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Selected Review and Evaluation of U.S. Safety Research Vis-a-Vis Foreign Safety Research for Nuclear Power Plants

Prepared by J. D. Stevenson, F. A. Thomas

Stevenson & Associates

Prepared for
U.S. Nuclear Regulatory Commission

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Previous Documents in Series:

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Stevenson, J.D. and F.A. Thomas, "Selected Review of Regulatory Standards and Licensing Issues for Nuclear Power Plants," USNRC Report NUREG/CR-3020, November 1982.

Stevenson, J.D. and F.A. Thomas, "Selected Review of Foreign Safety Research for Nuclear Power Plants," USNRC Report NUREG/CR-3040, November 1982.

ABSTRACT

A review of currently available nuclear safety research resources in a selected group of United States government national laboratories is presented. The current nuclear safety research interests of industry organizations, particularly the Electric Power Research Institute, are also identified. In addition, suggestions for potential joint or cooperative funding of light water reactor safety research in the U.S. between the NRC and other organizations, both foreign and domestic, are presented. The topics of research considered are associated with the areas of Siting, Structural Engineering, Metallurgy, Materials, and Mechanical Engineering.

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PREFACE

This report is a summary of currently available nuclear safety research resources and interests in a selected group of United States government national laboratories and research facilities. The report also includes a summary of current safety research interests of the Electric Power Research Institute and a discussion of the potential efforts for cooperative funding of light water reactor safety research between the NRC and other organizations both foreign and domestic. Topics of interest associated with the areas of Siting, Structural Engineering, Metallurgy, Materials and Mechanical Engineering are considered.

This is the fourth in a series of reports including the following: foreign licensing practice; foreign regulatory standards and current licensing issues; and a selected review of foreign safety research for nuclear power plants.

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1. INTRODUCTION

1.1 Scope and Purpose

The majority of nuclear safety research in the U.S. is sponsored by the U.S. Nuclear Regulatory Commission; therefore, current programs and projects are well-known and understood by the Nuclear Regulatory Commission. In addition, future research planning is contained in NUREG-0784, the Long Range Research Plan FY-1984-1988.(Ref.1) However, it is generally understood that the direction and thrust of nuclear safety research for light water reactors in the selected technical areas of siting, structural and mechanical engineering, and materials and metallurgy are expected to undergo significant change in the future. The anticipated safety research emphasis is expected to shift from siting and construction of new plants to maintenance and possible safety improvement of existing factilities to research eventually required for decommissioning of existing facilities. In order to plan for these anticipated changes, the Nuclear Regulatory Commission should have a comprehensive understanding of the physical and intellectual safety research resources available in various U.S. and foreign research organizations and facilities.

One of the objectives of this study is to indicate the currently available nuclear safety research resources in a selected group of U.S. government national laboratories and research facilities. The study also identifies additional possible major sources of nuclear safety research funding by other U.S. government agencies such as the Department of Energy.

Another objective of this study is to examine industry organizations which sponsor nuclear safety research, particularly the Electric Power Research Institute. Other industry organizations which sponsor nuclear safety research are electric utility owner's task groups which address specific licensing or regulatory issues, the Pressure Vessel Research Committee of the Welding Research Council, and in-house research programs of nuclear steam systems suppliers. An example of a utility-sponsored research program is the testing of seismic response of electrical cable raceways at the University of California shake table facility.

This study also examines nuclear safety research sponsored by foreign organizations and performed in the U.S. or abroad. Comparisons are made between selected safety research activities in the United States and similar activities in Canada, France, Japan, Sweden, United Kingdom of Great Britain and Northern Ireland, and the Federal Republic of Germany.

In addition, this study serves as a forum where suggestions for future research projects proposed by current national laboratory researchers are expressed. Finally, through the examination of research capabilities, interests, and future needs, this study identifies areas of potential cooperative funding of light water reactor safety research between the U.S. Nuclear Regulatory Commission and other organizations both foreign and domestic. These joint-funding efforts would augment the research activities or limit the total safety research expenditures on the part of the U.S. Nuclear Regulatory Commission.

1.2 Facilities and Organizations Surveyed

The particular facilities and organizations surveyed in this report are as follows:

(1) Argonne National Laboratory
Argonne, Illinois

(2) Brookhaven National Laboratory
Upton, New York

(3) Electric Power Research Institute
Palo Alto, California

(4) Idaho National Engineering Laboratory
Idaho Falls, Idaho

(5) Lawrence Livermore National Laboratory
Livermore, California

(6) Los Alamos National Laboratory
Los Alamos, New Mexico

(7) Oak Ridge National Laboratory
Oak Ridge, Tennessee

(8) Sandia National Laboratory
Albuequerque, New Mexico

These facilities were visited during 1982 by Dr. John D. Stevenson in order to obtain information concerning particular areas of expertise and research facilities available at the laboratory, except for Oak Ridge National Laboratory where information was obtained through correspondence. Also discussed during the visit were any potential sources of safety research funding from foreign or domestic sources in the technical areas of interest. Approximately one month prior to the visit, a letter questionnaire, as shown in Appendix A, was sent to the facilities surveyed identifying the requested information.

1.3 Organization and Scope of the Report

Th information obtained from the responses to the letter questionnaire sent to the laboratories and organizations listed above is discussed in Section 2 through Section 9 of this report. A summary is presented in Section 10, along with the conclusions derived from this information.

It should be understood that this report was prepared from information voluntarily supplied without reimbursement by foreign institutions and organizations, national laboratories, and the Electric Power Research Institute. For this reason, there is a considerable variation in the quantity of data presented in the text of this report as a function of the information provided by the various organizations surveyed.

In this report, only U.S. safety research conducted at the seven national laboratories listed in Section 1.2 or sponsored by the Electric Power Research Institute are identified. The U.S. NRC supports significant safety related research in Universities and other research organizations not identified and included in this report.

2. ARGONNE NATIONAL LABORATORY

2.1 Organization

The Argonne National Laboratory (ANL) nuclear safety research efforts in the technical areas of interest covered in this report are concentrated in the Reactor Research and Development Divisions Group (RRD) and Physical Research Diviions Group (PR) as shown in Figure 1. Within the Reactor Research and Development Divisions Group, the divisions most concerned with the areas of interest identified in this report are the Reactor Analysis and Safety Division (RAS), R. Avery-Director, and the Components Technology (CT) Division, R.S. Zeno-Director. Within the Physical Research Divisions Group, the Materials Science and Technology Division (MST), B.R.T. Frost-Director, is of most interest. The organization of these divisions is shown in Figures 2, 3 and 4.

Within the three divisions listed, most of the mechanical, structural, and materials work is concentrated in the following sections: in CT, under Structural Systems Analysis (C.A. Kot, Manager); in RAS, either Structural Mechanics (Y.W. Chang) or Computational Mechanics (J.M. Kennedy); in MST, the several groups under R.W. Weeks, who is the Associate Division Director in charge of the materials technology effort. The RAS division, while a major center for DOE-funded reactor safety research in the sodium breeder area, currently does not perform any work for NRC in the areas of interest in this report. It should be noted, however, that the engineering mechanics program of the RAS division does extensive structural mechanics computer code development work, and most of the codes would be directly applicable to LWR problems. Organizations sponsoring safety-related light water reactor research at ANL include the U.S. Department of Energy, Advisory Committee on Reactor Safeguards (ACRS), Atomic Safety and Licensing Board (ASLB), NRC Office of Nuclear Material Safety and Safeguards, NRC Office of Nuclear Reactor Regulation and the NRC Office of Nuclear Regulatory Research.

The ANL currently has approximately 35 separate NRC accounts and does about $10 million per year in NRC-sponsored work.

2.2 Topic Areas of Interest

Recently, the three ANL Divisions listed, RAS, CT, and MST have provided consultants to the ACRS and ASLB and have performed research in technical areas related to this study as indicated in this section.

2.2.1 Department of Energy

Argonne currently does not have any DOE-sponsored work in LWR's in the areas of siting, structural engineering, metallurgy, materials, and mechanical engineering. It should be noted, however, that Argonne has extensive DOE-sponsored work in these areas supported by the LMFBR program and by other energy technologies. The yearly research budget at ANL in these topics exceeds $30 million, with the largest single area

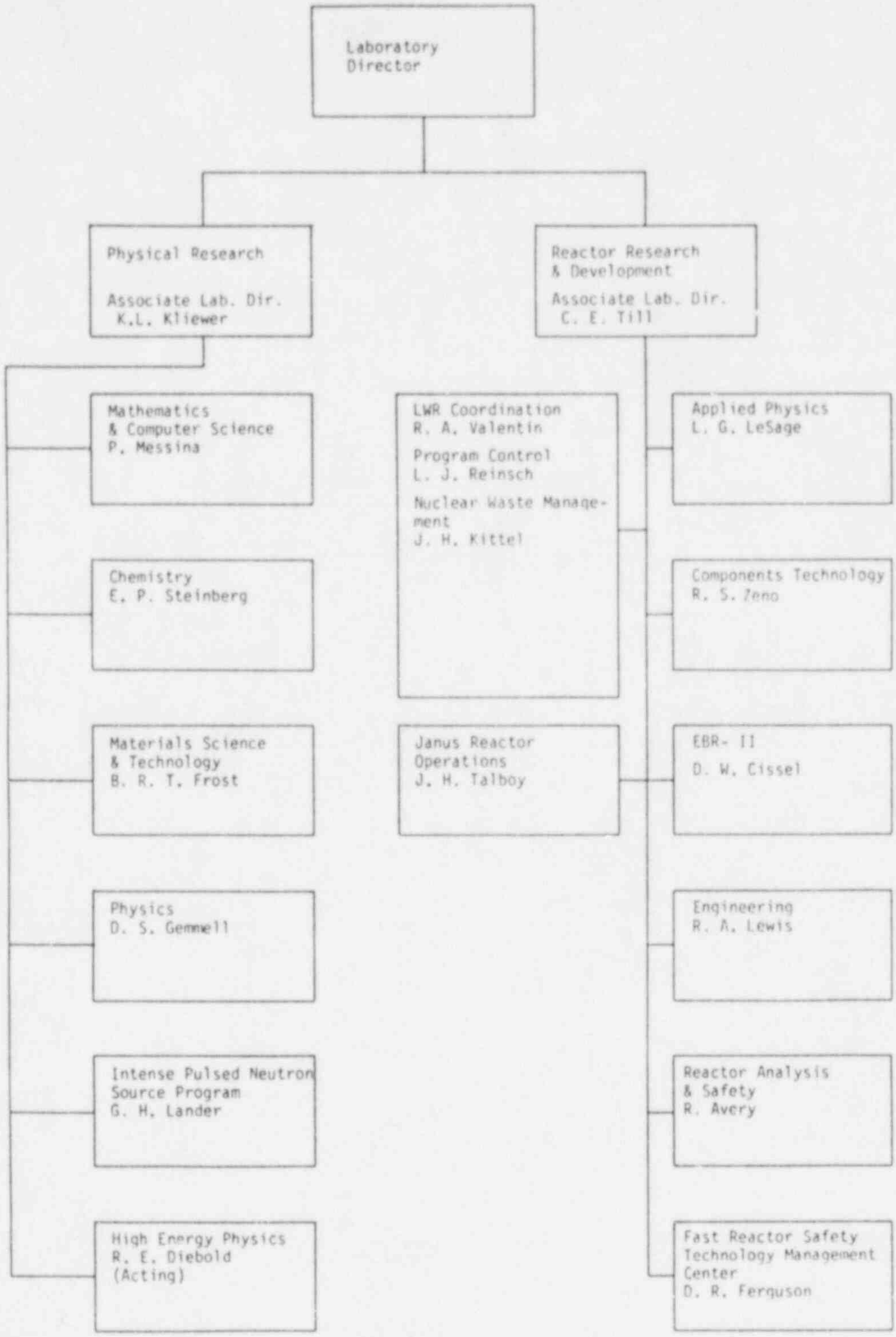

Figure 1 Organization of Argonne National Laboratory Divisions of Interest

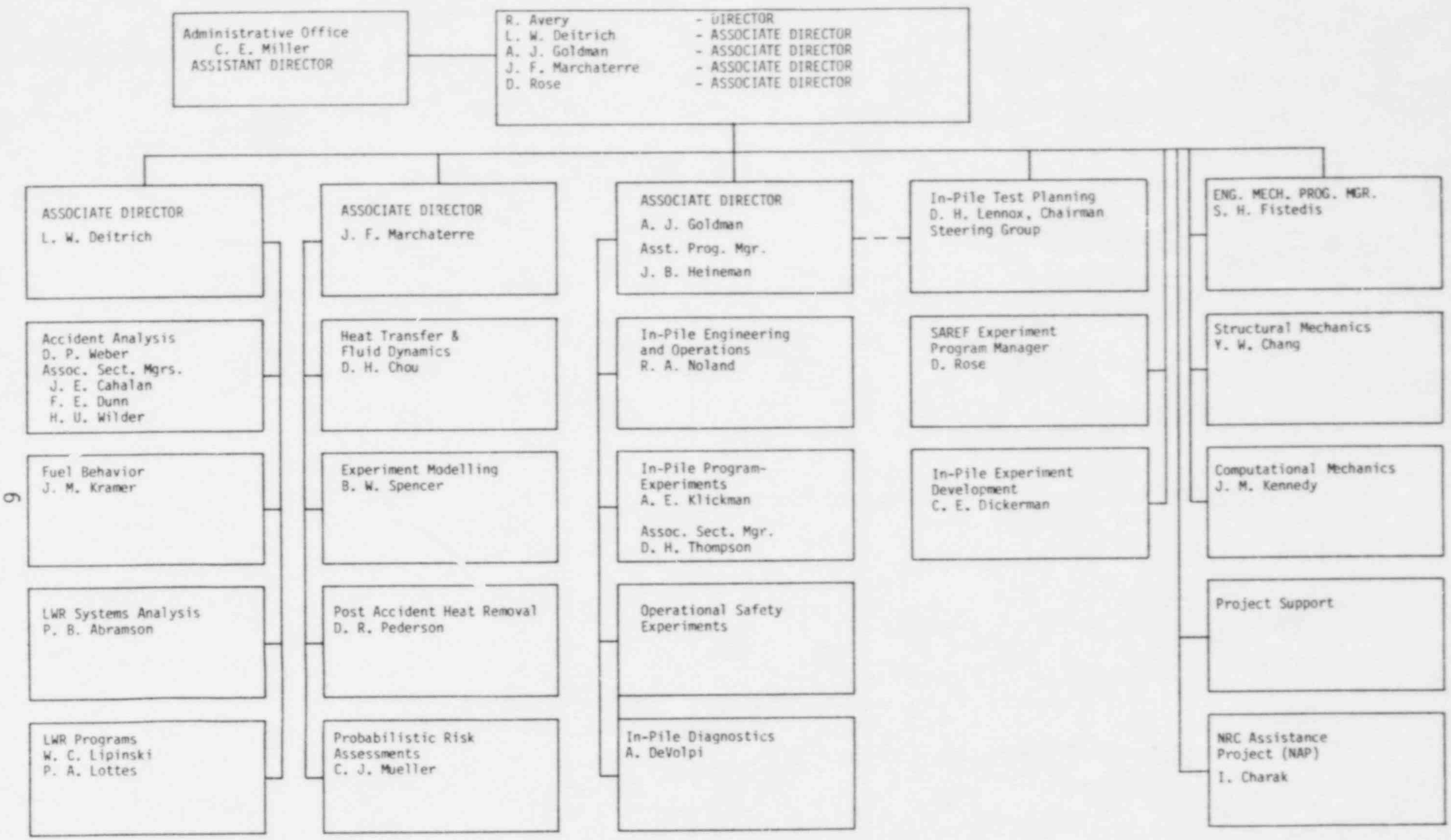

Figure 2 Reactor Analysis and Safety Division

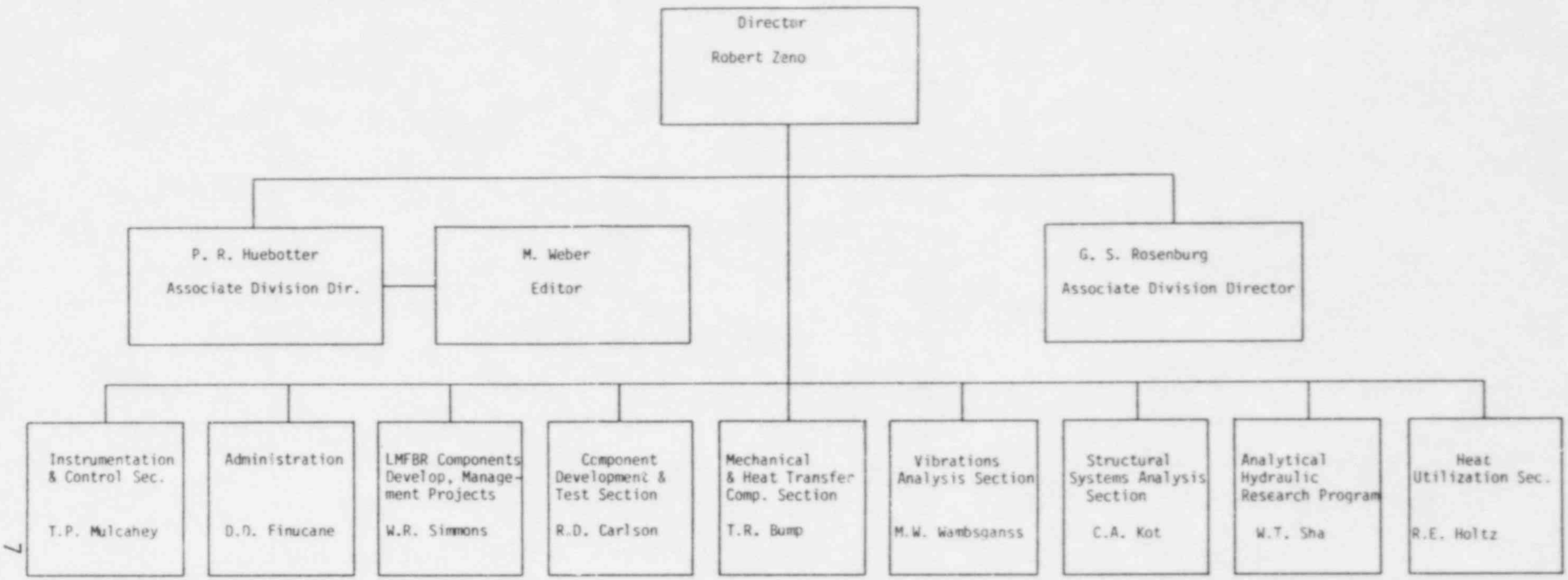

Figure 3 Components Technology Division

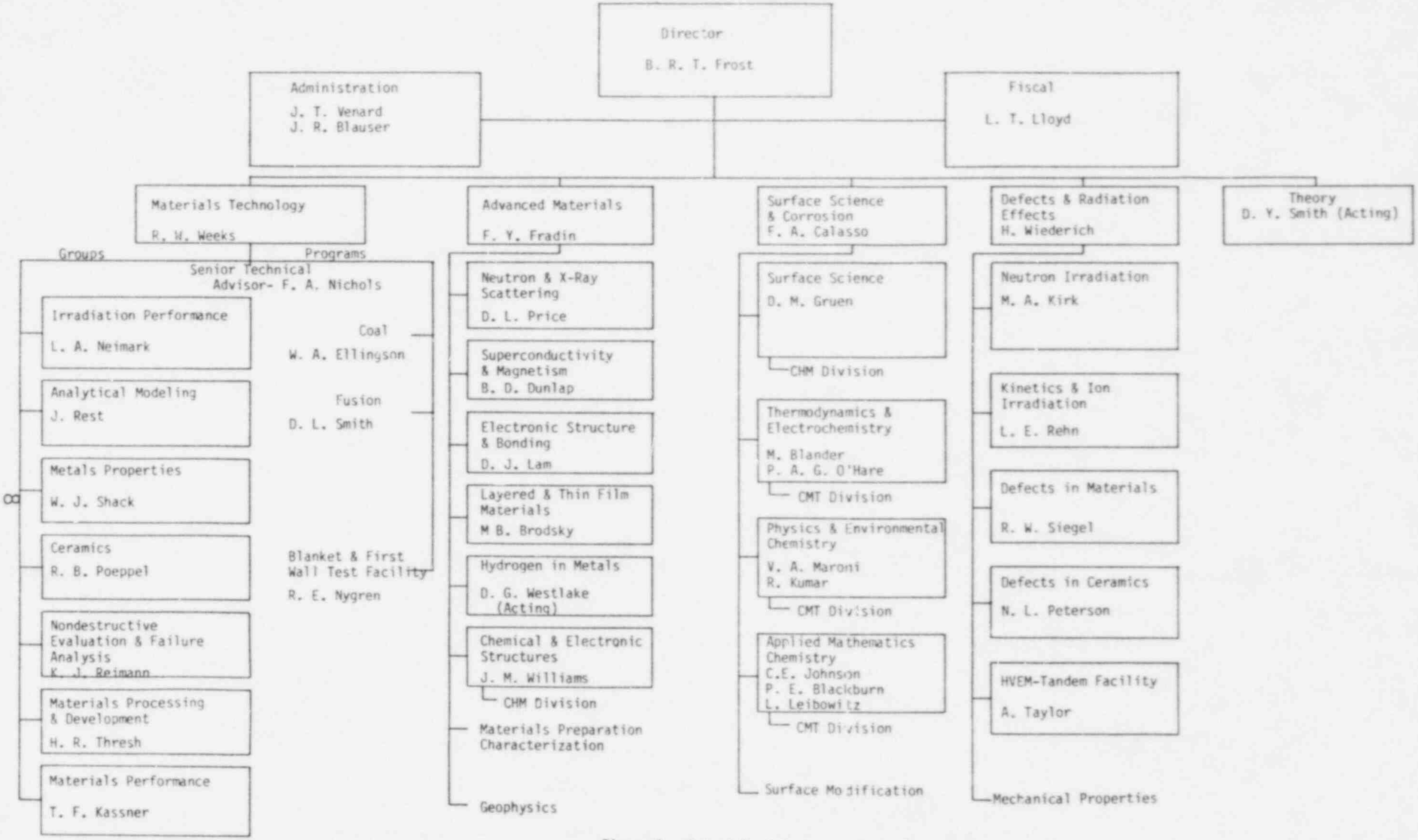

Figure 4 Materials Science and Technology Division

being materials. It is not feasible to list all such programs having direct application to LWR technology, however, a typical example is shown in Section 2.8.2 on computer software. That section lists ten large computer codes developed with LMFBR funding that are directly applicable to LWR problems. Similarly, a considerable fraction of ANL's $11 million per year applied materials technology effort has potential LWR application.

2.2.2 NRC Office of Nuclear Material Safety and Safeguards

(1) Environmental Impact Statement of the Decomissioning of the Kerr-McGee West Chicago Facility

2.2.3 NRC Office of Nuclear Reactor Regulation

(1) Licensing Program - Environmental Assistance

(2) Vibration Monitoring of Steam Generator Tubes

2.2.4 NRC Office of Nuclear Regulatory Research

(1) Analysis of Offsite Hazards and Their Effects on Nuclear Facilities

(a) Study of Hazards to Nuclear Power Plants from Large Liquified Natural Gas Spills on Water Transporation Routes

(b) Blast Hardening Design Criteria for Nuclear Power Plants

(2) Environmentally Assisted Cracking in LWR Systems

(3) Evaluation of Dynamic Testing Methods for Safety Assessment of Nuclear Power Plant Structures

(4) Loss-of-Benefits Estimation from Nuclear Plant Outages (Formerly Formalized Risk Decision-Making Processes)

(5) Modification of Backfill Materials under Repository Conditions

(6) Reliability Assurance Program

2.3 Facilities

2.3.1 Experimental

2.3.1.1 Flow-Induced Vibration Test Facility (FIVTF)

The FIVTF has four pumps with flow rates of 500, 1000, 2500, and 4000 gpm at 150 psig discharge pressure. The pumps discharge individually through their own control and by-pass valves, flowmeters and piping (including pressure and temperature indicators) to an accumulator. Thus, the flow can be varied to provide stable water flow from 50 gpm to a maximum of 8000 gpm by operating combinations of the pumps and valving. To allow ease of operation by the experimenter, all pumps and valves are operated remotely from the existing data acquisition facility via a control console.

To provide for "quiet flow," various design considerations have been employed. Piping with a minimum number of bends (all long radius), and of the largest diameter practical, is used to reduce turbulence. Structural-borne vibrations are minimized by using vibration attenuation pads at the pipe support locations, which are close together (for short span length). Flexible piping sections are used for connection to the pumps and at the entrance and exit to the test sections.

The 8000-gal accumulator at the entrance to the test legs has a maximum turnover rate of once per minute at the highest flow rate. This turnover rate, along with internal baffling and the small height-to-diameter ratio, tends to isolate the pump supply from the test item. Three large diameter nozzles (two horizontal and one vertical), for attachment of test items to the accumulator, are provided with isolation valves, so that one test can be performed while another is being assembled.

The pump supply tank is constructed of concrete to reduce vibration and has a large volume ($\sim$10,000 gal) to accommodate baffling and water-conditioning equipment. The conditioning equipment is used to maintain the quality of the water so that the transducers for the data acquisition system will not be adversely affected by variable water conditions. For example, during an experiment, water temperature will not vary enough to affect water viscosity, and water resistance is maintained at a high level so that transducer shorting-to-ground is minimized. Make-up water is circulated through the appropriate units so that its quality will be the same as that of the loop.

Water Quality Variables	Control Methods
Mean temperature	Cooling tower
Water resistance and chemical content	Ion exchange column
Particulate matter (up to 7 microns)	Filters
Bacteria content	Ultraviolet sterilization unit

An overhead crane with a 10-ton capacity services the area occupied by the FIVTF.

Principal uses of the FIVTF include the following:

(1) Flow-induced vibration tests of nuclear and nonnuclear systems and plant components including valves, reactor upper plenum components, instrumentation wells, thermal liners, and flow-directing baffles,

(2) Experimental studies of tube bundles in crossflow, single cylinders in skewed flow, and cylinders subject to inflow turbulence and wake buffeting, and

(3) Performance of tests, both scale-model and prototypic, to evaluate specific LMFBR component designs from the standpoint of flow-induced vibrations.

2.3.1.2 Material Science (MS) Division Experimental Facilities

The MS Division has the following major experimental test components available:

(1) 1250-ton Lake Erie extrusion press

(2) 350-ton Lombard extrusion press

(3) Harders 400-ton press

(4) Lake Erie 120-ton fast-acting press

(5) Loomis 30,000-psi isostatic press

(6) NRC 6-in. arc-melting furnace

(7) NRC electron beam-melting furnace

2.3.2 Computer Software

Over the past several years the Reactor Analysis and Safety Division has developed a number of computer codes for handling complex fluid-structure and other structural dynamics problems associated with the safety of commercial LMFBR plants which should also have some applicability to LWR facilities. These codes are identified as follows:

(1) The REXCO code system, a series of finite difference Lagrangian codes, particularly applicable to short duration, high intensity transients,

(2) ICECO, the Eulerian counterpart to REXCO, which can handle long duration events,

(3) STRAW, a finite element, 2-D fluid-structure code which has augmented third dimension capability,

(4) ICEPEL, a hydrodynamic-structural analysis code for complex piping systems with elastic-plastic capability,

(5) SHAPS, a hydrodynamic-structural analysis code for complex piping systems augmented with 3-D capability for treating flexural motions of pipe elements,

(6) ALICE, an arbitrary Lagrangian-Eulerian, Implicit-Explicit code,

(7) NEPTUNE, a finite-element, 3-D, fluid-structure interaction code for analysis of in-tank components and top closures for large LMFBR primary sodium vessels,

(8) SADCAT, a 3-D code for static or dynamic structural analysis augmented with thermal capabilities,

(9) DYNAPCON, an axisymmetric code which is used to model reinforced or prestressed concrete vessels (or cells) to determine response to energetic core disruptive accidents,

(10) SAFE/RAS, a finite element code used to analyze dynamic loadings on the upper internals located above the reactor core and reactor vessel head for LMFBR plants.

These codes represent the most advanced state-of-the-art which exists for containment and structural dynamics analysis in this area of LMFBR safety and form the basis for all foreign codes used in the area. It is important to note that most of this work has been experimentally verified and, further, that the analytical tools developed have been used directly in support of safety reviews for the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor project (CRBR). The program is well-documented throughout all phases of the work: (a) mathematical formulation, (b) code writing, (c) experimentation, (d) experimental verification and modification of codes, and (e) application to actual LMFBR reactor projects (in the US and abroad).

2.3.3 Research Staff Capabilities

2.3.3.1 Materials Science and Technology (MST) Division

(1) LWR Component Failure Analysis

MST has special facilities to conduct detailed failure analyses on radioactive LWR components removed from service, including fuel elements as well as structural components. Work of this type has been performed by MST for utilities as well as for the NRC.

(2) Nondestructive Evaluation (NDE)

MST has a full range of NDE facilities and equipment, including field as well as lab experience in most techniques. A special water-filled 10" piping facility allows for acoustic tests on cracked LWR piping. Radioactive pipe segments can be incorporated into this loop and have been in recent tests. State-of-the-art ultrasonics are used along with other NDE techniques to characterize cracks and flaws in LWR components.

(3) Fracture Mechanics

LEFM as well as elastic/plastic fracture mechanics and finite element techniques are available in MST. A number of MTS servo-hydraulic systems are in operation for conducting high temperature cyclic crack growth and fracture toughness experiments in air, in ultra-high vacuum, in liquid metals, and under LWR environmental conditions. Creep-fatigue testing is also carried out under both uniaxial and multiaxial loading conditions.

2.3.3.2 Components Technology (CT) Division-Structural Systems Analysis Section (SSA)

The SSA Section is devoted to the development of analytical and numerical methods for studying the response of structures, components and flow systems to various classes of normal and off-normal loading. Typical current work includes the study of pressure pulse propagation in LMFBR heat transport systems, effects of large sodium-water reactions, effects of offsite hazards on nuclear power plants, the structural analysis of devices associated with advanced energy systems and the evaluation of dynamic testing for safety assessment of nuclear power plants. The eight staff members have backgrounds in engineering, applied mechanics and structural analysis, or mechanical engineering.

Over the past two years, a series of computer programs (SWAAM, PTA, and STAC codes) have been developed by the staff under DOE sponsorship for the analysis of fluid transients in heat transport systems. These codes are capable of analyzing transients in complex piping networks both for single-phase and two-phase flow. Many unique features are included, for example, plastic pipe deformation, cavitation, filling of pipes, some fluid-structure interactions, and a precise treatment of pipe junctions in two-phase flow using the method of characteristics. The computer programs have been extensively verified by comparison with experiment and have been applied to the analyses of pressure transients in both LMFBR heat transport systems and LWR piping networks.

The SSA Section also operates a pressure pulse test facility which can be used to evaluate experimentally the response of various flow component models to pressure transients. Currently in progress are experiments evaluating the performance of pressure pulse suppression and energy-absorbing devices.

Another major computer program which was developed by SSA Section staff and others is the SCRAP Code. This code provides a core seismic analysis method which is capable of modeling the various physical phenomena of the complex fluid-structure-impact problems which arise particularly in LMFBR cores. However, the code provides considerable flexibility in choosing the model which makes it particularly suitable for conducting sensitivity studies and for extension to other applications.

In the past the SSA Section staff members have also been active in many areas of structural analysis, in particular, the field of dynamic plasticity and the development of viscoplastic models for engineering materials. In addition, some SSA staff members have participated, on a temporary basis, in the NRC review of applicant safety analysis.

2.4 Areas of Future Interest Expressed by Laboratory Researchers

Within the three Divisions, the following areas of interest identified by Argonne personnel for future consideration are summarized.

2.4.1 Reactor Analysis and Safety (RAS) Division

2.4.1.1 Improved Understanding of The Performance of Primary Containment Structures Under Core Melt Conditions

This topic can be studied by using existing codes, suitably modified, such as ALICE and SAFE/RAS. Treatment of slug impact loads on the reactor vessel head can also be accommodated using the REXCO and ICECO codes. DYNAPCON may prove to be useful in analyzing effects of corium spillage on the concrete portions of containment. DYNAPCON is now undergoing modifications to handle thermal effects; also, models have been developed to incorporate important phenomena such as water release from concrete under high temperatures.

2.4.1.2 Fluid-Structure Interaction in PWR Steam Generator

Loss-of-coolant events or other causes incur dynamic loads on the steam generator, particularly in U-bends of the tube bundle. Failure modes in this component dictate further failure phenomena in the system which is not well understood. The response of the U-tubes and their relationship to the influence of the surrounding fluid and fluid-structure interaction can be studied through use of a combination of codes developed by ANL/RAS such as SHAPS and SAFE/RAS. Furthermore, if it is necessary to study the effects of flexibility of the baffle plates in the steam generator, a 3-D model and code of the lower plenum region in the steam generator would need to be employed. Such a code is now under development in the RAS Engineering Mechanics Program and should be applicable to this task.

2.4.1.3 Integrated Analysis of Piping Systems

Nuclear piping systems are required to withstand safely the effects of many different loads and load combinations. Conventional piping analysis methods currently employed commonly treat these phenomena in a simple, uncoupled way. For example, fluid transients are often treated utilizing water hammer theory and a 1-D nonlinear method of characteristics formulation that, in effect, treats the hydrodynamics aspects of pressure pulse propagation along piping systems. In these simplified methods, various components such as elbows, valves, heat exchangers, etc., are replaced by some type of equivalent straight pipe segments. There are many simple assumptions necessary to carry out the analysis. Excessive overdesign and overconservatism results.

Argonne has proposed to develop a computer program which will have the capability for independent assessment (such as by NRC) of the entire piping system and its response to postulated accidental events, either individually or collectively, should be considered. Such a code development would build upon a firm base established in development of 3-D piping codes by ANL/RAS, such as SHAPS. (At present Brookhaven National Laboratory is being funded by the NRC in the development of a program for independent assessment of piping systems.)

