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Selected Review of Foreign Safety Research for Nuclear Power Plants

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Prepared for
U.S. Nuclear Regulatory
Commission

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ABSTRACT

A compilation and description of current foreign research related to regulatory standards and licensing issues in areas of interest associated with Siting, Structural Engineering, Metallurgy and Materials, and Mechanical Engineering are presented. Also included in this report and summary is a discussion of those research areas in which there exists a potential for joint sponsorship by the U.S. N.R.C. The particular foreign countries surveyed are Canada, France, Japan, Sweden, United Kingdom of Great Britain & Northern Ireland, and the Federal Republic of Germany.

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PREFACE

This report provides a synopsis of current nuclear safety research and unique research facilities related to regulatory standards and licensing issues in the technical areas of Siting, Structural and Mechanical Engineering, Metallurgy and Materials in a selected sample of those countries which have a well-developed nuclear power industry. The sample is further limited to those countries where either significant nuclear reactor safety research is sponsored or detailed codification and documentation of regulatory standards are available. It is intended that the information and observations contained in this report can be used to identify and to promote the potential for multi-national sponsored nuclear safety research.

This is the fourth in a series of reports, including the following: foreign licensing practice; foreign regulatory standards and current licensing issues; and a summary of U.S. nuclear safety research.

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NOMENCLATURE

Canada

AECB........Atomic Energy Control Board
AECL........Atomic Energy of Canada Limited
CSA.........Canadian Standards Association
WNRE........Whiteshell Nuclear Research Establishment

France

AFCEN.......French Society for the Design and Construction Standards of Nuclear Island Components
AFNOR.......Association Francaise de Normalisation
CEA.........Atomic Energy Commission
CEN.........Center for Nuclear Studies
DEMT........Department of Mechanical and Thermal Studies
DSN.........Department of Nuclear Safety
EDF.........French Electricity Authority
IPSN........Institute for Nuclear Protection and Safety
SCSIN.......Central Safety Department for Nuclear Facilities

Japan

BRI.........Building Research Institute
JAEC........Japan Atomic Energy Committee
JAERI.......Japan Atomic Energy Research Institute
JINS........Japan Institute for Nuclear Safety
MAPI........Mitsubishi Atomic Power Company
MHI.........Mitsubishi Heavy Industries
MITI........Ministry of International Trade and Industry
NUPEC.......Nuclear Power Engineering Test Center
REB.........Resources and Energy Bureau
STA.........Science and Technology Agency
TEPCO.......Tokyo Electric Power Company

Sweden

SA..........Swedish Plant Inspectorate
SKI.........Swedish Nuclear Power Inspectorate
SSPB........Swedish State Power Board

United Kingdom of Great Britain & Northern Ireland

BNDC........British Nuclear Design and Construction Company
CEGB........Central Electricity Generating Board
HSE.........Health and Safety Executive
NII.........Nuclear Installations Inspectorate
NNC.........National Nuclear Corporation
SRD.........Safety and Reliability Directorate

NOMENCLATURE (continued)

TNPG........The Nuclear Power Group
UKAEA.......United Kingdom Atomic Energy Authority

Federal Republic of Germany

BMFT........Federal Ministry for Research & Technology
BMI.........Federal Ministry of The Interior
DIN.........German Standards Institute
GRS.........Company for Reactor Safety
IRS.........Institute for Reactor Safety
KTA.........Nuclear Standards Commission
KWU.........Kraftwerk Union A.G.
NKe.........Nuclear Technology Standards Committee
RSK.........Reactor Safety Commission
SSK.........Radiation Protection Commission
TUV.........Technical Inspection Agency

United States

BNW.........Battelle Northwest Laboratories

SUMMARY REPORT ON FOREIGN RESEARCH

1. INTRODUCTION

1.1 Current Climate For Multinational Sponsored Nuclear Safety Research

Nuclear safety research programs in countries outside the U.S. are generally conducted as well as supported by government rather than industry organizations. Research capabilities are viewed as part of a country's technological resources; therefore, they are supported as a matter of national policy. Japan for example, established the Nuclear Power Engineering Test Center in 1976. One of its stated purposes was to "strengthen nuclear technology independence in Japan."(Ref. 1) 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 second country there is some expectation that the second country will in turn support various 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 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 the countries surveyed are 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 1980 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 commited. This shifting from a predominate design and new construction to an operating phase is certainly one manifestation of a maturing industry.

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

Moreover, it also should be understood that all of the European countries surveyed, France, Sweden, the United Kingdom and the Federal Republic of Germany 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.

This report will attempt to identify and to highlight opportunities where one or more of situations noted in (1), (2), or (3) above exist between the U. S. and the six countries surveyed. It should be understood that all of the countries surveyed are highly developed industrial nations with a strong interest in commercial development of their own national nuclear steam supply systems. As a result, the climate for future cooperative bilateral or multilateral safety research involving the U. S. is probably better in somewhat less developed nations which have not developed the broad expertise or the facilities in the research topics covered in this report.

It should also be understood that the USNRC has bilateral technical exchange and cooperative agreements in the field of research and development on reactor safety with all of the countries surveyed in this report except Japan, hence, it is expected that opportunities for multinational sponsorship of research will develop as a result of the technical exchanges associated with these agreements.

1.2 Current Licensing Issues Causing Significant Safety Research Activity

This report will also identify the current licensing issues which are considered of prime concern to the countries presented in this study. Licensing issues are identified and divided into two categories.

The first category addresses those issues which appear to be of general or common concern to all or most of the countries surveyed; these issues are identified and discussed Section 1.2.1 of this report. An example of a general licensing issue is seismic design since all countries surveyed consider seismic design resistance of nuclear power plants as a significant licensing issue.

The second category of licensing issues deals with those issues which are of particular interest to only one or a minority of the countries surveyed. Licensing issues such as these are identified and discussed in Section 1.2.2 for each of the countries surveyed. Examples of licensing issues in the second category include tornado resistant design in the U.S. and large (5000 Kg) aircraft crash resistant design for all nuclear power plants in the FRG. These topics are significant licensing issues for particular countries but are not considered to be of general interest to a majority of the countries surveyed.

1.2.1 Common Licensing Issues

1.2.1.1 Seismic

All countries surveyed currently consider earthquake resistance as a general design requirement for nuclear power plants. For some countries considered in this review, namely Sweden and the United Kingdom, seismic consideration has only recently become a design requirement. Examination of the design basis of all the countries surveyed in this report reveals that only Japan has considered a specific modern seismic design requirement for all its operating nuclear power plants. Because of the current trend toward seismic resistant design, an obvious general licensing issue is consideration of what steps, if any, should be taken to upgrade the seismic capability of the existing operating facilities which originally had little or no seismic resistant design. To date, only the U.S. through its NRC mandated Systematic Evaluation Program has formally embarked on a program to evaluate and to upgrade, where necessary, the seismic design resistance of older nuclear power stations.

Moreover, considerable differences have arisen among countries in terms of defining and implementing earthquake resistant design. These areas of significant differences include the following:

(1) Relative value of zero period ground acceleration to be considered in design

(2) Definition of specific level or levels of earthquake to be considered in design, (for example, SSE, OBE, S_1, S_2, DBE, etc.)

(3) Damping values to be considered in design

(4) Permissible structural behavior criteria as a function of earthquake level

(5) Earthquake input to be defined as a component or as a resultant motion and the means by which earthquake motions should be combined

(6) Intallation of automatic seismic scrams on power reactors

(7) Question of seismic loads as having a causative relationship to other design loads such as pipe break or design basis accident

To a greater or lesser degree, all of the items 1 through 7 should be considered as licensing issues in the countries surveyed.

1.2.1.2 Vessel or Major Component Rupture

Nuclear power plant containment and other engineered safeguard and protection systems for water-cooled reactors are not designed to accommodate the consequences of a major nuclear component rupture such as a reactor vessel, calandria, steam generator, reactor coolant pump, and pressurizer. For this reason, the structural integrity of these components for the design life of the plant must be assured with an extremely high degree of confidence. Because of this fact, a great deal of research and applied evaluation techniques in the behavior of relatively thick walled steel components has developed. Below is a list of such major research efforts:

(1) Examination techniques (usually non-destructive) to measure the following:

(a) ductility of metal as a function of forming process heat treatment and aging (irradiation)
(b) detection of crack or flaw size and orientation

(2) In-service inspection techniques to detect changes in original "as built" conditions

(3) Determination of how in-service changes in ductility and how crack or flaw size and orientation affect component design or safety margins

(4) Evaluation of fatigue and brittle fracture effects on design or safety margins

Existing pressure vessel construction and in-service inspection codes applicable to nuclear components have incorporated inspection and evaluation techniques, design procedures, and behavior limits all of which are intended to provide the necessary margins to assure that there will be no catastrophic rupture or degradation below an acceptable safety margin limit. In all countries covered by this report, major research efforts are continuing to develop improved material properties, inspection techniques, design procedures, and behavior limits in order to improve or to better define design and safety margins applicable to major components during construction and operation of the plant.

1.2.1.3 Degraded Core

Safety and protection systems in water-cooled nuclear power plants are generally not designed to accommodate a level of core damage which results in significant deformation of core geometry or failed fuel in excess of one percent. Therefore, much system design effort has been expended to provide redundant emergency core cooling systems to preclude significant core damage. In each of the countries considered for this report, these design measures (and their high degree of reliability and redundancy requirements) were generally found in the safety criteria developed by the regulatory authorities.

To better understand degraded core phenomena and thereby improved engineered safeguard performance, all of the countries considered in this report have ongoing major experimental or analytical research activities which consider various levels of core degradation.

In Sweden, the government has required that as a condition for continued operating permits for Barseback units 1 and 2, systems for filtered venting of the containments shall be installed and operable before September 1, 1986.

1.2.1.4 Probability Risk Assessment (PRA)

Probability Risk Assessment is a relatively new research activity which promises for the first time to provide a systematized means for a trade-off evaluation between normal and anticipated transient operation versus extreme environmental load and accident effects on the plant. As a result of PRA, it may be possible to optimize overall plant reliability and safety and thus permit concentration of the greatest engineering effort on those phenomena which have the greatest influence on public health and safety.

To date, PRA has had its greatest impact in the U.K. and the U.S. In the future, it is anticipated PRA will be used to an increasing degree in all countries to replace the minimax decision process(Ref. 3) and thereby provide more rational nuclear power plant design bases.

1.2.2 Licensing Issues by Country

1.2.2.1 Canada

(1) Environmental Extreme Loads

Because the CANDU PHW reactor system is quite different from the large volume vessels of PWR and BWR systems used in the U.S., it is difficult to define specific licensing issues relative to the topical areas of this report which are relevant for both heavy and light water reactor systems.

In the past, site-related, environmental, extreme load and input design criteria for Canadian nuclear power plants have been somewhat less stringent than the design criteria used for similar sites in the U.S. For example, with the exception of the Darlington site where a design basis tornado is established at the 10^{-6}/yr. probability of occurrence level, Canadian criteria do not require tornado resistant design. Moreover, values for the seismic zero period ground acceleration are typically lower (10^{-6} versus 10^{-7} probability level) than those values used in the U.S. for similar sites.

On the other hand, Canada has developed more stringent criteria than the U.S. with regard to structural behavior criteria under seismic loads. In addition, the design evaluation of the containment system for postulated leakage equal to that which could go undetected during normal operation is a requirement not considered in the other countries surveyed. More recent plant site seismic input criteria are of relatively higher intensity. This fact has also recently caused some reevaluation of seismic design adequacy of both the feeder pipe supports and the refueling machine which were originally designed to withstand the lower g levels typical of low seismic sites.

(2) Radiation Damage to Metals

Also related to feeder pipe design adequacy has been a concern regarding radiation growth of the feeder pipes within the design limits of their supports.

(3) Hydrogen Deflagration and Detonation

The use of large amounts of zirconium in the fuel, pressure and calandria tubes has the potential for generation of large amounts of hydrogen in containment due to a zirconium water reaction in the event of loss of core cooling. This has led to the construction of large, cylindrical and spherical hydrogen test chambers at the Whiteshell Nuclear Establishment where tests are currently being conducted to determine burning and ignition characteristics of hydrogen in a containment environment.

(4) Pipe Break

The AECL and the utilities have adopted a leak-before-break philosophy concerning pipe break design which would be used on plants currently in design and construction. This assumption would significantly curtail pipe break design requirements for Canadian plants as compared to those requirements currently used for U.S. plants. The AECB is currently evaluating the concept and as yet has not committed itself to acceptance of the leak-before-break criterion.

It should be understood that there also exists in Canada a large body of detailed criteria documents prepared by AECL and the various nuclear utilities, particularly Ontario Hydro, which contain detailed requirements for implementing and augmenting some of the more general requirements listed herein. These criteria documents are considered proprietary by the organizations which authored them and are not available for inclusion in this report.

1.2.2.2 France

(1) Seismic Design and Seismic Scram

The 900 MWe standard plants designed until recently have used a mean response spectra for 0.2 g ZPGA developed by EDF from a selection of 8-10 time histories for California earthquakes. For the 1300 MWe stations, it is anticipated that a mean plus one standard deviation spectra normalized to a 0.15 g ZPGA will be used.

At the present time, there is no intention to install automatic seismic scrams for the nuclear steam supply system. Apparently, Turbine-Generators installed in France have vibration trips without low frequency filters. The turbine-generator vibration trip would effectively trip the reactor in the event of a strong motion earthquake. A decision to return to power would be made by the operator after assessing available earthquake monitoring and other plant operation sensors. Some consideration is being given for installation of automatic scram for high seismic sites, but no decision has been made as yet.

(2) Seismic Response of Containment Founded on Piles

Concrete containment models, approximately 1/10 scale, were tested adjacent to a large fill compacted dynamically for the new Nice airport runway. Measured maximum acceleration in the model was equal to 0.1 g. Maximum response of the pile-founded containment model was 2-3 times that of an adjacent raft foundation model. Results of the research suggest that pile-mounted stations may be subjected to greater seismic loads than stations supported on raft foundations.

(3) Modification of PWR Pump Bearings to Resist LOCA Loads

As a result of an evaluation of LOCA induced loads on PWR reactor coolant pumps, pump bearings have been modified to carry such loads. No corresponding modifications seem to have been made on similar pumps in the U.S.

(4) Evaluation of 3 S_m Secondary Stress Design Limits

Experimental studies run at the Saclay research facility indicate that significant ratchetting or progressive increases in deformation are occurring in metals subjected to stress cycling below the 3 S_m level set as the elastic shakedown limit in design codes at an elevated temperature. These test results suggest to the researchers reporting the results that the 3 S_m secondary stress limits used in design may have to be reduced or that the effects of inelastic deformation be considered in analysis.

1.2.2.3 Japan

(1) Seismic Scram

Japanese nuclear power stations employ seismic scrams to automatically shutdown the plant in the event of a strong motion earthquake. Scram settings are normally set at 0.9 S_1 level. The technical basis underlying the requirement for automatic scrams is not clearly defined. At this time, it appears that this requirement is more a result of perceived public interest rather than a definitive safety need.

(2) Damping of Piping Systems

Results of field measurements of pipe system damping taken during start-up tests in Japanese power plants have generally yielded conflicting and inconsistent data. As a result, for the past three years the Japanese utilities in cooperation with the nuclear steam suppliers have been conducting an extensive laboratory experimental program on evaluation of dynamic response of piping systems, particularly the evaluation of damping. Evaluation of the results of this study has been hampered by the highly non-linear behavior of the systems where small gaps are present in the supports. Detailed results of the tests have yet to be published, but one pattern based on tests of individual restraints seems to have emerged. Damping for an individual restraint appears to increase dramatically from a value of about 1.0 percent to 7.0 percent critical as the motion of the pipe moves through the as-constructed gap in the restraint. In real systems such gaps typically vary from 0.05 to 0.50 inches. Once the gap is closed in a particular cycle, the damping tends to decrease typically to the 2.0 to 3.0 percent range. Obviously, in a real piping system where there are many supports and gaps, the resulting overall damping of the system is the integrated effect of the pipe contacting and moving through gaps in a cyclic manner.

Damping in piping is of particular interest in Japan because regulatory agencies have required relatively low values in the 0.5 to 1.0 percent range be used in current design. The Japanese utilities as a result of the tests sponsored in their nuclear steam system suppliers test facilities have established a recommended 3.0 percent damping value to be used in pipe design.

The Japan Atomic Energy Research Institute (JAREI) on behalf of the regulatory and licensing agencies has developed its own test program which is currently in progress to evaluate damping in pipe systems.

(3) Materials and Inspection Procedures

In line with the general licensing issue of vessel, major component or pipe rupture identified in Section 1.2.1.2 of this report, Japan has been and is currently conducting a major materials, metallurgical and inspection research effort.

(4) Pipe Rupture

The purpose of pipe rupture studies in JAERI is to perform model tests on pipe whip, restraint behavior, jet impingement and jet thrust force and to establish a computational method for analyzing these phenomena under a BWR operational condition of 6.77 MPa pressure and temperature of 285°C.

The pipe specimens are 114.3 mm (4 inch) in diameter and 8.6 mm in thickness and 4500 mm in length. Pipe whip restraints used in the tests are the U-bar type 8 mm in diameter and fabricated from type 304 stainless steel. The experimental parameters were the clearance gap (30, 50 and 100 mm) between the restraint and pipe wall and the overhang length (250, 400 and 1000 mm). The dynamic strain behavior of the pipe specimen and the restraints is investigated by strain gages and their residual deformation is obtained by measuring marking points on their surface. The pressure time-history in the pipe specimens is also obtained by pressure gages. Prior to the pipe whip tests, a jet thrust force test was performed to obtain the jet thrust force, using the pipe specimen of nearly the same size as that in the pipe whip tests.

(5) BWR Pressure Suppression Containment Behavior

A study of the containment pressure suppression effect in a BWR has been conducted in a full-scale segment Mark-II containment response test program. The test facility was constructed in March 1979. Through February 1981, a total of 19 tests were conducted to observe the effects of liquid and vapor line breaks and to provide line break data for containment pressures, pressure differentials, temperatures, water level, and structural responses in terms of strain and acceleration. The test series was completed in March 1982.

1.2.2.4 Sweden

(1) Seismic Design

Sweden has no provision for automatic seismic scram on any of its stations. To date, only two units, Forsmark 3 and Oskarshamn 3, are designed to be seismically resistant. Forsmark 3 is designed for an SSE ZPGA of 0.15 g and Oskarshamn 3 for 0.1 g. Since Sweden is considered to be one of the lowest earthquake potential areas on earth, these earthquake levels or larger are selected as having a probability of local occurrence of approximately 10^{-5}/yr. Some consideration is being given to a back fit of seismic resistance for other nuclear power plants being introduced during the next 10 years; however, no decision has been made to date. No OBE requirement is effectively considered because of very low acceleration levels established for the OBE. Other seismic design procedures appear similar to those used in the U.S.

(2) Tornado Design

Together with Italy and Canada, Sweden appears to be the only country other than the U.S. which has an explicit tornado design requirement. Forsmark III has been designed for an equivalent U.S. Zone 3 tornado. At this time, there is not a plan to back fit tornado resistant design to other stations.

(3) Other Extreme Loads

The Swedish position on other extreme loads is summarized as follows:

(a) LOCA and SSE loads are considered separately.

(b) There is no requirement for aircraft crash design, based on a high degree of redundancy (4 train emergency core cooling system) and good physical separation.

(c) Containment design for PWR's is being evaluated for the effect of a 75% zirconium water reactor.

(d) No generic external blast is considered in design, except for the effects of conventional bombs and acts of terrorism and sabotage.

(e) The Safe Shutdown Flood level is set at 10^{-5}/yr. return period. Design Basis Flood is set at 10^{-2}/yr. local probability.

(f) Pipe break restraints have not been used outside containment but back-fit analysis is being considered.

(4) Metal Behavior Criteria

In Sweden, allowable stresses in steel seem based on yield parameters and not on ultimate strength as is the case in the U.S. This practice tends to favor the use of steels with high-yield strengths.

The fatigue usage factor in ASME Code is limited to 0.5 rather than 1.0. The carbon content permitted in Swedish steels tend to be more restrictive than in the U.S. Swedish safety research is taking a leading role in fatigue and stress corrosion evaluations in steels. It is believed that by careful control of metal chemical properties, forming and fabrication procedures, progressive cracking in metals can be eliminated.

(5) Jet Impingement Tests

Recent tests at the Marviken facility indicate that interior jet pressure may be reduced to ambient within 3 diameters of the break opening for subcooled fluid jets.

(6) Cross Flow Induced Wear on Steam Generator Tubing

Relatively recent Westinghouse designs of PWR steam generators have tended to locate the feedwater inlet in the lower tube section of the steam generator. Feedwater cross flow appears to have accelerated heat exchange tube wear and caused appreciable tube thinning after only a few months of operation. This condition has become a significant licensing issue for two of the three Westinghouse designed stations in Sweden.

1.2.2.5 United Kingdom

(1) Vessel Rupture

The potential rupture of high energy, high pressure water reactor vessels and other major vessel components has been for some time a major licensing issue associated with the introduction of water reactors into the U.K.

(2) Probability Risk Assessment (PRA)

As identified in Section 1.2.1.4 the United Kingdom is engaged in significant PRA research aimed primarily at developing more rational basis for design loads to be considered in nuclear power plant construction. In addition they appear to be considering identifying loads in containment beyond nominal design basis up to the ultimate strength of the containment.

(3) Government Inquiry

A government sponsored public inquiry into all safety aspects of the PWR systems planned for installation at the Sizewell site, which is patterned after the PWR Callaway Station located at Fulton, Mo., began on 26 July 1982 (full hearings are expected to commence on 11 January 1983). The nuclear steam supply system for Sizewell is being provided by the National Nuclear Company (NNC) as a license to the Westinghouse Electric Co. to the CEGB. During the course of the inquiry, additional licensing issues are expected to be identified.

1.2.2.6 Federal Republic of Germany

Because the FRG has a general requirement for design against the aircraft impact effects of military aircraft operating at cruising speed and a blast overpressure impulse load not considered in the other countries surveyed, licensing issues have been developed based primarily on these particular extreme load phenomena.

(1) High Frequency Response

Postulated aircraft impact has the effect of high frequency excitation in the range of 20-80 Hz on equipment installed on common foundations with the impacted structure. Simplified dynamic elastic analysis would indicate several times g acceleration impacted to structures and equipment is more associated with the total impulse than with peak response amplitude. Acceptance criteria based on energy input (impulse) rather than force or acceleration have yet to be clearly established; hence, this area remains a current licensing issue.

(2) Characteristics of Impact and Blast Loading Response

Because of the loading phenomena associated with aircraft impact (penetration formulas, dynamic response of structures), research efforts are being expended to better define the response to these effects; thus, loading response continues to be identified as a licensing issue.

(3) Simplified Structural Analysis and Documentation

Because each state in the FRG, rather than the Federal Government, is the supreme licensing authority, there is generally a requirement to produce complete design and analysis documentation for each site even though a single standard plant design may have been used for a number of sites. As a result, industry has recently introduced the "convoy" system to minimize the amount of analytical (software) design documentation required for each plant. Efforts are underway to minimize the amount of analysis required to license a standard plant at a specific site.

1.3 Summary of Safety Research Funding

1.3.1 Introduction

Over the past two years, where data is available and in the opinion of many of the organizations contacted, research efforts, with the exception of those efforts specifically applicable to the TMI accident consequences, have or will tend to decline in terms of real spending capabilities. Table 1 shows data on the funding allocated for safety research on water reactors. For some organizations listed, it has been necessary to estimate the amounts because no distinction was made between reported development and safety research budgets.

