NUREG-1125
Volume 4
Generic Subjects A-G

A Compilation of Reports of **The Advisory Committee on Reactor Safeguards**

1957-1984

U.S. Nuclear Regulatory Commission

April 1985

ABSTRACT

This six-volume compilation contains over 1000 reports prepared by the Advisory Committee on Reactor Safeguards from September 1957 through December 1984. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and within project name by chronological order. Part 2 categorizes the reports by the most appropriate generic subject area and within subject area by chronological order.

PREFACE

This compilation has been prepared from the ACRS Report Notebooks that are kept in the ACRS Office. The notebooks are divided into two main sections, ACRS reports on specific projects, and ACRS reports on generic subjects. Normally, each report is filed in only one notebook subsection, with some cross referencing when appropriate. In one or two instances, a report is filed in more than one location to assist the notebook users. Every effort has been made to make this compilation as complete as possible, but due to the relative length of time covered by the notebooks and the variations in record keeping procedures, it is possible that some reports may have been inadvertently omitted.

This compilation does not contain ACRS reports that contain classified or other controlled information.

FOREWORD

The Advisory Committee on Reactor Safeguards (ACRS) was created in 1953 to provide advice to the Atomic Energy Commission (AEC) on the safety of reactor systems being developed at the onset of the era of civilian use of nuclear energy. In 1957, the ACRS was established as a statutory advisory body by the Atomic Energy Act of that year.

The ACRS has continued for over 30 years in this advisory role to the AEC and its successor in reactor safety regulation, the Nuclear Regulatory Commission (NRC). The Committee has played a central role in the development of safety standards and practices as nuclear power has grown from a glamorous scientific curiosity to a huge industry, beset with not only pains of spectacularly rapid initial growth, but also subsequent public disenchantment and controversy. The influence of the ACRS has been projected in a number of ways; through its direct contact with the AEC/NRC technical staff, the industry, the national laboratories, and the universities, and, since 1973, especially through its public meetings. However, its formal advice is given in the form of letter reports to the Commission it advises. These reports, expressing the collegial opinion of the 15-member ACRS have covered a wide variety of subjects, from statements of approval, often with caveats, for the licenses for every plant in the nation, to comments on significant technical issues. Some of the reports have been landmarks and have had a major influence on the development of nuclear power and of the safety of nuclear power. Most have been more mundane and served principally to help keep the regulatory system moving along a fair and responsible course. A few may have been unwise, and better forgotten or rescinded. But, we believe these reports, taken as a whole, provide an interesting view of the history of nuclear power in the United States and the rest of the world.

On the occasion of the 300th regular meeting of the ACRS, April 11-13, 1985, we have published these volumes of the Committee's collected reports. We trust they will be of value to those interested in the past and the future of the generation of electricity and other practical uses of nuclear power.

David A. Ward
Chairman, ACRS

TABLE OF CONTENTS

VOLUME I

	Page
ABSTRACT	iii
PREFACE	v
FOREWORD	vii

PART I: ACRS REPORTS ON PROJECT REVIEWS

Advanced Test Reactor (ATR)	1
Aerospace Systems Test Reactor (ASTR)	8
Air Force Nuclear Engineering Test Facility (AFNETF)	10
Allens Creek Nuclear Generating Station Units 1 and 2	13
Argonne Advanced Research Reactor (AARR)	18
Argonne Low Power Reactor - SL-1 (ALPR)	20
Arkansas Nuclear One Unit 1 (Formerly Russellville Nuclear Unit)	24
Arkansas Nuclear One Unit 2	31
Army Package Power Reactors:	
MH-1A (Floating Nuclear Power Plant)	38
SM-1 (Fort Belvoir)	44
SM-1A (Fort Greely, Alaska)	47
Arnold, Duane Energy Center	54
Atlantic Generating Station (See Floating Nuclear Plant)	
Babcock and Wilcox Test Reactor (BAWTR)	65
Bailly Generating Station Nuclear 1	70
Barnwell Nuclear Fuel Plant	77
Beaver Valley Power Station Units 1 and 2	80
Bellefonte Nuclear Plant Units 1 and 2	92
Bethesda Naval Medical Center - DASA-TRIGA Reactor	96
Big Rock Point Plant (Consumers)	97
Black Fox Station Units 1 and 2	115
Blue Hills Station	118
Bodega Bay Atomic Park Unit 1	121
Boiling Nuclear Superheater Power Station (BONUS)	128
Boiling Reactor Experiment V (BORAX V)	135
Braun, C.F., Standard Turbine Island Design	139
Brookhaven High Flux Beam Research Reactor (HFBRR)	142
Brookwood Station (See Ginna, R. E.)	
Browns Ferry Nuclear Power Station Units 1, 2 and 3	150

TABLE OF CONTENTS (CONT'D)

	Page
Brunswick Electric Steam Plant Units 1 and 2	174
Byron/Braidwood Station	186
California Department of Water Resources (CDWR)	193
Callaway Plant Units 1 and 2	196
Calvert Cliffs Nuclear Power Plant Units 1 and 2	205
Carolinas Virginia Tube Reactor (CVTR)	217
Catawba Nuclear Station Units 1 and 2	232
Cherokee/Perkins Nuclear Station Units 1, 2 and 3	239
Clinch River Breeder Reactor (CRBR) Plant Site	244
Clinton Nuclear Power Plant	251
Comanche Peak Steam Electric Station Units 1 and 2	256
Connecticut Yankee (Haddam Neck) Plant	263
Cook, Donald C., Nuclear Plant Units 1 and 2	278
Cooper Nuclear Station	306
Crystal River Nuclear Generating Plant Unit 3	313
Dairyland Power Cooperative (SPWR) (See also ICBWR)	321
Davis-Besse Nuclear Power Station Units 1, 2 and 3	323
Diablo Canyon Nuclear Power Station Units 1 and 2	336
Douglas Point Nuclear Generating Station Units 1 and 2	384
Dresden Nuclear Power Station Units 1, 2 and 3	387
Elk River Reactor (Rural Cooperative Power Association (RCPA))	437
Erie Nuclear Plant Units 1 and 2	459
ESADA - Vallecitos Experimental Superheat Reactor (EVESR) (See Vallecitos BWR)	
Experimental Beryllium Oxide Reactor (EBOR)	462
Experimental Boiling Water Reactor (EBWR)	465
Experimental Breeder Reactor (EBR-II)	472
Experimental Gas-Cooled Reactor (EGCR)	477
Experimental Low Temperature Process Heat Reactor (ELPHR)	483
Experimental Organic-Cooled Reactor (EOCR)	487
Farley, Joseph M., Nuclear Plant Units 1 and 2	491
Fast Flux Test Facility (FFTF)	499
Fast Reactor Core Test Facility (FRCTF)	522
Fast Reactor Test Facility (FARET)	524
Fast Breeder Reactor (FBR) Demonstration Plant	527
Fermi, Enrico, Atomic Power Plant Units 1 and 2	531
Fission Product Conversion and Encapsulation Plant (FPCE)	549
Fitzpatrick, James A., Nuclear Power Plant	551

Page

Floating Nuclear Plant (Includes Atlantic Generating Station and Platform Mounted Nuclear Plant) 559
Forked River Nuclear Generating Station Unit 1 610
Fort Calhoun Station Unit No. 1 616
Fort St. Vrain Nuclear Generating Station 624
Fulton Generating Station Units 1 and 2 636

VOLUME II

Gas-Cooled Fast Breeder Reactor (GCFBR) 641
General Electric Test Reactor (GETR) 645
Ginna, R. E., Nuclear Station Unit 1 (Formerly Brookwood) 654
Grand Gulf Nuclear Station Units 1 and 2 671
Greene County Nuclear Power Plant 684
Greenwood Energy Center Units 2 and 3 686
Ground Test Reactor (GTR) 689

Haddam Neck Plant (See Connecticut Yankee)
Hallam Nuclear Power Facility (HNPF) 691
Hanford No. 2 Nuclear Power Plant 705
Harris, Shearon, Nuclear Power Plant 708
Hartsville Nuclear Plant Units A-1, A-2, B-1 and B-2 724
Hatch, Edwin I., Nuclear Plant Units 1 and 2 727
Heat Transfer Reactor Experiment - 3A (HTRE-3A) 746
Heavy Water Components Testing Reactor (HWCTR) 748
High Flux Isotope Reactor (HFIR) 753
Hope Creek Generating Station Units 1 and 2 (See also Newbold Island) 764
Humboldt Bay Power Plant Unit No. 3 (Pacific Gas & Electric) 771
Hutchinson Island Plant Unit 1 790

Improved Cycle Boiling Water Reactor (ICBWR) (City of Los Angeles and Dairyland Power Cooperative) 795
Indian Point (Con Ed) Nuclear Generating Units 1, 2 and 3 805

Jamesport Nuclear Power Station Units 1 and 2 842

Kewaunee Nuclear Power Plant 846
Koshkonong Nuclear Plant Units 1 and 2 855

TABLE OF CONTENTS (CONT'D)

	Page
La Crosse Boiling Water Reactor (LACBWR)	865
La Salle County Station Units 1 and 2	879
Limerick Generating Station Units 1 and 2	884
Lithium-Cooled Reactor Experiment (LCRE)	894
LOFT Facility	895
Los Angeles, City of (Malibu Reactor)	900
Low Temperature Process Heat Reactor (LTPHR)	919
Maine Yankee Atomic Power Station	921
Malibu Nuclear Plant Unit 1 (See Los Angeles, City of)	
Marble Hill Nuclear Generating Station Units 1 and 2	929
Massachusetts Institute of Technology (MIT) Reactor	933
Materials Testing Reactor (MTR)	935
McGuire Nuclear Station Units 1 and 2	937
MH-1A (See Army Package Power Reactors)	
Midland Plant Units 1 and 2	943
Midwest Fuel Recovery Plant (MFRP)	973
Military Reactors (Atomic Energy Act of 1954, as amended, § 91.b.)	978
Millstone Nuclear Power Station Units 1, 2 and 3	980
Molten Salt Reactor Eperiment (MSRE)	1008
Montague Power Station Units 1 and 2	1012
Monticello Nuclear Generating Plant Unit 1 (Northern States Power)	1015
National Aeronautics and Space Administration (NASA):	
Mock-Up Reactor (MUR)	1029
Plum Brook Reactor (PBR)	1031
National Bureau of Standards (NBS) Reactor	1048
National Reactor Testing Station (NRTS)	1059
New England Power Company Nuclear Units NEP 1 and 2	1061
Newbold Island Nuclear Generating Station Units 1 and 2 (See also Hope Creek)	1064
Nine Mile Point Nuclear Station (Niagara Mohawk)	1079
North Anna Power Station Units 1, 2, 3 and 4	1097
Nuclear Fuel Services (NFS), Inc.	1137
Nuclear Power Plants in California (See also So. Cal. Ed.)	1144
Oak Ridge National Laboratory (ORNL):	
Research Reactor (ORR)	1151
X-10 Reactor (Annealing)	1153
Oconee Nuclear Station Units 1, 2 and 3	1154
Oyster Creek Nuclear Generating Station (Jersey Central)	1167

TABLE OF CONTENTS (CONT'D)

	Page
Palisades Plant	1185
Palo Verde Nuclear Generating Station Units 1, 2 and 3	1196
Pathfinder Atomic Power Plant	1207
Peach Bottom Atomic Power Station Units 1, 2 and 3	1219
Pebble Springs Nuclear Plant Units 1 and 2	1238
Perkins/Cherokee Nuclear Station Units 1, 2 and 3 (See Cherokee)	
Perry Nuclear Power Plant Units 1 and 2	1245
Phipps Bend Nuclear Plant Units 1 and 2	1256
Picatinny Arsenal Ordnance Corps Research Reactor (OCRR)	1259
Pilgrim Nuclear Power Station Units 1 and 2	1261
Piqua Nuclear Power Facility	1277
Platform Mounted Nuclear Plant (See Floating Nuclear Plant)	
Plutonium Recycle Test Reactor (PRTR)	1288
PM Reactors (See also Army Package Power Reactors):	
PM-1 Reactor	1295
PM-2A Reactor	1299
PM-3A Reactor	1302
Point Beach Nuclear Plant Units 1 and 2	1307
Pool Type Reactors	1316
Power Burst Facility (PBF)	1317
Prairie Island Nuclear Generating Plant Units 1 and 2	1329
Prototype Organic Power Reactor (POPR)	1336
Puerto Rico Water Resources Authority (Tortuguero Site)	1338

VOLUME III

Quad-Cities Station Units 1 and 2	1341
Radiation Effects Reactor (RER) (Lockheed)	1349
Rancho Seco Nuclear Generating Station Unit 1	1364
River Bend Station Units 1 and 2	1374
Robinson, H. B., Unit 2	1382
Rome Point Nuclear Generating Station	1392
Russellville Nuclear Unit (See Arkansas Nuclear One, Unit 1)	
St. Lucie Plant Units 1 and 2 (See Hutchinson Is. for CP Report)	1397
Salem Nuclear Generating Station Units 1 and 2	1410
Sandia Pulsed Reactor Facility (SPRF)	1422
San Joaquin Project	1424
San Onofre Nuclear Generating Station Units 1, 2 and 3 (See also Southern California Edison - Camp Pendleton)	1427

TABLE OF CONTENTS (CONT'D)

	Page
SAVANNAH, N.S. (Merchant Ship)	1444
Saxton Nuclear Experimental Corporation Reactor	1497
Seabrook Station Units 1 and 2	1509
Sequoyah Nuclear Plant Units 1 and 2	1519
Shippingport Atomic Power Station's PWR	1539
Shoreham Nuclear Power Station Unit 1	1548
Skagit Nuclear Power Project Units 1 and 2	1560
Small Pressurized Water Reactor (SPWR) (Jamestown Site)	1566
Sodium Reactor Experiment (SRE)	1573
Southern California Edison - Camp Pendleton (See also Nuclear Power Plants in California, City of Los Angeles, ICBWR)	1575
South Texas Project Units 1 and 2	1579
Southwest Experimental Fast Oxide Reactor (SEFOR)	1582
Spent Fuel Shipment Cask Program (See Transportation of Radioactive Materials)	
SPERT I, II and III Reactors	1594
Standardized Nuclear Unit Power Plant (SNUPPS) (See Callaway, Sterling, Wolf Creek and Tyrone Nuclear Plants)	
Sterling Power Project Nuclear Unit 1	1600
Summer, Virgil, Nuclear Station Unit 1	1604
Summit Power Station Units 1 and 2	1611
Sundesert Nuclear Power Plant Units 1 and 2	1615
Surry Power Station Units 1, 2, 3 and 4	1619
Susquehanna Steam Electric Station Units 1 and 2	1629
Three Mile Island Units 1 and 2	1637
Trojan Nuclear Plant Unit 1	1726
Turkey Point Nuclear Generating Plant Units 3 and 4	1734
Tyrone Energy Park Unit 1	1742
Vallecitos Boiling Water Reactor (VBWR)/(EVESR)	1745
Vermont Yankee Nuclear Power Station	1763
Vogtle, Alvin W., Nuclear Plant Units 1, 2, 3 and 4	1778
Wahluke Slope	1783
Washington Public Power Supply System (WPPSS) Nuclear Power Stations WNP 1 and 4, 2, 3 and 5	1785
Waterford Steam Electric Station Unit 3	1796
Watts Bar Nuclear Plant Units 1 and 2	1806
Westinghouse Testing Reactor (WTR)	1813
Westinghouse Ice Condenser Pressure-Suppression Concept (See Engineered Safeguards, Volume IV)	
Wolf Creek Generating Station Unit 1	1819

Page

Yankee-Rowe Nuclear Plant 1826
Yellow Creek Nuclear Power Plant Units 1 and 2 1850

Zero Power Plutonium Reactor (ZPPR) 1853
Zimmer, William H., Nuclear Power Station Unit 1 1855
Zion Station Units 1 and 2 1862

VOLUME IV

PART 2: ACRS REPORTS ON GENERIC SUBJECTS

Accident Analysis 1885
Aerospace 1889
Anticipated Transients Without Scram (ATWS) 1895

Babcock and Wilcox Reactors 1917
Babcock-205 Standard Nuclear Steam System 1921

Class 9 Accidents 1925
Combustion Engineering Standard Safety Analysis Report (CESSAR-80) 1995
Control Rods and Drives 2007
Criteria 2009

Decay Heat Removal Systems 2025
Decommissioning of Nuclear Power Plants 2031

Emergency Core Cooling Systems (ECCS) 2033
Emergency Planning 2091
Engineered Safeguards 2113
Extreme External Phenomena (See also pp. 3418-3419) 2143

Fire Protection 2183
Fluor Power Services (BOPSSAR) 2185

General Electric Company:
  BWR Design - BWR/6 2189
  GESSAR-II BWR/6 2192
  GESSAR-238 2195
  GESSAR-238 NSSS and GESSAR-251 2199
  GETR (See GETR, Volume II)
  Mark I Containment Acceptance Criteria 2202
  Mark III Containment Design and BWR Containments 2204
  8x8 Fuel Design 2207
Generic Items Relating to Light Water Reactors 2213

TABLE OF CONTENTS (CONT'D)

VOLUME V

	Page
High Temperature Gas-Cooled Reactor (HTGR)	2493
Human Factors	2495
Hydraulic Positioning (Hypo) Control System Concept	2519
Hypothetical Core Disruptive Accidents (HCDA) for LMFBRs	2521
Inspection and Enforcement	2525
Joint Committee on Atomic Energy (JCAE) (See also Procedures)	2539
Licensee Event Reports (LERs)	2571
Metal Components	2625
Miscellaneous Letters	2657
Mixed Oxide Fuels	2679
Power and Electrical Systems	2685
Procedures - ACRS/Regulatory/Legal	2719
Qualification Systems/Equipment	2895
Quality Assurance/Quality Control	2901
Radiological Effects	2907
Radioactive Waste Management	2957
Reactor Fuels	3003
Reactor Operations	3005
Reactor Pressure Vessels NDT Radiation Damage	3011
Regulatory Guides	3027

VOLUME VI

Reliability and Probabilistic Analysis	3107
Requests and Recommendations	3161
Review of Regulatory Practices and Reactor Safety (Allegations)	3197
Rules and Regulations	3351
Safeguards and Security (Sabotage)	3363
Safety Research	3377
Selected Safety Issues (See Review of Regulatory Practices...)	

TABLE OF CONTENTS (CONT'D)

	Page
Site Criteria	3535
SNUPPS (See Callaway Units 1 and 2, Volume I, pp. 201-203)	
SWESSAR	3563
Systematic Evaluation Program/Ten-Year Review	3579
Systems Interaction	3591
Transportation of Radioactive Materials	3601
WASH-1400 (Reactor Safety Study)	3625
Westinghouse Electric Corporation:	
RESAR-3S	3633
RESAR-41	3636
RESAR-414	3643
APPENDIX A: Other ACRS Publications	3647
APPENDIX B: List of ACRS Members by Calendar Year	3649
APPENDIX C: Background	3653

A

Accident Analysis

Aerospace

Anticipated Transients Without Scram (ATWS)

A

Accident Analysis

Aerospace

Anticipated Transients Without Scram (ATWS)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 8, 1978

The Honorable Morris K. Udall, Chairman
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Udall:

The Advisory Committee on Reactor Safeguards (ACRS) has considered the suggestion in your letter of January 27, 1978 for establishment of an independent, quasi-judicial board, patterned after the National Transportation Safety Board (NTSB), for accident analysis within the context of the current nuclear regulatory process. The Committee considered also the questions which you raised concerning the role of the ACRS vis-a-vis such a Board, should it be created.

Discussions with representatives of the NTSB's Bureau of Accident Investigation have indicated that, although the NTSB is responsible for investigating accidents in surface, air, and marine transportation, the criteria, procedures, and scope of the investigations vary depending on the specific mode of transportation involved. Air transport events, however, represent the bulk of NTSB work and range from minor incidents to serious accidents. It probably is the most well established area of NTSB's responsibility. In response to your inquiry the ACRS compared the nuclear power program requirements with air transportation investigation procedures.

While the NTSB reports on all aviation accidents, the bulk of the investigations, which are concerned with minor accidents or incidents, are delegated to the FAA, the involved regulatory agency. NTSB investigations are reserved for major accidents, generally involving fatalities. Analogous major accidents have not occurred in commercial nuclear power plant operation. Indeed, the Nuclear Regulatory Commission (NRC) and the ACRS devote a significant effort to reviewing operational experiences, proposed changes in operating procedures, and plant design features intended to forestall such accidents and continuing discussion of this process with the NRC Staff is planned. For this reason, the ACRS believes that existing institutional arrangements are adequate for the range of incidents thus far experienced in nuclear power plant operation. Should there be

an accident comparable in magnitude and significance to those now investigated by NTSB, it is within the mandate of the ACRS to conduct a comprehensive and independent investigation of it. Therefore, our opinion is that no need exists to establish an independent board to carry out this function.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 11, 1984

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

SUBJECT: ACRS COMMENTS ON THE POSSIBILITY OF AN ORGANIZATION LIKE THE NATIONAL TRANSPORTATION SAFETY BOARD (NTSB) FOR NUCLEAR SAFETY

Since the ACRS discussion with the Commission about the possibility of an NTSB-like organization for nuclear safety has been postponed, we would like to pass on some recommendations we would have made during the meeting.

You will recall that the Committee and the Commission reacted negatively to the idea when it first appeared in 1978, partly because up to that time there had not been accidents recognized as worthy of the kind of detailed study implied by the creation of a new agency, and partly because there was confusion about the role of the study of precursors in accident avoidance. Nonetheless, the Commission created the Office for Analysis and Evaluation of Operational Data (AEOD). It has provided independent analyses of operating events, and has contributed to the identification of precursors. We believe the time has come to take an additional step, by elevating its status to that of a Commission-level office.

Although the original proposal for an "NTSB" called for an independent organization (simply to assure its freedom to criticize the NRC), we have seen no evidence that the independence of AEOD has been infringed. Yet, it is a part of the NRC Staff structure and conflicts are not beyond reasonable conjecture. It seems to us prudent, now that AEOD has proven itself, to point to the future by removing this possibility.

Such conflicts are not unthinkable. The NTSB has just issued its findings on the amazing event in which an Eastern L-1011 lost all three of its engines because O-rings were left off the master chip detector plugs in the lubricating oil systems on all engines during the same maintenance operation. The NTSB placed much of the blame on the airline and the maintenance crew, but also placed substantial blame on the FAA inspectors for having failed to follow up earlier incidents of the same kind involving the same airline. Given the current reporting chain of AEOD, it is easy to see where problems might arise in a similar circumstance. We believe there is much to be gained and little to be lost by having AEOD report at a higher organizational level, especially as we move to a more operationally oriented NRC.

There is a clear administrative problem which we propose to solve by volunteering. We would suggest that Commission overview of AEOD be supported by asking ACRS to establish a standing subcommittee for that purpose. Such a subcommittee could, inter alia:

(a) provide a critical link to the full Committee and thereby an additional link to the Commission on the activities of AEOD,

(b) meet frequently with AEOD for mutual reinforcement,

(c) act as a technical bridge, to sharpen both groups' awareness of potential nuclear safety problems, and

(d) seek full Committee action and Commission support, as needed, to assure the independence and continuity of AEOD.

We believe the suggestion represents a reasonable compromise between the original idea of a wholly independent agency and the NRC organization now in place.

We would be happy to discuss these matters further with you.

Sincerely yours,

Jesse C. Ebersole
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C.

May 12, 1962

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Dear Dr. Seaborg:

Dr. Silverman, who was then Chairman of the Advisory Committee on Reactor Safeguards, was visited by Mr. William Finan, Deputy General Manager, in the Fall of 1960 to discuss the needs of the Atomic Energy Commission and participation of the ACRS in connection with the review of health and safety aspects of aerospace projects. Dr. Silverman replied that the ACRS would be willing to participate in this review in a letter to General Luedecke, dated October 14, 1960, of which a copy is attached, remarking that an early program briefing to acquaint the Committee with the nature and scope of the various aerospace projects would be very desirable. General Luedecke, in a letter to Dr. Silverman, dated November 14, 1960 (copy attached) indicated that the AEC would discuss this with the Committee after certain broad Commission decisions had been reached on the assignment of responsibilities for health and safety aspects of aerospace projects.

Prompted by the feeling that the space age is arriving at a rather brisk pace, I recently reopened this subject in an informal discussion with Mr. Lowenstein, Director of the Division of Licensing and Regulation. He agrees that an early program briefing of the ACRS is highly desirable and is arranging a briefing session of several hours on aerospace projects, tentatively planned for presentation during the June 28-30 meeting of the full Committee.

It appears to me, if the Committee is to undertake responsibility for safety review of aerospace nuclear projects, that it would be well to explore the bounds of this responsibility quite soon so that the Committee can develop the background necessary to deal with aerospace problems as expeditiously as possible. It was for this reason that I proposed this item for discussion at our May 10, 1962 meeting with you.

Sincerely yours,

/s/ F. A. Gifford, Jr.

F. A. Gifford, Jr.
Chairman

Att. Ltr. Silverman to Luedecke dtd. 10/14/60
Ltr. Luedecke to Silverman dtd. 11/14/60

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

October 14, 1960

A. R. Luedecke
General Manager
U. S. Atomic Energy Commission
Washington, D. C.

Dear General Luedecke:

On September 29, 1960, Mr. William Finan, Deputy General Manager, came to Boston to discuss the needs of the Commission for an independent review group for examining the health and safety aspects of the aerospace nuclear program. He indicated that rather than constituting a new group it would be preferable to have the Advisory Committee on Reactor Safeguards undertake such reviews if it was willing to do so since most of the problems appeared to lie within our area of concern.

I indicated a tentative affirmative answer at the time and upon consultation with my committee members I am able to confirm that decision by this letter.

The ACRS at the request of the Commission and in discharge of its obligation is willing to accept responsibility for safety review of the aerospace nuclear projects. We believe that the Committee can best serve the Commission on this matter if its safety reviews of these projects are general ones rather than detailed technical evaluations. Advice would be furnished primarily in terms of safety policy in regard to the nuclear components of these systems and the effects of their possible failure on the health and safety of the public.

If the Committee is to review the nuclear safety aspects of this program it is desirable to schedule an early general program briefing to acquaint the full Committee with the nature and scope of the various projects.

Sincerely yours,

/s/ LESLIE SILVERMAN

Leslie Silverman
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

November 14, 1960

Dr. Leslie Silverman, Chairman
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington 25, D. C.

Dear Dr. Silverman:

This is in reply to your letter of October 14, 1960, stating that the ACRS would be willing to participate in the safety reviews of aerospace nuclear projects should the Commission request it.

When the Commission has reached some broad decisions regarding the assignment of responsibilities for the health and safety aspects of aerospace nuclear projects, I shall wish to discuss with you the details of ACRS participation, to provide a solid basis of mutual understanding which can be reduced to writing.

Meanwhile, please express to the members of the ACRS our appreciation of their willingness to assist us in what we expect will be some very important and difficult determinations.

Sincerely yours,

signed A. R. LUEDECKE

General Manager

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C.

December 26, 1962

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Dear Dr. Seaborg:

On the previous occasions referenced below, the Committee has written to the Atomic Energy Commission about the problem of reviewing the health and safety aspects of the aerospace nuclear program. The Committee is particularly concerned with the implications of this program in regard to exposure of the general public to radiation hazards from such nuclear powered devices.

Aside from limited consideration of the SNAP-50 program in connection with the Lithium Cooled Reactor Experiment on which the Committee has reported, and a brief review of the SNAP program before our Reactor Safety Research Subcommittee at Idaho in August of this year, the Committee has not considered health and safety aspects of the aerospace nuclear program.

Because the aerospace program is expanding rapidly the Committee believes that a comprehensive review of the safety aspects appears to be essential. In order to avoid consideration of portions of this program at a date when any suggested change may cause substantial loss in time or money, the Committee requests that it be advised by the Commission whether or not this responsibility will be assigned to the Committee.

Sincerely yours,

/s/ F. A. Gifford, Jr.

F. A. Gifford, Jr.
Chairman

References:
1. Letter from Dr. Gifford to Dr. Seaborg, dated May 12, 1962.
2. Letter from Dr. Silverman to Gen. Luedecke dated October 14, 1960.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 19, 1966

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: AEROSPACE NUCLEAR SAFETY

Dear Dr. Seaborg:

At its sixty-eighth meeting, in November 1965, the Advisory Committee on Reactor Safeguards was briefed by the Space Nuclear Systems Division on the current status of the aerospace nuclear program with special emphasis on the organization and procedures for safety review of space nuclear operations. The Committee wishes to express its appreciation for being brought up to date in this area and hopes that from time to time further such information sessions can be planned.

At the briefing, the Committee was told that plans called for primary technical responsibility for review of the nuclear safety aspects of the aerospace program to reside in an interagency safety committee, made up of two members each from AEC, DOD, and NASA. The technical safety recommendations of this committee would be reviewed at a high level by representatives of AEC, DID, and HASA. The AEC might, for example, be represented by Commissioners. The recommendation of this review would then go to the Space Council, and finally to the President.

During discussion with the Commissioners at the Sixty-Ninth ACRS Meeting in January 1966, the Committee was advised that the AEC is now proposing that the interagency safety committee also include independent experts who do not have a direct organizational interest in the aerospace program. The Committee strongly endorses the intent of this proposed modification and urges that a sufficient number of independent members be included on the interagency safety committee to insure objectivity in this safety review which we believe is necessary to protect the health and safety of the public.

Sincerely yours,

/s/ David Okrent

David Okrent
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

May 10, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. Schlesinger:

The ACRS has provided comments to the Director of Regulation regarding a proposed position for protection against anticipated transients without scram. These comments are based on discussions during several ACRS meetings and twelve meetings of ACRS generic Subcommittees as well as discussions with reactor vendors during various project reviews since February, 1969.

A copy of this letter is attached for your information.

Sincerely yours,

/s/
C. P. Siess
Chairman

Attachment:
Letter from C. P. Siess to
L. Manning Muntzing

cc: L. Manning Muntzing, DR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

May 10, 1972

Mr. L. Manning Muntzing
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Muntzing:

Your letter of April 28, 1972, describes requirements for protection against anticipated transients without scram (ATWS) which the Regulatory Staff plans to impose on applicants for construction permits, and notes that ACRS comments on the proposed requirements received soon after the ACRS May meeting would be timely. The Committee wishes to make the following comments.

I. The ACRS recognizes ATWS as a low probability event. Nevertheless, it believes that, in consideration of the large number of water-cooled power reactors expected eventually to be in operation, and in view of the expected occurrence rate of anticipated transients (collectively, on the order of one per reactor per year), experience with scram systems of current design is insufficient to give assurance of an adequately low probability of occurrence of an ATWS event of possibly serious consequence. Accordingly, the Committee agrees with the intent of the ATWS position recommended, viz:

> "Applicants should be required to: (1) demonstrate that with their present designs the consequences of anticipated transients without scram (ATWS) are acceptable, or (2) make design changes which render the consequences of anticipated transients without scram acceptable, or (3) make design changes to improve significantly the reliability of the scram system."

II. The Committee has the following comments on the criteria proposed to be used in implementation of the basic position.

A. In respect to the proposed definition of "acceptable consequences" for implelemtation of either option 1 or option 2:

> 1. Concerning radiological consequences, we agree with the proposed condition: "The radiological consequences shall be within the guideline values set forth in 10 CFR Part 100".

2. Concerning primary system pressure:

a. We agree with the intent of the proposed condition: "The transient pressure shall be limited to less than that resulting in a maximum stress anywhere in the reactor coolant pressure boundary of the 'emergency conditions' as defined in the ASME Section III Nuclear Power Plant Components Code". However, we believe the wording should be changed so as to read along the following lines: "The transient pressure shall not be greater than that which results in reactor coolant pressure boundary stress conditions corresponding to those of 'emergency conditions' as defined in ASME Section III Nuclear Power Plant Components, 1971". We believe it should be noted that the intent of this provision is to obviate the need to consider a loss-of-coolant accident (LOCA) in conjunction with an ATWS event.

b. We agree with the intent of the proposed condition as expressed in the first sentence: "The transient pressure shall not exceed a value for which test and/or analysis demonstrate that there is no substantial safety problem with the fuel". However, we recommend that the second sentence be deleted.

3. Concerning fuel thermal and hydraulic effects:

a. The Committee believes that the proposed limit on enthalpy of the peak pellet of 280 calories per gram should not be adopted at this time. A single limit for all cases may not even be desirable. It is recommended that the criterion be changed to indicate that, in an ATWS involving a power excursion, the effects of rapid increase in fuel enthalpy shall not result in significant cladding degradation or in significant melting of fuel even in the hottest fuel zones.

b. We agree with the intent of the proposed condition: "A calculated initial heat flux event will not be acceptable unless the peak cladding temperature can be shown not to result in significant cladding degradation".

4. Concerning containment conditions, we agree with the intent of the proposed condition: "Calculated containment pressure shall not exceed the design pressure of the containment structure. Equipment which is located within the containment and which is relied upon to mitigate the consequences of ATWS shall be qualified by testing in the combined pressure, temperature and humidity environment conservatively predicted to occur during the course of the event".

B. We agree with the intent of the proposed requirement B:

"Any modifications made to comply with option 2 of the recommended position shall be shown not to result in violations of safety criteria for steady state, transient, or accident conditions and shall not substantially affect the operation of safety related systems."

C. We agree with the intent of the proposed requirement C:

"Design changes to make the consequences of ATWS acceptable should not rely on equipment or system designs which have a failure mode common with the scram system. The equipment involved in the design change shall, to the extent practical, operate on a different principle from equipment in the scram system. As an absolute mimimum, the equipment relied on to render acceptable the consequences of the ATWS event shall not include equipment identical to equipment in the associated scram system."

D. We agree with the intent of the proposed requirement D:

"Improvements must reduce considerably the potential for common mode failure of the scram system. Failures of identical equipment from a common mode should not disable sensing circuits, logic, actuator circuits or control rods to the extent that scram is ineffective. The addition of a separate protection system utilizing principles diverse from the primary protection system is indicated in order to meet this requirement."

III. In addition to the above comments on the proposed requirements, the Committee makes the following recommendations:

A. In any announcement of the basic position recommended, as well as in its implementation, care should be taken to make clear the fact that availability of options 1 and 2 is not to be construed as prejudicing in any way the importance of continuing effort to improve existing scram systems to the extent practical, irrespective of ATWS considerations.

B. As indicated in your letter, Regulatory Staff studies are continuing for the purpose of developing lists of transients to be considered, assumptions to be used, and acceptable evaluation models. We recommend that this effort be accelerated to the extent practical, in order that a maximum of guidance be available to the applicants upon commencement of implementation. It is also recommended that the list of transients to be treated be described as a minimum but not necessarily sufficient list, with the applicant responsible for identifying all relevant transients.

C. The Committee recommends that the proposed position, modified as above, also be implemented on a reasonable time scale in respect to those water-cooled power reactors under construction for which the ACRS letter and the Regulatory Staff safety evaluation associated with the construction permit identified the ATWS problem. The Committee assumes that, in due course, the Regulatory Staff will propose an appropriate course of action in connection with earlier plants.

IV. The Committee intends to continue working closely with the Regulatory Staff in the further study and development of criteria and procedures to be applied in the ATWS area.

Sincerely yours,

/s/

C. P. Siess
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

April 16, 1973

Mr. L. Manning Muntzing
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Muntzing:

Your letter of January 22, 1973, transmitted copies of a revised recommended licensing position on the need for protection against anticipated transients without scram (ATWS) for water-cooled power reactors, invited ACRS comment. The Committee considers this position to be generally consistent with your earlier proposed position favorably commented on by Committee letter of May 10, 1972, and supports the position.

It is suggested that, in those cases of "backfit" plants for which an OL stage or post-OL stage ACRS report or AEC Safety Evaluation identified ATWS as a continuing area of review, the philosophy used in evaluating the possible need for plant design changes should be similar to that for plants in Class I.B, recognizing the need for special consideration of the problems that may result if extensive modifications appear to be required.

The Committee also continues to believe that it is timely to begin implementation of the proposed ATWS position.

Sincerely yours,

/s/ H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

cc: J. M. Hendrie, L
J. F. O'Leary, L
P. C. Bender, SECY
A. Gaimbusso, L

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 17, 1975

Mr. Lee V. Gossick
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: ALTERNATIVES TO STAFF POSITION REGARDING ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Dear Mr. Gossick:

On September 12, 1975, Mr. Ben C. Rusche, Director, Office of Nuclear Reactor Regulation, transmitted to the Advisory Committee on Reactor Safeguards a working paper, dated September 12, 1975, and entitled "Discussion of Alternatives to the Previous Staff Position on Anticipated Transients Without Scram for Construction Permit Applications Made After October 1, 1976."

The working paper primarily discusses various possible new regulatory approaches to "Class A" plants as categorized in WASH-1270, "Anticipated Transients Without Scram for Water-Cooled Power Reactors," dated September, 1973. The Class A category of WASH-1270 requires design changes that would improve significantly the reliability of the shutdown system. This category was to apply to those applications docketed subsequent to October 1, 1976. The Class B category of WASH-1270 applies to reactors docketed from early 1968 to October 1, 1976, and requires that the consequences of a postulated ATWS must be acceptable and that those portions of the reactor shutdown system which might be particularly vulnerable to common mode failures be identified and corrected.

Subsequent to the publication of WASH-1270, the NRC Staff has continued to review the several vendor proposals for meeting Class A and Class B requirements. The NRC Staff has advised the Committee that it now believes it would be appropriate that Class A plants be treated in the same manner as Class B plants.

The ACRS has reviewed the working paper and discussed the matter with the NRC Staff during the Committee's 186th Meeting, October 9-11, 1975. The Committee concurs with the NRC recommendation.

Sincerely yours,

W. Kerr

W. Kerr
Chairman

REFERENCES:

1. U.S. Atomic Energy Commission Regulatory Staff's Technical Report on Anticipated Transients Without Scram For Water-Cooled Power Reactors, WASH-1270, dated September, 1973.

2. U.S. Nuclear Regulatory Commission Staff's Working Paper: Discussion of Alternatives To The Previous Staff Position on Anticipated Transients Without Scram for Construction Permit Applications Made After October 1, 1976, dated September 12, 1975.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 14, 1976

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: INTERIM REPORT ON ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Dear Mr. Anders:

At its 189th meeting, January 8-10, 1976, the Advisory Committee on Reactor Safeguards reviewed the Nuclear Regulatory Commission Staff's status reports on ATWS in water reactors and the analyses and proposals of four reactor vendors, The Babcock and Wilcox Company, Combustion Engineering, Inc., Westinghouse Electric Corporation, and General Electric Company, on this matter. Subcommittee meetings were held with representatives of the vendors, and with NRC Staff in Washington, DC, on December 11-12, 1975, and on January 7, 1976. The Committee had the benefit of the documents listed.

The Committee commented on the NRC Staff's proposal to revise the criteria for "Class A" plants, as categorized in WASH-1270 "Anticipated Transients Without Scram for Water Cooled Power Reactors," in a letter to Mr. Lee V. Gossick dated October 17, 1975. The Committee had previously commented on the regulatory position eventually published in WASH-1270, in letters to Mr. L. Manning Muntzing dated April 16, 1973, and May 10, 1972.

The ACRS endorses the general approach and safety objectives adopted by the NRC Staff including the use of a goal of 10^{-7} per reactor year as the maximum probability, from all causes, of ATWS with unacceptable conseguences.

Implicit in the use of a probabilistic goal is the application of probabilistic methods in the analysis of the reactor systems. Although WASH-1400 provides assistance in this area, data for some systems are still sufficiently sparse that engineering judgment must be used, both in synthesizing the analytical models and in choosing appropriate input data. Under the

circumstances there is a need for some conservatism in the choice of models and the selection of data. Even so, there are a number of approaches to the modelling that may prove to have equal validity and the ACRS suggests that both the NRC Staff and the vendors give further consideration to various alternatives. For example it may be feasible to treat the time-varying moderator temperature coefficient probabilistically.

The Committee also recommends that vendors be encouraged to continue to make design changes that decrease the probability of transients that are likely to cause difficulty and to make improvements that ameliorate the consequences. As appropriate these should be taken into account in the ATWS analysis. Continuing attention should be given to improving the reliability of the reactor shutdown systems.

During the course of these meetings comments were made which indicated that in some cases the NRC Staff needed further information from vendors in order to conclude its review of ATWS. The Committee urges that appropriate action be taken to obtain this information as soon as feasible.

The ACRS expects to complete its review of ATWS after further information has been developed and the Staff has completed its evaluation. The Committee urges that the NRC Staff and the vendors expedite efforts in this regard.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

REFERENCES:

1. Letter dated January 5, 1976, from Tobias W. T. Burnett, to T. G. McCreless concerning request for oral presentation to ACRS Working Group on ATWS on January 7, 1976 and to ACRS on January 8, 1976.

2. Draft Copy of Presentation to the ACRS entitled, "The ATWT Issues: The Industry Consensus As To Their Resolution," by T. W. T. Burnett, Chairman, ANS-51 ATWT Working Group.

3. NRC Staff's Status Report on Anticipated Transients Without Scram for Babcock and Wilcox Reactors, dated December 9, 1975.

4. NRC Staff's Status Report on Anticipated Transients Without Scram for Combustion Engineering, dated December 9, 1975.

5. NRC Staff's Status Report on Anticipated Transients Without Scram for General Electric, datd December 9, 1975.

6. NRC Staff's Status Report on Anticipated Transients Without Scram for Westinghouse, dated December 9, 1975..

7. Proposed American National Standard Evaluation of Anticipated Transients Without Trip on Pressurized Water Reactor Plants, N661, March 1975.

8. U.S. Atomic Energy Commission Regulatory Staff's Technical Report on Anticipated Transients Without Scram For Water-Cooled Power Reactors, WASH-1270, dated September, 1973.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 12, 1976

Lee V. Gossick
Executive Director
for Operations

Subject: RECIRCULATION PUMP TRIP TO HELP LIMIT THE CONSEQUENCES OF AN ATWS EVENT IN BWRs

It has recently come to the attention of the ACRS that not all operating boiling water reactors have installed a recirculation pump trip to help limit the consequences of an ATWS event. Such a trip was proposed by applicants for construction permits in 1971 and for operating licenses in 1972.

The Committee believes that the recirculation pump trip represents a substantial improvement in protection for BWRs and should be implemented promptly on all operating BWRs unless analyses have shown that such a trip is not required to mitigate the consequences of ATWS.

Please provide the Committee with a list of those reactors in operation, or nearing operation, which do not have such a trip and a schedule for installation of this trip on a timely basis.

Sincerely,

/s/ Dade W. Moeller

Dade W. Moeller
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 16, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON NRC STAFF REPORT ON ANTICIPATED TRANSIENTS WITHOUT SCRAM, NUREG-0460

Dear Dr. Ahearne:

At its 240th meeting, April 10-12, 1980 the Advisory Committee on Reactor Safeguards reviewed the Nuclear Regulatory Commission Staff Report, "Anticipated Transients Without Scram for Light Water Reactors" NUREG-0460, Volume 4. The Committee previously reviewed Volumes 1 and 2 at its 221st and 222nd meetings, September 7-9, 1978 and October 5-7, 1978, and reviewed Volume 3 at its 225th and 227th meetings January 4-6 and March 8-10, 1979. Subcommittee meetings were held to review Volumes 1-4 of NUREG-0460 on April 20, May 26, July 13, and August 1-2, 1978, January 31, and March 2, 1979, and January 25 and March 26, 1980. During these meetings the Advisory Committee on Reactor Safeguards discussed the report with representatives of the NRC Staff, Atomic Industrial Forum, Electric Power Research Institute, reactor vendors and some utilities. The Committee had the benefit of the documents listed.

Early Operating Plants

The ACRS agrees that the early operating plants (the list is given on page 53 of NUREG-0460, Volume 4) should comply with Alternate 2A as described in NUREG-0460, Volume 4. However it is recommended that discretion be used in implementing this requirement on these older plants to attempt to assure that those changes made in equipment or circuitry decrease rather than increase risk.

Plants Now Operating and Those Under Construction

For operating BWRs, and for those in advanced stages of construction, it is recommended that Alternate 3A be implemented. However, consideration should be given to the possibility that risk might be decreased by treating some plants, because of equipment arrangement, construction schedule, or other relevant considerations, as special cases.

For operating Westinghouse PWRs, and those under construction, it is recommended that the ATWS mitigating system circuitry as described in Alternate 3 of Volume 3 of NUREG-0460 be implemented. In consideration of the pressure relieving capability of the existing Westinghouse systems, and the possibilities for installation errors and unexpected systems interactions that can occur in backfitting, the Committee does not recommend changes in the scram system circuitry or in the scram breakers for these plants.

For B&W and for CE reactors now operating, and for those under construction, it is recommended that Alternate 3A, as described in NUREG-0460, Volume 4 be implemented. The information now available to the Staff indicates that calculated primary system pressures for B&W and CE plants exceed Service Level C stress limits. Because of these calculated pressures, the Staff is recommending installation (under Alternate 4A) of additional safety or relief valve capacity. The ACRS agrees that additional steps are needed if further investigation corroborates these calculations. However, before making a final decision on requiring additional valves, the Committee recommends that full consideration be given to mitigation capability inherent in possible adjustment of, better measurement of, or better calculation of the moderator temperature coefficient. There appear, for example, to be differences between the methods used for calculation of the peak pressures by the different vendors. These should be investigated to assure that appropriate assumptions and methods are being used. Consideration should also be given to any additional mitigating capability that may exist because of control rod insertion that may be initiated independently of the scram system logic.

The Committee recognizes the desirability of an implementation schedule which produces an expeditious resolution of the ATWS issue. However, the Committee recommends that whatever schedule is adopted permit sufficient time for the design, procurement, and installation of the necessary systems. Because of the large number of plant changes being required by TMI, the schedule being proposed by the Staff may be unrealistic or even counterproductive. It is important that the changes made contribute to risk reduction rather than risk enhancement.

New PWR Plants

For plants on which construction has not begun, the ACRS recommends that additional pressure relieving capability be added to CE and B&W plants. For Westinghouse reactors not yet under construction, the Committee recommends that both the NRC Staff and the vendors give consideration to changes in the scram system that might increase its reliability.

New BWR Plants and Those in Early Stages of Construction

For new GE plants and those in early stages of construction, it is recommended that increased reliability be incorporated in systems which can deliver water

to the vessel under the operating pressure expected after an ATWS. It is also recommended that further analyses of possible oscillatory behavior of reactor power level after failure to scram be made, and a larger capacity standby liquid control system (SLCS) be required if it is needed to reduce the oscillation to an acceptable level. The Committee also recommends that the SLCS capacity be evaluated giving consideration to the possibility of boron wash-out due to the addition of make-up water to the primary system during an ATWS event.

Plants at Sites With Higher Population Density

The ACRS believes that for those reactors at sites with higher population densities, additional considerations are appropriate. The Committee recommends that mitigation features be considered for Oyster Creek if study shows they are practical and may provide a significant reduction in risk. Limerick should be provided with a boron injection system having the reliability and reactivity reduction capability of Alternate 4A. Midland should, if practical, be limited to the Service Level C stress limits during an ATWS. Possible improvements in scram reliability should receive consideration as part of the ongoing review of risk reduction at Indian Point and Zion.

Schedule

The ACRS recommends that, in the interest of enhanced safety and efficient use of resources, the changes required by the NRC in connection with treating the ATWS issue be considered in the context of changes required because of the accident at TMI-2. For example, integration of the schedule for equipment installation required by the two programs may be feasible. It is also suggested that requirements imposed by ATWS that deal with specific subsystems (e.g. the auxiliary feedwater system and the requirement for containment isolation) be made compatible with those required because of TMI-2.

Mr. Jeremiah J. Ray did not take part in Committee deliberations regarding this matter.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Anticipated Transients Without Scram for Light Water Reactors", USNRC Report NUREG-0460, Volumes 1 and 2, April 1978, Volume 3, December 1979, and Volume 4, March 1980.
2. Letter, J. B. Randazza, Maine Yankee Atomic Power Company, to M. Plesset, ACRS, Subject: Anticipated Transients Without Scram (ATWS), dated April 8, 1980.
3. Letter, L. H. Heider, Vermont Yankee Nuclear Power Corporation, to M. Plesset, ACRS, Subject: Anticipated Transients Without Scram (ATWS), dated April 7, 1980.
4. Letter, F. Schwoerer, Westinghouse ATWS Owner's Group, to W. Kerr, ACRS, Subject: Comments on NRC NUREG-0460, Volume 4 Requirements, dated April 7, 1980.
5. Letter, Wayne H. Jens, Detroit Edison to W. Kerr, ACRS, Subject: Detroit Edison Position on ATWS, dated April 3, 1980.
6. Letter, D. L. Peoples, Commonwealth Edison Company, to W. Kerr, ACRS, Subject: Comments on ATWS, dated March 25, 1980.
7. Letter, J. Taylor, Babcock and Wilcox, to S. Hanauer, NRC, Subject: Transmittal of Addendum to B&W Topical Report, "Analysis of B&W NSS Response to ATWS Events", BAW-1610, dated February 26, 1980.
8. Babcock and Wilcox, "Analysis of B&W NSS Response to ATWS Events" BAW-1610, January 1980.
9. General Electric Company, "Assessments of BWR Mitigation of ATWS, Volume II (NUREG-0460 Alternate Number 3)", NEDE-24222, December 1979 (Proprietary).
10. General Electric Company, "Assessment of BWR/3 Mitigation of ATWS (Alternate 3)", NEDE-24223, December 1979 (Proprietary).
11. Westinghouse Electric Company report "Anticipated Transients Without Scram for Westinghouse Plants", December 1979.
12. Letter, R. H. Buchholz, General Electric Company, to S. Hanauer, NRC, Subject: BWR Capability to Accommodate ATWS, dated December 14, 1979.
13. Combustion Engineering Inc., "ATWS Early Verification-Response to NRC Letter of February 15, 1979 for Combustion Engineering NSSS's" CENPD-263-NP, November 1979.
14. Letter, R. J. Mattson, NRC to all NSSS Vendors, concerning generic ATWS analysis to meet requirements of NUREG-0460, Volume 3, dated February 15, 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 13, 1981

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: RESPONSE TO INQUIRIES CONCERNING THE BROWNS FERRY NUCLEAR POWER STATION UNIT 3 PARTIAL-FAILURE-TO-SCRAM

Dear Dr. Hendrie:

In response to Commissioner Ahearne's letter of December 12, 1980, the Advisory Committee on Reactor Safeguards has reviewed the issues contained in Congressman Morris K. Udall's inquiries which were prompted by the June 28, 1980 Browns Ferry Unit 3 partial-failure-to-scram. There appear to be four specific concerns expressed by Congressman Udall which include:

(1) the level of confidence placed in the Staff's ability to calculate the consequences of an anticipated transient without scram (ATWS);

(2) the level of confidence in the adequacy of actions taken subsequent to the Browns Ferry Unit 3 partial-failure-to-scram;

(3) the extent to which emergency procedures at operating plants contain instructions for the operator, given an ATWS; and

(4) an assessment of the causes of the Browns Ferry Unit 3 partial-failure-to-scram.

These matters, as related primarily to BWRs, were discussed during Subcommittee meetings with the NRC Staff held on March 10 and April 8, 1981 and during the 253rd ACRS meeting, May 7-9, 1981.

The Staff, through its contractor, Brookhaven National Laboratory (BNL), has the ability to calculate consequences of a full ATWS event. BNL's current capability to calculate consequences of a partial ATWS (where only a fraction of the control rods fail to insert) is not well developed. However, it is our belief that the consequences of a partial ATWS are bounded by those of a full ATWS. New, more sophisticated codes are being developed which should more accurately model both full and partial ATWS consequences. It is expected these codes will be completed by 1982.

Regarding the adequacy of actions taken subsequent to the Browns Ferry Unit 3 partial-failure-to-scram, we believe that the NRC Staff and the industry responded in a timely manner. The initial Bulletin resulting from this event (Bulletin 80-17) was issued on July 3, 1980 and was aimed at preventing the accumulation of water in the scram discharge volume (SDV), which was the probable cause of the Browns Ferry Unit 3 partial-failure-to-scram. We also take note of the ongoing effort by the NRC Staff to continue the review of BWR scram systems.