2.4.1.4 Pipe Whip Analysis

System response in a pipe whip event is very complex. As noted in Section 2.4.1.3 above for other piping loads, conventional methods used today tend to be very conservative and depend upon many simplifying assumptions. As in the integrated piping system response code development noted herein, SHAPS, now under development, provides an excellent base to develop a special purpose pipe whip code.

2.4.1.5 Seismic Load Characteristics and Nonlinear Response Methods

A major difficulty in nonlinear analysis of seismic problems is the great length of seismic records. A seismic record is generally nonstationary with increasing excitation in the first part of the record, followed by a maximum intensity stationary portion, and then a long decaying portion. A large part of these records is low level excitation with little potential for damage in nonresonant situations. Design methods currently avoid some of these difficulties by characterizing design earthquakes by means of linear response spectra methods. In these methods, the magnitude of the response is bounded by characterizing the seismic event by a response spectrum which gives the maximum excitation expected for a given frequency. The natural frequencies of the structure are found from a numerical model or by testing, and the maximum excitation is then found by taking the square root of the sum-of-the-squares of the excitation obtained for each of the frequencies. Since these methods involve the concepts of natural frequencies and superposition, they are not applicable to nonlinear structures.

The benefits of devising alternate load characterizations and analytical methods for nonlinear aspects of seismic response would be substantial. Response spectra methods which are currently predominant are not suited to nonlinear analysis; the entire concept of a response spectrum is embedded within the notion of linear response. Although the use of complete time-histories from specific seismic events such as EL Centro is applicable to nonlinear analysis, the relationship between the intrinsic factors in the record that govern the magnitude of the nonlinear response is clouded by this approach. Furthermore, scaling the input to other sites becomes a matter of considerable empiricism, and records with completely different nonlinear response characteristics can be generated.

Only by a careful study to determine the factors critical to nonlinear response can these inconsistencies be avoided. Thus, new load characterizations and analytical methods would enhance the confidence in the seismic safety of components and decrease the uncertainties in seismic response which in linear analysis are embodied in safety factors. These safety factors are desirable, but they may be unduly conservative and thus add significantly to the cost of these components.

2.4.2 Materials Science and Technology (MST) Division

2.4.2.1 Decontamination of PWRs

In order to reduce radiation exposure levels in LWR operations, on-line and off-line decontamination procedures are required. Off-line procedures exist that are applicable to BWRs, which because of the relatively high oxygen levels in the coolant present, form loosely-adherent oxide scales. In PWRs, which have much lower oxygen levels, corrosion products tend to form tightly-bound spinel layers on exposed surfaces, and to date, no satisfactory procedure exists for decontaminating PWRs. Both off-line and on-line procedures must be qualified with respect to their effectiveness, side effects, and safety considerations.

2.4.2.2 Improved NDE Techniques for Steam Generator Inspection

Because tube failures in steam generators can lead to serious accident sequences, improved NDE systems for the tubing have significant potential safety benefits.

A unique pulsed-eddy current system has been developed by the ANL-NDT Group. Although it is not commercially available, it has been employed successfully by Aerojet Nuclear Corp. and General Electric Company to inspect cladding for nuclear fuel. The system could also be used for in-service inspection of steam generator tubes and has the following unique advantages over conventional probes: (1) it averages signals over a very small area and thus can give readings at continually varying positions circumferentially around a tube; (2) it is small and can fit in a carriage that can travel down a tube with varying diameter; (3) it is more sensitive than conventional systems; and (4) it can be adjusted to be insensitive to lift off.

The system can be used to detect wall thinning (as a function of circumferential position in the tube) and very small anomalies in the tube wall. Measurements of variations in wall thickness of $\sim$1 mil has been shown to be feasible for a 7/8-in. diameter stainless steel tube with a nominal wall thickness under laboratory conditions. Pulsed-eddy current or multiple crystal ultrasonic probes may also be useful in measuring the extent of denting in steam generator tubing.

2.4.3 Components Technology (CT) Division

2.4.3.1 Offsite Hazards (Siting)

In reviewing the treatment of offsite hazards to nuclear power plants, it becomes clear that many aspects of the phenomenological modeling and accident scenario definition have only been addressed in a cursory fashion. Particular areas which come to mind are the dispersion of hazardous materials and the definition of threat and response in case of fire and explosion. The development of global models, which nevertheless contain the most salient features of the phenomenology, together with extensive sensitivity (parameter) studies should provide a much-improved description of the threat environments due to hazardous materials.

2.4.3.2 Fragility Methods

Probabilistic Risk Assessments require fragility distributions for structures and components under various loading scenarios. Seismic fragility methods are perhaps the most urgent need, but fragility due to other postulated events such as blast loading, internal pressure, and airplane impact present similar problems. Key questions include the following: definitions of failure; methods to properly account for loading uncertainties including loading rates, loading areas, and redistribution of loads during failure; methods to account for energy absorption (for example, higher damping at high deformations, plastic deformation); treatment of hardening nonlinearities (for example, gap closures and impact, structural stiffening); and methods to account for construction, installation, design, and quality control errors. The fundamental problem is to develop procedures for generating fragility distribution that are both comprehensive and defensible in the sense that such procedures treat all the potential failure modes and associated uncertainties and can be related to existing literature, fundamental experiments, and available test data.

2.4.3.3 Risk-Based Probabilistic Design Procedures for Reactor Components

One of the major limitations with the current (ASME code based) design procedures for LWR components is a lack of flexibility in addressing the relationship between high probability/low consequence and low probability/high consequence failure modes. Approaches that touch on this subject include classification of components, classification of loads, and load combination rules. A direct probabilistic formulation of the design equation would allow quantitative assessments of component failure and the associated risk for various design options and would also allow a design procedure which could meet preassigned risk criteria while minimizing cost factors. Such a design procedure would not penalize fail-safe failure modes in the same manner as fail-unsafe modes by virtue of the risk differentials and so might be expected to optimize safety of a component without unduly increasing the costs of producing it. In addition, such a procedure would directly address uncertainty in component fragility which would be useful to probabilistic risk studies.

2.4.3.4 Fluid-Structure Interaction Computer Code

A computer code could be developed to treat two-phase fluid blowdown, dynamic inelastic structural response, and the coupling between the fluid and structure. This code could then be used by the NRC to audit selected piping system analyses and to provide the applicant with verified benchmark solutions to check his analyses against. The structural side of the code would most likely be an existing finite element code with dynamic inelastic capability. The fluid side of the code could be adapted from the STAC code developed by ANL. This code uses a hybrid numerical technique to treat transient two-phase flows in piping systems and is particularly oriented toward handling the large gradient in-phase distribution near a blowdown boundary where flashing occurs.

2.4.3.5 Simplified Bounding Procedures

Techniques could be developed for the treatment of dynamic piping response to provide solutions to bounding problems in closed form or to problems requiring very little numerical computation. These techniques could be used by the NRC reviewer or the applicant to check the reasonableness of solutions obtained using more complex and expensive methods. The following are aspects of simplified modeling that would be investigated: the definition of simplified dynamic loads that reproduce the important features of the complex jet thrust loads, the development of a simple model for cross-sectional crushing, the selection of appropriate constants in simple material models to reproduce or bound the behavior of more physically realistic material models, and the development of simple solution methods compatible with these load and material models.

2.4.3.6 Application of and Extension of Computer Codes to LWR Problems

The computer codes developed in the SSA Section for LMFBR applications may have many unique features which make them valuable tools for LWR application. Thus, the core seismic analysis code SCRAP could readily be adapted to evaluate the seismic response of fuel-storage pools or LWR cores at high excitations. The series of fluid transient codes can also be extended to LWR problems and have, in one case, been used very successfully to analyze the blowdown transient in the pressure-relief piping of a PWR pressurizer.

2.4.3.7 Development of Dynamic Testing Approaches for Nuclear Power Plant Structures

A currently ongoing effort in the SSA Section is the evaluation of dynamic testing methods for the safety assessment of nuclear power plant structures. This work will culminate in a number of recommendations on the preoperational dynamic testing of nuclear power plants and the identification of existing and needed data bases for the verification of analytical and computational methods. A follow-on effort could be the development of specific testing approaches to generate needed data bases for analysis verification and the design of experiments to be used directly on as-built nuclear power plant structures as part of the licensing procedure.

2.5 Liaison with Other Foreign or Domestic Nuclear Safety Research Activities

The ANL will supply overall logistical support to the Seventh Structural Mechanics in Reactor Technology International Conference to be held in Chicago in August 1983. In the area of LWR safety, Argonne has only minor interactions with foreign programs (for example, NRC sponsored work on testing of nuclear power plant structures which involves use of data from the HDR in Germany). In the area of breeder reactor safety, physics, and components, the contacts are extensive. These contacts involve active participation in most of the US bilateral exchange agreements, transfer and training of foreign personnel, etc. Argonne's closest ties are with the UK and with Japan.

The ANL maintains a large educational program through the sponsorship of IAEA for the training of developing countries personnel in nuclear power plant licensing, construction and operational safety. There is usually in residence at ANL for 6 to 12 week periods 50 to 100 foreign engineers and scientists with prime interest in various aspects of light water nuclear reactor safety. The courses presented use well-recognized international and national experts as faculty, hence, provide a unique educational opportunity. If at all possible, these courses should be opened to U.S. student participation.

3. BROOKHAVEN NATIONAL LABORATORY

3.1 Organization

In the technical areas of interest covered in this report, the Brookhaven National Laboratory (BNL) nuclear safety research efforts have been concentrated in the Department of Nuclear Energy with H.J. Kouts, Manager.(Figure 5) More specifically, under the supervision of Dr. Morris Reich, the Structural Analysis Division of the Department of Nuclear Energy performs most of the work in areas of interest related to this study.(Figure 6) Brookhaven resources have focused primarily on current specific regulatory and licensing issues and have worked closely with the Division of Nuclear Reactor Regulation in providing detailed technical support. The Structural Analysis Division consists of 28 staff members and is subdivided into four groups: the Dynamic Response Evaluation Group, the Qualification Analysis Group, the Structural Probabilistic and Reliability Group, and the Soil-Structure Interaction Evaluation Group.

One unique feature of the Structural Analysis Group is that it makes maximum use of part-time consultants, primarily from university faculties in the New York City area, in performing its research tasks. As a result of this practice, an added dimension is brought to their technical capabilities, and research activities are performed in a highly cost-effective manner.

3.2 Topic Areas of Interest

Recently the Structural Analysis Division has performed research for the NRC in the following areas related to this study:

(1) Probabilistic Safety Evaluation of Category I Structures

- (a) Develop probability-based reliability analysis method for nuclear structures under static and dynamic loads.
- (b) Carry out reliability analysis for real reinforced concrete containments.
- (c) Generate fragility curves for probabilistic risk assessment for N.P. structures.
- (d) Develop load-combination methodology.

(2) Soil-Structure Interaction Methodology

- (a) Develop independent computer program for confirmatory analysis of Seismic I structures embedded in various soils.
- (b) Develop generic floor response spectra for various seismic zones.

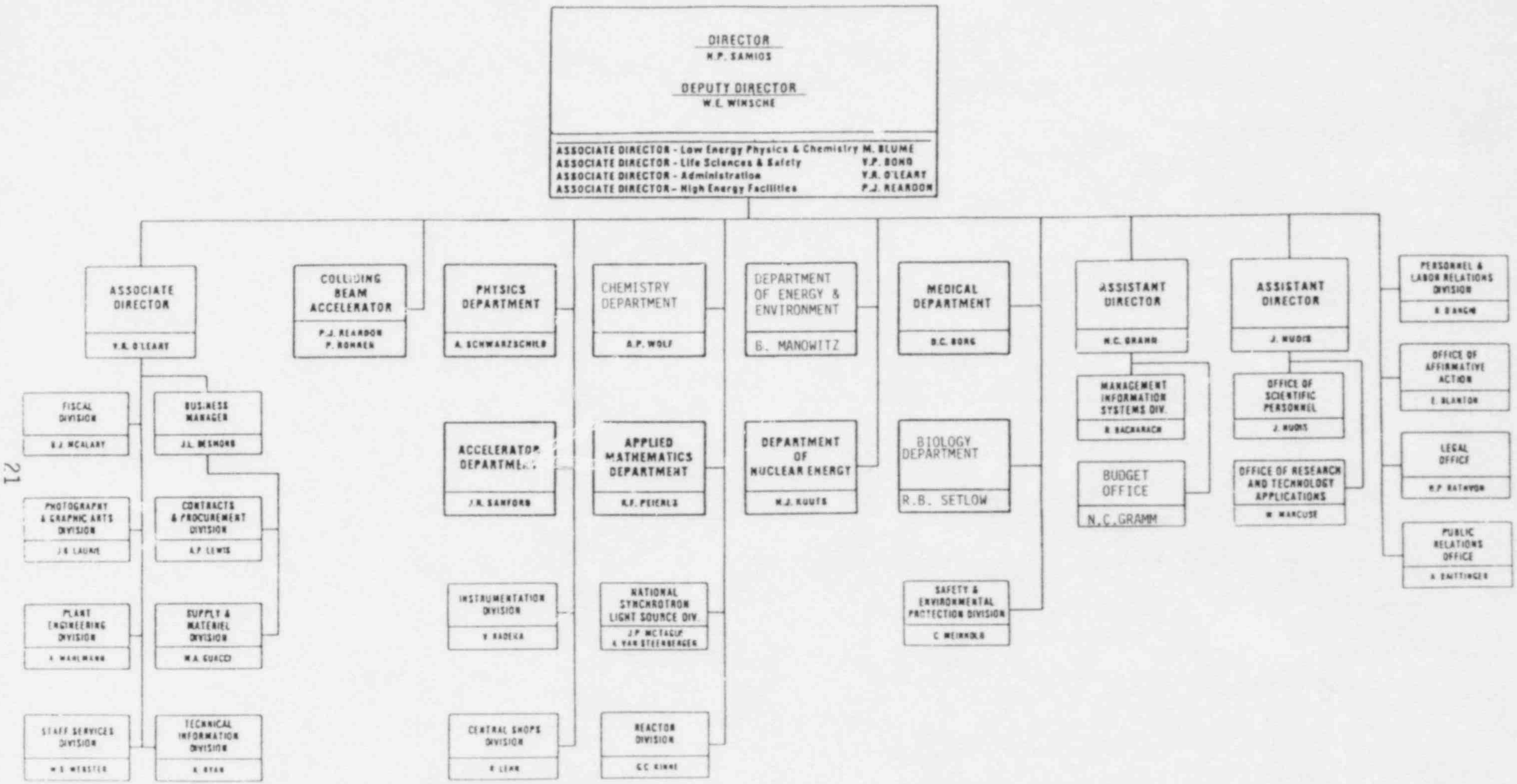

Figure 5 Brookhaven National Laboratory Organizational Chart

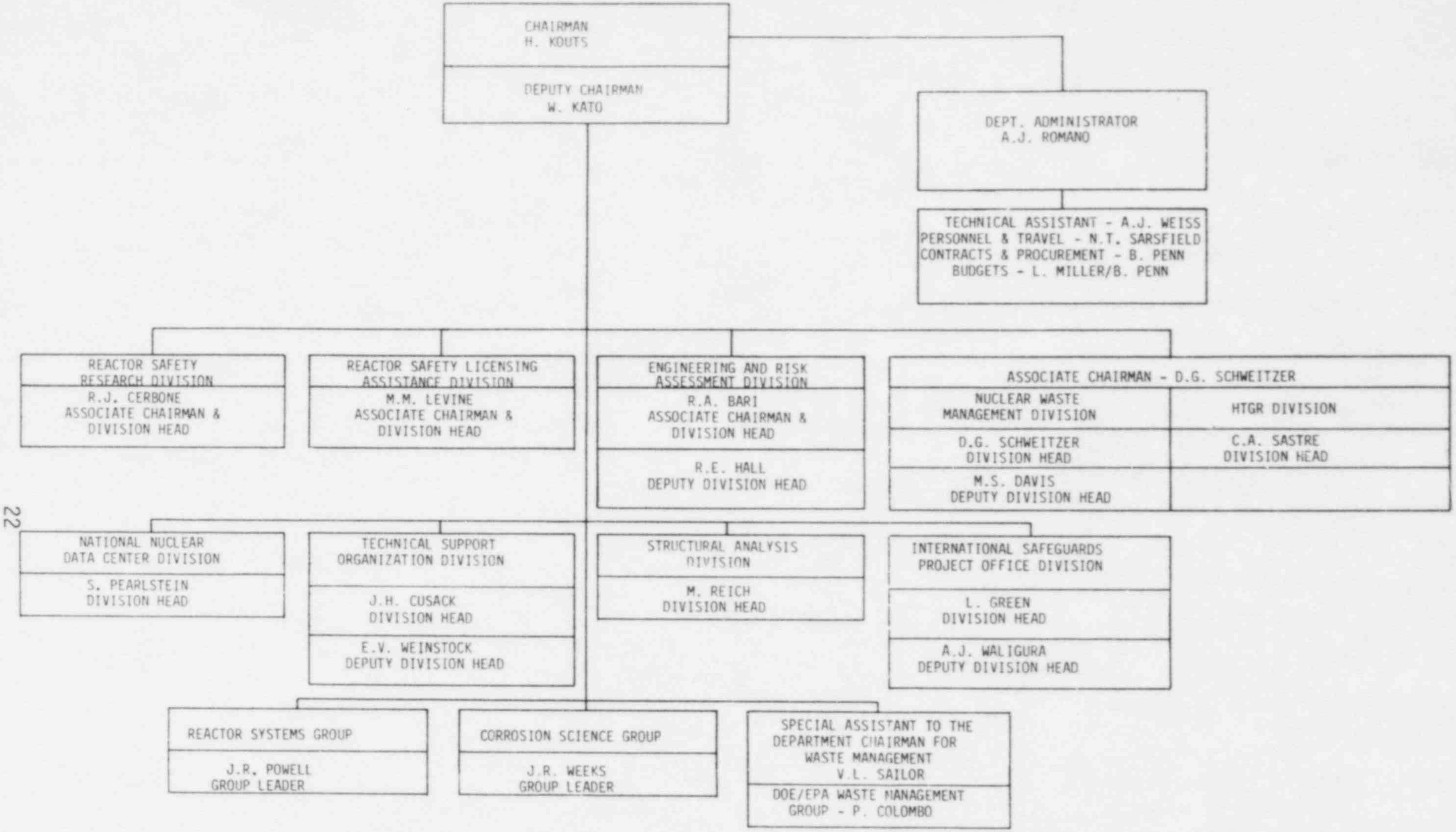

Figure 6 Department of Nuclear Energy Organizational Chart

(c) Check applicability of SSI analysis methods.

(d) Develop independent floor response spectra for Diablo Canyon Plant.

(3) Structural Engineering Benchmarks

(a) Independently develop computer code for inelastic structural analysis.

(b) Evaluate the ultimate capacity of R.C. containment structures.

(4) Dynamic Qualification of Safety-Related Mechanical and Electrical Equipment

(a) Carry out SQRT audits at plant sites.

(b) Issue qualification report on each item inspected during field visit.

(c) Develop NUREG for seismic and dynamic qualification of new plants.

(d) Develop NUREG for seismic and dynamic qualification for operating plants.

(e) Develop fragility curves for mechanical and electric equipment.

(5) Benchmarks for Mechanical Piping Problems

(a) Carry out independent confirmatory evaluation of ASME Class I, II and III piping systems.

(b) Independently develop computer program for analysis of piping system.

(c) Develop physical piping benchmark (testing).

In addition, the Division has also performed work in the areas of fracture mechanics, fatigue, creep, creep-fatigue-interaction, seismic-site spectra development, nonlinear response evaluations and nonlinear methods developments.

3.3 Facilities

3.3.1 Experimental

BNL's facilities for dynamic and environmental testing of structures and components are listed as follows:

(1) One 7,500 lb-force electro-magnetic shaker, variable or shaped input, frequency range 3-2,000 Hz,

(2) Two 2,500 lb-force electro-magnetic shakers, variable or shaped input frequency range 3-2,000 Hz,

(3) Numerous hydraulic actuators, whose input can be programmed for all types of acceleration, velocity and displacement characteristics. (The largest of these has a capacity of 250,000 lb-force and a stroke of 10 inches. A maximum stroke length of 36 inches is available on an actuator with a 75,000 lb-force capacity),

(4) One 14 x 14 independent phase incoherent, random phase biaxial simulator 50,000 lb-force,

(5) A long stroke phase coherent shaker (60 in/sed) 10,000 lb-force capacity,

(6) Environmental testing chambers (large capacity) for temperature-humidity-aging testing, and

(7) Valve LOCA facility with controlled shock impulses.

3.4 Areas of Future Interest Expressed by Laboratory Researchers

It is expected that probability safety evaluations will be a field of growing interest. In particular, the development of load combination methodology is of increasing interest.

3.5 Liaison with Other Foreign and Domestic Nuclear Safety Research Activities

The BNL Structural Analysis Division has numerous informal contacts with staff members of the U.S. National Laboratories, code committees, and selected Japanese and German researchers. Topics of mutual scientific interests are discussed at meetings with individuals of these organizations.

4. ELECTRIC POWER RESEARCH INSTITUTE

4.1 Organization

The Electric Power Research Institute (EPRI) nuclear safety research efforts in the technical areas of interest covered in this report have been concentrated within two departments: the Safety and Analysis Department and the Systems and Materials Department of the Nuclear Power Division. The Safety and Analysis Department, W. Loewenstein, Director, manages the Structural Integrity Subprogram with H.T. Tang as Program Manager. The Systems and Materials Department, K. Stahkopf, Director, manages the Structural Mechanics Safe Program with T.U. Marston as Project Manager.

EPRI is a research administrative organization funded by user fees on electric power usage contributed by member electric utilities. Most of the research funded is performed by consultants under contract to EPRI. Technical monitoring of each research project is performed by EPRI personnel. Current funding of nuclear power research both developmental as well as safety related is approximately $\$90 \times 10^6$/yr.

4.2 Topic Areas of Interest

4.2.1 Structural Integrity Subprogram

The Structural Integrity Subprogram has concentrated its research efforts in the following areas related to this study:

(1) Analysis/Design Methods-Algorithm Development

(2) Seismic/Vibration Response

(3) Fluid/Structure Response

(4) Impact/Impulse Response

(5) Structure/Component Performance

Details of these subprogram efforts are summarized herein.

4.2.1.1 Analysis and Design Methods

(1) Transient Continuum Mechanics Code Development

The objective of this project is to develop a general, portable, modular, and machine-independent, explicit, finite-difference computer code to address transient reactor design situations such as water hammer, soil-structure interaction, missile impact, piping flow, fluid-structure interaction and other transient accident or upset conditions.

Since the start of the project in late 1974, a series of codes have been developed. Table 1 summarizes the current status. STEALTH is the basic general purpose transient thermal-mechanical code and the rest are spin offs of STEALTH developed for special classes of problems. The basic general purpose STEALTH has undergone three releases (versions 1-1A, 2-2A, and 3-3A) with more than 100 copies distributed. The most advanced version (4-1A) which includes all 1D, 2D, and 3D capabilities has been qualified and documented and was formally released in the fourth quarter of 1982. Special purpose versions will be released in sequence following the general purpose release.

Currently, most of the STEALTH development has been completed. During 1982 and in 1983 only limited works relating to qualification and documentation of three-dimensional slide-line capability (for soil-structure interaction and dynamic impact analysis) will be performed to wrap up the development effort. The major portion of the proposed work in 1982 and 1983 will concentrate on full-scale applications in the area of soil-structure interaction, fluid-structure interaction, piping flow, water hammer, and other transient events.

(2) Nonlinear Structural Analysis Code

The objective of this project is to develop a nonlinear structural analysis code with special consideration of (a) economics and user-friendliness, (b) solution convergence and accuracy, and (c) versatile and convenient modeling. Specifically, the code is developed to improve structural analysis and component design, quantify existing design conservatism, evaluate structure and component safety margin, and correlate as well as interpret nonlinear dynamic experiments for response parameters identification.

The first phase of the report was to combine the pipe whip and seismic piping response development into one computer code (ABAQUS-ND)*. The overall development (ABAQUS-EPGEN)** will place emphasis on the following:

(a) Incorporating state-of-the-art research results in areas such as large deformation, strain rate dependent plasticity, intermittent impact, solution algorithim, fracture, concrete, etc.,

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* ABABUS-ND is for nonlinear dynamic analysis and is currently available from EPRI's Sofeware Center.

** ABAQUS-EPGEN is for general purpose analysis and is currently being qualified and documented.

Table 1 The STEALTH Family Codes and their Current Status

Name	Application	Expected Formal Release Date
STEALTH 4-1A* (1-, 2-, and 3-D)	General Purpose	11/82
STEALTH-PIPING	Piping Analysis	3/83
STEALTH-SEISMIC	Soil-Structure Interaction	6/83
STEALTH-IMPLICIT	Hydrodynamic Analysis	6/83
STEALTH-WHAMSE	Fluid-Structure Interaction	8/83

* Current version which is available from EPRI's Software Center is STEALTH 3-2A (1D and 2D only)

(b) Providing comprehensive pre-and post-processing package in terms of mesh generation, plot, and data check,

(c) Developing criteria for assuring solution convergence and reliability,

(d) Performing thorough qualification analysis, and

(e) Preparing comprehensive documentation.

Continued effort in the next few years will cover the following areas:

(a) Updating and code maintenance based on feedback obtained from code users,

(b) Further capability qualification and development for analysis of containment over-pressurization (concrete, rebar), piping local nonlinearity (snubber), heat transfer (thermal and residual stress), fracture (thermal shock), and other related areas,

(c) Further refinement of code architecture, input and output processing, solution algorithim, etc., to make the code more efficient and cost effective, and

(d) Extensive application studies.

4.2.1.2 Seismic/Vibratory Response

(1) Soil-Structure Interaction

The overall aim of this project is to investigate nonlinear soil-structure interaction behavior and to identify conservatisms embedded in current linear approaches under a strong earthquake condition.

The specific objectives of this project are to develop an experimental data base to quantify strong motion nonlinear soil-structure interaction and to create a verified soil-constitutive model that is compatible with the nonlinear response. Additional objectives of the project are to provide a qualified nonlinear analysis methodology and computer program through correlation of the strong motion experimental data and to explore seismic isolation concepts for seismic response mitigation feasibility.

In view of the need in the area of soil-structure interaction research, EPRI has adopted since 1976 an integrated approach to study this subject. This approach covers development of more refined soil constitutive models, in situ soil testing, SSI parameter identification, controlled experimental data base using explosives, nonlinear analysis methodology, and response mitigation concept and design. The objective of the research focuses on the establishment of a validated nonlinear SSI methodology applicable for plant design and licensing purposes. Continued efforts in the next few years will include further simulated testing, large-scale nonlinear SSI analysis, and technical exchange with foreign countries and similar matters.

(2) Piping System Response

The objective of this project is to develop a piping system response data base to study piping system damping, support-to-support interaction, pipe-structure interaction, and support-to-support correlation. Additionally, the experimental data collected will be used to qualify the analytical methodology and computer program developed under the structural integrity subprogram.

The primary aim of this project is to quantify conservative assumptions (damping, analysis procedure, multiple support excitation, etc.) embedded in current piping system design and analysis and thereby recommended improvements based on research results and findings.

The testing of an 8" feedwater line inside the Indian Point 1 containment constitutes the major part of the seismic piping system response project in the past several years. The first phase of this project will be completed with the formal final report to be published in 1983. Following this effort will be extensive post-test correlation to interpret in depth the physical behavior of the various configurations tested (lightly supported, mechanical snubber, hydraulic snubber, rigid strut, etc.) and to qualify the methodology and computer program that takes into account the nonlinear behaviors of support and pipe-support interaction.

(3) Seismic Isolation

The objective of this project is to investigate, both experimentally and analytically, the range of applicability of various seismic isolation concepts for nuclear power plant structural and equipment design. The outcome of the study should provide to the utility industry a reliable data base and innovative designs for more effective seismic resistant considerations by way of isolating undesired excitation characteristics.

At present, there is no on-going project for seismic isolation studies in EPRI's Structural Integrity Subprogram. The past projects in this area have been concentrated on experimental tests which utilize base isolation systems on scaled structures and equipment.

Limited data bases have been generated for the investigation of feasibility and applicability of a base-isolation concept.

4.2.1.3 Fluid/Structure Response

(1) PWR Hydrodynamic Loads

The objective of this project is to investigate, as well as understand, hydrodynamic loads associated with a postulated LOCA in PWR's and to develop an analytical methodology to describe these hydrodynamic loads.

The primary aim is to provide utilities with a qualified computer program for their licensing applications in the area of PWR hydrodynamic load issue.

The 1D, 2D, and 3D analytical methodologies have been developed by coupling STEALTH for the fluid description and WHAMSE for the structural description. These methodologies were developed in a stepwise manner and validated against experiments run semi-scale, at Battelle-Frankfort (RS16B) and at the large scale HDR facility. Currently an extension of the HDR model to a more prototypical geometry (a beam core) has been completed, and calculations are to be performed in 1982 and documented for project completion. User documentation of the coupled 2D and 3D codes is also to be completed.

EPRI has been participating in the HDR project. In exchange for EPRI's analytical calculations, EPRI receives the detailed experimental data traces from HDR. This allows validation of the coupled 3D fluid-structure interaction methodology. EPRI submits pre-test calculations of selected HDR experiments to the HDR project in a pre-arranged format (to allow quick post-test comparisons). This effort also allows access by EPRI to other structural and non-structural data taken during the multi-faceted HDR program.

There are no planned efforts beyond 1983. Current efforts (1982) are to wrap up the development with a qualified and well-documented package deliverable to EPRI's member utilities.

(2) Pipe Rupture and Depressurization

The objective of this project is to provide the nuclear power industry with test data on break-opening time by performing pipe rupture and depressurization experiments.

The aim of the project is to justify more realistic initial conditions in piping rupture-induced transient calculations and also to serve as benchmarks for validating existing calculation techniques.

This project began in 1982 with a study to define the pipe sizes, geometries, and service conditions most commonly encountered in PWR and BWR designs. This information, coupled with surveys of existing and future experimental programs in England, France and Japan, will be used to plan the proposed tests which will consist of extensively instrumented circumferential and axial bursts of pre-flawed pipes of various geometries over a range of steam/water conditions, external loadings, and boundary conditions.

4.2.1.4 Impact/Impulse Response

(1) Pipe Whip

The objective of this project is to develop analytical methodology to analyze pipe whip response and to conduct experiments to quantify pipe whip loads and their associated piping and structural responses. The aim of the project is to quantify conservatisms inherent in the current practice and thereby improve current design and analysis with minimum installation of pipe whip restraints and protective structures.

In the initial phase of the project, a review on the current state-of-the-design for pipe whip was conducted by TVA. In this review, all aspects related to pipe whip such as postulated break opening time, jet characteristics, pipe whip characteristics, pipe whip on concrete structures, pipe whip on steel structures, pipe whip restraints, etc., were reveiwed in depth to identify design/analysis adequacy, inadequacy, conservatism, as well as research that can lead to improvements. A state-of-the-art review of analytical methods and tools, in particular, finite element program, for pipe whip analysis was also conducted. Based on reviewing findings, a computer code with more improved capability in strain-rate dependent plasticity, large deformation, intermittent contact and impact, solution convergence and accuracy, etc., has been developed.

Current ongoing efforts in this project involve experimental testing of simulating an instantaneous high-energy pipe line (3" diameter) break and study of its impact on rigid, steel supports as well as deformable concrete slabs. This research is being performed by Framatome/CEA and is scheduled for completion by the end of 1983.

Following this effort will be analytical correlation and qualification of computer codes using the data collected. Further testing will be determined by evaluating results obtained elsewhere as well as by licensing requirements.

(2) Jet Impingement

The objective of this project is to conduct experimental as well as analytical studies to improve current understanding of pipe rupture jet behavior and thereby provide a more rational approach to design problems related to jet behavior such as designs against jet impingement forces, jet reaction forces and pipe whip.

The Structural Integrity Subprogram's effort with this project has been the co-sponsorship of the Marviken Full Scale Jet Impingement Tests, which include 6 free jet tests and 6 impingement tests with test nozzle diameter ranging from 7.87" (200 cm) to 19.69" (500 cm) and target distance from 23.62" (0.6m) to 78.74" (2.0m). Parallel to these test efforts, the analytical effort involves the use of K-FIX and SALE Codes to perform correlation studies of both Marviken as well as the German HDR jet tests. The data collected and the analytical methods developed could be used for more realistic evaluation of the conservatism embedded in the current licensing requirement on jet impingement load and design methods.

Most of the planned research and development was completed by the end of 1981. Beginning in 1982, efforts to develop an improved simplified jet load predictive model and method for more efficient licensing design evaluation have been performed. This proposed effort will take into account the in-depth understanding of jet behavior achieved through detailed analysis using sophisticated computer codes (K-FIX, SALE, etc.).

(3) Turbine Missile Impact

The objective of this project is to provide test data and supporting analyses that may be used to identify and quantify the conservations in current estimates of turbine missile risk in nuclear power plants. The project focuses on postulated failures of the shrunk-on disks in the low-pressure sections of 1800 rpm steam turbines, particularly in plants with a non-peninsula turbine arrangement. The three project areas in the program are impact of casing structures, impact of reinforced concrete structures, and probabilistic analysis of turbine missile risk.

Turbine missile impact research is scheduled to be completed in 1982 with the publication of eight final reports. Program results have also been documented in technical papers and in two narrated films available from EPRI upon request.

4.2.1.5 Structural/Component Performance

(1) Containment Load Carrying Capacity

The objectives of this project are: (a) to provide and qualify an analysis methodology for determining the overpressurization capacity of both reinforced and pre-stressed concrete containments, (b) to provide test data for existing reinforced and pre-stressed concrete design so that improved concrete constitutive models can be obtained for more realistic determination and prediction of the containment overpressurization capacity, and (c) to perform analysis to determine ultimate load carrying capacity for several prototypical reinforced and pre-stressed concrete containments.