The anticipated decrease in available real research funding is attributed to several conditions. One reason is the effect of a worldwide economic decline. Another reason is the overall view by research planners that the water-cooled power reactor technology which has been in widespread use in the current generation of nuclear power plants is now almost 20 years old and research necessary to resolve most of the generic licensing and safety issues identified to date has been performed or is in the development and planning stages. This is not to say that operating experiences such as TMI, which opened up the

Table 1 Trends in Nuclear Safety Research Funding[1]

			US$ x 10^6		
	1979	1980	1981	1982	1983
Canada					
Atomic Energy Control Board	-	2.3[2]	2.7[3]	-	-
Atomic Energy of Canada Ltd.	-	-	7.7	-	-
Ontario Hydro	-	-	6.1	-	-
France					
SCSIN	-	-	30.	-	-
EDF - Framatom - CEA - WEST.			(4)		
Japan[5]					
STA	-	-	35.	35	35
MITI	-	-	35.	15	15
Utilities & Nuclear Steam System Suppliers	-	-	10	8	8
Sweden					
SKI (STUDSVIK)	5.2	5.6	5.9	7.8	-
ASEA - ATOM, Utilities		-	1.0	-	-
United Kingdom					
NII (Direct)	-	-	2.0	-	-
NII (HSE)	-	-	3.0	-	-
UKAEA (SRD)	-	-	45.0	-	-
CEGB	-	-	(6)	-	-
Federal Republic of Germany					
BMFT			(7)		
BMI			17.		
TUV - Lander			(8)		
NSSS - Utilities			(9)		
United States					
NRC	-	-	200	-	-
DOE	-	-	2	-	-
EPRI	-	-	40	-	-
Other (Utilities, NSSS, A/E)	-	-	7	-	-

Notes:

(1) Limited to safety research applicable to water reactors

(2) Includes 1.6 x 10^6 for special IAEA Safeguards Program

(3) Includes 1.9 x 10^6 for special IAEA Safeguards Program

(4) Probably exceeds 10 x 10^6/year

(5) Additional support from Special Energy Trust Fund expected at $\$65.0 \times 10^6$/yr. through 1985 primarily to support 15 M Shaker Table Facility

(6) Direct funding probably does not exceed $\$2 \times 10^6$/year

(7) Total research budget $\$65 \times 10^6$/yr. Safety research probably does not exceed 20 percent of total.

(8) Safety research probably does not exceed $\$5 \times 10^6$/yr.

(9) Safety research probably does not exceed $\$3 \times 10^6$/yr.

broad area of probabilistic risk assessment and ultimate pressure capacity of containment, may not generate new licensing issues in the future. However, in the absence of such events, the trend in expenditures for water reactor safety research is expected to decrease.

In addition, it is expected that research expenditures will tend to concentrate more on licensing issues associated with operating reactors rather than on licensing issues affecting siting and new construction of power reactor facilities.

1.4 Organization of the Report

In Sections 2 through 7 for Canada, France, Japar, Sweden, United Kingdom, and the Federal Republic of Germany are presented descriptions of the organizations responsible for specifying safety research and a description of the research projects within the scope of this report which have recently been completed or are currently underway.(Refs. 2,3,4,5,6,7) Also, included are descriptions of unique facilities contained within those countries.

In Section 8 is presented a discussion of areas of potential joint or multinational sponsorship of research programs. In Appendix A are presented specific observations by Dr. S. Bush relative to research programs in the six countries surveyed in the areas of materials, metallurgy, nondestructive testing, and in-service inspection.

2. SUMMARY DESCRIPTION OF NUCLEAR SAFETY RESEARCH FACILITIES AND PROGRAMS IN CANADA

2.1 General Capabilities and Organizations

Nuclear safety research in Canada is usually performed under the sponsorship of one of three organizations, the Atomic Energy Central Board (AECB), Atomic Energy of Canada Limited (AECL), and Ontario Hydro which has extensive research facilities and directly funds some safety related research.

2.1.1 Atomic Energy Central Board (AECB)(Refs. 8,9)

The AECB is designated as a departmental corporation within the meaning and purpose of the Financial Administration Act and is an agent of the Government. Created by the Atomic Energy Control Act, the AECB functions as a regulatory body controlling the development, application, and use of atomic energy. The AECP receives its authority through the ACT and through regulations approved by the Governor-in-Council.

By means of a comprehensive licensing system, the Board controls all dealings in prescribed atomic energy substances and equipment for the purpose of assuring that such sustances and equipment are utilized with the proper consideration both for health and safety concerns and for national and international security. The Board's licensing system is administered with the cooperation of other federal and provincial government departments in the areas of health, environment, transport, labor aspects, and others.

In addition, the AECB has the responsibilty for identifying research and development needs related to its regulatory and safeguards functions. It carries out very little in house research but awards and administers contracts. The AECB's research organization is shown in Figure 1. During the fiscal year 1981/82, fifty-two research and development projects were either completed or continued. related to these areas:

Risk and Safety Evaluation	19
Health Effects	14
Environmental Processes	18
Special Safeguards	1
Security	1
Regulatory Process Development	1

2.1.2 Atomic Energy of Canada Ltd. Research Company (AECL)

Atomic Energy of Canada Ltd. (AECL) is a crown corporation 100% owned by the Canadian government. The AECL functions as a nuclear steam system supplier; some of its responsibilities are similar to the duties of an Architect/Engineer in the U.S. The organization also has the primary responsibility to provide the scientific and technological base for Canada's nuclear program. In addition, the AECL has engineered and has manufactured heavy water pressure tube nuclear steam supply systems

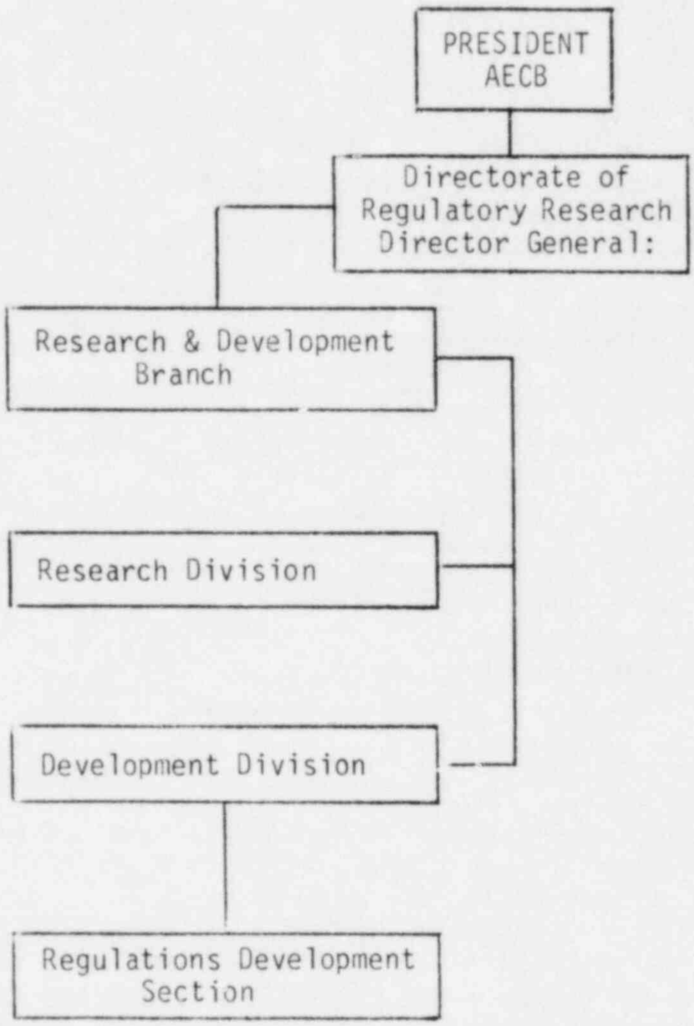


Figure 1- AECB: Organization for Regulatory Research

which are either under construction or planned in Argentina, Korea and Rumania.

The AECL is divided into five operating companies: Engineering - 2500 person staff, 800 engineers; Research - 2500 at Chalk River and 800 at Whiteshell; Radio Chemical - 200 to 300 persons; Heavy Water - 600 to 800 persons; and International Projects - 100 persons. The Engineering company has a total budget of \$60-80 x 10^6/yr.

The primary mission of the Research Company is to provide the scientific and technological base for Canada's nuclear program. The Company operates two major national laboratories: the Chalk River Nuclear Laboratories at Chalk River, Ontario, and the Whiteshell Nuclear Research Establishment at Pinawa, Manitoba, which tends to concentrate on accident analysis. The scope of work at the Engineering Company ranges from basic scientific research including reactor safety research to the transfer of technological developments to Canadian industry. The total budget of the Research Company is \$90-100 x 10^6/yr. With approximately \$8-10 x 10^6/yr. allocated to safety research, the balance of the funds is directed primarily toward basic scientific research and transfer of technology to Canadian industry.

2.1.3 Ontario Hydro

Ontario Hydro is a government owned electric utility which maintains research facilities near Toronto. Most of its research activities are aimed at improving the production and distribution of electric power. The utility generally conducts safety-related research as requested by the AECB and in cooperation with AECL. Ontario Hydro also helps to support safety research conducted at the AECL laboratories and at the Westinghouse Canada environmental test facilities. The total estimated funding spent by Ontario Hydro in 1981 on safety-related research was \$7,000,000.

2.2 Unique Facilities

2.2.1 Whiteshell Containment Test Facility (CTF)

A Containment Test Facility (CTF) has been constructed by AECL at the Whiteshell Nuclear Research Establishment (WNRE) to verify analytical models predicting the behavior of CANDU nuclear reactor containment systems under loss-of-coolant accident (LOCA) conditions initiated by a postulated pipe break. The CTF has been particularly interested in studying the behavior of hydrogen produced by the reaction between zircaloy fuel sheathing and steam and later released to the containment system with the coolant through the break. The manner in which hydrogen reacts (combustion, detonation) in the steam-air containment atmosphere affects its transient pressure and temperature. Since the containment system is the final of several barriers to the release of radioactive materials, there is considerable incentive to continue to improve the understanding of containment behavior, including hydrogen combustion and fission product chemistry, such that consequences of postulated accidents can be determined more precisely.

A basic research program to provide a data base for developing appropriate models to predict containment atmosphere behavior was initiated several years ago, and the CTF serves as the larger scale verification facility for these models. Although the primary focus has been on the combustion and detonation behavior of hydrogen, the CTF has been designed for flexibility so that variety of aspects important to reactor containment can be studied. The following considerations have influenced the design of the facility:

(1) Verification of hydrogen combustion models in distributed systems. The combustion behavior of hydrogen in weak concentrations or in systems in which the concentration varies in the different parts of the network is particularly important.

(2) Verification of detonation behavior and suppression in distributed systems. Because of the potential effect of detonation on containment integrity, it is important to verify the conditions under which detonation can occur and those under which it can be suppressed.

(3) Verification of predictions of steam-water-air pressure transients in multi-volume systems. In both this area and in hydrogen behavior, the momentum and mixing effects associated with branches and networks are particularly important.

(4) Investigation of flashing and condensation dynamics. Flashing dynamics can affect the pressure transient during the first second or so which is important to multi-unit designs. Condensation and other heat removal mechanisms have a large effect on the pressure transient at later times when fission product release may be important. Data would be obtained to verify the modelling of the processes.

(5) Fission product transport and depletion. A great deal can be learned about fission product transport and depletion using chemical simulants. This technique would be used to verify the fission product activity transport and deplection models.

(6) Investigation of leakage, seal, process equipment and safety instrumentation behavior. The experiments outlined above can provide pressure, temperature, and humidity conditions to test the behavior of leaks, seals, and components. The results of these experiments may be particularly significant in the hydrogen combustion situation.

(7) Special control systems. The effectiveness of various filter and fission product removal systems can be tested in the facility.

The CTF is located in a separate building designed specifically as a testing facility for potentially hazardous systems. The test area and control area are separated by a 300 mm thick reinforced concrete wall. The general layout is shown schematically in Figure 2 while Figure 3 presents a simplified process flow sheet.

Experiments at the CTF can be performed over a wide range of geometries ranging from individual vessels to various connected vessel systems. The main experimental components are two vessels, a 6.3 m^3 (220 ft^3) sphere and a 10.3 m^3 (360 ft^3) cylinder, and an interconnecting pipe to provide the geometric features of a CANDU multi-unit containment system with a vacuum building. Specific vessel dimensions are given in Table 2. The spherical vessel is mounted on an air pallet which permits moving the vessel with respect to the fixed cylinder, therefore allowing for different lengths of pipe. Provision has been made to add at a later date, a branch to another duct thus terminating in a third vessel. The design pressure of 10 MPa (1450 psi) permits experiments with detonations.

Each vessel has various ports which are listed in Table 3. The flanges of the manways are equipped with sight glasses for schlieren (or streak) photography. The 20" diameter main process connection determines the maximum diameter of the interconnecting pipe. Solid sample ports are included for fission product studies where coupons of specific materials can be inserted to determine plateout rates. The laser port permits the use of Raman diagnostic techniques (for example, CARS) to determine temperature and species concentrations remotely. The cylindrical vessel is equipped with connections for future experiments with dousing systems. Since many experiements will involve steam-air mixtures, the components have trace heating to prevent condensation on the walls.

The experimental system is supported by various auxiliary systems to provide various gaseous components in a controlled manner. These are all operated remotely.

The recirculation system serves to mix the gaseous contents of the main process and can also be used for fission product filtration studies. In addition, this circuit provides a means of remotely bleeding off to the atmosphere. The compressor is a liquid ring type using water as a seal between the impeller and the casing to prevent ignition. The compressor system also includes a cyclone separator and seal water heat exchanger.

The vacuum system serves to evacuate the system. It has a compressor system similar to that in the recirculation system. Hydrogen, air, and halon can be supplied in metered amounts either by appropriately setting the pressure control or by monitoring the flow rates and remotely throttling the valves. The air supply is also used to purge the system.

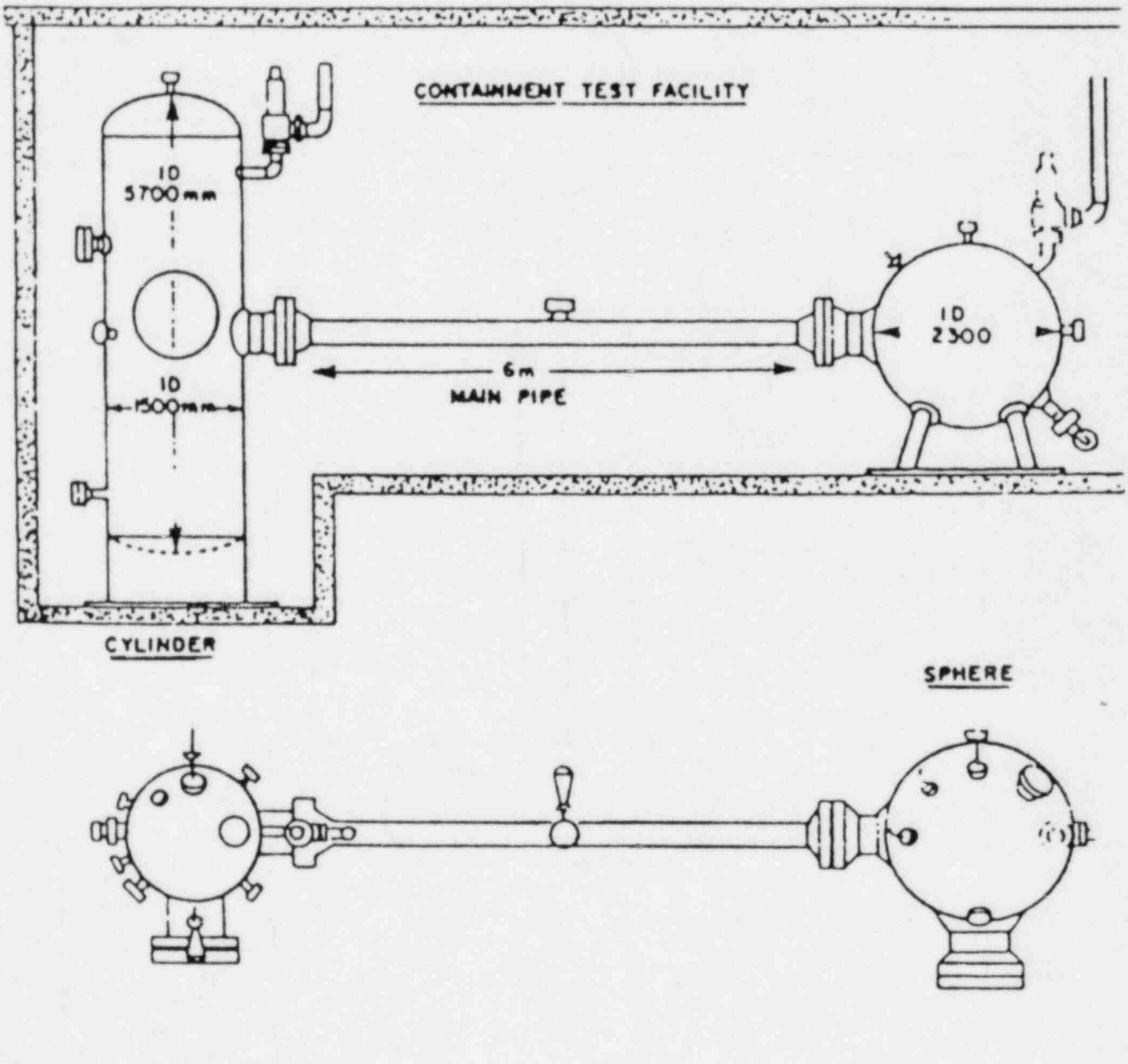


Figure 2 Schematic Diagram of the Containment Test Facility

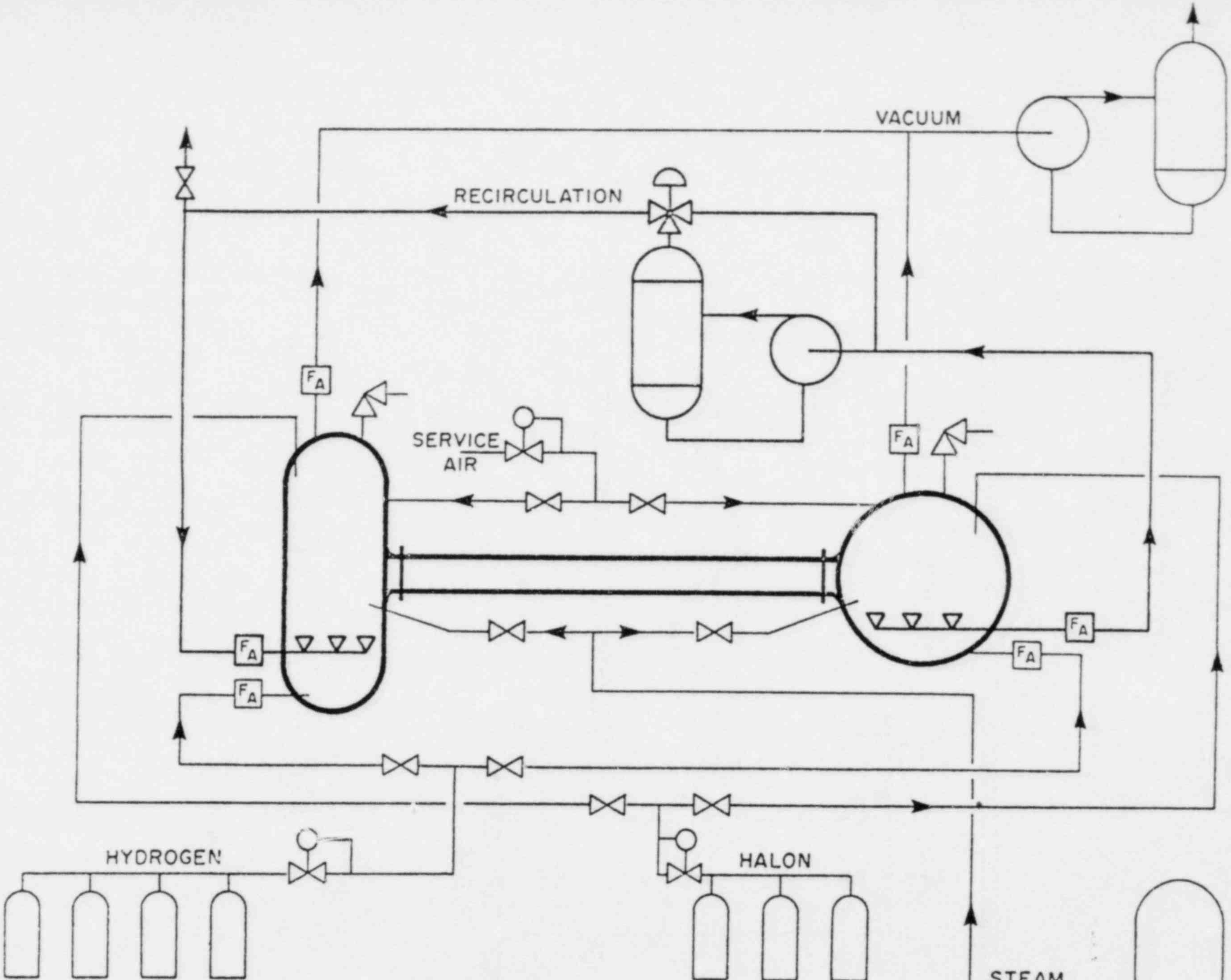


Figure 3 CTF Flowsheet (Simplified)

Table 2 Canadian Containment Test Facility Vessel Dimensions

	Sphere	Cylinder	Interconnecting Pipe	
			Initial	Maximum
Internal diameter	2.29 m	1.5 m	.29 m	.51 m
Length	-	5.7 m	6 m	24 m
Wall thickness	53 mm	73 mm	17 mm	
Volume	6.3 m^3	10.3 m^3	0.5 m^3	4.8 m^3
Design pressure	10 MPa	10 MPa	10 MPa	10 MPa

Table 3   Canadian Containment Test Facilities Vessel Connections

	Sphere		Cylinder	
	Quantity	Size	Quantity	Size
Manway	2	20"	2	20"
Main process	1	20"	1	20"
Recirculation loop	1	4"	1	4"
Pressure relief	1	4"	1	4"
Solid sample	2	6"	2	6"
Vacuum	1	1"	1	1"
Air	1	1"	1	1"
Steam	1	1"	1	1"
Laser port	1	2"	1	2"
Instrument port	2	2"	2	2"
Ignition source	1	1"	1	1"
Drain	1	1"	1	1"
Hydrogen	1	1/2"	1	1/2"
Halon	1	1"	1	1"
Gas sample	1	1/2"	2	1/2"
Pressure tap	1	1"	1	1"
Dousing system	-	-	2	1"

The steam supply is designed to provide 2-40 kg of water at 311°C into either vessel in 5-20 seconds. These rates were selected as appropriate for experiments coupling hydrogen behavior with loss-of-coolant blowdown dynamics.

Experiments are run remotely from the control room which is separated from the test area. Process instrumentation includes pressure, temperature, and flow-rate measurements at various points in the system as well as appropriate alarms and interlocks. Two closed circuit TV cameras are located in the test area for monitoring gauges, switches, etc., and are displayed in the control room.

Instrumentation available to support experiments include high speed thermocouples (10 μs response) and pressure tranducers (1 μs), two-velocity component Laser Doppler anemometry to determine turbulent gas velocities, schlieren photography to follow flame front travel, and Raman spectroscopy to determine chemical species concentration and temperature. Data collection and control are handled by the CTF microcomputer which can access the central DEC system PDP-10 computer of the WNRE site for data processing. Currently available are two channels of 10 MHz transient recording (4K memory/channel) to capture some of the detail across passing flame fronts and 32 channels at 100 KHz throughout. The event recorder has 1 μs resolution and is intended to provide accurate time of arrival of flame fronts and shock waves at various locations. The facility was operational in early 1981.

2.2.2 Canadian Westinghouse Environmental Test Facility

In Hamilton, Ontario, Westinghouse Canada maintains an environmental test facility which has been used extensively for environmental qualification of equipment. Its basic parameters are as follows:

(1) Horizontal Vessel - Overall length 20'-0" x 7'-8" I.D.; ASME rated to 80 psi. Capable of steam saturation in 7.0 sec.

(2) Vertical Vessel - Overall height 15'-0" x 7'-8" I.D.