As a result of the Browns Ferry Unit 3 partial-failure-to-scram, the Commission directed the Office of Inspection and Enforcement to conduct a survey to determine the existence of adequate emergency procedures for coping with ATWS events at operating power reactors. Bulletins issued following the Browns Ferry Unit 3 partial-failure-to-scram required checks to assure adequate ATWS procedures were in place at BWRs. The results of the survey made after the Bulletins were issued, indicated that all utilities which were operating BWRs then had procedures for coping with ATWS which were acceptable to the NRC. In addition, steps have since been taken to ensure that PWRs as well as BWRs now have ATWS procedures acceptable to the NRC. It is expected that further improvements will be made in ATWS procedures as in-depth examination of the matter continues. We note that these procedures, of themselves, do not substitute adequately for design improvements to mitigate and reduce the probability of an ATWS.

Although we believe that the Browns Ferry Unit 3 incident was caused by water in the SDV, a complete explanation as to why the water was present is not available. The most plausible explanation is that some type of flow resistance existed in the drainage system for the SDV, which prevented adequate drainage. The cause of the failure to detect this condition was poor hydraulic coupling between the scram discharge volume and the instrumented volume. This design deficiency is common to a number of the older BWRs. Later designs of the BWR have an improved hydraulic coupling which reduces this concern. Among the long-term fixes for this problem will be a combination of the two volumes as in the newer system design. It is unclear why the scram discharge system design was changed without older plants having been subjected to a reexamination. Some mechanism should be formulated or a better system developed to ensure design improvements are reviewed for applicability to existing plants.

It is our opinion that the Commission's previous responses to Congressman Udall reflected the best information available at the time those responses were prepared and that those responses still represent a reasonable assessment of the situation.

We recommend that instrumentation needed to accomplish safe shutdown of the plant following an ATWS event be qualified to perform acceptably following the high pressure or other environmental effects to which the instrumentation might be exposed during the event. We also recommend that the possible effects of high pressure on the integrity of primary system pump seals and the pressure boundary of instruments or other components not covered by the ASME Boiler and Pressure Vessel Code be explored, and that the NRC Staff assure itself that pump seal or other primary system coolant leakage caused by an ATWS does not significantly increase the likelihood of severe damage to the core.

To further decrease the probability of an ATWS, we recommend that efforts be made by plant operators to decrease the number of transients that challenge the reactor protection system. For example, operating experience has shown that a large portion of plant transients results from malfunctions of the feedwater system. Recent information also indicates that the number of reactor scrams per plant is significantly higher in the U.S. than in certain foreign countries (Japan for example). We recommend an investigation to determine if practical means exist to reduce challenges to the reactor trip system.

We are concerned about the potential for confusion when part of the reactor trip system is safety grade and part is not. Neither the diverse shutdown logic systems required by the Rule nor the heat tracing for boron solution storage systems are safety grade. The same is true for the alarm systems that indicate whether the heat tracers are operative. These matters may warrant further consideration.

The NRC Staff has suggested that an ATWS contribution to core melt frequency of 10^{-5}/reactor year or less would be acceptable. Since the risk to the public from such an accident may vary markedly between PWRs and BWRs, it is not clear that a single core melt probability criterion should be applied uniformly to all LWRs. If a major release of radioactive material from the containment is likely, the criterion of 10^{-5}/reactor year is, at best, marginal for any single cause, such as ATWS, and further reductions in likelihood should be considered. The NRC Staff's estimates of ATWS likelihood are best estimates, with considerable uncertainty, and thus may not reflect a reasonable degree of assurance that the actual risk is as low as it should be. We recommend that considerations of such risk be included, particularly for reactors having high surrounding population densities.

We recommend that designers of new plants give consideration to designs that will reduce reactor trip system challenges as well as enhance reliability. We note that while the Rule requires new BWRs to enhance the capability of the SLCS, it is silent concerning the need for additional mitigating features for PWRs. For example, there is no requirement for

additional pressure relieving capability for B&W and CE plants. We disagree with this position and recommend that the NRC require licensees to explore means for enhancing the ATWS mitigative capabilities for new PWR plants.

We support the provision that the schedule for implementation be one negotiated between the licensee and the NRC, but caution that care be exercised to assure the timeliness of the changes required.

We believe that the proposed Rule is an appropriate approach to dealing with the ATWS issue, but recommend that consideration be given to the above comments before a Rule is issued in final form.

The Committee plans a later review of the proposed amendment dealing with the Westinghouse trip system.

Sincerely,

J. J. Ray
Chairman

References:

1. Memorandum from M. L. Ernst, NRC, to R. F. Fraley, ACRS, "Amendments to 10 CFR Part 50 Related to Anticipated Transients Without Scram (ATWS)," dated June 27, 1983.

2. U.S. Nuclear Regulatory Commission, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," USNRC Report, NUREG-1000, Volume 1, dated April 1983.

3. U.S. Nuclear Regulatory Commission, "Generic Actions for Licensees and Staff in Response to the ATWS Events at Salem Unit 1," USNRC Report, SECY-83-248, dated June 22, 1983.

4. Letter from E. P. Rahe, Jr., Westinghouse Electric Corporation, to H. R. Denton, NRC, Subject: Westinghouse Comments on Proposed Amendment to ATWS Rule, dated June 22, 1983.

Although the actions taken by the NRC were timely as a direct response to the Browns Ferry Unit 3 incident, we believe that this incident should have inspired a very considerable acceleration of the Commission's efforts to resolve the ATWS issue. This has not been the case. We recommend strongly that the Commission act promptly and with high priority to resolve ATWS. In this regard, we reiterate the recommendations made in our letter of April 16, 1980 on ATWS.

Sincerely,

J. Carson Mark
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 15, 1983

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: ACRS REPORT ON THE PROPOSED FINAL RULE ON ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Dear Dr. Palladino:

During its 279th meeting, July 7-9, 1983, the Advisory Committee on Reactor Safeguards reviewed the proposed Rule on ATWS. A Subcommittee meeting was held with members of the NRC Staff on May 27, 1983. The Committee also had the benefit of the documents listed.

For BWR plants, the Rule requires installation of an alternate rod injection (ARI) system and a standby liquid control system (SLCS) of a specified control capacity. The Rule also requires that the SLCS must be capable of automatic initiation for new plants and for other BWR plants that already have the capability to automate this system. Automatic trip of the recirculation pumps is also specified, although we understand that this feature is already operational on all plants through Commission Order.

PWR plants are required to have equipment diverse from the reactor protection system installed to assure automatic initiation of auxiliary feedwater and of turbine trip. In addition, reactors manufactured by the Babcock & Wilcox Company (B&W) and Combustion Engineering, Incorporated (CE) are required to install a diverse scram actuation system. While not in the current version of the Rule, an amendment will be proposed to require Westinghouse reactors to install a diverse scram actuation system as well.

The NRC Staff is recommending that licensees establish a reliability assurance program for their plants' reactor trip systems. We support efforts to enhance reliability assurance for such critical plant systems; however, we believe a comprehensive program that encompasses all plant safety-related systems would be more effective.

B

Babcock and Wilcox Reactors

Babcock-205 Standard Nuclear Steam System

B

Babcock and Wilcox Reactors

Babcock-205 Standard Nuclear Steam System

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 15, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: BABCOCK AND WILCOX NUCLEAR PLANTS UNDER CONSTRUCTION

Dear Dr. Ahearne:

During its 240th meeting, April 10-12, 1980, the ACRS reviewed Mr. H. Denton's proposal, conveyed in his memorandum of January 22, 1980, not to halt construction of nuclear plants currently under construction which use Babcock & Wilcox nuclear steam supply systems. The plants in question are Consumers Power Company's Midland Units 1 and 2, Tennessee Valley Authority's Bellefonte Units 1 and 2, and Washington Public Power Supply System's (WPPSS) WNP Units 1 and 4. The ACRS had the benefit of discussions with the NRC Staff, and representatives of the Babcock and Wilcox Company and the Utilities. A subcommittee reviewed this matter on April 8, 1980.

In view of information supplied by the utilities on the status of construction, design improvements already undertaken, and the commitment of WPPSS to make further studies, the ACRS agrees that it is not necessary to halt construction of these plants. The ACRS understands that the possibility of imposing additional requirements will not be precluded by the decison to continue construction.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

Reference:
Memorandum, H. Denton to Commissioners, "Determination Whether B&W Designed Plants Presently Under Construction Be Allowed to Continue," January 22, 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 10, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON NUREG-0667, "TRANSIENT RESPONSE OF BABCOCK & WILCOX-DESIGNED REACTORS"

Dear Dr. Ahearne:

During its 242nd meeting, June 5-7, 1980, the ACRS completed its review of NUREG-0667. The Committee had the benefit of discussions with the NRC Staff and representatives of the nuclear industry and had received briefings from the NRC Staff during the 240th and 241st meetings of the Committee (April 10-12 and May 1-3, 1980). A Subcommittee meeting on the subject was held on April 29, 1980.

The recommendations of NUREG-0667 are consistent with conclusions reached by the NRC Staff in previous investigations and addressed in Draft 3, NUREG-0660, "NRC Action Plans Developed As a Result of the TMI-2 Accident." The ACRS has commented previously on the Action Plan.

The ACRS agrees in general with the recommendations of NUREG-0667 and believes their implementation will enhance the safety of Babcock & Wilcox (B&W)-designed reactors. The Committee has, however, several comments as follows:

1. The ACRS is concerned that the recommendations in this report show a continuation of a trend by the NRC Staff toward less and less flexibility in the resolution of safety issues. We believe that this trend is evidenced by what we have perceived as increasingly frequent recommendation of prescriptive solutions being offered and insisted on by the NRC Staff. The Committee recognizes that prescriptiveness is sometimes a consequence of lack of initiative on the part of the industry but believes that efforts should be made to encourage the industry to find alternative and potentially better solutions to safety problems.

2. The Committee recognizes the need to develop "fixes" as system deficiencies become apparent, but suggests that the preoccupation with specific problems related to B&W reactor designs not be permitted to detract unduly from attention to generic problems or to matters applicable to other types of light water reactors. For example, development of a seismically qualified dedicated shutdown heat removal system is a project which should be undertaken with a high priority.

3. In its section on "Instrumentation and Control", the report recommends that power buses and signal paths for nonnuclear instrumentation and associated control systems be separated and channelized to reduce the impact of failure of one bus. The Rancho Seco and Crystal River events demonstrated clearly that the operator can be deprived of critically necessary information in the event of such a failure. However, vulnerability to such loss may not be restricted to B&W nuclear steam supply systems.

   Detailed examination will reveal that many plants of even recent design have not included indicating, recording, or other operator information circuitry in the group for which separation and independence are required. We believe that reliable information for the operator is necessary, and more separation may be required. It would seem timely to review possible application of Regulatory Guide 1.75 to these circuits.

   Redundant indication and recording, if it means only two trains, leaves the problem of obtaining a proper response when contradictory information is provided. This matter requires further study.

   We suggest that the question of off-scale indications by instruments during upset conditions be addressed by the licensees to assure that information vital to the control of the course of an accident not be lost.

4. In its section on "General Areas For Improvement", the report recommends continued study of the criteria for tripping reactor coolant pumps during small-break LOCAs. In many plants tripping the pumps causes loss of the pressurizer spray function, the principal means for reducing system pressure. The study should include evaluation of the benefits of a separate source of spray water to control mild overpressure transients with pumps tripped.

With the reservations and suggestions mentioned above, the ACRS believes that the recommendations of NUREG-0667 should be implemented on a schedule which allows sufficient time for orderly design, installation, and testing of the suggested changes.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 18, 1977

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON BABCOCK-205 STANDARD NUCLEAR STEAM SYSTEM

Dear Dr. Hendrie:

At its 208th Meeting, August 11-13, 1977, the Advisory Committee on Reactor Safeguards completed its review of the application of Babcock & Wilcox Company (Applicant) for a Preliminary Design Approval (PDA) for its Standard Nuclear Steam System (NSS), Babcock-205. A Subcommittee meeting was held with representatives of Babcock & Wilcox Company (B&W) and the Nuclear Regulatory Commission (NRC) Staff in Washington, D.C., on July 27, 1977. The Committee had the benefit of discussions with representatives of the NRC Staff and the Applicant. The Committee also had the benefit of earlier reviews of other Standardized Nuclear Steam Supply Systems and of the documents listed.

The B&W Standard Safety Analysis Report (B-SAR-205) describes the B&W standardized two loop NSS for a 3820 MWt pressurized water nuclear plant with a core thermal power output of 3800 MWt. Its scope includes the Reactor, Reactor Coolant System, Pressurizer Relief System, Emergency Core Cooling System, Decay Heat Removal System, Chemical Addition and Boron Recovery System, Makeup and Purification System, Instrumentation and Controls, and Fuel Handling System. The reference design is similar to recent B&W 3620 MWt NSS designs reported on by the Committee: Bellefonte Nuclear Plant Units 1 and 2 (July 16, 1974), Washington Public Power Supply System Nuclear Power Stations 1 and 4 (June 11, 1975) and Pebble Springs Nuclear Plants 1 and 2 (February 11, 1976). Earlier ACRS reports on PDA applications for standard reference nuclear steam supply designs are the March 14, 1975 report on the General Electric Company GESSAR-238 Nuclear Island and the December 17, 1976 report on the General Electric Company GESSAR-238 and GESSAR-251 Nuclear Steam Supply Systems, the September 17, 1975 report on Combustion Engineering, Inc., CESSAR-80, and the September 18, 1975 and July 14, 1976 reports on Westinghouse Electric Corporation RESAR-41 and RESAR-3S.

The Applicant has selected horizontal ground accelerations of 0.3g and 0.15g as appropriate design values for the safe shutdown and operating basis earthquakes, respectively. Designs for the protection of Babcock-205 safety-related components and systems from other site-related hazards will be included in the balance of plant design. The reference NSS design includes the interface requirements information essential to a balance of plant safety design consistent with the assumptions used in the NSS design-basis accident analyses. Since the utility-applicant will be responsible for instituting the quality assurance programs necessary to assure that all safety-related design requirements have been met, the Committee will review these matters in more detail with the utility-applicants on a case-by-case basis.

The NRC Staff had listed thirteen outstanding issues in its report to the ACRS. The NRC Staff now believes that it has the information it needs to resolve eight of the thirteen issues, and that the remaining five issues can be resolved before the anticipated time of granting the PDA. The ACRS recommends that all outstanding issues be resolved in a manner satisfactory to the NRC Staff.

The Committee believes that for Babcock-205, as well as for other standardized nuclear steam supply systems, studies should be made to quantify the reliability, effectiveness and safety margins for the Emergency Core Cooling Systems. The studies should include evaluation of current NRC and EPRI programs on alternate ECC Systems to determine what, if any, improvements are practicable and desirable. These efforts will require the substantiation of the bases for best estimate analyses and verification of codes, and increased input to and participation in the related NRC programs for safety research.

New features in the instrument and control systems have been introduced through the specification of functional designs and design criteria which the NRC Staff has found to be adequate for the PDA. On all issues involving instrumentation and control, the Committee will use the case-by-case basis to ascertain progress of the work until the Babcock-205 design has progressed to the stage where Final Design Approval is achieved.

The Committee has encouraged Applicants to consider instrumentation with the necessary ranges and diversity to follow the course of an accident. Regulatory Guide 1.97 (Revision 1) embodies these recommendations. For standardized PWR's, such as Babcock-205, the Committee recommends that a study be made of the merits of including instrumentation to sense the water level in the reactor pressure vessel.

The Babcock-205 design includes provisions which anticipate maintenance, inspection, and operational needs of the plant throughout its service life. The Committee believes that the NRC Staff and the Applicant should continue to seek improvements in provisions of this type and, in addition should review procedures for removing accumulations of radioactive contamination so that maintenance and inspection programs and eventual decommissioning can be more effectively and safely carried out.

The Committee believes that B&W and the NRC Staff should intensify their review for design provisions that will further improve protection against sabotage and that provisions found timely should be incorporated in early applications of Babcock-205.

With regard to other generic problems cited in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 5," dated February 24, 1977, items considered relevant to Babcock-205 are: II-3, 4, 5, 6, 7, 9, 10; IIA-3, 4, 5, 6, 7; IIB-1, 2; IIC-1, 4, 5; IID-1, 2. These problems should be dealt with by the Staff and the Applicant as solutions are found.

The Committee believes that methods that seek to develop reference systems through standardization and through replication need to be coupled with ongoing programs that will permit design changes which improve safety and which, when justified, will be implemented in a timely manner. Use of reference systems should lead to more efficient and effective licensing reviews. Standardized designs such as Babcock-205 will contribute to this process. A transition period will be required in which the Committee will still give attention to the items noted, on a case-by-case basis.

The Committee believes that, subject to the above comments and successful completion of the R&D programs, the Babcock-205 design can be successfully engineered to serve as a reference system.

Sincerely yours,

M. Bender
Chairman

REFERENCES:

1. The B&W Standard Safety Analysis Report (B-SAR-205) and Amendments 1 through 16.
2. Report to the Advisory Committee on Reactor Safeguards by the Office of Nuclear Reactor Regulation, (ONRR), U. S. Nuclear Regulatory Commission in the Matter of Babcock and Wilcox Company Reference Safety Analysis Report, B-SAR-205, Docket No. STN 50-561, July 8, 1977.
3. Babcock and Wilcox letter to ONRR, Attention: Mr. Roger S. Boyd, Subject: B-SAR-205 - Outstanding Issues, dated July 15, 1977, Docket STN 50-561.
4. Babcock and Wilcox letter to ONRR, Attention: Mr. Roger S. Boyd, Subject: B-SAR-205 - Outstanding Issues, dated July 21, 1977, Docket STN 50-561.

C

Class 9 Accidents

Combustion Engineering Standard Safety Analysis Report (CESSAR-80)

Control Rods and Drives

Criteria

C

Class 9 Accidents

Combustion Engineering Standard Safety Analysis Report (CESSAR-80)

Control Rods and Drives

Criteria

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 7, 1980

The Honorable Victor Gilinsky
Commissioner
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Gilinsky:

Your memorandum dated March 25, 1980 asks for the Committee's thoughts on the feasibility and practicality of a containment concept which could withstand a core melt. In this letter the ACRS will provide some preliminary and necessarily incomplete comments on the subject.

1. The letter dated March 7, 1980 from Albert L. Latter, President, R&D Associates, to the Nuclear Regulatory Commission does not provide a technical description adequate to determine or evaluate the approach being proposed. Limited additional information was provided by R&D Associates at their meeting of April 25, 1980 with you and Chairman Ahearne. However, in the absence of considerably more information, the ACRS is unable to comment on the technical merit of this proposal.

2. The general question of the feasibility and practicability of a containment which could withstand a core melt should be examined within the context of some broad policy guidance. What cost is justifiable? Is 100% guarantee of containment integrity being sought? If not, what frequency of an uncontrolled airborne release of a large portion of the radioactive fission product inventory is acceptable? What frequency of a sudden gross release of the noble gas inventory is acceptable? What frequency of a penetration of the core through the containment foundation (or equivalent release to the ground water) is acceptable? Do the reactor site characteristics and the nation's energy supply situation bear on the definition of acceptable frequency?

   Prior to 1966, although the regulatory process did not include consideration of the probably strong correlation between core melt and containment failure, it was clearly recognized that paths existed for a loss of containment integrity concurrent with a core melt. These included the potential for large missiles arising from sudden failure of the reactor pressure vessel or other large components, as well as the possible failure of containment isolation mechanisms. Furthermore, natural events such as earthquakes and floods were known to provide a potential for a loss of containment concurrent with an accident which seriously damaged the core.

What developed in 1966 was a better appreciation that, for the medium power LWRs then under construction, containment failure was likely to be associated with core melt from whatever cause. As you know, in September, 1966 the ACRS was dissuaded by the AEC from transmitting a letter which would have recommended the development and implementation for LWRs of measures to cope with and mitigate serious accidents, and accepted as an alternate the establishment of a task force which was supposed to develop within a few months a recommended approach to pursue the core melt problem. The ACRS recommended in October, 1966 that the AEC initiate a vigorous, high priority safety research program on phenomena related to core meltdown and on design concepts to mitigate such an accident. The Committee has reiterated that recommendation many times in the ensuing years to both the AEC and the NRC with little success until the past several months.

For example, in a letter dated January 11, 1971 from R. F. Fraley to Milton Shaw, Director of the AEC Division of Reactor Development and Technology, the ACRS stated its belief that a core retention system could provide a substantial reduction in the probability of a fission product release to the environment. In addition, the ACRS noted some then-recent studies which offered encouraging results and recommended initiation of meaningful conceptual design work.

3. The response to your question depends heavily on what safety policy the NRC decides to adopt.

   (a) If the NRC policy were to become one which requires that there is no acceptable frequency for an accident involving both core melt and a loss of containment integrity, even very low power LWRs built underground would not satisfy this policy, since one can postulate scenarios, for example, involving terrorism, very large earthquakes, or a failure in an access path from containment, which could defeat any design, in principle.

   (b) If the NRC policy were to become one which requires that the design should have a very high probability of containing all core melt accidents, including those involving large internally generated missiles and should limit the maximum extent of airborne and liquid pathway release to man, some form of underground or earth-covered reactor plant may be required, with special attention given to earthquakes, floods, and groundwater conditions, and to the design of features having a high probability of successfully retaining a molten core and of retaining most of the radioactive material in case controlled containment pressure relief were called for.

(c) If the NRC policy were to become one which requires that the design should have a high probability of containing almost all core melt accidents which do not concurrently include a loss of containment integrity due to missiles, sabotage, very large earthquakes or similar postulated causes, the issue can probably be satisfactorily addresssed in terms of LWRs constructed above ground. There would still remain a policy question concerning the acceptability of design approaches which envisage the potential for deliberate venting of noble gases to prevent containment overpressure in the event of a serious accident.

(d) The NRC policy might involve a limited or selected set of additional requirements for mitigation, for example, a filtered venting system for containment to reduce the probability of containment failure due to overpressure, but no core retention within containment, relying instead upon acceptable hydrological conditions. Or, of course, the NRC policy might involve no additional accident mitigation requirements, placing all emphasis on prevention of serious accidents beyond the current design basis.

4. Policy alternative 3(c) above is examined briefly in this section for technical feasibility. The technical questions would, of course, be more complex if accidents such as pressure vessel rupture were also to be addressed per policy alternative 3(b).

A containment designed to withstand a core melt must consider both the problem of molten core retention and cooling, and the prevention or limitation of a significant atmospheric release of radioactive material. If one establishes a reasonable reliability goal for measures intended to protect against both the atmospheric and liquid release pathways, and the potential need for pressure relief of the containment with a resultant release of radioactive noble gases through a filtered, venting system, it appears that, for the large volume, high-design-pressure type of containment, core melt retention is probably feasible. Its practicality will depend on the cost, the reliability goals sought, and the benefits assigned to the accomplishment.

The attached, recently issued documents by Messrs. I. Catton, C. Kelber, A. Marchese and T. Speis, R. DiSalvo, and A. Benjamin and H. Walling provide some current thinking on the problems, prospects and issues involved in molten core retention. These documents have not yet been reviewed by the ACRS; an ACRS subcommittee meeting on the general subject is scheduled to be held this month.

It is clear that problems remain, for example, from the potential for an excessively large, sudden pressure rise within containment due to hydrogen deflagration or a very rapid thermal energy exchange between the molten fuel and water. Measures appear to be possible to cope with these and similar questions, but comprehensive and sufficiently detailed studies to evaluate and choose from various design approaches are not available.

Recent work suggests that steam explosions are not a likely source of a loss of containment integrity in a large, high-design-pressure containment. However, steam explosions may be locally disruptive to features intended to help retain a molten core. The decision as to whether it would be better to keep the region below the vessel dry or flooded with water cannot be readily made with the limited design and risk evaluation information now available.

The ACRS believes that, given reasonable reliability goals, goals which do not pose so unrealistic a demand that they cannot be confirmed either experimentally or theoretically, it should be technically feasible to design an LWR containment to withstand a core melt.

The Committee wishes to note that whatever policy the NRC adopts, thought has to be given to the approach which would be taken to recover the site and dispose of a nuclear plant which had been subjected to a core melt accident.

5. The ACRS believes that the issue of what protection is to be required to contain or mitigate accidents involving core melt in LWRs yet to be designed and constructed should be decided expeditiously. The NRC Staff has recently augmented its previously modest research efforts related to this issue. However, in view of the essential nature of this issue to any decision process regarding the design of future LWRs, the ACRS believes that the current efforts by the Staff are inadequate.

   If the NRC concludes that future LWRs will require protection to contain or mitigate serious accidents involving core melt, the Committee believes that the DOE should be requested to undertake, as soon as possible, the necessary research and development work. The DOE effort should be adequately funded from the very start in order to develop an effective and reliable protection system within a time frame that will not delay the design of future plants which would incorporate this system.

6. The ACRS believes that resolution of this general issue should be given high priority by the Commissioners themselves. The Committee believes that such a policy decision should be part of an overall NRC safety philosophy. The safety philosophy should also provide siting guidance for future reactors and reliability goals for design measures intended to prevent core melt accidents. It should also provide risk-based guidance to both the Staff and industry for the wide spectrum of possible accidents.

The ACRS believes that such a policy decision by the Commission should be made with recognition of the comparative risks from other energy sources and from other technologies, and in the light of the societal, economic, and political factors which bear significantly on the complex issues involved.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

Attachments:
1. I. Catton, ACRS consultant, memo to D. Okrent, ACRS, dtd. 4/25/80 re. Breach of Containment by a Core Melt

2. C. Kelber, RSR, memo to G. Quittschreiber, ACRS, dtd. 4/22/80 re. Input to Response to Commissioner Gilinsky's Questions on Core Melt

3. A. Marchese and T. Speis, NRR, memo to D. Okrent, ACRS, dtd. 4/25/80 re. "General Feelings on Containing a Core Melt"

4. R. DiSalvo, RSR, memo to G. Quittschreiber, ACRS, dtd. 4/24/80 re. Request for Input to Commissioner Gilinsky's Questions on Core Melt

5. A. Benjamin and H. Walling "Development and Analysis of Vent-Filtered Containment Conceptual Designs," SAND80-0887

cc:
Chairman Ahearne
Commissioner Kennedy
Commissioner Hendrie
Commissioner Bradford

Apı 25, 1980

TO: D. Okrent

OM: I. Catton

SUBJECT: Breach of Containment by a Core Melt

REFERENCE: Letter from Ivan Catton to David Okrent dated 6 March 1980

The question posed is whether or not it is feasible and practical to design a containment that can withstand a core melt. It is my opinion that to do so is both feasible and practical. Of course there will be a number or hurdles to overcome in arriving at a design. I will attempt to substantiate my opinion in the following paragraphs by first addressing existing plants and then give my ideas about new plants.

Before discussing LWRs, however, I would like to call your attention to previous work in this area for LMFBR's. Designs for core catchers were proposed for FFTF and CRBR. A number of crucible materials were evaluated and both passive (time delay) and active systems were considered. The German reactor SNR 300 will have an actively cooled crucible using depleted UO_2 as a sacrificial material. Several ideas for core retention have come out of efforts of the GE advanced reactor group. A firm in Germany was found that would make standard size bricks out of depleted UO_2 for what I remember to be a reasonable cost. The depleted UO_2 was needed to absorb the thermal shock from the melt and protect the active cooling system. The designs were not fully evaluated but had potential for being successful. When one considers that the fuel melt from an LMFBR has an energy density that is an order of magnitude greater than an LWR one sees that the design of a core catcher for a LWR will be less difficult.

A number of aspects of a core melt accident were discussed in the above referenced letter, which dealt with Indian Point and Zion. They are repeated here in part.

1. Steam explosions will probably not occur in-vessel if the pressure is above 7-10 bars. Even if a steam explosion were to occur in-vessel, recent SANDIA work shows that there is little chance of a missle that could penetrate the containment. The only missile that might be of concern was the control rod drive. Some plants have missile shields for this already and plants without could install one. An ex-vessel steam explosion will only occur if water is in the reactor cavity before the vessel is penetrated or enters shortly thereafter (before the molten pool solidifies and while gas is still being generated by concrete decomposition). The ex-vessel steam explosion will probably not do much damage and it appears that acceleration of missiles that will penetrate the containment is unlikely. Further confirmation of this opinion is needed to assure that damaging the shield wall, moving the vessel or some other aspect will not lead to containment penetration. High steam generation rate will occur if water precedes the melt and the resultant high steam generation rate needs to be a factor considered in seeking mitigation measures.

ATTACHMENT 1

2. In-vessel core coo _oility is presently not well e _gh understood to fully describe the core meltdown process. Programs presently underway in Germany and the US may yield sufficient information at some time in the future to describe the process. At this time one can only bound the problem and must assume that penetration of the vessel occurs early in the worst way. It should mentioned that it is not really clear what the worst way is. For example t of fuel resulting from a hole in the bottom of the vessel might erode a hole in the base mat with subsequent erosion of the hole being greater than if the entire vessel lower dome failed dumping all the molten fuel at one time.

3. Ex-vessel core debris coolability will depend strongly on whether or not water is in the cavity. If water is in the cavity in sufficient quantity before the vessel is penetrated, the core debris will be quenched as it enters. A sufficient quantity of water is a pool deep enough to prevent erosion of the base mat. It is not clear how deep this is. SANDIA programs underway, however, could help answer this question. If a reflux path is available the core debris will probably not dry out and re-melt. This opinion is based on past work at TREAT, UCLA, ANL and SANDIA that shows that $\varepsilon = 0.45$ is a reasonable void fraction and that an average particle size of 500 μm is to be expected. For $\varepsilon = .45$ and 500 μm particle sizes the entire core and a great deal of steel (125 tons of fuel and steel) will remain coolable.

If vessel penetration occurs when no water is in the reactor cavity, a great deal of penetration of the base mat may occur. The amount of penetration occurring during the period when the core debris is molten is predictable. Once it freezes a complicated process occurs and the amount of penetration is not predictable. Again, studies are underway in Germany (their strong interest results because they do not allow water into the reactor cavity) that will answer this question within the next couple of years. Use of a er in the cavity could buy time for plant personnel to get water into the cavity.

The debris could enter the dry cavity and become particulates. The gas flow from the decomposing concrete might block water added later from entering the bed. It is not known whether the cooling by the gases from the decomposing gases will be sufficient to preclude re-melting. This sequence needs further study if it cannot be shown that water will always pr ede the melt.

To summarize, in existing plants where water precedes the melt in sufficient quantities and can be resupplied, penetration of the base mat will most likely not occur. Under these conditions an ex-vessel steam explosion will probably take place with the possibility of a great deal of steam generation that must be accommodated. The possibility of damage of the biological shield or shifting of NSSS components leading to containment damage needs to be further assessed. When water is not available, the chances of base mat penetration are much greater. The conclusion is that a water supply needs to be assured. A cavity liner of depleted UO_2, Al_2O_3, MgO or some similar refractory or sacrificial material should be considered.

A containment building could be designed based on present information to preclude molten core penetration. A conceptual design that has redundant cooling capability as well could include the following features:

1. Concrete that minimizes gas generation on decomposition and has the best possible refractory characteristics.

   Several courses of depleted UO_2 bricks actively cooled at the concrete-UO_2 brick interface similar to the SNR 300 core catcher. The heat sink could be an existing plant system.

3. A steel liner to protect the UO_2 bricks.

4. A cavity flooding capability and a method of refluxing to insure that the cavity stays flooded.

Such a system requires very little new technology and depends on no new research. It should also be relatively inexpensive.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 22 1980

MEMORANDUM FOR: G. Quittschreiber, ACRS

FROM: C. Kelber, RSR

SUBJECT: INPUT TO RESPONSE TO COMMISSIONER GILINSKY'S QUESTIONS ON CORE MELT

Pursuant to your memo of April 18, 1980 I have called Dr. Okrent to relay these thoughts:

1. The Zion/Indian Point study emphasizes the need to review the systems interactions of containment systems taking into account the probable state of the core and coolant systems at and following the time the threat to containment arises. From this review a risk based balance of the effects of system failures should be used as a major factor in deciding what systems to put in place.

2. Once a risk based balance is accepted as part of the decision system, many options are available to resolve the questions. Options include, for example, filtered, vented containment systems, highly reliable power and water supplies for containment cooling, active and passive core catchers, either sacrificial or refractory, or strong containments that are externally coolable, as the Byblis B plant.

3. Another factor in the decision process is the political factor: an inadvertant release of all the nobles through a filtered vented containment system is without health effects, but a deliberate release is probably politically impossible to permit at this time. Thus, such a system might be without merit in the next few years simply because permission to use a filtered vented containment system would probably be denied if it were ever requested in advance.

4. The Zion/Indian Point study revealed these problems to be most troublesome:

   The steam spike arising from sudden mixing of a molten core with water. Sprays or an ice condenser will ameliorate this problem if they are available. The trouble with a passive system such as the ice condenser is that it will probably be used up by the time the steam spike arises.

   The pressure from a massive hydrogen burn. Satisfactory hydrogen control methods appear to be available, but, of course, they have to be used.

   If the molten core is not cooled rapidly, radiation from the core may cause failure of massive components such as the pressure vessel, and such failures might lead to extensive damage to neighboring systems.

ATTACHMENT 2

5. It is likely to be necessary to by-pass the containment through the let down line or similar systems to maintain vital plant functions. The release through such bypasses does not appear to be substantial.

Charles N. Kelber, Assistant Director
Advanced Reactor Safety Research
Division of Reactor Safety Research

cc: T. Murley, RSR
D. Okrent

Marchese/Speis Input

to Dr. Dave Okrent, ACRS

"General Feelings on Containing a Core Melt"

Re: Memo from Quittschreiber, "Request for Input to Commissioner Gilinsky's Questions on Core Melt," dated April 18, 1980

April 25, 1980

ATTACHMENT 3

1. Refractory materials exist (e.g., MgO) which are considerably more benign than concrete is in terms of interactions with molten core debris and which would significantly reduce the associated production of water vapor, non-condensible and combustible gases generated by melt-concrete interactions.

2. A core retention device could not only prevent failure of containment via preventing melt-through of lower basemat, but could have a significant mitigating effect on the upper containment loading conditions by decreasing the pressure, hydrogen, aerosol and activity transients.

3. We believe that a core retention device will mitigate significantly core meltdown accident consequences by:

   a. reducing both the airborne releases caused by sparging of activity out of the core melt (thus reducing the vaporization fraction of the atmospheric releases) and the higher containment pressure when core melt material interacts with concrete; and

   b. reducing the likelihood of containment basemat penetration, thereby reducing the likelihood of ground water contamination via melt-through.

4. If containment fails prior to the melt contacting the core retention device, the major value of such a device in this case (insofar as airborne releases are concerned) would be to reduce the driving forces for leakages caused by non-condensible gas generation and sparging of activity out of the core melt debris.

5. An actively cooled core retention device would have the added advantages of (a) permanently retaining the core melt debris within the confines of the containment building, and (b) dissipating the core melt decay heat to the atmosphere, rather than retaining the heat of the melt inside of the containment building.

6. The value of minimizing the sparging phenomena and the vaporization releases extends broadly across the core melt spectrum but would have greatest importance to the risk dominating sequences. This value would be achieved primarily through reduction of the sparging induced release of tellurium and to some lesser extent it would reduce other isotopes. Since tellurium may be one of the dominant contributors to the health risks (from airborne releases), a core retention device could have a significant value in reducing the health risks from airborne releases.

7. For those nuclear plant sites located on soils of high permeability and in close proximity to major water resources, the use of a core retention device would be of greater relative value insofar as liquid pathway releases are concerned. Also, the use of a passive core retention device would have some value in terms of providing added time for interdictive measures to be taken against ground water contamination, thus further reducing the probability of such contamination.

8. If a controlled-vent-filter containment system proves desirable, a core retention device would significantly reduce the gas, vapor, aerosol, and activity loadings on such a system.

9. A core retention device could eliminate the water vapor evolved by melt-concrete interactions, thereby reducing melt-water reactions and the associated H_2 production in the region of the core retention device.

10. Conceptual designs of core retention systems for each of the reactor containment types should be undertaken; studies should be of the integrated, system type.

11. Need to consider special backfit problems associated with installing a core retention system in existing plants.

12. Those existing plants that either have a poor liquid pathway situation (with respect to rapid transport of core melt activity) or are located in areas of high population density should be given special emphasis.

13. Need to decide on whether to <u>delay</u> the melt-through penetration of the basemat or whether to <u>permanently retain</u> the core debris within the confines of the containment building.

14. In connection with Item 13, both passive and actively cooled core retention systems should be examined. Studies of passive systems should also consider natural circulation cooling around the extremities of a refractory bed of material.

15. After conceptual design studies are completed, the required R&D can be better focused to support the final design of the most promising of the core retention systems.

16. Primary problems which will require core melt R&D are in areas of materials interactions, heat transfer and fission product behavior.

17. For future plants, we believe that it is both technically feasible and practical to incorporate a core retention system into the reactor containment building that will significantly mitigate the consequences of core melt accidents.

18. For existing plants, we feel that the feasibility and practicality has to be examined on a case-by-case basis, including but not limited to considerations of high population density sites, liquid pathway problems, and containment types. The practicality of installing a core retention device in the lower reactor cavity region should be examined in terms of space availability, access, shielding, radiation levels and costs.

19. Besides NRC and its contractors, Reactor Manufacturers and A&E firms need to take this problem seriously and perform actual conceptual design studies of real core retention systems.

20. The combination of a stronger containment (i.e., higher design pressure) coupled with containment heat removal and core retention systems is a very desirable concept for future plants to preclude the need for venting in order to relieve pressure following a core melt accident. Public acceptance of nuclear power would be greatly enhanced if we could claim that we can contain the worst of the nuclear accidents.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 24 1980

MEMORANDUM FOR: Gary Quittschreiber, Senior Staff Engineer
Advisory Committee for Reactor Safeguards

FROM: Raymond DiSalvo,
Probabilistic Analysis Staff,
Office of Nuclear Regulatory Research

SUBJECT: REQUEST FOR INPUT TO COMMISSIONER GILINSKY'S QUESTIONS ON CORE MELT

REFERENCE: Your Memorandum, same subject, April 18, 1980

I am not sure whether you have requested comments on the specific proposal offerred by Mr. Latter or on the more general issue his letter addresses as raised by Commissioner Gilinsky. Mr. Latter's letter is short on technical substance. Nevertheless, I will frame my comments within the context of his letter and trust that they will be applicable to the more general issue.

COMMENTS ON RATIONALE

1. Latter recommends changing regulatory policy to require "containment of all accidents" without defining acceptability criteria. Does "containment" mean zero release, a release less than TMI-2, a release giving doses less than those in 10 CFR 100, no release to groundwater, or what? Does "all accidents" include low probability externally initiated events which could destroy the containment building, or sabotage or human error which could reduce the effectiveness of any containment design? Feasibility of retaining a molten core is a red herring. The real issue is acceptability.

2. If the technology of containment is "well understood and reliable" enough to base changes in currect policy on, why are we bothering to study containment response during accidents within and beyond design basis?

3. The sentence "If the probability of a containment failure were estimated to be low..." is puzzling. To what "empirically determined data" does it refer; containment leak tests? Their applicability to the questions at hand is suspect at best.

4. Latter states, "...the critical question is whether adequate containment is technically possible and economically reasonable. On the basis of our preliminary work, we believe the answer is yes." Putting aside the lack of definition for "adequate," I have no reason to doubt the conclusion. In fact, if I substitute "improved" for "adequate," I agree with his answer. I disagree that this is the "critical question," however, preferring to think that effectiveness and necessity are more important.

APR 24 1980

COMMENTS ON CONCEPTS

1. Latter offers little technical content on which to base an evaluation. The generalities are characteristic of earlier dissertations on improved containment design except that Latter is the first soul I have found who is confident enough to predict "no releases of radioactivity."

2. Reducing static pressure may only be a partial answer to retaining containment integrity. Recent analyses by Battelle Columbus Laboratories and Sandia National Laboratories* indicate that pressure spikes from rapid generation of steam or hydrogen burning also challenge containment integrity in core melt sequences having relatively high probability of occurrence. Concepts have been proposed to overcome this potential problem.

3. Containing a molten core via a core retention device and a passive cooling system is feasible to the extent that on the order of 40 MW can be transferred from the core debris to its surroundings and an ultimate heat sink via natural circulation. In addition to being a "dense, inert, low melting point, and high thermal conductivity melting bed" (could it be lead?), it would be desirable that the material be economically available, workable into the proper configuration, and that its production and fabrication have no adverse effects on health, safety and environment.

4. In order to achieve the appearance of total containment of a core melt accident, all recognized containment failure modes must be precluded. This means eliminating failure to isolate in addition to the more spectacular failure modes Latter cites. The probability of isolation failure may put a lower limit on the feasibility of containing a core melt totally.

Please let me know if I can provide any further information or clarification of these comments.

* W. B. Murfin, "Summary of the Zion/Indian Point Study," SAND80-0617, NUREG/CR-1409, in publication.

Raymond DiSalvo,
Probabilistic Analysis Staff
Office of Nuclear Regulatory Research

cc: D. Okrent, ACRS
I. Catton, ACRS

SAND80-0887

Development and Analysis of Vent-Filtered Containment Conceptual Designs

A. S. Benjamin and H. C. Walling
Sandia National Laboratories
Albuquerque, NM 87185

ABSTRACT

Conceptual filtered-vented containment systems have been postulated for a reference large, dry, pressurized water reactor containment, and the systems have been analyzed to determine design parameters, actuation/operation requirements, and overall feasibility. The primary design challenge has been found to emanate from pressure spikes caused by core debris bed interactions with water and by hydrogen deflagrations. Circumvention of the pressure spikes may require a more complicated actuation logic than has previously been considered. Otherwise, major reductions in consequences for certain severe accidents appear to be possible with relatively simple systems. A probabilistic assessment of competing risks remains to be performed.

INTRODUCTION

The use of containment venting systems has been suggested by many as a means for significantly mitigating the risks from core melt accidents. Recently, the potential benefits of filtered-vented containment systems have been cited by such diverse groups as the California Energy Commission,[1] the Advisory Committee on Reactor Safeguards,[2] the TMI Lessons Learned Task Force,[3] the Rogovin Inquiry Group on Three Mile Island,[4] and the Swedish Government Committee on Nuclear Reactor Safety.[5]

In April 1979, a program was initiated at Sandia National Laboratories under contract with the U. S. Nuclear Regulatory Commission to investigate filtered-vented containment concepts for light water reactors. The program has the following objectives:

1. Development of conceptual designs of vent-filter systems which have the potential to mitigate the effects of accidents (particularly core melt accidents) that are beyond the current design basis.

2. Determination of the potential reduction in radioactive releases for core-melt accidents and the resultant reduction in overall risks.

3. Determination of the effect of the vent-filter on non-core-melt accidents and on normal operations.

4. Specification of system performance and safety design requirements for vent-filter systems.

5. Quantitative analysis of values versus impacts.

ATTACHMENT 5

The study considers several types of containment (i.e., large dry PWR, ice condenser PWR, Mark I BWR, and Mark III BWR) and includes both existing and new plants. A program schedule is presented in Figure 1.

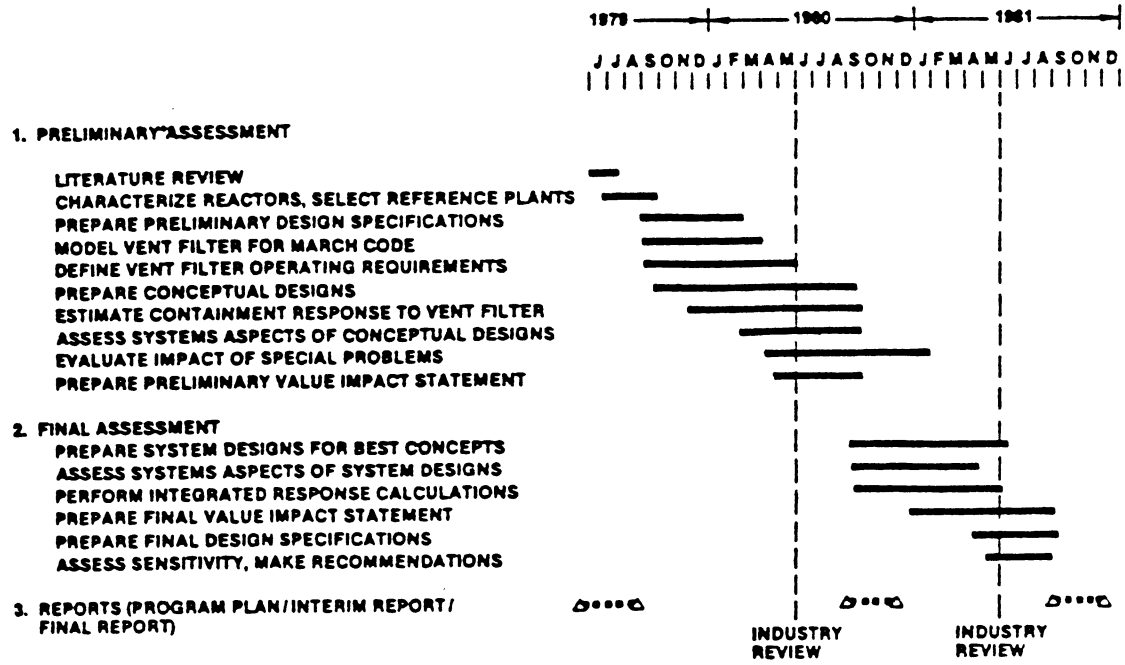


Figure 1. Program Schedule.

The risk reduction potential of vent-filter systems derives from their dual function of venting containment to prevent overpressurization from the generation of steam and noncondensibles and of filtering the effluent to limit the release of radioactive materials. In theory, post-accident filtration systems can reduce the risk from nuclear reactor accidents significantly; in practice, there are many engineering, technical, economic, and licensing questions to be answered before judgments on feasibility and effectiveness can be made. These questions include the capacity of the system to handle large pressure surges, possible interference with other engineered safety features, possible exacerbation of low-consequence accidents into high-consequence accidents, possible increase of hydrogen explosion potential, impact of uncertainties in various phenomenological and cost evaluation areas, and difficulties in reconciling vent-filter systems with the current regulatory position requiring essentially leaktight containment. These and other issues are discussed in the Sandia program plan for filtered-vented containment studies.[6]

The purpose of this paper is to provide a status report of the studies performed since the program plan was completed in October 1979, and to indicate the directions in which studies are progressing. Most of the analyses performed to date correspond to a reference large, dry, pressurized water reactor (Westinghouse design 4-loop plant) chosen because of its proximity to a population center. The results provided below correspond to this reference PWR.

TECHNICAL ISSUES

Accidents that Challenge the System

In the Reactor Safety Study,[7] and in subsequent studies based on the RSS methodology, a small number of accident sequences were found to dominate the overall risk for each reactor. For the large, dry PWR analyzed in the RSS, the dominating sequences were found to be TMLB' (i.e., loss of all AC power leading to failure of secondary heat removal), S_2C (i.e., a small LOCA with loss of containment sprays leading to loss of containment heat removal), and V (i.e., failure of the LPIS check valve leading to a LOCA outside containment).

In the present study, it has been considered important for the initial stages to consider not only those accidents which are thought to dominate the risk but also those which might provide the greatest challenge to a vent-filter system. For the reference PWR considered in the present study, the accident scenarios listed in Table I were judged to provide a reasonably complete bounding of accidents that both dominate the risk and challenge the vent-filter system.

Table I. Accident Scenarios Considered for Reference PWR Designs.

Accident Symbology	Accident Sequence	Limiting Characteristics
TMLB'	Loss of offsite and onsite AC power for 16 hours, resulting in loss of secondary heat removal, followed by the return of AC power and restart of the containment coolers.	Maximum pressure following reactor vessel failure (about 120 psia).
AB-Burn	Large LOCA plus loss of offsite and onsite AC power for 16 hours, followed by the return of AC power and restart of the containment coolers. The hydrogen ignites when the molten core drops into the cavity.	Shortest time for generation of a pressure exceeding containment design pressure (about 50 minutes).
S_2D-Burn	Small LOCA plus loss of ECCS injection capability, resulting in the loss of ECCS recirculation and containment spray recirculation capability. The hydrogen ignites when the molten core drops into the cavity.	Maximum potential pressure spike following reactor vessel failure (amount not yet established).
S_2C	Small LOCA plus loss of heat sink for containment coolers and containment sprays. This accident results in containment overpressurization before meltdown.	Maximum steam production (about 4×10^6 lbm).
TMLB"	Same as TMLB', except AC power returns after about 6 hours, leading to restart of containment coolers, containment sprays, and ECCS injection.	Most potential for system interactions during core-melt accident.
A-Vent	Large LOCA causing premature actuation of containment venting. All engineered safety features are assumed to operate on demand.	Potential for exacerbation of non-core-melt accident.

Pressure Spikes

A noteworthy feature of many of the accident scenarios that result in core meltdown is the occurrence of a sizable containment pressure spike at or near the time of reactor vessel failure (see, for example, Figure 2). The causes of the spike vary from case to case, but combinations of the following phenomena are generally responsible:

1. Steam release from the primary system to the containment when the reactor vessel fails at high pressure. (Accidents initiated by transients and small LOCA's, about 13 psi for reference PWR.)

2. Rapid steam formation caused by molten core interaction with water existing in the cavity at the time of reactor vessel failure. (Magnitude dependent on accident and amount of communication between sump and cavity.)

3. Rapid steam formation caused by flashing of some of the residual water in the primary loops when the reactor vessel fails, and by dumping of the remainder of this residual water onto the molten core in the cavity. (Accidents initiated by transients and small LOCA's, about 16 psi for reference PWR.)

4. Rapid steam formation caused by discharge of accumulator water at the time of reactor vessel failure and interaction of this water with the molten core in the cavity. (Accidents initiated by transients and small LOCA's, about 34 psi for reference PWR.)

5. Deflagration of the hydrogen produced by Zircaloy-steam reaction, triggered by the interaction of the molten core with the concrete in the cavity. (Accidents resulting in a flammable mixture, about 60 psi for reference PWR.)

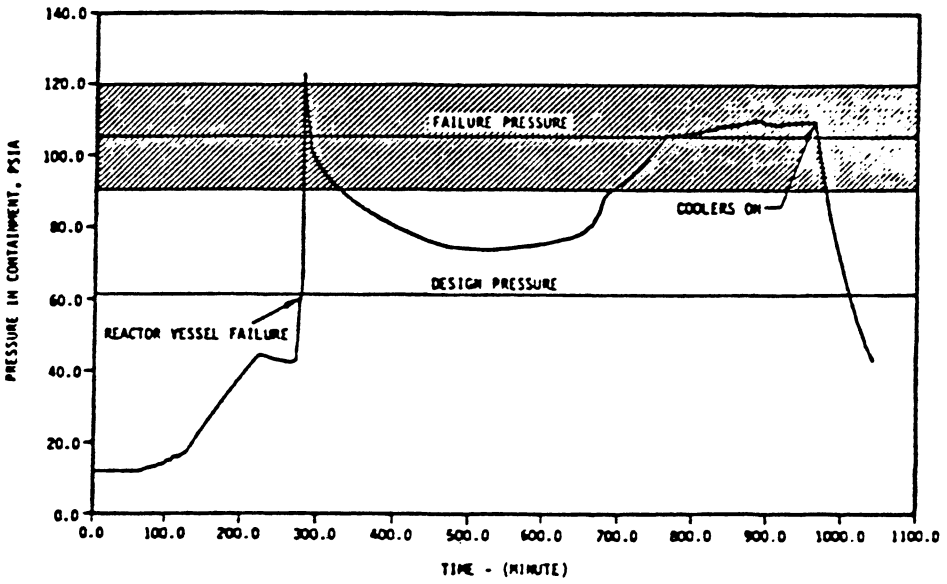


Figure 2. MARCH Code Calculation of Containment Pressure Versus Time for the TMLB' Accident in the Reference PWR.

The pressure spike in Figu. 2 was the cumulative result Items 1, 3, and 4, above.