The Structural Integrity Subprogram has plans for integrated analytical and experimental research for improving the state-of-the-art for making realistic estimates of containment capacity under loads exceeding design loads. The work began in 1982 focusing on response to overpressurization and in subsequent years will be expanded to include response to seismic loads. Both reinforced and prestressed concrete will be examined.

(2) Piping and Supports Dynamic Capacity

The objective of this project is to determine the ultimate dynamic load carrying capacity of a nuclear piping system including both piping and supports.

The aim of this project is to provide experimental data for quantified assessment of piping system safety margin under dynamic excitations, in particular, under external postulated strong earthquake events.

A laboratory-based testing program is ongoing. The objective of this project is to experimentally assess the dynamic design margins of representative nuclear plant piping systems designed to ASME code standards. The scope of the project includes dynamic testing of various pressurized multi-bend, multi-supported piping systems to stress levels many times beyond the ASME code allowed limits. With piping systems eventually reaching failure, the support (snubber) failure capacity as well as the associated piping behavior change and reserved load capacity with one or more supports' failure are to be determined.

4.2.2 Structural Mechanics Subprogram

The research activities which receive emphasis in the Systems and Materials Department are those related to safety, reliability and availability of light water reactors through improved understanding of structural materials. The scope of the Structural Mechanics Subprogram is to formulate and to conduct research projects which help insure the safe and reliable operation of nuclear power plant metallic components or structures such as pressure vessels, pipes, fittings, supports, heat exchangers, turbines, and containment structures and thereby maintain plant licensing and increase plant availability. Other EPRI programs focus on the overall performance of these same components including operability and functional capability while the Structural Mechanics Subprogram focuses on the material structural performance of these components. The scope of the program is indicated in Figure 7.

The program provides utilities with the tools to evaluate structures after in-service inspection. In addition, utilities can evaluate whether the result is a recordable indication or a real defect and whether the form or cause be cracks, degraded material or excessive loads.

The program is responsive to perceived utility needs. Major needs are identified as follows:

(1) Ability to assess safety significance of defects and degraded material properties,

(2) Ability to determine remaining useful life of a defective structure regardless of cause,

(3) Definition of remedial actions for nonacceptable components, and

(4) Transfer of developed technology to promote a rational balance between public safety and ratepayer power costs.

Safety research identified by components and their supports are summarized herein. Specific current or recently completed research projects are identified in Table 2 with specific research contracts listed in Table 3. For more detailed information concerning each contract see Reference 2.

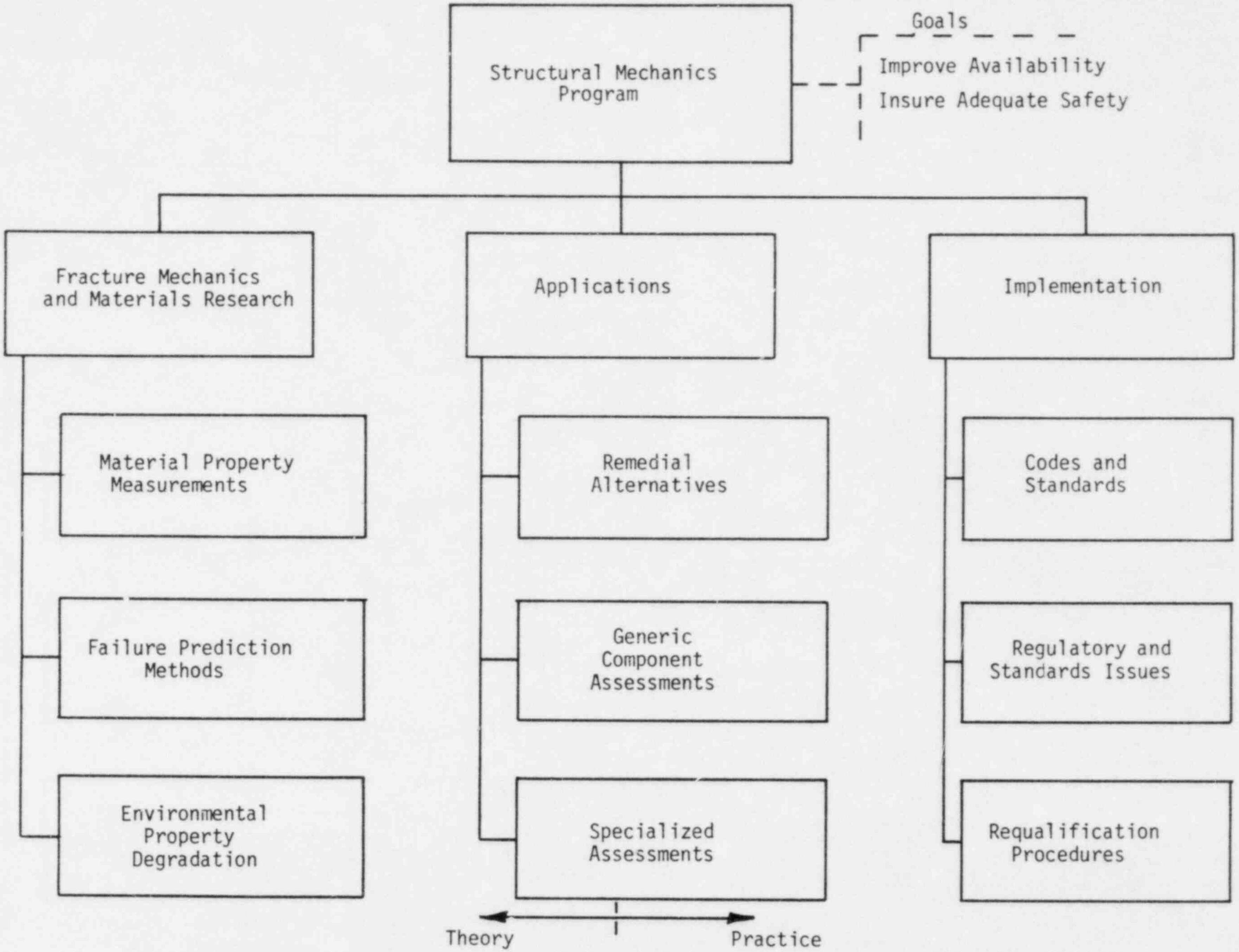

Figure 7 Scope of the 1981-1982 Structural Mechanics Program Organizational Chart

Table 2 EPRI Research Projects

Project Number	Project Name
RP603	Crack Growth in Residual Stress Fields
RP1021	Anneal and Radiation Embrittlement
RP1236	Repair of Nozzles
RP1237	Estimation Techniques for Dutile Fracture
RP1238	Fracture Toughness
RP1326	Irradiation Crack Arrest
RP1542	Structural Reliability Methodology
RP1543	Reliability of Piping and Fittings
RP1756	Component Requalification
RP1757	Code Support
RP2055	Support Structure Reliability
RP2056	TMI-2 Pressure Boundary Requalification
RP2180	Reactor Vessel Reliability
RP2227	Simplified Piping Handbook

Table 3 EPRI Research Contracts

Number	Topic
I. Fracture Mechanics and Materials Research	
RP1021-6,7	Kinetic Aspects of Neutron Embrittlement of LWR Steels
RP1326-1	Irradiated Crack Arrest Toughness
RP1542-1,2,3,4	Structural Reliability Methodology
RP1543-1	Fracture Toughness of Nuclear Pump and Valve Materials
RP2055-2	Data Base for Support Structure Reliability
RP2055-3	Reconstituted Charpy Specimens
II. Applications	
RP602-3	Residual Stress Distribution Near Growing Cracks
RP1237-1	Handbook for Elastic-Plastic Fracture
RP1237-2	Failure Analysis Diagrams for Ductile Fracture
RP1237-3	Nonlinear Line Spring for Ductile Fracture
RP1238-2	Experiments to Determine R-Curves and Instability Criteria
RP1326-2	Computer Analysis of Dynamic Crack Arrest
T118-1	Stress Corrosion Cracking Model for Large Pipes
T118-2	Safety Margins of Cracked Stainless Steel Pipe (Contract Extension: Fluid Flow in IGSCC Cracks)
T118-6	Probabilistic Modeling of IGSCC in Piping
T118-15	Relaxation of Residual Stresses in Piping
RP1543-7	Flaw Evaluation Criteria for Nuclear Power Plant Piping
III. Implementation	
RP1021-1,2	Thermal Annealing of Embrittled Reactor Vessel
RP1021-3	Steady-State Radiation Embrittlement of Reactor Vessels
RP1021-4	Fracture Toughness Correlations and Reference Curves
RP1236-1	Repair Welding of Heavy Section Steel Nozzles
RP1543-2	UT Detectability of Nozzle Surface Cracks
RP1543-3,4,5,6	Piping and Fittings Reliability Planning Group
RP1543-8	Environmental Fatigue Stress Rules for Carbon Steel Piping
RP1756-1,2	Component Requalification
RP1757-1,2	Code Support
RP2055-1	Support Structure Reliability
RP1056-1	Thermal Analysis of TMI-2 Pressure Boundary
RP2056-2	TMI-2 Primary System Boundary Characterization
RP2180	Reactor Vessel Reliability
RP2227	Simplified Piping Design Handbook

4.2.2.1 Reactor Vessels

The concern over reactor vessel integrity is heightened as a result of the current pressurized thermal shock issue. A significant portion of the EPRI program is focused upon the reactor vessel, and progress in 1981 was substantial. A procedure for testing the fracture toughness of specimens in reactor surveillance programs received NRC acceptance. The same work indicates that the current fracture toughness test techniques underestimate actual material toughness and provides the means for improving the estimates. The conservatism of the USNRC Regulatory Guide 1.99.1 in predicting transition temperature shift in reactor surveillance program materials is indicated. The generic feasibility of thermal annealing an embrittled reactor vessel is demonstrated. With regard to BWR vessels, the detection of cracks in feedwater piping by ultrasonic means is analytically assessed and found to be a strong function of prior service history and crack depth. An alternative repair welding procedure without stress relief using a multi-layer, gas tungsten arc process (instead of the ASME Code-approved half bead) appears to provide a fully satisfactory repair that is easily automated and produces fully refined heat-affected zones.

4.2.2.2 Piping

The principal emphasis in 1981 was placed on the development of flaw evaluation procedures for stainless steel nuclear piping. The work is in large part funded by the BWR Owner's Group. The acceptance criteria for flaw sizes in stainless steel are being reviewed by an ASME Section XI Committee for codification. The leak-before-break conditions are established for several generic BWR lines, such as recirculation, core spray, isolation condenser return and supply lines. Current research has led to the conclusion that a guillotine break in these lines is essentially impossible. A significant number of pipe experiments support this statement on instability. This fact greatly reduces the safety ramifications of pipe cracks. A new area initiated in 1981 evaluates the effects of designing nuclear piping systems for the low probability, high load events. The initial study indicates that a preoccupation with the high load events may actually reduce the reliability of piping systems for normal operation. Future work will focus on developing a design philosophy balancing safety and reliability.

4.2.2.3 Pumps and Valves

Thermal aging (or embrittlement) of cast austenitic steels is now recognized as a possible source of integrity concerns for reactor coolant pumps and valves. The research conducted in 1981 indicates that although a significant loss in impact resistance is possible, the remaining fracture toughness is in excess of the requirements imposed upon reactor pressure vessel materials.

4.2.2.4 Component Supports

The integrity of supports for heavy components is the subject of the NRC unresolved safety issue A-12. EPRI has funded a substantial amount of work to support the industry position coordinated by the Atomic Industrial Form.

4.3 Facilities

4.3.1 Experimental

EPRI does not maintain experimental facilities in the areas of interest associated with this report.

4.3.2 Computer Hardware and Software

Current computer software activities are summarized in Section 4.2.

4.4 Areas of Future Interest Expressed by Laboratory Researchers

It is anticipated that EPRI will continue to support research in topical areas of interest identified in Section 4.2. New areas will be added as a need is identified to support overall goals of safety, reliability and economy of nuclear power.

4.5 Liaison with Other Foreign or Domestic Nuclear Safety Research Activities

EPRI maintains liaison and supports foreign conducted research in a number of areas. As discussed in Section 4.2, EPRI is, or has recently, supported research in the following facilities:

(1) HDR Program in the FRG

(2) Jet impingement tests at Marviken, Sweden

(3) Pipe break tests conducted by Framatome/CEA in France

(4) Hydrogen detonation tests at the Whiteshell Facility in Canada

5. IDAHO NATIONAL ENGINEERING LABORATORY - EG&G IDAHO, INC.

5.1 Organization

EG&G Idaho nuclear safety research efforts in the technical areas of interest covered in this report are concentrated in the Science and Engineering Group, Nuclear Technology Division, L.J. Ybarrondo, Associate General Manager, as shown in Figures 8 and 9. Within the Nuclear Technology Division, research of interest is being performed by the Water Reactor Research Department with Dr. J.A. Dearien, Manager. Within the Water Reactor Research Department, the Code Assessment and Application Division under Mr. B.F. Saffell, Manager, is most concerned with areas of interest identified in this report. The Water Reactor Research Department also contains the Water Reactor Research Test Facilities Division which nas prime responsibility for the LOFT program.

Currently, technical assistance being provided to the NRC by EG&G Idaho is approximately evenly split between the Office of Nuclear Regulatory Research (approximately $\$7.0 \times 10^6$) and the Office of Nuclear Reactor Regulation (approximately $\$6.0 \times 10^6$).

5.2 Topic Areas of Interest

Within the last few years, EG&G Idaho has provided technical support to the Department of Energy and the Nuclear Reactor Regulatory Commission in the following areas related to this study.

5.2.1 Department of Energy

(1) Nondestructive Examination Imaging Research

This project encompasses the development of data acquisition and processing techniques for improved ultrasonic imaging of reflectors in structural materials. The objective of this research is increased accuracy in location and resolution of fractures in structural components.

(2) Engineering Analysis of Elastic-Plastic Fracture

The objective of this research is to improve design and analytical techniques for predicting the integrity of flawed structural components. The research is primarily experimental, with analytical evaluation guiding the direction of experimental testing. Tests will be conducted on a material exhibiting a range of fracture toughness but essentially constant yield and ultimate tensile strength. As test temperature increases, the specimen configuration-fracture toughness relationship will comply initially with requirements for linear elastic-fracture mechanics and eventually extend beyond the range of a J-controlled field. Specimens include compact, 3-point bend, and surface-flawed with through-thickness-oriented

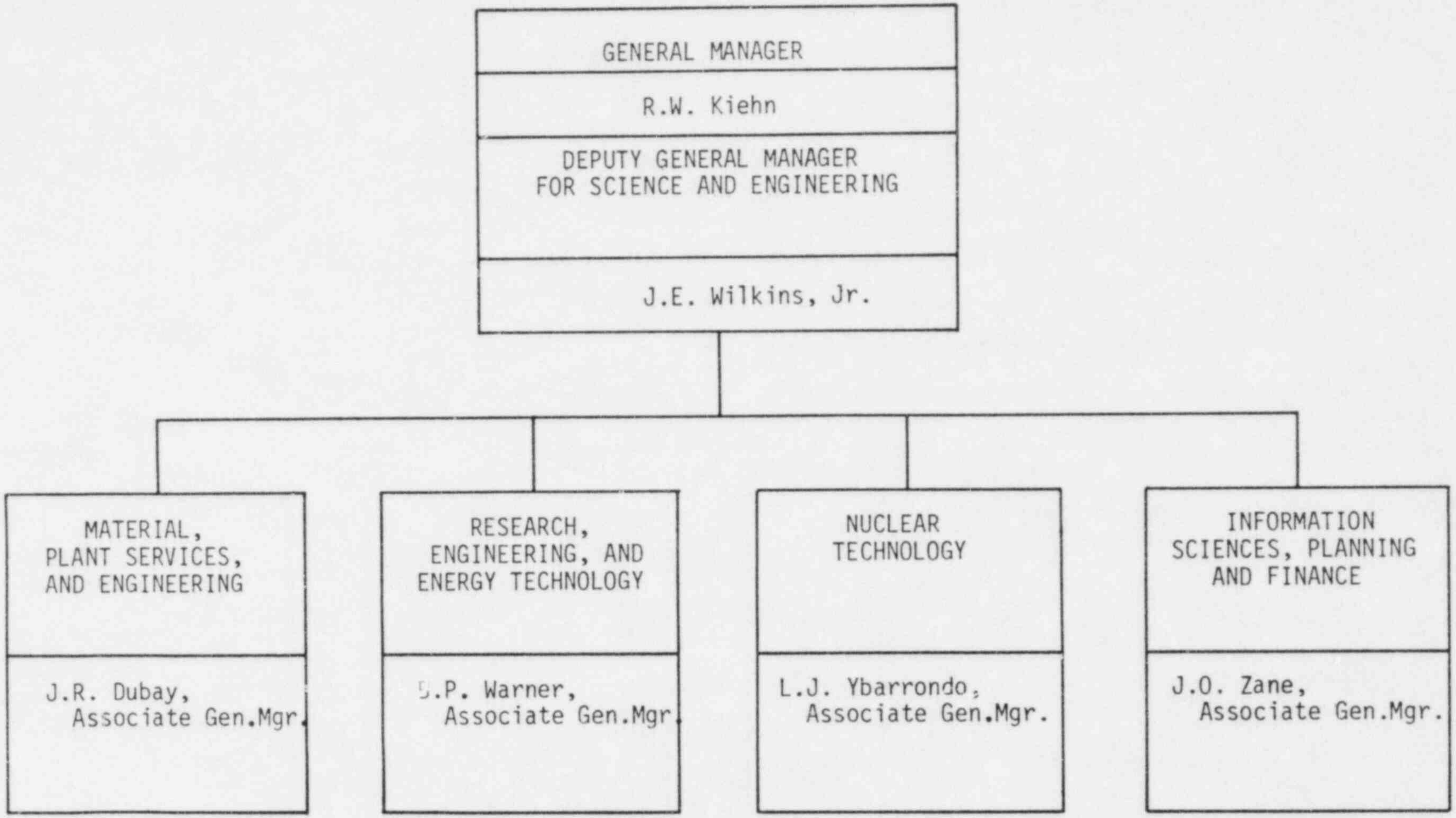

Figure 8 EG&G Idaho National Engineering Laboratory Organizational Chart

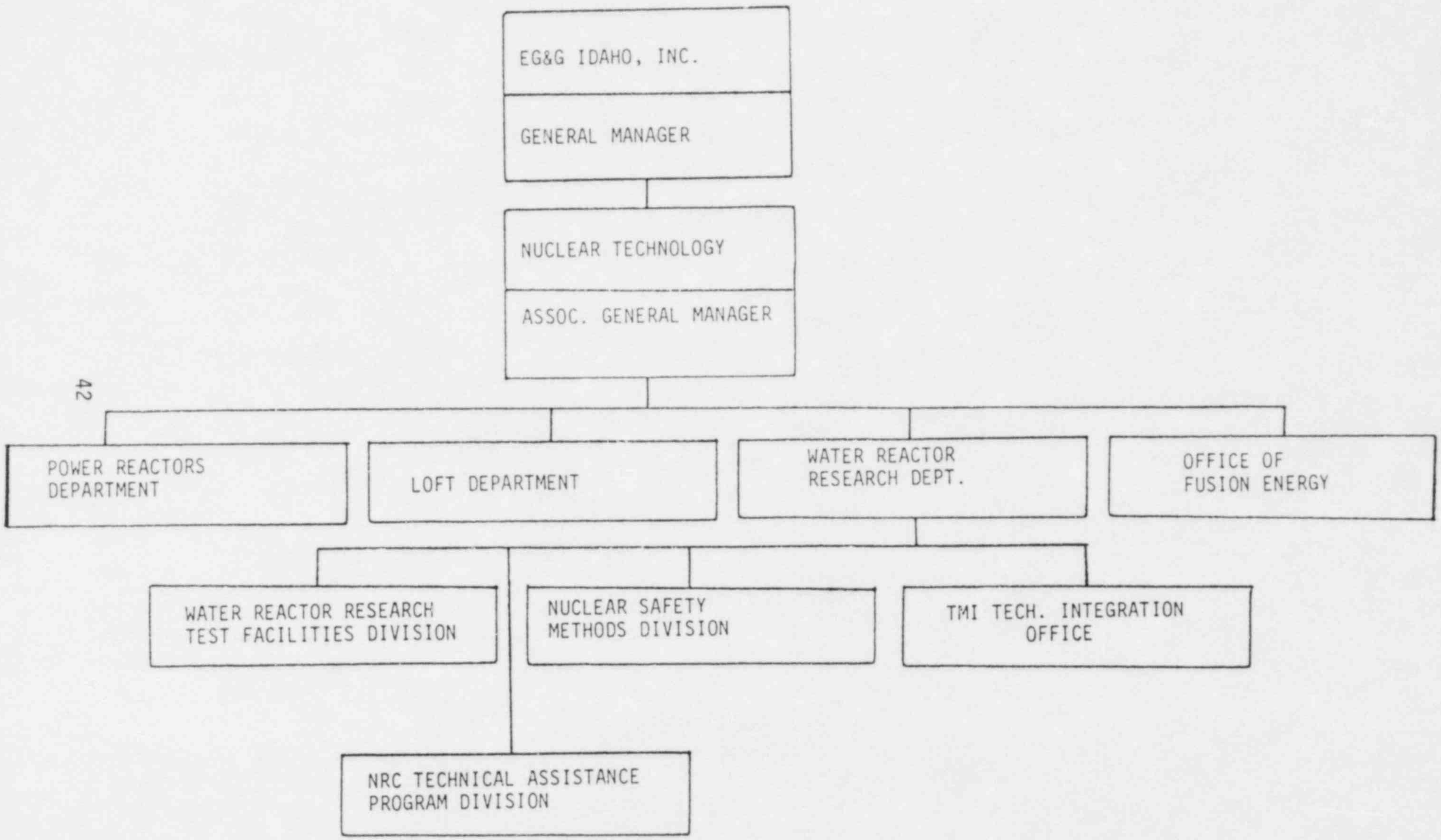

Figure 9 EG&G Idaho Nuclear Technology Chart of Organization

cracks. Material, specimen, and test data requirements have been defined and a ferritic steel has been procured. Both Charpy V-notch and drop-weight tests have been conducted and the data are consistent with programmatic requirements. Specimens have been designed and fabrication of both surface-flawed and E-399-81 fracture toughness specimens has begun.

5.2.2 Office of Nuclear Reactor Regulation

(1) Reactor Structures

(a) Buildings
(b) Piping
(c) Reactor internals
(d) Supports

(2) Materials

(a) Fracture toughness
(b) In-service inspection
(c) Non-destructive examination

(3) Probabilistic Risk Assessment of Big Rock Point

(4) Hydrology

(5) Equipment Qualification

(a) Environmental
(b) Seismic

5.2.3 Office of Nuclear Regulatory Research

(1) Computer Code Assessment and Applications - Thermal Hydraulic and Fuel Behavior

(2) Risk Analysis

(3) Equipment Qualification

(a) safety relief valve

(4) Structural Fracture and NDE Evaluations

(a) Heissdampfreaktor, HDR Piping Analysis (Federal Republic of Germany)

(b) HDR Damping Studies

(c) ASME Boiler & Pressure Vessel Codes Evaluation

(d) KUOSHENG - safety relief and blowdown prediction

(e) General pipe damping study

5.3 Facilities

5.3.1 Experimental

5.3.1.1 LOFT Facility

The objective of the Loss of Fluid-Test Facility (LOFT) facility, is to establish conditions in a nuclear reactor that are characteristic of accidents postulated for a large PWR so that methods can be developed and tested for analytical description, for accident recognition, and for manual and automatic plant stabilization and recovery.

The specific goals of this program are as follows:

(1) Acquiring data for the assessment and improvement of computer codes intended to predict the behavior of PWRs under a wide variety of accident conditions,

(2) Understanding the behavior of PWRs under accident conditions and the operator actions needed to stabilize and recover the plant,

(3) Interpreting and improving plant instrumentation needed to identify accident conditions and to assist the operator in recovering the plant, and

(4) Testing an advanced operator display system at a PWR under actual accident conditions.

The LOFT facility is the only nuclear-heated facility capable of carrying out tests of the response of the primary system of a PWR to loss-of-coolant accidents (LOCAs), to anticipated transients without scram (ATWSs), and to other off-normal and accident conditions. These tests provide the capability of ensuring that the codes used in full-scale plant analysis combine the effects predicted by various models in an accurate fashion and that no important effects are overlooked.

In addition, performance of these tests offers an opportunity to try out advanced instruments and data-reduction techniques, as well as an opportunity to understand how to enhance the capability of the operator to respond to accident conditions. The test program selected by the NRC is expected to be completed by early 1983. At that time, depending on current negotiations, the facility may be decommissioned or devoted to testing supported by an international consortium.

As the only operating scaled PWR, LOFT relates to the separate effects research programs as an integral systems test of individual phenomena and to the systems research programs as a large-scale nuclear-powered test. The early LOFT tests were first run on the Semiscale facility as LOFT counterpart tests to gain understanding of the response to be expected in LOFT. The LOFT-LOCA tests were simulated with individual fuel rods in the Power Burst Facility (PBF) reactor to give an indication of the response of the fuel in the LOFT core prior to running each test at LOFT. In general, LOFT provides an essential link from the small-scale system tests of Semiscale and the many large-and small-scale separate effects tests in NRC's research program to full-scale commercial plant behavior.

Through the direct participation by the reactor safety centers of ten other countries, supporting analytical and experimental results have supplemented the LOFT program. Results from the LOFT program, in turn, have been used to address safety issues in these and in other countries.

The LOFT program began nuclear-powered LOCA testing in December 1978. The first LOCA test used was pressurized fuel in the central core assembly and was conducted in fiscal year 1982. This test, designated L2-5, was a double-ended, cold-leg break with delayed emergency core coolant (ECC) injection. Final evaluation of the test results is expected to show that a full break with pressurized fuel and operating conditions consistent with the most conservative of Appendix K assumptions does not result in damage to the fuel cladding. Additional tests performed in FY 1982 included an ATWS, a series of four-operational transients, and a boron dilution transient from cold shutdown conditions. The ATWS and the boron dilution transient from cold shutdown were specifically requested by NRR for use in assessing vendor computer code capabilities.

NRC-sponsored testing in LOFT will be concluded in FY 1983. Following a second ATWS test, the test program will be concluded with a double-ended, cold-leg break with end-of-life fuel pressurization and excessively delayed ECC injection. The results of the last test will provide (1) a measure of conservatism in the licensing evaluation models and current ECC systems, (2) severe ballooning and bursting of the cladding in the central fuel assembly, and (3) a measure of the coolability of a ballooned and burst fuel bundle. The central fuel assembly will undergo extensive postirradiation examination (PIE) to obtain data on fuel rod cladding burst conditions and the fuel cladding damage distribution pattern in a 15 x 15 fuel rod assembly.

The PIE program for the damaged central assembly will provide data on stress levels in the vicinity of the clad burst, the distribution both longitudinally and radially of the ballooning and burst, the conditions present within the fuel rod at the time of burst, and the postburst cladding oxidation. The examination of the fuel is expected to involve staff from additional national labs, including Pacific Northwest Laboratory, Oak Ridge National Laboratory, and Battelle Columbus Laboratory. Analysis of the data should be concluded during the year and final reports on the results prepared.

The U.S. Department of Energy has established an international LOFT consortium to run a test program for an additional three year period. The consortium involves major participation by foreign countries, DOE, EPRI, and the NRC.

5.3.1.2 PWR Test Facility

The Water Reactor Research Test Facility (WRRTF) contains an integral, scaled PWR test facility and four test loops. With the five test systems, the WRRTF can simulate PWR behavior, calibrate instrumentation used in PWRs, study heat transfer phenomena, investigate the performance of instruments under transient conditions, and test components. Separate effects phenomena can be investigated under high pressure, high temperature, and single-phase or two-phase flow conditions.

The Semiscale test system began as a small, single-loop system that contained simulated reactor components. The program, initially an adjunct to the LOFT Program, has grown in complexity and usefulness. From 1974 through 1977, the Semiscale Mod-1 system was used to perform numerous separate and integral effects tests simulating PWR behavior during a LOCA. These tests included isothermal tests, blowdown heat transfer tests, reflood heat transfer tests, baseline and emergency core coolant (ECC) injection tests, alternate ECC experiments, and LOFT nuclear counterpart tests.

Alternate ECC injection concepts and steam generator tube rupture response are two important LOCA-related areas that were investigated during the last year of Mod-1 system operation. The effect on peak rod temperatures of the rupture of various numbers of steam generator tubes during an already occurring LOCA was also investigated. In August 1977 testing with the Mod-1 system was concluded.

The Semiscale facility was converted in early 1978 to a two-loop system (Mod-3). This conversion represents a major improvement because the facility is now more typical of large PWRs. The Mod-3 system included such improvements as a full-length (3.66-meter) non-nuclear core and a complete, active broken loop with pump, piping and steam generator all scaled to PWR counterparts.

After using the two-loop Mod-3 system for three years, the Semiscale facility was converted in the Summer and Fall of 1980 to the Mod-2A system. This system is basically an improved version of the Mod-3 system, designed: (a) to make Mod-2A more typical of a large pressurized water reactor, and (b) to improve the capability to determine and report thermal-hydraulic performance during plant transients.

As the need arose for additional testing of reactor components and measurement systems in support of major reactor simulation safety experiments, the Blowdown Loop, the Full Area Steady-State (FAST) Loop, the Two-Phase Flow Loop (TPFL), and the Steam-Air-Water (SAW) Loop were added.

The Blowdown Loop was installed in 1975 to provide separate effects test capability for the LOFT Program. Since then, the loop has been used to assess and to calibrate LOFT external fuel cladding thermocouples under transient conditions, to test the performance of LOFT flow instrumentation, to study basic blowdown heat transfer, to qualify the

Power Burst Facility (PBF) blowdown valves, and to test the performance of the Semiscale scaled high-speed pump.

The FAST Loop was added in 1977 to test and to qualify LOFT flow instrumentation in a full-size test section using single-phase water.

The TPFL can also be used to test relief and valves up to a pressure of 8.5 MPa. The relief valve testing is performed when the loop is modified with a relief valve attached to vessel 3 via a 4-in., Schedule 160 line.

The Steam-Air-Water (SAW) Loop, modified in early 1981, has been used to develop a counter-current flow correlation for the Westinghouse FLECHT-SEASET Test Facility.

5.3.1.3 Low Impedence In Situ Dynamic Testing

EG&G Idaho has the equipment and maintains the technical staff to perform in situ low impedence testing of components and structures to determine frequency, damping, mode shape, and transmissibility of such items. The equipment includes a Gen RAD 2508 experimental structural dynamic analyzer, electromagnetic and hydraulic shakers, and instrumentation system.

5.3.2 Computer Hardware and Software

5.3.2.1 Hardware

The Idaho National Engineering Laboratory (INEL) CYBER computer complex is operated for the U.S. Department of Energy (DOE) by the Scientific Information Division (SID) of EG&G Idaho, Inc. The main processors for this complex are located at the Computer Science Center (CSC) in Idaho Falls, Idaho.

The INEL dual CYBER 176 computer system provides scientific data processing services. It consists of the following major hardwares:

- CYBER 170 Model 196-A (Serial No. 201)
- CYBER 170 Model 176-B (Serial No. 206)
- Four CDC Model 18-20 Remote Terminals
- One Cope 1600 Remote Terminal
- DICOMED COM Microfilm Processor
- QUANTOR Q116 Microfiche Processor
- VERSATEC Model 3210-A Electrostatic Plotter (11 inch)
- VERSATEC Model 8222-A Electrostatic Plotter (22 inch)

Each of the above operates either independently or as a unit with the overall system. The CYBER 176 computers and the remote terminals generally operate within the combined system. The two COM recorders and the electrostatic plotters, however, operate off-line from the system, receiving their inputs in the form of magnetic tapes prepared on either the IBM 4341 or the CYBER system.

5.3.2.2 Software

In addition to utilizing classical solution techniques, numerous computer codes such as SAAS, SAP, FRAME, and ADINA are available at the laboratory to solve various stress analysis problems. In order to maintain state-of-the-art capabilities, development of new analytical methods is a continual on-going process.

The Applied Mechanics Branch also has a large number of state-of-the-art computer codes available to solve a wide range of dynamic problems. These computer codes include SAP, ADINA, ANSYS, NUPIPE-II, and HONDO.

5.3.3 Research Staff Capabilities

Staff capabilities at EG&G Idaho have been identified in the technical areas within the scope of this report.

5.3.3.1 Applied Mechanics Branch

The Applied Mechanics Branch offers state-of-the-art knowledge and capability in engineering mechanics, structural engineering, and mechanical engineering. Capability is available in such areas as pressure vessels, piping, pumps, valves, structural dynamics, seismic analysis, soil-structure interaction, thermal stress, fatigue, radiation growth, high temperature components, and ASME Boiler and Pressure Vessel Code, Section III and VIII components evaluations. The Applied Mechanics Branch is capable of analyzing static-dynamic, steady state-transient, and linear-nonlinear response in 1-, 2-, or 3-dimensions.

Experimental stress analysis support offers both analytical verification and field measurements. With the use of a fast fourier transform modal analyzer, modal analysis surveys of structures, systems, and components can be accomplished using excitation sources such as an instrumented hammer, an electromagnetic shaker or a hydraulic shaker. Collectively, this provides the capability in the field to experimentally predict or verify analytical predictions of frequencies, mode shapes, damping, and stresses related to structural response under dynamic loading conditions.