Both vessels have chemical spray capability.

3. SUMMARY DESCRIPTION OF NUCLEAR SAFETY RESEARCH FACILITIES AND PROGRAMS IN FRANCE

3.1 General Capabilities and Organization

Nuclear safety research in France is usually performed under the sponsorship of one of two organizations or groups, the Service Central de Surete des Installations Nucleaires (SCSIN) and a four-organization group consisting of EDF, Framatome, CEA, and Westinghouse.

Established in 1976, the SCSIN has several responsibilities: establishing licensing procedures for nuclear facilities; drawing-up and enforcing general technical rules and regulations; organizing the supervision of these facilities and carrying it out; and examining safety problems associated with choice of sites. The actual detailed technical review of the Safety Analysis Reports and related safety issues is performed by the Department of Nuclear Safety (DSN) which is part of the Institute of Protection and Nuclear Safety (IPSN) which in turn is part of Atomic Energy Commission (CEA).

Safety research directed and supported by SCSIN is proposed by the Institute De Protection et de Surete Nucleaire (IPSN) of the Commissariate a l'Energie Atomic (CEA). As part of IPSN, the Department de Surete Nucleaire (DSN) recommends needed safety research in the construction and operation of nuclear power plants. The recommended research is the result of DSN's detailed technical review of the plant and problems encountered during construction, start-up, in-service inspection and operation of the nuclear power plants. Current SCSIN funding of PWR safety-related research is at the $30 x 10^6/yr. level. Cost of the safety research program is paid by EDF in the form of licensing fees based on prior years costs. In general, the research is performed at one of the CEA research installations in France, primarily at Saclay or Cadarache. The current SCSIN, PWR related, safety research is summarized on a yearly basis in a CEA publication, "Fichier Des Etudes de Surete 1980, Reacteurs A Eau Ordinaire Sous Pression." (Ref. 2) Safety research results are also summarized in the Nuclear Safety Research Index. (Ref. 3)

Most nuclear safety research in France not otherwise funded directly by SCSIN is done through a four-party agreement among EDF, Framatome, CEA, and Westinghouse, although each party reserves the option to do independent research. Decisions regarding the research to be performed and the apportionment of cost to the four organizations is made by a committee composed of representatives of each organization. A general three day meeting to review all research activities and decisions as to development of priorities, continuation of future projects, and establishment of funding levels is held yearly, usually in March. The budget for the jointly sponsored safety research is not publically available, and while some of the projects, particularly those where CEA is involved are listed in References 2 and 3, not all such safety projects are necessarily listed.

The French Commissarit a l'Energie Atomique (CEA), which acts as a technical consultant to the nuclear licensing arm of the French Ministry of Industry, and one of the members of the EDF, Framatome, and Westinghouse industry group operates four laboratories at Saclay, Fontenay-aux-Roses, Grenoble, and Cadarache for general nuclear research and development.

Nuclear safety studies at the Centre d'Etudes Nucleaire (CEN) in Saclay are conducted within the Department of Mechanical and Thermal Studies (DEMT)-a group of about 165 technical personnel plus support staff. The DEMT consists of three sections: one dealing with solar and other alternative energy projects, a second addressing structural and thermal (fluid) mechanics topics, and a third systems section concerned with LOCA containment studies for all reactor types and fuel transport and reprocessing. Total DEMT funding amounts to about $\$20 \times 10^6$/yr. apportioned as follows:

general safety studies (20%)

fast reactor projects (30%)

PWR research (25%)

alternate energy souces (25%)

The structural and thermal mechanics section includes about 80 technical personnel organized into three groups, thermal hydraulics laboratory, structural mechanics, and large computer programs.

The following programs currently underway or recently completed at Saclay within the scope of interest of this study are as follows:

(1) thermal hydraulics experiments and computer code development

(2) static structural mechanics experiments and computer code development, pipe whip, fracture mechanics

(3) seismic studies and experimental evaluation of dynamic response of structures and equipment

(4) evaluation of thermal stress ratchet or increased plastic strains where elevated temperature cycling are kept within the 3 Sm or 2 Sy limits of the ASME Code

(5) testing of integrity of the other PWR incore instrumentation leads out of the bottom of the reactor assuming one has ruptured

Research studies at CEN Grenoble concentrate in the following areas:

(1) basic metallurgy and mechanical behavior of reactor materials and reactor fuel

(2) fundamental research in solid mechanics

(3) fundamental biological research

(4) thermal transfer

(5) fission reactors

(6) electronics

Research in the area of reactor materials and nuclear fuel is administered by the CEA Department of Metallurgy and Nuclear Fuel (DMECN), and is distributed among the four laboratories as follows:

(1) CEN Saclay (Department of Technology) - fast reactor materials, PWR pressure vessel materials, basic fracture mechanics research. Saclay also interacts with Electricite de France (EDF) and Framatome for investigating mechanical behavior of PWR fuel assemblies.

(2) CEN Fontenay-aux-Roses (Department of Plutonium Fuel) - basic plutonium research.

(3) CEN Cadarache (Department of Fuel Rod Development) - basic plutonium research, behavior of fast reactor fuel assemblies. Cadarache is also conducting out-of-pile studies (PHEBUS) on the behavior of defective LWR fuel assemblies (for example, fuel heatup due to loss of coolant flow, fission product release) under postulated LOCA conditions.

(4) CEN Crenoble (Department of Metallurgy) - basic materials research, mechanical behavior of reactor materials and nuclear fuel.

The Grenoble Department of Metallurgy (DMG) divides some 200 people roughly evenly between two groups: Radiological Studies, and Metallurgical Studies. The DMG investigates such basic materials phenomena as creep, fatigue, and crack propagation in stainless steels used for fast reactor components (for example, AISI316, Incaloy), and correlates electron microscope observatives of fracture surfaces with material properties. Mechanical testing of materials (for example, tensile tests, low-cycle fatigue tests, high-temperature fracture tests) is generally closely coupled with computer analyses of observed behavior. It was emphasized that no safety studies are performed at Grenoble, these being concentrated in Saclay within the Department of Mechanical and Thermal Studies. Instead, the Grenoble work in basic materials research supports the safety studies elsewhere, resulting in close cooperation between Grenoble and Saclay.

In summary, the Department of Metallurgy at Grenoble primarily provides basic research support to other CEA laboratories more directly involved in nuclear safety studies of U.S. NRC interest. Some thermal hydraulics safety research is performed in Grenoble within the Department of Thermal Transfer but most is concentrated in Cadarache. Similarly, structural research, human factors, and risk assessment are addressed mainly at Fontenay-aux-Roses and Saclay. As a result, the U.S. NRC will more likely find cooperative research opportunities with these other laboratories rather than in Grenoble.

Nuclear safety research is also conducted at Cadarache, described in individual projects as shown in Reference 3, but most safety-related studies are concentracted at Saclay while Cadarache is more active in reactor development projects.

3.2 Unique Facilities

At Saclay there are excellent facilities for shaker table and dynamic testing of structures and materials. There is also a missile test facility. At Cadarache there is a pipe break test facility. However, none of the facilities are unique in that there does not exist equivalent capabilities witnin the scope of this report in the U.S. or in at least one of the other six countries surveyed.

4. SUMMARY DESCRIPTION OF NUCLEAR SAFETY RESEARCH FACILITIES AND PROGRAMS IN JAPAN

4.1 General Capabilities and Organizations

In Japan as in Canada, France, Sweden, and the U.K., there is a strong industry as well as government-sponsored safety research program. However, in Japan, unlike the other foreign countries surveyed, much of the direct industry funded research is done in non-government laboratories as discussed in Section 4.1.2. Government-funded nuclear safety research in Japan is sponsored primarily by the Science and Technology Agency (STA) and the Ministry of International Trade and Industry (MITI). The STA has administrative responsibility for regulation and licensing of commercial nuclear plants in Japan. The MITI has responsibility for technical review for the licensing of commercial nuclear power plants in Japan.

4.1.1 Government Activities

In 1981 the Japanese government embarked on its second five-year program on Safety Research on Nuclear Power Plant and Its Related Research Facilities.(Ref. 5) This program has been developed by an advisory committee and is reviewed annually for necessary updates and modification.

The 1981-1985 program is categorized into the following eight areas:

(1) Safety of light-water reactor fuel

(2) Loss of coolant accident

(3) Structural safety of light-water reactor facilities

(4) Lessening radiation emission materials from reactor facilities

(5) Probability safety appraisal of reactor facilities

(6) Earthquake for reactor facilities

(7) Safety of nuclear fuel facilities

(8) Safety of nuclear fuel transport cask

Each research project is typically organized as follows:

(1) Purpose of research

(2) Contents of research

(3) Period of research

(4) Sponsorship of research

(5) Organization in charge of research

Safety research is funded both by direct government appropriations and by a special trust fund which is supported by a special energy tax. Currently, this tax is generating \$600 x 10^6 in revenue per year. Most of the fund is used as compensation for those impacted by adjacent power plant siting, but a portion (currently about 10 percent) is available to support safety research as needed. In particular, capital funding for major research facilities such as the new 15m x 15m shaker table at the Tadotsu Engineering Laboratory and the BWR blowdown test facility at the JAERI facility at Tokai was supported from the special energy tax subsidy special trust fund.

Government funding for water reactor safety related research not otherwise provided by the trust fund are provided by the Science and Technology Agency (STA) and the Ministry of International Trade and Industry (MITI). The 1981 STA safety research budget was approximately \$35 x 10^6, and the MITI budget was approximately \$15 x 10^6 exclusive of the trust fund contribution.

Government-sponsored nuclear safety research in Japan is typically performed in one of three major institutions.

(1) Japan Atomic Energy Research Institute (JAERI)

   (a) Tokai Research Establishment
   (b) Takasaki Research Establishment
   (c) Oarai Research Establishment

(2) Nuclear Power Engineering Test Center (NUPEC)

   (a) Tadotsu Engineering Laboratory
   (b) Isogo Engineering Laboratory
   (c) Katsuta Engineering Laboratory
   (d) Japan Institute of Nuclear Safety (JINS)

(3) Building Research Institute Tsukuba Technical Center (BRI)

4.1.1.1 Japan Atomic Energy Research Institute (JAERI)

Of the three JAERI establishments, only the Tokai facility is of particular interest within the scope of this report. The Takasaki Research Establishment was established as a R&D center on radiation chemistry. The Oarai Research Establishment consists primarily of a Material Testing Reactor, Radioisotope Utilization and Development Laboratories, Plutonium Fuel Research Laboratory and Radioactive Waste Treatment Plant.

The Tokai Research Establishment was founded in Tokai-mura, Ibaraki-ken, in July 1957, where various testing facilities have been set up, including four research reactors, a power demonstration reactor, critical assemblies and accelerators for the fundamental study of nuclear physics and reactor physics, hot laboratories and other nuclear research related facilities. A wide variety of research and development in many fields is being carried on in a comprehensive program.

Within the Tokai Research Establishment is located the Reactor Safety Research Center with a 250 person staff. The 1981 budget for the Safety Research Center was approximately $\$50 \times 10^6$. Most of the direct funding of the Reactor Safety Research Center is provided by STA and MITI. Funding comes from both regularly appropriated funds as well as from the special trust fund. The organization of the JAERI Nuclear Safety Research Center is shown in Figure 4. Recent and ongoing international cooperative programs for safety research are identified in Table 4.

4.1.1.2 Nuclear Power Engineering Test Center (NUPEC)

The Nuclear Power Engineering Test Center (NUPEC) was established in 1976 as a cooperative effort between Japanese industry represented by the electric utilities, equipment manufacturers and constructors and the government represented by STA and MITI. The primary purpose of NUPEC is to ensure a stable supply of energy for the future in Japan and to promote the reliable use and to improve the technology of nuclear energy.

The fundamental mission of NUPEC is to act as a fair, neutral and authorized agency to perform proof and verification tests on the safety and reliability of nuclear power plants using high-level technology and efficient management from the public and private sectors. As a result of these proof and verification tests, it is expected to establish the technology for nuclear power generation and to strengthen nuclear technology independence in Japan as well as to improve the availability of operation of nuclear power plants and management of quality assurance system.

The following are within NUPEC's task scope:

(1) PROOF TESTS

- (a) Seismic Proof Test on the Reliability for the Equipment and Components of Nuclear Power Plants
- (b) Proof Test on the Reliability of Valves
- (c) Proof Test on the Reliability of Pumps
- (d) Proof Test on the Reliability of Heat Affected Zones of the Welds
- (e) Proof Test on the Reliability of Inservice Inspection Technology
- (f) Proof Test on the Reliability of Fuel Assembly under Irradiation in the Reactor

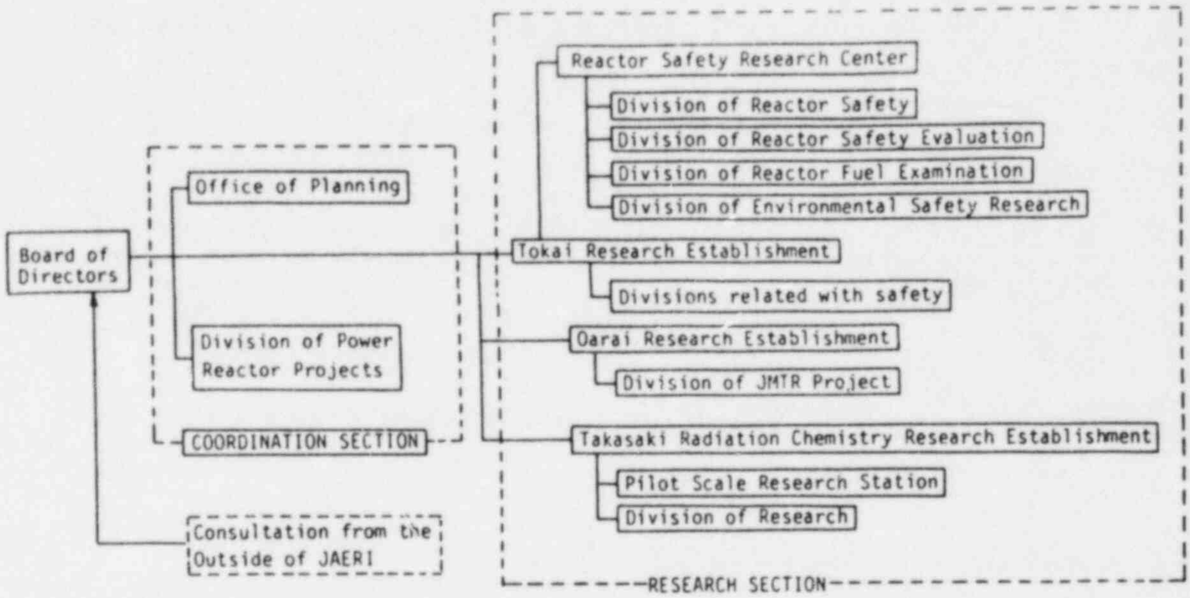


Fig. 4 Organization for Nuclear Safety Research in JAERI

Table 4 International cooperative programs for safety research

Research Items	Project Name	Host Organization	Research Objectives
Fuel Integrity Study	Halden	OECD/NEA	Fuel safety and veliability process computer application
	Studsvik	Studsvik Energiteknik AB	PCI study (Over-Ramp, Demo-Ramp I, Demo-Ramp II and Super-Ramp)
	HBEP	USDOE	High burn-up effects on FP release in fuel pellet
Integrity and Safety of Pressure Boundary Components	Steel Irradiation	IAEA	Behavior of neutron irradiation embrittlement of LWR pressure vessel steels
Engineered Safety Features of LOCA	LOFT	USNRC	PWR safety evaluation under LOCA
	2D/3D Reflooding	JAERI, BMFT, USNRC	Thermo-hydraulic behavior in refill and reflood phase of LOCA
Fuel Behavior under Accident Condition	PBF	USNRC	Fuel safety research under accident conditions
	PNS	KFK	Fuel behavior under LOCA heat up condition
	PHEBUS	CEA/Cadarache	Fuel behavior under LOCA
	NSRR	JAERI	Fuel safety research under reactivity initiated accident
Radioactive Waste Management	HLW Evaluation	IAEA	Safety evaluation of various high level waste solidified products in cold and tracer level

(g) Proof Test on the Reliability of Fuel Assemblies

(h) Proof Test on the Reliability of Electrical Instrumentation Equipment

(2) VERIFICATION TESTS

(a) Verification Test of Core Barrel for Internal Pump of BWR

(b) Verification Test of Advanced Fuel

(c) Verification Test of Seismic Analysis Code

(3) SAFETY ANALYSIS

(a) Perform Safety Analysis Calculation

(b) Improvement and Preparation of Safety Analysis Codes

(c) Collection and Analysis of Data for Safety Examination

The Tadotsu Engineering Laboratory with its 15m x 15m shaker table and large reaction wall due to be dedicated by the end of 1982 is meant primarily for seismic qualification testing. The NUPEC Isago and Katsuta Engineering Laboratories are meant to provide test facilities for mechanical and electric equipment qualification. The Japanese Institute of Nuclear Safety (JINS) which was established in 1980 as part of NUPEC by MITI and STA provides safety analysis calculations and development of safety analysis codes and data collection for safety analysis. The NUPEC organization is shown in Figure 5. The total operational funding for NUPEC in the period 1975 through 1984 is approximately $\$44 \times 10^6$ per year. The capital budget for the Tadotsu shaker table and reaction wall facility is over $\$200 \times 10^6$.

4.1.1.3 Building Research Institute (BRI)

The Building Research Institute (BRI) at the Tsukuba Technical Center provides nuclear safety research primarily in the fields of structural engineering and seismic design through the Institute of Seismology and Earthquake Engineering. The BRI has a very large reaction wall test facility as shown in Figure 6. This facility was recently used to test a full-scale seven story building structure for response due to earthquake displacements in cooperation with the U.S. National Science Foundation. The Tsukuba Science Center also has two other large shaker table facilities.

4.1.2 Industry Activities

When problems arise on particular stations requiring research, effort to resolve the research is usually funded directly by the utility affected with a research plan coordinated between STA and MITI and the

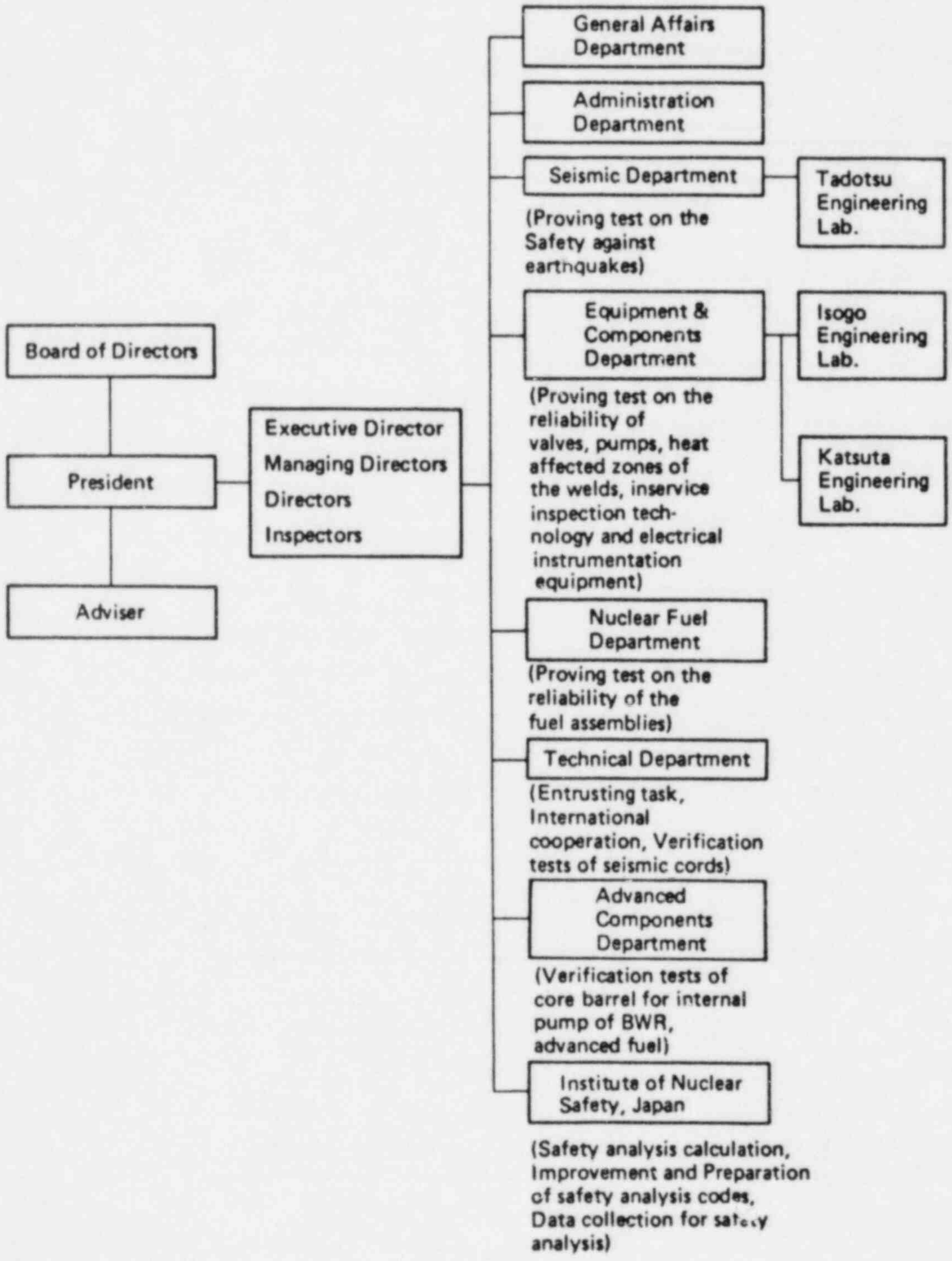


Figure 5 NUPEC Organization

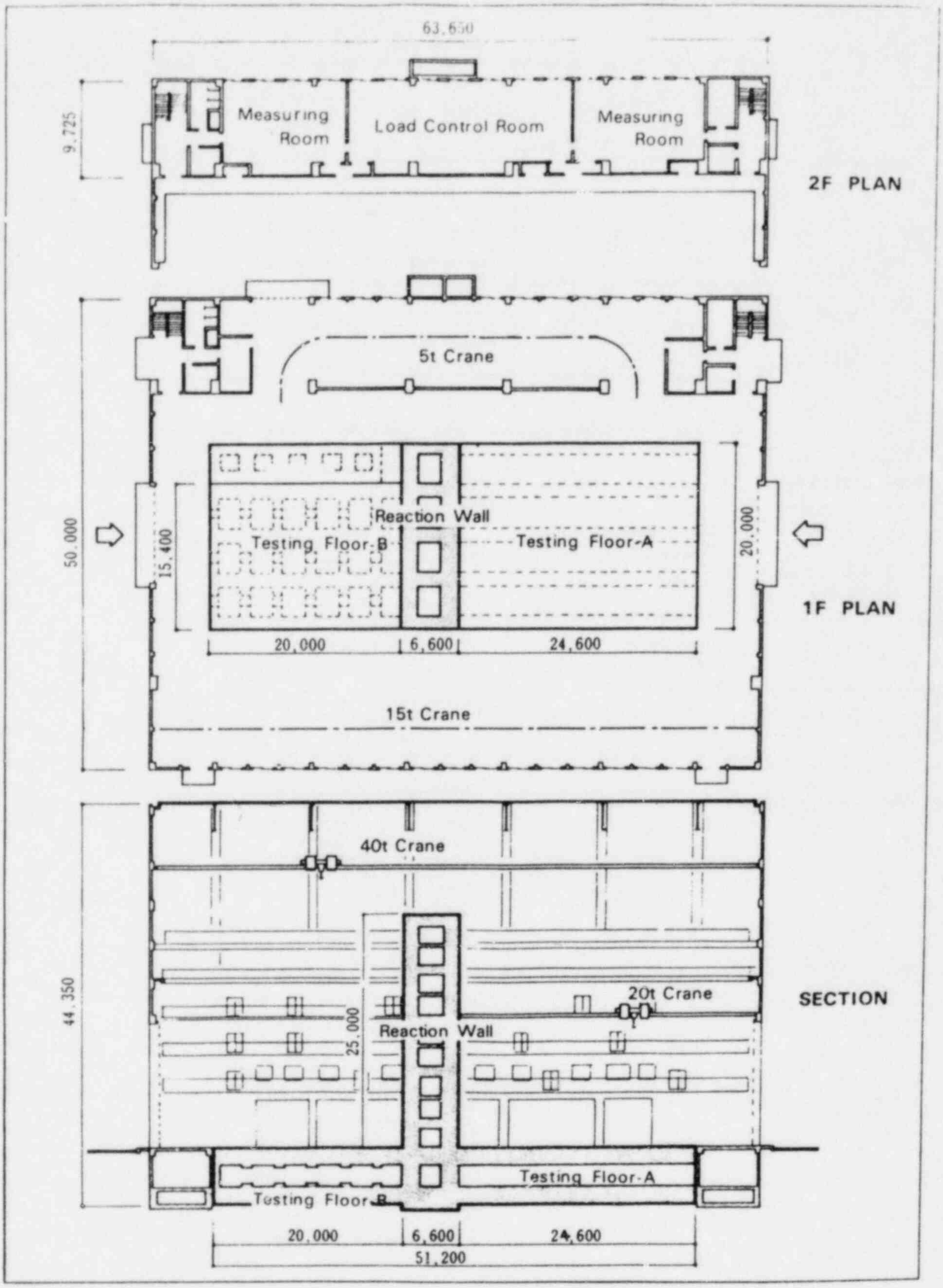


Figure 6 General Arrangement of the BRI Reaction Wall Facility

utility. When applicable, the nuclear steam system suppliers, MHI, Hitachi or Toshiba assist with funding and experimental facilities to perform the tests. In addition, generic research activities are also undertaken in much the same way utility owner's groups form in the U.S. to jointly fund research efforts affecting several plants.