The interactions of the core materials with water in the reactor cavity pose a particular concern. The rate of the interaction depends upon a number of difficult phenomenological questions, such as the size of the vessel rupre, the rate of dropping of the molten core into the reactor cavity, the egree of core fragmentation in the cavity and the resulting debris geometry, the possibility of steam explosions, and the question of whether the debris dries out and remelts or remains coolable. Since the data are inconclusive in all of these areas, it was considered best at present to make the apparently conservative assumptions that the vessel rupture area is very large, that the dropping of the core is immediate, that complete fragmentation occurs without dispersal out of the cavity, and that the debris does not dry out prior to the boiloff of the water. With these assumptions, the duration of the pressure rise caused by core-water interactions is about 15 seconds, the time required for the accumulators to discharge in the absence of a back pressure.

System Interactions

There are several plausible scenarios in which adverse system interactions could be caused by the venting of containment. During accidents such as S_2G (Table I), a rapid venting of containment can cause the recirculation pumps to cavitate as a result of sump flashing, leading to core uncovering and meltdown. During TMLB', the restoration of sprays and coolers after venting can create a severe vacuum which could cause containment failure in compression. During A-Vent, the premature venting of containment might degrade the reflood operation by removing the back pressure. Avoidance of these adverse interactions requires either design solutions, such as the incorporation of vacuum breakers, or preventive administrative procedures, such as a temporary realignment of the recirculation pumps to an outside source or a revision of set points for coolers and sprays. Evaluation of these interactions and their possible solutions is not yet complete.

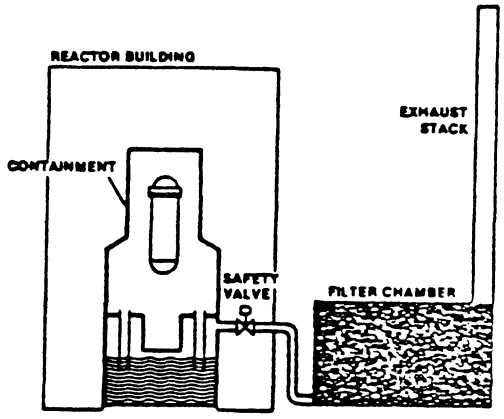
DESIGN POSSIBILITIES

Containment Vent Strategies for New Reactors

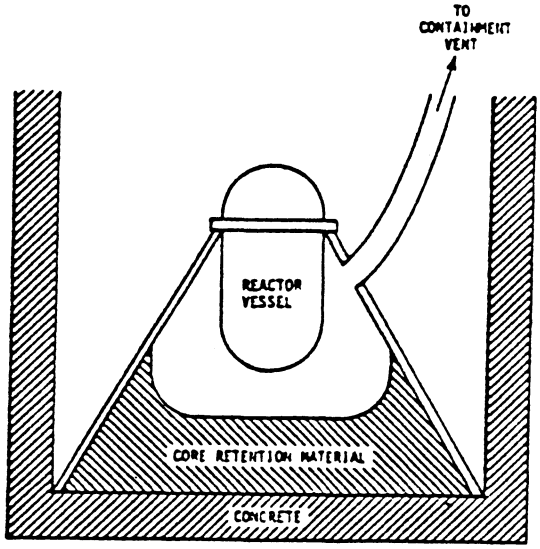
The primary challenge to a vent-filter system is its ability to mitigate the pressure spikes in containment. To accomplish this goal, it is much easier to formulate design concepts for new reactors (i.e., reactors that have not yet been built) than for reactors that already exist.

Three design possibilities for new reactors are shown in Figure 3. In one concept (Figure 3a), a large vapor suppression pool is placed within the containment to suppress a portion of the pressure spike as well as to remove the steam, cool the noncondensibles, and trap most of the particles and iodine. This design is similar to one suggested by the Swedish[5] for their boiling water reactors, except that the suppression pool is enlarged in order to accommodate steam generation during core melt accidents. Another design possibility being investigated is the use of a vented guard structure around the reactor vessel with core retention materials (Figure 3b). This concept diffuses and mitigates the containment pressure spike at the time of reactor

(a) Suppression Pool in Containment



(b) Reactor Vessel Guard Structure



(c) Vacuum Vent Building

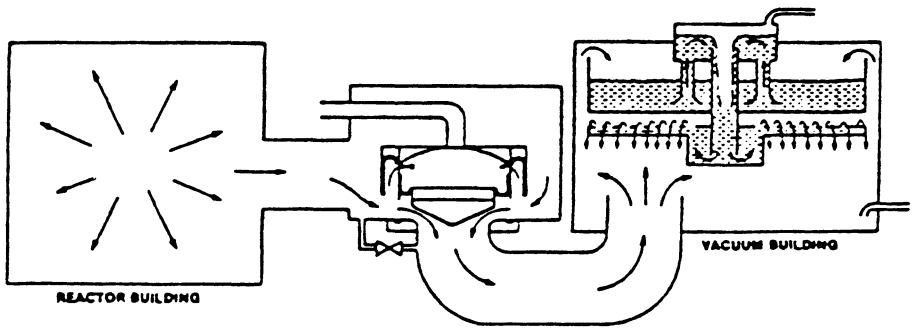


Figure 3. Schematics of Vent-Filter Design Concepts for New Reactors.

vessel failure by slowing the rate of primary system depressurization and accumulator discharge and by venting the primary system hydrogen before it mixes with the containment atmosphere. In another concept similar to that used in some Canadian reactors[8] for design basis accidents (Figure 3c), a large vent (on the order of 20 feet in diameter) may be used to connect the reactor containment to an evacuated vent building.

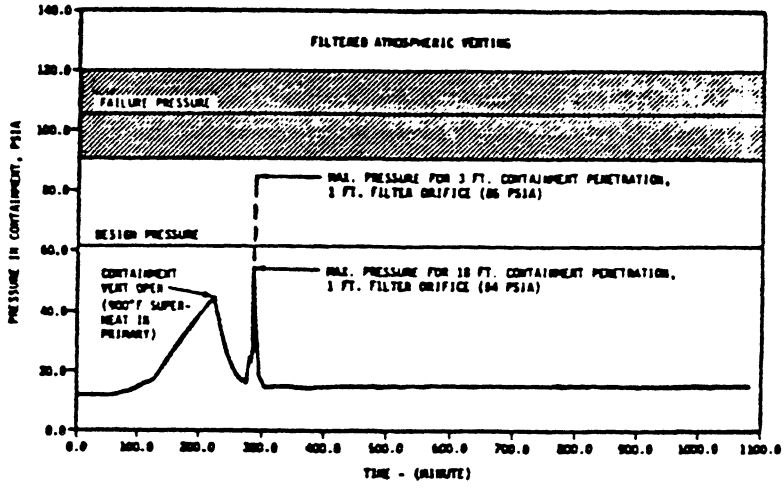
Containment Vent Strategies for Existing Reactors

The possibilities for retrofitting existing containments are limited by the fact that there is generally not room within containment for a large suppression pool or in the reactor cavity for a guard structure. Also, the creation of a large penetration in the containment boundary is prohibited for structural reasons. If it develops, therefore, that a rapid pressure spike does represent a serious threat to containment integrity (i.e., that it cannot be ruled out on phenomenological grounds), then one might consider several alternative strategies. One strategy might be to anticipate the reactor vessel failure and to initiate filtered atmospheric venting in advance. This strategy would reduce the containment pressure to a point where a sizable pressure spike could be accommodated without threatening the containment. A variation of this strategy for accidents initiated by transients or small LOCAs might include venting the primary system into the containment (or into the containment vent line) through existing primary system vent paths. Such an action allows the accumulators to discharge before the core melts down, which increases the chances for recovery and, if the reactor vessel still fails, reduces the magnitude of the steam spike. A different variation might include flooding the containment while the accident is progressing by gravity-induced flow from a large, elevated water tank. A million gallons of water in the bottom of the containment would offer a very large, passive heat sink that could function as an internal suppression pool. Still another vent strategy might be to use the existing equipment hatch to provide a large enough opening to vent a portion of the steam spike to a large external suppression pool or vacuum building. This strategy may be more costly to implement but is less likely to depend upon operator judgment.

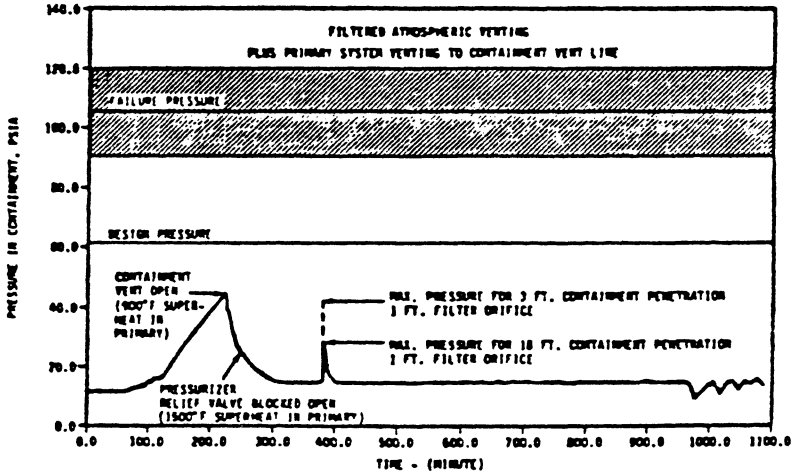
All of these strategies have implied risks, such as possible system interactions or human errors, that require a careful examination. When all the risks are evaluated, a simple vent strategy such as containment pressure relief at a setpoint above the design pressure, though perhaps less effective for the most severe accidents, may be more attractive overall.

The results of MARCH code[9] calculations of containment pressure and temperature response are shown for certain vent strategies in Figures 4 and 5. Figure 4 shows containment pressure histories for the TMLB' accident in the reference PWR for the following cases: (a) venting through filters to the atmosphere based on anticipation of reactor vessel failure, (b) venting through filters as in Case (a), but also with anticipatory primary system venting to the containment vent line via the pressurizer relief valve, and (c) venting from containment to a second building. Figure 5 shows the temperature of the containment atmosphere as a function of time for the various cases considered in Figure 2 (without venting) and Figure 4 (with venting). It may be observed that the utilization of containment venting lowers both the maximum containment pressure and the containment temperature.

(a) Filtered Atmospheric Venting



(b) Filtered Atmospheric Venting and Primary System Venting



(c) Venting to a Second Building

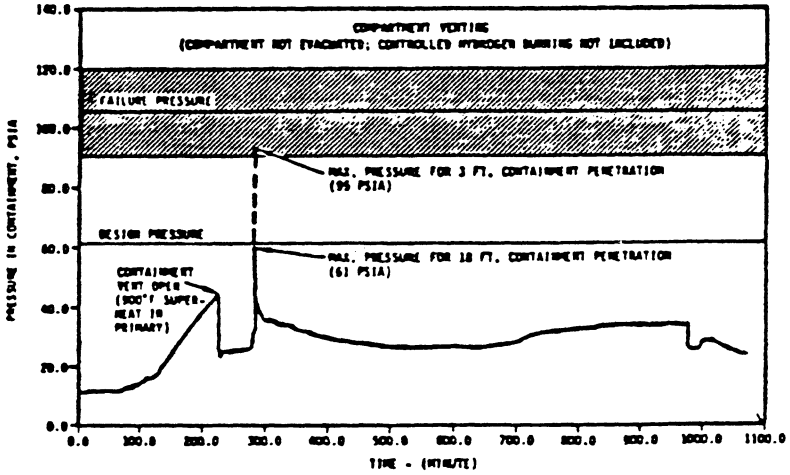


Figure 4. MARCH Code Calculations of Containment Pressure Versus Time for Various Venting Options During the TMLB' Accident in the Reference PWR.

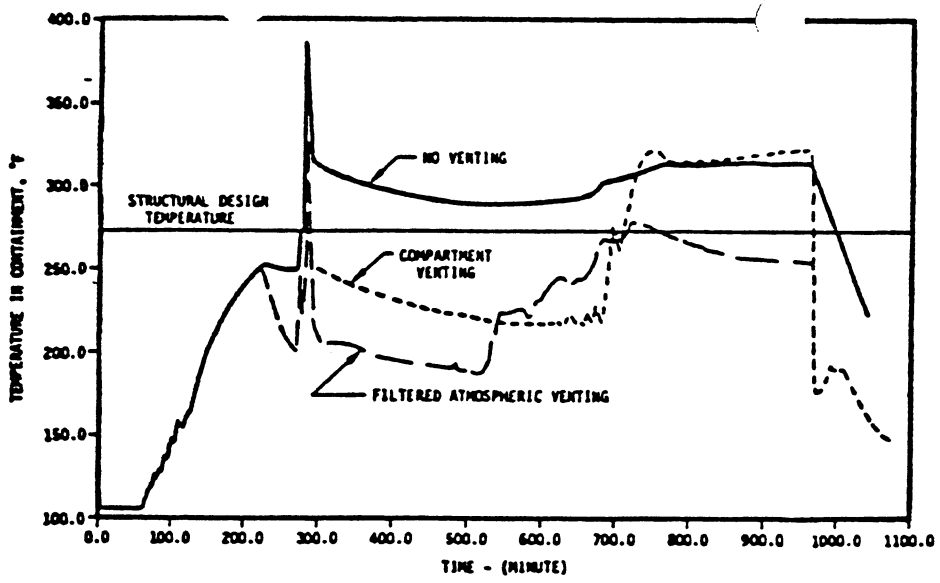


Figure 5. MARCH Code Calculations of Containment Atmospheric Temperature Versus Time for Various Venting Options During the TMLB' Accident in the Reference PWR.

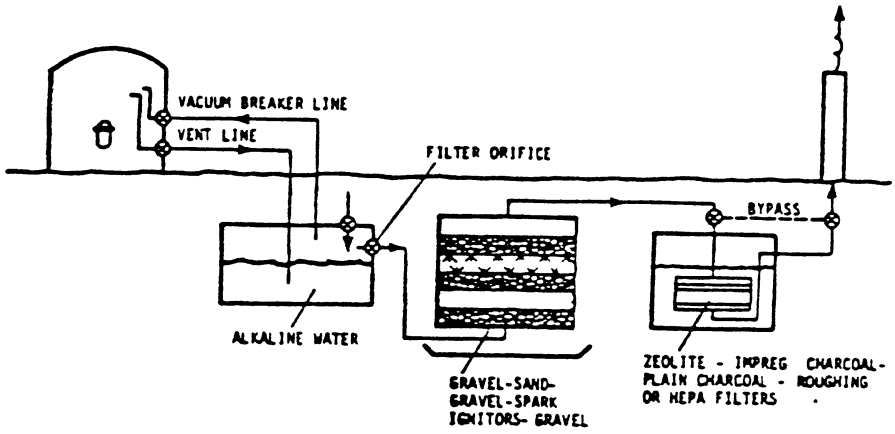
Cooler/Condenser and Filter Components

Various options are being considered for the external portion of the vent-filter system, with different degrees of complexity and different costs, corresponding to various levels of fission product entrapment. One of the options is shown schematically in Figure 6. The system is designed to operate successfully without AC power during a loss-of-power accident for a period of well over 16 hours, the time at which power is assumed to be restored. Thereafter, the operation of the system changes from a vent to a recirculation mode so as to eliminate further releases to the atmosphere. In the recirculation mode, the designs incorporate a heat exchanger to remove heat from the water and blowers to drive the circulating flow and to cool the charcoal filters.

The primary condensing/cooling component preceding the filter trains in the option shown is a vapor suppression water pool. The submerged portion of the pool (about 150,000 ft^3) provides enough heat sink to passively condense all the steam that is generated during the accidents TMLB', AB-Burn, and S_2D-Burn. The air space (also about 150,000 ft^3) allows for the additional amount of water produced by vapor suppression during accidents such as TMLB" and S_2G.

If the filters in the vent-filter system were designed to accommodate the flow rates required for anticipatory containment venting, the gravel-sand filter would have frontal dimensions of about 120 ft x 100 ft and a height of about 20 ft, including spark ignition sources for burning hydrogen. The adsorber system would have a frontal diameter of about 36 ft and a depth of about 6.5 ft, including a 4-inch zeolite guard bed to retain inorganic iodine, a 2-inch impregnated charcoal bed to capture organic iodine, a 5.5-ft (100 ton) plain charcoal bed to retain the xenon, and 2 inches of roughing or HEPA filters to prevent charcoal particles from escaping up the stack. The entire assembly, in a waterproof container, could be immersed in a 20,000 ft^3 water tank to remove heat via natural convection until power is restored.

(a) Without AC Power



(b) With AC Power

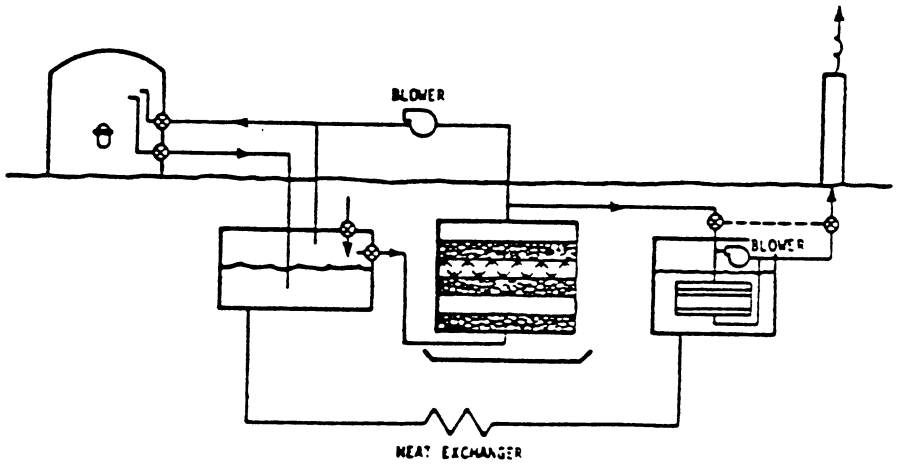


Figure 6. Filtered Atmospheric Venting Option. Estimated Collection Efficiencies: 99.98% Particles, 99.98% Inorganic Iodine, 99.95% Organic Iodine, 98% Xenon, 10% Krypton.

Simpler variations of the system in Figure 6 can be obtained by removing various components. Consequence evaluations for four variants of Figure 6 illustrate that for the TMLB' accident in the reference PWR, a large reduction in latent cancer fatalities and property interdiction and an elimination of early fatalities can be accomplished just by venting the containment through an alkaline suppression pool (See Figure 7). The consequence calculations were based on the Reactor Safety Study models applied to the reference PWR using site-specific weather and population data and a 5-mile evacuation radius (instead of a 25 mile evacuation radius). It was assumed that the vent-filter systems operate as designed and that the effluent from the filters is released (at ambient temperature) at an elevation of 180 ft. It should be emphasized that these calculations correspond to one accident only, and do not reflect the effect of vent-filter systems on overall reactor risks.

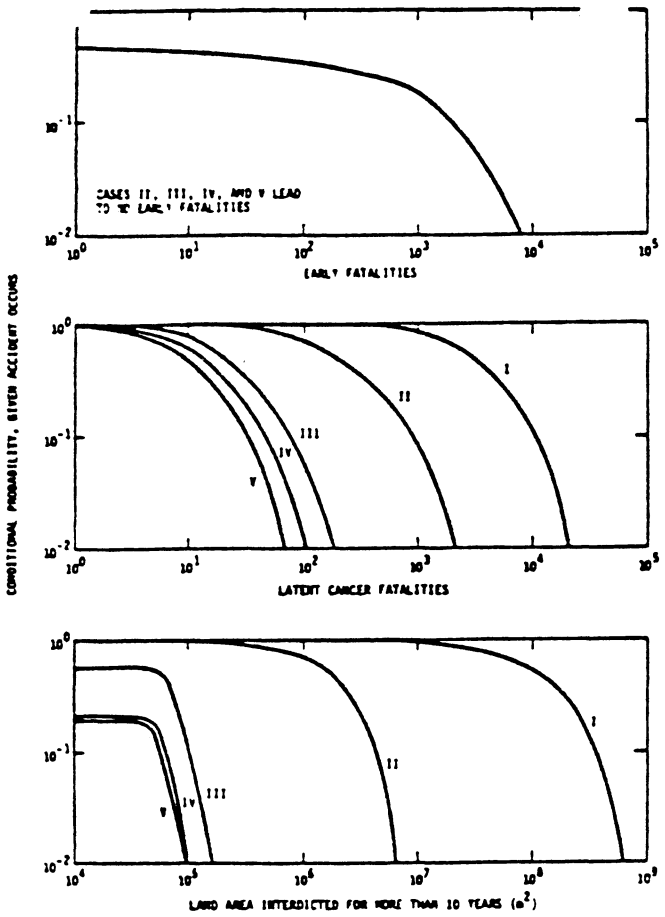


Figure 7. Probability of Early Fatalities, Latent Cancer Fatalities, and Land Interdiction for Various Filtered Venting Options, Given Occurrence of the Accident TMLB'.

CONCLUSIONS

The primary challenge to a filtered-vented containment system is the pressure spike that could occur in containment if the molten core penetrates the reactor vessel and drops into the cavity. The main contributors to the spike in a large, dry PWR are rapid vaporization of water in the cavity and the possibility of hydrogen deflagration caused by core-concrete interaction. Large phenomenological uncertainties are associated with these processes, and exploratory research is needed to better define the rate and magnitude of the pressure transient.

It presently appears that for certain severe accidents in large, dry PWRs, retrofitted vent-filter systems can be successfully utilized to circumvent containment overpressurization. For these accidents, major reductions in consequences appear to be possible with relatively simple systems. Because of space limitations and containment structural considerations, however, the

actuation and operation a retrofit system is likely require a greater degree of automatic control and/or operator participation than has previously been assumed. Before the overall risk reduction potential of vent-filter systems can be established definitively, a more detailed evaluation of a variety of accidents including considerations of actuation reliabilities, potential adverse system interactions, and possible failure modes including operator error is required. These analyses, which are now in progress, will provide the required inputs for a comprehensive assessment of competing risks.

ACKNOWLEDGMENTS

The authors wish to express their appreciation to the following people who contributed important technical support to this project: G. J. Boyd and D. C. Aldrich, Sandia National Laboratories; P. Cybulskis and R. O. Wooton, Battelle Columbus Laboratories; R. A. Haarman and D. T. Pence, Science Applications, Inc.; and R. DiSalvo, U. S. Nuclear Regulatory Commission. This work was sponsored by the Probabilistic Analysis Staff of the U. S. Nuclear Regulatory Commission.

REFERENCES

1. Staff of the California Energy Commission, "Underground Siting of Nuclear Power Reactors: An Option for California," Draft Report, June 1978.

2. "Interim Report No. 3 on Three Mile Island Nuclear Station Unit 2," letter from M. W. Carbon, Chairman ACRS, to J. M. Hendrie, Chairman USNRC, May 16, 1979.

3. Staff of the U. S. Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Final Report," NUREG-0585, October 1979.

4. M. Rogovin, et al., "Three Mile Island, a Report to the Commissioners and to the Public," NRC Special Inquiry Group, January 1980.

5. January 1980 Issue of Nuclear News, pp. 55-57.

6. A. S. Benjamin, "Program Plan for the Investigation of Vent-Filtered Containment Conceptual Designs for Light Water Reactors," SAND79-1088, NUREG/CR-1029, October 1979.

7. N. C. Rasmussen, et al., "Reactor Safety Study," WASH-1400, October 1975.

8. E. W. Fee and G. E. Shaw, "Vacuum Containment Systems for Multi-Unit Nuclear Power Stations," 7th Int. Congress on the Confinement of Radioactivity in the Utilization of Nuclear Energy, Societe Francaise de la Radioprotection, Versailles, France, May 1974.

9. R. O. Wooton and H. I. Avci, "A User's Manual for MARCH, A Light Water Reactor Meltdown Accident Analysis Code," Battelle Columbus Laboratories, Draft Report, December 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 8, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: NRC POLICY ON THE SEVERE ACCIDENT RULEMAKING AND RELATED MATTERS

Dear Dr. Palladino:

During its 262nd meeting, the ACRS discussed with the NRC Staff the proposed approach to dealing with SECY-82-1, "Severe Accident Rulemaking and Related Matters," dated January 4, 1982. We also considered the Commission's comments made at the January 6, 1982 Commission meeting and contained in the memorandum, Samuel J. Chilk to William J. Dircks, dated January 29, 1982.

We have the following comments:

(1) We believe that the licensing and regulatory process should take specific account of accidents more severe than those currently specified as design basis accidents.

(2) We believe the approach being taken to the problem of dealing with hydrogen is appropriate. We will continue to follow developments.

(3) Dealing with the issue of severe accidents during the review of future standard plants may be a feasible mechanism. However, if it is to be effective, we believe, and we note the Commission has also commented, that more guidance to applicants is needed than is provided in SECY-82-1.

(4) Although we have not yet given the question detailed consideration, we suggest that the guidance should include containment performance, the performance of systems for removal of decay heat from the core and of heat from containment, the use of containment venting, and designs which inhibit sabotage.

(5) We note that this approach in dealing with severe accidents does not specifically deal directly with operating plants and with plants under construction. We believe that, if the standard plant approach is adopted, an approach to severe accidents in operating plants and in plants under construction should be carried on in parallel.

(6) If the Commission were willing to establish an appropriate NRC working group with the mission of developing alternative approaches to resolving issues relevant to severe accidents, the ACRS would be willing to work with such a group on this task.

Sincerely,

P. Shewmon

Paul Shewmon
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 18, 1982

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS COMMENTS ON "NUCLEAR PLANT SEVERE ACCIDENT RESEARCH PLAN," NUREG-0900 (DRAFT)

During its 268th meeting, the ACRS discussed with the NRC Staff a version of "Nuclear Plant Severe Accident Research Plan," NUREG-0900 (Draft) which accompanied a draft of SECY-82-203A (August 1982). The NRC Staff had discussed an earlier version of NUREG-0900 (April 7, 1982) with a Subcommittee on May 28, 1982, and had discussed this most recent version of the report with the same Subcommittee on August 6, 1982.

In its report to the Commission on the FY 1984 and 1985 Research Budget, NUREG-0875 (July 14, 1982), the ACRS made the following comment:

> We have recommended repeatedly in our reports to the Commission and to the Congress that the research in the Accident Evaluation and Mitigation Decision Unit be structured to answer questions arising in connection with reactor regulation and licensing. In our recent reports, specific attention was called to the need for organizing the research under this Decision Unit to answer questions likely to arise in connection with the Commission's stated intention to modify the licensing process to take specific account of accidents more serious than those generally identified as Design Basis Accidents. However, there is still a lack of definition of even one approach to deal with the severe accident issue. Considering the difficulty of the problem, effort should probably be made to define several alternatives. Until these efforts have produced some initial results, guidance for research of the type described will not be available.

Neither the original nor the revised version of NUREG-0900 contains a delineation of an approach for dealing with severe accidents. This is needed to judge the appropriateness of the proposed research program. We continue to urge that the work necessary to provide one or more approaches be carried out. We look for requirements that might be placed on components or systems required to deal with severe accidents, description of what is now known about these, specification of what, if any, information is required to describe system performance with the necessary accuracy, some indication of whether the information can be obtained from research in the time and with the resources available, and what research is planned to obtain the needed information.

As an example, we note that, in the draft Implementation Plan for Safety Goals (July 16, 1982) provided to us, the NRC Staff concludes that it is not now feasible to specify the performance of containment systems. The NRC Staff further expressed an opinion that the information and approach needed for such a specification should be developed. We, therefore, looked at NUREG-0900 for a description of what new information is needed to specify performance of the various kinds of containments and containment systems now in use or proposed. Although there are elements of the program that could certainly contribute to more accurate specification of containment performance, we find no systematic description of what information is needed or what part of the proposed program is designed to provide the information.

The research program described in NUREG-0900 is elaborate, extensive and expensive. However, documents such as the Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation, SECY-82-1A (July 16, 1982), give the impression that the NRC Staff considers it likely that most operating plants will meet, perhaps with some minor changes, whatever criteria are finally devised for plant system performance in severe accident situations. Most of the research is said to be confirmatory. It is curious that the recommended program is needed for confirmation. The report and the associated documents do not make explicit what is to be confirmed.

Because of the importance of this issue, and of the associated issues of an appropriate Safety Goal and its implementation, the ACRS plans to devote considerable attention to these questions over the next several months. In our letter of February 8, 1982, we offered to assist the NRC Staff in developing a new approach for dealing with Severe Accidents. We repeat that offer.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 14, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON SECY-82-1A: PROPOSED COMMISSION POLICY STATEMENT ON SEVERE ACCIDENTS AND RELATED VIEWS ON NUCLEAR REACTOR REGULATION

During its 269th meeting, September 9-11, 1982, the Advisory Committee on Reactor Safeguards reviewed SECY-82-1A, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation." In its review the ACRS had the benefit of Subcommittee meetings held on August 6, 1982 and September 8, 1982, discussions with the NRC Staff, and the documents listed.

The ACRS recommends against publication for comment of the policy statement on severe accidents in the form proposed in SECY-82-1A. This recommendation is based in part on the following:

1. The NRC Staff, in SECY-82-1A, proposes to replace the long-term generic rulemaking effort relating to core melt accidents by multiple severe accident rulemaking actions designed to certify specific standard plant design applications, and by regulatory decisions based on generic evaluations regarding all classes of existing plants. For future plants, compliance with the recently implemented CP rule* would be required; in addition, a probabilistic risk assessment (PRA) would be required as part of the construction permit (CP) review. The applicant would also have to consider all Unresolved Safety Issues and commit to meet those requirements for design features for prevention, management, or mitigation of severe accidents that are shown in the course of the rulemaking to be cost-effective. Applicants would also be expected to address in their safety analysis reports external events, human errors, and sabotage.

   Although SECY-82-1A notes the problems associated with the immaturity of PRA, it seems to rely very heavily on the PRA process for benefit/cost decisions during the proposed rulemaking actions on "requirements for design features for prevention, management, or mitigation of severe accidents." Instead of attempting to have the Commission provide the designer with policy guidance, as feasible, prior to his making most major design decisions and having accomplished much of his detailed design, the proposed process appears to leave many safety-related design decisions to benefit/cost analysis during the different rulemakings,

* Federal Register Notice dated January 15, 1982, (47 FR, p.2286)

despite the likelihood that many issues will include uncertainties sufficiently large to render the benefit/cost ratio unclear.

If this approach is followed, what general guidance will be provided for the designer in deciding which approaches to use to prevent core melt accidents, to mitigate core melt accidents, to improve containment effectiveness, to reduce the likelihood of significant sabotage, etc.? Even if the Commission had adopted and made effective for this purpose a set of safety goals and quantitative design objectives similar to those in NUREG-0880, would PRA be adequate to guide the decision-making? Is it likely that features added as a result of the rulemaking will fit as well into the overall plant design as those initially incorporated by the designer? Furthermore, on what basis is it expected that an appropriate consistency in safety approach would be accomplished among several different rulemakings, each devoted to a specific standard plant?

If there were to be only one standard plant and there were an extended period for review and evaluation of several preliminary design concepts before the applicant had to choose a specific concept and perform the detailed design analysis needed for a CP, the approach in SECY-82-1A might be feasible and appropriate.

We believe that, before embarking on the course proposed for future CPs in SECY-82-1A, a concerted effort should be made by the NRC Staff and the ACRS to develop policy guidance on as many of the relevant safety issues as are tractable, and to propose an alternate approach to the Commission in which such policy guidance is provided to applicants for future standard plant designs. The time scale for accomplishment of such an effort might be two years, roughly the same period now envisaged as needed to make decisions concerning severe accident requirements for existing reactors, as well as to perform a first trial test of implementation of safety goals.

2. In SECY-82-1A the NRC Staff draws the conclusion that standard plant designs now at the Final Design Approval stage of development, when upgraded to conform with the recent CP rule, can be shown to meet the safety goal for a broad range of future sites, including consideration of severe accidents. The NRC Staff notes that it does not expect its present views on severe accident considerations to change substantially as a result of ongoing research with regard to the fundamentals of the present designs and their general adherence to NRC safety policy. The NRC Staff expects research results to permit further risk reduction by identifying worthwhile refinements in the design of operating reactors or their operating practices rather than identifying major redesign needs. Fairly strong conclusions have also been drawn in SECY-82-1A about the benefit/cost trade-off of several possible design features to mitigate accidents; however, the NRC Staff now believes that one of these conclusions may be reversed when external events are considered.

We believe it is premature to draw many of these conclusions. If the NRC Staff believes that strong technical bases exist for them, it would be well to have such bases made available for review, evaluation, and critique by the Commission, the ACRS, and others, prior to publication for comment of the policy statement in SECY-82-1A.

With regard to future plants, we believe that the NRC should examine and evaluate the safety-related changes now proposed or underway for LWRs in countries like France, the Federal Republic of Germany, Japan, Sweden, and the United Kingdom before arriving at its own judgment on what is appropriate for the U.S. For existing nuclear power plants, it is premature to assume that the available PRAs provide a generic basis for decision-making. On the contrary, despite their uncertainties, the PRAs indicate the existence of important plant-specific differences which need to be factored into the formulation of policy. Again, the specific backfitting approaches currently underway or contemplated for LWRs in other countries should be examined and evaluated for their relevance to U.S. policy.

3. Because of their close interrelation, one would expect that the proposed safety goals, the related implementation plan, the backfitting policy statement, and the severe accident policy statement would be closely integrated and would follow a coherent safety philosophy. Unfortunately, this is not the case. We recommend that an increased effort be made to accomplish such integration.

4. With regard to existing plants, we believe it would be productive for the NRC Staff to draft alternate positions on the most significant safety issues and to establish what would be needed in order to evaluate the alternatives. The ACRS would be willing to cooperate with the NRC Staff in such an effort.

Sincerely,

P. Shewmon
Chairman

References:

1. SECY-82-1A, Policy Issue (Affirmation) from W. J. Dircks, Executive Director for Operations, Subject: Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation, dated July 16, 1982
2. Letter from E. P. Rahe, Jr., Manager, Nuclear Safety Department, Westinghouse Electric Corporation, to Dr. Paul Shewmon, Chairman, Advisory Committee on Reactor Safeguards, Subject: severe accident policy in SECY-82-1A, dated September 3, 1982

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 10, 1983

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON SECY-82-1B: PROPOSED COMMISSION POLICY STATEMENT ON SEVERE ACCIDENTS AND RELATED VIEWS ON NUCLEAR REACTOR REGULATION

During its 273rd ACRS meeting, January 6-8, 1983, the Advisory Committee on Reactor Safeguards discussed SECY-82-1B, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," dated November 24, 1982. We also considered the memorandum of October 25, 1982, Samuel J. Chilk, SECY, to William J. Dircks, EDO, "Staff Requirements - Discussion of Severe Accidents - Policy Statement and Research Plan...." In our review, we had the benefit of a Subcommittee meeting held on December 21, 1982. The Committee has commented on earlier drafts of this SECY paper in reports dated February 8, 1982 and September 14, 1982.

As a result of the October 25, 1982 memorandum, S. J. Chilk to W. J. Dircks, the ACRS arranged a series of three Subcommittee meetings to discuss the proposed NRC research program in support of a regulatory approach for dealing with severe accidents as described in "Nuclear Plant Severe Accident Research Plan," NUREG-0900. The first of these meetings was held on December 21, 1982. You may recall that in our report of August 18, 1982 on NUREG-0900 and in our report of September 14, 1982 on SECY-82-1A, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," dated July 16, 1982, we expressed a number of concerns about what we considered to be the lack of a coherent and workable approach to dealing with severe accidents in the licensing of new plants and in the regulation of existing plants. We concluded that we could not judge the appropriateness or the adequacy of the research program without having examined one or several feasible approaches to which a research program could be related.

With these comments in mind, we requested the NRC Staff to present, during the Subcommittee meeting of December 21, 1982, whatever additional information had been developed on approaches to deal with severe accidents. We were surprised when we were informed that SECY-82-1B was, in the Staff's view, what the Commission is likely to adopt as its policy. The substance of SECY-82-1B is, so far as we can see, little different from that of SECY-82-1A.

In its statement of purpose, the policy statement is said to have been "revised to reflect Commission and ACRS comments." In our opinion, the policy statement of SECY-82-1B does not reflect the comments we have made in previous reports to the Commission.

Our comments on the various drafts of SECY-82-1 are summarized below together with some further recommendations. Additional details can be found in the attached excerpts from several previous ACRS reports.

1. As we understand the proposed policy, judgment as to whether an applicant for a license has dealt appropriately with severe accidents will depend heavily on the results of probabilistic risk assessment (PRA). Decisions will involve comparison of the results of the PRA with the numerical guidelines suggested in Revision 1 of "Safety Goals for Nuclear Power Plants," NUREG-0880. There is general agreement that large uncertainties exist in our ability to predict both the probabilities and the consequences of severe accidents. Furthermore, there is no generally agreed upon method for comparing the results of a PRA with the guidelines given in NUREG-0880.

2. For existing plants, it appears that some as yet undefined set of plant specific and generic PRAs will be used to draw generic conclusions about groups of plants. An effort will then be made to draw conclusions about specific plants. The process to be used is not yet defined, nor is it clear what methods will be used to define it. Indeed, we observe that experience gained with PRAs suggests that it may be inappropriate to use generic results in the evaluation of individual plants.

3. No specific guidance is given as to an appropriate balance between prevention and mitigation of severe accidents. Except for some rather general comments about the need to explore the behavior of containment systems, and some equally general comments about filtered vented containment systems and core retention devices, mitigation is largely ignored. It appears that in principle, under the proposed policy, only an appeal to prudent engineering practice or the use of ALARA in risk reduction could be used to generate containment specifications, for example, and requirements for other mitigation systems important to public health and safety.

4. We have in several reports expressed reservations about a strong dependence on PRA alone in decisions dealing with severe accidents. We note, however, that the Commission policy as expressed in SECY-82-1B would use PRA as a principal criterion in detecting and correcting weaknesses in design. We recommend that before issuing a policy statement on severe accidents, the Commission give consideration to the possibility of including more specific directions for systems or approaches for dealing with severe accidents. As examples we suggest:

(a) A statement that effort will be made to specify the performance of containment systems including subsystems for heat removal. It may not be feasible to do this at present, but an effort to do so can guide research that may be needed to determine if it is possible.

(b) Specifying improved performance for decay heat removal systems.

(c) Giving direction to a licensee that a plant design must include specific consideration of features to decrease the probability of damage from sabotage.

It appears to us that because of the close relationship that must exist among a safety goal, a policy on severe accidents, and a siting policy, a much more integrated approach is needed.

We recognize the considerable effort that has gone into the various drafts of SECY-82-1 and associated documents. We understand that the task is difficult. We nevertheless consider SECY-82-1B to be seriously flawed.

Sincerely,

J. J. Ray
Chairman

Attachment:
List of Relevant Comments from Previous ACRS Reports

References:
1. SECY-82-1B from W. J. Dircks, Executive Director for Operations, to NRC Commissioners, Subject: Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation, dated November 24, 1982.
2. SECY-82-203A, from W. J. Dircks, Executive Director for Operations, to NRC Commissioners, Subject: Revisions to Nuclear Plant Severe Accident Research Plan, NUREG-0900 (Draft), dated August 30, 1982.

Attachment to January 10, 1983 ACRS Report on SECY-82-1B - List of Relevant Comments from Previous ACRS Reports

- "We believe that, before embarking on the course proposed for future CPs in SECY 82-1A, a concerted effort should be made by the NRC Staff and the ACRS to develop policy guidance on as many of the relevant safety issues as are tractable, and to propose an alternate approach to the Commission in which such policy guidance is provided to applicants for future standard plant designs." (Ref. 5, p. 2)

- "With regard to existing plants, we believe it would be productive for the NRC Staff to draft alternate positions on the most significant safety issues and to establish what would be needed in order to evaluate the alternatives." (Ref. 5, p. 3)

- "Neither the original nor the revised version of NUREG-0900 contains a delineation of an approach for dealing with severe accidents. This is needed to judge the appropriateness of the proposed research program. We continue to urge that the work necessary to provide one or more approaches be carried out. We look for requirements that might be placed on components or systems required to deal with severe accidents, description of what is now known about these, specifications of what, if any, information is required to describe system performance with the necessary accuracy, some indication of whether the information can be obtained from research in the time and with the resources available, and what research is planned to obtain the needed information." (Ref. 3, p. 1)

  "As an example, we note that, in the draft Implementation Plan for Safety Goals (July 16, 1982) provided to us, the NRC Staff concludes that it is not now feasible to specify the performance of containment systems. The NRC Staff further expressed an opinion that the information and approach needed for such a specification should be developed. We, therefore, looked at NUREG-0900 for a description of what new information is needed to specify performance of the various kinds of containments and containment systems now in use or proposed. Although there are elements of the program that could certainly contribute to more accurate specification of containment performance, we find no systematic descriptions of what information is needed or what part of the proposed program is designed to provide the information." (Ref. 3, p. 2)

- "We recommend that alternate containment performance criteria be developed and evaluated for existing nuclear power plants as part of the trial implementation program. A separate set of alternate trial containment performance criteria should be developed and evaluated during the trial period for plants yet to be designed." (Ref. 4, p. 2)

- "With regard to future plants, we believe that the NRC should examine and evaluate the safety-related changes now proposed or underway for LWRs in countries like France, the Federal Republic of Germany, Japan, Sweden, and the United Kingdom before arriving at its own judgment on what is appropriate for the U.S. For existing nuclear power plants, it is premature to assume that the available PRAs provide a generic basis for decision-making. On the contrary, despite their uncertainties, the PRAs indicate the existence of important plant-specific differences which need to be factored into the formulation of policy. Again, the specific backfitting approaches currently underway or contemplated for LWRs in other countries should be examined and evaluated for their relevance to U.S. policy." (Ref. 5, p. 3)

- "In our recent reports specific attention was called to the need for organizing the research under this Decision Unit to answer questions likely to arise in connection with the Commission's stated intention to modify the licensing process to take specific account of accidents more serious than those generally identified as Design Basis Accidents." (Ref. 2, p. 9)

  "However, there is still a lack of definition of even one approach to deal with the severe accident issue. Considering the difficulty of the problem, effort should probably be made to define several alternatives." (Ref. 2, p. 9)

- "We find that the NRC program, as proposed, is not responsive to [previous] recommendations [that funding be reallocated to provide the information needed for the severe accident rulemaking]. The programs ... should be restructured so that the primary priority is to provide the information needed for decision-making concerning features to mitigate the consequences of accidents involving severe core damage or core melt, for reactors in operation and under construction and for reactors yet to be designed. This would allow the elimination of a substantial portion of the longer-term experimental and code development work." (Ref. 1, p. 9)

- "A focused priority effort is needed with respect to risk contributors such as seismic events, design errors, operator errors of commission, sabotage, and systems interactions to provide a methodology suitable for incorporation into PRAs on a trial basis or to identify and evaluate sources of uncertainty which make this impractical and to suggest regulatory approaches in light of these uncertainties." (Ref. 2, pp. 8-9)

- "Insofar as feasible, all accident initiators and risk contributors (other than sabotage) should be included in PRAs and in benefit/cost analysis. If the uncertainties are such as to make a meaningful

quantification for some initiator or contributor impossible, this should be documented in sufficient detail and an allocation of risk to this contributor justified." (Ref. 4, p. 3)

- "We believe that, in view of the continuing uncertainties to be expected in the art of PRA and a continuing inability to satisfactorily treat all initiators and other contributors to core melt frequency, and in view of the potentially very large differences in release magnitudes among different core melt accidents, containment performance design objectives are needed and should be developed expeditiously." (Ref. 6, p. 5)

Related ACRS Reports:

1. "Review and Evaluation of the Nuclear Regulatory Commisson, Safety Research Program for Fiscal Year 1983," NUREG-0864, dated February 1982
2. "Comments on the NRC Safety Research Program Budget for Fiscal Years 1984 and 1985," NUREG-0875, dated July 1982
3. "ACRS Comments on Nuclear Plant Severe Accident Research Plan," NUREG-0900 (Draft), dated August 18, 1982
4. "ACRS Report on the Draft Action Plan for Implementing the Commission's Proposed Safety Goals for Nuclear Power Plants," dated September 15, 1982
5. "ACRS Report on SECY 82-1A: Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," dated September 14, 1982
6. "ACRS Comments on the NRC Staff Questions to the Commission Concerning the Policy Statement on Safety Goals for Nuclear Power Plants," dated September 15, 1982

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 18, 1983

Mr. William J. Dircks
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

SUBJECT: ACRS REPORT ON THE ACCIDENT SEQUENCE PRECURSOR STUDY AND THE USE OF OPERATIONAL EXPERIENCE

During its 276th meeting, April 14-16 and its 277th meeting, May 12-14, 1983, the Advisory Committee on Reactor Safeguards reviewed some of the recent work and proposed programs related to the evaluation and use of operational experience, including the possible identification of accident sequences likely to lead to core damage. In addition to the report, NUREG/CR-2497, "Precursors to Potential Severe Core Damage Accidents: 1969-1979" (also known as the ASP Study), the Committee had the benefit of discussions with representatives of Oak Ridge National Laboratory (ORNL), Science Applications, Inc. (SAI), the Institute of Nuclear Power Operations (INPO) and the NRC Staff. Subcommittee meetings on the subject were held on February 9 and March 9, 1983. The latter meeting also included a summary report by a representative of the Electric Power Research Institute (EPRI) on an EPRI workshop held on the subject on February 28 - March 1, 1983. The Committee also had the benefit of the documents referenced. The Committee undertook this review at the request of the Office of Nuclear Regulatory Research.

The Accident Sequence Precursor Study

The ASP Study, which was performed by ORNL and SAI for the NRC, was initiated largely in response to a recommendation by the NRC's Risk Assessment Review Group to try to use operational experience to improve reactor safety. The ASP Study represents an interesting and useful first effort to categorize and numerically rank the significance of operating experience reported in Licensee Event Reports (LERs). It endeavors to apply the methodology of probabilistic risk assessment (PRA) to such reported experience in a pioneering attempt to quantify an area of investigation previously treated primarily by qualitative means. To this extent, it appears to be a worthwhile effort to reduce dependence on subjective evaluation, but it is not without shortcomings.

The original objectives were consciously vague. However, with the completion of the first phase of the work and the ensuing review and discussion, the purpose of the ASP program as now defined by the NRC Staff is to develop

and apply a formalized, systematic methodology for the evaluation of nuclear plant operational experience data. This methodology will be used to identify precursors to potentially serious accidents or incidents and to estimate their occurrence frequency. The specific objectives include the following:

- identification of strengths and weaknesses of existing PRA studies by the use of operational experience;
- interpretation of operational experience in terms of margins to core damage;
- evaluation of average, industry-wide risk as it emerges from operational experience.

Thus, the NRC Staff views the ASP program as complementary to the nuclear plant PRAs currently being performed by industry and by the NRC.

The contents of the ASP Study (NUREG/CR-2497) can be divided into three parts:

- definition and use of a screening process to select "significant" event sequences from the large mass of LERs available for the 1969-1979 decade;
- quantitative ranking, based on a definition of probabilistic margins to severe core damage, of those events judged significant for the above period;
- use of the probabilistic margin to core damage to formulate an experience-based, "industry-wide" estimation of severe core damage likelihood as it existed in the same 1969-1979 period.

The data base upon which this first phase of the ASP program was based was composed of approximately 19,400 computerized abstracts of LERs dated 1969-1979. A reading and selection process guided by written criteria and knowledgeable judgments was used to reduce this base of short abstracts to a group of 529 event reports before examining the complete documentation which composes each report. Given the superficial quality and scope of most LER abstracts, it is to be anticipated that a significant number of events were probably overlooked during the event selection process. The study estimated that at least 10% of the significant events were missed.

With regard to the screening process, the ASP Study concentrated on the identification of sequences containing multiple failures and resulting either in the complete loss of one function required to prevent core damage or in the degradation of more than one such function.

All available LER documentation was reviewed for the group of 529 events selected and a determination based on written criteria was made which led to the identification of 169 events as precursors to potential severe core damage. The events finally selected were categorized and mapped onto standardized (generic) event trees that described the sequence of possible mitigating actions or events following a given transient or accident (e.g., loss of offsite power, occurrence of a small LOCA, loss of feedwater, or steam line break). The event trees chosen were not plant-specific but did represent likely transient or accident situations which could have been affected by the identified precursors.

Initiating event frequencies, failure time intervals, and functional failure probabilities were developed where practical from information contained in the full LER documentation itself. Quantitative ranking of the postulated events was then performed by evaluating the conditional probability of severe core damage given the event considered. The parts of an event sequence that actually occurred were treated deterministically in this evaluation, whereas the parts that did not occur were treated probabilistically, but with an updating of the failure rate information available from prior PRA studies with the data contained in the selected event sequences. The top 52 events had conditional probabilities of 10^{-3} or greater and were selected as significant precursors for the purposes of the Study. Credit was not given in the standardized fault trees for plant-unique systems or differences, or for possible alternative pathways or processes that might have influenced the outcome. This is a recognized shortcoming.

An industry-wide, severe core damage (SCD) probability estimate was derived in the ASP Study by summing the conditional probabilities of SCD for the selected events and dividing the result by 432, the number of reactor-years of operation over the time period included in the Study.

A good deal of controversy has arisen concerning both the methodology and the numerical results obtained for the precursors quantified in NUREG/CR-2497. The authors of the report have recognized that their methodology, as employed, is subject to possible "overcounting" and have estimated that this might make their probabilistic results high by a factor of up to three. Other methods of analyzing the data have been suggested but none appears to be unequivocally "the right one" and the actual degree of "overcounting" remains difficult to quantify.

INPO applied the same methodology to the "significant" precursors identified by ORNL but made different interpretations of the data and employed detailed event sequences that frequently differed from the ASP Study with regard to available mitigative features. INPO obtained average numerical results which are substantially lower than those in the ASP Study, as well as a different ranking of the events in order of significance. Representatives of the NRC Staff reported a partial list of results that tended to support the ASP Study in some cases and the INPO Study in others.

Some of these differences will be reduced or will disappear as part of the continuing dialogue. However, it is clear that subjective opinion plays a considerable role in the ASP Study methodology, as it does in PRA in general, and large differences are likely to remain even if the same detailed event trees are employed by each group.

There exists a considerable school of thought in the industry that ASP Studies should be performed using plant-specific methods and data. This suggestion has its attractions. However, suitable plant-specific PRAs do not exist in sufficient number, and plant-specific data will be sparse. Thus, the resources required for event-significance quantification on a plant-specific basis would be far larger than those used in the ASP Study. Also, even plant-specific studies would be forced to rely on generic data in many instances, and there does not appear to be a universally accepted recipe for this procedure. Hence, there appears to be merit in both plant-specific and generic interpretation of operational experience, each properly executed.

It is noted that neither the ASP Study nor the INPO Study included an analysis of the uncertainty in the reported results or an assessment of their sensitivity to assumptions made in the analysis. This lack can be understood in a status report. However, we believe that future efforts along this line should include a careful evaluation of such uncertainties and sensitivities.

It should be noted that the authors of NUREG/CR-2497 were not trying to predict the future risk in making an estimate of the average likelihood of core damage; rather, they were trying to evaluate the existing risk during the years 1969-1979. Hence, improvements in safety made since 1979 are not reflected in the report. Because of differences in the methodology used and because each plant has its own safety characteristics (which may differ from the generic event trees used in the ASP Study), the results of the ASP Study should not be directly compared with those of the PRA for a specific plant.

The INPO SEE-IN Program and the Work of the NRC Office of Analysis and Evaluation of Operational Data (AEOD)

The ACRS reviewed briefly the INPO SEE-IN program and the work of AEOD in order to better understand their relationship to the ASP program. Both the SEE-IN program and the work of AEOD are largely qualitative and focus on an examination of specific operational events (or sets of similar events), screen these events for significance, and strive to extract lessons which can be used directly in improving safety. The efforts of both groups have been productive and both groups continue to improve their methodology.

Both efforts are needed, since they serve different users and provide somewhat different perspectives.

