libraries.
41. J. E. Mott, J. C. Robinson, D. N. Fry, and M. P. Brackin, "Detection of Impacts of Instrument Tubes Against Channel Boxes in BWR-4s Using Neutron Noise Analysis," Trans. Amer. Nucl. Soc., 23, 465 (1976). Available in public technical libraries.
42. R. C. Kryter, C. W. Ricker, and J. E. Jones, "Loose Parts Monitoring: Present Status of the Technology and Its Implementation in U.S. Reactors," Progress in Nuclear Energy, 1(2-4) (1977). Available in public technical libraries.
43. R. S. Booth, Noise Diagnostics for Safety Assessment, Quarterly Progress Report for October-December 1976, USNRC Report ORNL/NUREG/TM-97, February 1977.
44. R. C. Kryter, Noise Diagnostics for Safety Assessment, Quarterly Progress Report for January-March 1977, USNRC Report ORNL/NUREG/TM-112, June 1977.
45. R. C. Kryter, Noise Diagnostics for Safety Assessment, Quarterly Progress Report for April-June 1977, USNRC Report ORNL/NUREG/TM-143, September 1977.
46. R. C. Kryter and K. R. Piety, Noise Diagnostics for Safety Assessment, Quarterly Progress Report for July-September 1977, USNRC Report ORNL/NUREG/TM-161, November 1977.
47. F. R. Mynatt, comp., November 1977 Monthly Highlights for Office of Nuclear Regulatory Research Programs at Oak Ridge National Laboratory, USNRC Report ORNL/NUREG/TM-172, December 1977.
48. D. N. Fry, R. C. Kryter, J. E. Stoneking, and F. J. Sweeney, Noise Diagnostics for Safety Assessment, Standards, and Regulation, Quarterly Progress Report for January-March 1978, USNRC Report NUREG/CR-0145 (ORNL/NUREG/TM-207), June 1978.
49. J. C. Robinson, F. Shahrokhi, and R. C. Kryter, Inference of Core Barrel Motion from Neutron Noise Spectral Density, USNRC Report ORNL/NUREG/TM-100, April 1977.
50. D. N. Fry, R. C. Kryter, M. V. Mathis, J. E. Mott, and J. C. Robinson, Summary of ORNL Investigation of In-Core Vibrations in BWR-4s, USNRC Report ORNL/NUREG/TM-101, May 1977.
51. F. R. Mynatt, comp., February 1978 Monthly Highlights for Office of Nuclear Regulatory Research Programs at Oak Ridge National Laboratory, USNRC Report ORNL/NUREG/TM-197, March 1978.