The Tokyo Electric Power Company which is the largest non-public utility in the world funds approximately $\$50 \times 10^6$ of research per year. Of that amount, approximately $\$5 \times 10^6$ could be classified as nuclear safety related.

During the period 1979 through 1981, the Japanese utilities (primarily Kansai Electric and Tokyo Electric Power Companies) together with the Japanese nuclear steam system suppliers, MHI, Hitachi, and Toshiba funded and performed tests at the test facilities of MHI, Toshiba, and Hitachi to determine damping in nuclear power plant piping. Total cost of this research effort was approximately $\$15 \times 10^6$. A similar program is now underway to evaluate seismic design capabilities for nuclear power plant equipment.

Total nuclear safety-related research funded directly by industry coming primarily from the utilities is approximately $\$10 \times 10^6$/year.

4.2 Unique Facilities

4.2.1 NUPEC-Tadotsu Shaker Table

The unique facility for seismic simulation in Japan and in the world is the Large-Scale, High Performance Vibration Table located at the NUPEC Tadotsu Engineering Laboratory. This facility is due to be completed by the end of 1982. Its purpose is to perform proof tests and demonstrate seismic design adequacy of major nuclear components which heretofore have been too large or too massive to be tested. The shaker table characteristics are summarized in Table 5. The facility also includes a large 15m x 15m concrete reaction wall.

4.2.2 Building Research Institute Reaction Wall

The Building Research Institute reaction wall located at Tsukuba Technical Center is the largest in the world. The overall dimensions are shown in Figure 6 and loading facilities and testing systems in Figure 7.

4.2.3 Other

In addition to the NUPEC shaker table and the BRI reaction wall, the NUPEC-Isogo facility maintains one of the largest and most versatile environmental test facilities in the world. In JAERI at the Tokai research establishment is located the full-scale Mark-II Segment Containment test facility. This facility is scheduled to be deactivated during 1982.

Table 5 Summary of NUPEC-Tadotsu Shaker Table Characteristics

(1) Maximum Load Capacity = 1000 ton

(2) Table Test Area = 15m x 15m

(3) Simultaneous Independent Horizontal and Vertical Excitation

(4) Maximum Excitation Force - 3000 ton-$_f$.

(5) Maximum Overturning Moment: With Vertical Acceleration = 6500 T_f - m
Without Vertical Acceleration = 12000 T_f - m

(6) Maximum Acceleration: Horizontal 2.72 g (500 T) 1.84 g (1000 T)
Vertical 1.36 g (500 T) 0.92 g (1000 T)

(7) Maximum Velocity: Horizontal 75 cm/sec.
Vertical 37.5 cm/sec.

(8) Maximum Displacement: Horizontal $\pm$ 200 mm
Vertical $\pm$ 100 mm

(9) Building Floor Area: 4.100 m^2

(10) Frequency Range 0-30 Hz

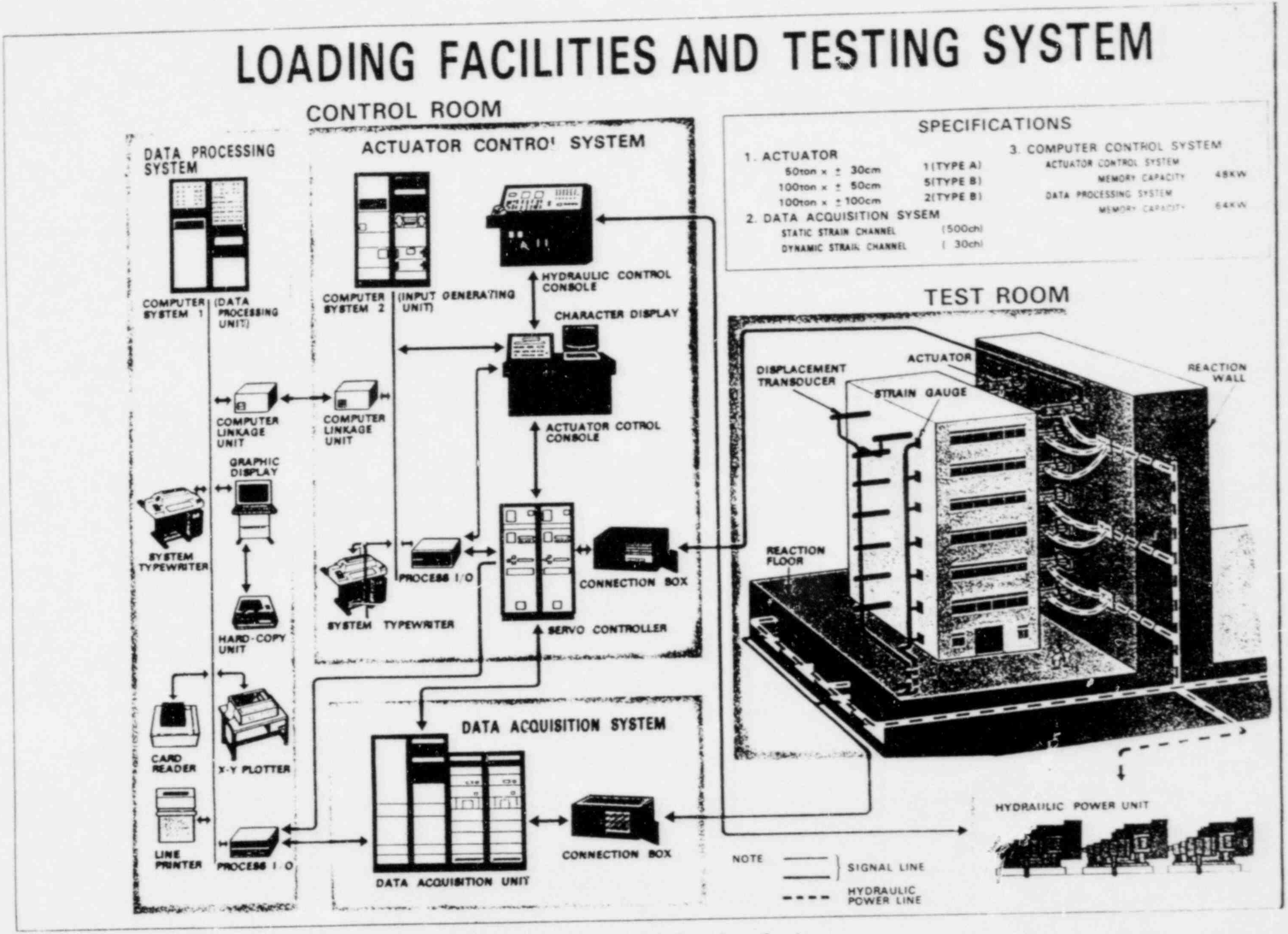


Figure 7 Loading Facilities and Testing System

In Table 6 can be found a summary of existing large seismic shaker table facilities in Japan.

Table 6 Summary of Large Shaking Tables in Japan

X-DIRECTION		
ORGANIZATION	SIZE (m)	MAXIMUM ACCELERATION
Kyoto Univ.	3 x 3	0.5 g
Tokyo Univ.	10 x 2	0.4 g
	12 x 4	3.0 (direct)
Japan Railway	10 x 2	0.4 g
CRIEPI	6.5 x 6	1.0 g
Mitsubishi (Hiroshima)	3 x 2.5	-
(Kobe)	3 x 2.5	0.7 g
Shimizu Co.	5 x 4	1.1 g
Obayashi Co.	3 x 3	1.0 g
X or Z-DIRECTION		
ORGANIZATION	SIZE (m)	MAXIMUM ACCELERATION
Science Univ. (Tsukuba)	12 x 12	X;0.55 Z;1.0 g
Japan Telephone & Telegram Co.	3 x 3	-
X + Z-DIRECTION		
ORGANIZATION	SIZE (m)	MAXIMUM ACCELERATION
Mitsubishi (Takasago)	6 x 6	0.6 g
Kajima Co.	4 x 4	-
NUPEC - Tadotsu	15 x 15	completed, 1982

5. SUMMARY DESCRIPTION OF NUCLEAR SAFETY RESEARCH FACILITIES AND PROGRAMS IN SWEDEN

5.1 General Capabilities and Organizations

Nuclear Safety Research in Sweden in the topic areas of interest in this report is usually performed under the sponsorship of the Swedish Nuclear Power Inspectorate (SKI). For specific plant or nuclear steam system safety research needs, the utility concerned with the cooperation of the nuclear steam system supplier, ASEA-ATOM or Westinghouse, may also help sponsor the research effort.

5.1.1 Swedish Nuclear Power Inspectorate (SKI)

In Sweden, the regulatory body responsible for administering the licensing process for nuclear installations and nuclear materials is the Swedish Nuclear Power Inspectorate (SKI) which comes under the Ministry of Industry. The SKI is responsible for reviewing license applications and advises the Ministry of Industry. The SKI organization has two main offices (inspection and regulations/research), one department for administration, and an information secretariate.

The primary objective of SKI is to promote nuclear safety in nuclear installations including both the facilities for handling, processing, and storage of fissionable materials and radioactive wastes and the means for transporting such materials. To achieve this objective, SKI reviews safety assessments, supervises and inspects nuclear installations and facilities with respect to nuclear safety and physical protection. The SKI also initiates, administers, and evaluates research and development within the field of nuclear safety.

The Inspectorate orders and administers safety R&D of relevance to the existing nuclear energy program and safety R&D of a more far-reaching nature. The Inspectorate sponsors safety R&D with the objectives to provide a foundation for the regulatory safety assessment, to broaden the basis for safety considerations, and to take cognizance of foreign safety R&D.

The Research Department which forms one of the four departments in the Office of Regulations and Research compiles information, opinions, proposals, and requirements from the following areas:

(1) the regulatory functions of the inspectorate

(2) systematized operating experience from utilities and vendors

(3) discussions within the advisory safety R&D reference group

(4) R&D organizations: Studsvik, ASEA-ATOM, Universities, consultants, etc.

(5) coordination with international programs

(6) participation in international working groups

(7) coordination with other Swedish sponsors: SSI, etc.

(8) discussions of priority within the R&D Department

An overall research program is formulated and approved by the Board of the Inspectorate. Before being sent to the R&D organization for performance of the R&D, the approved safety research is then broken down into definite projects after receipt of feed-back from the regulatory group, recommendations by the advisory R&D group, and approval by the Head of the Inspectorate.

The recent safety research budgets of the SKI are shown in Figure 8. The current allocation of funds for safety research by problem areas is shown in Table 7. The research may actually be performed at Universities, at the research facilities of ASEA-ATOM, or at AB Atomenergi (Studsvik). Most safety-related research is performed by Studsvik.

5.1.2 STUDSVIK ENERGITEKNIK AB

Studsvik was established in 1969 as a state-owned company conducting research and development in the energy field. The Company has a share capital of 30 million Sw.Cr. and about 900 employees. Most of Studsvik's research facilities are situated 90 km (56 miles) southwest of Stockholm.

Originally, the Company was mainly concerned with nuclear technology, but since the beginning of the 1970's it has expanded into other areas of energy technology. Studsvik's special fields include the following areas: energy-related safety and environmental questions; development of measurement techniques; process control; development and large-scale testing of materials, systems and components for optimal energy use; and consulting, license and patent services.

The Studsvik operates in the domestic and international markets, and its clients include state and municipal authorities, the power industry, fuel and equipment manufacturers, and steel and other process industries. The Company's activities are conducted through three separate operational divisions - the Nuclear Technology, Energy Technology and Technical Services Divisions.

The Nuclear Technology Division has the responsibility for Studsvik's nuclear research and service activities. This responsibility includes the operation of the Company's nuclear resources such as research reactors, laboratories and hot cells for the study and handling of irradiated materials and nuclear power components, facilities for active waste management, etc. The division is also responsible for the Company's security management and internal nuclear regulatory matters. Their activities cover consultant and R&D work for Swedish authorities, utility organizations and manufacturing industries. A large variety of

Table 7 ALLOCATION OF FUNDS FOR THE SAFETY R&D PROGRAM OF THE INSPECTORATE

PROBLEM AREA	CONTRACTS (Mill. Sw. Cr.) 1978/79	1979/80	1980/81	1981/82
1. MAN-MACHINE	1.080	1.112	2.0	3.5
2. MATERIALS	1.597	1.088	5.0	2.0
3. FUEL	1.425	2.126	2.0	3.5
4. THERMOHYDRAULICS: EXPERIMENTS AND CODES	4.985	10.441	6.5	7.0
5. COMPONENTS TESTING; INSTRUMENTATION	4.467	1.598	1.0	1.5
6. ACCIDENT AND SAFETY ANALYSIS	1.692	5.393	11.0	7.5
7. REACTOR MONITORING	.542	.455	.5	1.0
8. MARVIKEN PROJECTS	1.040	7.515	.5	1.0
9. SEISMIC MEASUREMENTS	-	-	.5	2.5
10. WASTE MANAGEMENT	.727	.255	1.5	1.5
11. MISCELLANEOUS	.244	1.132	.5	1.0
	17.799	31.115	31.0	32.0
TOTAL BUDGET	24.0	26.0	27.0	35.9

Fiscal year	1975/76	1976/77	1977/78	1978/79	1979/80	1980/81	1981/82 (proposal)
Budget (Mill Sw Cr)	7.0	14.0	18.5	24.0	26.0	27.0	35.9
(M $)	1.5	3.0	4.0	5.2	5.6	5.9	7.8

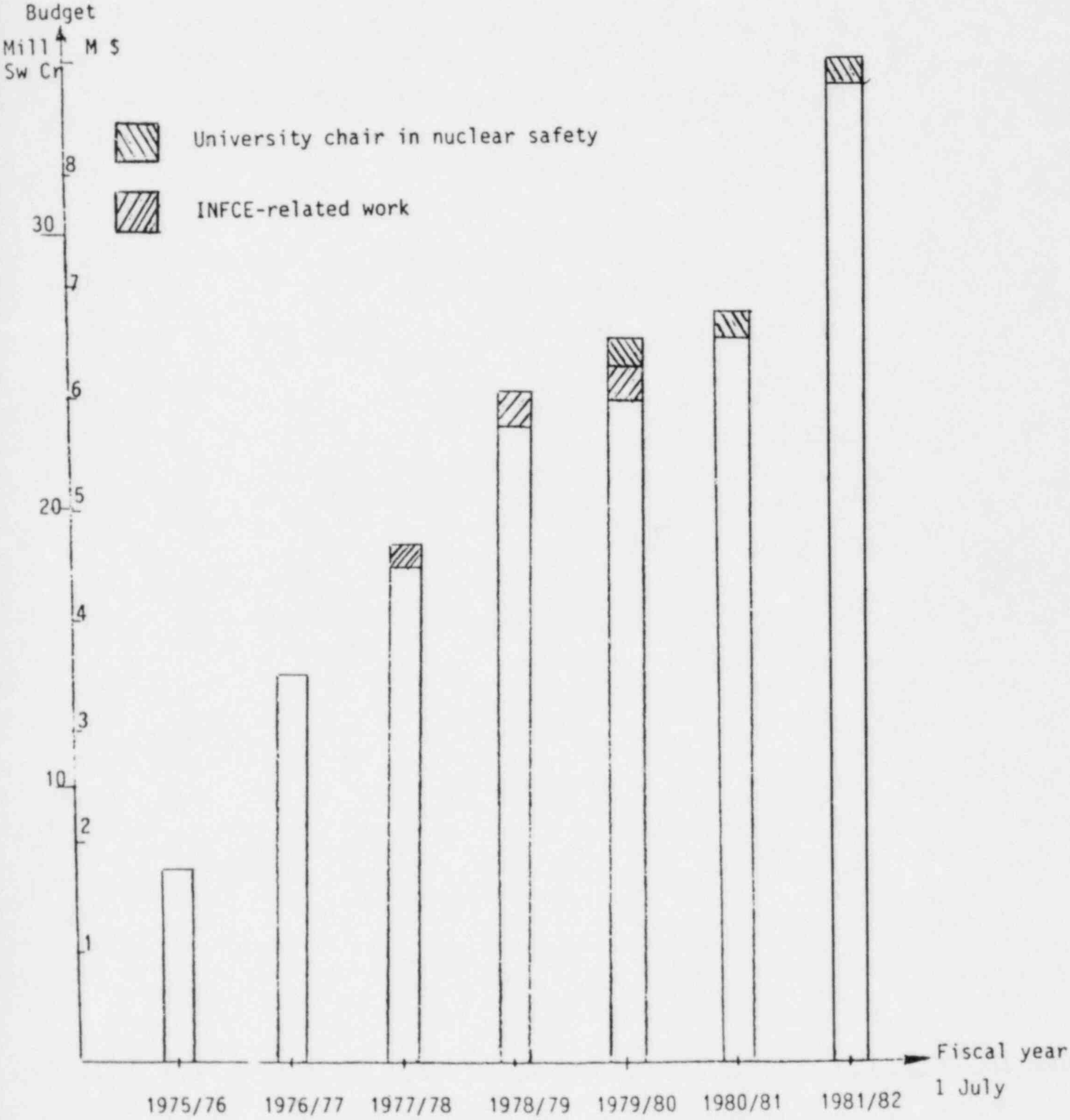


Figure 8 The Safety R&D Program Budget of the Inspectorate

radioactive services is carried out for utilities, industries, and hospitals.

Research agreements between the US NRC and STUDSVIK ENERGITEKNIK AB have been developed for the following projects:

(1) Technical exchange and cooperative arrangement in the field of nuclear safety

(2) PBF/HSST - Nordic Water Reactor Safety Research

(3) NORHAV-LOFT

(4) Aerosol behavior and filter system performance as related to vented filtered containment systems

(5) Marviken Jet Impingement Tests

(6) Fuel Testing - DEMORAMP 2

5.2 Unique Facilities

By far the largest test and research directed nuclear pressure vessel, containment, blowdown and pipe rupture facility in the world is located at the oil fired Marviken Power Station and is part of the Studsvik research facilities.

The Marviken Power Station is situated on a peninsula on the Baltic coast about 150 km south of Stockholm. The research facility area of the power plant is housed within a building of tower construction on top of which an auxiliary condenser is placed.

The test pressure vessel is situated in a pressure-suppression, PS containment. The PS containment has been modified to fit the blowdown and jet impingement test experiments. Openings have been made between drywell and wetwell and to the open atmosphere.

The pressure vessel, with its internals and auxiliary equipment such as electric power, steam generator, cooling facilities and other hardware, makes it possible to safely perform experiments on a scale which is representative of a full-size nuclear reactor plant.

Since 1972, the Marviken Power Plant has been used in internationally organized experimental programs to generate data which can impart a deeper understanding of the behavior of reactor containments and reactor system blowdown in accident situations.

The projects performed or planned to be carried out under the leadership of Studsvik Energiteknik AB (formerly Aktiebolaget Atomenergi) are designated as follows:

1	MX-I-CRT	(Containment Response Tests)
2	MX-II-CRT	(Containment Response Tests)
3	TECPO	(Theoretical Efforts on Containment Pressure Oscillations)
4 - 6	MARTIN I, II and III	(Marviken test-data interpretation)
7	MX-III-CFT	(Critical Flow Tests)
8	MX-IVT (postponed)	(Isolation Valve Tests)
9	MX-IV-JIT	(Blowdown Jet Investigation Tests)

These projects are summarized as follows:

(1) The first project (MX-I) consisting of sixteen full-scale blowdown experiments and performed during 1972 and 1973, included the studies of containment response, iodine transport, containment leakage and component behavior under loss of coolant accident conditions. The participants were from Denmark, the Federal Republic of Germany, Finland, Norway, Sweden, and the United States.

(2) The second project (MX-II) consisted of nine experimental investigations of pressure oscillations in the pressure suppression containment during simulated accidents. These experiments were performed during 1976. Additional organizations joining MX-II were from France, Japan, and the Netherlands.

(3) The third project (TECPO) was a theoretical and experimental study of the condensation pressure oscillations during blowdown related to the MX-II project. The experimental investigations comprising 29 tests were performed during 1975, 1976, and 1977 in the TESTA facility, which is a simplified small scale model of the Marviken PS-containment on the volumetric scale 1:1000. The participants were from Denmark, Finland, Norway, and Sweden.

(4) thru (6) The three MARTIN projects are performed by Studsvik Energiteknik AB in cooperation with AB ASEA-ATOM with the objective of evaluating the experimental data collected in the MX-I and MX-II projects.

(7) The seventh project, designated MX-III-CFT, was aimed at an investigation of the mass discharge from short, large-diameter pipes under critical conditions. A total of 26 tests were carried out during an 18-month period beginning late 1977. The organizations having joined the CFT-project were from Denmark, the Federal Republic of Germany, Finland, France, the Netherlands, Norway, Sweden, and the United States.

(8) The eighth Marviken project which is proposed (MX-IVT) is the full-scale testing of isolation valves for nuclear power plants. The critical flow test facility will be slightly modified to permit testing of BWR as well as PWR isolation valved. Blowdown periods of up to 10 pounds are projected with two-phase mass flow of various compositions. The isolation valve tests are mainly of a proof-type character, but the facility will clearly be available for all kinds of development work within the limitations imposed by the original design of the Marviken plant. This project has been postponed awaiting the accomplishment of a multi-national sponsorship.

(9) The fourth large-scale experiment conducted (MX-IV-JIT) has the objective of investigating blowdown jet phenomena. Two groups of tests were performed, referred to as the free jet expansion tests and the jet impingement load tests. A total of 11 tests were performed between September 1980 and October 1981. The organizations which joined the project were from Canada, Finland, Italy, Japan, the Netherlands, Sweden, and USA.

(10) Primarly because releases of radioactive materials during the TMI-2 accident were significantly lower than previously anticipated from model predictions, the Electric Power Research Institute (EPRI) has proposed that the Marviken facility be modified to measure transport of fission products and dense aerosols through a full-scale reactor primary system. The proposed 33-month program was prepared by EPRI in cooperation with Studsvik, Ontario Hydro (Canada), and KEMA (The Netherlands). Total estimated cost of the program is 31 million Swedish crowns, or about $7.5 M.