Some Concluding Comments and Recommendations

1. It was reasonable for the first phase of the ASP Study to develop somewhat specific objectives as the work progressed, since it was exploratory research. The future ASP program is expected to have better defined, albeit still flexible, objectives.

2. There is a class of information which the AEOD and SEE-IN programs may be treating only in part and which the ASP program largely has not considered. The ACRS suggests that further evaluation be made as to whether adequate attention is being given to operational experience in connection with matters such as the following:

   a. where improved maintenance practices are needed (e.g., the Salem circuit breakers),

   b. whether there are adequate diagnostic capabilities to indicate impending problems and how to control them,

   c. whether plant operating procedures are adequate and effective,

   d. deficiencies in engineering design, construction and application of plant systems, controls, and components,

   e. the stage at which accident consequences can be controlled most effectively,

   f. effects of plant aging on safety, and

   g. quality deficiencies that may have been overlooked.

3. The objectives of the AEOD work, the INPO SEE-IN effort, and the ASP program should be coordinated so that significant gaps do not exist when the combined efforts are considered.

4. The ASP program has highlighted deficiencies in most existing PRAs even though some of these weaknesses had been previously recognized. Examples include operator errors of commission and complex scenarios such as the Browns Ferry fire. Strong interaction in both directions between PRAs and the ASP program is needed in the future. Much more work is needed before quantitative results from the ASP program can be considered to be meaningful. However, there is merit in continuing to attempt to obtain average estimates of risk, as well as a quantitative ranking of event significance, by analyzing operational experience.

5. Some attention should be given to the following methodological issues:

   - Single failures should not be discarded automatically, as single failures of high frequency might be symptomatic of bad maintenance practices, or plant vulnerability to "external events" and other common causes or "bypass mechanisms." In particular, the screening process should be altered so that the ASP program includes single failures of on-line systems of special importance to safety.

   - The ASP Study considered only precursors which could lead to core damage, so that experience with containment was not a factor. Such experience would, of course, become relevant were one to attempt to evaluate public risk. The ACRS recommends that the scope of future studies be extended to cover mitigation systems such as the containment.

   - Sensitivity and uncertainty evaluation should be included as an integral part of the ASP program itself. A special effort should be made to identify the principal causes of uncertainty.

   - Since there appears to be no single, agreed-upon way to perform the quantitative evaluation of risk, either in relative or in absolute terms, the evaluation should be attempted in several alternative ways and studies made to explain the key reasons for differences which may materialize. The issue of "blending" plant-specific and generic data is one area in which one could gain beneficial insight by pursuing such a "multi-faceted" approach.

   - An evaluation should be performed and published to make clear the reasons for the different results obtained for conditional severe core damage probability by the ASP Study and the INPO critique (INPO 82-025) for the 52 significant precursors.

   - The handling of human error probability is highly subjective. It is potentially a large source of difference in the estimates developed. The matter should be specifically reviewed from several points of view, including: the usefulness of a common mode of treatment as one alternative in each such study to place similar studies on a comparable basis; the development of failure models more applicable to real occurrences; and the inclusion of errors of commission in the event trees.

6. The importance of the qualitative aspects of the evaluation of operational data and experience must continue to be emphasized. Study efforts should examine the chain of events in important incidents in terms of root cause.

7. The ACRS endorses continuation of studies of operational experience, both qualitative and quantitative.

Sincerely,

Jesse C. Ebersole
Acting Chairman

References:

1. Science Applications, Inc. Oak Ridge National Laboratory, "Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report," Prepared for USNRC, NUREG-CR-2497, Vols. 1-2, dated June 1982.

2. Institute of Nuclear Power Operations Confidential Report, "Review of NRC Report: 'Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report,'" NUREG/CR-2497, dated September 1982

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 2, 1983

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE SEVERE ACCIDENT POLICY

During its 281st meeting, August 31-September 1, 1983, the Advisory Committee on Reactor Safeguards continued its discussion with the NRC Staff of the development of a Severe Accident Policy for dealing with accidents more severe than those now analyzed as design basis accidents. This issue was discussed by the ACRS Subcommittee on Class 9 Accidents with the NRC Staff on August 23, 1983.

The focus for the discussion was a draft report developed by the Severe Accident Research Program (SARP) Senior Review Group and entitled, "Severe Accident Decisions for Existing Nuclear Power Plants," dated August 5, 1983. As indicated by its title, this report and the discussions were mostly concerned with power plants now in operation or under construction.

The report, drafted in response to a Commission request, describes on a preliminary basis "the approach (primarily deterministic, supplemented by probabilistic and systems assurance analysis), the Commission may use to arrive at severe accident decisions for existing nuclear power plants." The NRC Staff asked for comments "so that issues and the decision process can be refined, and so that any needed changes in the Severe Accident Research Program can be identified."

We encourage the NRC Staff to continue its development of this general approach. It appears to be an improvement over what has been proposed up to this time.

The NRC Staff has defined the problem in the form of a question to be answered, namely, "What changes, if any, should be made in nuclear reactor regulation to account for accidents involving core damage greater than the present design basis, including core meltdown accidents?" Although this is probably not the only way to define the issue, it is a reasonable approach.

The success of this approach depends on further, more detailed elaboration of the question, on identification of the information needed for such elaboration and on the ways in which information needed for its answer is to be developed and used in reaching a conclusion. We look forward to this further elaboration and expect to make further comments as more information becomes available.

Two key issues identified in the report are, "How safe are the existing plants with respect to severe accidents?" and, "Is additional protection for severe accidents needed or desirable?" An associated question is how the decisions are to be reached. We urge that priority be given, first to the method to be used for answering these questions, and second to information that may be needed to provide the answers. We do not view this task as primarily a research problem, especially the development of the way in which the decisions are to be made. We believe that some development of decision criteria, and the identification of needed information, at least on a tentative basis, are necessary before one can define any extensive research program.

We also agree with what appears to be a tentative conclusion that the decision process must include both probabilistic and deterministic considerations. Even though the appropriate balance between the two may be difficult to define, we believe that elements of both methods will be needed for a final resolution of the problem. There is a suggestion that Systems Assurance Analysis (SAA) could become a valuable tool in the analysis and prevention of severe accidents. Although this approach may be useful for some purposes, we do not consider it an alternative to the combined deterministic-probabilistic approach. We do not recommend application of SAA to this problem.

Because of the early stage of development of the program described in the draft we discussed, we are not able to comment on its completeness or adequacy. We do have the following comments.

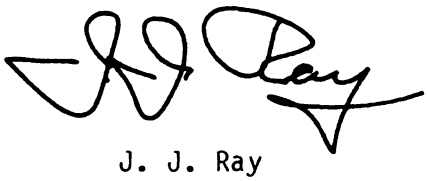
Eventually a policy must be developed for dealing with decisions involving areas of considerable uncertainty. One does not avoid this problem by deciding to use a deterministic approach. We recommend that immediate attention be given to this process. One area of uncertainty, not statistical, is the unexpected problem that may yet be encountered. This is alluded to in the report, but there are no suggestions for dealing with it. The formulation of the policy should take account of experience which has seen a continuing series of new safety issues discovered in the course of new risk analyses and other studies.

We note reference to a source term in this draft. We caution that there are many source terms, and that indeed the source term or terms used will depend markedly on the accident sequence or sequences finally chosen for analysis and decision making. Also, current or possible future changes in regulatory approach could lead to a major perturbation in the basis on which any new source terms may be applied.

The draft refers to important accident sequences. It is not clear what measure of importance is to be used. We believe that both probability and consequences should be included in judging importance and that sequences not now listed as "dominant" should be carefully screened for their possible significance.

The report implies that the severe accident issue can be dealt with for future plants without consideration of the information and the processes developed to deal with existing plants. Although we expect that there may be differences in the way in which one deals with these two classes of plants, we believe it is desirable that the approach proposed for new plants include what is learned in the development of a policy for existing ones.

Sincerely,



J. J. Ray
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 18, 1983

Mr. William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

SUBJECT: SEVERE ACCIDENT DECISIONS FOR EXISTING NUCLEAR POWER PLANTS

During its 282nd meeting, October 13-15, 1983, the ACRS met with members of the NRC Staff to continue discussion of a draft document dealing with severe accident policy development entitled, "Severe Accident Decisions for Existing Nuclear Power Plants." This report was also discussed with members of the NRC Staff during a Subcommittee meeting on October 12, 1983.

We consider the current draft to represent an improvement over the one on which we commented in our September 2, 1983 letter to Chairman Palladino. We endorse the current proposal to blend deterministic and probabilistic approaches in order to construct an appropriate regulatory framework. The Committee also recommends that a similar approach be used for future plants.

The general approach seems reasonable. We expect to study and comment further on the relationships being established among regulatory questions, policy issues, technical questions, and the research needed to provide information not currently available.

We observe that the Severe Accident Research Program was planned before this proposed approach was developed. We recommend that those managing and performing the research be asked to ensure that such research will contribute importantly to answering the regulatory questions.

We plan to continue our review of this activity.

Sincerely,

J. J. Ray
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 18, 1984

The Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON NUREG-1070, "NRC POLICY ON FUTURE REACTOR DESIGNS: DECISIONS ON SEVERE ACCIDENT ISSUES IN NUCLEAR POWER PLANT REGULATION"

During its 291st meeting, July 12-14, 1984, the Advisory Committee on Reactor Safeguards continued its discussion with the NRC Staff of the development of a Severe Accident Policy for nuclear power plants. The discussion focused on the description of the policy contained in the draft NUREG-1070 dated April 18, 1984. The report was discussed in Subcommittee meetings held on April 27 and May 31, 1984 and in the Committee's June 14-16, 1984 meeting.

The draft policy statement enunciates a policy that may be described for existing plants as follows:

> The risk from accidents more serious than the analyzed design basis accidents for nuclear power plants now in operation or nearing completion is acceptable subject to the resolution of Unresolved Safety Issues, to a decision on whether and how Safety Goals will be applied, and to the results of a source term rulemaking. Although some changes in equipment or procedures may be required after these programs are concluded, it is anticipated that these will not be major changes.

This conclusion is reached in light of:

a) the operating experience that has been accumulated;

b) the changes in required equipment, operating procedures and personnel requirements that have occurred as a result of the accident at TMI-2;

c) the results and insights gained from a significant number of PRAs for existing plants;

d) the results of studies such as the Severe Accident Precursor Program, the Accident Sequence Evaluation Program, and the Industry Degraded Core Rulemaking Program (IDCOR).

In our various meetings with the NRC Staff we have discussed the desirability of formulating some systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributors (sometimes called "outliers") that might be plant specific and might be missed absent a systematic search. Because previous experience indicates that systematic analysis using PRA or other methods may uncover such outliers, we believe that the policy statement should state explicitly that an appropriate approach will be developed and that an analysis will be made of any plant that has not yet undergone an appropriate examination. The examination should include specific attention to containment performance.

We note that resolution of the Unresolved Safety Issues will produce decisions for dealing with a number of issues that could have an influence on the risk associated with hypothesized severe accidents. Among these are:

a) an appropriate approach for assuring the reliability of decay heat removal systems, and

b) the appropriate reliability of electrical supplies for the power plant, both AC and DC systems.

There is under way a large-scale research program that concentrates on producing better information and increased understanding of the release of radioactive fission products from the reactor fuel during a severe accident, their subsequent transport outside the reactor primary pressure boundary, their behavior in the containment, and their subsequent release from the containment. We expect the results of this research to be incorporated in a rulemaking that is likely to influence the calculated risk from accidents involving severe core damage. If these new calculations have a significant influence on our present perceptions of severe accident risk, we expect them to be followed by whatever changes in regulations, plant equipment, or operational and emergency procedures are indicated.

There are a number of significant efforts on the part of the nuclear industry. Activities such as the IDCOR Program, and the organization of INPO and NSAC are industry initiatives which should lead to a decrease in the risk of severe accidents.

We support the consideration of cost-effectiveness as one of the important determinants in formulating an approach to risk management in existing plants. There will, however, always be substantial uncertainties in the calculation of risks, costs, and benefits, so that both prudence and sophistication will be required.

We believe that, taking into account the results of programs now in progress, and assuming a systematic examination of each plant, the proposed policy provides an acceptable basis for dealing with the severe accident issue for plants now in operation or under construction.

For new plants we interpret the policy statement to say that:

> New plants must meet existing regulations. They will be required to deal with the resolution of all the Unresolved Safety Issues. They will be subject to any new regulations that result from the source term rulemaking. The severe accident risk from new plants is expected to be dealt with in the foreseeable future through rulemaking for standard plants. One of the requirements of the rulemaking process will be a full scope PRA for the proposed plant. Severe accident risk will be dealt with primarily through consideration of the results and insights gained from the PRA.

We have several recommendations concerning the proposed policy for dealing with new plants. These recommendations follow:

- There should be a statement that the policy is expected to lead to new plants producing less risk than the older ones.

- The policy statement indicates that heavy reliance is to be placed on the results of the required PRA in deciding whether or not the severe accident risk associated with a proposed design is acceptable. Guidance on the required scope of the PRA and the way it is to be used are probably not appropriate to a policy statement. However, the policy statement should say that such guidance will be developed. We approve of the general approach of using a combination of deterministic and probabilistic considerations to provide the information on which a decision is to be based.

- The policy statement should speak to some balance between prevention and mitigation of risk. As a minimum, some clarification of containment performance expectations should be given. If the NRC Staff has concluded that performance criteria cannot be formulated at this time, the statement should say that such criteria or some appropriate description of expected performance will be formulated.

- The effectiveness of human performance, including that of management, has a substantial influence on risk. For this reason, we recommend that attention be given to these matters for both new and existing plants to assure that inadequate human performance at individual plants will not result in unacceptable risk. In particular, methods of analysis and associated data bases need to be developed which can properly account for both positive and negative human performance contributions.

- Although we recognize the uncertainty in dealing with sabotage, we believe the policy statement should indicate that the issue of both insider and outside threats will be carefully examined, and, to the extent feasible, taken into account in the design and in the operational procedures that are developed for new plants.

We conclude that in its present form that part of the policy statement that deals with new plants needs strengthening in the areas that we have indicated.

Additional comments by ACRS Members Robert C. Axtmann, Harold W. Lewis and David Okrent are presented below.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

Additional Comments by ACRS Member Robert C. Axtmann

Absent an urgent domestic need or a public appetite for new nuclear power plants of any design, the policy statement's emphasis on regulatory attention to future plants could be a misallocation of resources. While there may be an international market for advanced reactors, it is not clear why the U.S. public should underwrite creation of a regulatory framework that may not have domestic application for fifteen years or more. Should that schedule be foreshortened, I am confident that we can pick up where I propose we leave off in 1984.

Additional Comments by ACRS Member Harold W. Lewis

As I interpret the proposed policy statement it is that the Commission will not seek to further decrease the risk from existing reactors, unless programs now in being lead to the conclusion that it is necessary to do so. There is also a statement about new reactors, in a similar vein.

I am not persuaded of the need to issue such a statement at this time, although I share in a widespread recognition of the need to do something to relieve the apparent open-endedness of the regulatory process. I simply doubt that it is appropriate to formulate a policy on severe accidents without some sort of clarification of the overall objectives of NRC regulation of the nuclear enterprise. That the function of the Commission is to protect the health and safety of the public need not be repeated, but that in itself is hardly a guide to an appropriate level or direction of regulation. In the absence of guidance on this point, regulation has often become the objective in itself, imperfectly linked to its purpose.

The Commission has struck a glancing blow at this problem by promulgating a safety goal, for evaluation, test, and comment, but there remain open questions about the underlying rationale for the specific criteria chosen.

It would be far healthier to bring these scattershot efforts together into a coherent statement of Commission policy on an appropriate level of safety, and on NRC's perception of its role in achieving and/or maintaining that level. We said this in our letter of September 14, 1982. I believe that, however painful, an unhurried effort to generate an agency philosophy would make it far easier to deal with severe accidents and related questions in a coherent way. In the corporate world this is known as a long-range plan. A plan generated one element at a time -- safety goals, severe accidents, backfitting, etc. -- is doomed to incoherence.

The Committee letter, with which I have no serious disagreement, contains many items directed toward specifying a program plan, but avoids the questions raised here. I regret that we have thereby acquiesced in the current incoherent approach to safety assurance, and have thereby joined NRC in missing an opportunity for an integrated approach to resolve this matter.

Additional Comments by ACRS Member David Okrent

I generally agree with the ACRS letter. These comments are for purpose of emphasis or are supplemental.

1. I recommend that the Commission significantly modify the Staff's severe accident policy statement as given on the top of page 4 and again on page 15 of the April 18, 1984 draft of NUREG-1070. In this statement, it is said:

   "On the basis of currently available information, the Commission concludes that existing plants pose no undue risk to public safety and property and sees no present basis for prompt action on generic rulemaking or other regulatory changes for these plants because of severe accident risk."

   I would modify this statement both for technical reasons and for reasons of public policy. One might use something whose import was more like the following:

   "The currently available information does not lead the Commission to conclude that existing plants pose any undue risk to public safety and property due to severe accidents. However, the Commission plans to pursue a five-year program during which systematic examination of all existing plants will be undertaken by probabilistic risk assessment and/or other means to determine that there are no unacceptable large risk contributors and to help determine on a plant-by-plant (or generic) basis those safety improvements which it would be prudent to incorporate."

2. I believe that approval of the April 18, 1984 draft NUREG-1070 could easily appear to place the Commission at odds with a large segment, if not a majority, of the general public as well as many members of Congress, and with many respected individuals outside of government,

all of whom want nuclear power plants to be safer. The position of these groups seems to be supported by the several foreign countries (Sweden, France, England, Germany, Switzerland, etc.) who have adopted or propose to use safety requirements considerably more stringent than those of the USNRC. On the other hand, the proposed policy statement could be interpreted by others as an expression of satisfaction by the NRC with the status quo for existing reactors.

3. If the NRC had firm, indisputable data on the risk from each existing nuclear power plant, and it was convinced that the least safe was acceptable as is, it might be plausible for the NRC to take a position contrary to so much other opinion. However, such data do not exist. Only a limited number of plants have had good, "full-scope" PRAs, and their results are subject to large uncertainties. The existing PRAs cannot be accepted as representative of other individual plants because of the demonstrated importance of the specific features of a plant to estimates of core melt frequency and risk. Furthermore, several good PRAs have turned up one or more high probability core melt scenarios, so-called outliers, that required early remedy for the particular plant involved.

   A much less complete set of information is available concerning the performance during severe accidents of the many variants of containment design used in the U.S. For example, the ACRS has not had the benefit of a sophisticated report evaluating containment performance for some of the concepts currently in use.

4. Thus, it seems to be premature to draw so strong a positive conclusion concerning the safety of existing plants as that stated on pages 4 and 15 of the draft NUREG-1070. Furthermore, the proposed policy does not include a systematic examination of each plant, including its management and operation, for possible significant improvements in accident prevention and mitigation. I fear the overall safety posture is further aggravated by the way benefit/cost analysis has usually been used recently, namely a ratio of median estimates of benefit and cost, lacking a prudent regard for the large uncertainties. The stated intention to omit any credit for reduction in onsite losses will only worsen the situation, I fear, and may make few meaningful improvements in safety possible under the backfit rule. Since all costs are eventually born by large sectors of the public, including retirees whose annuities are linked to utility stocks among others, I find it more meaningful and equitable to balance the cost of an improvement against all benefits, offsite and onsite.

   To avoid excessive costs by being "prudent" in the inclusion of uncertainties in all benefit/cost analysis, one could include an overall limit on expenditures approved on the basis of prudence, say a few percent of the replacement cost of the plant.

5. Concerning future plants, I find the proposed criteria, in their summarized form on pages 4 and 5 of draft NUREG-1070, inadequate for the purpose. They place no emphasis on the need for effective containment. They do not set safer reactors as a goal for future reactors. They appear to place a very great reliance on the results from a PRA, despite the large uncertainties inherent in its results.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 6, 1984

Honorable James K. Asselstine
Commissioner
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Commissioner Asselstine:

SUBJECT: ACRS REPLY TO QUESTIONS REGARDING THE PROPOSED NRC SEVERE ACCIDENT POLICY STATEMENT

This is in reply to your memorandum of September 6, 1984 (a copy of which is attached). Your memorandum and the Committee report dated July 18, 1984 on NUREG-1070 were both addressed to the draft version of the Severe Accident Policy Statement dated April 18, 1984. Some of your concerns were with questions similar to some of those raised in the comments and recommendations in the Committee report, including its additional comments. Subsequently, in the proposed Federal Register Notice, Policy Statement, and the draft of NUREG-1070 dated August 23, 1984, a number of significant revisions were made to the earlier versions. Those revisions go far toward meeting some of the concerns raised in your letter, as well as in the Committee report.

In response to your particular questions, we have the following comments:

1. You ask for a qualitative judgment regarding the likelihood of an accident resulting in serious core damage before the end of the twentieth century. As you pointed out, a precise numerical estimate is not possible; but at the same time, the operational utility of any subjectively generated phrase such as, "somewhat unlikely" or "not unlikely" is limited.

Some admittedly simplistic observations may, nevertheless, be made. If one merely uses the statement in the present proposed Quantitative Safety Goal (fewer than one in 10,000 incidents per reactor year) and considers only nuclear power plants in the U.S., then, with $\sim$100 plants over a period of 15 years, one comes up with the probability estimate of 0.15. It is not really known, of course, whether the present population of plants meets this goal: by some current estimates, some existing plants fail to meet it by a factor of 3 or more, while others exceed it by a similar factor. In addition, by some estimates, the occurrence frequency of earthquakes beyond the seismic design basis is, in some cases, as large or larger than one in 10,000 per year. Such seismic hazard estimates are accompanied by very great differences of opinion among experts. Moreover, the general matter of component fragility includes many items on which subjective judgment provides the only

guidance presently available. Although there is reason to expect that some nuclear power plants can ride out an earthquake considerably larger than the seismic design basis, not all plants have yet been carefully examined on this point. We don't really know exactly how the present population stands with respect to the safety goal.

Two things must be kept in mind concerning both the general probability estimate of 0.15 and the more specific estimates of the probable frequency of some event or other obtained from probabilistic risk assessments (PRAs). On the one hand, such quantities should on no account be taken as representing an actual value, while on the other, they should not be completely ignored. The frequently reiterated attention called to the uncertainty bands associated with such numbers should be warning enough against assigning them literal significance. (For this reason, for example, their direct use in cost/benefit analyses must be treated with great caution. As a further aside, it may be worth noting that if only one quarter of the population of reactors should have, in fact, a probable frequency for core-damaging accidents four times that proposed for the Safety Goal -- a situation within the realm of possibility -- then the probability of a serious accident within this subgroup alone would equal 0.15, quite irrespective of the relative superiority of the rest of the population.) However, despite the necessarily fuzzy nature of the sort of numbers referred to, they ought not to be ignored, since they are capable of giving some indication of the state of affairs and, in many cases, the most useful indication available.

Whatever the "true" value of the probability in question may be (i.e., whether larger or smaller at this moment than the 0.15 arbitrarily mentioned above), it should be recognized that this quantity has been reduced considerably over the past five years and is likely to move toward lower values in the coming years. Without attempting to quantify this progress, it is the certain consequence of improvements in such things as operator training, plant design, instrumentation, procedures, simulator experience, and so forth, as well as of specific fixes. Of similar importance, since TMI, are the factors of increased awareness and attention to safety on the part of licensee management and the nuclear industry generally. Although most of the changes have been within the spectrum of design basis events and are not yet appreciably affected by specific consideration of severe accident problems, many severe accidents (e.g., TMI) are merely advanced stages of something which started as a design basis event. Continuing improvement in the actual accident frequency value for the reactor population will result from the programs addressed to Unresolved Safety Issues (USIs), generic issues, lessons from operational experience, and, perhaps particularly, from the correction of any major defects which may be identified in the planned plant-specific survey. All this is in addition to whatever may emerge from the Severe Accident Research Program.

Only some fraction of core-damaging accidents is expected to result in serious offsite consequences. This fraction is also not really known, though it has sometimes been pictured as being about one-tenth. The studies of containment performance may reduce this fraction over the coming years. The Source Term Program will affect the points considered

here, since it is important to estimates of the severity of consequences of such accidents as may occur.

In conclusion, and as an essentially intuitive judgment, we would consider it reasonable to suppose that, with our present (and proposed) program and a population of ~100 reactors, the probability of a core-degrading accident between now and the end of the century may lie in the neighborhood of one in ten -- within a factor of a few either way. Whether this should be phrased as "unlikely," or "not highly unlikely," or just how, we must leave to you.

2. Whether the occurrence of a severe accident within the next 16 years will "have very serious adverse consequences on public confidence" would probably depend (a) on the timing, (b) on the extent to which public opinion makers and policy makers can and do mature in their understanding of the risk from nuclear power, especially in its relationship to risk in general, and (c) on the nation's perceived need for nuclear energy. While we have no special expertise in evaluating anticipated public reactions, we would believe that another TMI-type accident in 1985 would have a strong adverse effect, whereas one in 1995 would probably cause much less reaction.

While the Commission should take responsible actions to assure the safety of nuclear power plants, it should limit its actions to those things required to provide an appropriate balance between risk and cost. The Commission should take the responsibility for determining how safe plants need to be and for evaluating the performance of plants to assure that they operate at or above that minimum level of safety. The Commission should not require expenditures of ratepayers', investors', or taxpayers' resources solely to improve the public perception of the safety of nuclear power. That responsibility must lie with the nuclear utility industry, or if it becomes a question of national policy, with the Congress. The Commission should assure itself that it has the necessary resources to fulfill its responsibility to assure an appropriate level of safety in nuclear power generation and take a clear, steady path to that goal.

The Commission's main responsibility with respect to public perception is to establish and maintain a strong reputation as a responsible, capable, independent regulatory body. It should not accept industry proposals without its own careful review and evaluation. Nor should it allow the impression, by acceding to unfounded criticism, that nuclear power risks are higher than they really are. It should, of course, listen to all.

We do not suppose that the matters of public confidence or public perception should be specifically discussed in the Policy Statement (although it is alluded to at the end of Section B.2 of the Policy Statement), but the Commission cannot escape having to take these views into account in all its activities. However, the intent of the Commission to "take all reasonable steps" (as spelled out in your "objective" statement) should be unmistakably clear. We have no doubt that this is the intent, but it does have to be read into the Policy Statement. Even though the presence or absence of some explicit phrase expressing the

intent might not significantly affect what may actually happen, it would serve the purpose of clarification and providing perspective. We believe that such an expression would improve the Policy Statement. As to where it might most appropriately appear, it would seem possible, since the second sentence of the Introduction is somewhat out of context in the present text of the Policy Statement, that the Introduction could start with this second sentence and be followed by something similar to your phrasing of the overall intent. This should be followed by a second paragraph discussing the documentation aspect of things -- as presently laid out.

3. The occurrence of a severe accident is, of course, more likely at some plants than others. In some cases, this may be attributable to deficiencies in systems or layout or procedures: things which may, for example, be identified through the proposed plant-by-plant survey. In all cases, management commitment and attention to safety are critical.

The quality of operation is not the same at all plants. The ability of operators, the quality of equipment maintenance and testing programs, the comprehensiveness of technical support, and the overall ability of plant management are important considerations in operation. There does not exist, at present, satisfactory means for fully evaluating these factors. The NRC should give high priority to programs that will improve, or encourage the utilities to improve, the quality of operation. The Institute for Nuclear Power Operations (INPO) has, and should continue to have, a key role in this area. But, it must be realized that the INPO role will be in assisting the utilities toward improving quality and there will remain a responsibility for the NRC to monitor and evaluate performance. This probably cannot be done effectively by strict and prescriptive methods, as can be done for hardware systems, but will require a fresh approach. Development of an effective and practical approach should be a high-priority matter for the NRC.

4. You asked what further steps could be taken. Our letter of July 18, 1984 recommended that "an appropriate approach" be developed for systematic analysis of each plant not previously analyzed. We do not recommend a full-scale PRA for each plant. So far as "focusing attention on weak performers" is concerned, if this means publishing periodically a list of rankings of weakness -- of course not. If it means having the NRC Staff direct special attention to points where it has perceived special problems -- by all means. The NRC Staff is not in a good position to assess weak management, or to do suitably constructive things about it when it is identified. This is an area where the INPO has a strong interest, as well as being in a much better position to be effective.

A second broad step, relating to potential vulnerabilities or inadequacies in design, is under way by such work as is being conducted under USI A-45, "Shutdown Decay Heat Removal Requirements." This work appears to have possibilities for acceleration and that would be desirable. "Assured core cooling" is a keystone goal of all regulatory activities, and it would seem that design improvement in some specific plants or nuclear steam supply systems may be desirable. For example, consideration might be given to possible upgrading of some of the General Design Criteria, such as Criterion 34 - Residual Heat Removal. The question of

the relative merits of redundancy in components versus diversity in design is complex and worthy of attention.

This is a further step, which would require time, thought, and effort to accomplish. The promotion and assurance of safety are the principal reasons for NRC's regulatory activities including inspection, enforcement, etc. Though not its only objective, the achievement and maintenance of safety is equally essential to the nuclear industry. The full recognition of this throughout the NRC Staff and all branches of the industry, along with such care in the conduct of NRC's activities that it becomes apparent to those others affected that this is the compelling end in view, could provide the basis for a common, cooperative, and even enthusiastic commitment to safety, and to excellence in performance on which safety finally depends. This step, or even progress in this direction, would be more important than any of those spelled out in the Policy Statement, or elsewhere in this letter.

5. We believe the present Policy Statement does not overemphasize the use of cost-benefit analyses and PRA. It does not, and we do not, recommend that the results of either of these types of analyses be used to yield automatic decisions based on numerical answers. Both cost-benefit analyses and PRA are useful tools that may furnish necessary inputs to engineering judgment, but as the ACRS noted in its letter of July 18, 1984, prudence and sophistication will be required in their use.

We question whether there is any such thing as qualitative engineering judgment. Engineering is by its nature a quantitative art and the issues to be addressed by a severe accident policy cannot escape having a major quantitative component. Until something better comes along, cost-benefit analysis and PRA are the best means for addressing these inherently quantitative questions. The important uncertainties in analytical results are implicit in the issues, and are not created by the methods of analysis.

6. You asked if the possible need for major changes should be made clearer. This point has been much improved in the present versions: it appears in Paragraph 3 of the Federal Register and in Paragraph 1 and Section C.1, Paragraph 3, of the Policy Statement. We recommend further changes. For example, the wording used in the "Commission concludes" sentence in the Federal Register would be preferable to that used in the analogous sentence in the Policy Statement, Section C, where the word "dismisses" has an unfortunate air. It would also be helpful (in these same two locations) if the whole content of the "ongoing programs, USIs, etc.," paragraph were to follow immediately (in the same paragraph) after the "Commission concludes" sentence; while the whole discussion concerning needs for and plans for plant-specific surveys was put in a following paragraph. This would strengthen the point that if major changes appear to be needed the Commission will require them.

7. Resolution of USI A-45 is important in consideration of a Severe Accident Policy in that it could be one of the most significant programs the Commission has under way to reduce the likelihood of occurrence of a severe accident. The present A-45 program is, as you suggest, "a broad

based program...." The resolution of A-45 may permit, however, "only... procedural changes to improve the reliability of existing decay heat removal systems..." in cases where only that sort of improvement is necessary. There is a broad range in reliability and capability of existing decay heat removal systems and the improvements required will account for that. Backfits are likely to be quite plant-specific.

Additional comments by ACRS Member Harold Lewis, Member David Okrent, and Members Harold Lewis, Dade Moeller, and David Ward are presented below.

Sincerely,

David A. Ward
Acting Chairman

Additional Comments by ACRS Member Harold Lewis

This is a generally excellent letter. I only wish to emphasize that uncertainties in quantitative estimates of probability go in both directions, and that there are those who believe that seismic risk, singled out in this letter for its large uncertainty, is a case of risk overestimation. Many of the PRAs used in our estimates were performed in a regulatory climate, for a regulatory purpose, and are infected with unquantified conservatisms. PRAs must be done realistically to earn credible use in a predictive role. All of our numbers should be read with this in mind.

Additional Comments by ACRS Member David Okrent

In its answer to question 1, the ACRS has been careful to identify the large uncertainties inherent in any estimate of average core melt frequency for the entire population of U.S. reactors.

However, in view of factors such as the considerable number of plants designed before 1970, the frequency with which the NRC best estimate of core melt frequency for existing full-scale PRAs exceeds one in ten thousand per reactor year, and the fact that even the better PRAs are incomplete, my own intuitive judgment is somewhat more pessimistic than that given by the ACRS, say by a factor of about 3. I would say that a core melt accident by the year 2000 is not unlikely.

Additional Comments by ACRS Members Harold Lewis, Dade Moeller, and David Ward

If the Commission becomes preoccupied with the issue of public perception of safety, it might shirk its responsibility to lead and to make definitive judgments about nuclear power safety. It has ready access to talents and resources about nuclear safety that the public does not, and that the Congress does not. It should use those resources. While the

Congress and the public properly have the right and power to make ultimate decisions, neither body has the ability to provide expert leadership. That is what the Commission must do.

Attachment:
Memorandum from James K. Asselstine, NRC Commissioner, to Jesse C. Ebersole, ACRS Chairman, "Severe Accident Policy Statement," dated Sept. 6, 1984

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE
COMMISSIONER

September 6, 1984

MEMORANDUM FOR: Jesse C. Ebersole
Chairman, ACRS

FROM: James K. Asselstine

SUBJECT: SEVERE ACCIDENT POLICY STATEMENT

On July 10, 1984, I wrote to you requesting the Committee's comments on several questions related to the staff's proposed Severe Accident Policy Statement. Following my memorandum, you and several other members of the Committee met with me to discuss my questions regarding the policy statement. In light of our discussion, we agreed that it would be useful for me to refine and clarify my questions to the Committee. Accordingly, I would appreciate the Committee's thoughts on the following:

1. It seems to me that it would be helpful to the Commission, in considering the adequacy and completeness of the Severe Accident Policy Statement, to have at least a general understanding of the likelihood of the occurrence of a severe accident in the foreseeable future, assuming the continuation of our present regulatory approach and program. I would therefore appreciate the Committee's views on the likelihood, given our present understanding of current LWR technology, the present capability, qualifications and attitudes of the utilities operating nuclear powerplants and our present regulatory program and approach, that an accident involving large-scale degradation or melting of the reactor core will occur between now and the year 2000. I should emphasize that I am not asking for some precise, numerical probabilistic judgment, which I think we agree is not possible. Rather, I am interested in a more qualitative judgment, such as whether such an accident is highly unlikely, somewhat unlikely, not unlikely, or highly likely, given our present knowledge of the factors listed above.

2. It seems to me that the occurrence of a severe accident (i.e., an accident involving large-scale degradation or melting of the reactor core) within the next 16 years would have very serious adverse consequences on public confidence both in our regulatory program and in the safety of the plants, even if the actual immediate health effects from the accident were limited. Would the Committee agree, and is this a factor that the Commission should consider in its evaluation of the

proposed Severe Accident Policy Statement? For example, would it be useful, in light of your answer to question one, to include in the policy statement a statement to the effect that it is a fundamental objective of the Commission's severe accident policy to take all reasonable steps to minimize the occurrence of a severe accident involving large-scale degradation or melting of the reactor core and to mitigate the consequences of such an accident should one occur?

3. Does the Committee believe that the occurrence of a severe accident is more likely at some plants than at others? If so, what are the significant contributors at the plants where a severe accident is more likely? In particular, how significant a factor is the licensee's management and operating capabilities and attitudes, and is a severe accident more likely at a plant with weak management and operating capabilities and attitudes?

4. If a severe accident is not unlikely during the next 16 years, what further steps could be taken, beyond those described in the proposed Policy Statement to minimize the occurrence of such an accident or mitigate its consequences? I note, for example, that the Committee's July 18, 1984, letter to the Commission seems to recommend a systematic examination of each plant. Should the Commission require a plant-specific PRA for each plant? What other measures might be useful? Would focusing attention on weak performers in the areas of management and operating capabilities and attitudes be helpful?

5. Cost-benefit analysis would play a key role under the proposed policy statement in considering further measures to address severe accidents. Does the policy statement over-emphasize cost-benefit analyses and PRA as a decision-making tool? Given the substantial uncertainties in calculating risks, costs and benefits in this areas, should the policy statement emphasize the use of more qualitative engineering judgment?

6. The proposed policy statement includes some statements that could be read to mean that major hardware changes will not be needed to address severe accidents for existing plants. Would it be more accurate to say that we cannot now identify one or more specific changes which are necessary and appropriate on a generic basis, but that such changes are possible depending upon the outcome of our severe accident research program and cannot be ruled out at this time?

7. How should the resolution of unresolved safety issue A-45, decay heat removal, fit in with the Severe Accident Policy Statement? For example, should the program to resolve A-45 be a broad-based program that examines in detail alternatives such as a new, independent decay heat removal train, or should

it be a limited program aimed only at procedural changes to improve the reliability of existing decay heat removal systems in the plants? Should this be discussed in the policy statement?

I greatly appreciate the Committee taking the time to consider these questions. I also want to express my appreciation to you and the other members of the Committee who met with me to discuss this subject. I found that meeting to be of exceptional value to me, and I hope that we can have similar meetings in the future on other subjects of mutual interest. This type of dialogue can only improve the quality of the Commission's decisions.

cc: Chairman Palladino
Commissioner Roberts
Commissioner Bernthal
Commissioner Zech
SECY
EDO

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 17, 1975

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Anders:

Subject: COMBUSTION ENGINEERING STANDARD SAFETY ANALYSIS REPORT - CESSAR-80

At its 185th Meeting, September 11-13, 1975, the Advisory Committee on Reactor Safeguards completed its review of the Application of Combustion Engineering, Inc. for a Preliminary Design Approval (PDA) for its Standard Reference System-80, Safety Analysis Report CESSAR-80. Subcommittee meetings were held with representatives of the Applicant, and the Nuclear Regulatory Commission (NRC) Staff in Windsor, Connecticut, on February 28 and March 1, 1975, and in Washington, D. C., on May 23 and July 25, 1975. The full Committee met with representatives of the NRC Staff and the Applicant at its 184th Meeting August 14-16, 1975. The Committee also had the benefit of the documents listed below.

The Reference System-80 design consists of the nuclear steam supply system (NSSS) with a rated core power of 3800 MW(t), the NSSS control system, reactor protection system, engineered safety features actuation system, chemical and volume control system, shutdown cooling system, safety injection system and fuel handling system. Combustion Engineering will provide, at the option of the user, certain other safety-related systems which are outside the scope of the Reference System-80 design. These non-standard systems will be dealt with in the user's Safety Analysis Reports.

The Reference System-80 has been designed for application to an envelope of plant sites which encompasses all sites approved to date for Combustion Engineering NSSS. CESSAR-80 provides seismic response spectra for all major components, and equipment and piping systems, and other information required to ensure that the balance of plant is designed to protect the Reference System-80 from all site-related hazards. Application of the Reference System-80 design will require an evaluation of each site to confirm its acceptability within the CESSAR-80 envelope. For multiple reactor units at a single station, CESSAR-80 requires that each important safety-related item of the Reference System-80 design be provided for each reactor unit.

CESSAR-80 will provide safety-related interface requirements information essential to the design of the balance of plant consistent with the assumptions used by Combustion Engineering in its accident analyses. Since the utility-applicant is responsible for instituting the quality assurance programs necessary to assure that all safety-related design requirements have been met, the Committee will review these matters in more detail with the utility-applicants on a case-by-case basis. The Committee recommends that, during the design, procurement, construction, and startup, timely and appropriate interdisciplinary system analyses be carried out to assure complete functional compatibility across each interface for an entire spectrum of anticipated operations and postulated design basis accident conditions.

The NRC Staff has identified several outstanding issues which will require resolution before the issuance of the PDA. The Committee recommends that these matters be resolved in a manner satisfactory to the Staff. The Committee wishes to be kept informed on the resolution of the following issues:

1. The emergency core cooling system evaluation.

2. The analysis of the effects of anticipated transients without scram.

3. Generic review of the effects of failures of reactor pump lubrication oil and component cooling water supply systems.

The most recent ACRS reports on nuclear power stations utilizing Combustion Engineering NSSS are the December 12, 1974 report on the application to construct the 2570 MW(t). St. Lucie Plant, Unit No. 2 and the June 10, 1975 report on the application to operate the 2570 MW(t). St. Lucie Plant, Unit No. 1. The Committee report on the 3390 MW(t). San Onofre Nuclear Power Generating Station, Units Nos. 2 and 3, selected by the Staff for reactor system design comparison with the Reference System-80 design, was issued July 21, 1972. Generic matters which include possible pump overspeed during a loss of coolant accident, transients associated with inadvertent operation of the emergency core cooling system or chemical and volume control system charging pumps, and analyses of postulated ruptures of the steam generator feed line, should be dealt with appropriately by the Staff. With regard to the rupture accident, the Committee recommends that the Staff perform an independent check on the calculation of steam generator blowdown force effects. It is expected that these items will be resolved in a manner satisfactory to the NRC Staff following the PDA and prior to the Final Design Approval (FDA). During the interim period, the Committee will continue to review these items on a case-by-case basis as well as through other appropriate Subcommittee and full Committee meetings.

The peak linear heat generation rate is reduced to 12.1 kw/ft in order to meet the ECCS final acceptance criteria of Appendix K, 10 CFR 50. The Committee recognizes that conservative restrictions used in the NRC-approved ECCS model and the use of a generalized containment envelope yielding low containment pressures may be factors contributing to the imposed reduction in the permissible linear heat generation rates. The reduced limit imposes restrictions on modes for plant operation and becomes dependent on in-core monitoring systems for verification that limits are not exceeded. The Committee recommends that for a standard reactor of this size, larger safety margins, such as obtainable from higher reflooding rates, should be demonstrated. Programs underway by Combustion Engineering, Inc., include analytical and experimental studies aimed at providing the technical base for ECCS model improvements, as well as studying possible changes involving augmented ECC systems. The Committee believes that these programs constitute a sufficient basis for proceeding at this time and that the demonstration of larger safety margins should be part of the first major revised version of the Reference System-80 design which, as stated by Combustion Engineering, Inc., is likely to be submitted for review in about two years.

The Committee needs to complete its review of the suitability of the new 16 x 16 fuel and modified core reactivity controls of the Reference System-80 design which are now scheduled for initial proof testing at Arkansas Nuclear One, Unit No. 2 and at St. Lucie Plant, Unit No. 2. The Committee also needs to complete its review of the new core protection calculator system and the computer-based core operating limit supervisory system which will be incorporated into the Reference System-80 design in the event they are successfully demonstrated at Arkansas Nuclear One, Unit No. 2. The Committee needs to be assured of the dependability of in-core neutron flux sensors for control of reactors operating at low core power peaking factors. For this purpose the Committee recommends that the Staff and the Applicant continue to gather pertinent information from operating CE reactors. The Committee will continue its review of these matters as appropriate documentation is submitted and the improvements sought can be evaluated.

The Committee recognizes the importance to safety and improved designs of developing computational methods to provide best estimate analyses of LOCA and other postulated accidents. The Committee encourages the Applicant and the NRC Staff to accelerate their efforts to this end. The Committee wishes to be kept informed.

The CESSAR-80 design should include provisions which anticipate the maintenance, inspection, and operational needs of the plant throughout its service life, including cleaning and decontamination of the primary coolant system, and eventual decommissioning. In particular, the Committee believes that the NRC Staff and Combustion Engineering, Inc., should review methods and procedures for removing accumulations of radioactive contamination whereby maintenance and inspection programs can be more effectively and safely carried out.

The Committee believes that Combustion Engineering and the NRC Staff should continue to review the Reference System-80 for design changes that will further improve protection against sabotage.

The Committee believes that methods that seek to develop reference systems through standardization and through replication need to be coupled with ongoing programs that will permit design changes to reference systems which improve safety and which, when justified, will be implemented in a timely manner. Use of reference systems should lead to more efficient and effective licensing reviews. Programs such as CESSAR-80 will contribute to this process. A transition period will be required in which the Committee will still give attention to the items noted, on a case-by-case basis.

The Committee believes that, subject to the above comments and successful completion of the R&D programs, the Combustion Engineering Reference System-80 design can be successfully engineered to serve as a reference system.

Sincerely yours,

William Kerr
Chairman

REFERENCES TO THE CESSAR-80 LETTER:

1. Combustion Engineering Standard Safety Analysis Report for System-80 (CESSAR) with Amendments 1 through 36

2. Report to the Advisory Committee on Reactor Safeguards from the Office of Nuclear Reactor Regulation, dated July, 1975

3. Supplement 1 to the Report to the Advisory Committee on Reactor Safeguards from the Office of Nuclear Reactor Regulation, dated August 8, 1975

4. Letter, dated March 24, 1975, Combustion Engineering, Inc., to DRL concerning information on the fuel transfer tube

5. Letter, dated March 10, 1975, Combustion Engineering, Inc., to DRL concerning radioiodine spiking effects on accident releases

6. Letter, dated January 15, 1975, Combustion Engineering Inc., to DRL concerning views on Anticipated Transients Without Scram

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 12, 1976

Lee V. Gossick
Executive Director
for Operations

Subject: ACRS REVIEW OF SWESSAR-P1/CESSAR-80

At its 191st meeting on March 4-6, 1976, the ACRS concluded that it would not be necessary for the Committee to make a specific review of the combined SWESSAR-P1/CESSAR-80 option since it has previously reviewed and reported on the SWESSAR-P1 and CESSAR-80 standard safety analysis reports and consideration was given to the interface between these facilities during this review. (See ACRS Reports of February 11, 1976 and September 17, 1975). The Committee plans to make a detailed review of the first application for a specific construction permit that utilizes the SWESSAR-P1/CESSAR-80 combination.

Sincerely,

Dade W. Moeller

Dade W. Moeller
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 15, 1981

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: ACRS REPORT ON FINAL DESIGN APPROVAL FOR COMBUSTION ENGINEERING, INC. STANDARD NUCLEAR STEAM SUPPLY SYSTEM (STANDARD REFERENCE SYSTEM 80)

Dear Dr. Palladino:

During its 260th meeting, December 10-12, 1981, the Advisory Committee on Reactor Safeguards reviewed the application of Combustion Engineering, Inc. for final design approval for its Standard Reference System 80 described in CESSAR. A Subcommittee meeting was held with representatives of the Applicant and the NRC Staff in Windsor, Connecticut on November 19, 1981. The Committee also had the benefit of the documents listed. The Committee's report on the preliminary design approval for this standard nuclear steam supply system (NSSS) was provided in a letter to the NRC Chairman dated September 17, 1975.

The System 80 design consists of a reactor system with a design rated core output of 3800 MWt and includes the reactor coolant system, reactor protection system, engineered safety features actuation system, chemical and volume control system, shutdown cooling system, safety injection system, and fuel handling system. The System 80 design provides safety-related interface requirements information essential to the design of the balance of plant. Combustion Engineering provides, at the option of the user, certain other nonstandard safety-related systems and services which are outside the scope of the System 80 design. Such systems will need to be dealt with in each user's Safety Analysis Report. The regulations governing the review of standard plant designs under the "reference system" option described in the Federal Register (42 FR 34395 and 43 FR 38954) are contained in paragraph 2.110 of 10 CFR Part 2 and Appendix O to 10 CFR Part 50.

CESSAR provides information required to ensure that the balance of plant is designed to protect the System 80 from site-related hazards. It envelops all plant sites approved to date for Combustion Engineering nuclear steam supply systems. When the System 80 design is applied, the related site must be evaluated to establish its acceptability within the System 80 envelope. For multiple reactor units at a single site, the reference design requires that each important safety-related item be separately provided for each reactor unit. The first plant using the System 80 design will be Palo Verde Nuclear Generating Station, Units 1, 2, and 3, of which Unit 1 is scheduled to load fuel during November 1982.

Because the utility-applicant is responsible for instituting the quality assurance programs necessary to assure that all safety-related requirements have been met, the NRC must review these matters with the utility-applicants on a case-by-case basis. The ACRS believes that Combustion Engineering should be required to evaluate the adequacy of the implementation of interface requirements, including such items as the influence of plant control system performance and reliability on NSSS integrity and function.

In recent years, the availability of reliable shutdown heat removal capability for a wide range of transients has been recognized to be of great importance to safety. The System 80 design does not include capability for rapid, direct depressurization of the primary system or for any method of heat removal immediately after shutdown which does not require use of the steam generators. In the present design, the steam generators must be operated for heat removal after shutdown when the primary system is at high pressure and temperature. This places extra importance on the reliability of the auxiliary feedwater system used in connection with System 80 steam generators and extra requirements on the integrity of the steam generators. The ACRS believes that special attention should be given to these matters in connection with any plant employing the System 80 design. The Committee also believes that it may be useful to give consideration to the potential for adding valves of a size to facilitate rapid depressurization of the System 80 primary coolant system to allow more direct methods of decay heat removal. The Committee wishes to review this matter further with the cooperation of Combustion Engineering and the NRC Staff.

System 80 employs some new design features for the steam generators, the core outlet flow region, control rod guidance and shrouding, and the core support structure. These appear to be acceptable, but, because they are new features, they should be monitored during early operation to determine if they perform as expected.

A number of items have been identified as Outstanding Issues and Confirmatory Issues. These include some TMI-2 Action Plan requirements. Progress on these matters is satisfactory, and we believe these issues can be resolved in an acceptable manner. The Committee wishes to be kept informed.

The manner of applying preliminary and final design approvals of the type proposed for System 80 will not be completely defined until System 80 has been used for several licensing actions at both the construction permit and operating license stages. The Committee believes that standard designs such as System 80 can be useful in assuring acceptably safe plants. However, a policy to establish when and how changes will be permitted to new or previously licensed plants is needed.

The Committee believes that, subject to the above comments and approval of the balance-of-plant designs, the System 80 design can be incorporated into nuclear power plants that can be operated without undue risk to the health and safety of the public.

Sincerely,

J. Carson Mark
Chairman

References:

1. Combustion Engineering, Inc., "System 80 CESSAR FSAR," with Amendments 1 through 5.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Final Design of the Standard Nuclear Steam Supply Reference System CESSAR System 80," NUREG-0852, dated November 1981.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 5, 1982

Mr. Willian J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

Subject: RELIABILITY OF THE SHUTDOWN HEAT REMOVAL SYSTEM ON THE SYSTEM 80 DESIGN

The ACRS in its December 15, 1981 report to Chairman Palladino on the Combustion Engineering, Inc. Standard Reference System 80 commented on the reliability of the decay heat removal system. These comments addressed the lack of a capability for rapid, direct depressurization of the primary system to allow feed and bleed operations and the reliance placed upon the secondary system for heat removal capability. The ACRS Subcommittee on Decay Heat Removal Systems met with representatives of Combustion Engineering, Inc. and the NRC Staff on March 16, 1982 to discuss these issues. The ACRS discussed these issues further during its 264th meeting, April 1-2, 1982.

Representatives of Combustion Engineering have defended their design, stating that:

1. The System 80 NSSS will be coupled with highly reliable emergency feedwater systems (EFWS) by addition of an interface requirement that the EFWS have an unavailability in the range of 10^{-4} to 10^{-5} per demand.

2. The System 80 NSSS is capable of achieving cold shutdown conditions using only safety grade systems even without offsite power and with an added single failure.

3. The System 80 steam generator design includes many features that will assure adequate tube integrity, minimizing concerns associated with operating reactors.

4. Even if all auxiliary feedwater supply were somehow lost, the secondary side of the steam generators could be depressurized to allow use of low head pumps which might be aligned to provide water to the steam generators from a number of sources.

5. Probabilistic analyses have not shown that installing PORVs will result in a significant improvement in safety. The added costs are not justified.

Combustion Engineering has proposed that the issues associated with the Committee's comments on the System 80 design be resolved in a continuing dialogue among the ACRS, the NRC Staff, and Combustion Engineering. It is the NRC Staff's intention to address these issues on an expeditious schedule with all applicants requesting licenses for Combustion Engineering NSSS designs which do not have capability for rapid depressurization independent of the steam generator. We concur with this approach and wish to be kept informed.