## LIST OF ACRONYMS

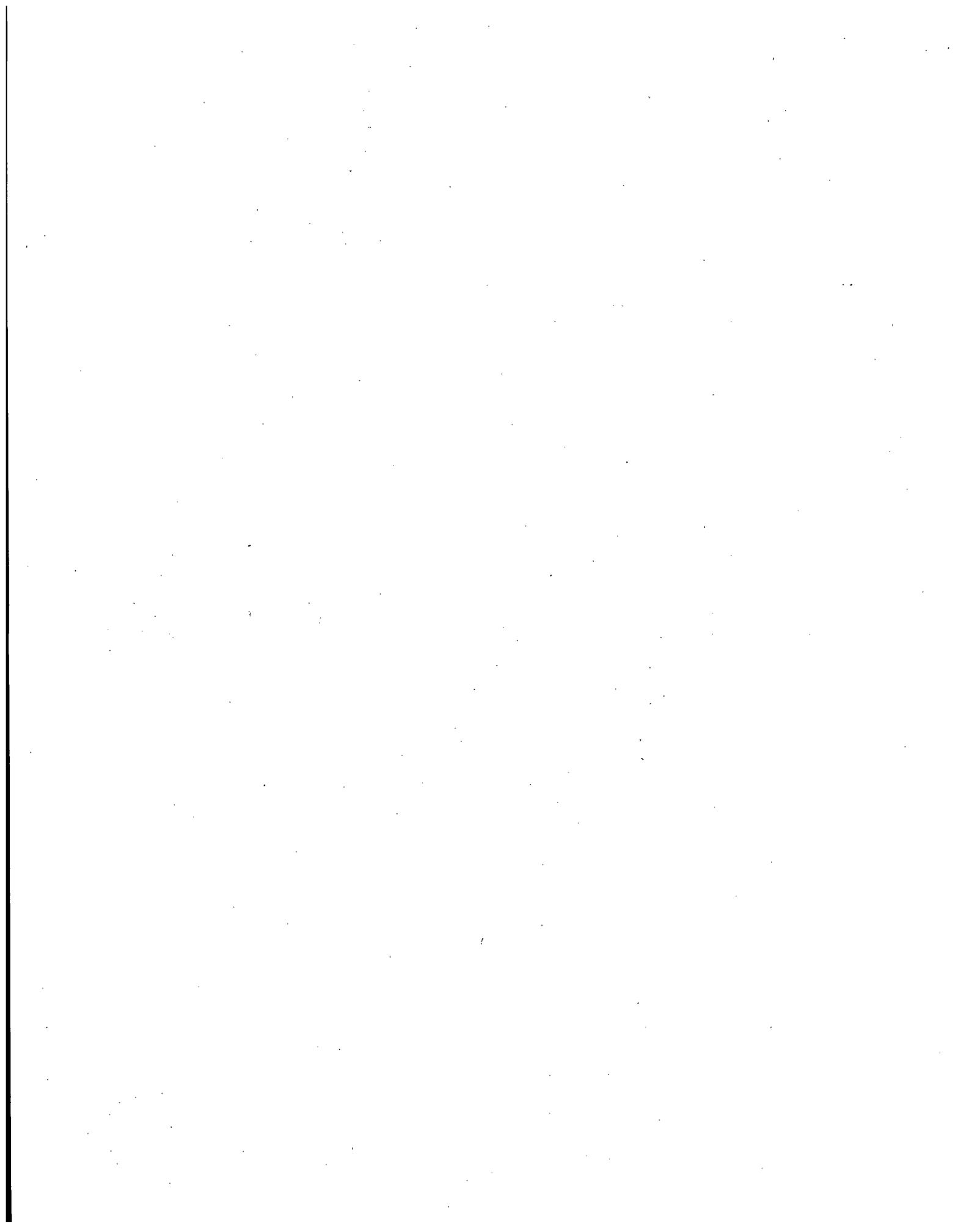
ACRS	Advisory Committee on Reactor Safeguards
AECL	Atomic Energy of Canada, Limited
ANL	Argonne National Laboratory
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
BCL	Battelle Memorial Institute - Columbus Laboratories
BD/ECC	Blowdown/Emergency Core Cooling
BDHT	Blowdown Heat Transfer
BE	Best Estimate
BNL	Brookhaven National Laboratory
BNL	Battelle - Pacific Northwest Laboratories
BT	Boiling Transition
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CISE	Centro Informazione Studi Esperienze (Italy)
DBA	Design Basis Accident
DBPS	Data Bank Processing System
DECL	Double Ended Cold Leg (Break)
DNB	Departure from Nucleate Boiling
DOE	U.S. Department of Energy
EBTF	ECC Bypass Test Facility
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System(s)
EG&G	Prime Contractor for the INEL
EM	Evaluation Model
EPRI	Electric Power Research Institute

FCMI Fuel-Cladding Mechanical Interaction  
 FLECHT Full Length Emergency Cooling Heat Transfer  
 FLECHT-SEASET System Effects and Separate Effects Test  
 FLECHT-SET System Effects Test  
 FRG Federal Republic of Germany  
 GE General Electric  
 HDR Heissdampf reaktor (Nuclear Power Plant in Germany)  
 HTDB Heat Transfer Data Base  
  
 IEEE Institute of Electrical and Electronics Engineers  
 INEL Idaho National Engineering Laboratory  
  
 LASL Los Alamos Scientific Laboratory  
 LEFM Linear Elastic Fracture Mechanics  
 LOBI Loop for Blowdown Investigations (Italy)  
 LOCA Loss of Coolant Accident  
 LOCE Loss of Coolant Experiment  
 LOFT Loss of Fluid Test (Facility)  
 LWR Light Water Reactor  
  
 MeV Million Electron Volts (Particle Energy)  
 MIT Massachusetts Institute of Technology  
 MSLB Main Steam Line Break  
 Mwd/MT Megawatt Day Per Metric Tonne  
  