The EPRI test program is divided into two parts:

- a main test series to investigate transport of relatively dense aerosols resulting from a reactor core melt, and
- a secondary test series to similarly investigate transport of fission products released before core melting actually occurred.

The Marviken Power Plant was originally designed and built as a boiling heavy-water direct cycle reactor, with natural circulation and provisions for nuclear superheating of the steam. The facility was completed up to the light-water commissioning tests, but was never charged with nuclear fuel. For several reasons, it was decided that an oil-fired boiler should be built to feed the turbine, leaving the reactor and most of the auxiliary systems, PS-containment, reactor hall and fuel handling area essentially intact.

When it was decided that full-scale reactor safety experiments should be performed at the station, mechanical and electrical adaptions including some minor modifications of the reactor building were carried out with these experiments in view.

Prior to the MX-II project, practically all of the internal parts of the reactor vessel were removed in order to create more open flow patterns, and a special heating device in the form of an electrical steam generator was installed. During the preparations for the MX-III project, a hole (diameter 1030 mm) was cut at the center of the bottom, where a connection piece was welded. To this piece, the discharge pipe with the stop valve was connected.

A lay-out of the test assembly as designed for the MX-IV project, which has recently been completed is shown in Fig. 9.

The pressure vessel has a 5.22 m inside diameter and has a height including the top-cupola of 24.55 m. The net volume of the vessel, that is, the free water space, is 420 m^3 after the removal of internal structures. The vessel is designed for a pressure of 5.75 MPa and a temperature of 272^oC.

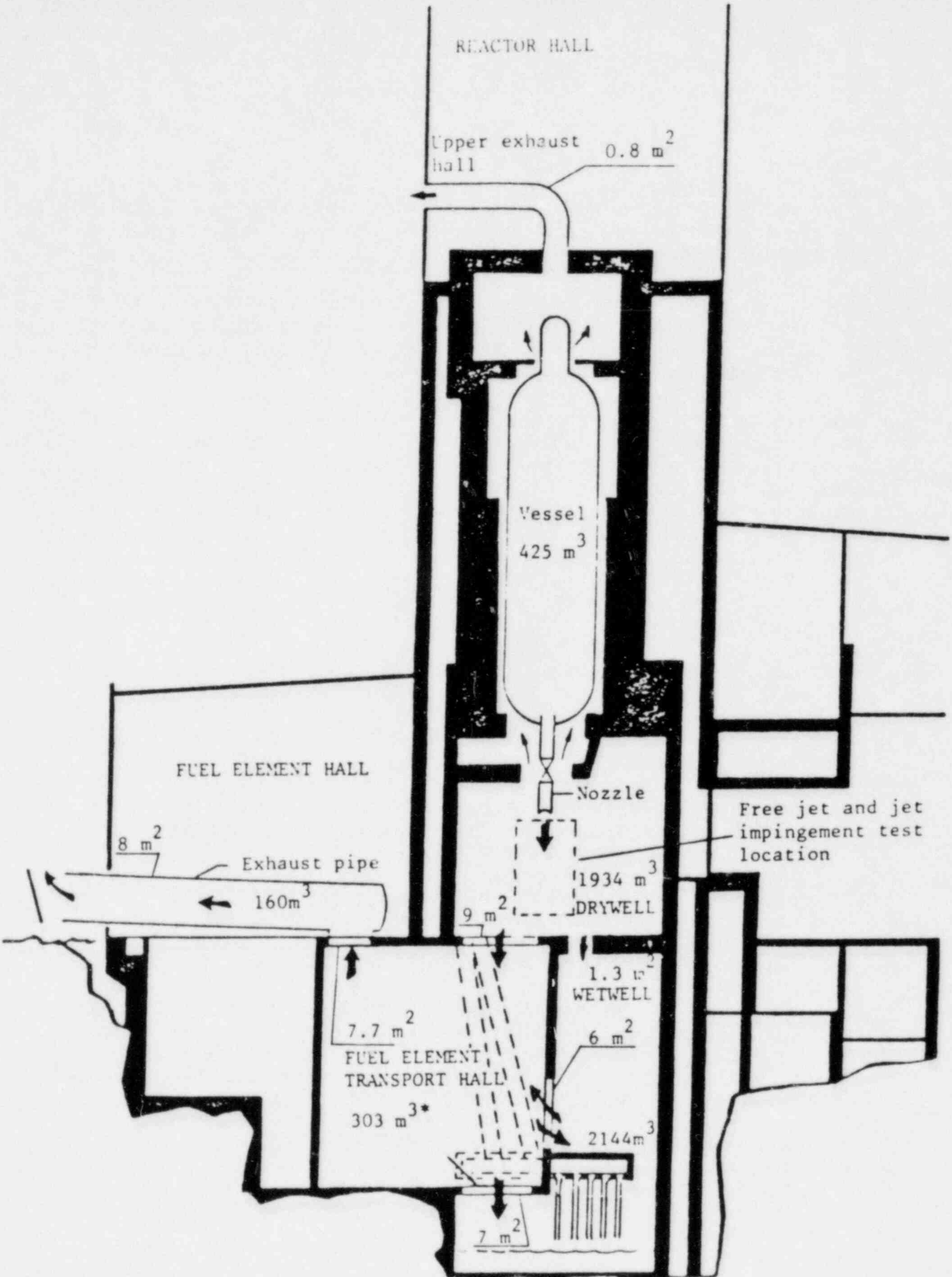
The connection piece at the bottom has a rounded inlet bolted to the upper end (inside the vessel). The inner diameter of the piece is 752 mm.

In contrary to the previous CRT-experiments, the containment now is connected to the open atmosphere, so that large steam quantities can be discharged outside the building. An overall view of the facility with the containment vent visible halfway up the wall is shown in Fig. 10.

The containment is divided into two principal spaces, the upper one called the drywell and the lower one called the wetwell. The spaces are separated by a heavy concrete floor at the ground level but communicate through a vent pipe/steam heater system and through holes made in the drywell/wetwell common wall before the MX-III project.

The main part of the drywell (lower drywell), in which the flow discharge takes place, is located underneath the pressure vessel and connects to the fuel element transport channel, which in turn has connections to the wetwell and to the open air through a discharge pipe in the fuel handling hall.

A smaller part (upper drywell) surrounds the pressure vessel and extends above the top of the vessel. From this part, there is the possibility of arranging an additional steam discharge pipe to the open air.



* Gross volume = 360 m3

Fig 9 Marviken test facility during MX-IV project

Figure 10 General View of the Marviken Test Facility

The total net volume of the drywell is 1760 m^3 and of the wetwell 2144 m^3. The design pressure is 0.41 MPa for the drywell and 0.33 MPa for the wetwell. However, re-calculations done prior to the MX-I-project indicated that a pressure of about 0.6 MPa could be accepted in both drywell and wetwell. The pressure differences across the d.w. floor must not exceed 0.12 MPa.

The heating system for the pressure vessel comprises an electric steam boiler with auxiliary pumps and pipe system. The water is taken from the bottom of the tank, by way of two pumps to the boiler, after which the steam is condensed in the tank with the assistance of water from the parallel coupled sprinkler system. The boiler can be worked up to full effect (about 5 mP) within half a minute and is capable of heating the tank from 20° to 260°C within a period of 20 - 40 hours, depending upon the content of water. If necessary, the system described above can be used to create temperature stratification in the tank.

Through the largest test nozzle (50 mm diameter) used in the MX-III project, approximately 15000 kg/s water was discharged at 30°C subcooling the vessel. Out of this amount, some 4500 kg/s flashed into steam when exposed to the pressure in the containment.

6. SUMMARY DESCRIPTION OF NUCLEAR SAFETY RESEARCH FACILITIES AND PROGRAMS IN THE UNITED KINGDOM OF GREAT BRITAIN & NORTHERN IRELAND

6.1 General Capabilities and Organizations

Nuclear safety research is usually performed under the sponsorship of one of two organizations, the Nuclear Installations Inspectorate (NII) and the Safety and Reliability Directorate (SRD) of the United Kingdom Atomic Energy Agency (UKAEA) through the Research Coordinating Committee which it chairs. Limited additional safety-related research, particularly, in the siting area, is sponsored by the Central Electricity Generating Board (CEGB).

6.1.1 Nuclear Installations Inspectorate (NII)

The Nuclear Installations Inspectorate (NII) is the nuclear regulatory arm of the Health and Safety Executive, the responsible agency for health and safety matters within the United Kingdom. The NII is responsible for all aspects of the nuclear licensing and inspection process. The total funding of the NII is approximately $9 x 10^6/year. Most of the funding of the NII comes directly from fees from the organization desiring a license to operate a nuclear facility which in the case of a nuclear power plant in England and Wales is the CEGB.

Safety research performed for the NII comes from two sources of funding. The first source is the Health and Safety Executive (HSE) research budget which are appropriated government funds. The total HSE Safety Research Budget is approximately $15 to 20 x 10^6/yr. with less than 20 percent being spent on nuclear safety research.

The second source of safety research funding for water reactors within the NII is of particular interest since this funding is generated by the individual inspectors in the various branch sections. Initially, the individual inspector identifies a need for some safety research effort. Next, available technical resources to perform the research (usually universities) are determined. The inspector then submits a request through the section manager and, with the manager's concurrence, a procurement authorization to perform the research is issued by the branch manager. The total procedure just outlined typically takes a maximum of two weeks. The total funding for this source of safety research is approximately $2 x 10^6/yr. for PWR research with an additional approximate $0.4 x 10^6/yr. being spent on siting research. Funding for this research is supplied directly by the licensee as an identified research part of quarterly payments which are paid to the NII from the CEGB licensee.

6.1.2 Safety and Reliability Directorate (SRD)

The origins of the Safety and Reliability Directorate stem from a re-examination or the organization for the control of health and safety in the United Kingdom Atomic Energy Authority (UKAEA) following an accident at the Windscale Plant in 1958. As a result of the lessons learned from this incident, the UKAEA set up a Health and Safety Branch in 1959, including a Radiological Protection Division at Harwell and a Safeguards Division at Risley. In 1971, the Safeguards Division became the Safety and Reliability Directorate, with its Director reporting directly to the Chairman of the UKAEA.

In 1976, it was agreed that in addition to its other work, the SRD would carry out scientific research into safety and reliability on behalf of the newly formed Health and Safety Commission. To this end, a new management board was set up to determine priorities for the Directorates activities and insure adequate allocation of resources.

The following are the formal functions of the Directorate:

(1) To advise the UKAEA on the formulation of their safety and reliability policy and to disseminate this policy for application by heads of management units.

(2) To apply this policy to the assessment and inspection of UKAEA reactors and plants (including laboratories).

(3) To coordinate and direct general reactor safety research in support of the nuclear power program as a whole and to undertake safety research in support of individual reactor systems.

(4) To carry out research and development work relating to safety and reliability on behalf of the Health and Safety Commission.

(5) To provide advice and services as required to the Ministry of Defense, and to other departmens and organizations at home and abroad on safety and reliability, including the safe transport of nuclear materials and the development of national and international standards.

(6) To provide a reference point for the Authority's external relations in the field of safety and reliability.

The SRD is the focal point of the UKAEA's external involvement in safety research and in its collaboration with other research organizations in the UK and overseas. This collaboration includes bilateral cooperative projects with countries such as US, France, Germany and also involvement in the work of international organizations such as the International Atomic Energy Agency (IAEA), Nuclear Energy Agency (NEA) and European Economic Community (EEC).

The SRD is currently performing PWR associated research at a total funding level between \$40-50 x 10^6/yr. Approximately eighty-five percent of this funding comes directly from UKAEA appropriations, the remainder is from the CEGB and British Industry representatives. The research effort is divided into two major programs. The first, titled the PWR Safety Research Program, is funded at the \$30-40 x 10^6/yr. level, and the second, called the General Safety Research Program, is funded at the \$10 x 10^6/yr. level. The General Safety Research

Program has joint research with the Federal Republic of Germany and France in the area of missile impact. In addition, collaborated efforts in the areas of fuel-air detonation and deflagration are ongoing with British and French Industry.

The SRD, also, has established two Special Topic Working Parties on Seismic and Missile Studies. Research is often conducted by so-called "working groups" within the SRD, whose participants periodically meet and confer on specific research topics. Few topical reports are issued, research results usually being documented in the form of annual progress reports. The existence of a working party may in some cases be the sole activity related to a particular topic. For example, there are currently no specific research projects related to seismic safety, only a working group.

It is interesting to note that the SRD performs its PWR safety research without formal coordination with the NII. This results in two different government agencies performing nuclear safety research essentially independent of each other. The SRD tends to use the personnel and laboratory facilities of the UKAEA while the NII tends to use university research facilities. The SRD, also, coordinates its safety research activities through the Research Coordinating Committee, which it chairs, and has representatives from the CEGB, NNC and NFC. The NII for water reactor safety research is not represented on the Research Coordinating Committee.

6.1.3 Central Electricity Generating Board (CEGB)

The CEGB is the national government-owned electric utility serving England and Wales. Together with the United Kingdom Atomic Energy Authority (UKAEA), the Nuclear Installations Inspectorate (NII), and the National Nuclear Corporation (NNC), the CEGB is also one of the principal sources of research funding in the United Kingdom. In the British nuclear licensing process, the CEGB decides on the final suitability of nuclear power plant sites subject to NII acceptance and performs safety reviews of nuclear power plant designs submitted for licensing. The CEGB is heavily involved in developing general siting criteria and performing siting studies and is also the primary responsible party for research related to operating plants.

The CEGB maintains three research centers at Letterhead, Berkley and Marchwood with the Berkley facility devoted primarily to nuclear research. The research performed directly by the CEGB in their own laboratories tends to concentrate on the operating and siting areas. The development of safety research priorities is a committee effort to which the CEGB is a party. The research coordinating committee is chaired by a representative of the UKAEA Safety and Reliability Directorate with members from the CEGB, NNC and BNFL.

6.2 Unique Facilities

None of the currently existing U.K. research facilities in the topic areas of interest in the study are unique in that facilities with their general capabilities are not otherwise available elsewhere in the five other countries surveyed or in the U.S.

The UKAEA has established a reactor safety test compound at the Atomic Energy Establishment, Winfrith, Dorset. This facility is meant as the prime test center for large-scale safety experiments undertaken by the UKAEA.

The first experiments started in December 1974 in THERMIR and since then a number of other plants have been brought into operation. Current activities on the RST Compound include the following:

(1) THERMIR (thermal interaction rig) - propagation effects in explosive metal/water interactions.

(2) CMR (containment modeling rig) - effect of a rapid energy release on reactor internals and containment vessels.

(3) THERMITE Rig A - rapid heat transfer from molten UO_2 to water.

(4) THERMITE Rig B - rapid heat transfer from molten UO_2 to liquid sodium.

(5) THERMITE FIRING RIG - development and firing of thermite charges producing up to 20 kg molten UO_2, and studies of heat transfer to trapped sodium.

(6) PCBR (pressurized circuit bursting rig) - consequences of over pressurization of deliberately weakened reactor components.

Additional facilities for safety experiments have been constructed in an unused part of the ZENITH reactor pit. These are as follows:

(1) MISSILE LAUNCHER LABORATORY - for studies of the damage produced by missile impact on containment structures.

(2) CMR 2 - a second facility for studying the effects of rapid energy release on structures and containment vessels.

In all cases, it is the aim of the experiments to reach a basic understanding of the physical phenomena involved and, hence, to lead to the development of validated calculational methods for plant design and safety assessment. The experimental activities are supported by extensive design, manufacturing, chemical, control and instrumentation, and computing facilities which are available on the Winfrith site.

7. SUMMARY DESCRIPTION OF NUCLEAR SAFETY RESEARCH FACILITIES AND PROGRAMS IN THE FEDERAL REPUBLIC OF GERMANY

7.1 General Capabilities and Organizations

Most nuclear safety research in the FRG is performed under the sponsorship of one of two organizations, the Federal Minister for Research and Technology (BMFT) and the Reactor Safety Division of the Federal Ministry of the Interior (RS-BMI).

7.1.1 Federal Minister for Research and Technology (BMFT)

Investigations on the safety of Light Water Reactors (LWR) being performed as part of the Research Program Reactor Safety (RS-Projects) are sponsored by the Federal Minister for Research and Technology (BMFT). The objective of this program is to investigate in detail the safety margins of nuclear power plants and their systems and the further development of safety technology.

The Reactor Safety Association (GRS), by direction of the BMFT, summarizes the status of such investigations by quarterly and annually publication of progress reports within the series GRS-F-Fortschrittsberichte (GRS-F-Progress Reports). Each progress report represents a compilation of individual reports about objectives, the work performed, the results, projected work, etc. The individual reports are prepared in a standard format by the contractors performing the work as documentation of their progress in work and published by the FB (Research Coordination Department), of the GRS, within the framework of general information of the progress in reactor safety research.

The individual reports are arranged according to the same classification systems being used in the Nuclear Safety Index of the CEC (Commission of the European Communities) and the OECD (Organization for Economic Cooperation and Development).

The BMI Section RS-I-3 acts as the liason between the BMFT and BMF in nuclear research and development to insure consistency and to avoid duplication of efforts.

7.1.2 Reactor Safety Division, Federal Ministry of Interior (RS-BMI)

In the FRG the individual States or each of the 11 Lander has the task of acting as the Supreme Licensing Authorities for nuclear proceedings. The Federal Government in the form of the BMI has the responsibility for overall technical supervision of the States.

One major area of responsibility of the RS-BMI is regulatory research. In addition to its licensing and regulatory function, the RS-BMI coordinates research programs in support of safety questions generated during licensing and operation of nuclear facilities. Since 1972 when nuclear regulation was separated from nuclear development, the Bundesministerium fur Forschung und Technologie (Federal Ministry of

Research and Technology or BMFT) has had the prime responsibility for energy research and development activities. However, the RS-BMI also requests and funds safety research. That research which is site or plant specific is paid by the utility plant owner. Generic safety research is funded by the Interior Ministry. Safety Research sponsored directly by the BMI amounted to approximately $\$17 \times 10^6$ in fiscal 1981.

Safety research is performed where the expertise is considered to exist. Thus, research may be performed by the nuclear steam system supplier, utility, government research facilities, universities or private contractors. No conflict of interest is seen by having the nuclear industry perform safety research since the data generated are normally evaluated independently. The safety research efforts of the BMI are periodically coordinated with the Ministry for Research and Technology who also funds development as well as nuclear safety research. Total nuclear research funding in 1981 by the Ministry of Research and Technology was approximately $\$65 \times 10^6$. The technical aspects of safety research for both the BMFT and RS-BMI are coordinated through the Gesellschaft fur Reaktorsicherheit (GRS). Annual reports are published on Reactor Safety Research Projects Sponsored by the Ministry for Research and Technology by the GRS.

7.1.3 Reactor Safety Associates (GRS)

The GRS which has two site locations, one in Cologne and the other in Garching near Munich, serves to provide technical support to both the BMFT and RS-BMI. In the Garching location, it also provides safety research test facilities. Currently, the GRS in Cologne and Garching has approximately 400 engineers. The GRS, in addition to providing technical services to the BMFT and the RS-BMI, also assists, as requested, the State Licensing Authorities and the individual TUV. It will also undertake funded research projects for industry.

The GRS receives some 30% of its total funding from the BMFT and another 30% of its funding, which is directed to nuclear research, from the BMI. Therefore, channels for U.S. NRC interaction with GRS research studies in the FRG are already in place through the NRC's existing agreement with BMFT.

The GRS also receives about 30% of its total funding from the German Technical Inspection Agencies (Technische Uberwachungsvereine, or TUVs) for work related to licensing topics. The remainder of its funding typically comes directly from German industry. As a result of the GRS's involvement with both government and industry funded research, it is in an excellent position to coordinate nuclear safety research in the FRG.

7.2 Unique Facilities

7.2.1 Meppen Missile Test Facility

The Meppen test facility, located approximately 200 km north of Cologne, is operated by the Bundeswehr; in fact, the actual missile tests are conducted by military technicians. Two basic projectile types have been considered to date, both nominally 600 mm in diameter and 6 m in length, each weighing approximately 1 ton (1000 kg). The projectiles are fired from a methane-gas rail gun and achieve typical impact velocities of between 150 and 250 m/s. The highly deformable projectiles typically crush to 20 - 35% of their original length. The impact of rigid missiles (addressing, for example, turbine missile impact) is not a part of the present study. In the U.S., the missile test facilities have similar test capabilities but a somewhat different missile propulsion system and to date have tended to concentrate more on rigid missile impact.

Two test series were defined for the Meppen program. Series I, which evaluated the force-time behavior of the two missile types for various impact velocities, used a "rigid" target consisting of an instrumented steel impact plate mounted on a suspended 3.7 x 3.5 x 3.0 m reinforced concrete slab. A total of eight Series I tests were completed and analysis of the test results is now in progress. Series II, which consists of a total of 18 tests, is investigating the kinetic bearing capacity and failure behavior of suspended 6.5 x 6.0 m reinforced concrete slabs of 40 mm to 70 mm thickness.

The current Series II will continue through the end of 1982. At present no "Series III" is planned. However, any future work would be expected to include the following:

(1) Extended evaluation of test results from Series I and Series II.

(2) Discussion with Winfrith (United Kingdom) of new projectile types. The Winfrith 6" air gun makes possible relatively inexpensive smaller scale tests on a wide range of missile and target characteristics.

(3) More extensive finite-element calculations to determine crack behavior (for example, crack patterns).

(4) Cooperative interaction with Interatom during the last two Series II tests to investigate global effects transferred to the containment building (that is, as opposed to local failure).

A schematic sketch of the Meppen missile test arrangement is shown in Figure 11.

If the test program is continued into a Series III, it will most likely concentrate in the area of building vibrational response.

7.2.2 Heissdampfreaktor (Superheated Steam Reactor) Kernforschungszentrum Karlsruhe

The Heissdampfreaktor Safety Program is a broadly based nuclear power safety research program supported by the German Federal Ministry of

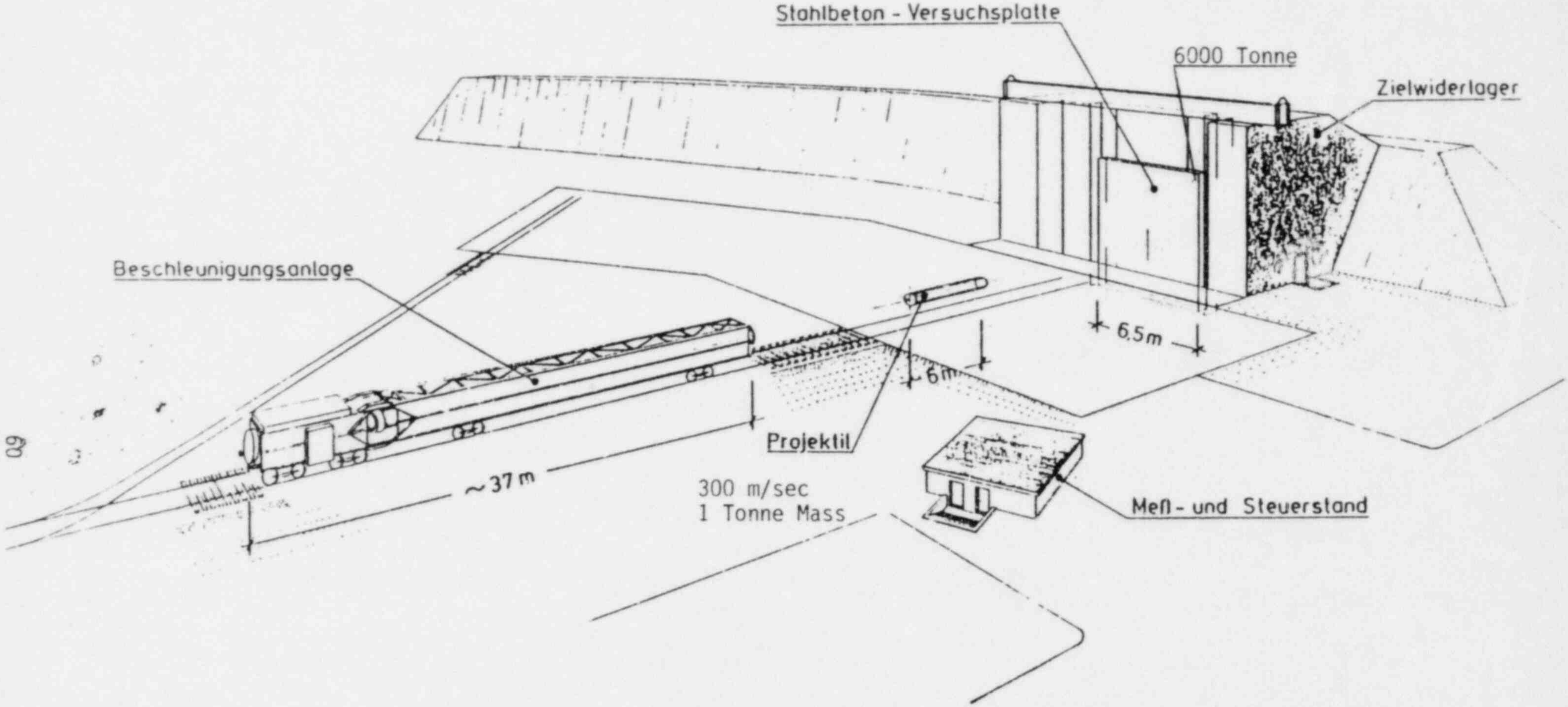


Figure 11 General Arrangement of Meppen Missile Test Facility

Research and Technology (BMFT) and executed under the direction of Kernforschungszentrum Karlsruhe Gmbh (Karlsruhe, Federal Republic of Germany).