The Combustion Engineering response to the Committee's comments on the System 80 design emphasizes the expected very high reliability of the feedwater systems and the integrity of the steam generators. We believe that these are necessary goals but note that past operating experience indicates that these goals are difficult to achieve. We believe that for this reason Combustion Engineering and the NRC Staff should consider further the addition of valves of a size to facilitate rapid depressurization of the System 80 primary coolant system as stated in the Committee's December 15, 1981 letter on the System 80 design. We believe that a plan for addressing this issue should be formulated in the near future. We wish to be kept informed and to discuss this further with Combustion Engineering and the NRC Staff.

We believe that, while this evaluation should be conducted expeditiously, its resolution should not now be a condition for operation of System 80 plants at full power, or of plants having similar features. The need for future hardware or procedural changes should be contingent upon results of this evaluation.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

May 6, 1960

Honorable John A. McCone
Chairman
U. S. Atomic Energy Commission
Washington 25, D. C.

Subject: CONTROL RODS

Dear Mr. McCone:

The Committee wishes to point out that the present state of the art of reactor control rod design leaves much to be desired, in particular for application to power reactors where reactivity is increased if the rod falls downward under the influence of gravity. It is important to stress the simplicity, ruggedness and reliability of control rod drives and to insure that the position of a rod is correctly indicated even should the rod be separated from its drive. Design problems appear to be particularly difficult for hydraulic rod drives. A control-rod-fall accident should be made as nearly impossible as practical.

We believe that this recommendation should be transmitted to all reactor designers concerned.

This letter does not modify any previous recommendations with regard to operating reactors.

Sincerely yours,

/s/ Leslie Silverman

Leslie Silverman
Chairman

cc: A.R. Luedecke, GM
W.F. Finan, OGM
H.L. Price

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 6, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: NEAR-TERM CONSTRUCTION PERMIT APPLICATIONS

Dear Dr. Ahearne:

During its 241st meeting, May 1-3, 1980, the ACRS reviewed the status of applications for near-term construction permits (NTCPs). In its review the Committee had the benefit of discussions with the NRC Staff and with representatives of the applicants for the NTCPs. A subcommittee meeting on this subject was held on April 9, 1980.

The six NTCP applicants and the reactor types involved are as follows:

Black Fox Station, Units 1 and 2, Public Service Company of Oklahoma, General Electric BWR/6, Mark III pressure suppression containment

Skagit Nuclear Power Project, Units 1 and 2, Puget Sound Power & Light Company, General Electric BWR/6, Mark III pressure suppression containment

Pilgrim Station, Unit 2, Boston Edison Company, Combustion Engineering custom NSSS, large dry containment

Perkins Nuclear Station, Units 1, 2 and 3, Duke Power Company, Combustion Engineering CESSAR System 80 NSSS, large dry containment

Allens Creek Nuclear Generating Station, Houston Lighting & Power Company, General Electric BWR/6, Mark III pressure suppression containment

Pebble Springs Nuclear Plant, Units 1 and 2, Portland General Electric Company, Babcock and Wilcox custom NSSS, large dry containment

The NRC Staff has approached this matter primarily by examining the Action Plan and judging the applicability and scheduling of each item to an NTCP. This procedure has resulted in placing many important items in a category wherein the NRC has yet to develop criteria applicable to construction permit applicants. Action Plan item II.A on siting introduces questions whose resolution must be achieved prior to issuance of a construction permit.

Item II.B on degraded or melted cores bears directly on containment design, as well as other safety features. Item II.C on reliability engineering and risk assessment could bear significantly on the design requirements for many important plant systems. There are many other items in the Action Plan and in the ACRS report of April 17, 1980 which also might impact directly on important design aspects of these plants.

Mr. Harold Denton advised the Committee that he envisaged permitting construction to proceed if there are no obvious site-related questions in terms of the Report of the Siting Policy Task Force (NUREG-0625) and if the containment design pressure were such as to withstand hydrogen combustion, on the assumption that other design aspects could be changed later if so required.

The utility representatives advised the ACRS that, in their opinion, there was a need for the resolution of several policy questions which relate to how and whether construction permit applications will be processed in the near term. The utilities identified the following six policy issues as being in most urgent need of resolution:

1. Siting
2. Emergency planning
3. Degraded core conditions
4. Control room design
5. Management for design and construction
6. Reliability and risk assessment

The utility representatives recommended that a concerted effort be undertaken to develop an acceptable interim approach to resolution by the Commission of such issues in the next few months. The ACRS supports this recommendation and urges that appropriate Staff resources be made available for this purpose. An ACRS Subcommittee plans to work actively with the Staff on the topic with the anticipation that the full Committee would review the NTCP matter within a few months.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. Memorandum from D. F. Ross, NRC, to R. F. Fraley, ACRS, Subject: Transmittal of NTCP Requirements List, dated April 22, 1980.
2. Memorandum from William F. Kane, NRC, to Addressees, Subject: Request for Review of Proposed TMI-2-Related Requirements for NTCP Applicants, dated April 4, 1980.
3. U. S. Nuclear Regulatory Commission, "NRC Action Plans Developed as a Result of the TMI-2 Accident," USNRC Report NUREG-0660 Draft 3, dated March 5, 1980.
4. U. S. Nuclear Regulatory Commission, "Report of the Siting Policy Task Force," USNRC Report NUREG-0625, dated August, 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 12, 1981

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: REQUIREMENTS FOR NEAR-TERM CONSTRUCTION PERMIT APPLICATIONS

Dear Dr. Ahearne:

During its 249th meeting, January 8-10, 1981, the ACRS again reviewed the status of the requirements for near-term construction permits (NTCPs). The Committee reported to you previously on this subject in a letter dated May 6, 1980. In the present review we had the benefit of a Subcommittee meeting on January 6, 1981 and of discussions with members of the NRC Staff and with representatives of applicants for NTCPs and Offshore Power Systems, the applicant for a manufacturing license (ML).

In our letter of May 6, 1980 we noted that the utility representatives had advised the Committee that there was a need for resolution of several policy issues which related to how and whether construction permit applications would be processed in the near term. The principal policy issues identified dealt with siting, degraded core conditions, reliability and risk assessment, and emergency planning. In May 1980, the utilities expressed a desire to have the chance to propose an acceptable interim approach to resolution of these issues. However, the utilities did not present any common proposal for dealing with this matter during the next several months.

The NRC Staff did develop a proposed policy and on October 2, 1980 the NRC published for comment in the Federal Register "Proposed Licensing Requirements for Pending Construction Permit and Manufacturing License Applications." The Federal Register notice identified the following three options as having been considered by the NRC Staff.

1. Resume licensing using the pre-TMI CP requirements augmented by the applicable requirements identified in the TMI Action Plan, NUREG-0660. In effect, this treats the pending CP and ML applications as if they were the last of the present generation of nuclear power plants.

2. Take no further action on the pending CP and ML applications until the rulemaking actions described in the Action Plan have been completed. This would, in effect, treat the pending applications as the first of a new generation of nuclear power plants.

3. Resume licensing using the pre-TMI CP and ML requirements augmented by the applicable requirements identified in the TMI Action Plan, NUREG-0660, and require certain additional measures or commitments in related areas, e.g., those that would be the subject of rulemaking.

The NRC Staff favored Option 3 as a suitable compromise and identified their current positions for NTCP and ML plants with regard to siting, degraded core rulemaking, reliability engineering and emergency preparedness.

The comments from representatives of the nuclear industry on the proposed licensing requirements generally opposed the Staff's preference for Option 3, and favored Option 1. In addition to opposing additional requirements for NTCP plants, the industry representatives argued that the Staff's position concerning degraded core rulemaking was open-ended and would lead to protracted delays and case-by-case adjudication of the matter at ASLB hearings. Industry representatives provided a varied set of comments concerning reliability engineering and argued against adoption of the NRC Staff's position on siting. Offshore Power Systems favored Option 1 but stated that they believed they could live with Option 3.

During the 249th ACRS meeting, the NRC Staff advised the Committee that it now favored adoption of a revised Option 3. The new NRC Staff position was described as follows:

Emergency Preparedness

The Commission has adopted a rule which addresses this subject. The NTCP Applicants will be required to comply with this rule.

Siting

In view of the demographic and hydrological characteristics of the proposed sites, no additional measures with regard to siting would be required in connection with these construction permit applications.

Reliability Engineering

Each applicant would be required to submit a site/plant probabilistic risk assessment as part of the application for an operating license.

Degraded Core Rulemaking

In order to minimize foreclosure of plant modifications in the structural design area, at least those applicants whose designs incorporate a relatively low-design-pressure reactor containment would have to strengthen the containment structure against internal pressure. In addition, all applicants would be required to commit to making provisions for an approximately three foot diameter, or equivalent, containment penetration which could be used in conjunction with a filtered venting design feature, should the latter be judged to be needed.

We agree with the NRC Staff's currently proposed approach on siting. We also agree with the current NRC Staff position on reliability engineering. During the discussion with us, the NRC Staff indicated that, although they did not propose making a formal requirement to that effect, one intent of the proposed position on reliability engineering was to strongly encourage each applicant

to perform the relevant portions of the probabilistic assessment early enough that the results could be factored into a safety-related reliability optimization of the design. We strongly support this point of view and recommend that each applicant give high priority to such efforts.

The NRC Staff's position on the degree of containment strengthening that should be required had not yet been definitively formulated by the time the 249th ACRS meeting was held. Since the NRC Staff's position was new, industry representatives did not have time to review the position and provide comments.

Furthermore, we were advised by representatives of the Houston Lighting and Power Company, the Applicant for the Allens Creek Nuclear Generating Station, that they had authorized a study of possible accident prevention and mitigation features for their plant in order to ascertain the advantages, disadvantages, and practicality of these features. The results of this study are to be presented to Houston Lighting and Power in mid-January and representatives of the company requested an opportunity to meet with the ACRS in early February to discuss these results.

We agree with the general approach outlined by Harold Denton at the 249th ACRS meeting concerning provisions for degraded core rulemaking on NTCP plants. However, we believe that the NRC Staff needs to define its proposal more precisely. We believe that both the NRC Staff and the ACRS should have the benefit of further discussions with the NTCP and ML applicants. Hence, we recommend that the Nuclear Regulatory Commission defer any final action on the overall matter at least until after the 250th ACRS meeting on February 5-7, 1981 during which this matter is scheduled for discussion.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 10, 1981

The Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: ACRS REPORT ON REQUIREMENTS FOR NEAR-TERM CONSTRUCTION PERMITS AND MANUFACTURING LICENSES

Dear Dr. Ahearne:

During its 250th meeting, February 5-7, 1981, the ACRS again reviewed the status of requirements for near-term construction permits (NTCPs) and manufacturing licenses (MLs). The Committee reported to you previously on this subject in letters dated May 6, 1980 and January 12, 1981. In the present review we had the benefit of a Subcommittee meeting on February 4, 1981 and of discussions with members of the NRC Staff and representatives of the Houston Lighting and Power Company, Offshore Power Systems, Boston Edison Company, and the General Electric Company.

In our letter dated January 12, 1981, we agreed with the general position outlined by Harold Denton to the ACRS but recommended that a decision be deferred while the NRC Staff better defined its proposal and the Houston Lighting and Power Company was provided an opportunity to present the results of their study of the merits of possible preventive and mitigative design features for the proposed Allens Creek boiling water reactor.

During the 250th ACRS meeting, the NRC Staff presented the attached proposed position regarding requirements for NTCP and ML applicants. We have the following comments on these proposed requirements:

Item 1 - Site/plant specific probabilistic risk analysis

The current NRC Staff position is similar to the Staff position of January 9, 1981 which the ACRS supported. The new position on reliability engineering is more specific in that it would require the applicant to submit the risk assessment within two years after issuance of the construction permit and call for an NRC review at that time to determine possible requirements for preventive and mitigative actions. The criteria which would be used in this selection process have not been specified nor are they easily specified at this time. The Committee suggests that the Commission consider stating as an aim the seeking of such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant, with the intent of encouraging each applicant to take those steps which are in harmony with such an aim.

Item 2 - Dedicated penetration for possible installation of systems to prevent containment failure

This is identical to the Staff position discussed in January and has the support of the ACRS.

Items 3 and 4 - Hydrogen control measures and containment strengthening requirements

These represent a modified statement of the position proposed by Harold Denton in January to strengthen relatively low-design pressure containments against internal pressure as practical, within the existing design concept and without excessive impact. Items 3 and 4 require hydrogen control measures and pose some specific requirements with regard to minimum internal pressure capability. The ACRS believes that the NRC Staff approach in this regard is acceptable. However, while the ACRS wishes to encourage applicants to provide containment strengthening of the type proposed in Item 4 a., we believe that, if proposed by any of the applicants, modest deviations from the specific requirements should be considered on their merits.

In a letter to you dated September 8, 1980 providing additional comments on hydrogen control and improvement of containment capability, the ACRS stated its belief that each licensee should be required to perform design studies of possible hydrogen control and filtered venting systems which have the potential for mitigation of accidents involving large scale core damage or core melting, including an estimate of the cost, the possible schedule, and the potential for reduction in risk. The Committee believes that such studies should also be made by NTCP and ML plants during construction and that the final choice of hydrogen control system for each plant should be made with the benefit of such broader studies.

Sincerely,

J. Carson Mark
Chairman

Attachment:
Staff Position With Regard to NTCP Requirements With Respect to Degraded Core Rulemaking, dated 2/6/81

STAFF POSITION WITH REGARD TO NEAR-TERM CONSTRUCTION PERMIT REQUIREMENTS WITH RESPECT TO DEGRADED CORE RULEMAKING - FEBRUARY 6, 1981

1. Applicants shall commit to performing a site/plant-specific probabilistic risk assessment and incorporating the results of the assessment into the design of the facility. The commitment must include a program plan, acceptable to the Staff, that demonstrates how the risk assessment program will be scheduled so as to influence system designs as they are being developed. The assessment shall be completed and submitted to NRC within two years of issuance of the construction permit. The outcome of this study and the NRC review of it will be a determination of specific preventive and mitigative actions to be implemented to reduce these risks. A prevention feature that must be considered is an additional decay heat removal system whose functional requirements and criteria would be derived from the probabilistic risk assessment study.

2. In order not to preclude the installation of systems to prevent containment failure, such as a filtered vented containment system, the containment design shall include provisions for one or more dedicated penetrations, equivalent in size to a single three foot diameter opening.

3. Hydrogen control measures shall be provided.

4. Applicants shall provide preliminary design information at a level consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

    a. Containment integrity will be maintained (i.e., for steel containments, ASME Service Level C based on ASME code specified minimum yield values and considering pressure and dead load alone. For concrete containments, an equivalent approach based on ASME Div. 2) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon-dioxide is the inerting agent depending upon which option is chosen for control of hydrogen. As a minimum, for steel containments ASME Service Level C (based on ASME Code specified minimum yield values and considering pressure and dead load alone) will not be exceeded at an internal pressure of 45 psig. For reinforced concrete containment structures, an equivalent standard based on ASME Division 2 is satisfied at the same internal pressure. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

b. The containment and associated systems will provide reasonable assurance that uniformly distributed hydrogen concentrations do not exceed 10% associated with an accident that releases hydrogen generated from 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

c. The facility design will provide reasonable assurance that, based on a 100% fuel clad metal-water reaction, combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

d. If the option chosen for hydrogen control is post-accident inerting:

   (1) Containment structure loadings produced by an inadvertent full inerting (assuming carbon dioxide) but not including seismic or design basis accident loadings, will not produce stresses in excess of the acceptable maximum for Service Level A specified in ASME Code Section III, Subsection NE (ASME Div. 2 for concrete containments).

   (2) A pressure test of the containment at 1.15 times the pressure calculated to result from carbon dioxide inerting can be safely conducted.

   (3) Inadvertent full inerting of the containment can be safely accommodated during plant operation and demonstrated by test.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 13, 1981

Mr. William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

Subject: APPLICATION OF TMI-2 ACTION PLAN TO NEAR-TERM CONSTRUCTION PERMITS AND MANUFACTURING LICENSES

During its 255th meeting, July 9-11, 1981, the ACRS heard presentations from the NRC Staff and the Applicant regarding application of the NRC TMI-2 Action Plan items to the Pilgrim Nuclear Power Station Unit 2.

The NRC Staff has established a review team especially for post-TMI-2 issues concerning near-term construction permit (NTCP) and manufacturing license applications. The Committee believes that this approach is providing effective reviews. The Committee concluded that it has no objection to NRC Staff approval of a construction permit for Pilgrim Unit 2, subject to the conditions in its letter of October 12, 1977.

In addition, the Committee concluded that it is not necessary for the ACRS to review application of TMI-2 Action Plan items to the Allens Creek Nuclear Generating Station Units 1 and 2, although the members would like a briefing regarding the resolution of questions regarding hydrogen generation and control for the Allens Creek Mark III containment. This briefing has tentatively been scheduled for the 257th meeting of the ACRS on September 10-12, 1981.

The Committee also desires an opportunity, with respect to the five additional NTCPs and the manufacturing license, to determine on a case-by-case basis if ACRS review of changes resulting from application of the TMI-2 Action Plan is appropriate.

Sincerely,

J. Carson Mark

J. Carson Mark
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 20, 1984

Mr. William J. Dircks
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Dircks:

SUBJECT: ACRS REPORT ON DRAFT TASK ACTION PLAN - CONTAINMENT PERFORMANCE GUIDELINES - JANUARY 4, 1984

During its 287th meeting, March 15-17, 1984, at the request of the NRC Staff, the ACRS reviewed the Draft Task Action Plan, Containment Performance Guidelines (CPGs), dated January 4, 1984. In its review, the ACRS had the benefit of discussions with representatives of the NRC Staff and of a Subcommittee meeting held on March 14, 1984.

Enclosure 2 of the Draft Task Action Plan poses a series of questions to the ACRS. The following comments will tend to be structured about the Staff's questions. The essence of the ACRS comments is that the containment should be a major contributor to the defense-in-depth philosophy, including coping with accidents beyond the design basis. The ACRS favors the development of CPGs.

Question 1

Does the ACRS agree with the problem statement? If not, how should the problem be stated?

Answer

The problem statement appears to be given on the bottom of page 3 of the draft document. It does not address the issue of why CPGs should or should not be established. It is necessary to address this issue in order to decide on the form the guidelines should take.

The problem description ties the usefulness of a containment design objective to the usefulness of current methodologies for assessment of containment performance. This is an unnecessary and inappropriate restriction. Had a similar restriction been applied to the development of other currently used design criteria, many would never have been adopted.

Question 2

Does the ACRS agree with the statement of objectives? Should there be any additions or deletions? Please list them.

Answer

The list of primary and secondary objectives is fairly extensive. However, containment performance for a specific reactor will hinge on the relative frequency of all the accidents in the entire spectrum. It is not clear how (or whether) the Staff plans to address this matter and factor it into the containment assessment. For example, recent NRC Staff actions on resolution of some unresolved safety issues seem to have focused on core melt probability, rather than risk, for perspective into decision making on acceptability of a proposed resolution. Since the accident sequences most important to risk are frequently not the largest contributors to core melt frequency, we question such an approach.

The listed objectives do not include an attempt to provide assurance that containment is a major contributor to defense-in-depth. We believe that this is a very important objective.

In item g, it is stated that various containment designs are to be treated in a relatively consistent manner; however, in item h, "special credit" for specific design features is proposed. It is not clear whether these two statements are partly in conflict.

In item g, it is not clear that, "to reflect the uncertainties, one must prescribe a range of values," or, if so, how one will use such a range. Under "Selected Phenomena," achieving improved containment performance by improved containment heat removal measures is not mentioned. The reason for this omission is not clear. Finally, it is not clear whether external phenomena are to be explicitly included in the assessment.

Question 3

Six proposed alternatives are the PRA, deterministic, a combination of the deterministic and PRA alternatives, translate the Safety Goals back to compatible containment performance goals and write a set of specifications based on the familar design basis approach taking care to make the new specifications consistent with existing regulations and the approach outlined by ACRS in NUREG-0739. What does the ACRS view as the advantages and disadvantages of each approach? Does ACRS have a preferred approach? If yes, why is it preferred? Is any approach more compatible with the safety goals than others? If yes, which one? and why?

Answer

We question whether using PRA alone will be workable, partly because of large

uncertainties, partly because of differences of opinion to be expected among different groups analyzing the problem.

Developing containment guidelines solely from the quantitative safety goals suffers equally from a strong dependence on PRA. It also might permit a situation where containment would not be needed because the safety goal was met by a combination of melt prevention and emergency procedures.

We question whether an approach which uses only existing regulations and past experience can provide a basis for deciding on containment performance requirements for severe accidents.

During the Subcommittee and Full Committee meetings, the NRC Staff said it had considered the matter further and had narrowed the field down to four alternatives which they labeled as follows:

1. probabilistic - threshold model
2. probabilistic - leak-before-break model
3. deterministic - figure of merit
4. deterministic - specification of performance criteria

We question the first two, if they are to rely solely on PRA. We also disagree with choosing a model which arbitrarily assumes how the loss of containment integrity will occur. We are skeptical of the deterministic models, in particular their use of subjective, numerical weights for a set of containment performance factors.

We prefer something like the hazard state approach described in NUREG-0739, by which containment would be required to contribute appropriately to defense-in-depth.

Question 4

What are the uncertainties that the ACRS believes are associated with the approach? How should the uncertainties be ranked? What weight of importance should be assigned to the uncertainty?

Answer

The ACRS, the NRC Staff, and others have identified various sources of uncertainty associated with containment performance. We need to be shown that the proposed system of rankings or importance weight is an appropriate way to proceed.

Question 5

The Staff is interested in receiving input from as many knowledgeable sources as possible. We plan to solicit input from outside NRC and are considering EPRI, INPO, AIF, NSSS vendors, public action groups, and foreign regulatory groups. In particular, in the latter two groups are there specific groups we should solicit views from?

Answer

We suggest discussing the matter (and ascertaining current containment philosophy) with regulatory groups in all countries developing their own approach to LWR regulation, including the United Kingdom, France, West Germany, Switzerland, Italy, Sweden, and Japan.

Question 6

Does the ACRS have an opinion on whether there should be different CPGs and design objectives for existing and future plants? If yes, how should they be different?

Answer

We believe that there should be different containment performance goals and design objectives for existing and future plants. The flexibility with which the design of existing plants can be modified is severely constrained. Some changes that would be relatively straightforward in new plants may be very costly or impractical in existing plants.

The CPGs for new plants should be more ambitious and should consider (and possibly include) the effect of such steps as are feasible to minimize the frequency or even the potential for large airborne releases of radioactive material.

Question 7

Does the ACRS believe it is preferable to have numerical guidelines or goals in the CPGs and design objectives? If yes, what would representative values be? How would one relate these values to the safety goals?

Answer

We believe it would be useful to try to define quantitative goals in the CPGs and design objectives. As noted above, these would be different for future and current plants, and may have to vary among classes of current plants. Such goals would have to be constructed in a way which allows for

some very low probability accidents that negate containment effectiveness. There may be need for a separate performance guideline on the frequency of such accidents.

The Safety Goal Policy should provide a rough perspective on such decision making. However, considerations of large uncertainties, of the desirability of having defense-in-depth, and of the large economic cost of a core melt (even if no significant off-site releases result) may lead to a choice of design objectives for core melt frequency and containment performance whose combined effect is not directly derived from the safety goals.

Question 8

If one defines an attribute as, "a characteristic of an alternative which is capable of being evaluated in terms of its utility contribution to the composite utility for a specific objective," the classes of attributes below are intended to fully characterize the six approaches in B.3. Is the set complete? What additions and deletions would ACRS recommend? Why? What "weight" would ACRS recommend? Why? What "weight" would ACRS assign to each class of attributes, if any, (and each attribute include later)?

a. Viability of alternatives;

b. Credibility of improvement;

c. Acquisition cost of achieving (implementing?) alternative;

d. Impact on NRC;

e. Impact on industry; and

f. Expected overall benefits.

Answer

We are skeptical that the decision on this matter should be made by a weighting of various attributes. Therefore, we will not at this time attempt to recommend "weights."

Question 9

Does the ACRS consider the existing containment performance requirements adequate for design basis accidents? If not, where do inadequacies exist and how should they be fixed?

Answer

The NRC Staff deleted this question from the list at the March 15-17, 1984 Committee meeting.

Question 10

Should as much credit be given to containment integrity for meeting the safety goal in severe accidents as for meeting the regulatory requirements for dealing with design basis accidents? If yes, why? If no, why not?

Answer

Yes, but in a context which does not evaluate severe accidents in the same way current design basis accidents are analyzed. One is interested in actual capability, not the meeting of design and regulatory requirements.

Sincerely yours.

Jesse C. Ebersole
Chairman

Decay Heat Removal Systems

Decommissioning of Nuclear Power Plants

D

D

Decay Heat Removal Systems

Decommissioning of Nuclear Power Plants

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 18, 1983

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: NEED FOR RAPID DEPRESSURIZATION CAPABILITY IN NEWER COMBUSTION ENGINEERING, INC. PLANTS

During its 282nd meeting, October 13-15, 1983, the Advisory Committee on Reactor Safeguards (ACRS) reviewed analyses of the NRC Staff and of a Combustion Engineering Owners Group (CEOG) regarding the need for addition of power operated relief valves (PORVs) to certain nuclear power plants designed by Combustion Engineering, Inc. (CE). This matter had been reviewed previously by a Subcommittee of the ACRS on October 4, 1983, and earlier on January 27, 1983 and March 16, 1982. PORVs are automatic and remotely operable valves installed on the reactor coolant system (RCS) pressurizer in most PWRs. The valves were originally intended to intercept overpressure challenges to code safety valves. These latter valves are prone to failure to automatically reclose tightly following pressure relieving actuation. The PORVs were perceived to be more manageable in this respect in that they can be closed on demand and can be isolated by a block valve.

Analysis and experience have shown RCS pressure to be more easily controlled than had been recognized earlier so that the need for PORVs in avoiding code safety valve actuation is not now believed to be an important consideration. For that reason, CE, in its most recent plant designs, has not included PORVs in the RCS. Their reasoning is that leakage and the potential for spurious actuation of PORVs (creating, in effect, a small or medium break LOCA) are detrimental to both safety and operating efficiency.

However, within the past few years the PORV has come to be seen as offering other advantages. For one, it is a means to rapidly depressurize the RCS when desired, for example, to minimize leakage to the secondary side following failure of a steam generator tube. A second advantage is as a controlled means to remove steam or hot water from the RCS so that cooler water can be injected by the high pressure safety injection (HPSI) pumps. This is the so-called "feed and bleed" cooling process by which heat can be removed from the RCS and hence the reactor core. Because these advantages must be weighed against the disadvantages mentioned above and the cost of installing PORVs, the NRC Staff and CEOG each have made an extensive analysis of the pros and cons.

The NRC Staff has concluded that the CE plants without PORVs meet all regulatory requirements, with some minor exceptions which can be rather easily corrected. Further, they have concluded that these CE plants, which are equipped with reliable, auxiliary pressurizer sprays (APS) can effect moderate rates of depressurization to accommodate certain transients more effectively than can be done in other PWRs which have PORVs, but which do not have APS. The NRC Staff has also analyzed on a probabilistic basis accidents beyond the design basis accidents, including:

- multiple steam generator tube failures,
- total loss of feedwater,
- small break LOCA without HPSI,
- pressurized thermal shock, and
- ATWS.

The NRC Staff has concluded that addition of PORVs could be advantageous in permitting "feed and bleed" heat removal following loss of all feedwater, and that there would be some advantage in having PORVs provide additional pressure relief for ATWS, and in the case of failure of a large number of steam generator tubes. For the other accident sequences, they conclude that PORVs would provide no improvement over existing systems in the CE plants. The NRC Staff's overall cost-benefit analysis concludes there would be a slight advantage in adding PORVs over not adding PORVs. They acknowledge that the advantage is small compared with uncertainty in the analysis. However, the Staff also states it is their judgment that PORVs will provide an additional margin of safety in providing an effective, alternative means for depressurizing the RCS and thus provide greater flexibility in means for emergency core cooling.

Based on this judgment, the NRC Staff has concluded that PORVs should be required to be backfitted to the CE plants in question. However, they have also concluded that implementation of this requirement need not be hurried, and should be integrated with new requirements for decay heat removal systems that evolve from Task Action Plan A-45.

Analysis by the CEOG has produced results similar to those of the NRC Staff. They conclude the plants meet all regulatory requirements with the minor exceptions alluded to above. Their cost-benefit analysis shows a very small disadvantage in adding PORVs. Several differences in assumptions and data used by CEOG and those used by the NRC Staff apparently account for this conclusion, opposite from that of the NRC Staff. These differences have not been resolved. However, as with the cost-benefit analysis by the NRC Staff, the calculated margin is small compared with uncertainties.

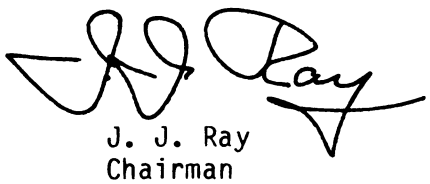
Although the CEOG acknowledges that PORVs could provide an emergency means to depressurize the RCS, they have concluded that depressurization by the APS or by rapid secondary side cooldown is much to be preferred. It is their judgment that PORVs should not be added.

The Committee believes there is so nearly a standoff between costs and benefits that extensive efforts to resolve differences or improve assumptions in the analyses are not warranted. A decision to require or not to require addition of PORVs must hinge on largely nonquantitative judgments.

Under some circumstances there might be significant safety advantage in having available an effective backup means to depressurize the RCS. On the other hand, maintaining integrity of the primary pressure boundary and removing heat through systems designed for that purpose, i.e., the steam generators, is generally preferable, even in emergency situations.

The Committee agrees with the NRC Staff's recommendation to integrate any new requirements for rapid depressurization into the more comprehensive new requirements for improvements to decay heat removal systems expected to be forthcoming from Task Action Plan A-45 within one year. We see no need for earlier resolution of the PORV issue.

Sincerely,



J. J. Ray
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 14, 1984

Mr. William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

SUBJECT: ACRS COMMENTS ON TASK ACTION PLAN A-45, "SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS"

During its 292nd meeting, August 9-11, 1984, the Advisory Committee on Reactor Safeguards received a status report by the NRC Staff and its contractor on the Task Action Plan (TAP) for resolution of USI A-45, "Shutdown Decay Heat Removal Requirements." The Subcommittee on Decay Heat Removal Systems had considered the status of the TAP during a meeting on June 5, 1984. This subject has been reviewed by the Committee and the Decay Heat Removal Systems Subcommittee on several other occasions since inception of the program.

The objectives of TAP A-45 have been described as an evaluation of the decay heat removal systems in current nuclear power plants to determine their adequacy and to evaluate the feasibility of alternative systems or methods, to determine the costs and benefits of the most promising alternatives, and finally, to develop a plan for implementing any new licensing requirements. The Task is complicated by the great variety and complexity of decay heat removal systems in existing plants.

We believe the issue being addressed by TAP A-45 is of major significance to the safety of nuclear power plants and should receive high priority in the allocation of agency resources. While the program appears to us to be addressing the major issues, and progress toward resolution is evident, we have several comments.

1) Because there is evidence that each nuclear power plant in the U.S. may have a nearly unique decay heat removal system and associated support systems, it may not be possible to develop a generic set of requirements, except at an abstract level, e.g., numerical reliability and cost goals. The present program is attempting to agglomerate all existing plants into a manageable number of groups, so that a representative system improvement can be defined for each. The NRC Staff should attempt to determine, early in the remainder of the program, whether this grouping to form generic types will ultimately be feasible. If not, an alternative approach must be developed in a timely manner.

2) We endorse attempts to make quantitative estimates of costs and benefits of proposed improvements. However, we caution that such

estimates can be used only as guides to judgment and should not be the sole basis for decisions. Serious consideration should be given to the way in which these estimates are developed. Unfortunately, for lack of other guidance, the NRC Staff has been evaluating health benefits by carrying the impact of any proposed change in a plant to the end consequence of human radiation exposure, and then using the ALARA standard of $1000/man-rem. This kind of bottom-lining reflects the weakest use of probabilistic risk assessment, and may, in the end, damage the credibility of an entire program.

3) We endorse the effort to apportion quantitative reliability goals among plant subsystems. We believe an attempt to be explicit can lead to a better understanding of the safety importance of the various systems and components and therefore holds promise for cost-effective improvements.

   We have some concern that the too-literal development of these reliability values from quantitative guidance of the present NRC trial safety goal may be burdening the A-45 program with the many ramifications of the safety goal issue. This quantitative use of the safety goal, or even of the reliability goals, seems to be an important policy decision. We suggest that, if such quantitative goals are to be important to resolution of USI A-45, some means be developed to have this policy thoroughly reviewed by the NRC on an expeditious basis.

4) The eight plants being studied in detail were judged by a qualitative screening method to be among those having the greatest weaknesses in existing decay heat removal systems. We think it is appropriate to concentrate further study on this end of the spectrum, but suggest that one or two plants from the middle or the other end of the spectrum should be included. The qualitative screening method, while seemingly appropriate, was judgmental and partly subjective. A study of a few plants outside of those chosen would provide additional confidence in the screening process.

5) In developing the accident scenarios, equipment environmental conditions should be considered which place demands on systems and components that might be subjected to hot water, steam, or radiation. This is especially important if feed and bleed is considered an effective means for removing decay heat.

We urge a high level of management attention to this important program and request that the Committee continue to be kept fully informed.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 15, 1978

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON DECOMMISSIONING OF NUCLEAR FACILITIES

Dear Dr. Hendrie:

For some time, the Advisory Committee on Reactor Safeguards has been active in a review of problems associated with the decommissioning of nuclear facilities. For example, the Committee referred to this matter in its report on the Platform Mounted Nuclear Plant, dated November 15, 1972, and in its report on the General Electric Standard Safety Analysis Report (GESSAR-238) dated March 14, 1975. Decommissioning was also the subject of a Subcommittee meeting held in Washington, D.C., on June 29, 1977. In addition it was considered at the 214th and 215th meetings of the full Committee held on February 9-11 and March 9-10, 1978, respectively. During the 214th meeting, the Committee discussed this item with representatives of the NRC Staff and the Atomic Industrial Forum, Inc.

As a result of these interactions, the Committee has had an opportunity to review NRC activities in this area. The Committee expresses its agreement with the increasing importance that the NRC Staff is assigning to this topic and its concurrence in the selection of the following specific items for implementation:

1. That the NRC Staff conduct a review of decommissioning problems associated with all types of nuclear facilities, including waste burial sites, so that priorities can be established for addressing those problems requiring primary attention. These efforts should include consideration of the manner and extent to which decommissioning is site related.

2. That the NRC Staff seek the advice of utility managers, architect-engineers, and vendors in the selection and evaluation of decommissioning alternatives to assure that, for given facilities, a range of choices is considered. This is particularly important because of the wide range of projected costs (person-rems and financial) associated with different approaches.

3. That these evaluations include an adequate review of problems associated with the management and disposition of the wastes resulting from decommissioning operations.

4. That the NRC Staff continue to develop criteria and standards for decommissioning.

5. That all such studies be designed to provide data that will enable licensees, suppliers, and architect-engineers to include in basic nuclear facility design key factors that will facilitate decommissioning operations.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

Emergency Core Cooling Systems (ECCS)

Emergency Planning

Engineered Safeguards

Extreme External Phenomena

E

Emergency Core Cooling Systems (ECCS)

Emergency Planning

Engineered Safeguards

Extreme External Phenomena

E

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

February 26, 1968

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT OF ADVISORY TASK FORCE ON POWER REACTOR EMERGENCY COOLING

Dear Dr. Seaborg:

The Advisory Committee on Reactor Safeguards offers the following comments on the recently issued Report of the Advisory Task Force on Power Reactor Emergency Cooling.

The Committee believes that the Task Force has performed a valuable service by assembling in a single document discussions covering many of the problems associated with postulated loss-of-coolant accidents and the phenomena important to proper functioning of emergency core cooling systems. Also, the Task Force has reviewed in a useful manner the many phenomena involved in the course of a postulated large-scale core meltdown.

Certain of the report's conclusions and recommendations appear to constitute expressions of judgment as to the adequacy or sufficiency of particular reactor safety provisions in respect to their capability for providing assurance against undue risk to the health and safety of the public. No attempt is made to comment on these. There are, however, a number of other conclusions in the report concerning which the Committee wishes to recommend emphasis, supplementation, or a differing viewpoint. Comments on these are given below.

In Conclusion 1, the report states in connection with the loss-of-coolant accident: "... for quantitative understanding of the accident, the analysis of such an event requires that the core be maintained in place and essentially intact to preserve the heat-transfer area and coolant-flow geometry. Without preservation of heat-transfer area and coolant-flow geometry, fuel-element melting and core disassembly would be expected. With the start of core disassembly there would be a major increase in the uncertainty of prediction of core behavior, and degeneration of the core to a meltdown situation could not be ruled out." The ACRS is in substantial agreement with this observation.

With respect to assuring that the core remains essentially intact during a loss-of-coolant accident, the report emphasizes the importance of properly assessing and designing for the hydraulic effects incurred, and lists several important specific aspects of the problem that must be recognized and dealt with in designing to cope with such effects. The ACRS agrees with this emphasis.

The possibility of fuel element failure from high internal pressure and high clad temperature during a loss-of-coolant accident is mentioned. In this connection, the Committee notes that present license applications show that a large fraction of fuel rods may fail in such accidents even though the emergency core cooling system works as designed. The Committee believes that, in addition to the work proposed by the Task Force, further research is needed to ascertain the modes of fuel rod failure and to determine that failures will not propagate or tend to block coolant flow excessively.

Conclusion 2 discusses further the importance of controlled and acceptable structural deformation during reactor blowdown in a loss-of-coolant accident. The ACRS agrees with this and calls attention to the need for considering deterioration during the life of the reactor and the role that periodic inspection could play in alleviating this potential difficulty. Also, more conservatism in design and fabrication may be needed where structural member response to accident-induced hydraulic forces is not testable. Further, the Committee continues to be concerned with the possibility of thermal shock effects on the pressure vessel, or other parts of the primary system, as a consequence of the rapid introduction of emergency cooling water in the unlikely event of a large primary system rupture.

The Committee endorses Conclusion 4 which recommends further testing of emergency core cooling at higher temperatures and for degenerated conditions such as core distortion.

The systematized approach to the design and evaluation of emergency core cooling systems described in Conclusion 5 appears potentially useful. However, deliberate allowance should be made for the possibility of aggravated accident conditions introduced by possible design errors, by weaknesses common to redundant components, or by other unexpected conditions, and full attention should be given to the potential advantage of diverse approaches to the design of emergency core cooling subsystems. It should be recognized, also, that new design features may introduce new potential safety issues in specific reactor designs.

The Committee endorses the recommendation of Conclusion 7 for improvements in primary system integrity to reduce still further the already low probability of primary system boundary failure.

The ACRS agrees with the statement of Conclusion 10 that: "If emergency core-cooling systems do not function ... the current state of knowledge regarding the sequence of events and the consequences of the meltdown is insufficient to conclude with certainty that integrity of containments of present designs, with their cooling systems, will be maintained." Recognizing that absolute certainty cannot exist concerning any facet of safety, the Committee strongly recommends that a positive approach be adopted toward studying the workability of protective measures to cope with core meltdown. Basic safety research experiments would provide valuable insight and, possibly, direct attention to potentially profitable avenues of design which eventually could lead to substantial additional protection in this area. The proposal in Conclusion 11 for study of preventive measures to be made effective prior to loss of containment integrity to minimize the ultimate hazard is a helpful step in this direction.

In summary, the Task Force Report presents considerable information of interest on primary system integrity, key phenomena effective during loss of coolant and core heatup, functional considerations for emergency core cooling systems, and phenomena and effects associated with large-scale core meltdown. However, there is provided little examination or discussion of the degree to which the efficacy of core cooling systems might be augmented by way of design modifications or new design concepts. Similarly, the report does not provide recommendations on design approaches to cope with large molten masses of fuel, or on the particular research and development problems related to these appraoches.

The ACRS endorses the Task Force recommendations for improvement in primary system boundary integrity and for additional research and development work on emergency core cooling systems. The Committee further recommends, as it did in its 1966 report on safety research, that a vigorous program be aimed at gaining better understanding of the phenomena and mechanisms important to the course of large-scale core meltdown. It also recommends that additional design and development effort be aimed at means for providing protection against the extremely low probability type of loss-of-coolant accident in which emergency core cooling systems of current design may not be effective. The ACRS urges that these matters be pursued vigorously by manufacturers of nuclear equipment, the electric utilities, and the AEC, as appropriate.

Sincerely yours,

/s/

Carroll W. Zabel
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 7, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D.C. 20545

Subject: REPORT ON INTERIM ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER POWER REACTORS

Dear Dr. Schlesinger:

At its 140th meeting, December 9-11, 1971, the Advisory Committee on Reactor Safeguards completed a review of proposed amendments to the Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors, published as an AEC Interim Policy Statement on June 29, 1971. The proposal was considered at a Subcommittee meeting held on December 4, 1971, in Washington, D. C. Further consideration was given to the interim criteria at the Committee's 141st meeting, January 6-8, 1972. The proposed amendments add new evaluation models by Combustion Engineering, Inc. and the Babcock and Wilcox Company to the existing list of acceptable evaluation models.

The Committee concluded that the proposed amendments were acceptable, on the same basis as the original Policy Statement evaluation models. Both the original and the amended criteria involve a number of provisions which are clearly conservative as well as some which may not be conservative, but on balance reflect adequate conservatism for interim use the plants similar in design to those which have been reviewed for construction permits. The Committee believes that each plant should be reviewed on a case by case basis to determine the extent to which these criteria are satisfied or modifications to the design or operation of the plant are required.

At the time adoption of the Interim Acceptance Criteria was under consideration, the Committee had opportunity to discuss the draft Criteria with the AEC Regulatory Staff and the Commission, and to offer comments. The Committee comments, made orally to the Commission on June 17, 1971 included the following:

> "The ACRS believes that the draft, 'Interim Acceptance Criteria for ECCS for Light-Water Power Reactors', dated June 17, 1971 is satisfactory for release subject to some clarification of the language.
>
> "The Committee concurs with the suggested approach of publishing the criteria in the Federal Register as an interim policy statement by the Commission. It agrees that Alternate A setting January 1, 1968 as a cut-off date is reasonable. It agrees with the relaxation in the criteria in Section IV B as promoting greater flexibility without freezing design and codes.
>
> "The Committee believes that the document represents an interim solution only, and that more work is required in code development, safety research oriented to LOCA-ECCS, and work on improved ECCS. The Committee is prepared to work closely with Staff and vendors through appropriate Subcommittees. We believe this approach may lead to an orderly solution of the outstanding problems.
>
> "With regard to the course of events of the last few months, the Committee now feels that there has been an over-all gain in that the level of work on ECCS evaluation has increased and more specific goals and objectives have been established. However, the Committee believes that more experimental work is needed to supplement the further analytical development of the evaluation models.
>
> "The Committee reemphasizes that the Commission should indicate publicly the need for continuing work on new ECC systems or significant modifications to current designs. It should be indicated that increases in power or power density will require larger margins."

Some of the restrictive assumptions imposed by the Interim Criteria have been introduced because of uncertainties in the behavior of the ECCS. The Committee believes such restrictions should be removed, as possible, either by design changes, or by demonstration that they are unnecessary. The Committee believes that it is timely for further implementation of the statement in the Commission's Policy Statement of June 29, 1971, that "in connection with the water power reactors yet to be designed and constructed the possibility of accomplishing by changes in design further improvements in the capability of emergency core cooling systems should be considered." Although the Interim Acceptance Criteria should be useful and helpful in the licensing process, the evaluation models prescribed in these criteria are recognized to have only limited usefulness as design tools for improving emergency core cooling systems. The nuclear industry should respond in a more direct fashion with realistic design methods, based upon additional scaled experiments and analytical studies. Design changes which would clearly eliminate any potential steam binding problems during a loss-of-coolant accident represent an example of the type of improvement considered to be of particular importance. The Committee recommends that design changes to improve ECCS capability should be sought and, to the extent practical, employed in plants for which construction permit applications are received in the future, irrespective of whether the plant design without such changes appears to meet the provisions of the Interim Acceptance Criteria.

The Committee believes that there is reasonable assurance that, with appropriate use of the Interim Acceptance Criteria and other applicable design and evaluation criteria, water reactors of current design can be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/ C.P. Siess

C.P. Siess
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 10, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: ACRS PARTICIPATION IN RULE-MAKING HEARING

Dear Dr. Schlesinger:

Subsequent to its meeting with the Commissioners on January 7, 1972, the ACRS has considered further the question of its participation in the forthcoming rule-making hearing on the interim Emergency Core Cooling Systems Criteria. The Committee continues to believe that its advice on reactor safety matters should be made available to the public only in the form of written reports which represent a consensus of the Committee. The Committee believes, therefore, that neither a member of the Committee, nor a member of its staff, should participate personally in the hearing.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

October 27, 1972

Nathaniel H. Goodrich, Esq.
Atomic Safety and Licensing Board Panel
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dr. John H. Buck
Atomic Safety and Licensing Appeals Board
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dr. Lawrence R. Quarles
Dean, School of Engineering and
Applied Science
University of Virginia
Charlottesville, Virginia 22901

Re: Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors (Docket No. RM-50-1)

Gentlemen:

Pursuant to the Commission's Order of January 26, 1972, and the Hearing Board's Order of September 22, 1972, in the proceeding, the Advisory Committee on Reactor Safeguards herewith submits for filing with the Board its responses to the interrogatories propounded on September 15, 1972, by the Consolidated National Intervenors.

I affirm that the attached answers to Consolidate National Intervenors interogatories are true and correct to the best of my knowledge and belief.

Sincerely,

/s/ C. P. Seiss

C. P. Siess
Chairman

Wash., DC October 28, 1972
Subscribed & sworn to before me this 28th day
of October, 1972 /s/ Ruth G. Davidow
Notary Public, DC
MY COMMISSION EXPIRES JULY 31, 1973

Distribution:
AEC Public Document Room
F. Karas, Public Proceedings Branch

In the Matter of)
)
ACCEPTANCE CRITERIA FOR EMERGENCY)
CORE COOLING SYSTEM FOR LIGHT-) Docket No. RM-50-1
WATER-COOLED NUCLEAR POWER REACTORS)

RESPONSES BY THE ADVISORY COMMITTEE
ON REACTOR SAFEGUARDS TO THE
INTERROGATORIES OF THE
CONSOLIDATED NATIONAL INTERVENORS

By motion dated September 15, 1972, the Consolidated National Intervenors (CNI) requested an order from the Hearing Board in the above-named proceeding requiring that the Advisory Committee on Reactor Safeguards (ACRS) answer a set of 86 interrogatories.

By Order of September 22, 1972, the Board allowed CNI interrogatories number 1, 2, 3, 9, 54, 56, 59, 61, 63, 67, 68, 75, 76, 83 and 85. Interrogatories 11, 45, and 55 are to be answered within limits or conditionally as specified by the Board's Order. The responses to the fifteen allowed interrogatories plus numbers 11 and 45 follow. The condition set by the Board in approving number 55 was determined to be not satisfied.

INTERROGATORY NO. 1: Does the ACRS believe that the PWR-FLECHT tests can be cited as an experimental demonstration of the effectiveness of the PWR ECCS? Please explain your answer in specific detail.

RESPONSE: No. The PWR-FLECHT tests are not a demonstration of system effectiveness, but are heat transfer tests which provide information for use in developing suitable models. Furthermore, the PWR-FLECHT tests address only that portion of the postulated loss-of-coolant accident designated as the reflood period.

INTERROGATORY NO. 2: Does the ACRS believe that the BWR-FLECHT test ZR-2 can be cited as an adequate demonstration of the effectiveness of the spray cooling system used on boiling water reactors? Please explain your answer in specific detail.

RESPONSE: No. The BWR-FLECHT test designated ZR-2 (correctly Zr-2K), like the PWR-FLECHT tests, is not a demonstration of system effectiveness, but is a heat transfer test which provides information for use in developing suitable models. Furthermore, the BWR-FLECHT tests address only a portion of the functions of the spray cooling system.

INTERROGATORY NO. 3: Does the ACRS believe that the PWR-FLECHT plate-simulated flow blockage tests can be accepted as the definitive basis for understanding the thermal/hydraulic effects of flow blockage in an open lattice pressurized water reactor core during a LOCA. Please explain your answer in specific detail.

RESPONSE: No. Other factors such as the actual core geometry and structural behavior and the redistribution of flow, and their interpretation in the analytical models, need to be considered further.

INTERROGATORY NO. 9: Does the ACRS believe that the quantity of data obtained in the two FLECHT programs is adequate for the purposes to which that data are used in the evaluation models? Please explain your answer in specific detail.

RESPONSE: Yes, for the purpose for which those data are used in the evaluation models. The ACRS believes that the purpose is to provide information for use in the development of empirical models for interim

application of the Interim Acceptance Criteria to evaluation of light water reactors of current design. The existing FLECHT data are relatively extensive but the range of parameters covered experimentally is limited. Extrapolation of these data, of necessity, leads to some uncertainty of prediction. Further experimentation and analysis is important and is underway.

INTERROGATORY NO. 11: Does the ACRS agree with Dr. L. J. Ybarrondo that the use of steady-state CHF and heat transfer correlations in the approved evaluation models is not necessarily conservative? Please explain your answer in specific detail.

RESPONSE: Yes. In the absence of exhaustive experimental data covering the effect of transients of all significant parameters on CHR and heat transfer correlations, it is necessary to agree that use of steady-state correlations, per se, in evaluation models involving transients is not necessarily conservative. However, the ACRS believes it is generally possible to use steady-state correlations in the evaluation models in such manner as to yield conservative results.

INTERROGATORY NO. 45: Does the ACRS believe that research should be performed to investigate the effects of an injection point on accumulator effectiveness? Please explain your answer in specific detail.

RESPONSE: Yes. Research involving effects of injection point of water into steam-water mixture flows under circumstances simulating possible conditions of a LOCA could improve our knowledge of the phenomena involved.

INTERROGATORY NO. 54: Does the ACRS believe that the approved evaluation models always calculate conservatively high estimates for the maximum cladding temperature for the LOCA under consideration? Please explain your answer in specific detail.

RESPONSE: No. The evaluation models include provisions for many detailed input parameters for which judgments must be made which affect the conservatism of the calculated maximum cladding temperature for the LOCA under consideration. However, the ACRS believes that with selection of appropriate values for these parameters and for other design and performance variables, such as peaking factors, cladding temperature can be evaluated in an acceptably conservative manner for an interim period.

INTERROGATORY NO. 56: Does the ACRS believe that the conservatism of the approved evaluation models has been quantified in a meaningful way? Please explain your answer in specific detail.

RESPONSE: No; however, although the degree of conservatism has not been quantified, the ACRS believes that with continuing reevaluation, the conservatism of the approved evaluation models has been sufficient for the interim purposes for which they have been used.

INTERROGATORY NO. 59: Have meaningful confidence limits been established for the approved LOCA transient analysis models? Please explain your answer in specific detail.

RESPONSE: The ACRS is aware of the sensitivity studies, of recent studies to establish integral errors, and of comparisons between evaluation model and best estimate results, but is not aware of any analyses that have established confidence limits as such.

INTERROGATORY NO. 61: Would the ACRS agree that a "conservative" analysis of a LOCA is one that calculates a higher maximum cladding temperature, or integrated time at temperature, than would actually occur in the LOCA under consideration? If the ACRS would not agree, please state its alternative definition. Please explain your answer in specific detail.

RESPONSE: No; we believe such a definition to be inadequate. The ACRS believes that an analysis of a LOCA that calculates a higher maximum cladding temperature than would actually occur in the LOCA is "conservative" with respect to calculation of cladding temperature, but does not necessarily comprise a "conservative" analysis of a LOCA. In a LOCA, there are aspects other than cladding temperature which are important to safety and which require evaluation for determination of conservativeness of the LOCA analysis.

INTERROGATORY No. 63: Hypothetical question: assuming that there are significant areas of uncertainty relative to the effect of maximum cladding temperature of various phenomena during a LOCA, can the "overall conservatism" of a LOCA transient analysis code be established without quantifying the uncertainties involved? Please explain your answer in specific detail.