The laboratory test equipment includes the analyzers, exciters, transducers and associated electronics equipment needed to dynamically excite and measure a structural systems dynamic response characteristics. The analyzers are minicomputer based structural analyzers. They include a Gen Rad 2508 8-channel system and a smaller Hewlett Packard 5423A 2-channel system. These analyzers measure vibratory forces applied to the structure, vibratory response of the structure and compute system natural frequencies of vibration, natural mode shapes of vibration, and system damping for each mode. System response parameters may then be printed and animated on the computer CRT display.

Dynamic excitation equipment includes an impact hammer, two impact sledge hammers, a 100# electromagnetic vibration system, and a 3300# hydraulic vibration system. The vibration systems are driven by a wave form generator which can generate sinusoidal, triangular, square, random or a combination of these wave forms. The impact hammers are instrumented with load cells to measure the force of blows to the structure. These dynamic excitation systems are used to apply a measured transient force to the structural system.

5.3.3.2 NDE Engineering Branch

The NDE Engineering Branch of the Materials Technology Division (MTD) is composed of fourteen engineering and scientific personnel executing a mixture of programs extending from the application and developing of NDE codes to state-of-the-art research programs in ultrasonic imaging. Also within MTD is the Materials Engineering Branch, which executes welding and fracture mechanics programs ranging from the practical administration of welding codes to advanced research.

The NDE Engineering Branch has assembled automated NDE systems ranging from the ultrasonic system to an automated, real-time radiography system for characterizing the contents of waste storage drums. The Branch is active in performing NDE Engineering for NRC Research and Licensing activities. NRC Licensing applicant inspection programs are reviewed, SER's prepared and relief requests evaluated. Controversial problems with ASME Section XI Code rules are investigated. Examples are requirements for calibration blocks for examination of large diameter pipe fittings and use of calibration simulators for ultrasonic test systems.

The Branch is also involved in advanced research programs to extend the resolution and accuracy of ultrasonic images and to prove the feasibility of using ultrasonic beams to sense the size and location of molten weld pools during the welding process. Both of these research programs center around the development of new digital signal processing methods.

5.3.3.3 Materials and NDE Laboratories

The following laboratory capabilities exist in the Materials and NDE areas:

(1) Materials Joining Laboratory

- Solidification Studies
- Arc Physics Studies
- Automatic Pipe and Heavy Section Welding Development
- Electron Bead Welding Research
- Laser Welding Research
- Automatic Welding Development
- Coating and Surface Alteration Development

(2) Materials Testing Laboratory

- Mechnical Properties
  - Creep
  - Fatigue
  - Impact
  - Tensile
  - Hardness
  - High and low temperatures
  - Wide range of atmospheres

- Physical Properties
  - Thermal
    - Conductivity
    - Capacity
    - Expansion
  - Optical
  - Electrical
  - Magnetic

(3) Materials Examination Laboratory

- Optical and Electron Microscopy
- Surface Elemental and Molecular Chemical Identification
- Structural Identification and Characterization
- Quantitative Metallography
- Surface Film Identification - Oxides and/or Absorbed Layers
- Surface-to-Bulk Chemical Gradient Determination

(4) Nondestructive Evaluation Laboratory

- Surface Examination Methods Development
- Radiographic Image Enhancement with Direct-Viewing X-Ray Techniques
- Remote, Automatic Ultrasonic Test Methods Development
  - Hardware
  - Software
- Schlieren Imaging and Holographic Interferometry Studies
- Acoustic Holography Reconstruction in Ultrasonic Formation Techniques
- Infrared Imaging to Define Thermal Profiles

5.4 Areas of Future Interest Expressed by Laboratory Researchers

5.4.1 Simulation of Earthquake Effects

This proposal involves subjecting the LOFT facility to severe ground shaking in a range comparable to that of a large earthquake. The German HDR seismic tests have been constrained to low levels of ground shaking (1 to 2% of gravity). This proposed experiment would extend the HDR results to a much higher range. The NRC has previously indicated interest in a test program that would extend to a Safe Shutdown

Earthquake (SSE) seismic level of ≈0.2 g. The LOFT facility is designed for an Operating Basis Earthquake (OBE) of 17% and an SSE of 37%. Stress calculations have indicated no structural damage or failure would occur in the event of an SSE, and those portions of the plant that are essential for plant safety would remain operable. Thus, the plant would be protected against a severe core damage accident and any possible breach of containment. LOFT could incorporate small-scale seismic tests into the program now as long as such tests stayed small. The loads that the test loop is subjected to during large Loss-of-Coolant Experiments (LOCEs) are much more severe than low level seismic loads. As long as the experiments were in the range of the HDR experiments, these types of tests could be conducted while the plant was nuclear. Explosives could be planted to initiate the seismic event and to attain the level of ground shaking required.

5.4.2 Structural Damping Studies

This proposal involves using the LOFT facility to conduct experiments on structural damping. Snapback, impact, and shaker tests would be conducted on various systems such as piping, supports, containment, and other structures to determine the magnitude and uncertainties of structural damping. This data could then be directly applied to PWR design assessments.

This experiment could be performed in conjunction with the experiments proposed in Section 5.4.1. For separate effects, experiments would involve imposing a displacement on the various structures and measuring the vibrations with accelerometers and displacement transducers. Experiments of this type have been conducted extensively by ANCO Engineers. Based on structural analysis, these experiments would stay well within the plastic limits of the structures.

The problem of typicality of LOFT pipe sizes and other system structures compared to nuclear power plants would have to be resolved. This is a topic which could be investigated concurrent with the existing LOFT testing schedule.

5.4.3 Seismic Scram Equipment Experiments

This proposal involves using the LOFT facility in conjunction with simulated seismic input to determine the effectiveness and sensitivity of instrumentation. This experiment could be coupled with proposals presented in Sections 5.4.1 and 5.4.2, utilizing both approaches to obtain the acceleration and frequency ranges for effective use of seismic scram devices. This topic is more attuned to the structural response with respect to ground motion. This work might also be conducted in smaller LOFT-related single-effects facilities.

5.4.4 Impact of Dynamic Restraint Design on The Behavior of Piping Systems

This proposal would involve instrumenting and testing the performance of active restraints (solid-state energy absorbing devices) in the LOFT facility. This type of testing could also be performed in the LOFT related smaller test facilities. This topic has been investigated with shakers and explosive tests, but only on a low level of damping.

5.4.5 Valve Qualification and Certification Program

This proposed effort was identified by a DOE study. In 1980, DOE elected not to fund these proposals. A program should be set up to qualify and to certify valves for nuclear service. The program would have broad aims and functions. Principal aims and functions are delineated below:

(1) Identify those valves important to safety and plant availability that should be qualified or certified for service in nuclear stations.

(2) Establish initial goals or benchmarks to judge the standards for qualification of each valve for a particular service. Analyze historical plant data on valve failures to determine:

(a) Those valves judged to meet qualification standards.

(b) The basic causes of valve failures that do not meet qualification standards.

(3) Based on historical plant data, establish a qualified products' list for valves important to safety and operation.

(4) Establish specification standards for those valves on the qualified products' list and others by fully defining those features of each valve that insure its suitability for service.

(5) Analyze the basic causes of valve failures with the view towards solving hardware problems.

(6) Recommend implementation of solutions to valve hardware problems by utilities and manufacturers.

(7) Develop maintenance guides for valves.

(8) Develop test specifications to qualify and to certify valves and to test valves on the qualified list to assure continued adequacy for service.

(9) Establish manufacturer facility inspections and verification similar to ASME code stamp certification process.

This program would require a broad scope of participation including architect engineers, manufacturers, utilities, industry societies, and the NRC. The qualification and certification program outlined above could conceivably be done in two phases. Phase One could establish the feasibility of the qualification and certification program. Industry conferences and committees would be held to establish the scope and funding necessary to implement a qualification program. Phase One could fully define the program, determine its objectives and further detail the necessary programs and capital expenditures required. The feasibility study would also give consideration to the legal aspects of such a program. Phase Two could be the actual implementation of the qualification and certification program as defined in Phase One.

Separately, or as a part of the valve qualification program, the development of an adequate reporting system for valve failures is necessary. In order to identify the basic causes of valve failures, adequate information is necessary. This information is not available from the present reporting system. These systems need updating to identify the actual valve by model number and manufacturer and the circumstances of the event and subsequent repair. In addition, separating the normal maintenance occurrences from the abnormal, that are a result of valve defects, is necessary.

It is recommended that proposals be solicited from a number of valve manufacturers to conduct research and testing to accomplish the following broad objectives:

(1) Develop improved packing materials and stuffing box design for valves.

(2) Develop alternative mechanical packing systems for valves such as mechanical seals and labyrinth systems.

(3) Develop specifications for valves in nuclear service to achieve low leakage.

A program to mitigate variances in valve actuator operation is necessary. For electric motor actuated safety system valves, a program should be undertaken to fully define their operational characteristics, the sizing of actuators, and the survival of actuators and/or valves under stall-torque conditions.

The objectives of this program should include the following:

(1) Determine stress data on all valve parts and their maximum permissible load.

(2) Develop an interface specification for:

  (a) Calculation and selection of electric actuators.

  (b) Analyzing maximum allowable stresses of valve components in relation to stall-torques determined by actual testing by motor-operator manufacturers.

(c) Selection of actuators with gear ratios compatible with the survival of the actuator due to its own stall.

(d) Selection of hand-wheel override capability compatible with valve strength.

(3) Test several valve systems in independent laboratories for qualification under actual design flow and simulated earthquake conditions.

Proposals should be solicited from a number of valve manufacturers to participate in the program outlined above. Alternately, a program should be funded to investigate the feasibility of modifying existing motor-operated safety system valves with newly-developed actuator systems.

A number of unique valve designs are known to be utilized in overseas nuclear plants in Canada, Germany, France, and Japan. Japan, in particular, has recently completed a seismic test program which has identified deficiencies in relief valve operation during seismic excitation. A mutual exchange program with each of these countries to share experiences in improving valves in response to problems and new developments would be valuable.

5.4.6 Applicability of ASME Code Section III Appendix F to NRC Regulations

The objectives of this project are to review those portions of the ASME B&PV Code used in the NRC licensing process and to assess whether these portions require modifications or expansion. Particular attention should be given to Appendix F of the Code, "Rules for Evaluation of Level D Service Limits."

The ASME B&PV Code provides a set of rules for the selection of materials and material properties, design, fabrication, inspection and testing procedures for pressure boundary components. Section III, V, IX, and XI of the Code are directly related to nuclear components. The NRC has accepted the ASME B&PV Code in its licensing philosophy; therefore, the Code must be periodically reviewed against present engineering knowledge and present licensing philosophy. Often differences in opinion exist between the Code and NRC philosophies, and shortcomings must be evaluated. Recommendations to adjust the Code to satisfy the needs of the NRC criteria need be made to the appropriate Code Committee. This research program will investigate potential shortcomings in the Code and itemize specific recommendations.

There is a need to review the adequacy of specific articles of the Code in light of current engineering knowledge, advances in in-service inspection techniques and the current philosophy of the NRC. This review will furnish apparent shortcomings in the Code relative to the treatment of intergranular stress corrosion cracking environment, residual stresses in pressure vessels and piping, nonseismic dynamic environments, shell buckling, ultrasonic testing, fatigue strengths, fatigue analysis of Class 2 and 3 piping, and strain-rate dependent material properties. Other possible shortcomings may also be found and explored. Particular needs are to review Appendix F of the Code and to assess the relationship between Appendix F allowable design stresses and true component margins of safety.

The approach to be taken in this project is to compare ASME B&PV Code rules with NRC staff positions, present engineering knowledge, and current in-service inspection techniques. In addition to the specific items enumerated in the above paragraph, a section-by-section review of the Code rules will be made. Specific shortcomings in the Code will be noted in detail and specific recommendations for the modifications and expansion of particular Code rules will be generated. Where appropriate, detailed recommendations for additional articles to the Code will be developed.

This study will require an in-depth review of current engineerings and in-service inspection practices and coordinated discussion with many NRC staff members. The study will also require some mathematical analysis to examine the relationship between Code requirements and true margins of safety.

In summary, this program will provide an in-depth knowledge of the adequacy of the ASME B&PV Code rules to satisfy the licensing requirements of the NRC. The program will also provide a set of specific recommendations to the ASME on how the Code should be modified to bring it in accord with the NRC licensing requirements.

5.4.7 Anchoring of Component Supports

Investigations of expansion anchors at Shoreham and North Anna Nuclear Stations showed deficiencies in installation practices. At Millstone, a water hammer-like event caused pipe supports fixed in concrete by expansion anchor bolts to break loose and fail. Evaluation of the strength of expansion anchor bolts has been based largely on static tests which do not simulate the dynamic environments to which such anchor bolts will be subjected, nor the coupling to the supported structures or equipment and concrete cracking due to cyclic loading. Because of current uncertainty in expansion anchor capacity, a safety factor of 4 or 5 for SSE seismic loads is being employed in design as compared to 1.4 for poured-in-place anchors. In addition, there have been developed a number of so-called ductile drilled anchors which manufacturers claim should be permitted to use lower safety factors in accordance with ACI-349 Appendix B Code requirements.

Testing is required to evaluate the performance of expansion anchor bolts under dynamic cyclic loads. In conducting these tests, coupling and interaction with the supported structure must be considered for concrete and steel-lined walls. In addition, degraded and improperly installed expansion anchor bolts must be tested using various means of embedment and installation to determine sensitivity to installation errors. Improved rules for designing such anchor bolts, focusing on appropriate safety factors but including necessary design details such as base-plate flexibility and bolt ductility, must be developed.

5.4.8 Bolting Examination Improvement

Bolting failures due to corrosion have occurred in several plants and conventional ultrasonic techniques have not proven capable of detecting this corrosion damage. EG&G proposes to develop procedure qualification requirements and special procedures to reliably detect corrosion damage by ultrasonic examination. As a parallel effort, through membership on the ASME Code Groups, results of this work would be used to implement ASME Code revisions to incorporate these improved requirements.

5.4.9 Experimental Structural Dynamic Characterization of Piping Systems

Structural dynamic characteristics (mode shapes, frequencies, and damping values) of piping systems and associated supports are necessary for accurate dynamic behavior prediction. However, due to the complexity of system geometry and support construction, as well as significant nonlinear behavior of some types of supports for large loads, measured and analytically predicted behavior may be significantly different for many typical piping systems. Thus, a need exists for experimental dynamic characteristic evaluation of typical piping systems, including commonly used supports for loads of various magnitude loads. EG&G has the capability (Gen Rad 2508 experimental structural dynamic analyzer, electromagnetic and hydraulic shakers, impact test system, and instrumentation system) and facilities (many typical in situ piping systems, for example, LOFT and facilities for construction of special systems, including the LOFT Test Support Facility) for performance of experimental piping system structural dynamic characterization. These capabilities and facilities may be used for a wide variety of experiments to determine dynamic characteristics (in particular, modal damping values) for various types and magnitude of input loading.

5.4.10 Pipe Repair Welding Coordinating Effort

A central laboratory should be identified to coordinate national efforts in nuclear pipe welding fabrication and repair. This liaison activity would improve planning and coordination of experimental activities and would avoid duplication. INEL interest, experience, and capability could be used to bring all appropriate developments together and interfaced to in-house work to facilitate the broad understanding required for regulatory guides.

Successful fabrication and repair depend on identification of critical process-controlled parameters (including heat treatment) from more than twenty available parameters in mechanized GTAW and their tolerances on the size, distribution, and orientation of weld discontinuities. Elucidation of the effects of these parameters on pipe integrity will be the goal of this study.

Specific requirements for repair procedures and regulatory guidelines can be developed from the results of the above items, and experiments can be designed to simulate repair procedures. Establishment of repair procedures will be based primarily on critical mechanical properties.

5.4.11 Irradiation Effects and Annealing

The Materials Technology Division is interested in participating in the Irradiation Effects and Annealing Research Program. The plan includes studies on pressure vessel embrittlement and removal of embrittlement by annealing. Embrittlement/annealing situations for specific plants and newer, higher-strength steels will be investigated.

(1) In addition to studies presently underway, the correlation of Charpy V-notch to large size specimens and identification of specific defect types responsible for decreases in fracture toughness should be studied. Mechanisms responsible for initiation and growth of these defects should be identified, which will suggest possible mechanisms affecting recovery and reembrittlement. This will complement, not duplicate, studies conducted at other facilities, such as at NRL.

(2) Since there is an effect of stress on swelling and recovery, effect of stress on recovery of original fracture toughness should be evaluated.

(3) Embrittlement/annealing requirements for specific plants should be established. This task will require the following:

(a) Characterization of types of steel and weldments with respect to metallurgical factors of chemistry, microstructure, etc.,

(b) Establishment of a data bank on effects of metallurgical factors (in addition to copper and phosphorus) on loss-of-fracture toughness, and

(c) Establishment of a data bank on effects of irradiation for newer, higher-strength steels.

5.4.12 Nondestructive Examination

Current NDE techniques applied to nuclear reactor components have often been found to be unreliable for timely flaw detection and inadequate for flaw sizing. NRC has several efforts ongoing or planned to alleviate these shortcomings. The following efforts should be considered:

(1) The field capabilities of the automated ultrasonic inspection system should be used in instances where NRC requires an impartial, third-person confirmation of PSI and ISI data.

(2) The LOFT reactor could be used as a test bed to qualify advanced techniques for future application to commercial reactors. Appropriate material discontinuities could be induced and candidate inspection techniques could be evaluated under field conditions requiring remote operation, difficult physical access, and typical environmental conditions (temperature, humidity, radiation, etc.).

(3) The considerable body of INEL field experience could be utilized to perform a cost-benefit analysis of the existing, automated inspection system versus manual inspection methods representative of commercial capability. NRC programs now active are intended to provide quantitative information on the reliability (the benefit) of both manual and automated systems. While this information can be achieved under laboratory conditions, the cost information is best achieved under actual field conditions. INEL reactors could be used to provide realistic test beds for the cost-benefit analysis of intended changes in reactor inspection codes.

(4) A need for an NDE method to detect and to locate voids in thick concrete sections has recently arisen. In the past two years, INEL personnel have been active in this research area and are prepared to extend the current practices to the point where such practices are codified in a form suitable for use on commercial reactors.

(5) INEL has developed codes and procedures for the UT inspection of reactor bolting. These procedures have been found to be very effective on INEL reactors. With the addition of a fracture mechanics analysis of critical flaw sizes and an NDE analysis of detection reliability versus flaw size, these procedures are ready for submission to the appropriate ASME Code Committees.

5.4.13 Advanced Ultrasonic Testing Implementation

The EG&G Advanced Ultrasonic Testing (AUT) System is ready for routine field operations as a result of thorough field testing of hardware, software techniques and procedures at LOFT during development. It has been demonstrated that the AUT System is an improvement on current manual methods, but it has not been possible to obtain the quantitative inspection reliability data necessary to establish the system's cost-effectiveness for wider service.

EG&G proposes the following program goals:

(1) Provide quantitative data on the inspection reliability of an advanced inspection system,

(2) Obtain a statistically valid comparison of the AUT System and current manual methods,

(3) Derive data from which to establish a cost-benefit ratio for the AUT System,

(4) Extend final recommendations to actual implementation by incorporating the recommendations into working documents.

Implicit in these goals is the desire to obtain the widest possible recognition that advanced techniques, as represented by the AUT System, are sufficiently mature for near-term application to reactor safety problems and to encourage the implementation of these advanced techniques on operating plants through appropriate Code changes.

5.4.14 Improved Surface Examination Processes

Penetrant examination is used extensively for ASME Code Section XI examinations. EG&G routinely uses fluorescent penetrants for in-service examination of piping welds. This practice is an allowable option by the present ASME Code. The use of fluorescent penetrants versus color contrast penetrants provides more reliable detection of in-service defects such as fatigue cracks. EG&G proposes to quantify this improved sensitivity and gather cost-benefit data for use on plant hardware. The resultant data will be used to implement appropriate Code changes.

ASME Section V does not control penetrant materials other than for sulfur and halogen content. No controls or important properties such as sensitivity, washability, viscosity, etc., are given. EG&G proposes to evaluate control requirement used for military and aerospace hardware and recommend appropriate requirements for reference by and/or incorporation into ASME Code and/or NRC documents.

5.4.15 Forging and Plate UT Examination Improvement

ASME Code ultrasonic NDE requirements and acceptance criteria are not realistic for some applications utilizing forgings and/or plate. Depending on a complete loss of back reflection for rejection makes accept/reject very dependent on the transducer used. The large flaws allowable may not be tolerable, for example, in a steam generator tube sheet. The three-inch allowable indications would extend through the wall between adjacent tube penetrations.

EG&G proposes to develop alternate ultrasonic requirements for incorporation into the ASME Code and define applications where implementation of these alternate requirements should be required.

5.4.16 Piping UT Examination Improvement

ASME Section III Code requirements for UT of pipe do not assure detection of centerline lack of penetration in welded pipe as indicated in IE Bulletin No. 79-03A. This type of flaw should be detectable with UT, provided a proper technique is used. The wide latitude in techniques allowed by the Code (for example, contact or immersion, no control or transducer size) does not always provide for a proper technique. EG&G proposes to draft procedure qualification requirements and improved procedures and implement these improvements through publication as Code rules.

5.5 Liasion with Foreign Nuclear Safety Research Activities

5.5.1 LOFT

The LOFT facility is by far the largest and most comprehensive of such facilities in the world built specifically for loss of flow experimentation and as such has attracted much foreign interest and liaison.

5.5.2 Relief Valve Test Stand

To the extent relief valve test facilities do not exist in foreign countries, it is felt the two-phase flow loop relief valve test configuration should be of interest to foreign organizations.

5.5.3 Evaluation of Pipe Damping

EG&G Idaho personnel hope liaison will be possible with Japanese industry and government sponsored test programs to better define pipe system damping and more optimum pipe support configurations.

5.5.4 Liaison with HDR and KUOSHENG

EG&G hopes to maintain continued liaison, support, and review of the Heissdampfreactor and the Kuosheng, Taiwan test programs.

6. LAWRENCE LIVERMORE NATIONAL LABORATORY

6.1 Organization

The Lawrence Livermore National Laboratory (LLNL) nuclear safety research efforts in the technical areas of interest covered in this report are concentrated in the Nuclear Systems Safety Program Office (NSS) with L.L. Cleland, Program Leader as shown in Figure 10. Within this group, the bulk of the safety research is concentrated in the Reactor Engineering and Seismic and Structural Safety Program organizations. These two groups have approximately 50 persons with current NRC Office of Nuclear Regulatory Research funding of approximately $\$5.6 \times 10^6$ and Office of Nuclear Reactor Regulation funding of $\$4.8 \times 10^6$.

The LLNL organization to a greater extent than most national laboratories has contracted the use of outside consultants to assist it in performing its technical tasks. This use of consultants assures that many viewpoints are considered.

6.2 Topic Areas of Interest

6.2.1 Office of Nuclear Regulatory Research

(1) Seismic Safety Margins Researcn Program

(2) Load Combination Program

(3) Seismic Scram

(4) Foreign Liaison/BWR Containment

(5) Seismic Characterization of the Eastern U.S.

6.2.2 Office of Nuclear Reactor Regulation

(1) Seismic Review for SEP

(2) Engineering geoscience source team evaluations

(3) Seismic qualification of auxiliary feedwater systems

(4) Failure analysis of materials

(5) Licensing reviews

6.3 Facilities

6.3.1 Experimental

The LLNL maintains staff at three sites and utilizes facilities at several other sites as needed. The Laboratory's main site in Livermore occupies 640 acres and is located 40 miles from San Francisco. Site 300

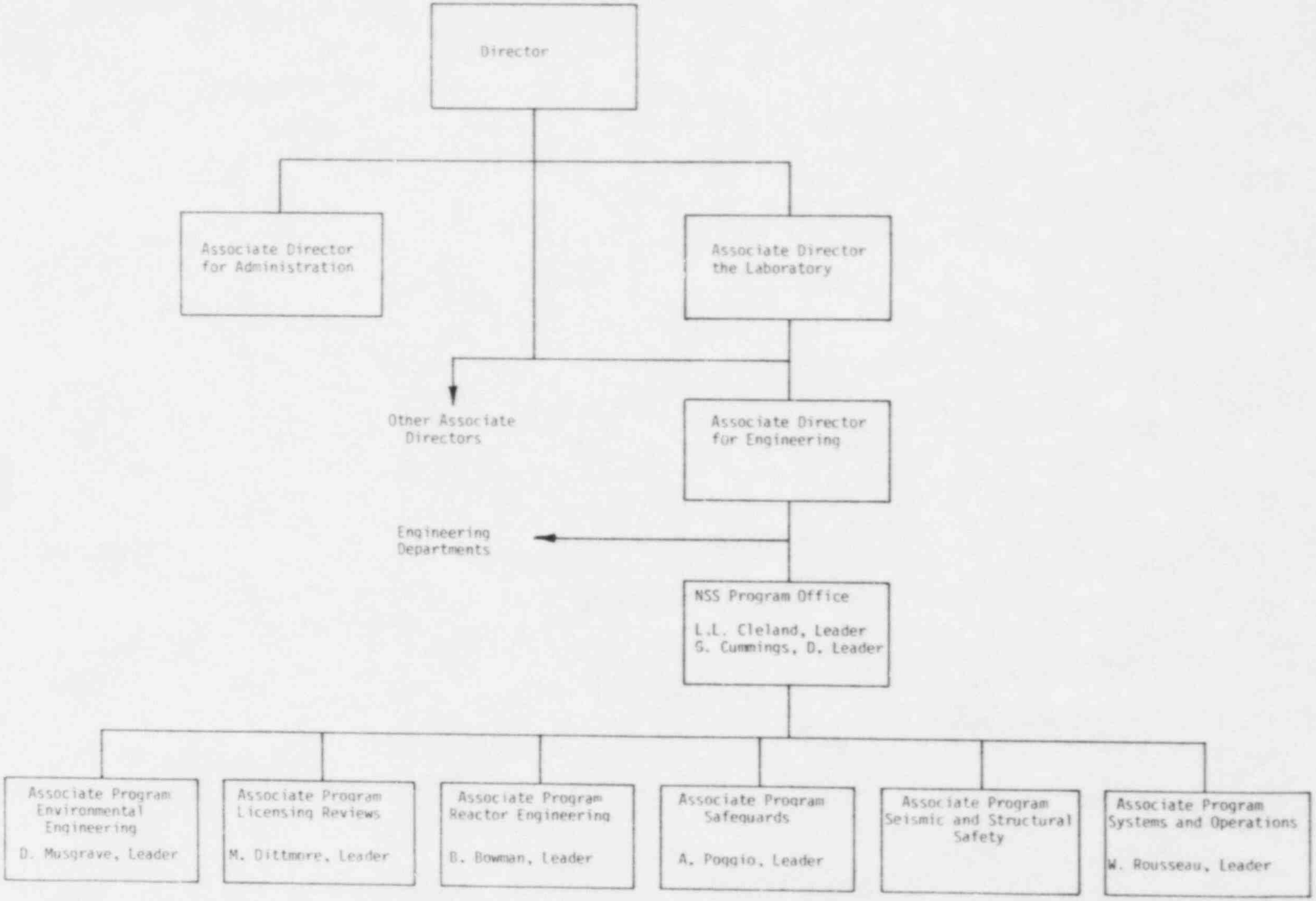

Figure 10 LLNL/NSS Program Organization Chart

covers 7,000 acres in the Corral Hollow area about 15 miles east of Livermore. In 1955 this site was established as a major research facility of the Livermore Laboratory to allow scientists and engineers to conduct experiments with high explosives in a remote environment. Currently hydrogen igniter experiments are being conducted at Site 300 for the NRC Office of Nuclear Reactor Regulation. To test Livermore-designed explosives, the Laboratory maintains a small, permanent staff at the Department of Energy's Nevada Test Site where nuclear explosives testing has been performed underground since 1963. Facilities at several other sites can be and are used by the Laboratory. The most notable site for seismic and structural experiments is the University of California, Richmond Field Station which is one of the country's largest shaker table facilities.

The sites available to Laboratory staff provide some unique experimental areas in which to conduct seismic and structural experiments. Both Site 300 and the Nevada Site may be used for seismic testing of large structures where high explosive charges in surrounding soil or rockets attached to the test structure are required for excitation.

The Nevada site has significant ground motion when nuclear explosives are detonated. This motion has been recorded and studied by Laboratory engineers and scientists. It is felt that this motion can be used to provide an earthquake-like environment for testing of full or scaled structures. Knowledge of the yield and location of the detonations relative to the test structures can provide "controlled" ground motion to simulate major earthquakes.

The LLNL has extensive testing facilities where experimental seismic and structural analyses and research can be carried out. A condensed summary of the key facilities are presented herein.

6.3.1.1 Multiple-Actuator Hydraulic Shaker Facility at Site 300

The LLNL multiple-actuator hydraulic shaker facility was developed in response to programmatic requirements for a high force, high amplitude, low frequency shock and vibration test facility. The facility is unique in its combination of performance capabilities, computer-controlled versatility, and ability to test hazardous materials.

Testing in the vertical axis is accomplished by driving a rigid fixture to which the test specimen is secured. Each actuator has a maximum dynamic thrust of 25,000 lbs. and a total stroke of 5 in. High-performance, 2-stage servo values rated at 0.5 ft.^{3}/S and having a response time of 2.6 x 10^{-4}s result in an actuator acceleration control bandwidth of 500 Hz, a velocity limit of 6.5 ft/sec, and an acceleration limit of 45 g.

The control room contains the computerized control system and data acquisition equipment. The control system is unique in that it is the only direct digital, multiple-actuator system for decoupled control of sine, random, and transient vibration testing. Under computer control, the actuators are easily programmed to perform virtually any test within the performance limits of the actuators.

The facility is equipped with a wide range of signal conditioning and data acquisition equipment. The general instrumentation equipment includes 24 channels of strain gage conditioning, 12 channels of thermocouple reference junctions, 24 channels of differential amplifiers, and 24 channels of piezoelectric transducer signal conditioners. Recording equipment includes a 32-track FM tape recorder and two 24-channel light-beam oscillographs. In addition, the computer control system can acquire and display a limited number of channels directly, while the remainder can be digitized and reduced immediately after testing is complete.

The flexibility and high performance of the multiple-actuator shaker facility have led to the system's use in a wide variety of applications. The applications include simulated vibration and shock tests for weapon systems, seismic proof testing of engineering power equipment, high rate tension/compression tests, and vibration testing of a large solid propellant rocket motor. Tests have been performed on specimens from 5 lbs. to 10 ton in size and over a frequency range of 1 Hz to 500 Hz.

An electrodynamic shaker with a 40,000 lb. peak force capacity is being added to this facility and should be operational by July 1982.

6.3.1.2 Dynamic Test Complex At Site 300

The Dynamic Test Complex provides facilities for the shock and vibration testing of systems, components, and assemblies containing high explosives or hazardous materials. The complex consists of two principal areas located approximately .5 mile apart. A wide variety of test technologies are represented at this complex.

At the main facility, a central control room provides instrumentation, data acquisition and control facilities for four test cells. Three 14-track FM tape recorders provide up to 42 channels of data acquisition for either piezoelectric or strain-gage-type transducers. An additional 42 channels of light-beam oscillograph recording are also available. For tests requiring thermal conditioning, up to 60 channels of thermocouple signal conditioning can be provided. Closed-circuit TV provides remote observation of activities in the test cells.

Three different shock testing machines are located in cells at the main facility. The largest of these, an 18 in. bore horizontal pneumatic actuator, is capable of generating a 400,000 lb. thrust with a rise time of 1 to 60 ms. With a 200 lb. load, accelerations of 100 g to 1000 g and a peak velocity of 325 ft/s can be achieved. The test item is attached to a 3 ft. wide carriage and is propelled by the actuator along a 70 ft. long track to impact a 20 ton block or to be braked to a slow stop. Alternately, lead crush pads on the face of the impact block are often used to shape the deceleration pulse. For shock testing of smaller specimens, a 6 in. bore vertical pneumatic actuator rated at 40,000 lbs. and a small drop tester are also available.

Two electrodynamic shakers, located in separate test cells, are used for simulation of high frequency ground and flight environments. The largest of these, a 30,000 lb. thrust MC C-210 exciter, has a maximum displacement of + 0.5 in., a velocity limit of 6 ft/s, and a freqency range of 3 to 2000 Hz. A hydrostatically supported slip table is used for lateral testing up to 300 Hz. Thermal conditioning chambers, used on both this shaker and a smaller 7,500 lb. thrust MC C-70 exciter, allow testing to be conducted over a temperature range of -100°F to 300°F. An 80-channel analog analyzer/equalizer is used for random vibration test control while sine control is accomplished with a conventional tracking-filter-type controller.

A 100 ft. drop tower provides a guided free-fall impact facility for specimens up to 4000 lbs. Test specimens containing high explosives or inert materials are dropped through the center of the tower and impact at velocities up to 80 ft/s.

Data acquisition and control of the experiments in this complex are handled by a minicomputer system. This system provides the following services: 1) interactive setup and calibration of instrumentation, 2) direct digital data acquisition, 3) online data analysis, and 4) digital control of the vibration shakers.

6.3.1.3 Livermore Shaker Facility

An electrodynamic shaker with a peak force capability of 36,000 lbs. also exists at Livermore. This capability is similar to that found at the Site 300 Dynamic Test Complex. The Livermore Shaker Facility, however, has no equipment for testing hazardous materials.

6.3.1.4 Instrumented Hammer

A modal analysis system exists that allows determination of mechanical vibration modes of structures. Data is collected by tapping parts in various locations with a hammer instrumented to provide input force. Structure responses are recorded and analyzed to give the natural frequencies, damping coefficients and mode shapes.

6.3.1.5 Mechanical Properties Testing Facility

A wide range of specimen preparation and mechanical properties testing is performed in this facility: preparation and vacuum annealing of test specimens; fracture-toughness, fatigue-cracking, and tensile testing; high-vacuum, high-temperature creep testing; computer control of tests and computer data acquisition; and equipment development and setup for in situ creep testing under 14 MeV neutron irradiation.