The Heissdampfreaktor is a decommissioned small superheated steam power reactor sited about 50 mm (30 miles) each of Frankfurt, FRG on the Main River near Kahl, FRG. In size, the reactor pressure vessel is considerably smaller than the Marviken facility but has a significantly higher pressure rating. The power facility was made available as a research facility and PHDR was initiated in 1974 with the first set of on-site experiments (seismic) being executed in late summer, 1975. The HDR has major experimental activities within the containment building scheduled into 1984. Figure 12 illustrates a general view of the HDR facility.

The research plan includes thermo-hydraulics, structure dynamics, fluid-structure interaction, and fracture mechanic and nondestructive testing studies. Physical components in the research program include the reactor containment building, the reactor pressure vessel, the reactor core barrell, piping systems, selected safety valves (feedwater check and steam isolation valves), and selected safety tanks. The HDR loading conditions include the following:

- pressurized heated reactor system states - 115 bar (1670 psi), 310°C (590°F);
- elevated pressure reactor system states - 150 bar (2175 psi), 50°C (120°F);
- blow down loading, including jet impingement;
- earthquake loading/structural dynamic testing; and,
- thermoshock loading.

The U.S.NRC currently maintains direct liason with this project through the Lawrence Livermore Laboratory which has a engineer stationed at the site.

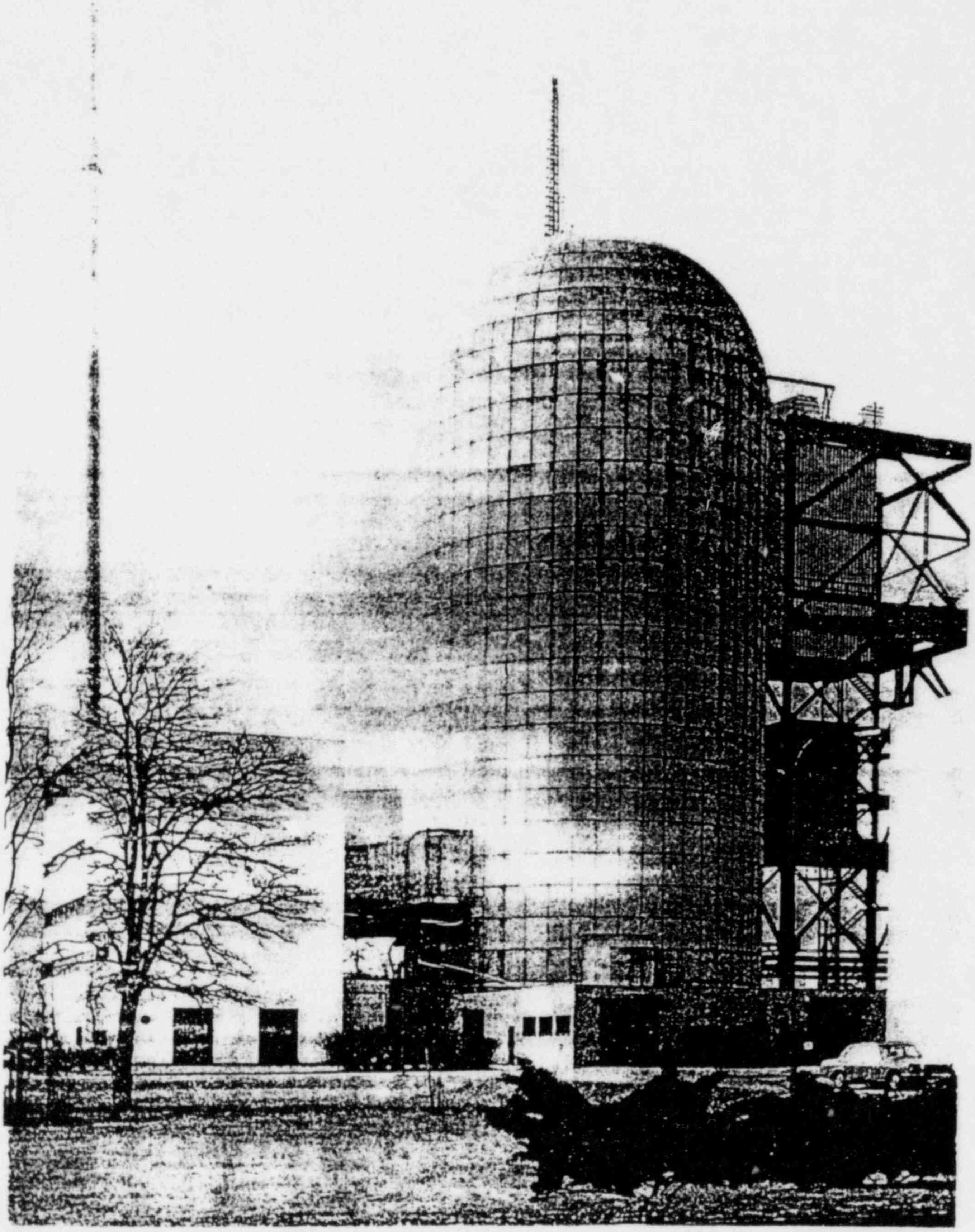


Figure 12 General View of the HDR Test Facility

8. RECOMMENDED AREAS OF POTENTIAL JOINT SPONSORSHIP OF RESEARCH PROGRAMS

8.1 Existing Multinational Nuclear Safety Research Programs and Facilities (JRC)

Before proceeding to recommendations concerning the potential for multinational sponsorship of safety research in the areas of siting, structural and mechanical engineering, materials and metallurgy, existing multinational programs and facilities should be identified. The Joint Research Center, Commission of the European Communities, Ispra Establishment, (JRC), has 5 major projects currently underway.

(1) Reliability and Risk Evaluation

(2) LWR Loss-of-Coolant Accident Studies

(3) Primary System Integrity

(4) LMFBR Core Accident Initiation and Transition Phase

(5) LMFBR Past Disassembly Phase

8.1.1 Primary System Integrity

Of these five projects, project (3) is of particular interest within the scope of this report. This project is broken down into the following tasks:

(1) Failure Detection in LWR Primary Circuit Components - PISC, Program for Inspection of Steel Components

The PISC I Program for the inspection of steel components was primarily conducted to assess the capability of the ultrasonic procedure prepared by the Pressure Vessel Research Committee (and based on the ASME Code Section XI) to detect, locate and size flaws or discontinuities in welds or heavy section steel plates. The results have shown the need for improvement in reliable detection of faults. Alternative ultrasonic techniques have already been applied by some of the PISC program participants showing that high sensitivity techniques (focussed beam probes, multiple orientation of the ultrasonic beams) improve the detection probability and correct sizing of defects substantially.

Based on these results, the PISC II was prepared in 1980 and endorsed by the CSNI (OECD). The JRC is "operating agent" of the program and is responsible for its management. The objectives of this activity are summarized as follows:

(a) evaluate the effectiveness of current and advanced non-destructive techniques for plate inspections and in-service inspections of reactor pressure vessel components with respect to in-service induced flaws.

(b) identify techniques for pre-service inspection and in-service inspection (ISI) which could be generally accepted, and

(c) bring the conclusions of the program to the attention of the Code, Standard and Regulatory bodies concerned with ISI.

As a complement to the round robin tests, a number of parametric studies are being conducted, addressing in particular the questions of: defect position and geometry, equipment characteristics, effects of vessel cladding and residual stresses. An important aid to this exercise is being given by the JRC Non-Destructive Techniques Laboratory at Ispra, which is engaged in equipment characterization.

(2) Models Development to Assess the Probability of Failure of LWR Primary Circuit Components

In order to get maximum operational safety of nuclear power plants, all reactor components are extensively tested and monitored prior to and during service. Using the data continuously supplied by these inspections and estimating the load function for a given time period, an updated calculation of the reliability (in terms of residual life) of pressure vessels and piping systems is provided.

To perform these calculations, the JRC is developing the code COVASTOL. Work has been concentrated in 1980 on the complete calculation of probability for the onset of unstable crack propagation in the welds of a PWR vessel for defects having widths from 3 to 18 mm and lengths from 8 to 2000 mm, located in any position through the thickness. The probability of existence of the defects and the probability of occurrence of LOCA accidents have been considered.

Due to the still limited experience in this field, an attempt will be made to show in a few integral small scale reactor vessel tests, the validity of the overall end of life prediction procedure. Acoustic emission techniques are being developed for continuous monitoring of crack growth and available NDT techniques will be used for periodic monitoring of crack localization and sizing.

8.1.2 Unique Facilities

The planned biaxial dynamic testing program located at the JRC facility at Ispra shown in Figure 13 currently has uniaxial dynamic load capability of 500 Tons with strain rates of $10^4 \times 10^3$ mm/sec. Biaxial test capabilities must await future appropriation of funds which will not be available until the 1984 budget. The facility uses the principal of elastically stored strain energy in high strength cables approximately 100 meters long to provide the large dynamic input.

The facility when completed would seem to have unique high load capacity biaxial dynamic loading capabilities. While strain rates are not compatible with the high energy initial missile impact or blast wave range, the facility would appear to be able to evaluate material properties in full size steel and concrete material specimens in both tension and compression at some distances from the impact point. In addition, with relatively minor modifications, it would appear the facility would be able to install a impact test table which would be able to economically evaluate the response of large mechanical and electrical equipment to high frequency, high g level impulse loadings. Such a facility would be able to quantify the damage differential between high and low frequency cyclic input to mechanical and electrical equipment.

8.1.3 Special Facilities with Existing Multinational Research Programs

As explained in Section 1.1 of this report, multinational nuclear safety research programs tend to be attracted to unique facilities not generally available within the countries desiring to have the results of the research. The facilities described in this report outside the U.S. which have attracted such multinational programs are identified as follows:

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

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

(3) Heissdampfreaktor - F[illegible] -
Seismic response of bu[illegible]dings and equipment, blowdown

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

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

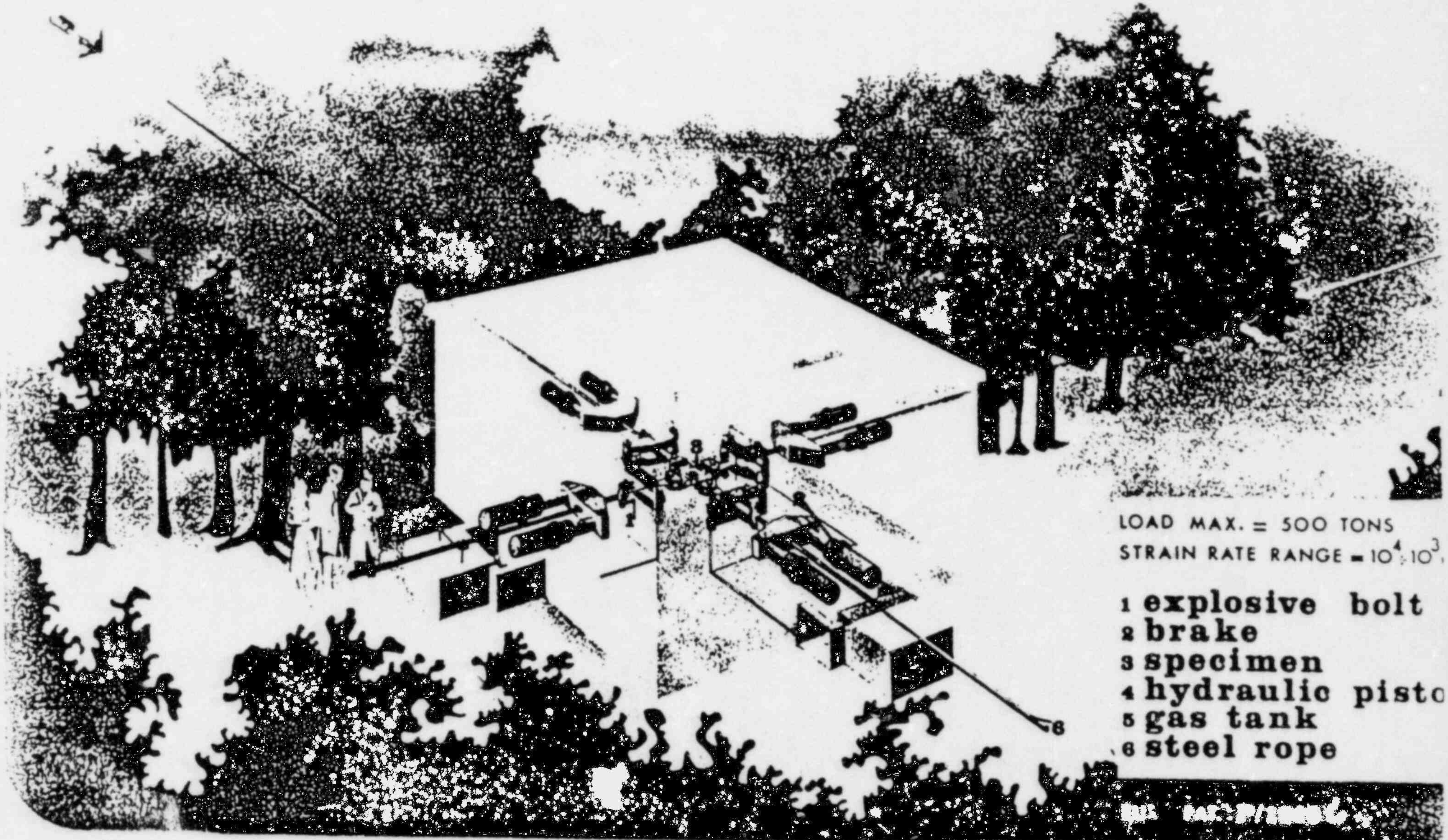


Figure 13 Biaxial Dynamic Testing Facility

8.2 Special Facilities with Potential for Multinational Research Programs

Currently, there are a number of special facilities not being used in multinational testing available in foreign countries which should be of interest in planning future safety research programs. These are listed as follows:

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

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

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

(4) NUPEC - Todatsu Seismic Test Facility - Japan - Full Scale Shaker Table Test Facility See also Table 6 for other Shaker Table Facilities in Japan

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

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

(7) 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

8.3 Suggested Projects for Multinational Research Sponsorship

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

8.3.1 Siting

8.3.1.1 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.

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

8.3.2 Structural Engineering

8.3.2.1 Seismic

There is a particular concern among all the countries surveyed with regard to 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 auxilary safety systems has recently become a major concern.(Ref. 10) 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 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

(3) Development of mean and lower bound damping data for structures and structural elements 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.

8.3.2.2 Containment Design

Nuclear containment vessels and structures have been designed such that under relatively high pressure (1 to 5 bar) they remain essentially leak-tight and 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. 11) 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, it is not clear how much of the test details will be available outside Japan.

Tests are about to start in Sandia National Laboratories under U.S. NRC sponsorship (Structural Safety)(Ref. 11) 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 $\$15 \times 10^6$. 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. Several countries have expressed an interest in the Sandia containment overpressure tests. Particularly strong interest was expressed in the U.K. (NNC) and France (Saclay).

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 their programs.

8.3.2.3 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 structure as a result of impact loads from water and steam hammer, sudden valve closure, suppression pool response to dynamic loads and impact from missiles. 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.

8.3.2.4 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 surveyed should be interested in this research area.

8.3.3 Mechanical Engineering

8.3.3.1 Seismic

In the area of Mechanical Engineering as well as in the Structural Engineering area, the Japanese have recently completed and are currently conducting or are about to start very large research programs in seismic qualification of mechanical 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

(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.

8.3.3.2 High Frequency Load

As discussed in Section 8.3.2.3, the resultant high frequency ringing of structures subjected to large impact loading using current spectral analysis methods 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.

8.3.3.3 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 have 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

(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.

8.3.4 Materials and Metallurgy

Through the JRC administered PISC Program, 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. See Appendix A for a more detailed review of the current status of safety research in this scope area.

8.4 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 share the total scope of a particular project or a trade off or exchange of information in unrelated areas can be negotiated. 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 8 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.

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Table 8 Summary of Candidate Program for Multinational Sponsorship with the U.S. NRC

A. Siting -

1. Small Aircraft Crash Probabilities and Effects

B. Structural Engineering -

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

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 and Structural Elements at High Seismic Stress Levels

4. Cooperative Agreement with NUPEC-Tadotsu, Japan Shaker Table Seismic Testing of Containment

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

6. High Frequency Cyclic Load Transmission through Structures

7. Research on Global and Local Ductility Limits on Structures

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

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

Table 8 Summary of Candidate Program for Multinational Sponsorship with the U.S. NRC (continued)

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 and Global and Local Ductility Limits on Equipment Distribution Systems and Their Supports

D. Materials and Metallurgy -

See Discussion Given in Appendix B

APPENDIX A

REVIEW OF SELECTED FOREIGN RESEARCH AND DEVELOPMENT IN FRACTURE MECHANICS, NONDESTRUCTIVE EXAMINATION, IN-SERVICE INSPECTION, AND PROPERTY DEGRADATION MECHANISMS

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INTRODUCTION

This report is a compilation of data obtained from selected countries into four categories, fracture mechanics, nondestructive examination, in-service inspection, and physical property degradation mechanisms. The preparation of this report was undertaken as part of a selected review of foreign safety research for nuclear power plants funded by the U.S. Nuclear Regulatory Commission.

The report includes the opinions and viewpoints of the authors as well as data furnished by others. Data compiled include information obtained from six countries (Canada, Federal Republic of Germany, France, Japan, Sweden, and the United Kingdom of Great Britain & Northern Ireland.)

SCOPE

Within the four categories listed (fracture mechanics, nondestructive examination, in-service inspection and mechanical property degradation mechanisms) two major problems are identified:

> In the context of failure mechanisms, can the reactor pressure boundary be subdivided into classes of systems or components with their own unique problems? (Obvious examples are the reactor pressure vessel, primary piping systems, steam generators and safety-related secondary piping systems).
>
> What mechanisms leading to degradation and/or failure of various parts of the reactor pressure boundary are postulated and are they considered of reasonable probability?

Given the four selected categories and the two basic problems identified, a number of additional questions arise. These are:

1. Assuming certain cracking or degradation mechanisms what is the current status as to reliability of detection of cracks in systems; for example, RPV, thick or thin ferritic or austenitic piping? What work is underway to improve the reliability of detection?

2. If flaws are detected, how well can they be sized and located in the various systems? What procedures offer the best reliability for sizing and location?

3. Recognizing that detection and sizing of flaws may not be completely reliable, what supplementary information is available on crack growth rates for various components and systems and materials-related mechanisms leading to a degradation in properties? (Obvious examples are irradiation embrittlement, hydrogen embrittlement, temper embrittlement, strain aging, and long-term aging).

4. Are there improved analytical techniques permitting a fracture mechanics analysis of the systems? (Ideally LEFM, EPFM, GYFM techniques should be available, preferably capable of use in both the deterministic and probabilistic modes).

5. What foreign research and development projects are completed, in progress or under consideration that will complement, supplement, or conceivably, replace techniques used in the U.S. What foreign projects are candidates for possible multinational support.

The purpose of this report is to present answers to question five considering the other problems and questions here enumerated.

REPORT ORGANIZATION

This report is divided into five major sections; Introduction, Scope, Organization, Overview, and a Country by Country Review. The Country by Country review includes a review of ongoing work being performed in the European Community through CEC and OECD. The General Overview is itself sub-divided into four categories:

- In-Service Inspection
- Nondestructive Examination
- Fracture Mechanics
- Probabilistic Fracture Mechanics

GENERAL OVERVIEW IN THE VARIOUS COUNTRIES

IN-SERVICE INSPECTION

In-service inspection is a requirement for continuing operation of nuclear power reactors in most or all countries having such reactors; however, the approaches vary from country to country. The United States approach will be used as a benchmark for purposes of comparison.

U.S. - ASME XI somewhat modified by 10CFR50.55 has the force of law. Many states adopt the various ASME Codes and require compliance. The USNRC accepts ASME XI with some limitations covered in Regulations or Regulatory Guides or Technical Specifications.

Japan - Tends to adhere closely to ASME XI with specific exceptions, such as DAC levels; for example, 20% not 100%.

Canada - Uses ASME XI as a basis for their ISI.

Sweden - Their ISI is modeled on ASME XI. They now have official "Rules for In-service Inspection of Nuclear Power Plants," April 1, 1981.

France - Their general approach tends to be similar to ASME-XI. Any deviations, flaw analyses, etc., are handled on a case-by-case basis with direct interfacing between regulators and regulated.

Federal Republic of Germany - The 1978-79 Code Comparison funded by TUV-Rheinland permits a good comparison of similarities and differences in the FRG and U.S. approach. In the past 3-4 years, a document KTA-3201, has been developed presenting the regulatory position with regard to NDE. It is understood this emphasizes NDE during construction. This document is under final review and should be released before the end of 1982.

United Kingdom of Great Britain & Northern Ireland - NDE has been applied to gas-cooled reactors for some time and fairly definite procedures exist. Typically the UK follows the British Standards approach with more flexibility permitted than is true under U.S. codes. For much of their NDE, particularly on naval reactors, they use ASME XI with modification in beam angle, DAC, etc.

NONDESTRUCTIVE EXAMINATION

Work on NDE is much too extensive to permit more than a superficial overview. The contacts made in Sweden, Norway, Belgium, Netherlands, France, Germany, United Kingdom, and Canada represent no more than a small fraction of the groups active in NDE.

In thick-secion steel analogous to material used in reactor pressure vessel, the PISC-II program appears to be the logical vehicle for transmittal of information, not only for PISC-II, but also for similar programs reported under that umbrella.

Late work reported by SWRI and IZFP permits a comparison of the relative reliability of detection of flaws with various advanced techniques; most of these techniques still are in the laboratory.

The FRP work is very extensive. Perhaps the most developed is the Tandem Technique where KWU has had excellent success for both detection and sizing of flaws. Several techniques for sizing such as focused probes and acoustic holography are well developed.

French work has concentrated heavily on focused probes where they have fully automatic systems for inspection of PRVs.

The P-scan system of the Danish Welding Institute appears to have considerable promise. Sweden has picked up and will be adapting it to in-service inspection.

The efforts, often interactive, at The Welding Institute, CEGB and UKAEA should provide extensive information on both detection and sizing reliability of thick-section weldments. They have severe time constraints because of the public hearings. Their extensive efforts on the fundamentals of UT should be of considerable interest.