RESPONSE: No. It is well established that, unless the uncertainties of the individual inputs are known, the uncertainty of the computed value cannot be determined by formal computation.

INTERROGATORY NO. 67: Do you believe that the independent analysis of ECCS effectiveness for PWR's is assisted by the fact that there exist four separate evaluation models and codes therefor? Please explain your answer in specific detail.

RESPONSE: The ACRS interprets an independent analysis to be one made by or for the AEC Regulatory Staff using models and codes derived or justified relatively independently of the development of the vendor models and codes. The ACRS believes that the independent analysis of ECCS effectiveness for PWRs (made by the Regulatory Staff using independently derived models and codes) has been assisted to some degree by the existence of a multiplicity of models and codes.

INTERROGATORY NO. 68: Do you believe that independent analysis of ECCS effectiveness for BWR's is hindered or rendered inadequate because there does not exist a code or evaluation model therefore, except that of General Electric? Please explain your answer in specific detail.

RESPONSE: The ACRS interprets an independent analysis to be one made by or for the AEC Regulatory Staff using models and codes derived or justified relatively independently of the development of the vendor models and codes. Capability for independent analysis of BWR-ECCS effectiveness has not been available until recently. The ACRS does not believe that this circumstance resulted from the fact that there was only one BWR vendor code or evaluation model.

INTERROGATORY NO. 75: Please list those aspects of each of the approved evaluation models that the ACRS does not believe to be presently proven to be conservative. Please explain your answer in specific detail.

RESPONSE: The ACRS takes the view that items in the approved evaluation models are proven to be conservative when fully confirmed by experimental evidence and supporting analytical studies. On this basis, the

following representative items are considered not proven to be conservative, and are undergoing investigation:

Analytical models and numerical methods.

Amount of initial stored energy in the fuel.

Treatment of phase separation and nonequilibrium effects.

Treatment of loop resistances.

Treatment of hot channel flow, including flow blockage and flow redistribution.

Treatment of break flows.

Treatment of decay heat.

Transient critical heat flux and heat transfer.

Treatment of clad ductility.

PWRs - Distribution of injected water.

Reflooding rates, reflood heat transfer, and carryover.

BWRs - Level swell.

Spray heat transfer.

While the above aspects are considered not proven to be conservative, the ACRS nevertheless believes that they can be handled in such manner that there is reasonable assurance that, with appropriate use of the Interim Acceptance Criteria and other applicable design and evaluation criteria, water reactors of current design can be operated without undue risk to the health and safety of the public.

INTERROGATORY NO. 76: Please list each item, matter or analysis not required to be done by each approved evaluation model which you believe should be required. Please explain your answer in specific detail.

RESPONSE: The Committee believes that the following representative additions and improvements should be made to the current evaluation models and their applications, on a reasonable time scale:

Analyses should be required of variance, systematic errors, and range of applicability of experimental data used to establish basic inputs, where practical.

Appropriate sensitivity studies should be made for key input parameters.

The use of values judged to be conservative should be required explicitly in the evaluation models for key parameters, such as heat transfer cooefficients.

Best estimate analyses, using most probably values for input parameters, also should be made.

Standard problems should be used to test each evaluation model.

Greater use should be made of means of confirming the numerical accuracy and internal logic of computer code calculations.

Suitable provisions should be included for the effects of flow blockage in the core.

Analyses for small breaks, using models approved for these break sizes, should be required in all cases, and appropriate safety limits for these breaks should be established.

Allowance for fuel densification and related effects should be included.

An acceptability limit based on time-at-temperature effects on clad ductility should be considered.

An approved independent evaluation model for BWRs should be made available.

As noted in the answer to Interrogatory No. 75, continuing studies are in progress. The results of these studies should be used to improve the evaluation models.

ADDITIONAL REMARKS BY MEMBER DAVID OKRENT: The design basis for the Dresden 3 reactor and succeeding BWR plants has been that either the core spray or the low pressure coolant injection (LPCI) mode alone should cool the core adequately in the event of a loss-of-coolant accident. I believe this diverse and redundant design basis to be important and that there should therefore be an approved evaluation model for the LPCI mode alone for BWR's.

I also recommend that in the future each evaluation model include, as practical, detailed justification of the physical assumptions and numerical methods employed in the model.

INTERROGATORY NO. 83: Would the ACRS agree that the Interim Policy Statement does not provide a uniform level of conservatism for all plants? Please explain your answer in specific detail.

RESPONSE: The ACRS believes that the Interim Policy Statement probably does not provide a uniform level of conservatism for all plants. However, the Committee believes that such uniformity is not essential.

INTERROGATORY NO. 85: Would the ACRS agree that the 0.8 core flow redistribution factor is arbitrary? If you would not agree, state the basis upon which you believe it not to be arbitrary. Is the 0.8 factor clearly supported by applicable experimental results? Please cite relevant results and please explain your answer in specific detail.

RESPONSE: The ACRS agrees that the 0.8 core flow redistribution factor is arbitrary. Continuing studies of flows under conditions postulated for the LOCA suggest that core redistribution factors may, under some circumstances, be either smaller or greater than 0.8. Thus, the handling of hot channel flow needs to be studied further.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 10, 1973

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

Dear Dr. Ray:

In response to your August 30, 1973, letter, the Advisory Committee on Reactor Safeguards at its 161st meeting, held on September 6-8, 1973, submits the following comments with regard to Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for Light-Water-Cooled Nuclear Power Reactors. The Committee previously commented on the Interim Acceptance Criteria for ECCS in its report of January 7, 1972. The Committee has also addressed matters related to the Interim Acceptance Criteria in its reports on various reactor applications during the past two years and in its response to a set of questions addressed to it by the Hearing Board on Docket No. RM-50-1.

The ACRS believes that the Concluding Statement of Position of the Regulatory Staff's Public Rulemaking Hearing on Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors (April 16, 1973) represents an improvement over the Interim Acceptance Criteria. The Committee notes that some sections still require interpretation. The Committee believes that the Position should represent a living document which will be responsive to added inputs of substantive information acquired through both analytical and experimental studies. The Committee believes also that it is essential that the Position provide flexibility for changes on a reasonable time scale.

In general, the Position retains previous restrictive assumptions, and several others have been added. Although there is evidence to support less restrictive evaluations, the Committee believes that a more substantial demonstration in terms of analyses and experimental data is required before relaxation of the restrictions can be effected. In achieving these goals, the development of the needed technical bases would be considerably enhanced if the proprietary aspects could be minimized.

The Committee continues to believe that more work is required on code development, safety research oriented to LOCA-ECCS, and improved ECCS. The Committee wishes to emphasize its opinion that a major portion of current and future research on LOCA-ECCS should be oriented toward the development of significant improvements in ECCS.

The Committee reaffirms its position that, in the future, design changes to improve ECCS capability should be sought and, to the extent practical, employed, irrespective of whether the plant design without such changes appears to meet the provisions of the Interim Acceptance Criteria and the proposed changes in these criteria. In its approach to nuclear safety, the Committee has sought to make allowances for the state-of-the-art knowledge on issues, to encourage further acquisition of background knowledge, to determine whether added safety improvements or margins are achievable, and then to encourage incorporation of practical improvements. It is the Committee's judgment, that for an expanding nuclear industry, the cumulative effects of the added improvements represent prudent goals. In particular, reactors proposed for standardized units should incorporate design improvements.

The Committee views the following as examples of measures which contribute to design improvements: 1) improved reliability of the ECCS and system components, including approaches intended to minimize the potential for common failure modes; 2) reactor core designs and operating modes which reduce the potential for high temperatures, clad swelling or perforation in postulated LOCAs; 3) ECCS whose proper functioning is relatively insensitive to reactor or ECCS design parameters and to proper functioning of other components such as steam generators or reactor containment; 4) ECCS having redundancy, diversity and abundance of flow such that its adequacy is subject to evaluation without undue requirement for complex evaluation techniques; and 5) other measures which further reduce the probabilities and consequences of a LOCA.

For existing ECCS designs, the ACRS encourages the ongoing efforts to develop evaluation methods, which are subjected to rigorous quality assurance measures, which are validated by experiment and theory, and which receive independent confirmation by the Regulatory Staff.

Acceptance Criteria for ECCS are not viewed by the Committee in isolation from other applicable design and evaluation criteria and guides. The Committee seeks to identify and to evaluate all pertinent factors for each case reviewed, as well as to be responsive to new information and experience. As noted in the Committee's December 18, 1972, report, Status of Generic Items Relating to Light-Water Reactors, resolution of some items on a generic basis is still pending. Other items which have been identified, studied or

are under study, include pressure vessel integrity, steam generator tube integrity, reactor coolant pump overspeed, independent capability to analyze the ECCS, evaluations of containment pressures, independent analysis of the ice-condenser containment time-pressure history, and structural responses to a LOCA. All such items are taken into account in arriving at Committee positions.

For those plants which are licensed to operate under acceptance criteria such as those of the Regulatory Staff's Position, the Committee recommends continued study of means to improve ECCS reliability and performance by such design and operational measures as appear practical and significant. When such studies indicate that a significant safety improvement can be achieved, it should be considered for backfitting on a timely basis in accordance with the Commission's backfitting policy.

The evaluation of ECCS capabilities must, of course, include the possible effects arising from fuel densification. Several restrictive assumptions have been introduced by the Regulatory Staff in their densification evaluation models that, when combined with the Interim Acceptance Criteria, result in more severe limitations being placed on maximum linear heat generation rates for PWRs and maximum average planar (bundle) linear heat generation rates for BWRs. These restrictions result in the requirements for more intensive incore monitoring. In some cases, such restrictions have also led to lower initial power ratings for reactors having operating licenses. In this new phase of bringing into operation reactors having both higher power and higher power density, the Committee agrees that some initial limits on power may be appropriate in achieving the following objectives:

a) prudent approach for gaining operating experience
b) more thorough demonstration of fuel behavior
c) more detailed measurements of maximum linear heat generation rates and peaking factors for PWRs, and maximum average planar (bundle) linear heat generation rates for BWRs
d) resolution of more generic items, and
e) implementation of the proposed new requirements for the acceptance criteria, including evaluation of analytical and experimental data relating to model and code developments.

The initial restriction on power ratings and/or on flexibility for power operations and the duration of these restrictions should be evaluated on a case-by-case basis until a standardized approach has evolved through implementation of the Position.

The Committee believes that the Concluding Statement of the Regulatory Staff represents an acceptable Position, that possible ECCS design improvements should be pursued for plants in operation or under construction, and that reactors filing for construction permits after January 7, 1972, should have significantly improved ECCS capability.

Sincerely yours,

H. G. Mangelsdorf
Chairman

References:

1) Concluding Statement of the Regulatory Staff entitled "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Reactors," April 16, 1973

2) Final Environmental Statement concerning the Proposed Rulemaking Action: Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, dated May 9, 1973 (Volumes 1 and 2)

3) Supplementary Concluding Statement of Position of the Regulatory Staff's Public Rulemaking Hearing on Acceptance Criteria for ECCS for Light-Water-Cooled Nuclear Power Reactors, dated August 9, 1973

FOR SEPTEMBER 11, 1973 LTR TO DIXY LEE RAY, TRANSMITTING MANGELSDORF MEMO TO MUNTZING RE FORKED RIVER, SAN ONOFRE 2&3, AND WATERFORD 3 ECCS DESIGNS, SEE PAGES 613-615 UNDER "FORKED RIVER".

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 15, 1974

L. M. Muntzing
Director of Regulation

The Committee recommends that there should be judicious and explicit inclusion of uncertainties in the representation of power distribution in the evaluation models formulated to meet the Final ECCS Acceptance Criteria.

W. R. Stratton
Chairman

cc: E. G. Case, L
L. Rogers, RS
P. C. Bender, SECY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 20, 1974

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON EVALUATION MODELS FOR COMMISSION CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

Dear Dr. Ray:

At its 175th meeting, November 14-16, 1974, the Advisory Committee on Reactor Safeguards completed a review of Evaluation Models which have been submitted in accordance with the Commission criteria set forth in 10 CFR 50.46. The following subcommittee meetings with reactor vendors were held in Washington, D. C.: March 28, 1974, Babcock and Wilcox; April 25, 1974, General Electric Company; April 26, 1974, Westinghouse Electric Corporation; and May 18, 1974, Combustion Engineering, Inc. Subcommittee meetings were held with the Regulatory Staff and their consultants in Washington, D. C., on August 6, 1974, September 28, 1974 and October 26, 1974. The Committee also had the benefit of the documents listed below. Previous reports to the Commission on interim acceptance criteria were made on January 7, 1972, and on the proposed changes on September 10, 1973. The Committee has also addressed the safety research programs and the latest report is on November 20, 1974.

The ACRS believes that the four light-water reactor vendors have developed Evaluation Models which, with the additional modifications required by the Regulatory Staff, will conform to Appendix K to Part 50.

Approved Evaluation Models will aid in conducting the licensing reviews, but a variety of specifics must be evaluated on a case-by-case basis. Items such as the particular features of a containment, sequencing of operations, single failure analysis and special features of the reactor design, are noted in the Staff's review of the vendor models. Additional items involving peaking factors and treatment of the uncertainties in the power distributions and monitoring of the power levels remain to be incorporated, case-by-case, in the Technical Specifications with appropriate conservatism.

The generic review of the vendor models proposed for Appendix K, like the reviews of the Interim Acceptance Criteria models, has contributed to improved understanding of the modeling techniques, including the applicability and limitations on current knowledge of thermal and hydraulic phenomena, and the need for more definitive safety research programs and code developments. The implementation of safety research programs, noted in the Committee's (November 20, 1974) report, and their results should have impact on the future evaluation methods and ECC systems.

The ACRS remains mindful that the Evaluation Models, in themselves are not the desired end products, but that effective, reliable emergency core cooling systems are the objective. The Committee acknowledges the contribution to reduced peak clad temperatures resulting from recent core design changes but reaffirms its position stated in the September 10, 1973 report that improved ECCS reliability and capability should continue to be sought and, to the extent practical, employed.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References Attached.

References

1) WCAP-8170 (P) dated June 1974, "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident"
2) WCAP-8200 (Rev. 2 (P)) dated June 1974, "WFLASH - A Fortran - IV Computer Program for Simulation of Transients in a Multi-Loop PWR"
3) WCAP-8301 (P) dated June 1974, "LOCTA - IV Program Loss-of-Coolant Transient Analysis"
4) WCAP-8302 (P) dated June 1974, "SATAN IV Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant"
5) WCAP-8327 (P) dated July 1974, "Containment Pressure Analysis"
6) WCAP-8339 (NP) dated June 1974, "Westinghouse Emergency Core Cooling System Evaluation Model, Summary"
7) WCAP-8340 dated July 1974, "Westinghouse ECCS - Plant Sensitivity Studies"
8) WCAP-8341 (P) dated July 1974, "Westinghouse Emergency Core Cooling System Evaluation Model Sensitivity Studies"
9) WCAP-8354 (P) dated July 1974, "Long Term Ice Condenser Containment Code - LOTIC Code"
10) BAW-10091, "B&W's ECCS Evaluation Model Report with Specific Application to 177 FA Class Plants with Lowered Loop Arrangement," August 1974
11) BAW-10092, CRAFT2 - Fortran Program for Digital Simulation of a Multinode Reactor Plant During Loss-of-Coolant," July 1974
12) BAW-10093, "REFLOOD - Description of Model for Multinode Core Reflood Analysis," July 1974
13) BAW-10094, "Revisions to THETA 1-B, A Computer Code for Nuclear Reactor Core Thermal Analysis," IN-1445, July 1974
14) BAW-10095, "CONTEMPT - Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," July 1974
15) CENPD-132, "Calculative Methods for the CE Large Break LOCA Evaluation Model," (P) August 1974
16) CENPD-134 (P) April 1974, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core"
17) CENPD-135 (P) April 1974, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program"
18) CENPD-136 (P) July 1974, "High Temperature Properties of Zircaloy and UO_2 for Use in LOCA Evaluation Models"
19) CENPD-137 (P) August 1974, "Calculative Methods for the CE Small Break LOCA Evaluation Model"
20) CENPD-138 (P) April 1974, "PARCH - A Fortran IV Digital Computer Program to Evaluate Pool-Boiling Axial Rod and Coolant Heatup"
21) CENPD-139, "CE Fuel Evaluation Model (FATES Thermal Performance and Densification Model)"
22) NEDO-20566 (DRAFT) (P) - "General Electric Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K"
23) Status Report By The Directorate of Licensing In The Matter Of Babcock and Wilcox ECCS Evaluation Model Conformance To 10 CFR 50, Appendix K (undated)

24) Supplement 1 To The Status Report By The Directorate Of Licensing In The Matter Of Babcock And Wilcox ECCS Evaluation Model Conformance To 10 CFR 50, Appendix K, November 13, 1974
25) Status Report By The Directorate Of Licensing In The Matter Of Combustion Engineering, Inc. ECCS Evaluation Model Conformance To 10 CFR 50, Appendix K (undated)
26) Supplement To The Status Report By The Directorate Of Licensing In The Matter Of Combustion Engineering, Inc. ECCS Evaluation Model Conformance To 10 CFR 50, Appendix K, November 13, 1974
27) Status Report By The Directorate Of Licensing In The Matter Of General Electric ECCS Evaluation Model Conformance To 10 CFR 50, Appendix K (undated)
28) Supplement 1 To The Status Report By The Directorate Of Licensing In The Matter Of General Electric ECCS Evaluation Model Conformance To 10 CFR 50, Appendix K, November 13, 1974
29) Status Report By The Directorate Of Licensing In The Matter Of Westinghouse Electric Company ECCS Evaluation Model Conformance To 10 CFR 50, Appendix K (undated)
30) Supplement To The Status Report By The Directorate Of Licensing In The Matter Of Westinghouse Electric Company ECCS Evaluation Model Conformance To 10 CFR 50, Appendix K, November 13, 1974
31) WREM: Water Reactor Evaluation Model, October 1974, Regulatory Staff-Technical Review
32) Water Reactor Evaluation Model (WREM): PWR Nodalization And Sensitivity Studies, October 1974, Regulatory Staff-Technical Review
33) Water Reactor Evaluation Model (WREM): BWR Nodalization And Sensitivity Studies, October 1974, Regulatory Staff-Technical Review
34) Evaluation Of LOCA Hydrodynamics, November 1974, Regulatory Staff-Technical Review

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 14, 1975

L. Manning Muntzing
Director of Regulation

Subject: LOCKING OUT OF ECCS POWER OPERATED VALVES

The Committee suggests that further attention be given to questions which arise in connection with procedures involving physically locking out electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS systems, based on the assumption that a spurious electric signal at an inopportune time could actuate the valves to the adverse position; e.g., closed rather than opened, or opened rather than closed. Examples where this procedure is being applied include Trojan, Salem, and Beaver Valley.

The ACRS believes that the matter should be studied using a system approach, both for existing and for future plants. Examples of suggested questions to be answered include:

(1) An evaluation of the probability of a spurious signal;

(2) Can the valve operator be reactivated in a reasonable time after indication of loss of electric power;

(3) If the circuit breaker is opened, are signal lights lost;

(4) If the signal lights are deactivated, is it possible to reactivate them through bypass circuits, etc.;

(5) Is it possible to lock a valve out in the apparently open position whereas it actually is closed since one depends on indicator position;

(6) Cases are documented where valve stems have broken leaving the gate in the closed position; is the indicator such that the lights indicate an open valve while it is actually closed
   (a) Is there a positive testing process applicable on a routine basis to establish that the critical valves are indeed correctly aligned, permitting a flow path;
   (b) If no testing process exists, can one use non-destructive examination techniques to unequivocally locate the gate position

(7) Would a more reliable ECCS system result if fewer motor-operated valves, more check valves, and more parallel paths were used; what other factors influence the choice of alternate designs;

(8) What are the pros and cons of a system where an incorrect action can be rapidly reversed by an alert operator compared to a system where electric power is locked out, but one or more valves are incorrectly aligned requiring the operator to go some distance in a very short time to reactivate the circuit breakers.

The Committee would be willing to discuss this matter with the Regulatory Staff through an appropriate subcommittee.

/s/ W. Kerr

W. Kerr
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 10, 1975

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: EMERGENCY CORE COOLING SYSTEMS EVALUATION MODEL (ECCS-EM) FOR THE NONJET-PUMP BOILING WATER REACTOR

Dear Mr. Anders:

At its 179th meeting, March 6-8, 1975, the Advisory Committee on Reactor Safeguards completed its review of the Exxon Nuclear Company's ECCS Nonjet-Pump Boiling Water Reactor Fuel Heatup Model. The initial application of this model is for the evaluation of a reload core at the Oyster Creek Nuclear Generating Station and requires the proper coupling with the approved General Electric Emergency Core Cooling Systems Evaluation Model (ECCS-EM) for the nonjet-pump boiling water reactor. The Jersey Central Power and Light Company is responsible for obtaining the overall ECCS-EM which meets the requirements of 10 CFR 50.46, Appendix K. Subcommittee meetings were held with the NRC Staff and the representatives of the Exxon Nuclear Company in Washington, D.C., on February 1 and 22, 1975, with attendance at the second meeting expanded to include participation by representatives from the Jersey Central Power and Light Company. The Committee previously reported on Evaluation Models for Commission Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors on November 20, 1974. The Committee also had the benefit of the documents listed below.

The ACRS believes that the Exxon Nuclear Company's portion of the ECCS-EM with the additional modifications required by the NRC Staff and when properly coupled with the approved General Electric nonjet-pump boiling water reactor model will conform to 10 CFR 50.46, Appendix K. Although the Committee believes that the coupling procedure is acceptable for the present application, the Committee recommends that future applications be based upon an integrated set of approved models.

Sincerely yours,

William Kerr
Chairman

References:

1. XN-73-34 (Rev 2), "HUXY: A Generalized Multirod Heatup Code"

2. XN-74-25 (Rev 1), "2DQ: A Two Dimensional Quenching Model"

3. XN-74-27 (Rev 1), "BULGEX: A Computer Code to Determine the Deformation and the Onset of Bulging of Zircaloy Fuel Rod Cladding"

4. XN-74-51, "ECCS Analysis for Oyster Creek Fuel Types IIIA, IIIE, and IIIF with the Exxon Nuclear Evaluation Model for BWR Loss of Coolant Accidents"

5. XN-174 and Suppl 1, "Densification Effects on Exxon Nuclear Boiling Water Reactor Fuel"

6. XN-235, "Exxon Nuclear Evaluation Model for BWR Loss of Coolant Accidents"

7. Report Regarding the Exxon Nuclear Company ECCS Non-Jet-Pump-BWR Fuel Heatup Model by the Office of Nuclear Reactor Regulation March 6, 1975

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 15, 1975

Honorable William A. Anders
Chairman
US Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON EXXON NUCLEAR COMPANY ECCS EVALUATION MODEL FOR RELOAD CORES IN PWR'S

Dear Mr. Anders:

At its 185th Meeting, September 11-13, 1975, the Advisory Committee on Reactor Safeguards completed its review of the Exxon Nuclear Company's ECCS Evaluation Model for reload cores in pressurized water reactors. A Subcommittee meeting was held with representatives of the Exxon Nuclear Company, Inc., and the Nuclear Regulatory Commission in Washington, D.C., on August 21, 1975. The Committee also had the benefit of the documents listed below.

Following its review of the Exxon Nuclear Company's ECCS Evaluation Model for reload cores for nonjet-pump BWR's, the Committee recommended in its March 10, 1975, report to the Commission that future applications by the Exxon Nuclear Company present an integrated set of models. The Exxon Nuclear Company, in responding to this recommendation, has used the Nuclear Regulatory Commission's Water Reactor Evaluation Model (WREM) code package as the starting point and has developed the Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model. Since the NRC Staff's WREM model did not satisfy all requirements of Appendix K to 10 CFR 50, a significant effort has been required to demonstrate conformance of the new evaluation model with Appendix K. Deficiencies identified by the Staff have been addressed, and the Committee believes that the Exxon Nuclear Company can complete the remaining requirements set forth by the Staff. Even with an approved generic evaluation model, applications of the model will require case-by-case evaluation by the NRC Staff. Applications are scheduled for H. B. Robinson Unit 2, Yankee Rowe, and Palisades.

The Committee believes that, with the additional requirements set by the NRC Staff, the Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model will conform to Appendix K to 10 CFR 50.

Sincerely,

W Kerr

William Kerr
Chairman

REFERENCES TO EXXON NUCLEAR COMPANY ECCS EVALUATION MODEL FOR RELOAD CORES IN PWR'S LETTER:

1. Division of Technical Review, Nuclear Regulatory Commission, report entitled: "WREM: Water Reactor Evaluation Model, Revision 1," dated May 1975, (NUREG-75-056).

2. Division of Technical Review, Nuclear Regulatory Commission, report entitled: "TOODEE 2: A Two Dimensional Time Dependent Fuel Element Thermal Analysis Program," dated May 1975, (NUREG-75-057).

3. Exxon Nuclear Company, Inc., report entitled: "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model," (XN-75-41)
Volume I, dated July 25, 1975
Volume II, "-Model Justification," dated August 1, 1975
Volume II Appendix, "- 3-Loop Westinghouse Sample Problem," dated August 1, 1975
Volume III (Revision 2), "-Small Break Model," dated August 20, 1975
Supplement 1, "-Further Definition and Justifications to Reflood Heat Transfer Models," dated August 14, 1975
Supplement 2, "-Supplementary Information Related to Blowdown and Reflood Analysis," dated August 14, 1975
Supplement 3, "-Supplementary Information Relating to Blowdown and Heatup Analysis"
Supplement 4, "-Supplementary Information Relating to Blowdown and Heatup Analysis," dated August 20, 1975

4. Exxon Nuclear Company, Inc., Nuclear Regulatory Commission, report entitled: "Core Physics Methods and Data Used as Input to LOCA Analysis," (XN-75-43) dated August, 1975

5. Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, report entitled: "Safety Evaluation Report Regarding Review of the Exxon Nuclear Company Pressurized Water Reactor Generic ECCS Codes and the H. B. Robinson Reactor ECCS Evaluation Model for Conformance to All Requirements of Appendix K to 10 CFR 50," dated September 11, 1975.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 14, 1976

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: REPORT ON WESTINGHOUSE ELECTRIC CORPORATION'S ECCS UPPER HEAD INJECTION EVALUATION MODEL

Dear Mr. Rowden:

At its 197th meeting, September 9-11, 1976, the Advisory Committee on Reactor Safeguards completed a review of the Westinghouse Electric Corporation's ECCS Upper Head Injection (UHI) Evaluation Model. Six Subcommittee meetings have been held with representatives of the Westinghouse Electric Corporation and the Nuclear Regulatory Commission, the first being held at Monroeville, Pennsylvania, on June 25, 1975, and the remaining meetings at Washington, DC, on October 4 and December 20, 1975, and March 16, June 15, and September 2, 1976. The Committee also had the benefit of the documents listed below.

The NRC Staff has taken into account the special features of UHI and the supporting research and development in order to formulate evaluation model requirements which suitably conform to 10 CFR 50, Appendix K. Further, sensitivity studies are being performed by Westinghouse to provide assurance that sufficiently conservative bounds for the evaluations will be included.

The ACRS believes that the Westinghouse Electric Corporation's ECCS UHI Evaluation Model, with the requirements set by the NRC Staff, will conform to 10 CFR 50, Appendix K. The approved evaluation model, with appropriate use of plant-specific parameters on a case-by-case basis, will aid in the licensing reviews.

The ACRS encourages the NRC Staff to continue its deliberations in seeking accelerated and coordinated programs for establishing meaningful experimental facilities and independent analytical tools for studying the performance of UHI-ECC systems.

As noted in its November 20, 1974 Report on Evaluation Models for Commission Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, the ACRS "remains mindful that the Evaluation Models, in themselves are not the desired end products, but that effective, reliable emergency core cooling systems are the objective." Continuing efforts are needed for implementing the safety research programs to provide bases for confirming design margins and for improving ECCS reliability and capability.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

REFERENCES:

1. "Status Report to the Advisory Committee on Reactor Safeguards in the Matter of Westinghouse Electric Corporation ECCS-Upper Head Injection Evaluation Model Conformance to 10 CFR 50, Appendix K," Proprietary (August 1976) U.S. Nuclear Regulatory Commission - Office of Nuclear Reactor Regulation.
2. WCAP-8400 - "ECCS Heat Transfer Experiences with Upper Head Injection," Proprietary, Volume I (October 1974), Volume II (January 1975), Volume III (August 1976), Westinghouse Electric Corporation.
3. WCAP-8479, Revision 1 - "Westinghouse Emergency Core Cooling System Evaluation Model Application to Plants Equipped with Upper Head Injection," Proprietary (August 1976) Westinghouse Electric Corporation.
4. WCAP-8582 - "Blowdown Experiments with Upper Head Injection in G-2, 17X17 Rod Array," Proprietary, Volume I (January 1976), Volume II (August 1976), Westinghouse Electric Corporation.
5. WCAP-8793 - "G-2, 17X17 Refill Heat Transfer Tests and Analysis," Proprietary (August 1976) Westinghouse Electric Corporation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 16, 1977

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: REPORT ON EXXON NUCLEAR COMPANY WREM-BASED NON-JET PUMP BOILING WATER REACTOR ECCS EVALUATION MODEL

Dear Mr. Rowden:

At its 202nd meeting, February 10-12, 1977, the Advisory Committee on Reactor Safeguards completed its review of the Exxon Nuclear Company's ECCS Evaluation Model for non-jet pump boiling water reactors. Subcommittee meetings were held on July 21-22, 1976, in Richland, Washington, and on December 21, 1976, and January 29, 1977, in Washington, D.C. During its review, the Committee had the benefit of discussions with representatives of the Exxon Nuclear Company and the NRC Staff. The Committee also had the benefit of the documents listed.

The ACRS previously reported on an evaluation model developed by Exxon for non-jet pump boiling water reactors in the Committee's letter of March 10, 1975. That model required the coupling of the General Electric non-jet pump boiling water reactor model for blowdown with the Exxon model for the fuel heatup. The Committee recommended that future applications be based upon an integrated set of approved models. The first Exxon integrated model reviewed by the Committee was approved in a September 15, 1975 report on an evaluation model for a PWR.

The present non-jet pump BWR integrated evaluation model uses several portions of the Exxon WREM-Based Generic PWR Evaluation Model, and the Committee concurs with the NRC Staff that the blowdown portions are treated in an acceptable manner.

The NRC Staff has not accepted the proposed transient critical heat flux correlation, and Exxon has agreed to use a conservative procedure proposed by the Staff. The Committee concurs with the Staff in imposing this requirement.

Use of the Evaluation Model will require review by the NRC Staff of plant-specific items for case-by-case applications. The Committee believes that, with the requirements set forth by the NRC Staff, the Exxon Nuclear Company WREM-Based Non-Jet Pump Boiling Water Reactor ECCS Evaluation Model will conform to Appendix K to 10 CFR 50.

Drs. S. H. Bush and M. W. Carbon did not participate in this review.

Sincerely yours,

M. Bender
Chairman

REFERENCES:

1. Exxon Nuclear Company, "GAPEX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients," XN-73-25, August 13, 1973, as found in XN-174 "Densification Effects on Exxon Nuclear Boiling Water Reactor Fuel," November 1973, and Supplement 1, November 29, 1973.
2. Exxon Nuclear Company, "2DQ: A Two-Dimensional Quenching Model," XN-74-25, Revision 1, October 30, 1974.
3. Exxon Nuclear Company, "BULGEX: A Computer Code to Determine the Deformation and the Onset of Bulging of Zircaloy Fuel Rod Cladding," XN-74-27, Revision 2, December 1974.
4. U.S. Nuclear Regulatory Commission, "WREM: Water Reactor Evaluation Model," Revision 1, May 1975.
5. Exxon Nuclear Company, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model," XN-75-41, Vol. I, Vol. II, and Appendices A, B, C, and D, Vol. III (Revision 2), and Supplements 1-7, July-November 1975.
6. Exxon Nuclear Company, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option - User's Manual," XN-CC-33(A), July 28, 1975.
7. Exxon Nuclear Company, "Core Physics Methods and Data Used as Input to LOCA Analysis," XN-75-43, August 1975.
8. Exxon Nuclear Company, "The Exxon Nuclear Company WREM-Based NJP-BWR ECCS Evaluation Model and Application to the Oyster Creek Plant," XN-75-55, Revision 2, August 1976, Supplement 1, September 1976, and Supplement 2, December 1976.
9. Intermountain Technologies, Inc., "Comparisons of the RELAP-4 Computer Code with Experimental Data," December 1976.
10. U.S. Nuclear Regulatory Commission, "Revision 1 to Safety Evaluation Report Regarding Review of the Exxon Nuclear Company Non-Jet Pump Boiling Water Reactor ECCS Evaluation Model for Conformance to Appendix K to 10 CFR 50 by the Office of Nuclear Reactor Regulation," January 14, 1977.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 16, 1977

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: REPORT ON THE EXXON NUCLEAR COMPANY WREM-BASED GENERIC PWR-ECCS EVALUATION MODEL UPDATE, ENC-WREM-II

Dear Mr. Rowden:

At its 202nd meeting, February 10-12, 1977, the Advisory Committee on Reactor Safeguards reviewed the modifications to the Exxon Nuclear Company Water Reactor Evaluation Model which provides a generic evaluation model (ENC-WREM-II) for the analysis of the emergency core cooling systems (ECCS) for pressurized water reactors without upper head injection and having a low containment pressure such as that for an ice-condenser pressure-suppression system. This matter was considered at the 201st ACRS meeting, held in Washington, D.C., on January 6-8, 1977, and at ECCS Subcommittee meetings held on July 21-22, 1976, in Richland, Washington, and December 21, 1976, in Washington, D.C. During its review, the Committee had the benefit of discussions with representatives of the Exxon Nuclear Company and the NRC Staff. The Committee also had the benefit of the documents listed.

The previous Exxon PWR Evaluation Model utilizing WREM was approved by the Committee in its September 15, 1975, report. For ENC-WREM-II, the principal modifications involve the evaluation of flow blockages, reflood heat transfer coefficients, and the hot wall delay time. The Committee concurs with the NRC Staff on the approval of these modifications.

Use of the generic model will require review by the NRC Staff of plant specific items for case-by-case applications. The Committee believes that, with the requirements set forth by the NRC Staff, the Exxon Nuclear Company Evaluation Model Update, ENC-WREM-II, will conform to Appendix K to 10 CFR 50.

Drs. S. H. Bush and M. W. Carbon did not participate in this review.

Sincerely yours,

M. Bender
Chairman

ADDITIONAL COMMENTS BY MEMBERS DAVID OKRENT AND MILTON PLESSET

The ENC-WREM-II ECCS Evaluation Model is intended for application to the D.C. Cook Unit 1 on which the ACRS reported on January 14, 1977. It is proposed that this reactor be operated with a maximum F_Q of only 1.95.

We believe that, particularly for such an application, the proposed modifications in the ECCS evaluation model must be treated with some reserve. Reflood heat transfer coefficients should be more reliably determined, particularly under conditions of low pressure and low reflood rates. Also, further studies, both analytical and experimental, are needed to justify that conservatism exists under all conditions for assumption of a large reduction in flow blockage for a narrow range of fuel element gas pressure.

We believe that further studies, including sensitivity studies and statistical analysis of flow blockage, heat transfer coefficients, and the effects of significant steam generator leakage, should be made before this particular evaluation model is accepted for application to a low pressure containment.

REFERENCES:

1. Exxon Nuclear Company, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," XN-75-21, April 1, 1975.
2. U.S. Nuclear Regulatory Commission, "WREM: Water Reactor Evaluation Model," NUREG-75/056, Revision 1, May 1975.
3. Exxon Nuclear Company, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model," XN-75-41, Vol. I, Vol. II, and Appendices A, B, C, and D, Vol. III (Revision 2), and Supplements 1-7, July-November 1975.

4. Exxon Nuclear Company, "Definition and Justification of Exxon Nuclear Company DNB Correlation for PWR's," XN-75-48, October 6, 1975.
5. Exxon Nuclear Company, "Flow Blockage Model for LOCA Analysis," XN-75-6(A), January 1976.
6. Exxon Nuclear Company, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC-WREM-II," XN-76-27, July 1976, Supplement 1, September 1976, and Supplement 2, November 1976.
7. U.S. Nuclear Regulatory Commission, "Revision 1 to Safety Evaluation Report on the Exxon Nuclear Company WREM-Based Generic PWR-ECCS Evaluation Model Update ENC-WREM-II for Conformance to Requirements of Appendix K to 10 CFR 50 by the Office of Nuclear Reactor Regulation," January 5, 1977.
8. Exxon Nuclear Company, "Flow Blockage and Exposure Sensitivity Study for D.C. Cook Unit 1 Reload Fuel Using ENC WREM-II Model," XN-76-51 Supplement 1, January 1977.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 13, 1977

Lee V. Gossick
Executive Director for
Operations

PROPOSED REVISION OF 10 CFR PART 50, APPENDIX K

During its 209th meeting the Committee discussed with the NRC Staff four possible alternatives with respect to proposed changes in Appendix K. The Committee concluded, as reflected in the transcript at this meeting, that it endorses Alternate 1 at this time.

The ACRS Subcommittee on ECCS will continue to work with the NRC Staff, both in Reactor Regulation and in Safety Research, in their efforts to develop methods for quantifying safety margins which exist in Appendix K and for preserving safety margins taking into account new results which reduce the uncertainties in the data bases. The ACRS Subcommittee will report back to the Committee at an appropriate time (approximately 6-8 months).

R. F. Fraley
Executive Director

cc:
R. Boyd, DPM
E. G. Case, NRR
S. Levine, RES
R. Mattson, DSS
D. Ross, DSS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 14, 1978

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: WESTINGHOUSE CRITICAL HEAT FLUX CORRELATION AND THERMAL DESIGN PROCEDURE

Dear Dr. Hendrie:

During its 215th meeting, March 9-10, 1978, the Advisory Committee on Reactor Safeguards reviewed the changes being proposed for the Westinghouse critical heat flux correlation and the accompanying thermal design procedure. These matters were first introduced in the review of the Donald C. Cook Nuclear Plant Unit No. 2, and the Committee recommended in its December 21, 1977 report that a generic review be completed prior to implementation of this new thermal design analysis. The ECCS Subcommittee met with the NRC Staff and with representatives of the Westinghouse Electric Corporation, in Washington, DC, on February 16, 1978 to discuss the bases for the changes being proposed. The Committee also had the benefit of the documents listed below.

The Committee recognizes that the regulatory process must be responsive to new data and new analyses and that a strengthened technical base may justify some relaxation in previously acknowledged conservative positions. The Westinghouse proposals for a new critical heat flux correlation and for a new thermal design procedure are examples of such an approach. The application of these proposals, which the Committee considers a generic matter, could lead to greater flexibility of plant operations and to higher power densities.

The Committee concurs with the NRC Staff position, noting that a conservative safety margin is still being retained.

The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the conservatisms recommended by the NRC Staff, there is reasonable assurance that the Westinghouse critical heat flux correlation and the accompanying thermal design procedure can be used as a regulatory basis for evaluating nuclear power reactor operations without undue risk to the health and safety of the public.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

REFERENCES:

1. Westinghouse Electric Corporation, "Improved Thermal Design Procedure," WCAP-8567, July 1975.
2. Westinghouse Electric Corporation, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762, July 1976.
3. U.S. Nuclear Regulatory Commission, "Draft of Safety Evaluation of the Westinghouse WRB-1 Critical Heat Flux Correlation," January 1978.
4. U.S. Nuclear Regulatory Commission, "Draft of Safety Evaluation of the Westinghouse Improved Thermal Design Procedure," January 1978.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 8, 1978

Lee V. Gossick
Executive Director for Operations

SUBJECT: NRC OFFICE OF NUCLEAR REGULATORY RESEARCH RECOMMENDATIONS ON ECC BYPASS RESEARCH

Dear Mr. Gossick:

During the 220th meeting, August 3-5, 1978, the Advisory Committee on Reactor Safeguards was briefed by members of the NRC Office of Nuclear Regulatory Research and the Office of Nuclear Reactor Regulation regarding future research on ECC Bypass. The ECCS Subcommittee met with the NRC Staff on July 17-18, 1978, to discuss this program.

The NRC Staff reported its conclusion that a new, large test facility is not now needed for research in this area. This conclusion has taken into account plans to upgrade present programs with improved instrumentation and higher steam pressures, basic physics tests, increased sensitivity studies with TRAC, and data from the Upper Plenum Test Facility in the 2D/3D program. The Committee agreed with this conclusion.

R. F. Fraley
Executive Director, ACRS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 16, 1980

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: ADDITIONAL ACRS COMMENTS ON THE RCP TRIP AND HPI TERMINATION CRITERIA

Dear Dr. Ahearne:

In your letter of April 1, 1980, you requested that we clarify our concerns with the present reactor coolant pump (RCP) trip and the high pressure injection (HPI) termination criterion. You also indicated in a memorandum to R. Fraley on February 22, 1980 that you would welcome our comments on NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors."

The present requirements for RCP trip and HPI termination have developed from the lessons learned from the Three Mile Island accident and from the extensive number of small break LOCA calculations subsequently carried out. There are two distinct requirements in the I&E Bulletins issued, as referenced below, which can be considered separately. The first concerns the directive which requires prompt shutdown of all reactor coolant pumps in PWRs following a depressurization transient which initiates safety injection. The second is the requirement that the safety injection system continue to be operated until a specified degree of subcooling is attained in the primary system.

The prompt reactor coolant pump trip mandated by the Bulletins followed analyses by the vendors of nuclear steam supply systems which seemed to show that there was a "window" of break sizes and pump trip delay times which would lead to calculated peak cladding temperatures in excess of the 2200°F licensing limit. These same methods of analysis indicated that with prompt pump trip the peak cladding temperatures would remain below 2200°F. The NRC Staff prepared a useful critique in NUREG-0623 of these vendor calculations and, while this report clearly presented the deficiencies in the analytical methods used, the report agreed with the vendors' conclusions. The short-term action by the Staff therefore was the requirement of prompt trip of the reactor coolant pumps; as a long-term action the Staff recommended that licensees propose and submit design changes that will assure automatic trip of all reactor coolant pumps.

We do not, at this time, disagree entirely with the Staff's requirement of prompt coolant pump trip, but in view of the analytical limitations upon which prompt trip is based we believe that the emphasis on immediacy of the trip and on eventual automatic trip may not be desirable. Recent experimental data has put doubt on the existence of the "window" which is the basis

for requiring prompt pump trip. Additional experimental data will become available before the end of the year. The prompt trip has been carried out in four transients since the Bulletins have appeared. In none of these was there a LOCA in the primary system; all of these transients arose from disturbances on the secondary side. No significant plant damage ensued in these transients and there was no harm to plant personnel or to the public. There has been complaint, however, that without reactor coolant pump flow the operator loses reactor pressurizer control since, in many PWRs, pressurizer spray flow depends on coolant pump flow. Further, natural circulation must also be established to remove decay heat. It must be said that the Staff's hope to develop a clear distinction between depressurization from a small break on the primary side and depressurization from a secondary side transient seems quite optimistic.

We believe that reactor coolant pump trip upon primary depressurization is an acceptable procedure, but we see no urgency at this time for installation of automatic pump trip. With regard to primary pressure control, we believe that it is desirable to provide pressurizer spray flow which is independent of main coolant pump flow.

The present set of requirements for HPI termination criteria is based upon achieving a specified degree of subcooling in the primary coolant system along with, in some cases, a specified water level in the pressurizer and steam generators. These requirements are intended to prevent a recurrence of the TMI-2 situation in which HPI flow was terminated while still necessary; these requirements, however, do not address the conditions in which HPI should be terminated when not required. We are concerned that relatively frequent system transients which activate HPI might progress to liquid discharge through safety valves or PORVs, valve failure under liquid flow, and a resultant small break LOCA. It should also be pointed out that Westinghouse has recently reported a significant deficiency under 10 CFR 50.55(e) for a number of reactors with high head centrifugal charging/safety injection pumps. Failure to stop these pumps promptly when high pressures are reached could result in pump failure from low flow - a common mode failure of the redundant HPI pumps. Changes in operational procedures may also affect the design limits of other components. These interactions need to be carefully reviewed.

We note that a number of plant transients that have occurred in the past year have been affected by the NRC approved HPI termination and RCP trip criteria. These include events, as referenced below, at North Anna, Unit 1, September 26, 1979; Prairie Island, Unit 1, October 2, 1979; and ANO, Unit 2, January 29, 1980. Some changes have been made in criteria in response to these events. We believe that continued Staff attention in this area is required.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, "I&E Bulletin 79-05A," April 5, 1979.
2. U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, "I&E Bulletin 79-06A," April 14, 1979.
3. U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, "I&E Bulletin 79-06B," April 14, 1979.
4. U. S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, "I&E Bulletin 79-05C and 79-06C," July 26, 1979.
5. NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," November 1979.
6. Letter, C. M. Stallings, VEPCO, to J. P. O'Reilly, NRC, Submitting Licensee Event Report for September 25, 1979 North Anna Number 1 Cooldown Incident (October 9, 1979).
7. Letter, L. O. Mayer, NSP, to J. G. Keppler, NRC, Submitting Licensee Event Report for October 2, 1979 Steam Generator Tube Rupture Incident (October 16, 1979).
8. U.S. Nuclear Regulatory Commission Preliminary Notification of Event or Unusual Occurrence, PNO-IV-80-05, January 30, 1980.
9. Letter, D. C. Trimble, AP&L, to R. W. Reid, NRC, Submitting Startup Report, Supplement 2 for ANO-Unit 2, March 6, 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 9, 1980

Mr. William J. Dircks, Acting
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Dircks:

SUBJECT: CLADDING SWELLING AND RUPTURE MODELS FOR LOCA ANALYSIS - NUREG-0630

During its 245th meeting the Advisory Committee on Reactor Safeguards discussed the final version of the NRC Staff report: "Cladding Swelling and Rupture Models for LOCA Analysis" - NUREG-0630. The Committee initially discussed this item with the NRC Staff at its 235th meeting during November 1979. A draft version of NUREG-0630 was discussed at a combined meeting of the ECCS and Reactor Fuel Subcommittees on February 14, 1980. A final version was discussed at a Subcommittee meeting on Reactor Fuel on September 3, 1980.

Based on research done in the U.S. and overseas, the Staff believes that some of the current vendor evaluation techniques fail to assure that for the fuel cladding in a LOCA "the degree of swelling and incidence of rupture are not underestimated" as required in Appendix K of 10 CFR 50.46. The procedures proposed in NUREG-0630 introduce additional conservatisms by prescribing new curves for clad strain and flow blockage. It is not at all clear that the approach presently used leads to a lack of conservatism in estimating cladding strain that would occur during a LOCA.

At low reflood rates, Appendix K prohibits consideration of the cooling due to water droplets in the steam, when indeed they contribute significantly to the cooling provided. Also, the Staff believes Appendix K requires that clad ballooning must be prescribed so as to provide a negative effect on cooling, when in fact there is clear evidence that it increases cooling in most of the situations of concern. This approach both constrains the application of research results by the Staff and thrusts a strong component of artificiality into the entire proceedings.

In light of the above considerations, the ACRS recommends that implementation of the NUREG-0630 models and other significant changes in the evaluation models be held in abeyance until a thorough revision of the Appendix K requirements can be undertaken by the NRC Staff. This revision should take into account research results so that both the nonconservative and overly conservative aspects of the evaluation model can be properly evaluated.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References

1. U.S. Nuclear Regulatory Commission, "Cladding Swelling and Rupture Models for LOCA Analysis," USNRC Report NUREG-0630, April 1980

2. U.S. Nuclear Regulatory Commission, "Cladding Swelling and Rupture Models for LOCA Analysis," USNRC Draft Report NUREG-0630, November 1979

3. Memorandum for NRC Public Document Room from D.A. Powers, NRC, Subject: Formal Comments on Draft NUREG-0630, dated March 14, 1980

4. Memorandum from R.J. Budnitz, Director, Office of Nuclear Regulatory Research, NRC, to H.R. Denton, Director, Office of Nuclear Reactor Regulation, NRC, dated June 6, 1980, Subject: Comments on Draft NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," D.A. Powers and R.O. Meyer, March 24, 1980

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 13, 1982

Mr. William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

SUBJECT: ACRS COMMENTS ON NRC PROGRAM TO ADDRESS CONCERNS WITH THERMAL HYDRAULIC BEHAVIOR OF BABCOCK AND WILCOX PLANTS DURING TRANSIENTS AND ACCIDENTS

During its 270th Meeting, October 7-8, 1982, the Advisory Committee on Reactor Safeguards met with the NRC Staff and representatives of the Babcock and Wilcox (B&W) Owners Group to discuss NRC Staff concerns regarding the dynamic thermal hydraulic behavior of B&W plants during transients and accidents, particularly small break loss of coolant accidents.

For some time, the NRC Staff has identified a need for experimental data for investigation of specific plant phenomena and for assessment of analytical calculations of B&W plant response to transients and accidents. Recently, a Test Advisory Group composed of NRC Staff members and representatives of the B&W Owners Group was formed to evaluate alternatives available for obtaining the desired test data. The Owners Group has proposed use of two industry test facilities (GERDA and SRI-II) in response to the NRC Staff's concerns.

While we support the cooperative effort between NRC and the Owners Group, it appears that the GERDA and SRI-II facilities as now proposed will be inadequate to satisfactorily address the NRC Staff concerns in this matter. Although the data obtained from these facilities may be useful, we believe that a more adequate facility, similar to the proposed Semiscale MOD-5 configuration, is necessary to address the major operational questions of concern. We also wish to emphasize that the timely acquisition of such data and associated analyses are required in order that NRR can make use of B&W plant accident analyses confidently.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 16, 1984

Mr. William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

SUBJECT: ACRS REVIEW OF GENERAL ELECTRIC SAFER/GESTR-LOCA ECCS EVALUATION MODEL

During its 289th meeting, May 10-12, 1984, the Advisory Committee on Reactor Safeguards completed its review of the General Electric SAFER/GESTR-LOCA revised ECCS Evaluation Model. The ACRS had previously considered elements of this review during its 278th meeting, June 9-11, 1983. Subcommittee meetings were held on this matter on December 2-3, 1982, February 17-18, 1983, and April 24, 1984. During our review, we had the benefit of discussions with representatives of the General Electric Company and the NRC Staff. We also had the benefit of the documents referenced.

The General Electric Company has developed an improvement in the methods used to evaluate the performance of emergency core cooling systems in boiling water reactors (BWRs) of the jet pump design. They have proposed that this improved method be accepted for routine use in demonstrating that BWR cores meet licensing requirements. The NRC Staff has reviewed the proposal and has concluded that the new methodology, with certain restrictions, can be used for licensing. The ACRS has also reviewed the proposal, and we concur with the NRC Staff.

The improved method of analysis takes advantage of data and understanding which have been developed in the past decade regarding thermal-hydraulic behavior during large-break loss-of-coolant accidents (LBLOCA) in light water reactors. It will permit some reduction in the present, unnecessarily large margins from limits on peak cladding temperature required with present methods, while still fully meeting the letter and intent of 10 CFR 50.46 and Appendix K to Part 50. The reduction in required temperature margin will, in turn, permit BWR licensees more flexibility in the operation of these reactor cores and in associated fuel management practices. Significant economies in power generation will result. We believe there is an additional advantage in the improved method in that it provides a more straightforward and understandable assessment of the performance of an important safety system.

The improved ECCS evaluation method uses nominal or realistic values of inputs and correlations, rather than the conservatively biased values which have been used in the past. To meet licensing requirements, a term, "Adder"

is added to the realistic estimate of peak clad temperature (PCT). This Adder incorporates all of the biasing requirements specified in Appendix K in a direct, "nonstatistical" way. A second evaluation of PCT is then made, using the realistic models along with data and evaluations of variability and uncertainty inherent in the analysis to provide an "upper-bound" estimate of PCT. It is estimated that at this value no more than 5 percent of actual PCTs in a population of hypothetical LBLOCA's would be in excess of the calculated PCT. When operation with a given core is approved, NRC will require a demonstration that the PCT value calculated by the licensing method always exceeds the upper-bound value.