This facility has the following capabilities:

(1) Specimen preparation of large and small specimens; high-vacuum, high-temperature annealing and other special treatments used to prepare specimens,

(2) Tensile testing on the Instron universal testing machine from 0-2400 C under vacuum to 13 μPa (10^{-7} Torr),

(3) MTS fatigue-crack growth for studying the effect of various atmospheres on fracture toughness,

(4) High-vacuum creep at temperatures up to 2400 C for studying the creep of materials under stress and pressures down to 0.13 μPa (10^{-9} Torr),

(5) Computer control and data acquisition for mechanical-properties tests, fatigue-crack monitoring, and remote testing, and

(6) Setup and staging for the in situ creep-testing unit used in studies on 14-MeV neutron radiation damage performed at the LLL RTNS-II neutron source.

The equipment includes sample-preparation and polishing equipment, optical microscopes, a scanning transmission electron microscope (STEM), a scanning electron microscope (SEM), an electron microprobe, and several fully-equipped dark rooms to document the results observed with these instruments.

6.3.1.6 Materials Test & Evaluation Facilities

These facilities determine the mechanical behavior of engineering materials, systems, and assemblies under specific stress and environmental conditions. LLNL maintains capabilities to conduct mechanical tests according to ASTM specifications or by unique tests designed for specific program requirements. Experiments can be carried out at temperatures ranging from near absolute zero to 2500 C, at strain rates from essentially static (creep tests) up to 10 in/in/sec, and at loads up to 500,000 pounds. Loads can be applied in tension, compression, bending, torsion, and biaxially. Fracture, fatigue, creep, and impact tests can also be conducted. The wide variety of testing machines, load cells, extensometers, strain gages, and test fixtures permits evaluation of all types of engineering materials including metals, alloys, ceramics, glasses, organic materials, and fiber composites. LLNL personnel have a high level of engineering and technical expertise in mechanical properties measurements and in the special fields of fracture mechanics, experimental stress analysis, composite materials evaluations, high-strain rate testing, acoustic emission, and stress corrosion cracking.

(1) Materials Testing Laboratory

Tests or experiments are conducted for the following purposes:

(a) Determining mechanical properties,

(b) Characterizing the mechanical behavior of materials, and

(c) Evaluating the environmental response of assemblies or sub-assemblies to force, temperature, or other parameters.

In the performance of the above tests, extensive fixturing design, computer assigned data acquisition and analysis, and detailed engineering calculations are required.

(2) <u>Acoustic Emissions/Ultrasonics Laboratory</u>

The work activities of AEUL include the following:

(a) Proof testing parts, components, and materials for quality control,

(b) Monitoring flaw growth and determining flaw location during proof testing of structures,

(c) Determining bulk properties of materials by ultrasonic testing and other special applications,

(d) Monitoring flawed and unflawed metal and composite samples to study deformation and microfailure processes in these materials,

(e) Identifying and characterizing sources of AE in metals and composites,

(f) Improving data gathering and data processing to extract additional information from AE data, and

(g) Conducting experiments to study wave propagation of AE in structures.

(3) <u>Fracture Mechanics Laboratory</u>

Test programs are developed and initiated to certify material is of acceptable fracture toughness, to investigate design alternatives, and to establish fracture models (or fracture criterion) for the given material or design. The Laboratory also investigates the behavior of materials in various environments at different temperatures and strain rates.

(4) <u>Mechanics of Composite Materials Laboratory</u>

Equipment and facilities are designed for the characterization of the mechanical properties of composite materials which are dependent on time-history and environment. Servo-hydraulic control testing machines and dead-weight creep and interrupted creep machines provide a wide range of mechanical excitation sources. These two classes of test units permit testing down to the time range of 0.01 sec and 1000 sec, respectively. Adequate data acquisition is provided by seven microprocessor computer systems with data rate ranges of 10 kHz and 1 kHz. These data-acquisition systems will be connected together as a computer network system. With these systems, creep-relaxation function and material damage due to creep and fatigue can be determined.

(5) High Rate Test Facility

High rate testing is conducted using a Hopkinson Pressure Bar technique. Rates to 10μ/s, temperatures to 500 C degrees and strains to 100% are accomplished. A high-speed camera records strain. The facility is supplemented by a commercial High Rate Test Machine (MTS). Test speeds of 420 mm/s in closed-loop and 6000 mm/s in open-loop operation are possible.

6.4 Areas of Future Interest Expressed by Laboratory Researchers

6.4.1 Seismic Margins

For some time, many in the nuclear industry have believed that excessive margin exists in the NRC seismic safety requirements. Recently, work on the SSMRP on SONGS-1 has provided results illustrating one approach to quantifying a certain aspect of seismic margin. The following additional work should be performed:

(1) Extend the work from in-structure spectra to include margins in piping moment,

(2) Obtain a better understanding of margins, and

(3) Obtain a better understanding of the strengths and weaknesses of seismic safety requirements.

This work could be done by prolonging the extensive work already performed on SONGS-1 and ZION-1.

6.4.2 Attributes of Improved Seismic Design

Recently, considerable advances have been made in understanding the relationships between and relative importance of the various elements of seismic design. Studies should be performed to develop a design process that recognizes the limitations of our ability to predict seismic response and provides more balance between the various analytical efforts used in design.

6.4.3 Seismic Qualification Test Data

A considerable body of knowledge and data resides in the laboratories that have performed seismic testing over the years. However, for various reasons, this information is generally unavailable. Research should be performed to develop well-defined approaches for extracting the important information from test reports, for correlating this information, and for summarizing this information.

These approaches should then be executed by the test laboratories under close interaction with a technical cognizant body to develop reports which make the appropriate information available to the industry, but at the same time, preserve the anonymity of the basic test data.

6.4.4 Nonlinear Soil Response

The effects of site response (SR) can lead to significant modifications in the seismic hazard definition at many sites. Soil-structure interaction (SSI) remains a controversial area after many years of analytical studies. Both SR and SSI are heavily dependent on the nonlinear response of soil; therefore, a key first step in resolving these issues is to obtain a better understanding of how to model soil. The body of test data acquired at the Nevada Test Site over the years is a unique available resource that could and should be used for this purpose.

6.4.5 Reliability of Seismic Testing

More and more seismic testing is being performed around the world in an attempt to obtain a better understanding of the possible effects of earthquakes. However, rarely if ever, are the statistical uncertainties recognized. Research should be done to develop the following:

(1) A better understanding of the real inferences that can be made from test results,

(2) Approaches to define conditions and test items under which the data sought is predominantly statistical and when it is not, and

(3) The use of statistics to develop test plans that maximize the acquisition cf good data.

6.4.6 Statistical Load Combinations

Recent work has led to advances in our ability to use reliability theory to combine the extreme loads used in the nuclear industry. The results of this work should be applied to selected issues in nuclear design.

6.4.7 Core Status

The reactor core experiences an ambient vibration due to normal operation. This vibration can be monitored to obtain a normal "signature" of the core. It is believed by some that this signature is relatively sensitive (compared to other nuclear structures, systems and components) to core damage. Research should be performed to determine if this method can provide added useful information on core status during accidents.

6.4.8 Design and Construction Errors

Design and construction errors have a significant influence on nuclear power plant structures, for example, Diablo Canyon piping support installation errors. Some preliminary work is underway relating such errors to piping systems as part of the Load Combinations Program. A more general study of this problem as it relates to nuclear power plant equipment and structures seems warranted. Such a study should take into account the relationship of quality assurance and in-service inspection programs to the probability of serious design and construction errors. As a start, definitions of what constitutes a design or construction error need to be established. Work on such definitions has already been started as part of the Seismic Safety Margins Research Program.

6.4.9 Independent Evaluation

Various agencies around the world use a spectrum of methods to review the safety of various structures, offshore platforms, airplanes, and so forth. These methods should be surveyed, summarized, and compared with methods used by the NRC on nuclear power plants.

6.4.10 Electrical Failures Caused by Dynamic Loads

Electrical components appear to be particularly sensitive to seismic or other dynamic loads. Relays and circuit breakers will chatter and sometimes fail due to such loading. Studies of the effect of chatter on the safe electric power supply and equipment performance at nuclear power plants need to be made to assess the importance of this problem. The probability of chatter caused by dynamic loads is much higher than its failure probability assuming chatter does not cause failure. If chatter does result in failure to perform a safety function, the probability and nature of such failures need to be determined.

6.5 Liaison with Other Foreign and Domestic Nuclear Safety Research Activities

The LLNL personnel expressed an interest in setting up a technical center for foreign technical liaison. Other than the engineer LLNL has supplied to the HDR project in Karlsruhe, FRG, and their host activities to visiting foreign research engineers, LLNL does not have any direct interaction with foreign or other domestic research efforts. However, in the past LLNL carried out an extensive foreign liaison program for the NRC concerning BWR Containment Blowdown load testing in Germany, Japan, and Taiwan.

7. LOS ALAMOS NATIONAL LABORATORY

7.1 Organization

The Los Alamos National Laboratory (LANL) safety research efforts in the technical areas of interest in this report have been concentrated in Energy (Q) Division with J. Jackson as Division Leader as shown in Figure 11. Within the Q Division, the groups most concerned with nuclear safety research are Advanced Engineering Technology, Q-13, under Group Leader, C. Anderson and Safety Assessment, Q-6, under Group Leader, R. Haarman.

7.2 Topic Areas of Interest

The Advanced Engineering Technology Group, Q-13, currently is performing research funded by the Nuclear Regulatory Commission in the following areas related to this study:

(1) Margins to Failure Program

  (a) Determine margins to failure of Category I structures other than containment. Dynamic Testing of Shear Wall Structures at high stress levels by means of both pulsator and shaker table input.

(2) Seismic Response of Nonlinear Systems by Computer Analysis of One and Two Degree of Freedom Systems

(3) Analytical Response of Concrete Containment Structures to Internal Hydrogen Detonation

(4) Steel Containment Buckling Program

  (a) Evaluate Area Replacement Method in Reinforcing of Circular Opening of Cylindrical Shells Subjected to Buckling Loads by Tests

  (b) Develop Benchmark Experiments to Evaluate Buckling Computer Code Predictions

  (c) Develop Dynamic Testing and Analytical Program to Predict Dynamic Buckling and Criteria Development

The Safety Assessment Group, Q-6, currently is performing research as indicated.

7.2.1 Department of Energy

(1) Accident Analysis Code Development - EVENT and TVENT codes developed to simulate explosively driven transients (explosions and tornadoes) within nuclear facilities.

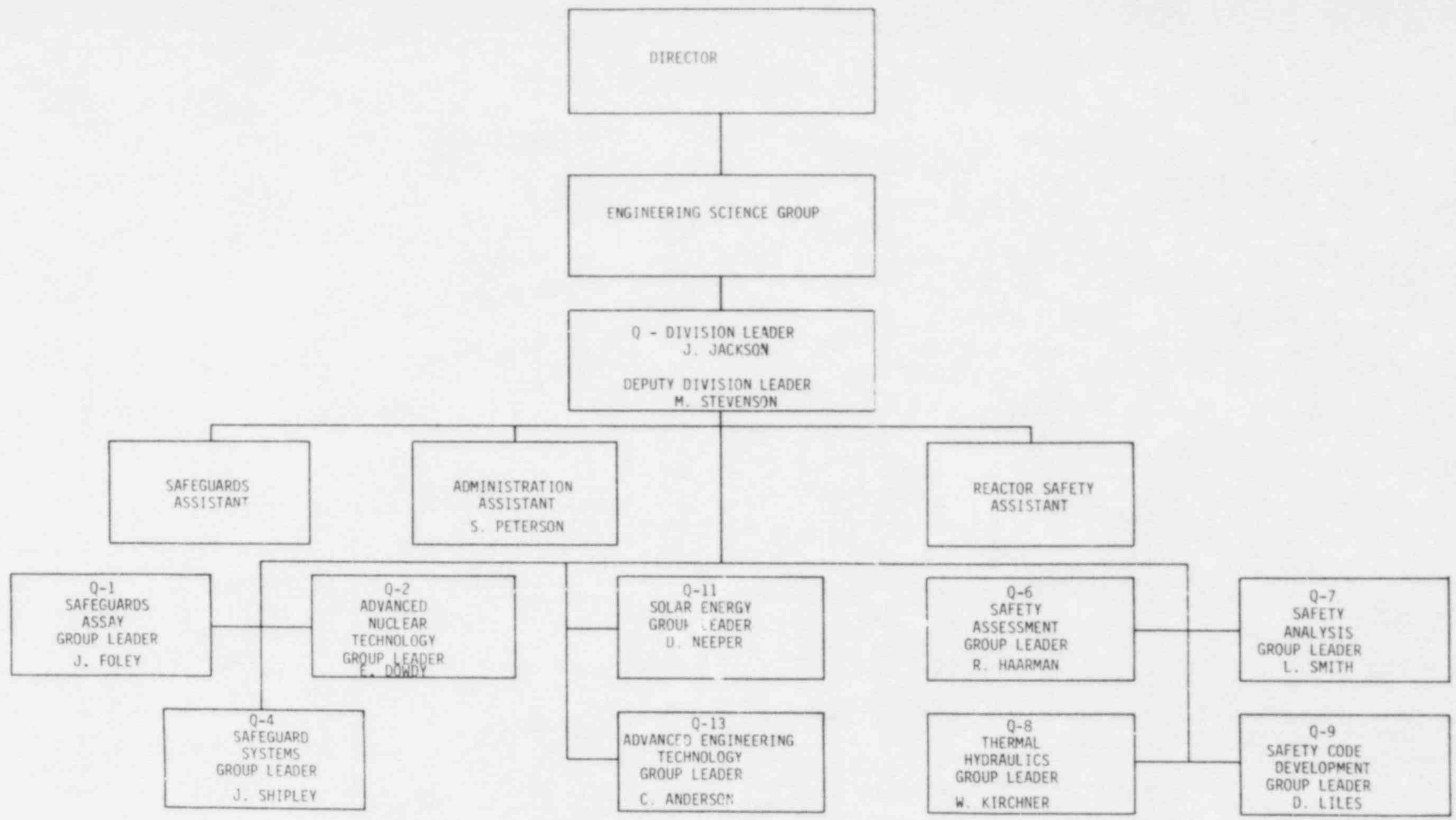

Figure 11 Los Alamos National Laboratory Organizational Chart

(2) Experimental Tests performed to determine operating characteristics of typical blowers and valves in nuclear facilities to benchmark the EVENT and TVENT computer codes.

(3) Tests performed on simulated ventilation systems for pressure transients (moderate to shock) to provide verification of EVENT and TVENT code development.

(4) Tests performed on filtration devices (HEPA filters) to determine effectiveness in confinement mode.

7.2.2 Office of Nuclear Reactor Regulation

(1) The margin of conservatism in existing containment design codes such as COMPARE-MODIA was compared with the more advanced best-estimate containment code, BEACON-MOD3. In particular, the reactor cavity forces and movements were compared to determine any significant difference.

7.2.3 Office of Nuclear Regulatory Research

(1) Multi-year program to develop Accident Analysis Handbook and develop computer codes to examine accidents in facilities typified by confined structures without windows and minimum door openings. These programs are used primarily by fuel cycle facilities.

(2) Develop a fire code, FIRAC, to predict fire spread within confined structures.

(3) Experimental program to determine response of containment devices (fitters, blowers, and valves) to accident-induced stress.

7.2.4 Other

(1) Provide assistance to application of TVENT to non-Los Alamos facilities. (EG&G)

(2) Provide technical expertise to determine effectiveness of filtration devices in containments. (SNL)

7.3 Facilities

7.3.1 Experimental

7.3.1.1 Large Volume Press

The Los Alamos 5000-ton static draw press provides a general purpose reaction frame for large volume and high load experiments. This apparatus consists of a cluster of four simple piston and cylinder pressure-generating systems of 500 tons each surrounding a main 3000 ton system. Cluster and main rams are operable either individually or in tandem. A 350 ton ejection ram located in the base of the frame can be used to position large samples. Because all frame components are cast rather than welded, modifications to the press might include pressure capabilities to 1000 bars, effective stress capacity to 1500 bars and temperatures to perhaps 300°C without decreasing the machine's inherent stiffness. A test cell to be designed for triaxial experiments using fluid pressure may include a torsion component for internally heated samples. Using conventional length to diameter ratios, specimens up to a meter in diameter can be tested.

7.3.1.2 Explosives Capability

LANL possesses a strong capability for the fabrication, testing, and use of many common and developmental high explosives. Of interest to the NRC should be use of this explosives' technology in the testing of model structures (concrete, steel, or mixed concrete and steel) to simulate blast loading on reactor structures and components of nuclear power plants.

7.3.1.3 20,000 lb. Force Shaker

LANL possesses a 20,000 lb. force electrodynamic shaker with a slip table. The frequency range of the output from the shaker, 20-2000 cps, while out of range of prototype earthquake pulse frequencies, is well within the frequency range of earthquake pulses scaled for testing model structures of reactor components. This shaker has a random excitation capability and good ancillary recording and control equipment.

7.3.1.4 Material Property Testing Machine

At Los Alamos there are several multiaxial test machines for studying the static or low strain rate dynamic failure characteristics of metals and plastics under conditions of biaxial stress. Temperature condition equipment is available.

7.3.1.5 Stress Wave and Impact Testing

The two main pieces of equipment used in this type of testing are a 200-mm diameter single-stage gas gun and a 60-mm diameter split-Hopkinson Bar. The gas gun can fire an impact specimen at velocities from 10 to 400 m/s. Extreme care is taken to produce highly planar impacts, which are a necessity at very low velocities. The desired goal is to attain planarties of only 60 microradians. With this gun, strain rates as low as 2 or 3 times 10^4 s^{-1} can be produced. The target chamber and catch tank are evacuated, and provision is made for "soft" catching of the target. Both real-time and recovery experiments can be conducted. The Hopkinson Bar has an axial preload capability up to 0.2 GPa (30,000 psi). Also, by the end of 1982, plans called for the addition of a confining vessel, providing for confining pressures of the same magnitude and a temperature to 250°C (489°F). Pore pressure can be independently controlled. The Hopkinson Bar will operate in the strain-rate range from about 10 to 10^4 s^{-1}.

7.3.1.6 New Mexico State University (NMSU)

The Los Alamos test facility is located on the NMSU campus with operation and testing services provided by the NMSU Mechanical Engineering Department.

The apparatus at this test facility are used to test the capability of filtration devices under abnormal conditions. Using these apparatus, we are able to generate varying degrees of flow transients to simulate both natural and man-caused accidents. The facility also has been used by Karlsruhe Nuclear Research Center, Federal Republic of Germany, to test their filtration devices. A large wind tunnel that will also be used to obtain experimental data on re-entrainment and deposition is under construction at NMSU. Some of the experimental apparatus are described in greater detail below.

(1) Blowdown Apparatus

The purpose of the blowdown apparatus is to impose relatively slow (0.5 s to 6 s) pressure pulses across ventilation system components. The system is capable of generating pressure levels of 27.5 kPa (4 psi) and volumetric flows of 11.8 m^3/s (25,000 cfm). The system consists of 2 large pressurized tanks, sonic nozzles, a prefilter chamber, and a wind tunnel.

(2) Shock Tube

The purpose of the shock tube facility is simulating low-grade explosions and thereby creating shock waves that can be imposed on ventilation system components. The shock tube is 914 mm (36 in.) in diameter with a 11.2 m (36.9 ft.) driver section and a 35.4 m (116.1 ft.) driven section. The facility also includes other test devices that are used to evaluate filtration devices for explosive transients. The devices include the large shock tube, a small shock tube, and an aerosol efficiency test device.

(3) Scale-Model Ventilation Systems

The primary purpose of the two scale-model ventilation systems is providing system pressure and flow data for comparison with the TVENT tornado computer code predictions. The scale-model ventilation system test facility also will be used for future material transport and fire code verification studies.

(4) Wind Tunnels for Material Transport Studies

At present, no known computer code will handle the complex problem of modeling ventilation system pathways, predicting the energy propagation away from an event, predicting the flow of accident-generated gases and aerosol-laden air, and keeping track of material accumulation on the HEPA filters. However, much of the needed basic information that is not presently available can be obtained from aerosol transport experiments. In addition, the explosion (also fire and tornado) computer code needs basic material transport models.

Two wind tunnels are in use. One tunnel has a 0.5 by 0.7 m (20 by 28 in.) cross-section and a top speed of 18 m/s (60 ft/s). A second tunnel is under construction and will feature higher speeds, interchangeable sections, improved visibility, and a larger cross-section. The new tunnel will have a 1.2 by 1.2 m (48 by 48 in.) cross-section and a top speed of 46 m/s (150 ft/s).

(5) Facility Modifications for Fire Experiments

Fire and smoke tests can be performed using the existing filter loading facility modified to accept the smoke generator, with eventual testing to be continued by using the existing larger scale-model fire chamber, which has a volume of 17.4 m^3 (613 $ft.^3$) and has 50.8 mm (2 in.) thick steel walls connected to 0.6 by 0.6 m (2 by 2 ft.) ducts.

7.3.2 Computational Facilities

The Los Alamos National Laboratory (LANL) has been a leader in the development of computational methods that utilize large computers. Besides internally developed special purpose software, LANL is using several well-known general purpose codes for reactor safety programs.

Five engineering structural analysis codes currently being used on reactor safety programs are ADINA, ADINA-T, NONSAP-C, SAPV and SPAR. The NONSAP-C code is a severely modified version of the NONSAP code directed at the strength and creep analysis of concrete structures. The modifications were done at LANL.

These finite element codes run on either in-house CRAY or CDC-7600 computers. Currently LANL features four CRAYs, four CDC-7600s and one CDC-6600, a common file network for moving data and codes between machines, and state-of-the-art common graphics systems. The use of this unique facility for engineering analysis in LWR safety programs has not as yet reached its full potential.

7.4 Areas of Future Interest Expressed by Laboratory Researchers

7.4.1 Category I Structures

LANL's review of the present status of knowledge concerning the behavior of reinforced concrete Category I structures, beyond their design limit, results in a continuing need for research in this area. Building on the background and capabilities that have been, or will be, established in the course of the present research effort, proposals have been made to conduct additional research in one or both of the following areas:

(1) Loading of Category I structures, beyond their current design limit, due to nearby explosions (sabotage). Most of the construction techniques, instrumentation, and analytical tools developed for the present contract could be used. Furthermore, LANL has extensive facilities and know-how in the field of exposives testing.

(2) Interaction of the structure and large equipment items when either is loaded beyond its design limit. Large pipe breaks, vessel rupture, or partial structure failure all impose loads which may involve nonlinear behavior of connecting and support points. The ability of racks, support walls, hangers, etc., to survive, and hence, limit damage, is of great importance. Facilities should be developed to experimentally investigate the problems; these efforts could build upon a considerable amount of existing facilities.

7.4.2 Steel Containment Buckling

The analytical program for nonlinear collapse analysis (buckling) predictions using the Finite Element (FE) method should be expanded. Current nonlinear codes are deficient in this area for one or more reasons. The proposed plan would incorporate features into the ADINA computer code (for example, Rik's method, etc.) that would allow confident predictions of collapse loads. Because of match-up of needed and available large capacity computing power for these problems and ADINA FE development experience, LANL feels it is uniquely qualified to carry out this research.

There should also be a development of a failure mode experimental program for stud-supported, steel-lined concrete containments. Today, it is possible to accurately calculate stress fields well into the nonlinear regimes for these containments (for example, Zion/Indian Point studies). The difficulty comes in trying to assess their influence. Biaxial effects, liner buckling from thermal effects, anchor support, and shear loads and weld effects would be incorporated into this program.

7.4.3 Fire and Explosion Effects

Work in the areas of fire and explosion effects on nuclear facilities are of major future interest at the Los Alamos National Laboratory. New initiatives in the area of cable tray fire simulation have been proposed.

A unique fire/explosion test facility is in the design stage at Los Alamos. This facility features multi-chambers with an interconnected ventilation system network. Thermal and material transport dynamics will be studied. This facility will also provide experimental verification for fire and explosion accident analysis codes that are under development.

7.5 Liaison With Other Foreign and Domestic Nuclear Safety Research Activities

With regard to foreign interest, LANL personnel believed that the Kajima, Ohbayashi-Gumi Construction organizations and Mitsubishi Heavy Industries of Japan would be interested in cooperative efforts and possible funding of containment buckling and design margin research applicable to both steel and concrete containment. Domestically there may be interest and support in the containment buckling program from the Chicago Bridge and Iron Co.

Karlsruhe Nuclear Research Center has used the LANL test facilities over the past several years to subject German-made filtration devices to pressure transients. Three test series have been performed and a fourth is planned for 1983.

The Los Alamos National Laboratory has been asked by Dornier Systems (a Germany company) to assist them in learning how to apply the LANL facility accident-analysis codes. LANL plans to work with Dornier in 1983.

LANL has participated in an international working group concerned with behavior of air-cleaning systems in nuclear facilities (reactor and fuel cycle). This group was sponsored by the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD). A final draft report has been prepared identifying needed research in the field of air-cleaning systems. Facilities fan testing under accident conditions are also identified in this report.

In April of 1983, LANL plan to host an OECD Specialist Meeting on Fire and Explosion Interactions with Ventilation Systems of Nuclear Facilities. Thus far, 40 papers have been accepted for presentation at this meeting.

8. OAK RIDGE NATIONAL LABORATORY

8.1 Organization

The Oak Ridge National Laboratory (ORNL) is a multidisciplinary, multiprogram facility. ORNL utilizes a matrix management system, and essentially all of the LWR safety research work is concentrated under the direction of the Nuclear Regulatory Commission (NRC) Program Office. The research work within the scope of this report is performed in the functional divisions listed below:

(1) Engineering Technology Division

- (a) Pressure vessel studies and experiments
- (b) Thermal hydraulic studies and experiments
- (c) Piping and structures evaluations
- (d) Structural design methods development

(2) Engineering Physics Division

- (a) Shielding studies and experiments
- (b) Probabilistic Risk Analysis

(3) Metals and Ceramics Division

- (a) Materials evaluation (Irradiation effects, ASME code support studies)
- (b) Nondestructive testing development and evaluation
- (c) Materials testing

Essentially all of the LWR safety research ongoing at ORNL is sponsored by the NRC. The only Department of Energy (DOE) sponsored work is that associated with the Three Mile Island Unit 2 (TMI-2) rehabilitation. There is also some EPRI sponsored research at ORNL, conducted mostly in conjunction with ongoing NRC sponsored programs. However, none of the EPRI sponsored programs are within the scope of this report.

An organizational chart of the NRC Program is provided as Figure 12. The program structure has the following features:

(1) Most of the NRC sponsored work at ORNL is performed for the NRC Office of Nuclear Regulatory Research. As shown in Figure 12, work is done for five separate divisions of the Office of Nuclear Regulatory Research (RES), and a program manager is assigned responsibility for each of the five categories of work.

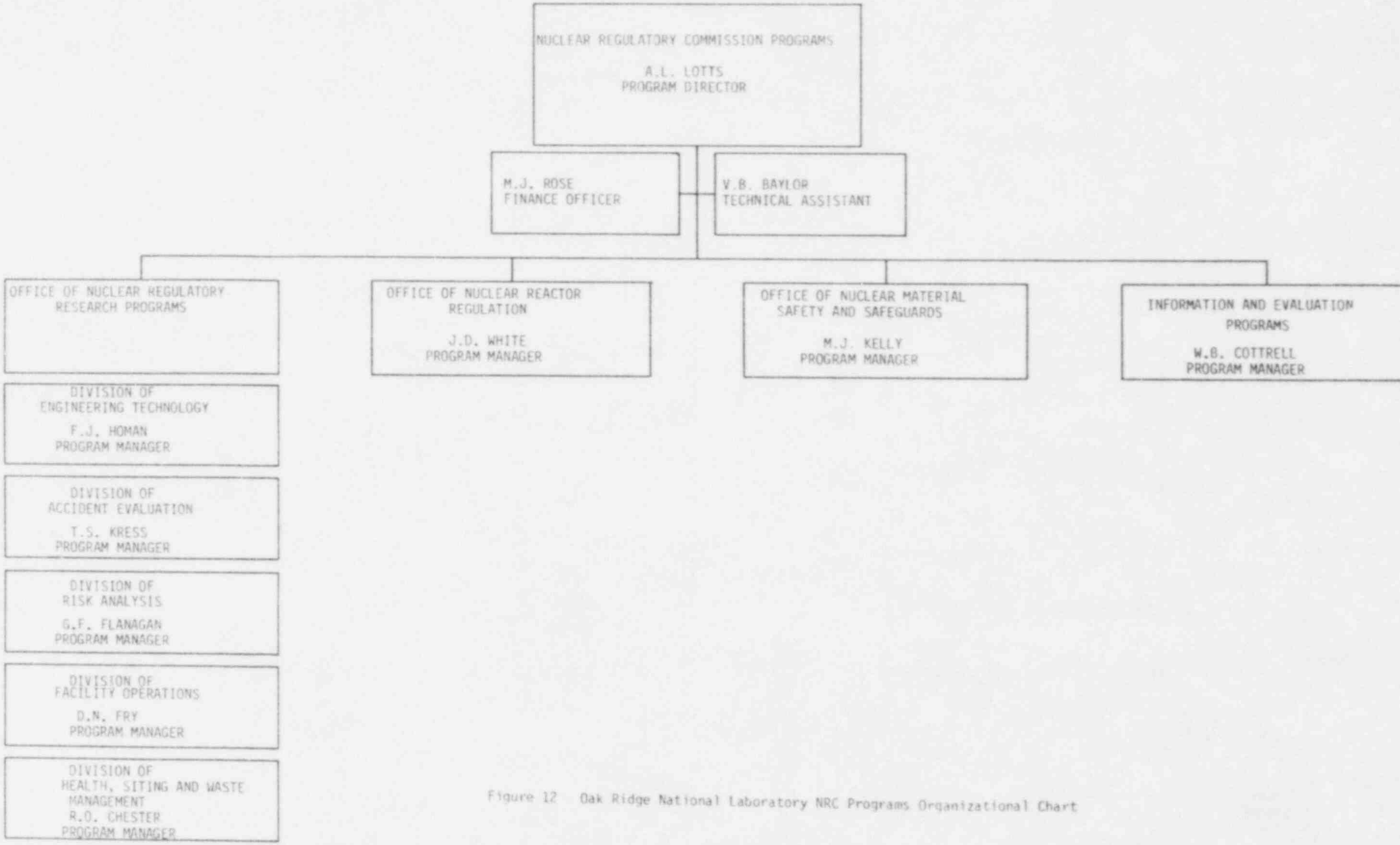

Figure 12 Oak Ridge National Laboratory NRC Programs Organizational Chart

(2) Work is sponsored by three other NRC Offices

   (a) The Office of Nuclear Materials Safety and Safeguards (NMSS)

   (b) The Office of Nuclear Reactor Regulation (NRR)

   (c) The Office of Information and Evaluation Programs (I&E)

Each of these categories of work also has a program manager, reporting to the Director of the NRC Programs.

8.2 Topic Areas of Interest

8.2.1 Department of Energy

As indicated in Section 8.1, the only DOE sponsored work at ORNL in the LWR Safety area is that associated with TMI-2 rehabilitation. This work is divided into two categories: work performed immediately after the accident, and work in progress now. However, none of this activity is in the topic areas of interest in this report.

8.2.2 Office of Nuclear Reactor Regulation

A complete listing of all the NRC-NRR Programs is included in the NRC Program organization chart (Figure 13). As can be seen from the Figure, there is a wide variety of activities. While space does not permit a detailed description of each program, some generalities can be made:

(1) The programs sponsored by NRR tend to be analytic rather than experimental

(2) The programs tend to be plant specific, or issue specific rather than generic.

(3) The programs tend to be short-term (1 to 2 years) in duration, and involve a review or evaluation of some issue of immediate concern to the NRC.

8.2.3 Office of Nuclear Regulatory Research

Again, a complete listing of all the NRC-RES Programs is included in Figure 14. The range of activities is quite wide, but in contrast to the NRR Programs, the RES Programs are predominantly experimental, and predominantly generic in nature. The following programs in the scope of this report are associated with the highest priority regulatory issues:

(1) Heavy Section Steel Technology (HSST) - pressurized thermal shock problem.