 NASA National Aeronautics and Space Administration  
 NESC National Energy Software Center (Formerly Argonne Code Center)  
 NRC U.S. Nuclear Regulatory Commission  
 NRR (Office of) Nuclear Reactor Regulation (NRC)  
 NRU Nuclear Reactor "U" (Chalk River, Canada)  
 NSIC Nuclear Safety Information Center (Oak Ridge, Tenn.)  
 NSRR Nuclear Safety Research Reactor (Japan)  
 NTIS National Technical Information Service  
  
 ORNL Oak Ridge National Laboratory  
 OSD Office of Standards Development (NRC)

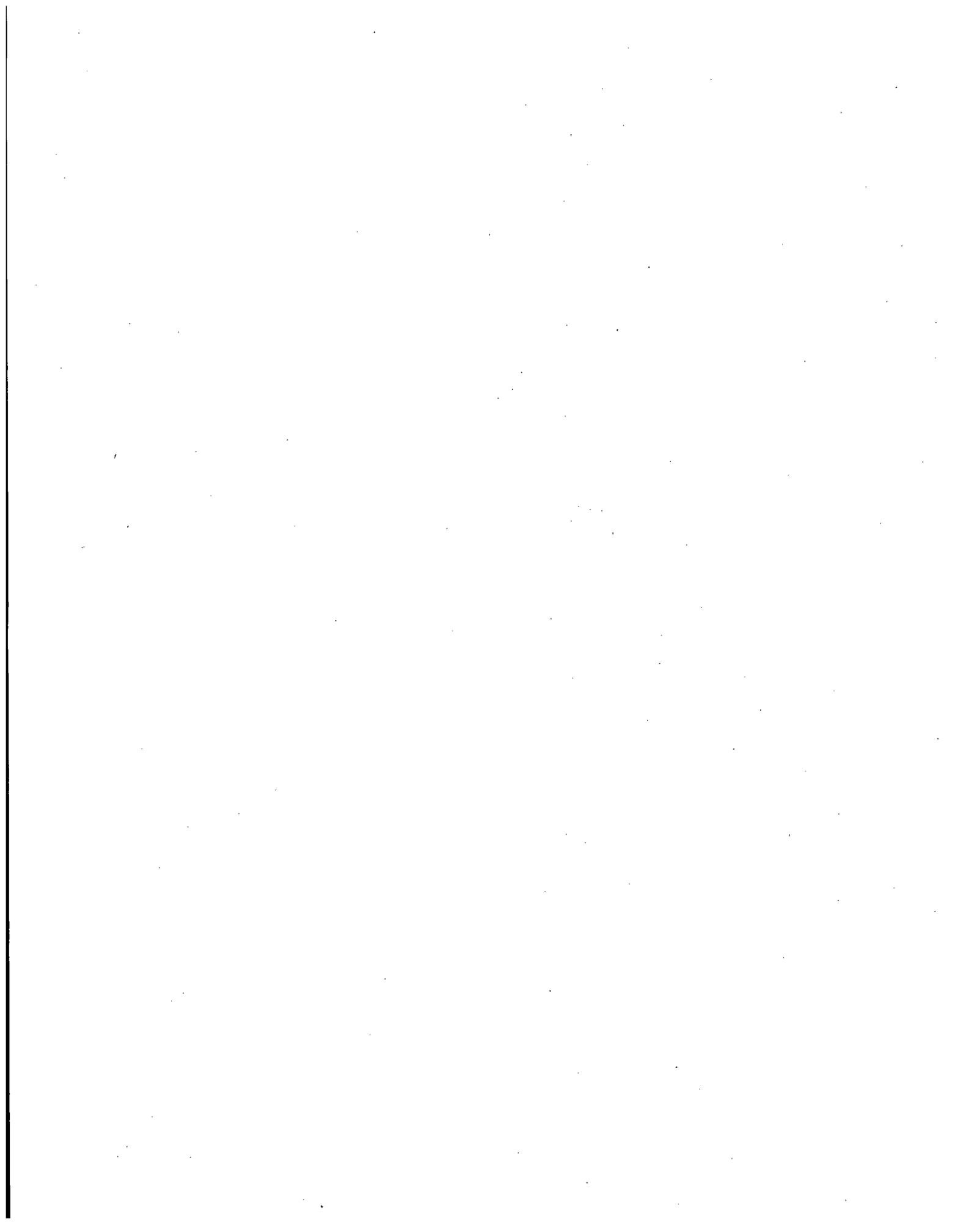
PBF	Power Burst Facility
PCM	Power Cooling Mismatch
PCT	Peak Cladding Temperature
PKL	Primarkreislauf (Hydraulic Test Facility in Germany)
PWR	Pressurized Water Reactor
RES	(Office of) Nuclear Regulatory Research (NRC)
RIA	Reactivity Initiated Accident
ROSA	Rig of Safety Assessment (Japan)
RPI	Rensselaer Polytechnic Institute
RSR	(Division of) Reactor Safety Research (RES)
SCC	Stress Corrosion Cracking
SEM	Scanning Electron Microscope
THTF	Thermal Hydraulic Test Facility
TLTA	Two Loop Test Apparatus
UHI	Upper Head Injection
W	Westinghouse Electric
WREM	Water Reactor Evaluation Model
WRSR	Water Reactor Safety Research, an Office in RSR