The NRC Staff review of this item has progressed in two phases: Review of the SAFER and GESTR-LOCA codes, and review of the application methodology designed to demonstrate conformance with the requirements of 10 CFR 50.46 and Appendix K to Part 50. The NRC had earlier issued Safety Evaluation Reports indicating their acceptance of the SAFER/GESTR-LOCA models with the provision that their use was subject to Staff approval of an acceptable application methodology, and so informed the ACRS. The Staff has now issued an SER which approves the General Electric application methodology for these codes, subject to specified limitations on their use.

The ACRS believes that revision of the ECCS evaluation methodology to permit the expected economies in operation is well justified. We also believe there is a positive contribution to overall nuclear power plant safety with the improved analysis in that it provides a more understandable evaluation of ECCS performance along with more explicit assessment of margins of uncertainty and variability. We support the General Electric revision. We also expect other vendors and licensees to engage in a similar effort in the near future. We believe, however, that the NRC Staff must continue to provide careful review of all such proposals and changes to assure that appropriate overall plant safety margins are maintained.

Sincerely,

Jesse C. Ebersole
Chairman

References:

1. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report on General Electric ECCS Evaluation Methodology," Draft Report dated March 1984.
2. U. S. Nuclear Regulatory Commission, "Emergency Core Cooling System Analysis Methods," SECY 83-472, dated November 17, 1983.
3. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report on the SAFER Code," dated July 12, 1983.

4. U. S. Nuclear Regulatory Commission, "Safety Evaluation of the General Electric Company Topical Report, NEDE-23785-1, Volume 1-GESTR/LOCA, A Model for the Prediction of Fuel Rod Thermal Performance," September 1983, report approved November 2, 1983.
5. General Electric Company Topical Report, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss of Coolant Accident," NEDE-23785-1-P (Proprietary); Volume 1, "GESTR/LOCA - A Model for the Prediction of Fuel Rod Thermal Performance," December, 1981; Volume 2, "SAFER - Long Term Inventory Model for BWR Loss-of-Coolant Analysis," December, 1981; Volume 3, "SAFER/GESTR Application Methodology," March 23, 1984.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 14, 1984

Mr. William J. Dircks
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Dircks:

SUBJECT: GENERIC ISSUE 23, "REACTOR COOLANT PUMP SEAL LEAKAGE"

During its 292nd meeting, August 9-11, 1984, the Committee considered a proposal by the NRC staff to designate Generic Issue 23, "Reactor Coolant Pump Seal Leakage," as an Unresolved Safety Issue (USI). The ECCS Subcommittee had reviewed this proposal during a meeting on July 31, 1984.

Designation of this issue as a USI will give it higher priority with regard to assignment of agency resources and attention from management. This will, we surmise, lead to a more prompt and satisfactory resolution of the issue.

The Committee concurs with the NRC proposal, believing that the issue of pump seal failures in certain types of reactor coolant pumps has important safety implications and should be resolved within a reasonable time. We understand that a new Task Action Plan is to be developed, and we suggest that the following items be considered in that undertaking.

- Seal failures caused by station blackout can cause a LOCA at a time when there is no containment cooling. Particular attention must be paid to whether containment overpressure could result.

- Accurate knowledge of the range of possible seal leak rates will be of great importance in resolving this issue and should be obtained as early as possible. Experimental work may be required, because of the complexity of the flow path resulting from a failed seal.

- Initial information indicates that seal failure may be of significant safety importance only in station blackout scenarios. Therefore, resolution of this USI will have to be carefully coordinated with the resolution of USI A-44, "Station Blackout."

The Committee wishes to be kept informed concerning resolution of this issue and particularly about the development of the Task Action Plan.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 14, 1975

L. Manning Muntzing
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Muntzing:

Several nuclear power plants recently considered or currently under review by the ACRS will be located in states which heretofore have not been the site of a commercial nuclear electric generating station. Examples are the Summit Nuclear Power Station in Delaware, Grand Gulf Nuclear Power Units 1 and 2 in Mississippi, and the Waterford and the River Bend Nuclear Power Stations in Louisiana.

In its review of such plants, one of the items considered by the Committee is the capability of appropriate State regulatory agencies to monitor the facility under normal conditions and to respond with emergency action in case of a major environmental release of radioactive materials. In the examples cited, however, the latest published information (DHEW Publication FDA 75-8006) indicates that the total funds spent on environmental radiation surveillance activities during Fiscal Year 1973 in the three States cited were less than $6,000.

Although hopefully each of these States is now developing a comprehensive environmental surveillance program, the Committee has no factual information to confirm this. In fact, published data show that the total funds being devoted nationwide to environmental radiation surveillance at the State level have decreased during recent years. Since we are sure you share our belief that surveillance programs are an essential part of the spectrum of activities necessary for the demonstration of safe operation of nuclear plants, we wanted to bring this potential problem to your attention and to encourage you to take whatever action you believe is appropriate to assist in promoting the development of a comprehensive environmental surveillance capability in all States, and most especially those in which commercial nuclear power plants are being built for the first time.

Sincerely yours,

/s/ W. Kerr

W. Kerr
Chairman

Attachments:
See Page 2

Attachments:
Pages 39 and 41 from the
U. S. Dept. of HEW, "Rpt of
State & Local Radiological
Health Programs, Fiscal Year 1973"

Reference: U. S. Department of Health, Education, and Welfare, "Report of State and Local Radiological Health Programs, Fiscal Year 1973" DHEW Publication (FDA) 75-0086 (July, 1974).

Table 1. - Total Expenditures[a] for State and Local Radiological Health Activities by Program Area and State, Fiscal Year 1973

State	Total	Basic Planning and Administration	X-ray Survey and Control	Environmental Surveillance	Radioactive Materials	Electronic Products	Other Radiological Health Activities
Total	$8,276.7	$1,366.3	$2,585.6	$1,493.6	$1,512.4	$363.4	$739.9
Alabama	181.0	27.2	61.3	37.3	44.7	3.1	7.6
Alaska	3.3	3.3	0.0	0.0	0.0	0.0	0.0
Arizona	179.9	13.5	45.0	9.0	39.4	5.6	67.3
Arkansas	127.5	12.8	38.3	25.5	38.3	6.4	6.4
California	N/R	N/R	N/R	N/R	N/R	N/R	N/R
Colorado	510.8	52.6	46.9	78.9	86.9	13.7	231.8
Connecticut	96.7	11.6	59.9	13.5	6.8	2.4	2.4
Delaware	20.7	3.1	12.4	2.1	1.0	1.0	1.0
District of Columbia	82.0	19.9	20.3	22.3	6.4	7.1	6.0
Florida	396.4	79.3	79.3	79.3	99.1	39.6	19.8
Georgia	413.2	41.3	103.3	103.3	103.3	41.3	20.7
Hawaii	24.0	3.6	9.6	1.2	3.6	6.0	0.0
Idaho	40.2	6.0	10.1	2.0	18.1	2.8	1.2
Illinois	293.3	55.7	184.8	17.6	8.8	2.9	23.5
Indiana	41.0	10.3	18.5	4.1	2.1	2.1	4.1
Iowa	39.5	3.9	13.8	19.7	1.2	0.8	0.0
Kansas	114.5	17.2	40.1	17.2	28.6	5.7	5.7
Kentucky	107.7	21.5	30.2	21.5	25.9	3.2	5.4
Louisiana	158.7	15.9	47.6	3.2	71.4	4.8	15.9
Maine	27.6	0.7	9.0	12.7	2.2	0.7	2.2
Maryland	259.6	23.4	80.5	90.9	62.3	2.6	0.0
Massachusetts	250.0	27.5	85.0	82.5	40.0	12.5	2.5
Michigan	N/R	N/R	N/R	N/R	N/R	N/R	N/R
Minnesota	56.9	5.1	18.2	31.9	0.6	0.6	0.6
Mississippi	69.7	13.9	31.3	0.3	23.0	0.3	0.7
Missouri	92.2	36.9	36.9	4.6	4.6	1.8	7.4
Montana	19.8	7.0	9.2	1.6	1.0	1.0	0.0
Nebraska	6.6	6.8	21.5	9.1	12.4	1.1	5.7
Nevada	N/R	N/R	N/R	N/R	N/R	N/R	N/R
New Hampshire	50.1	8.4	12.8	17.2	10.1	1.6	0.0
New Jersey	400.0	80.0	80.0	80.0	80.0	60.0	20.0
New Mexico	94.9	20.9	33.1	9.6	7.9	1.2	22.3
New York	1,588.3	254.6	642.4	246.0	266.9	53.9	124.5
North Carolina	186.9	31.8	102.8	28.0	18.7	0.0	5.6
North Dakota	27.2	3.5	9.6	3.8	9.8	0.5	0.0
Ohio	175.6	40.4	101.8	7.0	7.0	3.5	15.8
Oklahoma	116.3	26.7	58.2	17.4	11.6	2.3	0.0
Oregon	184.1	37.0	46.3	52.3	38.4	0.0	10.0
Pennsylvania	439.7	85.5	179.1	66.0	84.7	4.4	20.0
Rhode Island	65.8	19.8	17.5	16.2	1.6	5.7	4.9
South Carolina	233.6	35.0	23.4	116.8	49.1	7.0	2.3
South Dakota	20.6	3.1	14.4	1.0	0.0	2.1	0.0
Tennessee	194.0	50.4	50.4	3.9	35.9	3.9	48.5
Texas	243.7	24.4	48.7	24.4	114.6	24.4	7.3
Utah	38.0	5.3	14.1	4.4	1.8	4.4	8.0
Vermont	60.0	18.0	28.8	6.0	1.8	1.2	4.2
Virginia	77.0	15.4	27.0	27.0	3.9	1.9	1.9
Washington	148.5	34.2	52.0	37.1	22.3	3.0	0.0
West Virginia	34.2	2.8	29.3	0.3	1.0	0.5	0.2
Wisconsin	143.3	20.1	94.6	21.5	1.4	4.3	1.4
Wyoming	19.3	9.7	0.6	5.8	0.6	1.2	1.5
Puerto Rico	69.7	17.4	20.9	10.5	10.5	7.0	3.5
Virgin Islands	2.9	1.9	0.6	0.0	0.3	0.0	0.1

[a] In thousands of dollars.

Table 3. - Radiological Health Personnel[a] Man Equivalence by Program Area and State, Fiscal Year 1973

State	Total	Basic Planning and Administration	X-ray Survey and Control	Environmental Surveillance	Radioactive Materials	Nonionizing Radiation	Other Radiological Health Activities
Total	514.08	63.29	182.99	110.88	96.25	16.69	41.98
Alabama	12.50	1.20	4.55	3.90	2.50	.11	.24
Alaska	.10	.10	0	0	0	0	0
Arizona	6.40	1.05	2.85	.50	1.30	.50	.20
Arkansas	10.50	2.00	2.00	1.50	4.00	.50	.50
California[b]	51.20	2.34	22.53	7.57	16.21	.45	2.10
Colorado	16.20	1.65	1.65	3.15	2.80	.60	6.35
Connecticut	6.15	.65	2.15	2.50	.45	.20	.20
Delaware	2.00	.20	1.40	.20	.05	.10	.05
District of Columbia	5.88	1.43	1.45	1.60	.46	.51	.43
Florida	23.20	1.75	7.10	7.30	4.75	2.05	.25
Georgia	13.05	1.05	4.80	3.00	4.00	.20	0
Hawaii	2.05	.42	.52	.22	.42	.47	0
Idaho	2.00	.20	.55	.10	.95	.15	.05
Illinois	24.00	1.80	18.40	1.20	.50	.10	2.00
Indiana	3.00	.45	1.20	1.10	.05	.05	.15
Iowa	3.32	.45	.99	1.75	.07	.03	.03
Kansas	6.32	1.20	2.75	.83	.92	.20	.42
Kentucky	8.40	1.45	2.15	2.00	2.00	.30	.50
Louisiana	6.30	.60	1.95	.10	2.55	.20	.90
Maine	1.85	.05	.60	.85	.15	.05	.15
Maryland	11.75	.30	4.10	4.00	3.05	.20	.10
Massachusetts	10.25	1.10	3.10	3.75	1.70	.50	.10
Michigan	8.60	1.19	3.52	2.83	.50	.01	.55
Minnesota	10.50	.75	3.70	5.55	0	0	.50
Mississippi	4.28	.89	1.94	.15	1.04	.16	.10
Missouri	3.90	.85	1.95	.45	.30	.11	.24
Montana	.90	.35	.30	.10	.05	.10	0
Nebraska[c]	6.18	.38	2.05	.85	1.80	.52	.58
Nevada[b]	2.00	N/R	N/R	N/R	N/R	N/R	N/R
New Hampshire	3.33	.33	1.40	1.05	.50	.05	0
New Jersey	18.00	1.90	5.15	7.40	1.30	1.55	.70
New Mexico	6.18	1.83	1.50	.70	.30	0	1.85
New York	62.30	11.20	16.25	11.70	14.10	1.90	7.15
North Carolina	18.06	.81	6.75	4.30	1.35	0	4.85
North Dakota	1.70	.22	.54	.52	.40	.02	0
Ohio	11.55	1.55	7.25	.60	.48	.31	1.36
Oklahoma	7.10	1.20	3.68	1.32	.82	.08	0
Oregon	14.50	1.60	3.50	3.65	4.10	0	1.65
Pennsylvania	22.29	4.49	8.26	3.54	4.48	.32	1.20
Rhode Island	3.32	.72	.65	1.47	.06	.22	.20
South Carolina	13.45	.90	2.05	8.30	1.80	.22	.18
South Dakota	1.30	1.00	.10	.05	0	.15	0
Tennessee	12.51	2.00	3.05	.15	4.06	.25	3.00
Texas	18.50	1.50	5.70	.60	7.30	2.00	1.40
Utah	1.90	.30	.80	.25	.10	.25	.20
Vermont	5.33	.98	1.20	2.75	.15	.15	.10
Virginia	4.41	.76	1.55	.95	.35	.15	.65
Washington	7.65	1.10	3.50	1.50	1.50	.05	0
West Virginia	3.50	1.00	2.25	.05	.13	.07	0
Wisconsin	6.00	.60	3.90	1.35	0	.02	.13
Wyoming	1.07	.34	.09	.31	.11	.11	.11
Puerto Rico	7.25	1.05	3.60	1.32	.28	.45	.55
Virgin Islands	.10	.06	.02	0	.01	0	.01

[a] Excludes Federal personnel on loan to States.
[b] 1972 data.
[c] Includes three full time personnel from University of Nebraska.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 8, 1975

Honorable William A. Anders
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: EMERGENCY PLANNING

Dear Mr. Anders:

In the course of its reviews of reactor facilities, one of the items taken into consideration by the Advisory Committee on Reactor Safeguards is the degree to which applicants have prepared a sound plan of action in case of emergency situation. This matter was the subject of meetings of the ACRS Environmental Subcommittee on July 26-27, 1974, and January 17, 1975, and was most recently discussed at the 179th and 180th meetings of the full Committee on March 6-8 and April 3-5, 1975.

On the basis of its evaluations, the Committee has concluded that an effective emergency plan can play a significant role in the protection of the nearby population in the unlikely event of a major accidental release of radioactive material for a nuclear installation. Reviews by the Committee, however, of emergency plans for nuclear power plants currently in operation, or nearing completion of construction, show that much work remains to be done. In this regard, the Committee has identified the following items:

1. Action Criteria

There is a lack of well defined criteria for the initiation of remedial actions. A comparison of emergency plans for various nuclear power plants shows that there are large differences in dose rates at which similar protective actions would be initiated. The Committee notes that the Environmental Protection Agency is developing a set of uniform protective action guides for use by State agencies. The Committee recommends that NRC lend appropriate support to assure that this effort be completed as soon as feasible.

2. Accident Scenarios

Sound emergency planning requires the ability to cope with a wide range of accident situations. Inquiries by the ACRS indicate a lack of development of an adequate series of scenarios to cover the range of emergencies which might take place and of methods for minimizing the resulting consequences. Such scenarios need to be developed and drills incorporating appropriate responses should be conducted. Following such drills, emergency plans should be revised and updated as necessary.

3. Accident Instrumentation

A key factor in the ability to respond intelligently to an emergency is the availability of instrumentation capable of determining the nature of, and following the course of, an accident. Although all plans have useful instruments of one or more types, few plants are equipped with instruments of a sufficient variety and range of applicability. Further work is needed to assure that appropriate instrumentation is provided and that the resulting data are readily available and can be interpreted promptly.

4. State Response Plans

There is a need for further improvement of response plans on the part of State and local agencies who will be responsible for protection of people in the neighborhood of plants. This includes the development of a well defined division of responsibilities and of the coordination required among people responsible for on- and off-site aspects of protective actions. There is a need also for improved coordination of emergency planning for those installations located near the border between States, and for installations located near our borders with Canada and Mexico.

Additional observations by the Committee are that the response plans of many States responsible for dealing with population groups in the neighborhood of nuclear power plants are only in the planning stages or, if completed, show a need for more professional knowledge in this subject area. Compounding these problems is the fact that Federal funds to lend support to the development of State response plans, which the Committee understood were to be made available through the Federal Disaster Assistance Administration, have never materialized. Although there have been several candidate states for the development of model state plans, no such plan has yet been completed.

The Committee is pleased to learn of the recent visits by an interagency training cadre to evaluate the emergency response capabilities of selected States. The Committee recommends that this program be expanded and that the NRC assume a role of leadership in coordinating the necessary efforts to foster the development of adequate State emergency response capabilities.

5. Action Verification

In case of accident response, there is need for the development of means for verifying that protective actions have actually been carried out by responsible individuals. Criteria need also be established for terminating emergency actions. The Committee recommends that the NRC assure that emergency plans include designation of the responsible individuals and the establishment of the necessary critieria for handling these matters.

6. Professional Support

Many key decisions in emergency actions will be dependent upon sound interpretation of environmental-radiation-related information. Reviews of existing emergency plans often reveal a lack of understanding of the importance of professionally qualified radiation protection personnel in such actions. The Committee believes that such personnel must play a key role in the design and implementation of emergency response measures and recommends that greater attention be given to this matter.

7. Federal Emergency Assistance

For a number of years, the United States Atomic Energy Commission has maintained a Radiological Assistance Program through which interagency Radiological Assistance Teams were available for immediate response in emergency situations. The Committee recommends that NRC assure the continuation of this Program.

8. Research and Development

Research and development efforts on several aspects of these problems are underway at several of the National Laboratories and by the NRC and ERDA headquarters staffs. Examples include the Atmospheric Release Advisory Capability (ARAC) at the Lawrence Livermore Laboratory, the Clinch Valley Study at the Holifield National Laboratory, and a rapid version of the Aerial Radiation Monitoring System (ARMS) being considered for development by the headquarters staffs. In addition, studies of the evaluation of risks of evacuating population groups are underway within EPA. Such efforts, however, appear to be in need of improved

coordination to assure that results will be in a form suitable for ready implementation by NRC licensees, State and local units, and appropriate Federal agencies. Also needed are studies to assure that the results of these R&D efforts are disseminated to user groups.

To assure that the best methods for population protection are being incorporated into emergency planning, additional research is needed on appropriate countermeasure actions for reducing the population intake of radioactive materials, particularly through food and milk. Attention also needs to be directed to the long range implications of potential radioactive contamination of land areas with respect to possible methods of cleanup and any necessary restrictions on land use.

Sincerely yours,

/s/ William Kerr

William Kerr
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 8, 1978

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: EVALUATION OF ALTERNATIVE SITES FOR THOSE WITH HIGH POPULATION DENSITIES

Dear Dr. Hendrie:

During its 218th meeting, June 1 and 2, 1978, the Advisory Committee on Reactor Safeguards reviewed the NRC Staff proposal to use the Calculation of Reactor Accident Consequences (CRAC) Code as a tool for the assessment of alternative sites in areas of high population densities. The proposal was also considered during a Subcommittee meeting on May 3, 1978. The Committee also had the benefit of the documents listed.

The studies to date have shown that the CRAC Code can provide additional understanding of the impact on the public health of accident consequences exceeding the limits of 10 CFR 100. However, there are many factors influencing the application of the Code that have an important bearing on the computational results but which the Code does not address adequately. These include regional meteorology (particularly for coastal and river valley sites), plume geometry, and effluent particle size distribution. In addition, the Code does not address the behavior of radionuclides within containment prior to release.

Because of these recognized limitations, the ACRS recommends that caution be exercised in the use of the Code in alternative site evaluations. In addition, efforts should be continued to develop improved input data for the Code.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

References

1. U.S. Nuclear Regulatory Commission Staff Paper, "Assessments of Relative Differences in Class 9 Accident Risks in Evaluations of Alternatives to Sites With High Population Densities," SECY-78-137, dated March 7, 1978.

2. U.S. Nuclear Regulatory Commission, "Reactor Safety Study, An Assessment of Accident Risk in U.S. Commercial Nuclear Power Plants," Appendix VI, WASH-1400, NUREG-75/014, dated October 1975.

3. Wall, I. B., et. al., "Overview of the Reactor Safety Study Consequence Model," USNRC Report NUREG-0340, dated October 1977.

4. Church, Hugh W., "The Atmospheric Dispersion Model as Used in the Reactor Safety Study, WASH-1400," Reprint from "Third Symposium on Atmospheric Turbulence, Diffusion and Air Quality," Raleigh, N.C. (October 26-29, 1976).

5. Acharya, S., "Some Observations on the CRAC Code and WASH-1400," USNRC, dated March 23, 1978.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 6, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON PROPOSED EMERGENCY PLANNING RULE (10 CFR Part 50)

Dear Dr. Ahearne:

The purpose of this letter is to provide you with ACRS comments on the Proposed Rule on Emergency Planning (10 CFR Part 50) as published in the Federal Register (Vol. 44, No. 245) on December 19, 1979. In preparing these comments, the Committee had the benefit of discussions with the NRC Staff on May 1, 1980. The ACRS Subcommittee on Site Evaluation also met with the NRC Staff on April 22, 1980 to discuss this matter.

Subsequent to the meeting on April 22, 1980, the Subcommittee Chairman was informed that the Proposed Rule had been extensively revised by the NRC Staff. However, a copy of this newer version was not made available to either the Subcommittee or the full Committee in time for the preparation of these comments. If you desire, the Committee would be pleased to offer comments on the revised Rule at a later date. Because of scheduling difficulties, the earliest that this could be accomplished would be approximately the middle of July. Although this would probably necessitate a delay in the implementation of the Rule, we believe there are benefits to be gained through additional review.

The ACRS concurs with the NRC Staff view that there is a need to review and upgrade the status of emergency preparedness at commercial nuclear power plants. Those provisions in the proposed regulations that concern definition of roles, identification of proposed actions, and testing of the performance of equipment and personnel are clearly desirable. However, our review of the Proposed Rule has revealed a number of questions and problem areas. The more significant of these may be summarized as follows:

1. The Proposed Rule includes two alternative approaches for implementing the proposed changes. On the basis of clarifications provided by the NRC Staff, the ACRS would endorse Alternative A. In case of problems with State and local government emergency response plans, this Alternative would require action by the NRC to shut down a plant, instead of automatically requiring shutdown under the regulations.

2. The NRC Staff notes in the Proposed Rule that "while emergency planning is important for public health and safety, the increment of risk involve[d] in permitting operation [of existing reactors] for a limited time in the absence of concurred-in plans may not be undue in every case." The Committee agrees with this conclusion but questions whether it is compatible with the assertion that the Commission views "emergency planning as equivalent to, rather than as secondary to, siting and design in public protection ...." Safe day-to-day operation would be impossible without adequate siting and design and proper operation of a safely designed and sited reactor would probably not represent an unacceptable risk for several months and probably years.

   A preferred statement would recognize that siting, design, and emergency planning, as well as responsible operation, are separate but interrelated considerations that constitute the overall safety package. It is not clear that the NRC policy of elevating emergency planning to the same level as engineered safety features is wise or necessary. The role of emergency planning should be defined as supplemental to the decisions to allow operation of a plant.

3. In the Foreword to NUREG-0654 (See Reference 2) emphasis is placed on there being minimum acceptance criteria for emergency preparedness and planning. There are also implications in this report and in the Proposed Rule that these criteria will be made mandatory for licensees and for the acceptability of emergency plans developed by State and local agencies. Insistence on strict compliance with detailed criteria could prevent proper coordination of nuclear power plant emergency planning with other emergency preparedness activities of State and local agencies, and could also delay the modification of specifications for key factors, such as evacuation times and distances, as better information is developed through ongoing emergency planning.

   In addition, the Committee has noted an absence of technical justification for many of the requirements associated with the Proposed Rule and the criteria by which compliance will be judged. If, in the final analysis, a decision is made to retain these criteria in the Rule, then, as a minimum, efforts should be made to test them on a range of nuclear and major nonnuclear accidents that have occurred in the past. Such tests would be particularly useful in showing how successful the specified actions would have been in alleviating the effects of the given events.

4. The Proposed Rule specifies that "the capability will be provided to essentially complete alerting of the public within the plume exposure pathway EPZ within 15 minutes of the notification by the licensee of local and State officials." The ACRS agrees that providing such capability is desirable but believes that emergency plans should reflect the fact that there is less urgency for immediate notification of people living at greater distances from the site and that, in the majority of cases, the promptness of notification should have the important input of human evaluation and assessment. This might be accomplished through application of a graded scale of timing tied into distance, coupled with on-the-spot evaluations of local weather and other conditions. Supporting this approach are the results of recent research which indicate that prompt evacuation of people residing beyond five miles of a site may not be beneficial on a risk assessment basis except under the most unusual circumstances. Furthermore, there is need to consider the possible risks associated with notification of the public prior to the police and other officials being ready and available to direct and control the responses of people residing near a power plant.

5. The Proposed Rule and accompanying proposed criteria request that applicants provide detailed information on evacuation, including "an analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations." In no case, however, does the Proposed Rule provide information as to what times would be considered acceptable, even though, in the case of evacuation, the risks resulting from transportation accidents are often related to the hastiness of the action. As written, the Rule also appears to allow no alternative to evacuation. This implies that the applicant is not likely to be permitted to provide a better alternative, such as having the population remain indoors while the plume passes. This is a situation that reduces itself to the now familiar issue of specifying "how to" rather than providing the desired goal and allowing the licensee or State government to seek the best solution. In some locations, evacuation from the plume Emergency Planning Zone is obviously impractical. If evacuation is to be the favored emergency planning alternative, this choice and the requirements for it should be well-substantiated.

6. The Proposed Rule calls for "the yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information such as the possibility of nuclear accidents, the potential human health effects of such

accidents and their causes, methods of notification, and the protective actions planned if an accident occurs...." Although the last two of these items appear reasonable, the ACRS suggests that the dissemination of information of the types described in the first two items cannot be expected to provide any improvements in emergency preparedness. The Committee therefore recommends that these two items be deleted.

7. The Proposed Rule specifies that exercises to test the adequacy of an emergency plan should be conducted at a frequency of once every three or five years. Because of the rapid turnover in staff personnel at all levels in all the organizations involved, the ACRS recommends that such exercises be conducted at three-year intervals. The Committee also urges that the exercises be utilized for purposes of instruction as well as for evaluations of compliance.

   Although the Proposed Rule calls for licensees to provide an independent review of their emergency preparedness program every twelve months, no mention is made of participation by State and local authorities. This omission should be corrected.

8. One alternative in the Proposed Rule requires that corrective measures to prevent damage to onsite and offsite property be identified. The ACRS believes that protection of property is less important and less feasible than protection of health and safety and, in fact, may divert effort from the latter aspect. The Committee recommends therefore that this requirement be omitted from the Rule.

9. As written, the Proposed Rule will require in-depth discussion and subsequent concurrence in the emergency preparedness program by the applicant and the NRC, as well as by State and local governmental authorities. The ACRS is concerned that this could constitute a third-party veto of the operation of a nuclear power plant based on considerations that may be unrelated to health and safety. The ACRS believes that such a requirement should not be included in the Rule without some safeguards against such action by a third party. Furthermore, a de facto veto power on operation appears to exist with each local government entity within ten miles of a nuclear power plant if it chooses not to permit establishment of the warning facilities required to meet the criteria. If the Proposed Rule poses such a possibility, it introduces complex societal issues. The ACRS recommends that the wording of the Rule be altered to permit the NRC sufficient flexibility to cope with this situation and not mandate such power to local governmental entities in the absence of a Federal law addressing the matter.

10. The ACRS would also like to comment on the role of the Federal Emergency Management Agency (FEMA) as related to the Proposed Rule. Although the NRC Staff stated that FEMA would simply notify them of their decision relative to the adequacy of a State and local emergency plan, a nonconcurrence on the part of FEMA might also represent a "veto" action on a given application. There are also questions as to the adequacy of the resources or the staffing of FEMA to assume these new responsibilities. In addition, the ACRS sees a need for clarification of its future role relative to FEMA and to reviews of emergency preparedness planning for nuclear facilities.

11. In a sense, the NRC is serving as a pioneer in the area of emergency preparedness. It should be recognized that there are many other technological aspects of society which pose hazards comparable to, or larger than, those from nuclear power plants. FEMA is in the process of developing guidance with regard to emergency preparedness in a general way; however, the rate of implementation proposed for nuclear plants by this Rule appears to be much more rapid, and the requirements possibly more stringent than those required for other types of facilities. The Committee believes that the NRC-FEMA approach to emergency preparedness for nuclear reactor accidents should be developed and implemented within the framework of a broad societal approach to emergency situations in general.

The Committee will be pleased to discuss the above items with you at your convenience. In the meantime, we trust these comments will be helpful to you and the NRC Staff.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. Proposed Emergency Planning Rule, Federal Register Vol. 44, No. 245, December 19, 1979.
2. NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," January, 1980.
3. NUREG-0628, "NRC Staff Preliminary Analysis of Public Comments on Advance Notice of Proposed Rulemaking on Emergency Planning," January, 1980.
4. NUREG/CP-0011, "Proceedings of Workshops on Proposed Rulemaking on Emergency Planning for Nuclear Power Plants," January, 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 11, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON DRAFT FINAL RULE ON EMERGENCY PLANNING (10 CFR PART 50 AND PART 70)

Dear Dr. Ahearne:

During its 242nd meeting, June 5-7, 1980, the ACRS discussed with the NRC Staff the draft copy of the Final Rule on Emergency Planning (10 CFR Part 50 and Part 70). The ACRS Subcommittee on Site Evaluation also met with the NRC Staff regarding this matter on May 21, 1980.

These discussions and our review of the draft copy of the Final Rule show that the NRC Staff has seriously considered all, and has incorporated many, of the suggestions offered in our report to you of May 6, 1980 on the Proposed Rule. The Committee very much appreciates this response on the part of the NRC Staff and believes that the draft Final Rule represents a significant improvement over the Proposed Rule.

Nonetheless, there are several remaining issues on which the Committee would like to offer further comments. These are summarized below:

1. The Committee continues to believe that the NRC-FEMA approach to emergency preparedness for nuclear reactor accidents should be developed and implemented in a manner so as to encourage State and local agencies to incorporate these efforts into their plans for coping with all types of emergency situations. For example, plans for responding to nuclear reactor accidents could be combined with those for other emergencies. The Final Rule, as currently drafted, does not reflect this approach.

2. The draft Final Rule requires that State and local governmental authorities "have a capability for notification of the public (within the plume exposure EPZ) during a serious reactor emergency with a design objective of completing the initial notification within 15 minutes." Although the ACRS agrees that providing such a capability is desirable, the Committee continues to believe that emergency plans should reflect the fact that there is less urgency for immediate notification of people living at greater distances from the site. Although the NRC Staff has, to some degree, incorporated this thought into the supplementary material accompanying the draft Final Rule, we believe it is of such importance that it should be included in the Final Rule itself.

3. The draft Final Rule requires that licensees submit to the NRC the emergency response plans of responsible State and local governments and that these plans be judged acceptable by the FEMA and the NRC before a commercial nuclear power plant is permitted to operate. The Committee notes that this situation gives veto power on the operation of nuclear power plants to State and local agencies. The Committee would appreciate clarification of the intentions of the Commission in this regard.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 16, 1981

MEMORANDUM FOR: W. J. Dircks
Executive Director for Operations

FROM: J. C. Mark, Chairman, ACRS

SUBJECT: EMERGENCY PLANNING AT NUCLEAR FACILITIES DURING AND FOLLOWING NATURAL EVENTS

In its review of nuclear power plants, particularly those in highly seismic areas, the ACRS has raised several questions regarding emergency planning and preparations, and the degree to which they take into account natural events such as earthquakes. For example, will systems and features used to alert the public to an accident in the plant, the equipment to monitor environmental conditions, the communications network, and emergency evacuation routes be available during or following a major earthquake.

The Committee recommends that the NRC Staff give further consideration to the development of emergency plans and the operability of equipment needed to deal with nuclear emergencies which result from natural disturbances.

In this connection, the Committee suggests that in the planning of exercises for the NRC Emergency Operations Center, consideration should be given to postulated scenarios including natural catastrophies such as earthquakes and related plant failures, particularly the failure of equipment or systems that are not Seismic Class I and the impact of such failures on systems important to safety.

It would seem advisable also that the FEMA be requested to review the capabilities of local emergency and disaster organizations to cope with multiple emergencies such as might result from a major earthquake followed closely by an accident at a nuclear power plant.

cc:
Commissioners
S. Chilk, SECY
H. Denton, NRR
V. Stello, I&E

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 11, 1982

The Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

Subject: ACRS REPORT ON EMERGENCY RESPONSE CAPABILITIES AT NUCLEAR POWER PLANTS

During its 265th meeting, May 6-8, 1982, the ACRS reviewed the subject matter described in SECY-82-111, "Requirements for Emergency Response Capability." This matter was considered in meetings of an ACRS Subcommittee on January 5, March 17, and May 5, 1982. During this review we had the benefit of presentations by the NRC Staff as well as by representatives of the nuclear power industry and interested organizations.

We have concluded that the overall plan outlined in SECY-82-111 to bring about improvements in emergency response capabilities at licensed nuclear power plants is generally sound and should be adopted. We have concerns however about several issues.

Analysis and review of the TMI-2 accident have identified many improvements that could be made in the design and operation of nuclear power plants to provide for better control of abnormal conditions and management of accidents. Since 1979, this has resulted in a growing list of proposals for added requirements to be placed on licensees and applicants. This list has not been well defined and has included inputs from varied elements of the NRC Staff with interrelated and overlapping interests. Uncertainty about what final requirements would eventually evolve from this body of information has proven to be an impediment to actual plant improvements in emergency response capabilities. Licensees and applicants have been understandably reluctant to proceed with improvements until they could develop some understanding of what new requirements might be forthcoming.

In the face of failure of the more traditional regulatory practice of developing and promulgating requirements that are as explicit as possible, we believe the NRC Staff has successfully integrated the many proposals for emergency response improvements and proposed a process to assure their implementation. As we understand the process, the licensees will be told that they will be required to adopt, in some form, the several new systems

and operating practices described in SECY-82-111. The many NUREG reports and Regulatory Guides referenced in this document will serve only as guides, not firm requirements. Using NRC Project Managers with technical support from branches in the NRC Staff, the NRC will then negotiate with each licensee a detailed agreement on how the several general improvements will be incorporated in each specific plant. This negotiated, detailed agreement will then become a set of requirements for that particular plant. While we believe this process can be made to work, it does place a heavy burden of responsibility on the Project Managers. It should be constantly monitored by NRC management.

The following matters should be given additional NRC Staff attention.

- The timing for implementation of new Emergency Operating Procedures (EOPs) and the Safety Parameter Display System (SPDS) appears to be a problem. Apparently, EOPs will be written with the SPDS as an important element of the control room information management system. New EOPs are scheduled, we understand, to be put into use in most plants later this year. SPDS installation in plants is two or three years in the future. There may be a temptation to delay upgrading of EOPs pending completion of SPDS installations. We believe use of new EOPs is very important and should be implemented without delay, if necessary, using non-SPDS versions in the interim.

- The SPDS has been singled out by the NRC Staff to be implemented on a higher priority than some other elements of the program. This is because of the Staff's judgment that the SPDS will be highly beneficial in reducing the type of operator error which contributes most significantly to risk. There is some opinion that SPDS implementation should not be singled out but should evolve out of the more general control room information management evaluation and upgrading described in NUREG-0700, "Guidelines for Control Room Design Reviews." While we have some sympathy with this view, we believe the SPDS implementation should proceed as the Staff suggests.

- We believe that implementation of the SPDS should not be forced at a rate which will preclude its orderly development. The SPDS has considerable potential as a diagnostic tool to assist operators in the effective management of a wide range of possible abnormal occurrences. We believe the industry should be encouraged to develop designs which can be expanded to incorporate confirmatory and diagnostic functions and should be permitted the flexibility to do this.

- We suggest that additional attention be given to some specification of reliability for the SPDS. We encourage the industry, in cooperation with the NRC Staff, to develop appropriate standards.

We are skeptical about the need for the comprehensive analysis of control room information management systems called for in NUREG-0700. We believe that many of the benefits coming from such a review will be in control board improvements which will tend to raise the "skill-based" performance of operators. However, our understanding of the contribution of operator error to risk is that cognitive error is the major factor. Because resources are limited, we recommend that priorities be assigned to the guidelines of NUREG-0700 so that improvements that provide important risk reduction will be emphasized.

With the exceptions noted above, we endorse the program described in SECY-82-111 and urge its expeditious implementation. We would like to be kept informed about progress of the program.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

April 18, 1963

Honorable Glenn T. Seaborg
Chairman
U.S. Atomic Energy Commission
Washington, D.C.

Subject: TESTING OF ENGINEERED SAFEGUARDS

Dear Dr. Seaborg:

In 10CFR100, the Atomic Energy Commission established criteria according to which sites for nuclear reactors could be assessed from the standpoint of potential hazards. The methods used in this and in the related Technical Information Document, TID-14844, lead to the establishment of an exclusion area, a low population zone, and a population center distance, based on estimates of the effects of radioactivity releases following an extreme accident to the reactor.

The increasing pressure to locate nuclear power plants near population centers requires that designers of these plants incorporate in their designs engineered safeguards to reduce the magnitude of postulated radioactivity releases.

The Advisory Committee on Reactor Safeguards is of the opinion that, as a principle, where such engineered safeguards are incorporated in plants, they should be so designed and built that their reliability can be tested continuously or at least frequently under design conditions, or under conditions such that conservative extrapolation to design conditions can be made. This would insure that no deterioration occurs that could lead to increases in the postulated effects of accidents.

Sincerely yours,

/s/ David B. Hall

D. B. Hall
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

February 25, 1964

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: SEISMIC DESIGN OF NUCLEAR POWER PLANTS

Dear Dr. Seaborg:

The question of appropriate seismic design of nuclear power plants has recently occupied the attention of the Advisory Committee on Reactor Safeguards. Most of this study has been called for because of proposals to construct nuclear power plants in seismically active zones. There have, however, also been questions relative to plants to be built in regions where seismic activity is low or where the seismic history is based on uncertain documentation of earth movement that took place long ago.

The Committee believes that, especially in cases of the latter kind, uncertainties in the geology or the seismicity may cause a trend toward choosing design methods or design parameters with more conservatism than is necessary. Adequate seismic design guides that provide protection comparable with that provided by other safety features of nuclear plants are needed.

Before seismic design guides or parameters are adopted for zones of relative seismic quiet, the Committee suggests further consideration of the question of earthquake probability and magnitude for these zones, and of the question of adequate engineering for small earth movements.

Sincerely yours,

/s/ Herbert Kouts

Herbert Kouts
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

April 9, 1964

Honorable Glenn T. Seaborg
Chairman
U.S. Atomic Energy Commission
Washington, D.C.

Subject: REACTOR CONTAINMENT TESTING

Dear Dr. Seaborg:

The Advisory Committee on Reactor Safeguards has met with representatives of the AEC Regulatory Staff to discuss the problems related to testing of reactor containments. A survey report prepared by the staff has been reviewed by the Committee. The Committee's conclusions concerning promulgation of this report are contained in the attached letter to the Director of Regulation.

Sincerely yours,

/s/ Herbert Kouts

Herbert Kouts
Chairman

Attachment:
Letter to Mr. H. L. Price dated April 9, 1964.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D.C.

April 9, 1964

Mr. H. L. Price
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REACTOR CONTAINMENT TESTING

Dear Mr. Price:

On March 21, 1964, a subcommittee of the Advisory Committee on Reactor Safeguards met with representatives of the Regulatory Staff to discuss the problems in testing reactor containments. In particular, the document entitled "A Survey of Leakage Characteristics of Steel Containment Vessels for the Purpose of Evaluating Leakage Criteria and Testing Requirements" by R. R. Maccary, dated December 1963, was reviewed.

The first portion of this document contains a careful review of containment vessels used in existing reactors, including the major problems of design, construction, testing and maintenance. The Committee believes that this portion of the document is a valuable assembly of useful information and suggests that this information be made available in the form of a technical report.

The last part of the report proposes criteria for use in connection with testing of containments as engineered safeguards for reactor installations. The Committee is in general agreement with these criteria. The Committee suggests that these criteria be made generally available in some form for constructive suggestions and opinions from the interested public in order to prepare the way to their practical application.

Sincerely yours,

/s/ Herbert Kouts

Herbert Kouts
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 18, 1964

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON ENGINEERED SAFEGUARDS

Dear Dr. Seaborg:

In response to the request of the Commission, the Advisory Committee on Reactor Safeguards has reviewed the question of "engineered safeguards" as they are used now or are currently considered for use in nuclear reactor power plants. This letter summarizes the results of the Committee's reviews.

The number of sites economically suitable for the location of large nuclear power reactors is limited. Apart from safety considerations, the chief requirements are (a) a sufficient cooling water supply must be available for the turbine condensers and (b) the cost of electrical transmission from the reactor plant to the area of use must be kept low. The first makes it necessary to select sites bordering on large rivers, large lakes or the seacoast; while the second makes it desirable to locate as close as feasible to civic or industrial centers. Presently stated site guides do not favor selection of many economically desirable locations.

The health and safety of the public are assured if fission products are retained within the reactor or its containment. To accomplish this aim, a strong program stressing nuclear reactor safety has been adopted in the United States. Strenuous effort is made to be sure that accidents will not occur. This effort starts in the original design groups and is carried along through the construction and operation phases with reviews and approvals required at appropriate intervals. It is the intent of this effort to eliminate reactor accidents. However, since the field of nuclear reactor technology is new and since there may still be some hazards which have been overlooked, dependence must also be placed upon safeguards engineered specifically to limit the consequences in the unlikely event of a major accident, or to prevent such an accident.

In each case, the engineered safeguard must be considered with respect to the specific nuclear plant for which it is intended. It is equally important to assess the degree of confidence that the safeguard will function properly in an emergency. An engineered safeguard that must remain operable for the life of the plant but cannot be tested without ruinous effects would not usually be accorded the same importance as one that can be tested periodically.

It is important to recognize that engineered safeguards are designed to allow the siting of reactors at locations where, without such safeguards, protection of the public would not be adequate. The advantages of a remote site cannot be exactly balanced by engineered safeguards. On the other hand, the advantages of a remote site may be temporary, if appreciable increases in population density occur near the reactor. Few sites presently in use are such that some engineered safeguards are not desirable. Thus, the protection of the public ultimately depends on a combination of engineered safeguards and adequate distances. Engineered safeguards which can justify decrease of the distances must be extraordinarily reliable and consistent with the best engineering practices as used for applications where failures can be catastrophic. To be worthy of consideration, engineered safeguards must be carefully designed, constructed and installed, equipped with adequate auxiliary power, and continuously maintained. Certain designs are based on sound engineering principles supported by materials acceptance tests; others require developmental and proof testing. In any case, provisions for regular and careful testing are required where deterioration may be expected. The acceptance of engineered safeguards to mitigate unfavorable aspects of reactor sites should continue to be based on positive evidence that these design objectives can be attained. In addition, there will probably be a continuing need to develop new devices and design concepts as reactors are proposed for less and less remote sites.

The principal engineered safeguards include: (1) containment and confirement, (2) containment pressure and temperature reducing systems, (3) air cleaning systems to remove fission products, and (4) core spray and safety injection systems. Discussion of the important features of these is appended hereto, as are conclusions regarding their value.

An extract of the conclusions is as follows:

1. When competently designed, constructed, maintained, and tested, containment and certain confinement systems are considered to be effective safeguards.

2. With careful attention to the containment vessel and to the external connections, such as the use of reliably actuated valves in series in primary system lines and a means of preventing pressure buildup from afterheat, a pressure suppression method is an acceptable engineered safeguard against primary coolant line rupture.

3. A building spray system can be a valuable adjunct to containment, if it is carefully designed and maintained, if it has adequate water supply and backup emergency power, and if it is periodically tested for proper functioning of its components. Conservative estimates of the ability to reduce containment pressure are then considered acceptable.

4. With the same reservations as are applied to building spray systems, heat exchange methods of limiting containment pressure buildup are considered to be acceptable engineered safeguards.

5. Because it has not been shown that all accidents which could occur in a reactor are of a sequential nature involving first, a release of stored energy, and later of fission fragments, and because the engineering design of pressure control systems is in the process of development, a confinement system based on sequential release is not generally considered to be of sufficient reliability for use as an engineered safeguard in populated areas.

6. The nature of fission product releases to be expected in the unlikely event of a major accident is not yet well enough known to permit more than conservative lower bounds on the efficiency to be determined for air cleaning systems. Reliable lower limits may, however, be assumed when individual cases are reviewed.

7. Core spray and safety injection systems cannot be relied upon as the sole engineered safeguards in a nuclear power plant. Nevertheless, prevention of core melting after an unlikely loss of primary coolant would greatly reduce the exposure of the public. Thus, the inclusion of a reactor core fission product heat removal system as an engineered safeguard is usually essential.

It is the opinion of the Advisory Committee on Reactor Safeguards that the including of properly engineered safeguards in reactor plants can

permit the reduction of distances required for protection of the public, and that engineered safeguards of selected type should make feasible the siting of power reactors at many locations not otherwise considered as suitable.

Sincerely yours,

/s/ Herbert Kouts

Herbert Kouts
Chairman

Report on Engineered Safeguards
Addendum #1

CONTAINMENT AND CONFINEMENT

Containment is designed to prevent the escape of injurious amounts of fission products resulting from any failure in a nuclear system. In the usual plant, fission products are held within sequential barriers: first, in the fuel; second, inside the fuel cladding; third, within the primary cooling system; and fourth, in the plant containment or confinement system. This discussion is directed primarily to the last, implying that failure of the first three to hold fission products must usually be considered credible.

The term containment or confinement as used here will mean the containment shell or structure, and its accessories as described later. Containments are intended to hold the fission products and primary coolant which may be released, except for inherent leakage. Confinements usually allow the materials discharged by the accident including noble gases to be released in a directed manner during which time most of the radioactive halogens and particulates may be trapped and retained.

Various versions of these systems are given below, with the expected performance in retaining fission products in the unlikely event of a major accident:

1. Double containment, i.e., two envelopes, in which leakage through the inner may be (a) pumped back to the contained space, or (b) directed to a stack with or without air cleaning. Fission products released may be held below a very small fraction per day of the content of the containment gas.

2. Single containment of the nuclear system including the entire primary coolant system. Fission products released might vary from a fraction of a percent to several percent per day of the containment contents.

3. Single containment of the nuclear system except that, as in some boiling water reactors, the primary coolant may pass outside the containment to a turbine and condenser and be returned. In these partial containments, the isolation valves in the

coolant lines are essential parts of the containment. Expected fission product release may be about as in (2) above.

4. Confinement consists of a structure from which gases and vapors may be drawn by a blower through air cleaning systems and vented. Since air cleaning systems ordinarily do not remove noble gases, as much as half of the radioactive content of the gases in the confinement might be released. Confinement systems may include designs for release of the large amount of energy in non-radioactive vapors and gases, such as pressurized water, before release of fission products from fuel cladding, e.g., in case of failure of a pipe in the primary cooling system.

Design maximum pressure in the containment may vary from inches of water to several atmospheres. The containment or confinement structure may vary from an industrial-type building which is nominally gas tight, to a steel or reinforced concrete pressure container. Concrete without lining is usually not impervious enough to act as a high pressure container.

There are various codes applicable to the design, construction and test of containment structures such as the Uniform Building Code, and those of the American Society of Mechanical Engineers and the American Concrete Institute. Codes give the minimum requirements for design, fabrication and testing. Mere compliance with appropriate codes may be inadequate for this nuclear service. Competent designers will be aware of such additional needs.

There are many accessories and auxiliaries that if incorporated must be considered as integral parts of the containment structure. A representative list includes the following with parenthetical notes of some of their important features:

(a) access openings and closures including air locks, (seals; airlock door interlocks; certainty of door closure during reactor operation),

(b) vacuum relief valves or systems (assurance of closure when needed; safe disposal of effluent; protection in case of relief failure),

(c) ventilating air inlet and outlet isolation valves (double valving and last-ditch manual control),

(d) penetration seals for pipes and electric and control leads (age deterioration of sealing compounds),

(e) valves for closing pipes penetrating the containment (double valving; supports to prevent mechanical failure at the penetration),

(f) thermal insulation to enable keeping the temperature of steel containment vessels within the ductile range,

(g) protection against missiles,

(h) special protection against earthquakes, tsunamis, high winds, etc.,

(i) essential electric power, water and compressed air supplies and emergency back-up for them (alternate automatic operating power sources; controls for valving and auxiliaries with last-ditch manual control; suitable operating controls for auxiliaries in emergencies), and

(j) chutes or canal gates for fuel handling.

Testing of the containment and of its accessories and auxiliaries is the only assurance of their capability and reliability. Initial tests for mechanical strength after construction are routine. Leak testing involves detection of the leaks and the determination of total leakage rate. Since many leaks, such as those through access seals and seals around electrical leads, may increase as the sealant ages, periodic testing is necessary. Leaks through metal cracks or fissures do not generally show increased flow with pressure other than that expected from hydro-dynamics. Sealed joints, however, may be tight at low pressure and leak at higher pressure. Testing these below service or design pressure is not always definitive. In many cases, double seals at penetrations enable testing of these seals without pressurizing the entire containment. This is a desirable and effective test of such seals, but it tells nothing about the rest of the containment.

Increased leakage rate of the composite containment during service is an indication of deterioration. Continuous monitoring of leakage where possible is desirable as a check against misoperation. Careful design and testing of gas handling equipment is required.

Containment effectiveness is greatly dependent on administrative control. This includes assurance that operations such as the closing of openings and valves and the maintenance of seals, valves, closure operating mechanisms, controls and auxiliary services are perfectly carried out. Follow-up checks of critical operations and independent re-checks after all maintenance is necessary and should be made a matter of written record. Periodic inspection of operability of apparatus and control and sensing mechanisms should be made and recorded.

The type and design of each containment must be dependent on the needs of the nuclear plant of which it is a part. The amount of leakage permissible depends on the type, power, design and operation of the plant, as well as the nature of the site. The containment is a machine and must be designed, constructed, operated, checked, serviced, repaired and maintained as one. Since safety of many people may depend on its performing as intended in an emergency, each of these operations should be better done than comparable non-nuclear operations.

When competently designed, constructed, maintained, and tested, containment and certain confinement systems are considered to be effective safeguards.

* * *

Report on Engineered Safeguards
Addendum #2

PRESSURE REDUCTION SYSTEMS

Containment systems for water reactors (boiling and pressurized) are required to maintain their integrity upon release of the energy evolved in the unlikely event of nuclear excursions, metal-water reactions, and complete primary system depressurization as might occur following a major rupture. The principal contribution of energy, and hence containment system pressures, would result from the flashing of primary system water. Several methods may be employed to remove heat and thus to reduce containment design size and pressure.