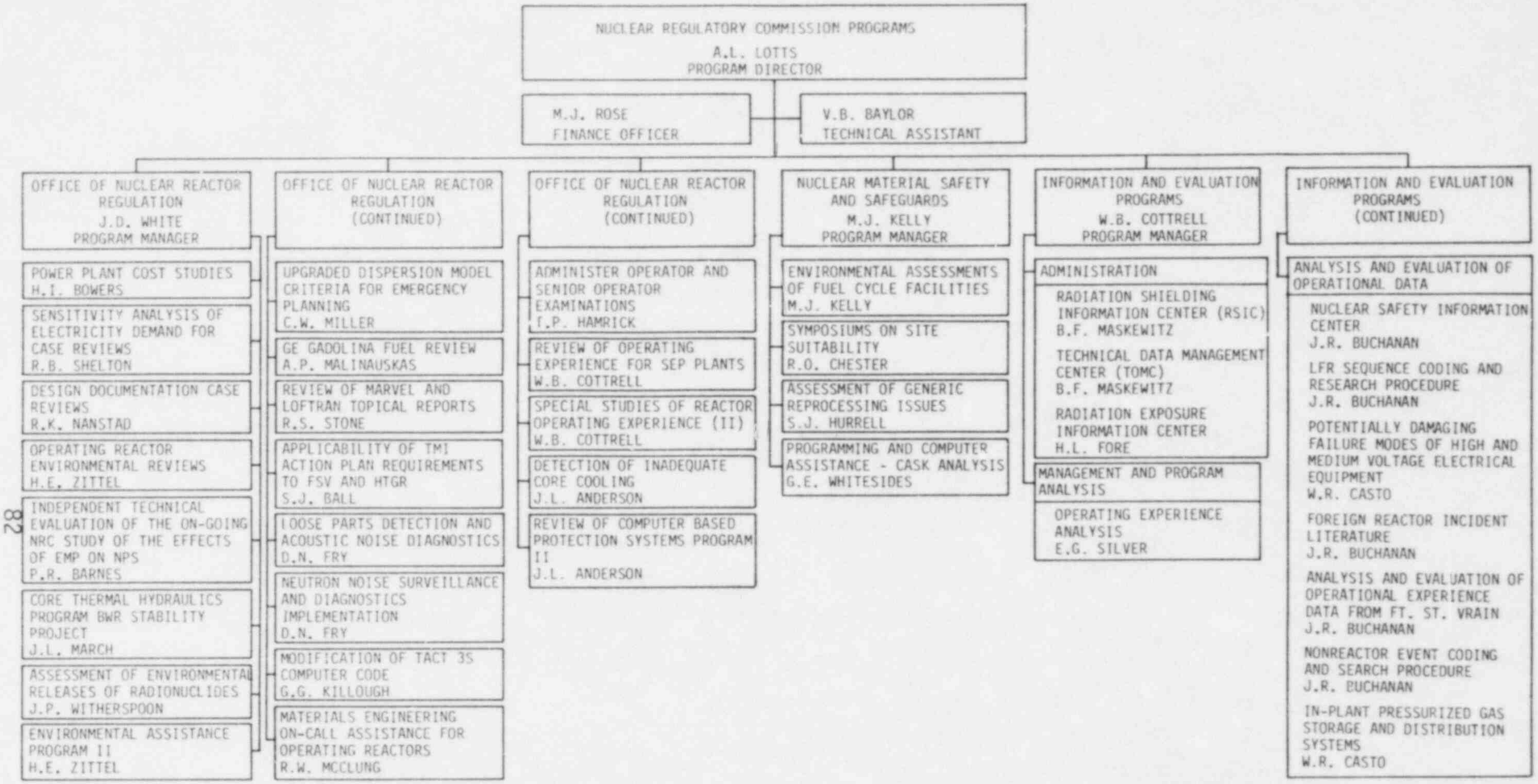

Figure 13 List of NRC-NRR Programs at ORNL

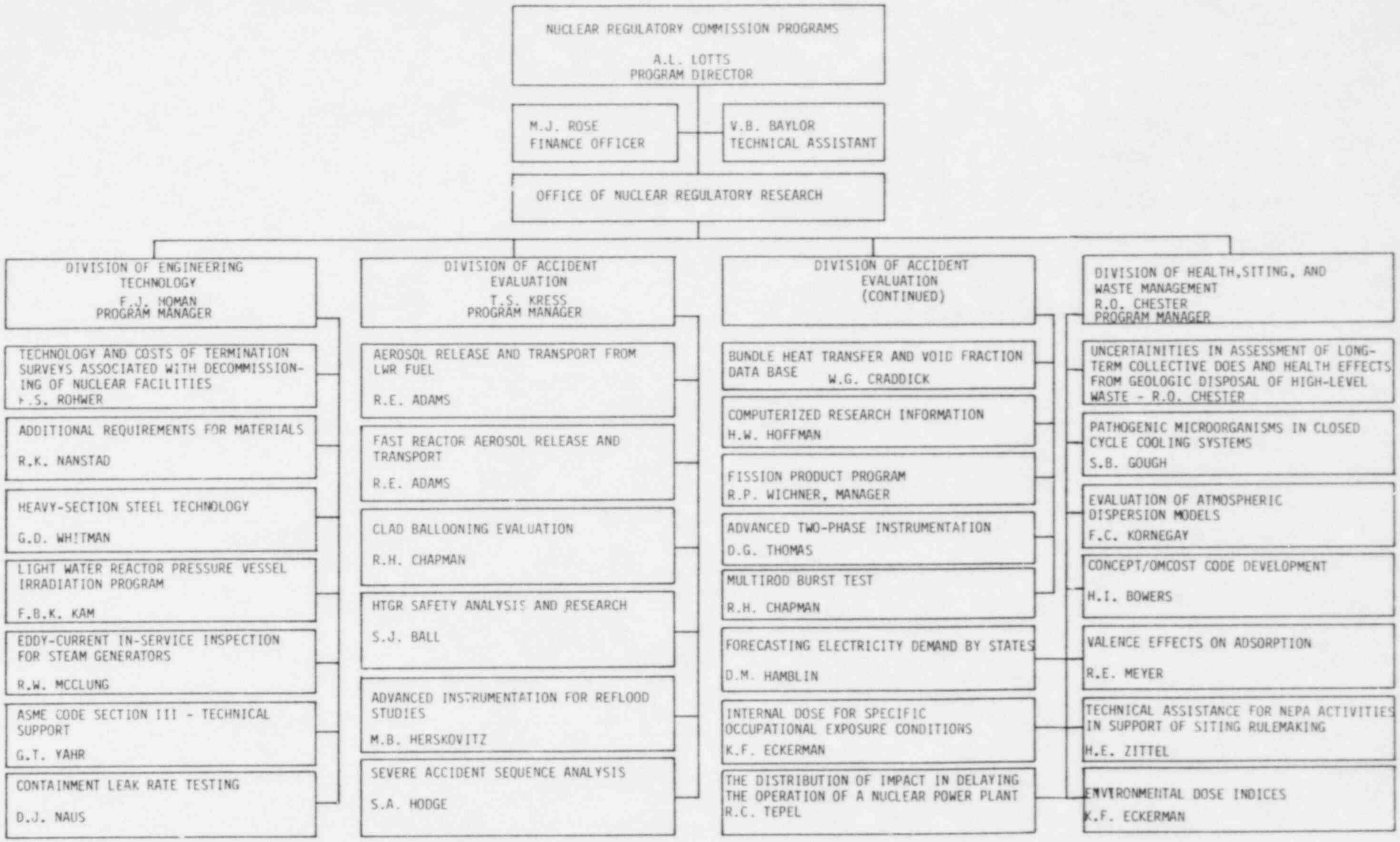

Figure 14 List of NRC-RES Programs at ORNL

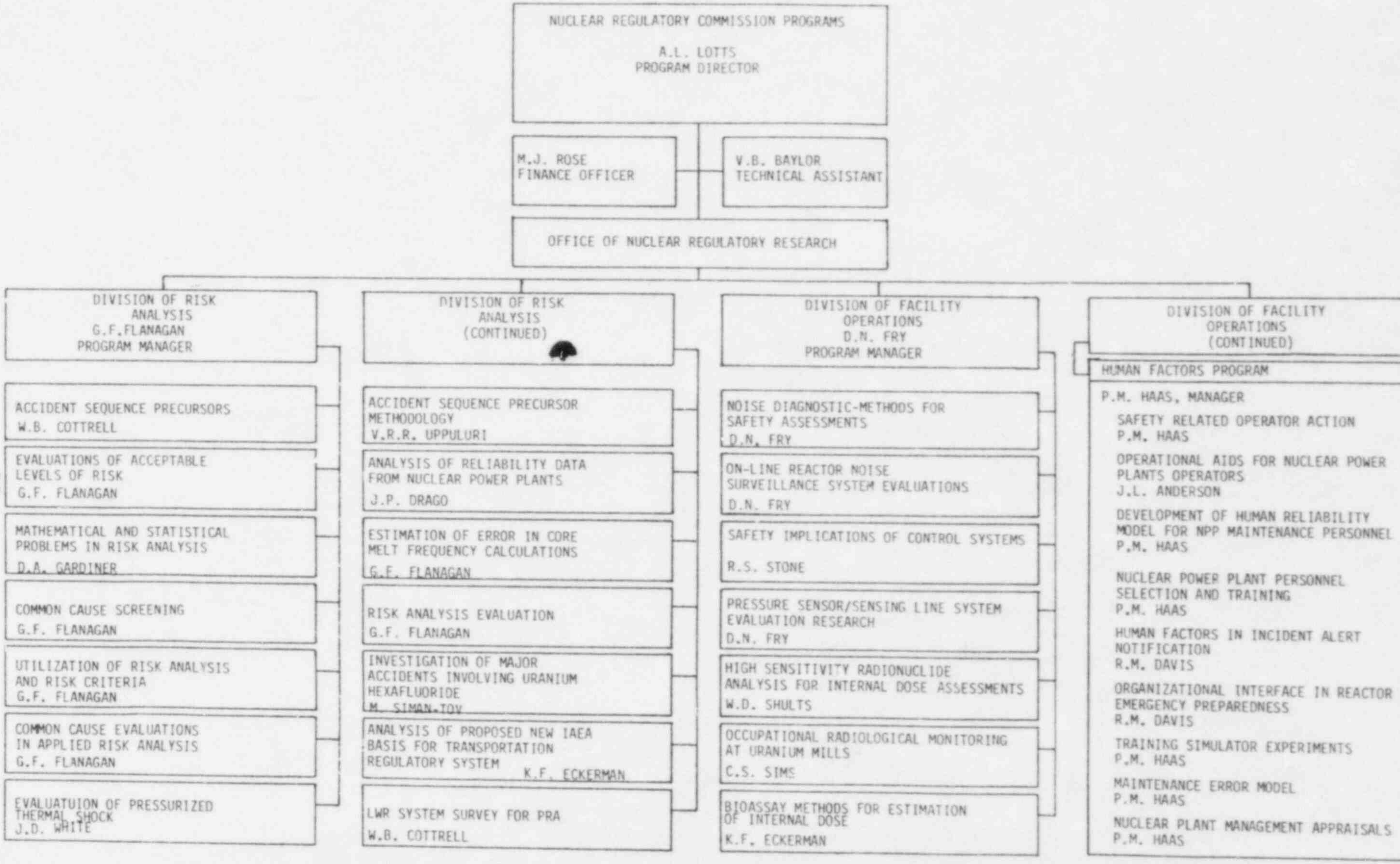

Figure 14 List of NRC-RES Programs at ORNL (continued)

(2) LWR Pressure Vessel Irradiation Program - embrittlement of pressure vessel steels due to neutron irradiation.

8.3 Facilities

8.3.1 Equipment

The facilities and equipment at ORNL which have been used in the past for LWR Safety Research (or are being used at present) within the scope of this report are summarized herein.

8.3.1.1 Reactors

There are six reactors at ORNL. The reactors which have been used for LWR Safety Research are as follows:

(1) The ORR (Oak Ridge Research Reactor) has been used to establish benchmark fields for development of LWR pressure vessel surveillance methodology. In the near future the ORR will be used for irradiation of LWR pressure vessel test specimens to more accurately quantify the effects of irradiation on material embrittlement.

(2) The PCA (Pool Critical Assembly) has been used to establish LWR benchmark fields in support of the LWR Pressure Vessel Irradiation Program.

8.3.1.2 Pressure Vessel Test Facilities

Thermal shock tests are conducted in a facility in Building 9201-3. Intermediate test vessels (8 in. thick) can be rapidly cooled from the inside to simulate an overcooling accident (OCA) in a light water reactor vessel without pressurization.

Pressurized thermal shock tests will be accomplished in a facility now being built in the abondoned power plant building at the K-25 site. This facility can simulate an OCA with the vessel under pressure.

8.3.2 Computer Hardware - Software

ORNL and the Oak Ridge complex have extensive computer capability, including large digital machines, analog computers, and hybrid computers. Some of the computer codes and applications being used in support of the current NRC Programs include the following:

(1) RELAP IV and V and RETRAN thermal hydraulic codes for accident analysis.

(2) Fracture mechanics codes: ORVIRT-2D, ORVIRT-3D, NOZ-FLAW, FMECH, OCA-1, OCA-2, and BIGIF.

8.3.3 Research Staff Capabilities

The staff currently working on the ORNL NRC Program are listed in Figures 12, 13, and 14. Their capabilities are associated with the projects listed in the organization chart, and the functional disciplines outlined in Section 8.1.

8.4 Areas of Future Interest Expressed by Laboratory Researchers

Each of the projects listed in Figures 13 and 14 has an assumed duration. Projects are being completed and added continually. Over the past couple of years the program mix has changed from predominently big programs associated with thermal hydraulic simulations, LOCA evaluations, rod burst studies, to smaller, more analytic projects. ORNL program management expect the following trends over the next several years:

(1) Completion of the current projects associated with pressurized thermal shock, steam generator NDE, and pressure vessel surveillance.

(2) New projects in the area of aging effects, including mechanical, electrical, and structural components. Both safety systems (called upon to act only under accident conditions) and normal operating systems will be included. Specific work will involve both components (pumps, valves, etc.) and systems. The overall goal of such work would be aimed at defining a qualified lifetime for each component or system.

(3) Seismic analysis. Determination of the safety-related trade-offs between the static piping systems in use in current LWRs as compared with more dynamic (less-rigid) designs.

8.5 Liaison with Other Foreign or Domestic Nuclear Safety Research Activities

(1) Technical advisory role with NRC staff on Marviken experiment in Sweden.

(2) Resident Engineer in Germany. An engineer from the ORNL staff is on assignment at MPA in Stutgart, West Germany to provide technical liaison with the German pressure vessel development and testing program.

(3) Advanced Instrumentation of PWR Reflood Studies. ORNL has developed and is supplying special instrumentation for reflood experiments in Japan and West Germany.

9. SANDIA NATIONAL LABORATORY

9.1 Organization

The Sandia National Laboratory (SNL) light water nuclear safety research effort in the technical areas of interest covered in this report have been concentrated in Program Area 4400, Nuclear Fuel Cycle Programs with A.W. Snyder, Director, as shown in Figures 15 and 16. Within this program area, the LWR Safety Department consisting of approximately 60 persons has taken a lead role with G.R. Otey as Department Manager.

SNL, unique among the laboratories and topic areas considered in this report, supports both DOE-funded Light Water Reactor Safety Technology Programs as well as NRC-funded nuclear safety research programs. Under the sponsorship of the DOE Nuclear Development Division, the DOE Program is intended to reduce the probability and consequences of reactor accidents by developing and assisting implementation of Light Water Reactor (LWR) systems and concepts which offer significant improvements to safety.

9.2 Topic Areas of Interest

9.2.1 Department of Energy LWR Safety Program

Research efforts sponsored by the DOE in the areas of interest related to this study are as follows:

1) Support of Research Being Performed by the Pressure Vessel Research Committee of the Welding Research Council

2) Analytical and Experimental Study of non, self-similar fatigue crack growth being performed at the Institute of Fracture Mechanics at Lehigh University

3) Evaluation of Failure of Safety Class 1 and 2 Relief Valves

4) Evaluation of Standard LWR Floating Plant at an Inshore Site

Details of these DOE projects can be found in the SNL Light Water Reactor Safety Technology Program Quarterly Reports.(Ref. 3)

9.2.2 NRC Funded LWR Safety Program

The NRC sponsored research in the technical areas related to this study are listed as follows:

1) Hydrogen Behavior inside Containment

2) Two-Phase Jet Load Prediction on Target Structures

3) Structural Safety Margins for Containment

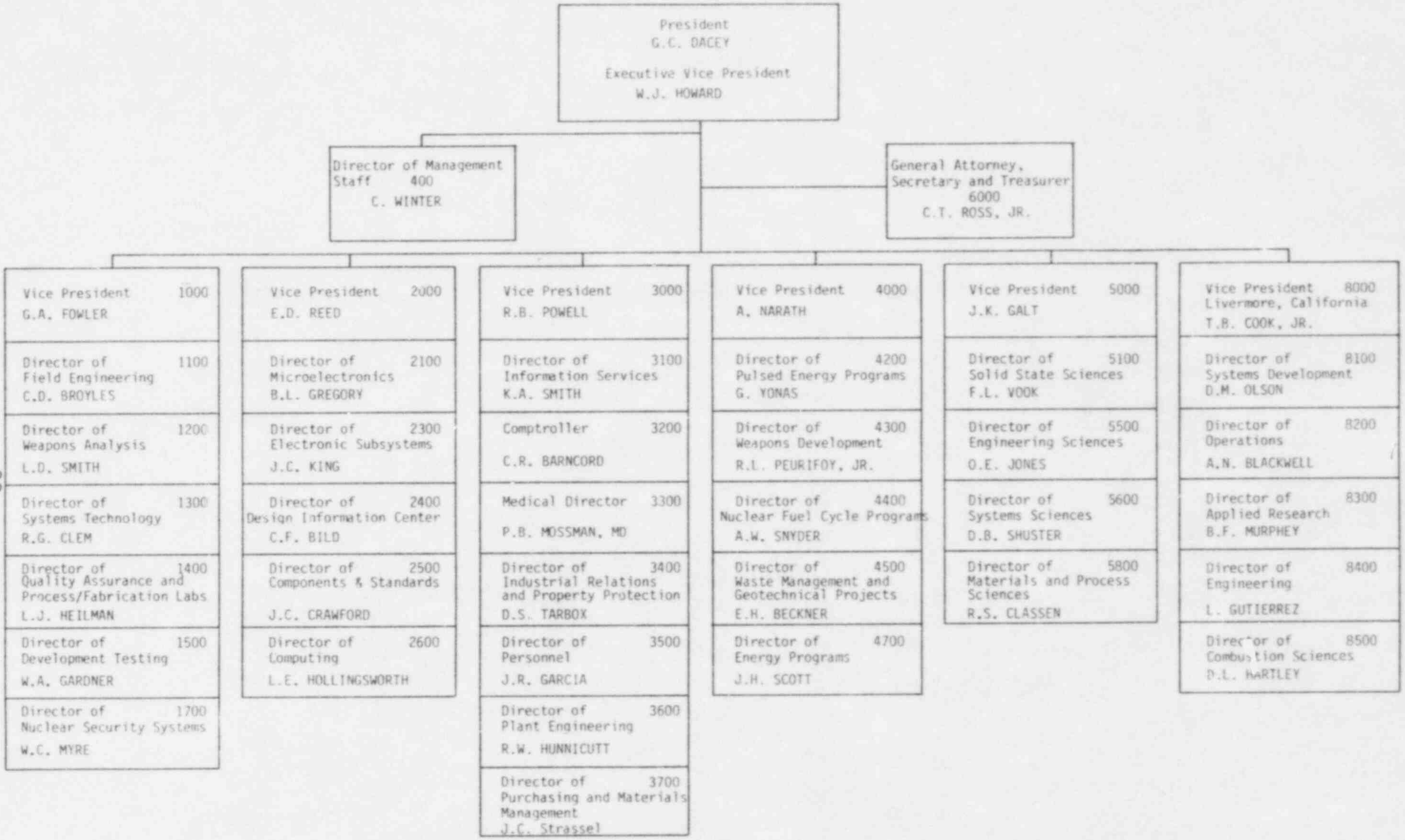

Figure 15 Sandia National Laboratories Organizational Chart

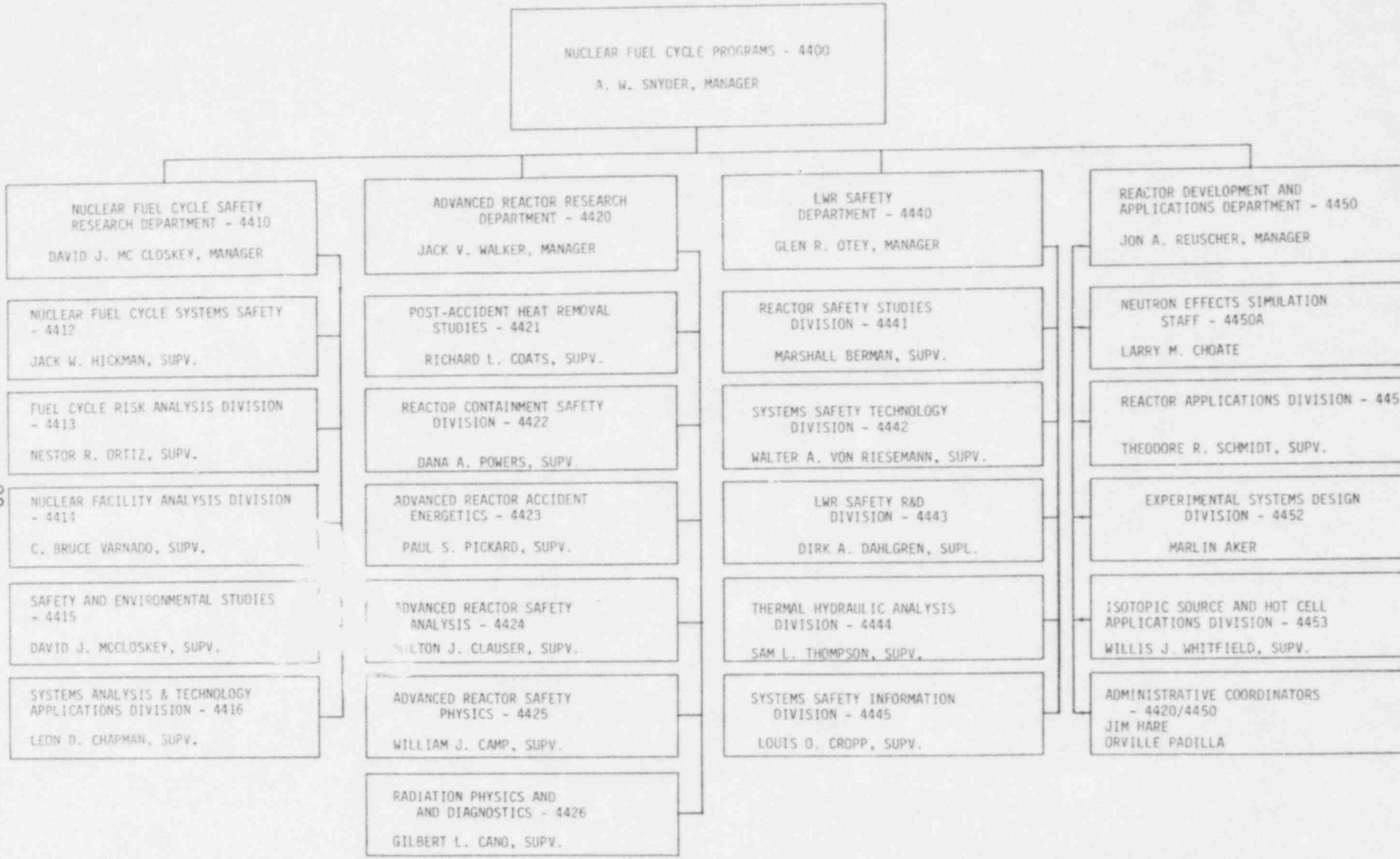

Figure 16 SNL Nuclear Fuel Cycle Programs Organizational Chart

9.3 Facilities

9.3.1 Experimental

9.3.1.1 Pneumatic Actuator Facility

Mechanical shock tests can be conducted using the pneumatic actuator facility to produce a programmed acceleration loading profile that simulates the environment resulting from blast loading. The actuator is used in the thrust column driven sled mode for this test series. This mode consists of an air-driven actuator applying a force through a piston-driven thrust column to a sled mounted between two rails 610 mm apart. The maximum force that can be applied to the sled is 2.45 MN with a maximum stroke of 0.9 metre. Velocities available range from 80 m/s with a 115-kg sled to 40 m/s with a 900-kg sled. Accelerations up to 2000 x gravity may be obtained using this sled mode.

To achieve higher accelerations or a short-duration shock pulse, the actuator can be used in a two-sled impact test mode. In this method the actuator imparts a velocity to the first sled which impacts a second sled via a mechanism that controls the forces between the sleds. Up to 10,000 x gravity can thus be applied to a test specimen on the second sled. Other test methods involving multiple sleds and force programmers have been developed to simulate various field environmental conditions.

9.3.1.2 Centrifuge Impact

The energy-absorbing capabilities of insulation materials can be evaluated using the 10.7 metre-radius centrifuge. A test specimen is secured to the end of the centrifuge arm by a cable. When the centrifuge is stabilized, with the specimen traveling at the desired impact velocity, the cable is cut, allowing the specimen to travel freely into a 14,000-pound steel target. Hardwire instrumentation is used to measure deceleration forces on the specimen during impact.

The facility can provide impacts from 10 to 550 ft/s, with a maximum velocity of 210 ft/s on a 10,000-pound test specimen.

9.3.1.3 Blast Testing

The effect of a blast-wave environment enveloping a structure (tank, car, building, re-entry vehicle) can be studied using a shock tube to produce the desired shock wave. One such tube is 1.8 metres in diameter and 61 metres in length, with a driving charge of high explosives at one end and the experiment at the other. Detonation of the explosive produces an aerodynamic shock wave which moves down the tube to envelop the experiment. In a typical experiment, a charge of 150 kg of PETN explosive in the form of primacord is used in the shock driver section at a loading density of 11 kg/m. At this density the explosive energy is contained by the driver.

Blast overpressures of 1.4 MPa and 0.5 MPa can be generated in the 1.8-m and 5.8-m test sections, respectively. Other shock tubes up to 2 m in diameter are available for generating blast pressures up to 4 MPa. These, however, are driven by larger explosive charges with energy densities that cannot be contained. Thus, a driver section must be expended with each test.

9.3.1.4 Random-Vibration Testing

Since random-vibration testing is computerized for improved control accuracy, the time required for preparing a test is reduced. Side advantages of considerable significance are the near real-time data analyses and the power spectral density plots obtained. Control spectra may be of any form that can be represented by straight-line segments and may range from 0.5 Hz to over 3 kHz with rms forces of up to 155.7-thousand Newtons (35,000 pounds) and displacements to 200 mm (7.87 inches) at the low frequencies. The computer system functions with up to 1024 discrete frequency lines in either pseudo or true random mode. Capability is available on the minicomputer system to perform spatial modal analysis resulting in eigenvector or eigenvalues at monitored points. This technique allows development of mathematical models of subsystem package.

9.3.1.5 Vulnerability of Structures to Airborne Objects

A test facility used to evaluate the vulnerability of structures to tornado-borne objects has been completed and evaluated. Among objects successfully thrown at the structure are 12-foot x 4-inch x 12-inch wooden planks at 200 mph a 3-inch diameter pipe 10 feet long at 100 mph, and a 3000-pound car at 50 mph.

The capability of the facility includes subjecting concrete slabs 17 x 17 feet square, and up to 24 inches thick, to the impact of tornado-generated missiles of any variety. For instance, telephone poles at velocities of 150 mph, pipe to 300 mph, planks to 300 mph, etc., can be accelerated into concrete slabs or any other material.

9.3.1.6 Dynamic Mechanical Properties

A plate-impact test has been developed, together with high-resolution instrumentation, to provide quantitive data for material response at the highest strain rates that can be obtained by propellant-driven gun facilities which permits launching of flat flyer plates for planar impact against stationary target plates. Quartz stress gages or laser interferometers capable of nanosecond time resolution allow measurement of the fine structure of loading shock waves and unloading release waves from which dynamic yield strengths, relaxation spectra, polymorphic phase-change kinetics, melting kinetics, high-pressure equations of state, and dynamic spall fracture data are deduced.

A variety of ultrasonic test facilities provides routine sound-speed measurements over wide ranges of temperature and pressure. Both pulse echo and interferometric methods can be used. In addition, facilities have been developed with the wide frequency and high powers necessary to accurately determine dispersion and absorption curves for lossy materials. These techniques have been used successfully with porous materials, polymers, and laminated and fibrous composites.

Current facilities include the following:

- Gun driven flyer-plate facilities
  - Impact velocities to 5 km/s
  - Strain rates to 10^8/s
  - Shock pressures to 1.0 TPa
  - Shock temperatures to 10^4 K
- Ultrasonic facilities
  - Low temperature to 1 K
  - High temperature to 600 K
  - Static high pressures to 3 MPa

9.3.1.7 Static Mechanical Properties

Facilities have been developed for testing materials under states of combined stress. Static or low-rate tests involving biaxial stresses (via tension or compression), external or internal pressurization, and torque loading of tubular specimens are applied to materials such as structural metals to determine combined stress-yield surfaces and to composites to determine combined stress-failure envelopes. Static or low-rate confined compression tests, involving so-called triaxial loading within a static high-pressure vessel, have been applied to measure failure envelopes and strength of rocks. Both biaxial and triaxial machines have computer feedback control, allowing sophisticated programming of loading paths as well as direct computer-controlled data acquisition and reduction.

Current activities include the following:

Biaxial facilities
- 800 kN end load
- 70 MPa pressure
- 8 kNm torque

Triaxial facilties
- 1.8 MN end load
- 1.0 GPa pressure

9.3.1.8 Thermal Mechanical Properties

Test capabilities exist for determining all conventional mechanical properties and some unconventional ones. Creep and stress relaxation tests can be made on metals, polymers, ceramics, and composites under controlled temperature and atmospheric environments. Conventional uniaxial stress-strain properties can be obtained at temperatures from that of liquid nitrogen to the melting points of refractory metals. In addition, feedback-controlled, time-dependent, thermo-mechanical environments can be applied to electrically conducting materials during which mechanical properties can be obtained as a function of the history of temperature and stress or strain.

Stress-corrosion cracking tests are performed in a variety of environments with controlled humidity and in pressurized liquid. These tests can also be used in fracture-mechanics studies except for molten salts and pressurized liquids.

Current facilities include the following:

Uniaxial loading facilities
- Load - 800 kN
- Strain rate - 10^{-5} to 10^{2}/s
- Temperatures - 65 to 3000 K
- Pressures - 10^{-6} Pa to 100 MPa

9.3.2 Engineering Analysis

Capabilities have been identified at SNL in the following engineering analysis areas which may have application to nuclear safety research.

(1) Static Stress Analysis

(a) Thermal stress analysis
(b) High temperature creep

(c) Plastic collapse
(d) Pressure-vessel analysis

(2) Shock and Vibration Analysis

(a) Mode shape and frequency determination
(b) Shock spectra
(c) Lumped parameter models
(d) Non-parametric models

(3) Seismic Studies

(a) Near field transmission
(b) Media-structure interaction
(c) Primary structure response

(4) Response of Structures At and Beyond Failure

(a) Brittle fracture
(b) Ductile fracture
(c) Crush

(5) Analytical Technique Development

(a) Time integration procedures
Stability
Frequency shifts
Damping

(b) Constitutive modeling
Finite strain plasticity
Cyclic plasticity
Combined creep and plasticity
Cumulative damage

(c) Nonlinear static deflection algorithm
Tangent modulus
Initial modulus
Approximate tangent modulus

(d) Mesh generation
3-D interactive graphics
2-D self-organizing

(e) Finite strain transient response

9.4 Areas of Future Interest Expressed by Laboratory Researchers

SNL personnel have expressed a particular interest in expanding the Structural Safety Margins Program applicable to various types of Containment Structures.

9.5 Liaison With Other Foreign and Domestic Nuclear Safety Research Activities

A trip to Europe was taken by Dr. W. Von Riesemann in November 1982 to enlist European support for the Structural Safety Margins Program for Containment scaled model testing. Significant interest appears to have been developed in Sweden and the United Kingdom.(Ref. 4)

10. SUMMARY AND CONCLUSIONS

10.1 Introduction

In the U.S., nuclear safety research activities in the technical areas of interest related to this report are funded by a variety of sources. The vast majority of such research is funded by the U.S. Nuclear Regulatory Commission, primarily the Office of Nuclear Regulatory Research. The Office of Nuclear Reactor Regulation also contributes a significant amount of funds. The next major source of funds is supplied by the Electric Power Research Institute. Additional resources are provided by the Department of Energy, Electric Utility Owners Groups, in-house research supplied by NSS Suppliers, the Pressure Vessel Research Committee (PVRC) of the Welding Research Council and in-house research by Architect Engineers. Foreign organizations engaged in safety research in the U.S. have not been significant sources of funding.

Because the organizations surveyed generally allocate their funds for a broad scope of research activities, it is very difficult to accurately define funds spent on research activities in the areas of interest related to this study. Moreover, it is difficult to distinguish between funds spent on water reactor research and funds spent on sodium breeder reactor safety research. Similarly, it is difficult to determine funds allocated to product development as opposed to funds directed to safety research. A rough breakdown of funding per year in 1982 for the organizations surveyed in this report is as follows:

(1) U.S. NRC - $\$70 \times 10^6$

(2) Electric Power Research Institute - $\$8 \times 10^6$

(3) Department of Energy - $\$5 \times 10^6$

(4) Electric Utility Owners Groups - $\$4 \times 10^6$

(5) NSS Suppliers - $\$3 \times 10^6$

(6) Pressure Vessel Research Committee - $\$0.5 \times 10^6$

(7) Architect/Engineers - $\$0.5 \times 10^6$

10.1.1 NRC Research Objectives

Within the scope of this report, the major objective of the NRC research program is to provide the understanding of phenomenology and the verified analytical methods to permit identification and realistic (or best-estimate) analysis of important accident sequences and their consequences. To this end, the NRC research program consists of a mixture of experimental work and code development work that is aimed at understanding a complex system of input and output transients. In order to obtain improved cost effectiveness, the emphasis in planning future research focuses on smaller-scale experiments providing data that are applied to nuclear plant safety through carefully scrutinized analyses using thoroughly checked codes. Large, complex, inherently atypical facilities tend to yield few data points, and data points of doubtful applicability. Additional major NRC research objectives are to provide the methodology to make more effective use of probabilistic risk assessment in the regulatory process and to improve confidence in the data base for risk assessment. This combination of experiments, code work, and risk analyses determines significant factors that should be taken into account in the regulatory process.

The pre-TMI regulatory practice of attempting to characterize a wide range of possible accidents with one or several "limiting case" or upper bound sets of accident assumptions not only may lead to unwarranted conservatism but also may result in design requirements that inhibit normal operation and the ability to cope with actual accidents or precursors of potential accidents. The result of such practice may be a reduction in overall plant reliability and safety.

The current NRC research effort is directed toward several end uses:

(1) Better understanding of input parameters particularly those associated with extreme loads, system response and safety margins provided by plants in real-world accidents and transients,

(2) Better use of probabilistic risk analysis with better ability to differentiate relative safety significance of regulatory issues or prospective requirements, and an

(3) Understanding of a sound basis for the design of accident mitigating systems.