<b>NRC FORM 335</b> (7-77)		<b>U.S. NUCLEAR REGULATORY COMMISSION</b> <b>BIBLIOGRAPHIC DATA SHEET</b>		1. REPORT NUMBER <i>(Assigned by DDC)</i> <b>NUREG-0006</b>	
4. TITLE AND SUBTITLE <i>(Add Volume No., if appropriate)</i> <b>WATER REACTOR SAFETY RESEARCH PROGRAM</b> <b>A Description of Current and Planned Research</b>				2. <i>(Leave blank)</i>	
7. AUTHOR(S) <b>Division of Reactor Safety Research</b>				5. DATE REPORT COMPLETED MONTH <b>July</b> YEAR <b>1978</b>	
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS <i>(Include Zip Code)</i> <b>Office of Nuclear Regulatory Research</b> <b>U. S. Nuclear Regulatory Commission</b> <b>Washington, D. C. 20555</b>				DATE REPORT ISSUED MONTH <b>February</b> YEAR <b>1979</b>	
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS <i>(Include Zip Code)</i> <b>Same</b>				6. <i>(Leave blank)</i>	
				8. <i>(Leave blank)</i>	
				10. PROJECT/TASK/WORK UNIT NO.	
				11. CONTRACT NO.	
13. TYPE OF REPORT <b>Program Plan</b>			PERIOD COVERED <i>(Inclusive dates)</i>		
15. SUPPLEMENTARY NOTES				14. <i>(Leave blank)</i>	
16. ABSTRACT <i>(200 words or less)</i> <p>The NRC's confirmatory research program on light water reactors (LWRs) is structured to provide additional and/or independent information to gain confidence that the margins of safety identified in the licensing review are well defined and quantified. The objectives, present status, research program and milestones of the NRC LWR safety research are given for each program element. These program elements are systems engineering (thermal-hydraulic data under LOCA/ECCS conditions), fuel behavior, computer code development, metallurgy &amp; materials (integrity of the primary system pressure boundary), and research support (includes reactor operational safety research).</p>					
17. KEY WORDS AND DOCUMENT ANALYSIS <b>Light Water Reactor</b> <b>Reactor Safety Research</b> <b>Code Development</b>			17a. DESCRIPTORS <b>Metallurgy &amp; Materials</b> <b>Fuel Behavior</b> <b>Systems Engineering</b>		
17b. IDENTIFIERS/OPEN-ENDED TERMS					
18. AVAILABILITY STATEMENT <b>Unlimited Distribution</b>			19. SECURITY CLASS <i>(This report)</i> <b>Unclassified</b>		21. NO. OF PAGES <b>159</b>
			20. SECURITY CLASS <i>(This page)</i> <b>Unclassified</b>		22. PRICE <b>\$</b>









UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

OFFICIAL BUSINESS  
PENALTY FOR PRIVATE USE, \$300

POSTAGE AND FEES PAID  
U.S. NUCLEAR REGULATORY  
COMMISSION



120555004118 4 R1R2R3R4R5RPR  
US NRC  
RES ANALYSIS DEVELOPMENT  
BRANCH CHIEF  
1130SS  
WASHINGTON DC 20555