FRACTURE MECHANICS

The general feeling is that the necessary answers with regard to Linear Elastic Fracture Mechanics (LEFM) are available. In fact, efforts in the support of LEFM are quite limited other than in the area of predictive tests to correlate Elastic Plastic Fracture Mechanics (EPFM) values with LEFM values. The big efforts are in EPFM or General Yield Fracture Mechanics (GYFM).

Major efforts in probabilistic fracture mechanics are in Sweden, France, Norway, and starting in England. These are discussed elsewhere.

In EPFM the leaders are considered to be France (Framatome), and England with the CEGB R-6 approach. Another major program is on Crack Tip Opening Displacement (CTOD) at the Welding Institute. This is the old COD plus its follow-up in the R-Curve where loading beyond initiation has been examined extensively. Both R-6 and R-Curve have been applied extensively to real structures containing flaws to decide whether to continue operation or to repair. Their programs should be of major interest to the NRC.

Another program of interest is the planned EPFM round robin. Presumably this one pursued by UKAEA (Risley) under Ray Nichols will be better defined than the previous one reducing scatter in results and permitting a definitive comparison of the front-runner EPFM techniques.

Although aware of substantial work in the FRG; their work is not considered to be plowing new ground or markedly extending the state of the art.

PROBABILISTIC FRACTURE MECHANICS

One of the problems in the use of probabilistic fracture mechanics is the lack of data defining probability density functions covering flaw size and mechanical property and stress distributions. One can synthesize the stress distributions, and, to a degree, the mechanical property distributions; however, there may be large sources of error on the use of an incorrect continuous distribution; for example, Guassian when it should be log-normal, etc.

Data should be forthcoming from the following sources:

FRG - raw data on about 2730 meters of weldment in relatively thick sections

Norway - data from the joint Scandinavian program

France - data from the 100-200 meters of weldment

U.K. - prior data is available

If possible, the probability density function which best fits each batch of data (to determine if the same continuous distribution is applicable to all lots of data) should be determined. It might be possible to synthesize a distribution applicable to weldments over a substantial (2-12 inch) range of thickness.

A Det Norske Veritas paper given in Bratislava in 1979 is of particular interest to piping systems. They examined the effects of environment (air versus corrosive) and of eccentricity alone or together. In the corrosive environment after 20 years the failure probability was about 10^{-8}; a given eccentricity at 20 years gave a failure probability of about $10^{-2.5}$ while both eccentricity and corrosive environment gave 10^{-1} indicating the major contribution of eccentricity to failure.

COUNTRY BY COUNTRY REVIEW

EUROPEAN COMMUNITY

In essence, because of the competition for monetary units, CEC supports very little other than work at Ispra or proprietary work (non-nuclear) to which access is virtually impossible. Historically the Joint Research Center at Ispra, JRC-Ispra, has placed emphasis on fast reactors competition with various national programs in the European Community.

Of the four areas considered, the following represent a capsule version.

1. NDE with emphasis on UT- Funding level substantial. Effort all directed into support of PISC-II. Their contributions are picked up separately. JRC-Ispra serves as technical manager and referee laboratory for PISC.

2. NDE- Other detection systems. They have ongoing work in acoustic emission that tends to fit into overall European programs. Programs are reported routinely. Alternatively they can be followed, together with other work, through CSNI.

3. Degradation Mechanisms- Because of their emphasis on fast reactors they are concerned with a higher temperature regime with regard to material damage. They have ongoing work in creep-fatigue of austenitic stainless steels. This is considered outside the scope of this project and not reviewed.

4. Fracture Mechanics- There is a substantial effort at JRC-Ispra in fracture mechanics. They have participated in the CSNI round robin on EPFM and they have a substantial program on constitutive laws of materials under dynamic loading that tends to be closer to the seismic tasks area than to fracture mechanics. Again, their emphasis is on fast reactors so their fracture mechanics concentrates on austenitics at higher temperatures and in a neutron environment. Their slow crack growth work is in sodium environment. They have provided direct support of the CSNI EPFM round robin and a report exists. Unfortunately, this round robin wasn't too successful and another is planned with the United States invited to participate.

OECD-CSNI

Examples of relevant programs under the Committee of Safety of Nuclear Installations (CSNI) of OECD are:

1. CSNI Specialist Meeting on instrumented precracked Charpy tests - Palo Alto, December 1-3, 1980.

2. CSNI - Sample Programs for Comparison of Critical Defect Sizes Calculated using ASME XI Appendix A and EPFM Procedures, R. W. Nichols and T. Ingram. The earlier EPFM round robin used the ASTM three-point bend test data for analysis.

Two CSNI documents briefly describe alternate EPFM concepts: one by Burdekin and Harrison in 1979 discusses thirteen concepts; another by Harrison and Ingram discusses six of the thirteen in slightly more detail.

SWEDEN

With regard to NDE-UT a deliberate decision was made in Sweden <u>not</u> to compete in the development of new techniques. Rather, they carefully assess some of those developed elsewhere as a potential, then devote their efforts to taking such techniques and establishing field feasibility. Specific examples include:

1. The P-scan system developed by the Danish Welding Institute

2. Immersion focused search units for sizing of defects (French-CEA)

3. A limited effort on detection of IGSCC using RTD or Vincotte search units.

All the Swedish activities noted are under AB Statens Anlaggningsprovning (the Swedish Plant Inspectorate analogous to U.S. third-party inspectors-insurance companies). Sweden is participating in PISC-II using mechanized equipment (probably P-scan) developed by Tekniska Rontgen Centralen AB (TRC).

Material Degradation

There is ongoing work on the aging of thick sections of RPV steel (A 533 B) at Studsvik and similar work at the Swedish Institute for Metals Research. Final results are expected in 1983.

Studsvik has had programs in residual stress evaluation. In fact, this was a major task under PISC-II, tabled because of lack of funds. In this area, there is a great deal of unpublished work from several countries that needs to be pulled together to determine what, if anything, needs to be done.

Since residual stresses represent a major input into RPV accident analyses, it is unfortunate that this work isn't being completed.

Fracture Mechanics

There are several active programs in fracture mechanics in Sweden. In fact, this represents the most probable area for bilateral pursuit. Studsvik has work in fracture mechanics and fracture toughness. The majority of programs are in the Swedish Royal Institute. They have done some of the better analytic work in probabilistic fracture mechanics.

In-Service Inspection

Sweden has used ASME XI for their reactor pressure vessel examination; however, their requirements for piping differed substantially. A paper by Kornvik gives an excellent and up-to-date picture of how in-service inspection is carried out. They issued their official "Rules of In-Service Inspection of Nuclear Power Plants" in April 1, 1981, a translation is available. Areas not covered are accept-reject criteria for flaws and flaw evaluation techniques. There are several specifications for NDE covered in 13 different procedures specifications.

Status NDE-R&D

Work on focused probes tends to follow the French CEA approach. The project may be completed this year. No major value to NRC is seen through Sweden since the French contacts are well established.

Limitations in UT probes for NDE of coarse-grained austentic alloys. A joint program with SA, Sulzer, KWU, TRC, and RTD using KB Aerotech, Vincotte, SWRI, RTD probes with work by RTD (Netherlands). All probes had definite limitations. The SWRI performed most satisfactorily.

Development of P-Scan Ultrasonic Examination Method. First stage is complete. An excellent description of the system was given by Niels Nielsen at the 1980 International Conference on NDE in the Nuclear Industry and was reprinted in January 1981 in Welding Research Abroad (pp 2-26). P-Scan (Projection Image Scanning Technique) is a microprocessor controlled UT system that is capable of measuring and recording all signals with one dB resolution. These signals are recorded as projected images (from top surface through thickness) plus a visualization of maximum echo amplitude in a logarithmic scale. It is both fast and flexible.

JOINT SCANDINAVIAN PROGRAM RT AND UT

An ongoing program (since 1977) has some potential value to nuclear secondary and terciary piping systems. Co-sponsorship is supplied by organizations in Denmark, Sweden, Finland, and Norway. The program consists of examinations with RT and UT by certified operations (8 each) of butt-welded mild steel plates and pipes in thickness to 25 mm. A total of 250 m of weld containing a variety of defects will be examined, NDE results confirmed by destructive testing and the results analyzed statistically. The program was scheduled for completion in 1982. This represents one of the few comparisons known of RT and UT for extensive footage of weldment. While the welds are not truly typical of production runs, they should provide a cross-check on the landmark Danish work on weldment defect probability density functions used in the Marshall Report and in other probabilistic fracture mechanics studies.

Det Norsk Veritas is a not-for-profit foundation similar to U.S. insurance companies for third-party inspection. A listing of titles of possible relevance is given as follows:

1. P. Dalberg, AA Four-Probe Technique for Accurate Crack Sizing Using Ultrasonic Diffraction/Scattering," DNV 81-0051, January 9, 1981.

2. Det Norsk Veritas: Research and Development 1980-1981, pp 39-49. List R&D projects in 1981; articles (only those in English) lectures and papers issued in 1980; and R&D reports in 1980.

3. Papers on NORDTEST Project 72-76, "A Comparison of Radiographic and Ultrasonic," NDE Pilot Study of a Statistical Model for Evaluation of Examination Performance.

4. Fracture Mechanics Papers

    a. Tenge, P. and Karlsen, A., "Significance of Defects in Heavy Section Welds for Offshore Purposes."

    b. Bokalrud, T. and Korsgren, P., "Some Aspects of the Application of Probabilistic Fracture Mechanics for Design Purposes."

Both above presented in Colloquim on Practical Application of Fracture Mechanics, July 10, 1979, ISI, IIW Bratislava.

FEDERAL REPUBLIC OF GERMANY

The FRG authorities use the ASME Codes (III, XI) and various publications on linear elastic and elastic-plastic fracture mechanics; also they use or are aware of the CEGB R-6 method.

The above procedures have been applied to systems, particularly piping. In at least one instance an extensive fracture mechanics anlyses was made of the STADE reactor pressure vessel. Papers are available describing some of the analyses.

Reference was made to analyses by KWU. No written response from KWU was received, only informal comments concerning work in UT.

There is work underway on probabilistic fracture mechanics by DVM. Presumably a report will be released sometime in 1982.

There has been UT by three teams on approximately 800 meters of weldment 90-120 mm thick (3.5-4.5 inch). There were 385 indications detected with varying success. All found 32%. In other studies on thinner (60-120 mm) and thicker (120-250 mm) sections, similar flaw populations exsisted. A total of 2730 m of weld were examined or about one indication per 5 m of weld (385 + 375 + 240 = 1000).

Defect size distributions are skewed to the left, for example, the preponderance were small, less than 30 mm long (approximately 60%).

There appear to be programs underway at Bundesanstalt fur Materialprufung (BAM) of interest to the NRC in resolution or definition of certain safety concerns. Each is separately discussed with comments concerning relative values:

1. Reliability of detection of flaws in coase-grained austenitic materials- BAM has substantial programs in this field, and have published several papers. It is understood that much of this work is coming to an end. They have a proposal for developing UT techniques for the region between the weld center line and the heat affected zone. This field is of interest to the NRC in the general context of NDE-UT reliability with respect to austenitics. There is general access to the data and nothing sufficiently unique to justify bilateral agreements is seen.

2. Sizing and location of flaws in weldments using specialized UT techniques- BAM applies the philosophy-use whatever is reliable for detection. Once detected highly specialized equipment can be justified for sizing. They have explored acoustic holography (AH) extensively, including examinating real flaws in pressure vessels. In addition, they are using contact focused probes (FP) for sizing. There are several publications on AH, but not in English. The AH work is similar to that at Babcock and Wilcox. The focused probe requires comparison to the extensive CEA effort in France. They expect to use both techniques on the MPA pressure vessel if the government approves funding.

3. Detection of flaws in the near field range- BAM has worked extensively with creeping waves to detect near field defects very near the surface or immediately under cladding. The creeping wave technique has been applied to vessels, turbines, etc., for 5-7 years with substantial success for detection of defects 0-15 mm below the surface. Supposedly they have statistical data as to detection reliability correlated with actual sizes. This program should be of major interest in the context of NRC concerns with the cold repressurization accident and defects immediately below the cladding. Another technique that is quite new and on which no definitive papers have been published in English is Wittig's work with pulsed eddy currents. This technique appears to be quite effective in locating under-cladding cracks.

4. A general catchall project participated in by BAM, Drautkramer, KWU, and MAN covers the broad field of techniques and UT equipment for in-service inspection. This project (RS-2704) has been discontinued. However, data on probe and equipment development, coupling effects, cladding effects, NDE of nozzle corner region, etc., were developed. Some of the information has not been published.

Similar work on near field detection of flaws is being done by RTD Netherlands; however, the BAM work seems to be the most effective for detection of near field flaws among the six countries.

COMPARISON OF SIX UT TECHNIQUES FOR CLADDED PIPE

Examinations

Techniques	phased arrays, controlled signals, spatial averaging, restricted beams, short pulses, multiple beams
Specimens	Cladded or welded; specimens containing penny-shaped cracks
Statistical Analysis	41-86%
Table 3.	describes specimens

Table 4.	gives Type III plates and notch dimensions
Spatial Averaging	poor
Controlled Signals	fair-poor
Phased Array	fair-poor
Restricted Beam	good-fair
Shear Long	better
Short Pulse	good-poor
Shear Long	much better
Multiple Beam	good
Shear Long	about same

The KWU has been withdrawn from government funding on the basis that progress is slow and actions restricted. Their tandem technique has been very successful for examination of thick weldments typical of reactor pressure vessels. In fact, they were the most successful participant in PISC-I in detecting the sizing flaws. They recognize the accessability limitations of the tandem technique in discontinuity regions and are exploring single probe techniques to complement the tandem.

The United Kingdom, in anticipation of their upcoming public hearings, have prepared test plates with small difficult-to-detect weld defects and are having two of the best in the business, KWU (using the tandem technique) and CEA (using focused probes), examine the plates.

The KWU is considered to be one of the best, if not the best, pratitioner with tandem UT.

In the FRG they have been developing an official position document aimed at more rigorous requirements than under a code such as ASME-III. This document KTA-3201 has been cited extensively in the first 1-2 years; however, there has not been an official release. Presumably it is under final review and an official copy in 1982 has been promised. This document is aimed at the construction stage (not operation) although they infer some aspects of in-service in portions of the document.

It is suggested that the NRC, if it has not already do so, should formally request the released version.

It is uncertain whether the version to be released contains a calculational procedure for K_I. Earlier versions did not, and they used ASME XI, Appendix A. They infer use of K_I. In fact, the use of K_{IC} rather than K_{IR} is cited.

The largest programs is fracture mechanics in the FRG are those at MPA. They are similar to the HSST in some aspects; however, testing of a full-size vessel is included. Generally, the work in the FRG tends to concentracte on development of test specimens to simulate system loads, and obtaining of test data rather than coming up with new analytic procedures. For example, BAM has been working on a specimen to simulate reactor vessel biaxial stress states and the Fraunhofer-Institute fur Werkstoffmekanik has been collecting J-integral data for EPFM.

The various facilities in the FRG conducting work in fracture mechanics in the 1970's were:

MPA (Kussmaul/Sturm) covers vessels and piping

KWU - Klausnitzer and Schmitt - EPFM

IFRM - Sommer - Blauel - LEFM, COD. Also Winkler on shock loading

Peter - TUV - RWE; Aurich - BAM; Azoda - IRS: Bazent - BBC; Spahn - BASF.

A report by RSK in 1978 analyzed the probability of failure of the Stade RPV after irradiation. Faulted loads were assumed including LOCA and cold-water injections. The existence of cracks under the cladding to 5 mm deep by 30 mm long was assumed and the system modeled with the VAK-I and VAK-II computer programs. Generally the approach was similar to that under generic issue A-11 on low upper shelf toughness. They used a J-integral approach with input from experimental programs on irradiation damage.

FRANCE

A paper by Dufresne of CEA given at Cannes in 1981 has processed (not raw) information on flaw-size distributions in PWR weldments. The data should be of assistance in probabilistic fracture mechanics.

The "new" French reactor construction code developed to comply with the Official Regulations regarding the reactor pressure boundary exists in several volumes with more planned. Basically, it is modeled after ASME III in format. For example, Vol I - contains several sub volumes

A - General
B - Class 1 (like NB)
C - Class 2
D - Class 3
G - Internals (in course of preparation)
H - Supports
Z - Fracture Mechanics (like Appendix G-ASME)
2 - Materials
3 - Method de Controle - Test techniques both destructive and nondestructive similar to ASME Section V
4 - Welding
5 - Fabrication

The NRC should obtain copies of these.

The Annex Z (fracture mechanics) will be revised soon. A translated LWR status document should also be available soon.

No flaw standards or evaluation procedures for flaws in operating reactors exist. All such are handled on a case-by-case basis. It is planned to develop documentation concerning generic procedures; however, there will be nothing comparable to IWB-3600.

Currently ASME XI is used as an unofficial guide for in-service inspection. It is possible that EDF in conjunction with Framatome may develop such a code in the future.

Framatome is concentrating on real systems for their fracture mechanics analyses. Current studies include:

1. Effects of flaw shape factor in the RPV beltline regions including a variety of loading conditions such as small LOCA, large LOCA, safety injection, etc.

2. Potential effects of underclad cracks.

3. The nozzle-safe end potential for failure. Specific experiments are planned to analyze crack behavior in the buttering layer (309 Cb) on the nozzle. They assume cracks interact at the ferritic interface then move into the softer austenitic layer. In fact, they have experienced a very large flaw in such an interfacial region that was detected by RT.

4. They are conducting analyses on steam piping regarding failure mechanisms.

In FY-82 emphasis is on the effects of flaw aspect ratios and system geometry, varying a/c from 1/3 to 1/10 and R/t from 2 to 2.3.

It is felt that there is some information exchange primarily through ASME-XI and CSNI/OECD.

Another area of fracture mechanics being investigated is the nozzle problem. They are analyzing corner cracks in the inlet nozzle by altering the crack shape. This will be a parametric study holding K_I constant as the crack from advances and determining changes in flaw a/c values. In recognition that cracks may occur away from the nozzle corner they are looking at cracks in the bore region.

Electricite de France purchased rights to the BIGIF Code to permit a comparison of the strengths and weaknesses to the French and the BIGIF computer codes. As noted later CEGB did the same to permit a comparison of Chell's codes with BIGIF.

A substantial effort, and one of potential interest to the NRC, is the return to fundamentals both analytically and experimentally to justify an elastic plastic fracture mechanics approach such as use of the J-integral. They will be examining crack tip behavior without a tie to J-integral. This is comparable to EPRI-supported work at Stanford, work at the British Welding Institute discussed later, and some of the CEGB work. All are attempting to develop a better understanding of how cracks are initiated at the microscopic level and what factors influence their propagation. This fundamental effort is considered necessary to justify to the regulatory authorities that there is a real understanding of how cracks initiate and propagate and to prove that the analytic method adequately models the behavior.

The French work on piping appears to be fairly extensive. They are extending early work of BMI on axial flaws with experimental and analytic tests. I believe they are examining growth of real cracks to permit examination of crack profile instead of limiting to a trapezoidal flaw. The piping effort is shared among various groups; for example, EDeF concentrates on material properties.

Framatome cooperates with CEA in evaluating experimental (CEA) data. There is a pipe rupture group at Cadarache (CEAO as well as an analytic group under Roche (CEA).

In addition to the flaw testing they are examining aging effects on welds and changes in properties of case austenitic alloy.

The CEA efforts in fracture mechanics are in the following areas:

(1) Toughness and fatigue behavior of bimetallic welds.

(2) Fracture mechanics applied to austenitic stainless steel.

(3) Mechanical factors affecting the realiability of structures. This program is in its sixth or seventh year and emphasizes fatigue, etc., as validation check on ASME Codes.

There were large number of papers covering the overall French fracture mechanics program at SMiRT in August 1981. Some of these papers contain new data worthy of review by the NRC.

It is felt that the French program in fracture mechanics is quite good. Their work in the probabilistic area is more advanced in some respects than work elsewhere. The CSNI has served as a fairly good vehicle for information exchange; however, it has been in the doldrums lately. The piping work represents the area with most potential for an information exchange. By and large the vessel fracture mechanics is an extension of techniques studied elsewhere rather than a unique program with no work elsewhere. Their fracture mechanics work has surfaced frequently through ASME-XI, CSNI, etc.

The principal effort by CEA in NDE is with focused probes. They now have six fully automated systems for UT of reactor pressure vessels. The MISE (Machine Inspection Service) system performs UT to a close approximation of the ASME-XI Code. They have examined the same vessel ISI using three different MISE systems with high consistency of detection and of signal reproducibility. Their system for RPV's uses about 50 probes. They have conducted a total of 40 pre-service and in-service inspections with MISE.

Framatome has been examining UT of austenitic weldments. They have developed a technique for examining RPV nozzle corners which is incorporated into MISE.

The focused probe technique had results comparable to KWU on the PISC-I plates with regard to both detection sizing and locating.

With regard to flaw analyses the French handle each incident on a case-by-case basis. Their UT-ISI tends to follow ASME-XI. They meet the fabrication/construction requirements incorporated into law. An earlier order, February 27, 1974, established fabrication/construction regulations for "pressure vessels in water-cooled nuclear steam generators." This pertained to the primary circuit of such systems. There is a new French Code dealing specifically with NDE during the fabrication phase paralleling ASME III and V.

With regard to ISI per se, there appears to be no effort to generate a Code. A new status report on LWR's translated into English should have been available in July, 1982.

They have construction rules for LMFBR's, Turbines and Concrete Structures with input from EdeF, NOVATOME, and FRAMATOME.

UNITED KINGDOM

The Welding Institute

The Welding Institute represents somewhat of an anomaly. It limits its work to members; a great deal of the work is proprietary. It does consulting for its members. Even so much of their work is published, particularly that related to nuclear because sponsors such as the Nuclear Installation Inspectorate want the information in the open literature.

It is felt that the NRC could profit through participation in some of the Institute's programs. They represent one of the major, if not the major, repository of information relevant to welding and to the behavior of weldments. DOE is a member so the NRC could operate through that membership. A NRC membersip could be valuable since DOE distribution of documents is spotty.

The major field of interest to the NRC is their work in the field of fracture mechanics. They are strong advocates of the R-curve approach discussed later. More significantly they back up the analytic techniques with the most extensive experimental programs on weldments known. A large share of expeimental data throughout the world relevant to fracture mechanics is on wrought products such as rolled plate or forgings. The amount on weldments and heat-affected zones is relatively limited with the great majority produced by the Institute.

A second and more limited area is in NDE-UT. They are late comers to the field. However, in conjunction with other laboratories they have made marked progress.

Finally, they have done substantial work on mechanisms affecting material properties either positively or negatively with emphasis on welds and heat affected zones. Some examples include reheat cracking, measurement and control of residual stresses, hydrogen-induced under-clad cracking. Some programs significant to NRC interests not yet public include:

(1) Literature review and experimental programs relevant to reheat cracking and embrittlement of thick section weldments of A-533-B and A-508-C13.

(2) An extensive study on residual stresses in thick stainless steel sections. (National Nuclear Company)

(3) The repair welding program.

(4) Under-clad cracking (Framatome, NII, etc.)

The fracture mechanics people have definite reservations concerning use of the J-Integral where K_{IC} obtained from J_{IC} in certain toughness-transition regimes can be nonconservative.

In NDE the Institute contribution has been in producing test blocks containing a variety of fabrication flaws. They have worked closely with Harwell and CEGB on developing UT techniques so their expertise has increased substantially. They are active in the program aimed at producing information for the PWR public hearings; also in the PISC-II program.

As cited earlier NDE is a relatively new program. There is substantial document entitled "Size Measurement and Characterization of Weld Defects by Ultrasonic Testing." This was a seminar given March 17, 1981, touching on Phases II, III, IV.

Phase I - Non-planar defects was released previously covering U.S.

Phase II - Planar defects in ferritic steel

Phase III - Effect of metallurgical features of welds in ferritic steel

Phase IV - Investigations of welds in more complex geometries and site application

Phase II - A joint effort by The Welding Institute AERE-Harwell, National Coal Board

Phase III - Welding Institute

Phase IV - Welding Institute, CEGB NDT Centre, AERE NDE Centre

This report appears to be quite detailed and worthy of careful review; however, the final report will be necessary to intercompare detection reliabilities.