Areas of particular research emphasis are identified as follows:

(1) Small-scale specific tests of particular accident or extreme load phenomena

(2) Applications of codes (computer) to regulatory problems

(3) Improved safety system design and operation

(4) Evaluation of pressure boundary safety, operability of equipment, integrity of electrical connections, and structural integrity of aging plants

(5) Development of nondestructive examination techniques

(6) Application of risk-assessment techniques to better focus the regulatory process on safety issues

(7) Approaches to decay heat removal and improved containment systems.

Over the next few years, the NRC will devote a significant effort to support the reassessment of the regulatory treatment of severe accidents. This work includes studies of transients leading to fuel or cladding damage, the behavior of damaged fuel, fuel melt, fission product release and transport, and severe accident mitigation concepts.

Increased emphasis in the future will also be given to assessing relative priorities of regulatory activities and to developing further improvements and probabilistic risk assessment techniques that RES will apply to research planning efforts. The NRC will increase its use of such methods in the licensing review process. This effort will be augmented by increased use of systems' reliability analyses and evaluation.

In addition, the NRC has decided to develop a safety goal and related safety guidance with initial emphasis on individual and societal risks that might arise from the reactor accidents. The purposes of this project are to develop a general approach of risk acceptability and safety-cost tradeoffs and, to the extent possible, to specify qualitative safety goals and quantitative safety guidance and standards for review of rules and practices.

Also, the NRC has major concerns regarding the pressurized thermal shock problem, and future research of the NRC will concentrate heavily on this problem. The pressurized thermal shock problem is just a part of the overall primary-system-integrity area of concern. The NRC research effort is constantly analyzing other parts such as the conduct of an extensive steam generator tube failure research program and an evaluation of nondestructive examination techniques.

It should be understood that NRC-sponsored research is primarily aimed in two areas: developing a technical basis to support regulatory decisions, rulemaking; and the development of standards, and resolving generic safety issues. The nuclear industry and DOE also have a major responsibility to perform safety research to ensure that nuclear power plants and other nuclear facilities are designed and operated safely and reliably. There should be cooperation and coordination among the NRC, DOE, and the nuclear industry to ensure that the appropriate level of effort is directed to resolving safety issues and to preventing unnecessary duplication of effort. The RES technical staff attempts to stay abreast of work sponsored by the nuclear industry, DOE, and foreign organizations through meetings, discussions, and the exchange of information. The RES staff attempts to sponsor appropriate regulatory research that does not duplicate other efforts.

10.1.2 Department of Energy

The Light Water Reactor Research Act-PL 96-567 passed in 1981 requires DOE to establish a safety development and demonstration program for developing practical improvements in the generic safety of nuclear power plants during the next five years. Until recently, this program has been involved in an evaluation phase to determine what areas should be addressed and in what priority. It is expected that an implementation phase will be initiated in 1983 at which time close cooperation between NRC and DOE is expected to develop to assure project interfacing and coordination between the two safety research activities.

10.1.3 Electric Power Research Institute

In the area of nuclear safety research, the EPRI utility sponsors are primarily interested in those programs that assist them in the resolution of outstanding safety issues. For this reason, EPRI-sponsored research tends to respond to NRC perceived and identified safety related issues. EPRI-funded safety research tends to focus in the following areas:

(1) More realistic and less conservative definition of the phenomenon identified as causing a safety issued by the NRC,

(2) More realistic and less conservative definition of the consequences of a phenomenon identified as a safety issue by the NRC,

(3) More realistic and less conservative assumptions on parameters used in design and analysis then those permitted by the NRC, and

(4) More cost effective methods or computer algorithm used in design and analysis.

Safety research sponsored by the other organizations associated with nuclear industry except the PVRC tends to focus on specific issues affecting only a small number of facilities and is developed and administered on an ad hoc basis. PVRC programs tend to be developed on the basis of a broad industry concensus of technical personnel and usually center on perceived shortcomings of existing industry standards and to the better development of such standards.

10.2 Summary of U.S. National Laboratory Experimental and Computational Facilities and Future Interests

In reviewing the facilities and interests of the various National Laboratories there appears to be significant overlap between the various facilities. In general, this means there is more than one National Laboratory capable of performing a particular research program. In Table 4 is listed a summary of research interests identified by each National Laboratory plus a listing of the projects currently being sponsored by EPRI. In Table 5 is listed a summary of experimental and computational facilities available at each national laboratory and computational code development sponsored by EPRI for the research topics within the scope of this report.

Table 4 Summary of National Laboratory and EPRI Research Activities and Interests

A. National Laboratory	Siting	Structural Engr.	Mechanical Engr.	Material & Metallurgy
Argonne	o Environmental Impact Statement o External Blast Phenomena + Improve Hazardous Material Siting	o Blast Resist Des. o Backfill Desification o Evaluation of Dynamic Test Procedures + Nonlinear Seismic Analysis + Development of Fragility for Structures	o Vibration Monitoring Steam Generator o Reliability Assurance + Fluid Structure Interaction in S.G. + Piping Analysis Code Development + Fragility Development for Equipment + PRA based Design + Pipe Whip Code Development	o Cracking in LWR Materials + Improved Non Dest. Testing for Steam Generator
Brookhaven		o PRA Eval. of Structures o Develop Load Combinations Methodology for Structures o Develop Generic Floor Response Spectra o Inelastic Structure Analysis Code Development o Evaluate Ultimate Strength of Concrete Cont. o Develop Seismic Qual. Proc. for New & Oper. Plants	o Develop Generic Floor Response Spectra o Development Benchmark Problems Problems for Piping Anal. o Develop Fragility Curves for Equip. o Develop Seismic Qual. Proc. for New & Oper. Plants	
EG&G-Idaho	+ Simulation of E.Q. Effect at a Site	o PRA Evaluation of Big Rock Point + Simulation of E.Q. Effects on Real Structures + Study of Damping in Real Structures + Evaluate Seismic Scram in a Reactor Facility Subject to Seismic Strong	o Evaluate HDR Piping Analysis o Safety Relief Value Qualification o Evaluate Pipe Damping + Simulation of E.Q. Effects on Real Equipment + Evaluation of Active Seismic Restraints + Valve Qualification & Certification Prog. + Evaluate ASME Service Level D Criteria	o Review of Fracture Mechanics & NDE on the HDR + Evaluate Irradiation Effects on Annealing + Evaluate NDE Procedures + Implement Advanced Ultrasonic Testing Proc. + Improve Surface Examination Processes + Evaluate Pipe Welding & Repair Procedures
Lawrence Livermore	o Seismic Safety Margins Program o Seismic Scram o SEP Seismic Review o Seismic Source Term Evaluations	o Seismic Safety Margins Program o SEP Seismic Review o Evaluation of Design and Construction Errors in Plant Design + Improved Seismic Design Methodology + Nonlinear Soil Response	o Seismic Safety Margins Program o Load Combination Program o Seismic Qualif. of Aux. Feedwater Systems + Improved Seismic Design Methodology + Evaluation of Existing Seismic Qualification Test Data + Evaluation of Seismic Test Methods + Evaluation of Electrical Faults due to Seismic Excitation	o Failure Analysis of Materials

Table 4 Summary of National Laboratory and EPRI Research Activities and Interests (continued)

A. National Laboratory	Siting	Structural Engr.	Mechanical Engr.	Material & Metallurgy
Los Alamos		o Margins to Seismic Failure Program o Testing of Concrete Shear Walls o Seismic Response Analysis of Nonlinear One & Two Degree of Freedom Systems o Analysis of Concrete Cont. for Hydrogen Detonation o Testing & Analysis of Steel Containment & Openings for Buckling o Development of Dynamic Buckling Criteria + Ultimate Capacity of Structures to Resist Explosions	+ Primary-Secondary System Structural Analysis including Nonlinear Support Response	
Oak Ridge	o Methodology for Flood Risk Analysis o Evaluation of Atmospheric Dispersion Models	o Evaluation of Greased Tendons in Prestressed Concrete Cont.	o Acceptable Level of Risk Determination and Methodology o Common Cause Failure Modes o Evaluation of Reliability Data	o Fracture Mechanics Analysis & Investigation o Irradiated Material Analysis o Thermal Shock Evaluation o Pressure Vessel Integrity Evaluation
Sandia	o Evaluate Floating Nuclear Plant at a Shore Site o Evaluate External Blast Phenomenon	o Hydrogen Behavior Inside Containment o Two Phase Jet Load Prediction on Target Structures o Safety Margins in Containments	o Safety Relief Valve Failure Evaluation	o Crack Growth in Metals
Electric Power Research Institute		o Evaluate Soil Structure Interaction Methodology o Better Definition of Containment Load Carrying Capacity o Structural Reliability Methodology	o Development of Computer Codes for Better Analysis of (1) Water Hammer (2) Soil-Struct Int (3) Missile Impact (4) Fluid Struct Interaction (5) Transient Load Analysis (6) Pipe Whip (7) Large Deformation, Rotation & Strain Analysis Program o Improved Piping System Response Analysis o Evaluate Seismic Isolation Hardware o Better Definition Pipe Rupture & Depressurization Phenomena o Improved Jet Impingement Analysis o Better Definition Turbine Missile Impact o Better Definition of Piping and Supports Dynamic Capacity o Reliability of Piping and Fittings	o Define Crack Growth in Residual Stress Fields o Radiation Embrittlement o Estimation Techniques for Ductile Fracture o Review of Fracture Toughness o Irradiation Crack Arrest

Table 4 Summary of National Laboratory and EPRI Research Activities and Interests (continued)

A. National Laboratory	Siting	Structural Engr.	Mechanical Engr.	Material & Metallurgy
Electric Power Research Institute (continued)			o Simplified Piping Analysis Handbook o Component Requalification o Support Structure Reliability o Reactor Vessel Reliability	

o Research Areas Where Research Organization is Currently Active
+ Research Areas of Future Interest and Capability

Table 5 Summary List of Computational and Experimental Facilities Available at National Laboratories, Including EPRI Software Development

National Laboratory	Computational Software/Hardware	Experimental
Argonne	Software: o The REXCO code system, a series of finite difference lagrangian codes, particularly applicable to short duration, high intensity transients. o ICECO, the Eulerian counterpart to REXCO, which can handle long duration events. o STRAW, a finite element, 2-D fluid-structure code which has augmented third dimension capability. o ICEPEL, a hydrodynamic-structural analysis code for complex piping systems with elastic-plastic capability. o SHAPS, a hydrodynamic-structural analysis code for complex piping systems augmented with 3-D capability for treating flexural motions of pipe elements. o ALICE, an arbitrary Lagrangian-Eulerian, Implicit-Expliucit code. o NEPTUNE, a finite-element, 3-D, fluid-structure interaction code for analysis of in-tank components and top closures for large LMFBR primary sodium vessels. o SADCAT, a 3-D code for static or dynamic structural analysis augmented with thermal capabilities. o DYNAPCON, an axisymmetric code which is used to model reinforced or prestressed concrete vessels (or cells) to determine response to energetic core disruptive accidents. o SAFE/RAS, a finite element code used to analyze dynamic loadings on the upper internals located above the reactor core and reactor vessel head for LMFBR plants.	o Flow-Induced Vibration Test Facility (FIVTF) Cap: 8,000 gal/min o Lake Erie Extrusion Press Cap: 1,250 Ton o Lombard Extrusion Press Cap: 350 Ton o Harders Press Cap: 400 Ton o Lake Erie Fast-Acting Press Cap: 120 Ton Loomis Isostatic Press Cap: 30,000 psi o Arc Melting Furnace Cap: 6 in. o Electric Beam Melting Furnace
Brookhaven	Software: o a Soil Structure Interaction Code for Structures embedded in various soils o a Inelastic Structural Analysis Code o a Piping System Stress Analysis Code for ASME Class 1, 2 and 3 systems	o One 7,500 lb-force electro-magnetic shaker, variable or shaped input, frequency range 3-2,000 Hz. o Two 2,500 lb-force electro-magnetic shakers, variable or shaped input, frequency range 3-2,000 Hz. o Numerous hydraulic actuators, whose input can be programmed for all types of acceleration, velocity and displacement characteristics. The largest of these has a capacity of 250,000 lb-force and a stroke of 10 inches. A maximum stroke length of 36 inches is available on an actuator with a 75,000 lb-force capacity. o One 14 x 14 independent phase incoherent, random phase biaxial simulator 50,000 lb-force. o A long stroke phase coherent shaker (60 in/sec) 10,000 lb-force capacity. o Environmental testing chambers (large capacity) for temperature-humidity-aging testing. o Valve LOCA facility with controlled shock impulses.
EG&G Idaho	Hardware: o CYBER 176	o LOFT o PWR Test Facility o Low Impedance In Situ Dynamic Properties Testing
Lawrence Livermore		o Multiple-Actuator Hydraulic Shaker Facility at Site 300 Cap: 25,000 lbs Vel: 6.5 ft/sec Acc: 45 g Disp: $\pm$ 2.5 in

Table 5 Summary List of Computational and Experimental Facilities Available at National Laboratories, Including EPRI Software Development (continued)

National Laboratory	Computational Software/Hardware	Experimental
Lawrence Livermore (continued)		o Two Electrodynamic Shakers Cap: 30,000 lbs each 3-300 Hz Vel: 6 in/sec Disp: + 0.5 in o Electro Dynamic Shaker Cap: 36,000 lbs o 100 ft Drop Tower Cap: 4,000 lbs Vel: 80 ft/sec o Instrumental Hammer o Mechanical Properties Test Facility o Specimen preparation of large and small specimens; high-vacuum, high-temperature annealing and other special treatments are used to prepare specimens. o Tensile testing on the Instron universal testing machine from 0-2400 C under vacuum to 13 Pa (10^{-7} Torr). o MTS fatigue-crack growth for studying the effect of various atmospheres on fracture toughness. o High-vacuum creep at temperatures up to 2,400 C for studying the creep of materials under stress and pressures down to 0.13 Pa (10^{-9} Torr). o Computer control and data acquisitions for mechanical-properties tests, fatigue-crack monitoring, and remote testing. o Setup and staging for the in situ creep-testing unit used in studies on 14-MeV neutron radiation damage performed at the LLL RTNS-II neutron source.
Los Alamos	Hardware: o 4 CRAYS o 4 CDC-7600 o 1 CDC-6600 Software: o ADINA o ADINA-T o NONSAP-C o SAP V o SPAR Software: o EVENT o TVENT computer codes to simulate explosively driven transients o FIRAC code to predict fire spread within containment	o 5,000 Ton Static Press o Electrodynamic Shaker Cap: 20,000 lb 20-2000 Hz o Material Property Test o Stress Wave & Mechanic Impact Test o Blowdown Apparatus o Shock Tube o Scale Model Ventilation System o Wind Tunnel o Fire Experiment Test Facility
Oak Ridge	Software: o ORJINT-2D o ORVIRT-3D o NOZ-FLAW o FMECH o OCA-1 o BIGIF	o High Pressure Vessel Test Facility Cap: 50,000 psi pressure 50 ft diameter x 20 ft high
Sandia		o Pneumatic Actuated Shock Tester Cap: 2.45 MN force 0.9 meter disp. 80 to 40 m/sec vel. 2,000 g acc. o Centrifuge Impact Tester Cap: 10-550 ft/sec vel. 210 ft/sec for 10,000 lb test specimen o Blast Tube Tester o Random Vibration Tester Cap: 0.5 to 3,000 Hz 35,000 lbs + 4.0 in o Missile Impact Test Facility Cap: 3,000 lbs @ 50 mph 250 lbs @ 300 mph 17' x 17' x 2' target slab o Dynamic Material Properties Tester Cap: Imput Vel. 5 Km/sec Strain Rate 10^{-6}/sec Shock Press 1.0 TPa

Table 5 Summary List of Computational and Experimental Facilities Available at National Laboratories, Including EPRI Software Development (continued)

National Laboratory	Computational Software/Hardware	Experimental
Sandia (continued)		o Static Material Tester Biaxial - 800 KN Axial - 70 MPa Pressure Triaxial - 1.8 MN Axial - 1.0 GPa Pressure o Thermal Material Properties Cap: 800 KN Force 10^{-5} to 10^{-2}/sec Strain Rate 65 to 3,000° K Temp 10^{-6} Pa to 100 MPa Pressure
Electric Power Research Institute	Software: o STEALTH, a basic general purpose transient thermal-mechanical code (water hammer, missile impact, piping flow and fluid-structure interaction o ABAQUS-ND, a nonlinear dynamic analysis code for pipe whip and seismic analysis o ABAQUS-EPGEN, a general purpose program for analysis for stocastic loads and nonlinear response	

Most of the National Laboratory capabilities have been developed in support of national defense requirements, in development of new energy facilities, and not in the support of nuclear safety research. Given the current level of funding by the NRC and DOE it is highly unlikely that any substantial new experimental facilities will be developed, particularly now that the research emphasis has been focused toward simple separate effects' tests rather than integrated large-scale experiments. Thus, the trend toward simpler separate effects' tests, except perhaps in Japan with the development of the Todatsu Shaker Table, appears to be a world wide phenomenon aimed at getting more information or data for the research expenditure.

10.3 Summary of Foreign Experimental and Computational Facilities and Future Interests

Nuclear safety research in countries outside the U.S. are generally conducted as well as supported by government national agencies. In most well-developed countries, research capabilities are viewed as part of a country's technological resources; therefore, they are supported as a matter of national policy. As a result, nuclear safety research activities focus on existing or planned research capabilities as well as perceived research needs. The technical areas in which countries direct and emphasize their research activities are influenced both by the degree of risk perceived by the regulatory authorities responsible for safety research and by the research resources in terms of facilities and personnel available.

An exception to the general rule of nationally supported and conducted research can be found where unique research facilities exist. In these instances, bilateral and multinational sponsorship of research activities is apparent. Multinational sponsorship of nuclear safety research is also evident in areas where the scope of the research can be shared, or there is some hope of research reciprocity between countries. Thus, if the research budgets of one country support research performed in a another country, there is some expectation that the second country will in turn support research performed in the first country. Such sharing of research cost is much more likely to occur at the initiation of a new research activity. If the research has already been performed and funded by one country, there appears to be a substantial reluctance on the part of any other country to contribute financially to obtain the results of the research. In particular, the U.S. has been at a decided disadvantage in developing joint research programs with other countries. Details of most U.S. sponsored nuclear safety research programs are made public as a matter of policy. Thus, unless a formal agreement exists to share information, other countries do not see any advantage in sharing their detailed research results with the U.S. since comparable U.S. safety research data will generally be available to them.

Currently, safety research budgets in most developed countries are at best relatively constant and generally are not keeping pace with inflation. This reduction in research funding may be due to a growing feeling among nuclear safety research policy makers that the construction and operation of water-cooled and moderated power reactors of the BWR, PHWR, and PWR types is a maturing technology which no longer requires the level of safety research once considered necessary. During 1982, for the first time in both the U.S. and the countries surveyed, the percentage of water reactor power plants that were operating or at least 90 percent complete exceeded 50 percent of the total number of plants committed. This shifting from a predominate design and new construction phase to an operating phase is certainly one manifestation of a maturing industry.

Theoretically, reduced research funding should increase the potential for more multinational supported research activities since this practice should be seen as a means for funding research which otherwise would not be performed. However, the practical requirements of maintaining present national research facilities and personnel staff levels with existing or declining research budgets provide little incentive for spending part of that budget in the support of foreign research.

Moreover, it should be understood that most of the European countries already support the multinational nuclear safety research facility of EURATOM located at Ispra, Italy. As a result, these countries are reluctant to enter into other international agreements if the existing EURATOM facility is capable of accomplishing the desired research.

The current climate is such that multinational sponsorship of research is not likely to be achieved unless one or more of the following situations exist:

(1) Research must be performed in a unique facility not available in the foreign countries desiring to have the results of the research.

(2) The research is to be performed in topic areas where little expertise exists in the foreign country desiring to have the results of the research.

(3) There is an agreement for sharing the total scope of a particular project or reciprocal financial support of research between the countries involved.

In highly developed industrial nations with a strong interest in commercial development of their own national nuclear steam supply systems, it is unlikely that other than the first situation will be met. As a result, the climate for future cooperative bilateral or multilateral safety research involving the U.S. is probably better with somewhat less developed nations which have not developed the broad expertise or the facilities in the research topics covered in this report.

In Table 6 is a summary of recent research interests within the scope of this report of foreign nations with well developed safety research programs and an independent nuclear steam system supply. In Table 7

Table 6 Summary of Selected Recent and Future Safety Research Interests in Specific Foreign Countries

Country	Siting	Structural Engineering	Mechanical Engineering	Materials & Metallurgy
Canada	- Assessment of Small Aircraft Crash Probabilities in the Vicinity of Airports - A Stochastic Approach to Extreme Floods	- Behavior of Concrete Containment Under Overpressure Conditions - Hydrogen Behavior in Containments	- Periodic Inspection for Safety of CANDU Heat Transport Piping Systems - A Probabilistic Approach - Evaluation of Seismic Equipment Qualification	- Ultrasonic Sizing of Fatigue Cracks
France	- Comprehensive Approach of Seismic Risk, Safety Margins in Structures of Nuclear Power Plants - Methodology for the Calculation of Reference Earthquake Spectra Based Upon Physical Parmeters - Processing of Seismotectonic Data in France - Formation and Atmospheric Dispersion of Drifting Clouds of Explosive or Toxic Gases or Aerosols as a Consequence of an Accident on a Chemical or on a Nuclear Plant - Identification and Description of Extreme Natural Events - External Impacts on Nuclear Plants: Unconfined Chemical Explosions Due to an Industrial Environment or to Communication Routes - Studies of Environmental Characteristics of Sites with Respect to Safety Establishment Siting Criteria - Synthetic Seismic Standard Signal Studies - Compilation of Recordings of Near Field Motion and of Information on the Corresponding Damage - Instrumental Monitoring of Seismicity Surrounding Nuclear Sites	- Soil Mechanics the Vicinity of Installation and Soil-Structure Interaction in Seismic Safety - Seismic Analysis of a Nuclear Power Plant Soil Structure Interaction - Mechanical Factors Affecting Reliability of Structures - Overall Behavior of Prestressed Concrete Containment Subjected to Internal Pressure Greater than the Design Value - Primary Shield Wall Behavior of PWR's in Case of Restricted Pressure Vessel Rupture - Local Behavior of Reinforced Concrete Walls Under Hard Missile Impact - Problem of Rare Events in the Reliability Analysis of Nuclear Power Plants (PWR) - Containment Behavior Beyond the Design Conditions - Behavior of Typical Structures Under Seismic Excitation Shake Table Tests - Evaluation of the Effect of Piles on Containment During Earthquakes	- Behavior of Piping in the Event of an Accident - Failure Probability Calculation of a PWR Vessel - Qualification Tests of Reactor Safety-Related Class 1E Equipment on the Expected Range of Normal Service Accident and Post Accident Conditions - Electrical Characteristics Evolution of Class 1E Equipments After Aging and Accidents Simulations - Flow Induced Vibrations in Pipes - Nuclear Reactor Vibratory Characteristics Tracking - Evaluation of 3 S_m Code Limit Applied to Secondary Stresses	- Accelerated Aging Nuclear Qualification Tests of Polymeric Materials - Cumulative Law for Fatigue Damage Under Complete Loadings - Fracture Mechanics Apply to Ductile Materials (Stainless Steel) - Analysis of the Performance Neutron Irradiation of Improved Steel to Contain a Nuclear Reactor under Pressure - Crack Arrest Methodology for Nuclear Pressure Vessel Steels - Research Program on Irradiation Embrittlement of Pressure Vessel Steels. AIEA Coordinated Program. - Toughness and Fatigue Behavior of Bimetal Welds - Corrosion Aspects of Nuclear Pressure Vessel Steels - Research Program on Irradiation Embrittlement of Pressure Vessel Steel - Ultrasonic (and Radiographic) Testing of Dissimilar Metal Welds - Ultrasonic Detectability of Flaws under Compression - Acoustic Emission Development of Equipment and Methods - European Program of Ultrasonic Testing of Heavy Section Steel Plates - Under Cladding Cracks Detection and Sizing by Eddy Currents Method - Study of Optical Filtering of Radiographic Films
Japan	- Evaluation of Turbine Missile Behavior at Nuclear Power Plant - Establishing Test Methods and Data Evaluation Procedure for Safety Analysis Using Wind Tunnel - Atmospheric Diffusion Under the Very Low Wind Condition for the Purpose of Estimation of Effluent Gases - Experimental Study on the Atmospheric Diffusion under the Very Low Wind Condition for the Purpose of Estimation of Effluent Gas Diffusion - Research on Three Dimensional Ground Motions and The Response Analysis for Aseismic Design of Nuclear Power Plants - Develop More Rational Design Analysis of Nuclear Power Plants - Evaluation of Earthquake Motions on Rock - Research on Seismic Waves Measuring Data Collection, Sorting and Analysis - Researches on Seismic Resistance of Building Foundation and Evaluation of Characteristics of Soil Movement by Shock Waves for a Pilot Nuclear Reactor - Effects of Shock Waves to Underground Electric Power Plant - Measuring Methods of Soil Pressure Changes and the Data Application - Dynamic Characteristics and Dampening Characteristics of Silt having a Wide Range of Strains	- Structural Design of a New Type Reactor Steel Containment Subject to High Seismic Loads and Dynamic Buckling - Full-Scale Mark II Containment Response Test - Study of Sea Water Intake System in Tsunami - Evaluation of Dynamic Load to BWR Containment Under the Loss-of-Coolant Accident - Study of Prestressed Concrete Reactor Vessel (PCRV) Structures - Experimental Verification of Japanese Technical Code for Concrete Containment - Evaluate Behavior of Reinforced Concrete Containment Models Under the Combined Action of Internal Pressure and Lateral Force - Evaluate Behavior of Concrete Shield Wall Subjected to Thermal Stress and Earthquake - Seismic Analysis of Support Structure of Nuclear Steam Turbine - Effectiveness of Integrated Building Construction for Nuclear Power Plant Buidling Complex for High Magnitude Earthquake Zone - Nuclear Power Plant Design Isolating from Earthquake Shaking - Experimental Study on Strength of Steel Plates Subjected to Missile Impact - Research on Seismic Safeguard Design for Nuclear Reactor Building - Research of Building Structure using Theory of Absorbing Vibration Energy - Safeguard Design for Nuclear Reactor Containing Building Considering Dynamic Characteristics of the Foundation and Uneven Foundation - Dynamic Analysis of Anti-Seismic Design for Reactor Built on a Slope - Analysis of Shock Wave Transmission Through Soil Layer Utilizing 3-Dimensional F.E.M.	- Experimental Determination of Damping Characteristics of Nuclear Power Plant Piping Systems - Seismic Tests on Steam Generator Tube Bundle - Study on Structural Safety of Nuclear Pressure Vessel Against Internal Pressure - Seismic Test of Moving Equipment/Machinery in Nuclear Power Plant - Pressurizer Safety Valve and Relief Valve Performance Tests - Improve Dependability of Low Pressure Service Valves - Reliability Study of Pressure Boundary Components - Vibration Analysis Between Fluid and Liquid Container having Symmetrical Axis - Nonlinear Vibration Analysis for Piping System - Research on Energy Dampening on Piping System for Nuclear Power Plants - Seismic Proof Tests of Electronic Instruments - Improved Seismic Resistance Design of Nuclear Power Plant Equipment	- Experimental Study of the Safety Margin of the Simplified Elasto-Plastic Fatigue Analysis Method - "Nonlinear Fracture Mechanics Analysis and Experiment on Thermal Shock Behavior of PRV Plates" - The Experimental Study on the Safety of Heavy Section LWR Pressure Vessel Steel Plates - Develop Stress Corrosion Properties of Nuclear Grade 316 Stainless Steel - Study on Evaluation of Structural Safety for Nuclear Pressure Vessels by Acoustic Emission - Acoustic Emission Characteristics of Structural Materials - Post Irradiation Examination on the Surveillance Specimen from the Light Water Reactor's Pressure Vessel - Influence of BWR Reactor Water Environment on Fatigue Crack Growth and Fracture Behavior of Pressure Vessel - Neutron Irradiation Embrittlement of Steels for Nuclear Reactor Vessel - Analysis of the Behavior of Advanced Pressure Vessel Steels Under Neutron Irradiation - Remotely Operated Method of Ultrasonic Testing - Stress Corrosion Cracking of BWR Cooling System Pipe - Study of Crack Growth Under Cyclic Loading - Assessment of Neutron Irradiation Embrittlement on LWR Pressure Vessel Steels - Prevention of Stress Corrosion Cracking of Stainless Steel Piping for LWR's - Research on Residual Stress Reduction of Pipe by Induction Heating - Applicability of AE Techniques to Detection of Stress Corrosion Cracking (SCC) - Research & Development on Ultrasonic Testing System for Nuclear Power Plant

Table 6 Summary of Selected Recent and Future Safety Research Interests in Specific Foreign Countries (continued)

Country	Siting	Structural Engineering	Mechanical Engineering	Materials & Metallurgy
		- Simulation of Three Dimensional Earthquake Ground Motion Along Principle Axes - Research on the Relationship Between Soil Conditions and Nuclear Reactor Containing Building Construction - Time Phasing Dynamic Analysis for a Combined Building and Ground System by 3-Dimensional Thin Layer Element Model - Vibration Dampening Analysis for Prestressed Concrete Structure - Research on Nuclear Reactor Containing Building for Analytical Method of Earthquake Energy Discharging to Ground - Dependability of Concrete Container of Nuclear Power Reactor for Earthquake - Vibration Tests of Prestressed Concrete Container - Research on Anti-Seismic Dependability Analysis on the Major Nuclear Power Plant Structures - Safety Factor Check for Nuclear Power Plant Structures Against the Horizontal Force		- Post Irradiation Examination of the Surveillance Test Specimen from Reactor Pressure Vessel - Study on Dynamic Fracture Toughness of Nuclear Pressure Vessel Steel - Quantitative Processing of Metal Defects for Thick Plate for Nuclear Reactor Vessel - Proof Test on the Reliability of In-Service Inspection of Nuclear Power Plant Components
Sweden	- Seismicity and Earthquake Risk in Sweden	- Reactor Containment Modification: Project FILTRA Swedish Vent-Filter Conceptual Design Study - Design Considerations for Implementing a Containment Vent-Filter Plant at Barsebaeck, Sweden - Marviken Test-data Interpretations, Applicable to Pressure Oscillation and Spikes in BWR Containments	- Large Scale, Two-Phase Jet Impingement Experiments at Marviken - Evaluation of Crossflow Vibration on Steam Generator Tubing - Pressure Vessel Integrity Under Fault Conditions	- The Effect of Strain Aging on the Toughness Pressure Vessel Steel for Nuclear Power Plants - Development of Ultrasonic Examination Method with Focused Search Units - Ultrasonic Examination of Austenitic Stainless Steel Welds for Intergranular Stress Corrosion Cracks - Residual Stresses in Weldments of Pressure Vessel Steels - Development of P-Scan Examination Method - Evaluation of Ultrasonic Testing in Austenitic and Inconel Weld Material
United Kingdom	- Development of Methods and Computer Codes for Seismic Analysis with Application to PWR's - Theoretical Studies of Gas Cloud Explosions: Mechanical Effects - Assessment of Seismic Hazard Within the United Kingdom - The Consequences of the Accidental Release of Toxic or Flammable Vapors to the Atmosphere - The Atmospheric Dispersion of Radioactive Material in the Event of an Accident to a Nuclear Installation	- Internal Missile Effects - Evaluate PWR Plant and Core Structural Dynamic Response - The Experimental and Theoretical Study Local Effects in the Impact of Missiles on Structures		- Development of Acoustic Emission Measurement - Corrosion Fatigue Crack Growth - Metal Fracture (Crack Growth) - Metal Fracture (Stability) - UK Contribution to IAEA Coordinate Program on Analysis of the Behavior of Advanced Reactor Pressure Vessel Steels Under Neutron Irradiation - Mechanics of Plane Strain Fracture - Fracture Behavior of Structural Steels - Corrosion Fatigue of Pressure Vessel Steels - The Characterization of the Microstructure and Embrittlement Susceptibility and Associated Effects on Upper Shelf Toughness of Pressure Vessel - Assessment of Upper Shelf Toughness of PWR Pressure Vessel Steels - Elastic/Plastic Fracture Toughness
Federal Republic of Germany	- Radioactive Release Control at Nuclear Power Plants Sites - Investigations of Remote Sensing Methods with Respect to Their Suitability to Measure Meteorological Parameters in the Atmospheric Boundary Layer - Atmospheric Diffusion Models for Particular Meterological Situations - Investigation on the Application of Non- steady State Dispersion Models for Reactor Accident Risk Analysis - Investigation on the Atmospheric Dispersion of Radioactive Substances in Local Range (up to 15 km Distance), Emission Height above 100 m - Investigation on the Atmospheric Dispersion of Radioactive Substances in the Mesoscale (more than 15 km distance) - Scale Influence on the Evaluation of Bench Test Results in Comparison with Real Explosions	- Development and Verification of Codes for Analysis of Dynamic Stresses and Deformations of LWR-Containments During LOCA - Reliability Assessment of the Secondary Containment of PWR - Experimental Investigation of the Hydrogen Distribution in the Containment of a Light Water Reactor Following a Loss of Coolant Accident - Energy Absorption Capacity of Reinforced Concrete Structural Members Under Impact Force - Experimental Studies Concerning Energy Absorption of Reinforced Concrete Members Subjected to Impact Load - Investigations on Safety Technology at the HDR Plant to Improve the State of Knowledge Concerning the Properties and Structural Behavior of LWR System and Components	- Comparison and Application of International Technical Codes-Represented by the ASME Boiler Pressure Vessel Code and Corresponding Rules in the Federal Republic of Germany - Experimental Data Acquisition and Processing of the Dynamic Behavior of the Pressure Vessel Tests Internals in the HDR Blowdown Experiments - Development and Verification of Coupled Fluid - Structure Dynamics Code for Analysis of Dynamic Stresses and Deformations of Reactor Vessel Internals During LOCA - Investigation of the Influence of an Overpressure Transient on the Safety of PWR Pressure Vessel Integrity - Experiments on the Critical Discharge from Pipe Leaks	- Supplementary Investigations on the Behavior of Reactor Concrete During the 4th Phase of a Hypothetical Core Meltdown Accident - Planning and Evaluation of the BETA Experiments - Material Investigations in the Framework of BETA Experiments - Experiments on the Interaction of Steel Melts and Concrete - BETA Facility - Erosion of Concrete by Steel Melts - Investigation of the Melt Front Velocity - Crack Arrest Behavior - Ultrasonic Testing of Austenitic Welding Joints - Development and Testing of Mechanized Welding Methods for Spherical Reactor Containments

Table 6 Summary of Selected Recent and Future Safety Research Interests in Specific Foreign Countries (continued)

Country	Siting	Structural Engineering	Mechanical Engineering	Materials & Metallurgy
	- Explosion Characteristics of Fuel-Air-Explosives as Standard of Comparison for Gas Explosions - Blast Wave in the Close Region of a Detonating Real Gas Cloud - Inflammability Conditions and Explosions Vehemence on Nonhomogeneous Gas-Air Mixtures - Analytical Determination of the Gas Explosion Pressure Field	- Reinforced Concrete Construction Loaded by Crashing Aircraft-Theoretical Utilization of the Experiments at Meppen Especially Considering the Behavior of Materials - Structural Behavior of Pressure Suppression System - Stress on Building Structures of Nuclear Plants with Complex Gas Explosion Loading Function - Stress on Structures of Nuclear Power Plants at Supposed Detonation of a Real Gas Cloud - Pressure Build up in Consequence of Hydrogen Deflagration - Theoretical Investigations on the Kinetic Bearing Capacity of Reinforced Concrete Slabs under the Impact of Strongly Deformable Metal Missiles - Explosion Possibility of Mist/Vapor/Air or Mist/Gas/Air-Mixtures - Experimental Investigation of the Hydrogen Distribution in a Model Containment (Preliminary Experiments II) - Investigations on the Nonlinear Behavior of Reinforced Concrete Structures under Seismic Loads - Analytical Activities of the GRS in the BMFT Research Program on Reactor Safety Standard Problems - Ultimate Bearing Capacity of Reinforced Concrete Slabs under Time-Dependent Loads (for example, Aircraft Crash) - Pressure Load in Consequence of Collapsing Steam Bubbles - PSS Structural Dynamics, Calculations - Analytical Activities of the GRS in the Research Program on Reactor Safety Pressure Suppression Systems and Condensation - Ultimate Bearing Capacity of Reinforced Concrete Beams Under Impact Load - Investigation of the Phenomena Occurring Within a Multi-Compartment Containment after Rupture of the Primary Cooling Circuit in Water-Cooled Reactors - Possible Initiation of Detonation - Like Explosive Modes in Free Gas Air Mixtures and Resulting Nuclear Power Plant Loads - Loads of Nuclear Power Plant Structures During Gas Explosions - Response of Nuclear Power Plant Structures to the Air Pressure Wave and the Induced Ground Wave Caused by Gas Cloud Explosions Similar to Detonations - Experimental Investigations to Determine the Pressure Field in Consequence of Interaction Between Pressure Waves and Building Structures - BWR Containment Pool Dynamics	- Investigation on Effects of Whipping Pipes on their Surroundings in Nuclear Power Plants - Noise Measuring on the Primary Circuit of the Cooling System in Nuclear Power Plants to Locate Leakage - Special Stress Conditions in Pipe Bends - Laboratory Experiments for Validation and Enhancement of Fluid/Structure Dynamics Codes to Initial Phases of LOCA - Design, Precomputation, and Evaluation of the HDR Blowdown Experiments on Dyanmic Loadings and Deformations of Reactor Pressure Vessel Internals - Investigations on the Failure of Core Support Structures - Analytical Activities - Rupture Phenomena with Vessels and Pipes - Loadings and Response of a Feedwater Line Due to Pipe Break and Ensuing Check Valve Closure - Investigation into the Phenomena Involved in the Depressurization of the Water-Cooled Reactors, Experiments Using a Steel Vessel 11.2 m in Height with Internals - Phenomenological Pressure Vessel Burst Experiments	- Low Cycle Fatigue Behavior of Steel, 10CrMoNinb 9 10, in Consideration of Hold Time - Determination of the Microstructure from Pressure Vessel Steels with Magnetic Induced Measuring Quantities - Improved Phenomenological Description of Acoustic Emission Signals and Their Analysis with Respect to a Better Evaluation of Defects - Multi-Eddy Current Testing - Investigations for Defect-Sizing with Acoustical Holography - Reconstruction by Signal Focus Curves - Development and Application of Signal Averaging Procedures Concerning and Ultrasonic Fault Testing of Coarse Grained Austenitic Materials and Welded Joints - Transposition of Fracture and Crack Propagation Characteristics from Specimens to Structures - Optimization of the Acoustic Linear Holography at Clad Components with Complicated Geometry - Study of Long-Term Stability of the Micro-Structure of the Steel 8 CrMoNiNb 9, 10 at Elevated Temperatures - Research in Analysis of the Behavior of Advanced Reactor Pressure Vessel Steels and Neutron Irradiation - Analysis of the Behavior of Advanced Reactor Pressure Vessel Steels Under Neutron Irradiation - Nondestructive Examination Under the PISC II-Program - New Methods for Examination of Reactor Components Using X - and Gamma Rays - High Speed Tensile Tests on Large Specimens, Stage II: Construction of a 12 MN High Speed Tensile Testing Machine, Experiments with Large Specimens - High Speed Tensile Tests on Large Specimens, Stage II: Construction of a 12 MN High Speed Tensile Machine, Experiments with Large Specimens - Irradiation Tests in the Framework of the Research Program Component Safety - Supplementary Evaluation of Safety Related Estimation of the Results Concerning the Research Project Safety Components, Project Part Materials Mechanics Research - Onward Development of Nondestructive Testing Methods for In-Service Inspection of Reactor Power Plants - Basic Inspection and In-Service Inspection of Austenitic Reactor Vessels and Components with Ultrasonic Methods

Table 7 List of Selected Foreign Experimental Research Facilities Suitable for Multinational Sponsored Safety Research

(1) Cadarache Pipe Break Test Facility - France -
Test of Pipe Whip and Break Phenomena

(2) Canadian Westinghouse Environmental Test Facility - Canada -
Containment Environmental Simulation and Testing

(3) Heissdampfreaktor - FRG -
Seismic response of buildings, equipment, and piping

(4) JAERI - Tokai Pipe Break Test Facility - Japan -
Test of Pipe Break and Pipe Whip Phenomena

(5) JRC Biaxial Dynamic Testing Facility - Ispra, Italy -
Dynamic Testing of Large Structural Element. Tests for Response of Structures and Equipment to High Frequency Input Motion (Currently limited to single axis testing)

(6) Marviken Test Facility - Sweden -
Blowdown, fluid jet characteristics, reaction and impingement effects, containment characteristics

(7) Meppen Missile Test Facility - FRG -
Effects of missiles on concrete and steel targets

(8) NUPEC - Isago Environmental Test Facility - Japan -
Containment Environment Simulation and Testing

(9) NUPEC - Todatsu Seismic Test Facility - Japan -
Full Scale Shaker Table Test Facility - 15M x 15M Table

(10) Scalay Missile Test Facility - France -
Small Diameter Missile Effects

(11) Whiteshell Containment Test Facility - Canada
Hydrogen burn inside containment

(12) Winfrith Missile Launcher Laboratory - United Kingdom -
Scale model tests of missile effects

can be found a summary of existing foreign experimental and computational facilities which lend themselves to multinational research activities.

10.4 Identification of Current Selected Research Capabilities and Interests for Research Facilities in Canada, France, Japan, Sweden, the United Kingdom, Federal Republic of Germany, and the United States

In Table 8 can be found a listing of particular research topics and the organizations within Canada, France, Japan, Sweden, the United Kingdom, the United States, and the Federal Republic of Germany which have recently or have indicated a future interest in performing research in the areas indicated. With respect to the U.S., only the programs of national laboratories listed in Section 1. of this report and the Electric Power Research Institute are given. The U.S. NRC supports significant safety related research in Universities and other research organizations not identified and included in this report.

The key to the organizations listed in Table 8 is presented as follows:

A	Argonne National Laboratory - U.S.
B	Brookhaven National Laboratory - U.S.
C	Atomic Energy of Canada Ltd. (AECL)- Canada
C1	Whiteshell Nuclear Research Establishment - Canada
D	Atomic Energy Control Board (AECB) - Canada
E	Electric Power Research Institute (EPRI) - U.S.
F	French Atomic Energy Commission (CEA) - France
F1	Institute of Protection and Nuclear Safety (CEA) - France
F2	Saclay (CEN) - France
F3	Cadarache (CEN) - France
G	Japan Atomic Energy Research Institute (JAERI) - Japan
G1	Tokai Research Establishment (JAERI) - Japan
H	Nuclear Power Engineering Test Center - Japan
H1	Tadotsu Engineering Laboratory (NUPEC) - Japan
H2	Isogo Engineering Laboratory (NUPEC) - Japan
I	Building Research Institute (RRI) - Japan
J	Industry Research - Japan
K	Japan Institute of Nuclear Safety (JINS) - Japan
L	Lawrence Livermore Laboratory - U.S.
M	Idaho National Engineering Laboratory - U.S.
N	Los Alamos National Laboratory - U.S.
O	Oak Ridge National Laboratory - U.S.

Table 8 - Summary of Current Safety Research Capabilities and Interests in the U.S. and Selected Foreign Countries

I. SITING

A. Seismic:	
(1) Simulation, Monitoring and Study of Earthquake Effects at Site	L, M, Q, F2, G
(2) Seismic Source Term Evaluation	L, Q
(3) Comprehensive Approach to Seismic Risk	L, F2, P
B. External Blast:	
(1) Evaluation of Phenomenon	A, C1, F, G, R, T
(2) Improved Hazardous Material Siting	A, F, R, T, U, V
(3) Measurment of Radiological Release	U
C. Flood:	
(1) Methodology for Flood Risk	D, O
D. Meteorlogy and Atmospheric Dispersion:	
(1) Development and Evaluation of Atmospheric Dispersion Models	F, G, O, R, U
E. Aircraft and Missiles:	
(1) Aircraft Probabilities	D
(2) Missile Characteristics	F, G, U
F. General Siting Criteria Development:	
(1) Identification and Description of Extreme Natural Events	F

Table 8 - Summary of Current Safety Research Capabilities and Interests in the U.S. and Selected Foreign Countries (continued)

II. STRUCTURAL ENGINEERING

A. Load Combinations & Fragility:

(1) PRA Evaluation and Fragility of Structures	A, B, M, F
(2) Structural Reliability Methodology	B, E, F, U
(3) Safety Margins and Load Combinations	B, T, E, F

B. Seismic Design:

(1) Evaluate Dynamic Test Procedures on Structures	A, M, F2, G, X, N
(2) Nonlinear Seismic Analysis	A, B, L, N, X, U
(3) Develop Generic Floor Response Spectra	B
(4) Evaluate Seismic Scram	M
(5) Study Damping	M, G, X
(6) Study of Seismic Risk and Safety Margins in Structures	L, N
(7) Improved Seismic Design Methodology	L, G, N
(8) Soil Structure Interaction Effects	A, B, L, E, F, G, R
(9) Development of Seismic Isolation Systems	G
(10) Development of Seismic Qualification Procedures for Operating Plants	B

C. Blast:

(1) Blast Resistant Design of Structures	A, U
(2) Analysis of Containment for Hydrogen Retension	N, T, C, U
(3) Hydrogen Behavior in Containment	C1, T, E, U

D. Aircraft and Missile Impact:

(1) Response of Concrete Walls and Slabs to Missile Impact	F2, U
(2) Response of Steel Plates to Missile Impact	G
(3) Experimental and Theoretical Effects of Missile Impacts and Pipe Break	S, U, W

Table 8 - Summary of Current Safety Research Capabilities and Interests in the U.S. and Selected Foreign Countries (continued)

E. LOCA and Pipe Break Response:	
(1) Two Phase Jet Load Prediction on Target Structures	T, Pl
(2) Evaluation of Dynamic Loads on BWR Containment	Gl, S, P, U, Vl
F. Ultimate Strength Capacity of Containment:	
(1) Evaluate Ultimate Strength of Containment and Other Structures	B, N, T, C, F, G, U
(2) Determine Buckline Capacity of Containment	N, G
III. MECHANICAL ENGINEERING	
A. Load Combinations, Fragility and Probability Risk Assessment:	
(1) Component Reliability	A, O, E, G
(2) Probability Risk Assessment and Fragilities	B, B, O, C, F
B. Seismic Testing Analysis and Design Components:	
(1) Piping Analysis, Testing and Design	M, L, E, G, J, Hl
(2) Computer Code Development for Component Analysis	A, B, L
(3) Primary - Secondary Support Interaction	N, E, M
(4) Equipment Seismic Qualification	B, M, L, E, C
(5) Damping Determination in Equipment & Piping	M, Gl, J
C. Pipe Break & Pipe Whip Phenomena:	
(1) Pipe Break and Pipe Whip Phenomena	A, E, F, Pl, U
D. High Frequency Loading:	
(1) Vibration Monitoring of Components & Pipe	A, E, F, G

Table 8 - Summary of Current Safety Research Capabilities and Interests in the U.S. and Selected Foreign Countries (continued)

E. Qualification of Components:	
(1) Relief Valves	M, T, E, G, U
(2) Steam Generator	A, G, P
(3) Piping	A, B E, C
(4) Electrical Equipment	B, C, F
F. Design Code Application:	
(1) Evaluation of Component Design Against Existing Design Codes	B, M, E, L
IV. MATERIALS & METALLURGY	
A. Cracking:	
(1) Radiation Embrittlement	A, M, O, E, F, G, R, U
(2) Fatigue	A, C, F, G, R
(3) Growth and Arrest Methodology	O, E, F, G, R, U
B. Non-Destructive Examination and In-Service Inspection:	
(1) Radiographic	M, F, G, P, U
(2) Ultrasonic	M, C, F, P, R, U
(3) Surface	M, U
(4) General	M, F, G, U
C. Fracture Mechanics:	
(1) Elastic Fracture Mechanics	F, G, R
(2) Elastic-Plastic and Plastic Fracture Mechanics	F, F, G, R
(3) Fracture Toughness	E, F, G, R

Table 8 - Summary of Current Safety Research Capabilities and Interests in the U.S. and Selected Foreign Countries (continued)

D. Failure Analysis of Materials:

(1) Thermal Shock	L, G
(2) General	L, G, Y
(3) Testing (Heavy Plate)	F, G, R, U
(4) Corrosion (Stress)	G, P
(5) Residual Stress	P
(6) Strain Aging	A, U
(7) Concrete	A, U
(8) Melting	A, U

V. COMPUTATIONAL CAPABILITIES

A. State-of-the-Art of New Computer Code Development:

(1) Seismic	A, B, E, F2, K
(2) Pipe Whip and Break, Missile Impact	A, E, G
(3) Nonlinear	E, F2, K, Y
(4) Fluid-Structure Interaction	E, F2, K, Y
(5) General Purpose	E, F2, K, Y

P Swedish Nuclear Power Inspectorate (SKI) - Sweden

P1 Studsvik Energiteknik AB - Sweden

Q Central Electricity Generating Board (CEGB) - U.K.

R UKAEA Safety & Reliability Directorate (SRF) - U.K.

S AEE Winfrith Dorset (SRD) - U.K.

T Sandia National Laboratory - U.S.

U Federal Ministry for Research and Technology (BMFT) - FRG

V Federal Ministry of Interior (RS-BMI) - FRG

V1 Reactor Safety Associates (GRS) - FRG

W Meppen Missile Test Facility - FRG

X Heissdampfreaktor Test Facility - FRG

Y Joint Research Center - ISPRA - European Comm.

Z Canadian Westinghouse Enviro. Test Facility - Canada

10.5 Potential for Bilateral or Multilateral Sponsorship of Nuclear Safety Related Research

In this section specific research projects which should be of interest to the U.S. and one or more foreign countries are suggested. These projects are considered preferential candidates for multinational sponsored research.

10.5.1 Siting

10.5.1.1 Small Aircraft Crash

The extent to which small aircraft crash is considered as a design requirement in nuclear facilities varies in the countries surveyed. Only France has a well-defined requirement to design nuclear power plant safety-related facilities to resist small (> 5000 Kg) aircraft crash. Currently Canada has a program to review probabilities of small aircraft crash. In the U.S., it has generally been assumed that the tornado design requirement envelops the small airplane crash, but there does not appear to be a definitive evaluation available of small aircraft crash characteristics or probability of occurrence which would rule out such an event as a design basis for nuclear plants in the U.S.

10.5.1.2 Seismic Source Term Evaluation and Comprehensive Approach to Seismic Risk

There appears to be a growing worldwide sentiment to develop more rational site seismic design criteria based on actual seismic risk rather than on the somewhat arbitrary current deterministic methodology.

Countries which should be interested in such a multinational research effort are the U.S., Canada, Sweden, the U.K., and Japan.

10.5.2 Structural Engineering

10.5.2.1 Seismic

There is a particular concern among all the countries surveyed regarding the development of more realistic seismic design criteria. The perceived negative interaction of seismic design requirements with the service life and reliability of the reactor coolant and auxiliary safety systems has recently become a major concern.(Ref. 5) Since seismic response of the foundation-structure and the building structure serve as input to the reactor coolant and auxiliary safety systems, it is essential that such response be realistically defined. This evaluation would include joint research activities in the following areas:

(1) Worldwide collection of data and correlation of structural damage to actual strong motion earthquake input levels,

(2) Correlation of damage levels for actual earthquake with structural analysis (linear and nonlinear) of building structures, and

(3) Development of mean and lower bound damping data for structures and structural elements at high seismic stress levels.

This evaluation would also include the relative importance of the OBE and SSE as a seismic design basis.

The countries which should have a particular interest in this research area are the U.S., Japan, France, and the FRG.

10.5.2.2 Containment Design

Nuclear containment vessels and structures have been designed simultaneously for relatively high pressures (1 to 5 bar), to remain essentially leak-tight and to sustain high earthquake loads. While containments are routinely tested up to 1.15 times design pressure before being placed into service, there is little information (Ref. 6) concerning their behavior up to failure pressure and essentially no information regarding how such composite structures behave when subjected to design basis earthquake level loads. The planned seismic tests of 1/4 scale PWR and BWR containments at the new NUPEC-Tadotsu shaker table in Japan should go a long way to define leak-tight integrity during strong motion earthquakes. However, lacking any cooperative agreement in this area, Japan is not clear how much of the test results' details will be available outside Japan.

Tests are currently underway at Sandia National Laboratories under U.S. NRC sponsorship (Structural Safety)(Ref. 1) to evaluate containment ultimate pressure capacity. It would seem logical to attempt to develop a detailed exchange of information regarding the seismic capacity of containments developed in Japan with containment over pressure tests in the U.S. However, it should be understood that the Japanese containment seismic test program will cost in excess of $\$40 \times 10^6$ while the NRC's containment program current and projected budget is less than $\$15 \times 10^6$. During a recent visit to Europe by Sandia personnel, particular interest was expressed by the NII in the United Kingdom and the SKI in Sweden in the Sandia containment test program.

In addition to the containment pressure test program, a closer liaison between the U.S. and Japan, associated with the Seismic Safety Margin Research Program, might be developed as an added inducement for the exchange of information on the Japanese seismic containment tests. Through the Electric Power Research Institute (EPRI), which is helping to support aerosol studies at the Marviken Test Facility and containment hydrogen research at the Canadian Whiteshell Containment Test Facility, the U.S. NRC should be kept abreast of the results of these programs.

10.5.2.3 High Frequency Cyclic Loads

Another area of relatively new concern is the transmittal of high frequency cyclic loads (40 to 80 cycles at 1 to 5 g peak response) resulting from impact through the building structure to the supports of safety equipment. These loads result from ringing of the structural as a result of impact loads from water and steam hammer sudden valve closure, suppression pool response to dynamic loads and impact from missiles and aircraft. If such loads are treated in the same manner as low frequency cyclic loads (seismic), they will often control design of equipment. Countries which should be particularly interested in multinational research efforts in this area are the FRG, the U.S., Sweden, and Japan.

10.5.2.4 Ductility Limits

Response of structures and components just prior to failure is normally into the inelastic or plastic material range. Consistent with the anticipated response into the inelastic range, there is a need to define limits of ductility both local and global at failure for a range of loadings, materials, and structural configurations. Only in this way can safety margins associated with the design basis loads be quantitatively defined. All countries should be interested in this research area.

10.5.3 Mechanical Engineering

10.5.3.1 Seismic

In the area of Mechanical Engineering as well as in the Structural Engineering area, the Japanese have recently completed and are currently conducting a very extensive test program in seismic qualification of mechanical and electrical components. These programs are summarized as follows:

(1) Damping in Piping Systems - Sponsored by Japanese Industry (1979-1981) Total Funding $\$15 \times 10^6$

(2) Damping in Piping Systems - Sponsored by Japanese Government (1980-1982)

(3) Seismic Qualification of Equipment - Sponsored by Japanese Industry (1981-1983) Total Funding $\$15 \times 10^6$

(4) Seismic Qualification of Major Nuclear Components - Sponsored by Japanese Industry and Government (NUPEC) (1983-1986) Total Funding is $\$400 \times 10^6$ or approximately $\$100 \times 10^6$/year

Given the large scope and cost of these programs, any agreement to share this information with the U.S. would probably require a very significant commitment on the part of the NRC to provide funding or results of other U.S. safety research. As in the case for structures, there is a need for a joint research activity in the following areas:

(1) Worldwide collection of data and correlation of structural, leak-tight integrity and functional damage of industrial, mechanical, and electrical equipment and distribution systems to actual strong motion earthquake input levels,

(2) Correlation of damage levels in mechanical and electrical equipment and distribution systems in actual earthquakes with structural analysis (linear and nonlinear) of such components, and

(3) Development of mean and lower bound damping data for mechanical and electrical equipment and distribution systems at high seismic stress levels.

The countries which should have a particular interest in this research area are the U.S., Japan, France, and the FRG.

Canada is continuing a study on seismic qualifications of equipment and may be interested in a cooperative effort with the U.S. With the exception of Japan, all of the countries surveyed have some concern regarding seismic qualification of equipment in older operating plants which were not designed to be seismically resistant to current levels and methods of design. In the U.S., through its Systematic Evaluation Program, a large number of older plants have been seismically reevaluated. As a result, there exists a large quantity of data which might be correlated with judgment in a multinational research program to permit seismic qualification of the older facilities without recourse to detailed structural analysis.

10.5.3.2 High Frequency Load

The resultant high frequency ringing of structures subjected to large impulse and impact loading resulting from suppression pool oscillation or large missile impact using current spectral analysis method may control design of equipment supported by those structures. Observed damage of equipment subjected to high frequency cyclic loads is significantly less than when subjected to lower frequency excitation at similar accelerated levels. Multinational research in this area should receive the support of the U.S., FRG, Sweden, and Japan.

10.5.3.3 Pipe Break Design Criteria

As in the case of seismic design, there is a growing concern that the use of pipe whip restraints on high energy piping systems may be having a negative effect on the service life of such systems. It is believed that better ISI and the use of improved leak detection methods should reduce the potential high energy pipe break to the point where pipe whip restraints no longer need be applied. There is a need for a joint research activity in the following areas:

(1) Worldwide collection and evaluation of data pertaining to pipe rupture in industrial piping,

(2) Evalution of leak detection and monitoring devices and methods, and

(3) Development of ductility limits to be used in the design of pipe whip restraints to minimize their size.

All countries including the U.S. should have an interest in this research area.

10.5.4 Materials and Metallurgy

Through the JRC administered PISC (Program for Inspection of Steel Components) effort, there already exists significant multinational cooperation in the areas of materials and metallurgy. While most of the research is being nationally funded and conducted in national laboratories, there is an agreement for sharing the total scope of the effort which is collectively established. It is anticipated that future nuclear safety research in this area will be conducted within the framework of the existing international cooperative PISC effort.

Additional candidates for multinational sponsorship of research in the materials and metallurgical areas would be in the development and experimental verification of elasto-plastic and plastic fracture mechanics using both probabilistic as well as deterministic methodology. Countries with a particular interest in this area would be the U.K., Japan and the U.S. Also, long term radiation embrittlement of cast materials is a likely candidate for a multinational research effort.

10.5.5 Summary and Conclusions

Multinational sponsorship of nuclear safety research should receive wide support in the countries surveyed in this report, assuming agreements can be negotiated which allow for sharing the total scope of a particular project or which provide for a trade off or exchange of information in unrelated areas. In general, there has been considerable reluctance for any transfer of funds between countries in the support of multinational nuclear safety research unless the research is being performed at a unique facility.

Given the ground rules just stated, the projects identified in Table 9 would appear to be good candidates for multinational agreement and sponsorship of safety research of interest to the U.S. NRC within the topical scope of this report.

Table 9 Summary of Candidate Programs for Multinational Sponsorship Among Selected Foreign Countries Canada, France, Japan, Sweden, U.K., and FRG with the U.S.

A. Siting -

1. Small Aircraft Crash Probabilities and Effects

2. Seismic Source Term Evaluation and Comprehensive Approach to Seismic Risk

B. Structural Engineering -

1. Worldwide Collection of Earthquake Data and Correlation of Structural Damage to Actual Strong Motion Earthquake Input Levels for Modern Industrial Facilities

2. Correlation of Damage Levels for Actual Earthquakes with Structural Analysis (Linear and Nonlinear) of Building Structures

3. Development of Mean and Lower Bound Damping Data for Structures, Structural Elements at High Seismic Stress Levels

4. Cooperative Agreement with NUPEC-Tadotsu, Japan Shaker Table Full or Large Scale Seismic Testing of Containment and Other Major Nuclear Power Plant Structures

5. Cooperative Agreement with Other Countries for Sandia Laboratory Test Program for Containment Pressure Test to Failure

6. High Frequency Cyclic Load Transmission through Structures and Development of Structural Acceptance Criteria for High Frequency Cycle Loads

7. Research on Global and Local Ductility Limits on Structures and Structural Elements

C. Mechanical Engineering -

1. Cooperative Agreement with Japan to Obtain Results of Damping Tests on Piping and Equipment and NUPEC-Tadotsu Shaker Table Tests of Major Nuclear Components

2. Worldwide Collection of Data and Correlation of Structural, Leak-Tight Integrity and Functional Damage of Industrial, Mechanical, and Electrical Equipment and Distribution Systems to Actual Strong Motion Earthquake Input Levels

Table 9 Summary of Candidate Programs for Multinational Sponsorship Among Selected Foreign Countries Canada, France, Japan, Sweden, U.K., and FRG with the U.S. (continued)

3. Correlation of Damage Levels in Mechanical and Electrical Equipment and Distribution Systems in Actual Earthquakes with Structural Analysis (Linear and Nonlinear) of such Systems

4. Development of Mean and Lower Bound Damping Data for Mechanical and Electrical Equipment and Distribution Systems at High Seismic Stress Levels

5. Development of Simplified Techniques for Seismic Qualification of Existing Equipment and Distribution Systems in Operating Nuclear Power Stations

6. Development of Design Criteria Applicable High Frequency Cyclic Loading of Equipment and Distribution Systems

7. Worldwide Collection and Evaluation of Data Pertaining to Pipe Rupture in Industrial Piping

8. Evaluation of Leak Detection and Monitoring Devices and Methods

9. Research on Global and Local Ductility Limits on Equipment Distribution Systems and Their Supports

D. Materials and Metallurgy -

1. Continued Participation in the PISC-II Study

2. Development of a Coordinated Program for the Study of Elastic-Plastic or Fully Plastic Fracture Mechanics

3. Development of a Coordinated Program for the Study of the Long Term Irradiation Embrittlement of Cast Materials

REFERENCES

(1) Offices of Nuclear Regulatory Research, Long Range Research Plan FY 1984 - FY 1988, NUREG-0784, U.S. Nuclear Regulatory Commission, August 1982.

(2) Electric Power Research Institute, Structural Mechanics Program: Progress in 1981, EPRI NP-2705-SR Special Report, October 1982.

(3) Sandia National Laboratories, Light Water Reactor Safety Technology Program Quarterly Report, July-September 1981, SAND 82-0101, December 1981.

(4) Private Communication, J.D. Stevenson to W. Von Riesemann, January 1983.

(5) S. Bush, "Piping System Design," Letter from S. Bush, Pressure Vessel Research Committee to NRC Chairman Palladino, 8/20/81.

(6) Sebrell, W.A. (Editor). "Proceedings of the Workshop on Containment Integrity", NUREG/CP-0033, U.S. Nuclear Regulatory Commission, October 1982.

APPENDIX A

LETTER QUESTIONNAIRE SENT TO NATIONAL LABORATORIES

Appendix A
Letter Questionnaire Sent to National Laboratories

Dear Sir:

This letter is meant to confirm our telcon of concerning the status of water reactor safety research at in the technical areas of structural, mechanical engineering, siting and metallurgy materials. I am preparing a report detailing current, anticipated and projected future water safety research. This is part of a contract consulting effort I am performing for the Research Division of the U.S. NRC under the technical monitorship of Dr. J. O'Brien.

While I would appreciate a brief summary of the current DOE and NRC sponsored safety research, the main thrust of my effort is to determine your interests for future research in the technical areas indicated, given the funding were available. A discussion of any particular areas of expertise and special or unique facilities available at the lab would also be appreciated. Finally, I would also like to discuss potential sources of safety research funding other than the NRC either foreign or domestic that you think might be available in the technical areas of interest.

Specifically, I hope you might be able to provide the following information when I visit on

1. A current table of organization and staffing for those groups which are conducting safety related research in structural-mechanical engineering, materials and siting.

2. The latest monthly or quarterly report on NRC, DOE or other agency sponsored safety research in technical areas identified.

3. In lieu of reports identified in Item 2, a research summary sheet with format as typically shown in Figure A.1.

4. A listing and brief description of those test, analytical and computational facilities and resources at which you consider give you a special capability to perform safety research in the technical areas identified.

5. A summary description of those areas or projects of safety related research which you would like to perform in the future and your identification of possible sources of cooperative or joint funding (foreign or domestic) of safety related research other than the U. S. NRC.

The information provided will be compiled in a survey report of U.S. and foreign safety related research in the areas of structural-mechanical engineering, materials, and siting which you will have the opportunity to review prior to publication.

Thank you for your assistance.

Sincerely,

John D. Stevenson

JDS:kab

RP1236-1 Repair Welding of Heavy Section Steel Nozzles

Prime Contractor: Babcock & Wilcox Company (P S. Ayres)

Duration: October 13, 1978 to February 13, 1982

Project Cost: $938,800

EPRI Project Manager: T. U. Marston

Objective: The development program has as its primary objective: A)defining an alternate repair procedure that will be superior to the half bead technique with regards to ease of implementation and service suitability of the resulting repair, and B) defining the metallurgical and mechanical properties of repair welds made with both the alternate and half bead techniques.

Strategy: Develop an alternate repair procedure through analytical modeling techniques and experimental selection (Task I). Characterize the residual stresses and fracture mechanics based mechanical properties produced by the half bead and alternate weld repair procedures (Task II). Compare, analytically and experimentally, the service suitability of repair welds made in BWR feedwater nozzles with both repair procedures (Task III).

Status: Literature search has been completed in weld repair without post-weld stress relief; a topical report is in preparation. A BWR feedwater nozzle mockup has been repaired using the conventional bead technique referenced in the ASME Code. The most promising alternative weld repair procedure appears to be a TIG/MIG procedure. Residual stresses have been calculated in order to optimize the welding procedure (minimize residual stresses).

"Repair Welding of Heavy Section Steel Nozzles," Semiannual Progress Report.

J. M. Bloom, "Analytical Assessment of the Effects of Residual Stresses and Fracture Properties on Seismic Performance of Various Weld Repair Processes," ASME Paper 80-C2/PVP-140.

Figure A.1 Typical Research Summary Sheet

Follow-up Letter Sent to National Laboratories

Enclosed herein please find the rough draft text of our NUREG/CR report of the NRC "Selected Review and Evaluation of U.S. Safety Research Vis-A-Vis Foreign Safety Research for Nuclear Power Plants," it is intended to identify U.S. research resources and interest of the various national laboratories in the topical areas identified in the Abstract attached hereto and Introduction to the report. Individual national laboratories chapter texts have been generally organized as follows:

X.1 Organization

X.2 Topic Areas of Interest

X.2.1 DOE
X.2.2 NRC - Nuclear Reactor Regulation
X.2.3 NRC - Nuclear Regulatory Research
X.2.4 Other

X.3 Facilities

X.3.1 Experimental
X.3.2 Computer Hardware - Software
X.3.3 Research Staff Capabilities

X.4 Areas of Future Interest

X.5 Liaison with Other Foreign or Domestic Nuclear Safety Research Activities

I would appreciate any comments or additional information you may have by . I expect to finalize the report by .

Thank you for your assistance.

Sincerely,

John D. Stevenson

JDS:clj

NRC FORM 335 (11-81) U.S. NUCLEAR REGULATORY COMMISSION

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16. ABSTRACT (200 words or less)

A review of currently available nuclear research resources in a selected group of United States government national laboratories is presented. The current nuclear safety research interests of industry organizations, particularly the Electric Power Research Institute, are also identified. In addition, suggestions for potential joint or cooperative funding of light water reactor safety research in the U.S. between the NRC and other organizations, both foreign and domestic, are presented. The topics of research considered are associated with the areas of Siting, Structural Engineering, Metallurgy, Materials, and Mechanical Engineering.

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