The following summarizes key parameters:

<u>NDE</u>

14 differenct specimens

t - 34 to 94 mm

Welding Processes - SA, MMA

Joint Prep - Single and Double V and U

Defects - solidification cracking, incomplete fusion, large crack, hydrogen cracking, slag inclusions, HAZ crack.

Defects - 28 in all

Surface Prep - as-welded, machined (one or both surfaces)

Examinations - RT, UT (B-scan, C-scan, Accuscan, time-domain, conventional), Holography

Crud Sizing - freeze break, on sectioning

Sizing (Tables 4,5) X $\pm$ 2 (various techniques)

Prediction of type of defect - 16 correct, 9 manual, 3 not identified

Conventional UT - Welding Institute

Manual B-scan - Welding Institute

Accuscan - CEGB (NDTA)

Acoustic Holography - CEGB and Harwell

Time Domain - Harwell

RT - ? (Probably WI)

Immersion C-scan - Welding Institute

Only cumulative data reported not individual.

Central Electricity Generating Board-Berkeley Laboratory

Both CEGB and UKAEA (Risley) were quite negative toward acoustic emission. The precise reasons were not given, however, they seemed not to be very confident with crack detection. There was a general reaction that they felt less strongly with regard to leak detection.

The majority of time at Berkeley was spent on their fracture mechanics programs (Appendix IVa). These will be touched on briefly.

(1) Berkeley - Chell, has been developing a computer code somewhat similar to BIGIF, based on Green Functions. The two codes are FRACPAC and FACPAC. There is a document being prepared that will compare FRACPAC-FACPAC to BIGIF. Advantages cited for the computer package are cheaper, easier, quicker, open-ended; however, there are existing limitations such as handling statistical variations in 2-D data.

(2) The major area of development is the R-6 procedures used in fracture mechanics analysis of pressurized systems. A great deal of work has been done and several options are included. These will be discussed as a subset under fracture mechanics. The latest work will attempt to bring probabilistic fracture mechanics into R-6.

(3) In property changes and degradation, work includes strain hardening in austenitic alloys. Work on strain aging of A-533 Grade B at ambient temperatures (approximately 30°C), temper embrittlement in A-533 Grade B. Both studies are by CERL. Although the temper embrittlement studies are preliminary, the results seem to indicate that it is as severe as irradiation embrittlement.

(4) The fracture mechanics properties and sensitivity to stress corrosion cracking of A-533 Grade B weldments are being examined in 5-inch plate using manual metal and submerged arc. They will be examining both weld and HAZ properties in these weldments.

(5) The CEGB has promised reports in analytic work on the effect of flaw aspect ratios in various sizes of piping; the variation of c/a are being checked experimentally. This seems to parallel CEA (France) work as well as FRG work at Freiburg. One aspect has to do with the potential for plastic collapse in the presence of very deep flaws in the R-6 model. As a result an a/t ratio of 0.8 is being used.

Of interest is the fact that R-6 has been used to justify leaving in cracks. They have made a substantial number of such analyses most of which are proprietary or privileged information. Most such are on magnox plants or non-nuclear plants.

A relatively new area of work is development of hot cell facilities to permit testing of fracture mechanics specimens such as pre-cracked Charpy specimens under three-point bending over temperature range of 100-400°C. They will be able to conduct unloading compliance as well as testing compact tension specimens.

In the NDE areas of Berkeley program has as its major objective the assessment of NDE (UT) reliability. Their long-term approach is to do so through a fundamental assessment of reliability of modeling. The models use simplifying assumptions so the approach is quasi-fundamental rather than fundamental. Hopefully, more sophisticated models will be the result. The current effort is to assess embedded surface, and through-thickness flaws. The model is complete and they have moved to the validation stage. The validation will use PISC-I data. They have the raw data computer tapes and expect to do an extensive analyses. They will tie this to a spectrum of defects based on fracture mechanics input and then tie to required NDE reliability. This is a possible area for information exchange.

Modeling to ASME V and FRG Tandem techniques have predicted high detection reliability. Results are presented in several figures with various tilt and skew values. Comparisons of theory to experiment are given. Generally, there is good to excellent agreement.

The CEGB does not use specific codes and standards. Such items as ISI and flaw evaluation are handled on a case-by-case basis. Programatically, ASME XI is used then modified by adding more search angles, lower DAC levels, etc. There is a trend toward the FRG tandem technique (KWU) plus focused probes for sizing. They use higher sensitivities than KWU.

The NDT Centre also is using fundamental theoretical models to establish the interaction of UT beams with reflectors. They are working with the University of Manchester. This is aimed at beam skewing and scatter in anisotropic media.

Of interest is a very recent decision to coordinate PWR-related R&D through one agency rather than separately funding NII, CEGB, UKAEA. It appears that Risley (UKAEA) will provide this function. This change should not affect ongoing programs; however, it may change future programs.

There are seven major programs to be studied in the 1980-1985 period. These include the following:

(1) Consultancy;
(2) Determination of Ultrasonic Reflectivity of Flaws in the PWR Pressure Boundary;
(3) Influence of Austenitic Steel Cladding on the Ultrasonic Inspection of the Base Material in PWR Pressure Vessels;
(4) Ultrasonic Inspection of Austenitic Components in PWRs;
(5) Effectiveness of Ultrasonic Testing of PWRs;
(6) NDT of PWR Steam Generator Tubing;
(7) Automatic and Semi-automatic Ultrasonic Inspection of PWRs.

UKAEA-Risley

Risley has discontinued work on LEFM. Emphasis (in support of the PWR program) will be EPFM or GYFM with the upper shelf region of major concern. They will be conducting a paper study plus testing of weldments in plate. In essence the preceding is a mini-W, EPRI, NRC program using center cracked panels, SENB, DENB samples. They hope to get dJ/da data to validate R-6 and the R-curve.

There are substantial reservations in some quarters concerning the validity of the K_{IA} (crack arrest) curve in ASME-XI for faulted conditions. Because of this they are planning a crack arrest program to establish if it is a material property.

A problem is the possibility of SCC in A-508 Cl-3 or A-533 Grade B with emphasis on weldments.

Work is planned to see if the Chell-Milne approach to fracture in the transition regime is valid.

Work on degradation of properties include aging, strain aging, temper, embrittlement. Some work at Harwell on aging is using an exaggerated approach to emphasize temperature shifts.

Another program being planned considers the combined effects of pressure loads and thermal shock. They hope to use a spinning cylinder to develop axisymmetric loads then spray the inner core with water to thermally shock it.

Another study nearly complete is an analytic comparison of three of the front runner techniques for EPFM analyses as a forerunner to an EPFM round robin. The three are:

COD (removing conservatisms)

J - integral

R-6

Many of the above items were expected to appear in the second Marshall report.

Flaw evaluation in real systems - generally UKAEA uses the ASME XI Appendix A and WRC-175 approach with conservative inputs.

Modeling - Some 2-D and 3-D work; most emphasis is on visualization rather than detailed computer codes. There are experimental and analytic programs supporting the effort. Much is classified; however, there are some unclassified data.

ISI - They have been using ASME-XI with some added requirements such as more angles, lower sensitivity levels and data digitilization.

They are examining factors influencing calibration through the examination of a spectrum of ASME, FRG, clad, unclad calibration blocks.

Flaw Sizing - Work is underway to develop time-of-flight equipment and focused probes.

Near Surface UT - the BAM technique, using twin crystal probes can detect flaws 1.5 mm in height in 25 mm under the cladding. They have been investigating techniques for near field in the inner ratio of nozzles with limited success and expect to shift to another approach.

CANADA

Atomic Energy of Canada, Ltd.

The AECL does substantial work on properties of materials directly related to CANDU, namely, AISI-4340 steel, A-516, Grade 70, A-403, ZR-2.5% Nb, etc. Since section thicknesses usually are well below 4-inches, emphasis is on J_{IC} measurements which may or may not be converted to K_{IC}.

A program recently initiated is of potential interest. They intend to measure initiation times of crack from defects of known sizes and geometries. The initiation times are expected to be in nature so that many tests will be necessary to develop the appropriate probability density function(s).

There is an effort to incorporate fracture mechanics into some of the Canadian Codes. This will tend to expand on ASME XI Appendix A as a source. Their primary interest is in fracture mechanics as a tool so they have only limited interest in theory.

Probably the most interesting program is that on their pressure tubes, a Zr-2.5% Nb Alloy. There have been instances of cracking occurring due to a combination of high surface residual stress and hydrogen. Their fracture mechanics model does a good job of predicting observed behavior. To my knowledge the work on tube bursting of sections containing flaws is one of the few definitive programs. It's less comprehensive than that of Eiber of BCL but leads to similar conclusions regarding leak-before-break or instability.

They are quite strong in eddy current and improvements in ultrasonics. Efforts underway on UT include Rayleigh waves for detection of surface fretting and wear; controlling signal characteristics with EMAT and crack growth monitoring by measurement of the crack shadow. Their UT work is seen as good but not plowing new ground.

In eddy current they have done and are doing excellent work. They emphasize single frequency ET using absolute rather than differential probes because they feel such an approach simplifies visual analysis and flaw prediction. To a major degree their decision has been influenced by their very low steam generator or heat exchanger tube failure rates and to the limited number of failure mechanisms. They are familiar with the multi-frequency multi-parameter approaches and are prepared to move in that direction if it becomes necessary.

The work in improved equipment for on-line analysis is good. Particularly impressive is the work to develop ET for examining ferromagnetic materials. They have been successful in mildly ferro-magnetic materials such as Monel-400; now they are attempting to develop a technique for ferritic tubes to take advantage of the speed of ET. It may be necessary to couple with follow-up UT when saturation is not possible such as in the tube sheet region. If they succeed, there will be a major payoff, particularly, in the fossil industry where many steel tube heaters, etc., are used.

They have an ambitious program to use ET (or other techniques) to locate leaking tubes in heat exchanger and steam generators. The last two items should be classified as having only as limited probability of success; however, both would have a major payoff and represent the two items worthy of follow-up by NRC.

<u>Ontario Hydro</u>

Ontario Hydro work should be classified as applied R&D strongly tied to their operating plant needs with an emphasis toward the nuclear plants.

In NDE they are strong in equipment development aimed more at signal handling and automation than at advanced UT techniques. They are engaged in developing transducers capable of surviving a few mega-rads at temperatures somewhat above ambient. They found that none of the transducers on the market could survive the gamma and neutron fields of CANDU or did not have sufficient sensitivity if they survived.

A recent program is using pulsed (chirped) eddy currents with radar. While their major interest is UT they haven't figured how to obtain chirped UT signals in the range of frequencies of interest. This can be done with ET. This appears to be similar to a FRG program reported briefly at Lindau. This latter item may interest NRC.

Fracture Mechanics

Ontario Hydro has carried ASME XI Appendix A solutions as far or further than anyone in the U.S.

Their fatigue work was performed mostly at room temperature and in air adds to the data bank but isn't too relevant to NRC needs.

Also of interest is their work to correlate fatigue and growth in flat plates under both bending and tension with crack growth in CT specimens. The program outlined will tend to repeat earlier work of Iida in Japan using another material and should improve our perspective of crack growth under various types of loads.

Their work on analysis of flaws in real systems follows ASME XI Appendix A closely. It represents sophisticated analyses of flaws in various components and have all been accepted by the Canadian reactor safety board. Examples of applications include manholes, pump impellers, fly wheels, nozzle regions, etc.

In Table 1 is a listing of recent Canadian publications in the areas of materials and metallurgy with a brief comment as to information contained.

AECL Docments

(1) V.S. Cecco, "Design and Specification of a High Saturation Absolute Eddy Current Probe with Internal Reference." Mat. Eval. 37 No. 13 (ACEL-6763), 8 pp. - Specifics of absolute probes using Monel-400 data to show application in weakly ferromagnetic environment.

(2) G. Van Drunen and V.S. Cecco, "Eddy Current Inspection of a 17-year old Nuclear Steam Generator," 9 pp, Source? (ANS?). - Tests on NPD - Special single frequency design successfully detected. Neither visual nor multi-frequency were successful.

(3) G. Van Drunen, V.S. Cecco, J.R. Carter, "Eddy Current Detection of Corrosion Damage in Heat Exchanger Tubes," AECL-6965, May 1980, 29 pp. - A general overview paper giving pros and cons of differential and absolute probes - applicable.

(4) Anon, "Novel Ultrasonic-Technique for Remote Measurement of Fretting Wear Grooves Under Pipe Hangers, 1 pp. AECL. - A discussion on how Rayleigh waves can detect fretting wear.

(5) R.I. Coote, "Ultrasonic Assessment of Crack Size in Candu Pressure Tubes- no source cited. - Extensive UT on cracked Zr alloy tubes correlated with distribution evaluation. Compares EDM notches also. Benchmark regarding real flaws.

(6) W. Licht, J.B. Hallett, "Mapping the Ultrasonic Defect Shadow in a Pitch-Catch Mode," AECL-7101, June 1981, 10 pp., Presented 4th Int. Conf. on NDE on Nuclear Ind. - Landau, Germany, May 1981. - Pitch-catch technique to size flaws using shadow actual versus UT size compared to real and artificial flaws.

(7) M. J. Ward, "Potential for Ultrasonic Inspection of Heat Exchanger Tubes," CRNL-2114, August 1980. Not for Pub (Presented at Canadian Nuc. Assn. Sem. on HX Reliability, Toronto, May 1, 1980. - UT for HZ tubes - no definitive data.

(8) W. J. Langford and L.E.J. Mooder, "Fracture Behavior of Zirconium Alloy Pressure Tubes for Canadian Nuclear Power Reactors," Int. J. Pres. Ves. and Piping 6 1978, pp. 275-318. - Extensive dissertation on cracking mechanism in CW Zr alloy tubes in-reactor. Also includes burst test data.

(9) E.C.W. Perryman, "Pickering Pressure Tube Cracking Experience," Nucl. Energy 17 1978, No. 2, pp. 95-105 (AECL-6059). - Similar to (8) without burst data.

(10) R.R. Hosbons, "Methods of Determining Allowable Defect Sizes in Structures and Their Applicability to Nuclear Reactor Components," Met. Soc. C.I.M. Annual Vol. 1978, pp. 145-154. - General LEFM approach per ASME XI; no data.

(11) R. R. Hosbons, "Future Trends in Fracture Mechanics Theory and Applications," AECL-6198, May 1978, 21 pp. - Broad overview of fracture mechanics (lecture) not specific to AECL. Good review paper.

(12) L. A. Simpson, "Initiation COD as a Fracture Criterion for Zr-2.5% Nb Pressure Tube Alloy," Fracture, 1977 3 Waterloo, Can. (I CF4) June 19-24, 1977, pp. 705-711. - Summary discussion of COD.

(13) L.A. Simpson, "Effects of Specimen Geometry on Elastic-Plastic R-Curves for Zr-2.5% Nb," Advances in Fracture Research 1980, pp. 833-841. - Similar to work elsewhere regarding R-Curves and geometry.

(14) L.A. Simpson, "Expressions for Calculating J-Resistance Curves," Int. J. of Fracture 16, 1978, pp. R247-R249. - Usual approach to developing J-R curves.

(15) L.A. Simpson, "The Relationship Between Stress Intensity Factor, Crack Opening Displacement and J-Integral in Zr-2.5% Nb," J. Engineering Mat. & Tech. 102, January 1980, pp. 97-100. Relation J versus COD - usual- equation.

(16) L.A. Simpson and C.F. Clarke, "An Elastic-Plastic R-Curve Description of Fracture in Zr-2.5% Nb Pressure Tube Alloy," ASTM STP 668 Elastic Plastic Fracture 1979, pp. 643-662, J.D. Landes, J.A. Begley, G.A. Clarke, eds. - Fairly extensive data to develop R-curve. Only indirectly applicable to LWRs.

(17) L.A. Simpson and B.J.S. Wilkins, "Prediction of Fast Fracture in Zr-2.5% Nb Pressure Tubes Using Elastic-Plastic Fracture Mechanics," Mechanical Behavior of Materials, eds. K.J. Mills and R.F. Smith, August 1979, Vol. 3, ICM-3, pp. 563-572. - R-curve approach to predicting fast fracture in Zr-alloy tubes. Also uses burst test data per Eiber's piping tests.

Ontario Hydro Documents

(1) J. A. Baron, "Automated Ultrasonic Inspection of Bruce NGS "A" Headers Report," 80-456-K, Ontario Hydro, November 25, 1980. - Compares UT (ISIO on Bruce Nuclear Power Plant Components.

(2) O. A. Kupcis, "Nondestructive and Fracture Evaluation Section - 1980 Review and 1981 Work Program," Report 81-7-K, February 5, 1981. - Lists accomplishments, future plans, plus publications.

(3) Bruce Generating Station A Reference Plan - Periodic Inspection Program for UNIT 4 - BGA 09342-24.4. - Presents the UT plan of attack.

(4) M.K. Vanderglas, "Pickering GS "A" Unit 3 Manway Defects - ASME Section XI Fracture Analysis," 80-435-K, November 3, 1980. - Presents specific example of LEFM applied to a detected flaw in a nuclear component.

(5) 1982 Research and Development Program Plan - Fracture Mechanics Units - Nondestructive and Fracture Evaluation Section. - Lists plans in fracture mechanics research and development for 1982.

JAPAN

University of Tokyo, ISES and JAERI

On October 14, 1981 a meeting was held between Dr. Stevenson and Drs. Ando and Yogawa of the University of Tokyo, Dr. Fugimura, Technical Research Association for Integrity of Structures Elevated Service Temperatures (ISES) and Dr. Nozawa of JAERI. The purpose of the meeting was to discuss the current status in Japan of nuclear safety research and development in the areas of metallurgy, materials, NDE, and ISI in Japan.

Dr. Ando summarized three papers which had been prepared for the post SMiRT seminars on Nondestructive Examination in Relation to Structural Integrity and Fracture Resistance of Reactor Components held in August 1981. The three reports are summarized as follows:

(1) Reliability Assessment of Nondestructive Examination in Japan - To evaluate the safety of component quantitatively, it is necessary to understand the safety margins based on the flaw detectability of current nondestructive examination technique as well as adoptability of advance technique. Two activities currently undergoing in Japan are introduced in this paper. One is the proving test program of in-service inspection sponsored by the government to prove the reliability of NDE and to rationalize inspection procedure to reduce radiation exposure. Second is the round robin test by JPVRC to evaluate defect detection probability with the international cooperation.

(2) Nondestructive Examination Relating to Structural Integrity of Light Water Nuclear Power Plants; Recent Trend in Japan - With the increase of nuclear power plant, the importance of the nondestructive examination in relation to structural integrity has been widely recognized. First, a review is made on codes and regulations with incorporated the linear elastic fracture mechanics recently. New design considerations for easier in-service inspection and the recent developments of ultrasonic examination and eddy current examination systems are presented next. Finally outline of recent research program for unstable fracture of stainless steel piping is introduced.

(3) Some Recent Developments on Application of Fracture Mechanics to Reactor Components and Materials in Japan - The aim of this paper is to show some of the recent research projects in Japan concerning the fracture mechanics applications to reactor components and related materials.

Dr. Fujimura then presented a summary of the ISES Program.

The ISES is organized by 21 private companies in the field of heavy industry as identified in the registered members list, and is operated by their co-working groups, joined with researchers in universities and national research institutes.

The Japan Atomic Energy Research Institute (JAERI), Power and Nuclear Fuel Development Corporation (PNC), and Electric Power Companies have assigned to ISES several research programs concerning the structural integrity for the light water reactor, fast breeder reactor, high temperature gas cooled reactor, fusion facility, and coal conversion plants respectively.

The ISES has the following working committees:

No. 1 Committee (For FBR development)
- a) Fatigue-creep interaction problems,
- b) The secondary stress and elastic follow-up evaluation,
- c) Piping structures test,
- d) Application of bellows for essential piping.

No. 2 Committee (for HTGR development)
- a) Structure strength test using Hastelloy-X alloy,
- b) Survey of high temperature design code.

No. 3 Committee (for sodium environmental strength)
- a) Post-sodium-immersion test,
- b) Reflecting to FBR design code on the sodium environmental effects.

No. 4 Committee (for materials data gathering)
- a) Gathering of domestic structural ferritic steels, stainless steels and high alloys data,
- b) Gathering of fatigue and creep data for FBR design.

No. 5 Committee (for structural safety problems of LWR)
- a) Extreme loading evaluation,
- b) Seismic design problems.

No. 6 Committee (for internal missile problems of LWR)
- a) Large-scale test concerning internal missile of LWR,
- b) Computational analysis.

No. 7 Committee (for integrity evaluation of primary cooling piping of Monju, Proto-type FBR)
- a) Survey of fabrication and inspection process of piping system.
- b) Computation of analysis of crack propagation.

No. 8 Committee (for nuclear fusion development)
- a) Survey on the structural integrity of JT-60,
- b) Quality assurance process of JT-60.

No. 9 Committee (for integrity on primary piping of BWR)
- a) Establishment of leak-before-break conception,
- b) Crack propagation and unstable ductile fracture analysis.

No. 10 Committee (Japan-Germany collaboration for nondestructive examination)
- a) Information exchange,
- b) Cooperative test program.

No. 11 Committee (Computational analysis)
- a) Connection to O'Donnel Inc. and Science Application Inc. in USA,
- b) Consultation.

No. 12 Committee (Engineering for coal conversion)
a) Test program of materials development
b) Engineering cooperation to large-scale components fabrication.

No. 14 Committee (Fracture mechanics approach to actual components)

No. 15 Committee (Seismic test of active components)
a) Fundamental tests,
b) Evaluation for actual-scale test.

No. 16 Committee (Robots development for nuclear facilities)
a) Maintenance robot
b) Inspection robot
c) Working robot,
according to Prof. Funakubo's conception.

No. 17 Committee (Nuclear structural materials)
a) Clud problems
b) Development of Cr-Mo steels

No. 18 Committee (Piping bellows)
a) Design of piping systems
b) Experimental and computational analysis of be'lows piping.

The Registered Members List is as follows:

Asahi Chemical Industry Co., Ltd.
Fuji Electric Co., Ltd.
Hitachi, Ltd.
Hitachi Shipbuilding & Engineering Co., Ltd.
Ishikawajima-Harima Heavy Industries Co., Ltd.
The Japan Steel Works, Ltd.
Kawasaki Heavy Industries, Ltd.
Kawasaki Steel Corporation
Kobe Steel, Ltd.
Kubota, Ltd.
Mitsubishi Heavy Industries, Ltd.
Mitsubishi Metal Corporation
Mitsui Shipbuilding & Engineering Co., Ltd.
Nippon Benkan Kogyo Co., Ltd.
Nippon Kokan K.K. (Japan Steel & Tubu Corp.)
Nippon Steel Corporation
Nippon Welding Rod Co., Ltd.
Nippon Yakin Kogyo Co., Ltd.
Sumitomo Metal Industries, Ltd.
Sumitomo Shipbuilding & Machinery Co., Ltd.
Toshiba (Tokyo Shibaura Electric Co., Ltd.)

Dr. Nozawa of JAERI outline briefly the activities of JAERI and indicated that their research programs are generally identified in both the Annual Program of Safety Research(Ref.) and in the OECD report.(Ref.)

ATTACHMENT I

GLOSSARY OF ABBREVIATIONS

AE	Acoustic Emission
AH	Acoustic Holography
CSNI	Committee on the Safety of Nuclear Installations
CTOD	Crack Tip Opening Displacement
EPFM	Elastic Plastic Fracture Mechanics
ET	Eddy Current Test
FM	Fracture Mechanics
GYEM	General Yield Fracture Mechanics
HAZ	Heat Affected Zone
ISI	In-Service Inspection
LEFM	Linear Elastic Fracture Mechanics
NDE	Nondestructive Examination
PISC II	Program for Inspection of Steel Components
RPV	Reactor Pressure Vessel
UT	Ultrasonic Testing

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