One means of pressure reduction, known as the pressure suppression system, involves use of a large reservoir of water through which all contained vapors must pass via multiple submerged vents. Steam, the principal contributor to containment pressure, is condensed in passage through the water, and non-condensible gases are cooled. This method has been specifically adapted to the direct cycle boiling water reactor in which only the reactor vessel and a few auxiliaries and connecting piping are contained. The method is dependent on maintenance of water level in the suppression pool, and on reliability of valving in steam, feed water and other piping penetrating the containment. The entire method has not been tested under accident conditions, but elements have been satisfactorily tested under simulated accident conditions. Since reliability of the method does not depend on action of personnel or mechanical components, pressure suppression performance tests do not appear necessary. With careful attention to the containment vessel and its external connections, such as the use of reliably actuated valves in series in primary system lines and a means of preventing pressure buildup from afterheat, the pressure suppression method provides an adequate engineered safeguard against primary coolant line rupture.

A second method of pressure reduction involves the use of a spray system within a containment structure. This system is effective in condensing water vapor and removing sensible heat from contained gases. It may be actuated either automatically or manually. The effectiveness of a spray system must be carefully evaluated for each application, using a conservatively calculated sequence of events. Testing under accident conditions is usually not practical due to potential damage to electrical and control equipment. A building spray system can be a valuable adjunct to containment, if it is carefully designed and

maintained, if it has adequate water supply and backup emergency power, and if it is periodically tested for proper functioning of its components. Conservative estimates of the ability to reduce containment pressure are then considered acceptable.

A third method of pressure reduction involves the removal of vapor space heat by heat exchangers (see discussion under Air Cleaning Systems). Heat generated within the containment during normal operation as well as heat produced following an accident may be controlled or removed by air recirculation. Such systems usually consist of multiple fans and heat exchangers. Heat accumulated in the heat exchangers may be transferred to the normal coolant systems, or to a separate or alternate supply of coolant water. The use of such systems during normal operating times gives a higher assurance of availability under accident conditions. Back-up coolant supplies and alternate or parallel power supply are essential to make certain such units will function adequately and for the necessary time after an accident. The system is of greatest value in maintaining reasonable pressures and temperatures in containment structures for long periods following accidental releases. With the same reservations as are applied to building spray systems, heat exchange methods of limiting containment pressure buildup are considered to be acceptable engineered safeguards.

A fourth method of pressure control involves release of water vapor to the environment following a system rupture, and then establishing containment prior to core melting and fission product release. The accident for which this system is designed consists of a loss of coolant in which the steam release precedes the fuel melting by a predictable period. This system requires the use of very large dampers which must close tight, necessary reliable administrative control, and a proper understanding of the accident conditions. Such a system, at the present time, is not considered of sufficient reliability as an engineered safeguard for use in populated areas.

* * *

Report on Engineered Safeguards
Addendum #3

AIR CLEANING SYSTEMS

The function of an air cleaning system as an engineered safeguard is to remove and to retain fission products from an unlikely partial or total fuel melt-down. Fission products that are thus retained should be fixed in a form that prevents redispersion. Components of the air cleaning system should be so located that decontamination and essential handling can be accomplished readily and without hazard to the health and safety of the public.

For cleaning or decontamination purposes, the released fission products from a reactor fuel melt-down may be divided into four groups. These are: the noble gases (krypton and xenon); the halogens (bromine and iodine); volatile solids (such as tellurium, selenium, cesium and ruthenium); and other solids (primarily strontium, yttrium and barium).

Because of their chemical nature and short half-life, radioactive noble gases can usually be treated only by containment or controlled release from elevated locations such as tall stacks. Therefore, with noble gases consideration must be given to meteorological dispersion and dilution as influenced by characteristics of the surrounding environment.

It is convenient to divide the remaining fission products into two physical groups -- gases and particulates. Gases (essentially iodine and bromine in elemental form) are removed by adsorbents such as activated charcoal, by chemi-sorption on silver, or by absorption in a reactant solution. Particulates which range in size from several microns down to less than 0.1 micron can be removed by impingement scrubbers, electrostatic precipators, or filters. The final device in a particulate cleaning train is usually one based on mechanical filtration principles. Containment spray or "dousing" systems for condensing steam may also serve as decontamination systems because of the gas contacting and impingement action of the spray droplets. The multiple contacting which is possible within a contained gas volume makes a containment vessel with sprays equivalent to a scrubbing chamber.

Halogen gases may be adsorbed upon the surfaces of released particulates and may react with them. Hence, it is necessary to use a combination of adsorption, absorption, and filtration devices to remove halogens. Because iodine may occur in both inorganic and organic states, the gas cleaning system must be capable of removing both. Since the halogens

adsorbed on particulates are not irreversibly bound, it is necessary to follow the filter with an adsorber. A liquid scrubber should be followed by both when maximum decontamination is necessary.

The air cleaning components of a reactor safety system include: a ventilation system, a heat removal device, air cleaners, and an air mover with motor. The system must be capable of working continuously in hot, saturated steam environments for a period of time long enough to remove the required portion of the released fission products from the containment or confinement system. To handle the anticipated release, the air cleaning system must have sufficient capacity in flow, in adsorbent and chemically reactive materials, and in filtration surface. Adsorption and filtration systems must be designed and installed so that the decay heat of collected fission products will not cause combustion or destruction of their media or overheating to the point where collected fission products will be redispersed. The media must also be protected against shock waves, missiles, moisture entrainment, liquid slugging, and radiation damage, as well as corrosion and chemical attack. The duct work and filter housing should be protected against mechanical injury or missile damage to avoid bypassing or leakage of untreated air. The system should be leak tested at the same pressure differential as it would have to endure under accident conditions.

Because electrical power is necessary for circulating and recirculating both air and water, it is essential that backup power be available for maintaining minimal flow rates.

The decontamination efficiency required of an air cleaning system will depend on whether the system is once-through or recirculating. The decontamination factors needed will be based on the dose to the environment and the dilution to be assumed for stack dispersion. In recirculating systems, the decontamination factor is related to the number of containment volumes passed through the cleanup system. Decontamination factors of 10 to 1000 or more may be required in most applications.

Gas leaks which bypass filters or adsorbers in effect decrease air cleaning efficiency. When an iodine removal efficiency of 99% is projected, a bypass or leak around the beds of 1% leads to double the iodine release. Appropriate design and testing of associated gas handling equipment is required.

The reliable performance of an air cleaning system must be assured by frequent "in-place" testing which includes monitoring with gases and

particulates that simulate the expected fission products. Ease of testing for leaks, and access for inspection of seals, gaskets and clamps are necessary. A continuous monitor of resistance or pressure drop through the cleaning system is desirable where the decontamination unit is always in use. In the case of emergency or standby cleaning systems, at least quarterly operation and checking of both air mover and cleaner is desirable.

The nature of fission product releases to be expected in the unlikely event of a major accident is not yet well enough known to permit more than conservative lower bounds on the efficiency to be determined for air cleaning. Reliable lower bounds may, however, be assumed when individual cases are reviewed.

* * *

Report on Engineered Safeguards
Addendum #4

CORE-SPRAY AND SAFETY-INJECTION SYSTEMS

The terms core-spray and safety-injection are used to designate systems designed to supply coolant to the core of a water reactor to prevent meltdown in the event of an unlikely loss of coolant accident or a major coolant leak. These systems are intended to remove the fission-product heat produced in the core after the reactor has been shut down. However, they might not function for several reasons in the event of an accident, such as, severed lines to the reactor vessel and low water supplies. Therefore, reliance cannot be placed on systems such as these as the sole engineered safeguards in the plant. Nevertheless, prevention of core melting after an unlikely loss of primary coolant would greatly reduce the exposure of the public. Thus, the inclusion of a reactor core fission product heat removal system as an engineered safeguard is usually essential.

The core-spray and safety-injection systems consist of the piping, which connects the reactor-plant water storage facilities to the reactor coolant system, and also any necessary pumps, valves, auxiliary piping, spray nozzles or rings in the vessel, and instrumentation. The design details will vary from one nuclear plant to another. In the unlikely event of a major loss-of-coolant accident, these systems are designed to introduce large volumes of water into the reactor coolant system to replace water passing out of the rupture into the containment vessel. When this is done through separate spray rings located in the reactor vessel that distribute water over the core, the system is called a spray system; in other forms it is called a safety-injection system. A second function may be to circulate and cool, through heat exchangers, the water that has accumulated in the containment vessel. If the reactor requires a soluble neutron poison for cold shutdown, the system should draw water from a storage tank maintained with the necessary concentration of poison and heated if necessary.

Chief among the factors that must be considered in the design of a core-spray or safety-injection system are:

1. Assurance that the system will start when needed and will deliver the required amount of water. Adequate emergency power sources must be assured.

2. Provisions to prevent damage to supply lines during normal operation and any accident

conditions that might involve motion of the reactor vessel.

3. Flow distribution over the core. Many reactor designs involve components in the structure which could interfere with the injection of emergency coolant. It is often difficult to establish the circumstances under which adequate distribution can be achieved. Careful analyses and tests are required to determine whether the safety injection system as designed can prevent core meltdown and even metal-water reactions (in certain cores) in the unlikely event of a loss-of-coolant accident.

4. Provision for missile protection and accidental injury to important components of the systems.

5. Avoidance of thermal shock problems. Failure due to thermal shock could impair the operability of the system.

6. Protection of external parts of the safety injection system against freezing.

7. Provision for periodic check on operability of components.

* * *

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 9, 1967

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: ICE-CONDENSER PRESSURE-SUPPRESSION CONCEPT

Dear Dr. Seaborg:

At its ninety-first meeting, on November 2 and 3, 1967, the Advisory Committee on Reactor Safeguards reviewed a concept of an ice-condenser pressure-suppression scheme for a large pressurized-water reactor facility. A Subcommittee met on October 27, 1967, in Washington, D.C., to review this concept. During its review, the Committee had the benefit of discussions with representatives of the Westinghouse Electric Corporation and their consultants, and the AEC Regulatory Staff and of the documents listed below.

The ice condenser provides for rapid absorption of thermal energy, released in the unlikely event of a loss-of-coolant accident, by condensing the released steam in a heat sink that consists of ice located in an annular refrigerated compartment. An operating deck separates the containment into upper and lower chambers. The lower chamber houses the reactor and primary cooling system. The ice condenser connects the two chambers so that, in a loss-of-coolant accident, the steam and air from the lower chamber flow to the condenser, significantly reducing the peak and average pressures.

The Westinghouse Electric Corporation has erected a test facility representing a full-scale 1/180-segment of the ice condenser for a large power reactor. Representatives of Westinghouse reported results from this facility and from previous small-scale exploratory tests. Westinghouse plans further tests and analyses.

As with any new system, the design features of the ice condenser should be thoroughly explored and the appropriate margins evaluated by an applicant prior to submission in connection with a license application. The submittal should be fully documented to provide an adequate justification of the suitability of the system for the proposed facility.

The Committee believes that, subject to confirmation by additional tests and analytical studies, the ice-condenser concept can be successfully engineered to provide a satisfactory pressure-suppression system.

Sincerely yours,

/s/ N. J. Palladino

N. J. Palladino
Chairman

References:

1. Westinghouse Electric Corp. letter dated July 13, 1966; WCAP-2951, Ice Condenser Reactor Containment, dated June 1966.

2. Westinghouse Electric Corp. letter dated April 4, 1967; WCAP-7040, Ice Condenser Reactor Containment (Report of March 3, 1967 Presentation to the Atomic Energy Commission Regulatory Staff), undated.

3. Westinghouse Electric Corp. letter dated July 21, 1967; WCAP-7079, Preliminary Design and Evaluation of the Ice Condenser Reactor Containment and Associated Engineered Safeguards, dated July 1967.

4. WCAP-7079, Supplement No. 1, Supplementary Information to WCAP-7079, dated September 1967.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 8, 1974

L. M. Muntzing
Director of Regulation

SYSTEMS ANALYSIS OF ENGINEERED SAFETY SYSTEMS

With the current effort to standardize the design of certain types of nuclear power plants, the Committee believes that attention to the evaluation of safety systems and associated equipment from a multi-disciplinary point of view to identify potentially undesirable interactions between systems becomes increasingly important. The attached illustrative examples represents an initial and not necessarily complete listing of some problem areas.

The Committee would appreciate the Regulatory Staff reviewing these comments and discussing their ideas with an appropriate Subcommittee. Based on these discussions a mutually beneficial procedure for handling such issues may be developed.

W. R. Stratton
Chairman ACRS

Attachment:
List of Illustrative Examples

cc: P. Bender, SEC
E. Case, DL

Illustrative Examples of Questions to be answered by Systems Analysis and Quality Assurance

The following comments and questions are suggested for consideration as additional guidance in the review of Engineered Safety Systems:

1. Comment: Designers and architect-engineers frequently delegate responsibility for systems analyses to teams with functional engineering specialties such as "civil," "electrical," "mechanical," or "nuclear" with the team effort coordinated by managers responsible for controlling costs and avoiding schedule delays. With the same standard design applied to a number of plants, an intensive systems analysis effort which integrates the functional engineering specialties, is feasible. The scope and approach of the related Quality Control/ Quality Assurance effort should be commensurate with the Project Design effort. Consideration should be given to identifiable multi-disciplinary analyses of safety-related systems and associated systems, as part of Quality Control in design, procurement, construction, operation, and maintenance activities.

   General Question:

   What are the respective roles played by Project Design and Quality Assurance/Quality Control in the multi-discipline analyses of safety systems and associated systems?

2. Comment: As an aid in identification, safety related systems and associated equipment may be categorized as follows:

   a) Those systems or items of equipment which must be de-energized on demand (to a zero energy state) with extremely high reliability to:

      (1) Perform a safety function;

      (2) Prevent fire or other damaging consequence.

   b) Those systems which must be capable of long-term active operation to preserve control over radioactive materials (examples are fuel and environmental cooling and lighting and communications services).

   c) Those systems not directly related to a safety function but whose malfunction could have safety consequences because of secondary effects. It should be noted that such systems may not ordinarily be included in the set for which "conditions of design" are defined.

Question: In the design of such systems, is an interdisciplinary systems analysis performed to assure redundancy and separation appropriate to the category of the system? Does it consider all modes of normal operation, operation following any of the design basis events plus additional incidents such as pipe failures, loss of all active inputs to the system, and operation of part of the active components combined with the failure of others (for example, the operation of a large and critical motor in a space where the ventilation has failed)?

3. Comment: In addition to systems and equipment, space allocation and arrangement are crucial to safety. Both Unit and Station systems must be analyzed to assure adequate independence and separation of all vital functions. The analysis should consider the possibility that adverse "feedback" or other effect from one unit may leave other units without adequate redundancy.

   Such an analysis should help to provide a basis for establishing reliability, redundancy, and separation requirements. It should also provide information concerning the degree of separation necessary to protect against mechanical damage, fire or sabotage.

   Questions:

   a) Are design efforts and systems analyses directed to avoid concentration of vulnerability from various causes in one safety class structure, room, or zone?

   b) Are field located and field run equipment and systems examined to see if localized vulnerability has been created?

   c) Is space allocation a conscious responsibility in design?

   d) Are field inspections of space occupied by safety equipment and systems made by the cognizant design engineer to assure non-encroachment?

   e) Do changes in space allocation or arrangement require special approval?

4. Comments: Control systems may require communication lines, (electrical, pneumatic or hydraulic) that traverse significant distances and pass through several compartments.

   Questions:

   a) Is attention given in system design to physical locations of "field-run" impulse or static lines, including lines that provide information regarding ECCS functions, and other electrical-mechanical-hydraulic-pneumatic control systems which perform

safety functions, to assure that an unacceptable interaction between these and other systems is avoided?

b) Is specific attention given to assuring that field location follows that specified in the design?

5) Comments: Electrical systems and equipment should be analyzed to assure that over-current or other fault protection is sufficiently reliable and redundant to assure appropriate limitation of damage potential to other safety systems.

An example is the electrical power supply to primary system pumps. Failure of the circuit breakers could result in damage to the electrical penetrations and loss of containment under post-LOCA conditions.

Questions:

a) Are the circuit breakers for electrical power circuits that pass through containment penetrations set to trip in the event of arcing faults within the penetration?

b) Are such circuits designed with ground fault trips to protect the penetrations?

c) Are ground fault trips provided on all power circuits within the physical safety complex to reduce the fire hazard?

d) Are emergency lighting systems and internal communication systems safety grade?

e) Are control and power cables of widely differing voltages and currents intermixed in cabletrays, raceways, or conduits?

f) Are magnetic forces and molten copper considered in specifying the separation required between cables?

g) Are differences between laboratory test conditions for flame resistant cable insulation, and conditions that could exist in a cable way under faulted conditions, considered in defining separation requirements?

h) In determining the adequacy of separation, is consideration given to "foreign" sources of damage such as vehicle impact, use of welding equipment, explosive gas accumulation, or acts of sabotage?

i) Is fireproof, rather than fire "retardant" insulation required in vital areas? Is potential damage from radiation exposure from nearby components, such as air filters and charcoal adsorption beds, taken into account?

j) Is the timing of loss of offsite power considered in the prediction of the consequences of an accident? (For example, the most disadvantageous time may be just as motor operated valves are about to open or large pumps are almost up to operating speed).

6. Comments: Some ventilation systems may not be given attention as engineered safety features, however, situations may arise in which they can have important effects on safety.

Questions:

a) Are auxiliary systems such as containment or reactor building air cooling systems analyzed to see if their failure can lead to the failure of safety systems?

b) Are dynamic as well as static differential pressures on containment ventilation ducts and isolation dampers considered?

c) Are local effects of flow pressure gradients resulting from pipe ruptures analyzed for phenomena such as the collapsing of ventilation ducts, which could result in closing vent areas?

d) In evaluating the adequacy of the protection provided to operators in the control room following a LOCA, is consideration given to the possibility that a ventilation (or large electrical) penetration of the containment has failed and is leaking the containment atmosphere into the adjacent space?

7. Comments: Experience has indicated that fluid systems deserve special attention in both static and dynamic situations. Particular attention should be given to stresses resulting from valve action, pump starts, and water slugs, including backflow and check valve action, as well as flow action under severe accident conditions or fault modes.

Questions:

a) Is consideration given to the effect of fluid system dynamics on mechanical stresses in components and equipment?

b) Are the consequences of the failure of check valves to close properly in various fluid systems examined for normal and faulted conditions?

c) Is evaluation made of the possibility and effects of crushing and/or rupturing one group of control rod drive hydraulic lines during a LOCA? Are combinations of ruptured and crushed lines also considered?

d) In PWRs, are the consequences of multiple steam generator blowdown considered?

e) In the evaluation of a system's ability to perform its required service is consideration given to potential flooding effects resulting from roof drain obstruction (potential roof collapse), rupture of non-Class I tankage, continued operation of a leaking system, or reverse flow through normal or ruptured pipe that could be siphoning liquid from some storage source?

f) What controls are placed on the use of "plaster" or glass wool type thermal insulation within containment, that could foul or possibly cause failure of ECC systems?

g) Is an analysis performed to determine when pool boiling would occur, if during refueling (with the reactor vessel head removed) both of the two canal cooling systems become disabled?

What would be its consequences? How long would the operators have to restore cooling?

8. Comments: Fires may have unusual consequences in reactor systems and deserve special attention.

Questions:

a) How are fires analyzed for potential effects on safety?

b) Are the storage of flammable materials in vital spaces and the passage of flammable gases or liquids through vital spaces prohibited?

c) Are safety enclosures, including doors, for diesel generators designed to withstand a diesel runaway, fire, or combined fire-explosion?

d) Does analysis of electrically generated fires consider the following for each power circuit:

1) The change of a circuit short or overload in a circuit within a safety class structure?

2) The chance of a branch overload or short circuit followed by failure to clear the fault?

3) The chance of fire from circuit overheating at or below normal current load?

4) The possibility that fire will propagate to:

a) Disable one vital electrical division if the circuit is not already in a vital division?

b) Disable two or more vital divisions in a local area where minimum allowable separation is employed?

5) The potential consequences of combustion of fumes from fire in confined spaces?

9. Comments: After careful analysis and design it is essential that operation or tests in the field follow the resulting specifications.

Questions:

a) Does environmental qualification allow for or test for the possible lack of discipline in field installation which may result in a field installation that is significantly different from the qualification test setup? Do the qualification tests represent a condition of long term (multi-year) normal operation followed by short term, very severe environmental conditions?

b) Are special instructions for operation and maintenance identified after being developed by a disciplined systems analysis? Examples of such instructions are "use no flame", "no traffic area", "do not operate if ...," "no welding without prior approval of fire protection personnel", "do not use mercury-containing instruments", "do not overtorque", "no substitutes for this material".

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 9, 1979

The Honorable Victor Gilinsky
Commissioner
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Gilinsky:

In response to your letter of September 12, 1979, we offer the following comments relative to the emission of xenon from the TMI-2 nuclear plant and the possibility of reducing such releases in similar accidents which may occur in the future.

So far as we know, the comments on the various approaches for estimating the total quantity of xenon released following the accident appear reasonable. In terms of controlling xenon releases in accidents which may occur in the future, we believe that chilled or cryogenic charcoal adsorption beds of adequate capacity would be helpful. Adsorption systems using this principle are commercially available and are in successful operation for handling routine releases at several nuclear power plants. The major problems would be possible decreases in the effectiveness of such beds under conditions which involve the larger volume and moisture content of releases accompanying an accident, and the difficulties in assuring that all the xenon emissions are collected so they can be effectively directed to and treated by the adsorption system. Other methods, such as low temperature liquefication and fractional distillation may also be usable for removal of radioactive noble gases.

As noted above, the technology exists and has been applied in BWRs and PWRs to remove and retain for decay purposes certain radioactive noble gases before discharge as gaseous effluent. The systems have not, however, been applied under accident conditions where multimegacurie releases may occur essentially in bursts or over relatively short time intervals, such as was experienced during the TMI-2 accident. The Committee believes that a study should be undertaken to determine the applicability and desirability of available technology to minimize the release of radioactive noble gases during various postulated accident scenarios. The study should include assessment of the various potential pathways for radioactive gaseous releases as well as considerations of accelerated rates of treatment of large gas volumes such as those existing in large containments.

Anticipating the need for answers to the types of questions raised in your letter, the Committee requested about a year ago that an ACRS Fellow develop a report summarizing experimental data on the performance of charcoal beds under a variety of parameters. It is anticipated that this report will be completed within the next two to three months, and we will plan to provide a copy to you at that time.

Sincerely,

Max W. Carbon
Chairman

cc: Joseph M. Hendrie, OCM
Richard T. Kennedy, OCM
Peter A. Bradford, OCM
John F. Ahearne, OCM
Samuel Chilk, SECY
Lee V. Gossick, EDO

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

September 11, 1961

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: NUCLEAR REACTOR SYSTEM MECHANICAL DESIGN

Dear Dr. Seaborg:

Since the primary loop of any nuclear reactor system is the principal and sometimes the first containment of radioactive fission products produced in the nuclear reaction, adequate mechanical design of these systems is important to the integrity of the plant and to the health and safety of the public.

Design of vessels, piping, and other components for these systems must be of a quality far superior to that ordinarily acceptable for commercial or defense installations. The generally recognized standards for commercial installations have been the Pressure Vessel Code of the American Society of Mechanical Engineers, and the Pressure Piping Code of the American Standards Association. These have been incorporated in many state codes and laws, and modified for marine and naval service. These codes are attempting to meet the changing needs for superior quality by the issue of revisions or addenda. Both organizations have in preparation complete new code sections for nuclear installations. These new codes will take some years for completion and adoption.

The Navy has prepared "NR-S-1 Tentative Structural Design Basis for Components of Reactor Coolant and Associated Systems" which is new and in some respects may appear to be a more lenient code than the two mentioned above. The apparent lenience is balanced by the more complete and rigid inspection of materials, fabrication, and testing by the Navy.

It is important to recognize that the codes specify minimum requirements and that additional specifications will necessarily be added by the purchaser or the manufacturer of the vessel or apparatus.

Experienced and reputable manufacturers fabricating nuclear vessels go beyond the minimum required by the codes or by the purchasers' specifications in inspection of materials and control of fabrication. If apparatus is purchased on a competitive price basis alone, the successful bidder may be inexperienced and so not supply the needed extra unspecified quality, or a competent bidder may be forced to sacrifice quality to be competitive. Bids should not be accepted for nuclear vessels without assurance that the bidders can adequately fabricate and inspect, and are of demonstrated competence in the construction of similar nuclear or nonnuclear equipment.

In addition to the usual design and inspection problems in pressure vessel manufacture, considerations must be given to deterioration of materials under irradiation. The most clearly recognized problem is the rise in nil ductility temperature in some metals. This might result in failure of reactor vessels and in case of such failure reactor containments are likely to be violated. In event that reactor internal parts are made of such materials, they might remain under stress when cold resulting in possible internal structural failures. This general problem is currently under study by a number of specialized groups, and the available data and their application to operating and new reactor vessels is being considered by the Division of Licensing and Regulation.

It is the recommendation of the ACRS that:

(1) Except in special units, all nuclear vessels and piping be designed, built, and inspected in accordance with one of the recognized codes and that no deviation be permitted except as provided by the code used;

(2) In selection of vendors for supplying equipment for use in nuclear installations, recognition be given to the need for care beyond code requirements in selection of materials, control of fabrication, and testing;

(3) In the Hazards Summary Report, (a) the code used be specified, and (b) assurance be given that adequate concomitant material selection, manufacturing procedure, and testing will be used;

(4) Every effort be made to expedite the laboratory and field study of radiation damage of metals as related to nil ductility temperature;

(5) The Division of Licensing and Regulation continue to pursue vigorously the determination of the status of present reactor vessels, the necessary steps to avoid failure from brittle fracture, and definition of the basis for preventive design of future installations.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

March 31, 1965

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Dear Dr. Seaborg:

The design of nuclear reactor plants to withstand earthquakes has recently been the subject of growing attention. Considerable difficulty has arisen in achieving a consensus of expert opinion as to the maximum earth motion and accelerations to be expected at any given site.

These are not easy problems to resolve, and the problems will take on increasing importance as more and larger nuclear reactors are built. The Advisory Committee on Reactor Safeguards recommends that a panel composed of especially well-qualified scientists and engineers drawn from academic, industrial, and governmental institutions be established, and that this panel be asked to generate a meaningful evaluation of the present knowledge about earthquakes, and to provide a basis for determining earthquake protection requirements in a general way, if possible. The panel could also be expected to recommend a long range research program designed to add to our knowledge of seismic phenomena and our ability to establish earthquake protection needs.

The Advisory Committee on Reactor Safeguards would like to make one further recommendation pertaining to the seismic problem at this time. There is some chance that in the future a large earthquake may occur in the vicinity of some operating reactor. This particular reactor presumably will have been designed to withstand a ground acceleration having a certain amplitude-frequency relationship. If no record of the actual seismic effects at the reactor site is made, doubt may arise as to whether or not some vital, but inaccessible, component of the reactor complex has been subjected to dangerous forces. The likelihood that such a circumstance will arise could be reduced considerably if appropriate instrumentation were provided and

used routinely at each power reactor site. The Committee recommends that the feasibility of such monitoring be investigated, and that such a program be instituted, if it is found to be practical.

Sincerely yours,

/s/ W. D. Manly

W. D. Manly
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

March 14, 1967

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C.

Subject: GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Dear Mr. Price:

The Advisory Committee on Reactor Safeguards has reviewed the February 6, 1967, draft of "General Design Criteria for Nuclear Power Plant Construction Permits" and offers the attached comments. It is requested that these comments be incorporated into the criteria to the maximum extent practical. Any statement regarding ACRS review of the criteria should be cleared with the Chairman of the ACRS before release.

The Committee notes that the preamble to the criteria released on November 22, 1965, contained the following sentence in the second paragraph: "It should be recognized that additional criteria will be needed for evaluation of a detailed design, particularly for unusual sites and environmental conditions, and for new and advanced types of reactors." The Committee believes that these criteria should not be released without this or an equivalent sentence.

Sincerely yours,

/s/

N. J. Palladino
Chairman

Attachment:
Comments on February 6, 1967 Draft of "General Design Criteria for Nuclear Power Plant Construction Permits".

Comments on February 6, 1967 Draft of "General Design Criteria for Nuclear Power Plant Construction Permits".

Criterion 1. This criterion should call for the identification of vital from non-vital equipment.

Criterion 3. We are licensing a plant for a certain power; if at the construction permit stage we don't think that power is safe (under this criterion) we should say so. Hence, this should be in Category A.

Criterion 4. "process oscillations" should be clarified.

Criterion 5. Change to "The reactor shall be designed ...". Taken literally, the present wording allows, for example, an intrinsic positive power coefficient if some coolant temperature controller makes an "apparent" negative coefficient.

Criterion 6. Reinstate "exceedingly low probability of gross rupture ..." Suggested rewording: "The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of significant leakage throughout its design lifetime."

Criterion 8. Add sentence at end of present criterion as follows: "It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause."

Criterion 9. Delete the first "process".

Criterion 10. Needs to be reorganized and clarified. Original wording was clear. "Means ... for monitoring and maintaining control over ... disposition of fuel" could be instrumentation, fuel hold-down, or something else.

Criterion 11. Precede by a statement that the applicant shall establish acceptable fuel damage limits.

Criterion 13. This criterion still does not contain the thought that the instrument range must be extended to indicate large, large releases that the designers think incredible. Insert "anticipatory transients" after "normal operations".

Criterion 18. (a) Delete "process" everwhere.
(b) Change "shall not negate the minimum redundancy" (double negative) to "leaves intact a system satisfying all".

Criterion 21. Change "functional operability" to "that no failures or loss of redundancy have occurred". Delete "and to determine component or circuit failures".

Criterion 24. (a) "Subcritical" at "hot operating" - a contradiction.
(b) Do we enforce this on GE? Their liquid poison system is nominal only.
(c) This criterion is not clear. Under what circumstances are the reactivity control systems to work sufficiently fast?

Criterion 25. Even under loss-of-coolant transients?

Criterion 26. How does this differ from the last sentence of Criterion 25?

Criterion 27. We do not now enforce this for the rod-drop (GE) or rod-ejection (PWR) accidents. Are these "accidents" malfunctions? This is not clear.

Criterion 28. This is inadequate; there should be lots of margin.

Criterion 30. Suggested rewording of the first sentence: "The reactor coolant pressure boundary shall be designed to minimize the possibility of rapid propagation type failures."

Criterion 31. (a) Temperature-pressure restrictions should be imposed when the upper shelf of the Charpy transition curve shows a low energy value.

(b) In the first sentence, change "reactivity induced" to "rapid".

Criterion 33. Suggest some other word than "Accommodate", e.g., "cope with".

Criterion 34. Difficult to read.

Criterion 35. The old 9.1.2 was better. The alternate power sources must be provided before all off-site power is lost. "A capability" is not enough; where is redundancy, testing, capacity margin, etc., etc.?

Criterion 37. Delete "reliability and". This criterion is part of reliability. Why is the very important last sentence deleted?

Criterion 38. Now unacceptable. The design must be such that operation of safeguards can not make things worse. The revision implies that interlocks should prevent protective (engineering safeguard) action at the wrong time - a dangerous kind of safeguard indeed.

Criterion 39. Suggest adding two sentences as follows: "The performance of the ECCS shall be evaluated conservatively in each area of uncertainty. Independent and preferably diverse redundant methods of accomplishing abundant emergency cooling shall be provided."

Criterion 44. Suggested rewording: "The containment structure, including access openings and penetrations, shall be designed to accommodate or dissipate without exceeding the design leakage rate, the largest credible energy release, including a considerable margin to cover effects like metal-water or other chemical reactions that could occur as a consequence of improper function of emergency core cooling systems."

Criterion 47. Should state that each heat removal system should be full capacity.

Criterion 48. "multiple" should be "redundant".

Criterion 52. Why not work in a positive sense? The intent may be ambiguous in that testing under the conditions demanded in the accident case may not be possible. Change "whether" back to "that no". The point (often repeated, and many times reinstated in this document) is that a valve can fail and yet a test can show no leakage because the backup valve is tight. Tests must reveal the first flaw so it can be fixed, else redundancy doesn't pay.

Criterion 53. Add the "torus" to the list of example items.

Criterion 61. "Favorable" geometries are those favorable to criticality! "Geometrically safe configurations" is the correct term.

Criterion 62. Too many words between "ensure damage" and "is prevented". Reword.

Criterion 65. Add at the end of the last sentence, ...", except that reduction of the dosage levels recommended in these parts may be required where the location of a plant is such that effluents can affect large populations".

GENERAL COMMENTS

1. A suitable criterion covering leak detection from the primary system should be developed.

2. There is insufficient coverage on the need to design to minimize the possibility of fire or difficulty from fire.

3. Throughout the document, "protective" should be replaced by "protection" as preferred image; see the recent IEEE criteria. "Protective" means a coating.

4. It should be required that appropriately complete fabrication records of important components be kept.

5. It should be required that different principles be used where necessary to achieve true independence of redundant instrumentation components.

6. Restrictions for sharing redundant systems between reactors should be stated.

7. Capability of retesting the containment at design pressure should be required.

8. In the second paragraph of the preamble, change "unnecessary" to "inapplicable" (second sentence) and insert "acceptable" between "alternative" and "criteria" (third sentence).

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

May 13, 1971

H. L. Price, Director of Regulation

SEISMIC RESEARCH PROGRAM

The geological and seismological consultants to the ACRS have expressed concern that seismic conditions in the eastern United States are poorly understood and have recommended that emphasis be placed on the early development of information that would aid the AEC Regulatory groups in the determination of conservative parameters for the Safe Shutdown (or Design Basis) Earthquake.

Of course, several broad and comprehensive sets of recommendations for earthquake and earthquake engineering research have been published in recent years. However, these generally would involve considerable financial expenditure to implement and have not yet been pursued. In response to a request by the ACRS for definition of a specific, limited effort which has promise of providing early assistance in the difficult judgment process of determining seismic design parameters, our consultants have recommended that the National Ocean Survey (NOS) and the United States Geological Survey (USGS) be asked to accelerate a task that they have already begun, namely, the preparation and publication of a comprehensive and systematic treatment of the seismic data east of the Rockies, including adjacent parts of Canada. To the extent practical on an expedited time scale, such a study should be correlated with existing tectonic and basement geology maps, and other pertinent geophysical data. The seismicity data should be so presented that not only are all known epicenters available, but also that selected sets of epicenters (such as all those above a certain magnitude) are also readily available. The availability of such study would not only assist ongoing judgments of seismic design parameters but enable participation by various other groups in the seismic community in the development of a better understanding of tectonic conditions in the eastern United States, and in the application of various statistical methods to the prediction of earthquake recurrence intervals.

The ACRS believes that a better basis for judging seismic design parameters east of the Rockies is important, and that early steps should be taken to expedite completion of the study by NOS and USGS as described above, including the use of AEC financial support, if necessary.

The ACRS consultants have also recommended an accelerated study of the Charleston, South Carolina earthquake of 1886, which is the major seismic event in the populous eastern seaboard of the United States. They believe it very important to reach an early appraisal of the validity of localizing the Charleston event at Charleston or the validity of assuming that the Charleston event could occur at other localities in the Coastal Plain or Piedmont areas of the eastern seaboard. If the latter were proven to be true, it would greatly increase seismic risk throughout this area and result in greater acceleration as a design criterion for the Safe Shutdown Earthquake. The ACRS believes it is desirable, therefore, to establish as soon as practical the geologic and seismological facts of the Charleston event and to assure the early installation of a sufficient number of strong accelerographs in the Charleston area.

Because of the potentially large difference in vibratory motion characterisitics of earthquakes east and west of the Rockies, as well as to better establish the nature of several other seismically active areas east of the Rockies, the ACRS consultants recommend and the Committee supports the position that the AEC, through NOS (or perhaps with the cooperation of appropriate universities) accomplish the early installation of strong motion instruments in each of the several other areas of great seismic interest east of the Rockies.

/s/ R. F. Fraley

R. F. Fraley
Executive Secretary

cc: G. Seaborg, Chairman, AEC
R. Hollingsworth, GM

ADVISO COMMITTEE ON REACTOR SA. EGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

May 16, 1973

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON SEISMIC RESEARCH

Dear Dr. Ray:

The seismicity of the Eastern United States is considerably less than that of the Western United States; however, there have been a number of large earthquakes in historical times in the eastern region. The geological and seismological consultants to the ACRS have expressed concern that seismic conditions in the Eastern United States are poorly understood and have recommended that emphasis be placed on early development of information that would aid the AEC Regulatory Groups in the determination of appropriate parameters for the Safe Shutdown (or Design Basis) Earthquake. Three areas are of particular interest: South Carolina, the St. Lawrence Valley, and southeastern Missouri. Western Ohio and the Cape Ann region of Massachusetts also have had significant shocks and hence are areas of interest.

The ACRS agrees with its consultants and believes it is particularly important to determine the cause of the 1886 Charleston, South Carolina earthquake. This shock may represent a localized source; however, if it does not, it can become the determining factor in establishing the potential earthquake risk for the entire Atlantic coastal plain and Piedmont Province from New Jersey to Florida.

In the western part of the United States, it is usually possible to correlate occurrence of earthquakes with known active faults. Many of the earthquakes in Western United States are accompanied by visible fault displacements, and it is possible to relate the occurrence of these earthquakes to the tectonic framework of the region. In Eastern United States, the earthquake sources are not well understood and at the present time we must depend almost completely on the historic records to project a future pattern of earthquake occurrence. The ACRS endorses the recommendation of its consultants that vigorous investigation be carried forward in the Eastern United States to better understand the reasons for and sources of the earthquakes -- particularly in those regions where we know large shocks have occurred

in the past. NOAA and the U. S. Geological Survey have been collaborating in bringing up to date the historic seismicity in the Eastern United States and correlating it with the tectonic framework of the region. More detailed and sophisticated studies need to be carried forward, with high priority given to the Charleston area.

The most significant historic earthquake near Charleston occurred in 1886; this earthquake had a magnitude somewhere in the range of Richter 7, and the epicentral effects were such that a number of buildings and bridges were badly damaged and railroad tracks distorted. Some earthquake activity had occurred in the Charleston region prior to the great shock in 1886, and there have been a number of aftershocks in the succeding years, with small earthquakes continuing to occur.

A study was made of the Charleston earthquake shortly after its occurrence; however, this was before much was known of the mechanism of earthquake occurrence and some vital evidence may have been overlooked. Instrumentation for recording earthquakes had not been developed in 1886, so there are no useful instrumental records of the shock. The mechanism of the shock is uncertain. It probably occurred as a result of slippage on a fault that was deep enough so that there was no surface breakage. Because of the considerable damage at the surface, it has been inferred that the depth was not very great. The orientation of the fault is unknown and reasonable arguments may be presented for either a northeast or a northwest trend. The historical evidence could be interpreted to indicate a localized source but, even if the source is highly localized, a tectonic explanation is important.

The locations and frequency of occurrence of earthquakes in the region are only imperfectly known. There is only one seismograph station in the State of South Carolina, and the number of stations in this part of the United States is not sufficient to determine the local seismicity. A suitable network of strong-motion stations (at least six) should be put in the area so that the locations of all earthquakes of Richter magnitude 2.5 and up in the coastal plain and the Piedmont Province can be established and their time and space distribution determined. To supplement this network, micro-earthquake studies should be made throughout the region to determine if there are zones along which micro-earthquakes are frequent and whether or not these zones correlate with the larger shocks. Both the network and the micro-earthquake studies should cover an area considerably larger than the immediate vicinity of Charleston. The Committee understands that some micro-earthquake studies have been initiated by NOAA in this area in recent months.

Detailed geological studies need to be done in the Charleston area to look for any remaining evidence from the 1886 shock that may now be interpretable in light of what we now know of earthquake mechanisms. The geology, both surficial and subsurface, needs to be re-examined and compiled to develop any information pertinent to the seismicity of the coastal plain and Piedmont Province of South Carolina. Additional aeromagnetic and gravity studies should eventually be made for correlation with the seismic and geological investigations.

After a preliminary delineation of the seismicity by a modern network of seismograph stations and a re-examination of the geology, it will be necessary, almost certainly, to verify suspected structures by detailed reflection seismic profiles. The extent of such profiling will depend on the seismicity and geological investigations but could involve several hundred miles of profiling. Ultimately some drilling may have to be done for final verification.

If there is not a recent first order triangulation and level net in the Charleston area, the relevant state and federal agencies should be encouraged to update the existing nets. Should another large earthquake occur in this area, there may be significant changes in level or position that would help determine the nature of the source.

A program similar to that outlined for the Charleston area should follow in other areas in the Eastern United States where there have been large earthquakes in historic times.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Acting Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

June 11, 1973

L. Manning Muntzing
Director of Regulation

PROPOSED SEISMIC AND GEOLOGIC SITING CRITERIA, APPENDIX A TO 10 CFR PART 100

The Committee believes it acceptable to publish the draft Seismic and Geologic Siting Criteria, dated June 7, 1973. However, the Committee notes that the minimum acceptable acceleration currently in use (0.1 g SSE) frequently leads to considerable debate among seismic experts concerning an acceptable seismic design basis, even in regions of low seismicity. The Committee recommends that a study be initiated to determine the practicality and cost of raising the minimum design basis acceleration to 0.15 or 0.2 g for future reactors, except where foundation conditions and geological and seismic information are recognized by a clear consensus of expert opinion to warrant a lesser value. The Committee expects that an increased minimum design basis acceleration would assist in standardization, would provide increased margin to cover uncertainty, and would facilitate seismic aspects of regulatory review for many sites, without involving an important increase in cost, if factored into the original design.

H. G. Mangelsdorf
Chairman

cc: L. Rogers, RS
R. Minogue, RS
P. Bender, SECY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 13, 1973

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON "RELATIONSHIPS OF EARTHQUAKES AND GEOLOGY IN WEST TENNESSEE AND ADJACENT AREAS"

Dear Dr. Ray:

At its 163rd meeting, November 8-10, 1973, the Advisory Committee on Reactor Safeguards completed its review of seismic design bases proposed by the Tennessee Valley Authority (TVA) for use at potential reactor sites in Western Tennessee between Miles 170 and 240 on the Tennessee River. The matter was previously considered at a Subcommittee meeting held in Los Angeles, California, on Ocober 26, 1973. In its review, the Committee had the benefit of discussions with representatives of TVA and its consultants and of the AEC Regulatory Staff and its consultants. The Committee also had the benefit of the documents listed.

The very large New Madrid earthquakes of 1811-1812 dominate the historical seismic records of this part of the United States, and the major matters entering into this review relate to the possible location of a future earthquake of the same magnitude. Nearly all large historic earthquakes in this area have occurred within the western part of the New Madrid Faulted Belt in an area that is called the Reelfoot Tectonic Structure. However, the applicant proposes to use the eastern boundary of the New Madrid Faulted Belt as the epicenter of a postulated major earthquake, thereby increasing the estimated vibratory effects in the siting region under consideration. The applicant has also employed a conservative attenuation basis derived from the limited data available. The applicant has concluded that site intensities varying with location from a low Modified Mercalli (MM) IX to a high MM VIII should be employed for the safe shutdown earthquake at sites on bedrock between Miles 170 and 240 on the Tennessee River.

The Committee believes that the proposed intensities appear to be reasonable for this section of the Tennessee River and should provide a satisfactory basis for the development of the response and seismic design parameters for specific sites, provided proper attention is given to the actual foundation conditions at each site. The Committee notes, however, that the attenuation assumptions for the siting area considered in this review may not be applicable to regions outside this area. The Committee also notes the desirability of further studies to determine the cause of the New Madrid earthquakes and to delineate better the seismic zone where such earthquakes might occur.

Sincerely yours,

/s/ H. G. Mangelsdorf

H. G. Mangelsdorf
Chairman

References attached

References

1. TVA Report, "Relationships of Earthquakes and Geology in West Tennessee and Adjacent Areas"

2. Amendment 1 to the TVA Report, "Relationships of Earthquakes and Geology in West Tennessee and Adjacent Areas"

3. "Review and Comments on TVA Report, 'Relationships of Earthquakes and Geology in West Tennessee and Adjacent Areas'" transmitted by United States Department of the Interior, Geological Survey letter, dated September 11, 1973

4. "Review of the Tennessee Valley Authority Report, 'Relationships of Earthquakes and Geology in West Tennessee and Adjacent Areas'" transmitted by United States Department of the Interior, Geological Survey letter, dated November 6, 1973

5. "SAB-L Review of the Report by TVA Entitled 'Relationships of Earthquakes and Geology in West Tennessee and Adjacent Areas,'" dated August 21, 1973

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 16, 1975

Lee V. Gossick
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Gossick:

During ACRS review of the application for a construction permit for the Callaway Plant, Units 1 and 2, Dr. Stepp of the NRC Staff made a presentation which summarized well the current Staff approach to the evaluation of seismic design bases for most of the U. S. east of the Rockies. The Committee believes it would be useful to have the Staff position written up in some detail, so that it may more readily be reviewed and utilized as appropriate. The Committee recommends that this be done.

Sincerely yours,

Original Signed by:
W. Kerr

W. Kerr
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 15, 1977

Mr. Lee V. Gossick
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REGIONAL TECTONICS OF THE PACIFIC NORTHWEST

Dear Mr. Gossick:

The Advisory Committee on Reactor Safeguards met and considered matters related to the regional tectonics of the Pacific Northwest at its 209th and 211th meetings. Subcommittee meetings have been held on this subject on September 1-2, 1977 in San Francisco, California and on October 27-28, 1977 in Portland, Oregon. The Committee and its consultants have had the benefit of presentations on this matter by representatives of the Nuclear Regulatory Commission (NRC) Staff, the U. S. Geological Survey (USGS), Puget Sound Power and Light Company, Portland General Electric Company, Washington Public Power Supply System, their consultants, and members of the public.

The Committee and its consultants have given extensive consideration to the possible intensity of the 1872 Wenatchee earthquake. The USGS, as a consequence of the manner of application of the Modified Mercalli (MM) criteria, considers that the assigned intensity of this earthquake cannot be less than MM Intensity VIII and is probably MM Intensity IX, based primarily on the occurrence of a large landslide at Ribbon Cliffs, believed to have been caused by this earthquake. This higher intensity differs from the previously cited Intensity of VII - VIII, usually given in prior studies and discussed in the Coombs Review Panel Report. The Committee and its consultants have considered this matter and have concluded that the 1872 Wenatchee earthquake should be considered as an Intensity VIII. In reaching this conclusion, the Committee considered the bases on which the relation between MM intensity and horizontal ground acceleration relationships were derived. The Committee recognizes that higher MM intensities often are reported when seismic data are scanty and multiple MM criteria cannot be used. The extensive data compiled by the utilities and their consultants have been reviewed by the ACRS and its consultants. These data pertain not only to the 1872 earthquake but also to the regional tectonics, regional geology, and the various mechanisms considered responsible for the seismic events observed throughout the Pacific Northwest. Based on a review of these data, the Committee agrees with the

opinions expressed by most participants and ACRS consultants that arbitrary movement over extended distances of a prototype 1872 earthquake, for purposes of seismic design, should be reexamined. This is particularly true for the Columbia Plateau which appears to be either a different tectonic province than that in which the 1872 earthquake occurred, or a region of lower seismic activity with regard to frequency and intensity.

The Committee believes that the studies currently being performed on the tectonics of the Pacific Northwest will better identify the seismic potential of the various regions within the Pacific Northwest. The studies being performed for the Columbia Plateau Region are particularly promising with respect to better defining its seismic potential. The Committee wishes to be kept informed of the progress of these studies and to discuss these matters with the NRC Staff when they have completed their review.

Sincerely yours,

M. Bender
Chairman

References:

1. H. A. Coombs, "Report of the Review Panel on the December 14, 1872 Earthquake," dated December, 1976.
2. Puget Sound Power and Light Company: "Preliminary Safety Analysis Report for Skagit Nuclear Power Project, Units 1 and 2," Chapter 2.5 and Appendix 2J, "Investigation of the December 14, 1872 Earthquake in the Pacific Northwest," by Bechtel Corporation, Inc., dated June, 1975.
3. Portland General Electric Company: "Preliminary Safety Analysis Report for Pebble Springs Nuclear Plant, Units 1 and 2," Chapter 2.5.
4. Washington Public Power Supply Systems: "Preliminary Safety Analysis Report for WPPSS Nuclear Power Projects 1 and 4," Chapter 2.5 and Amendment 23.
5. Weston Geophysical Research, Inc., "The 1872 Earthquake - Significant Data and Conclusions," undated.
6. Woodward-Clyde Consultants, "Review of the Pacific Northwest Earthquake of December 14, 1872," undated.
7. N. H. Scott, "Evaluation of the Epicenter and Intensity of the Pacific Northwest Earthquake of December 14, 1872," dated September 13, 1976.

8. Shannon and Wilson, Inc., "Reconnaissance Investigation of the Ribbon Cliff Landslide, Entiat, Chelan County, Washington," dated August, 1976.
9. Shannon and Wilson, Inc., "Geologic Studies of Columbia River Basalt Structures and Age of Deformation, The Dalles-Umatilla Region, Washington and Oregon, Boardman Nuclear Project," dated November, 1973.
10. Shannon and Wilson, Inc., "Regional Geologic and Seismic Investigations, Boardman Nuclear Project," dated January, 1973.
11. Letters from D. L. Renberger, Washington Public Power Supply System, to O. L. Parr, Office of Nuclear Reactor Regulation, dated September 14, 1977 and September 2, 1977, respectively, concerning the results of field investigation of the Fraser River - Straight Creek Fault Zone.
12. E. S. Cheney, "Interim Report on the Seismic and Geologic Hazards to the Proposed Skagit Power Site, Sedro Wooley, Washington," Revision of October, 1977, unpublished manuscript.
13. U. S. Geological Survey, "A Study of Earthquake Losses in the Puget Sound, Washington Area," Open File Report 75-375, dated 1975.
14. U. S. Geological Survey, "Maximum Intensity of the Washington Earthquake of December 14, 1872," dated April 1, 1977.
15. U. S. Geological Survey, "Status of Review - Skagit, Washington," by W. Hays and S. Brockman, dated July 21, 1977.
16. Portland General Electric Company: "Final Safety Analysis Report for the Trojan Nuclear Plant," Chapter 2.5.
17. Washington Public Power Supply System: "Preliminary Safety Analysis Report for WPPSS Nuclear Project No. 2," Chapters 2.6 and 2.7.
18. Letter from Edson G. Case, Acting Director, Office of Nuclear Reactor Regulation, to R. F. Fraley, Executive Director, Advisory Committee on Reactor Safeguards concerning the review of the geology and seismology for the Skagit Nuclear Project, dated July 28, 1977.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 15, 1980

Mr. William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: SEISMIC QUALIFICATION OF AUXILIARY FEEDWATER SYSTEMS

Dear Mr. Dircks:

In a letter to you dated June 10, 1980, the Advisory Committee on Reactor Safeguards expressed its concern that the NRC Staff's level of effort in reviewing the seismic qualification of auxiliary feedwater systems might not be consistent with timely resolution of the matter. We recommended that the necessary manpower be committed to assure completion of the Staff's short-term review in two or three months. We recommended also that the affected licensees perform independent reviews of this matter on the same time scale.

In a memorandum to Chairman Ahearne dated August 27, 1980 Harold R. Denton transmitted a memorandum dated August 8, 1980 from Roger J. Mattson to Darrell G. Eisenhut which described an interim risk study. Mr. Denton stated that the study indicates that the risk to the public health and safety for the next three years is acceptable and that the plants could be permitted to continue to operate during this period. Mr. Denton also stated that a bounding analysis, using conservative assumptions regarding seismic damage to auxiliary feedwater equipment, shows that operation for the next several months would not result in an unacceptable increase in risk to the public.

We continued to review this matter during our 246th meeting, October 9-11, 1980. In our review we had the benefit of a Subcommittee meeting on October 8, 1980.

We offer the following observations and recommendations:

1. The interim risk analysis performed by the Staff is useful. The Committee notes that this study does not appear to have been subjected to independent peer review or given the necessary degree of quality assurance that a risk analysis, which may enter importantly into safety decision making, should receive.

2. The August 8, 1980 memorandum from Roger Mattson does not provide the expected (or mean) value of risk. Since detailed knowledge of all the components, equipment, and systems important to the auxiliary feedwater function is not available to the Staff, it is not clear that a sound basis exists for the terms "conservative" or "very conservative" which are used in this memorandum. Under these circumstances, it is possible that for one or more of the plants, the risk will be found to be larger than that estimated.

3. The Staff's estimated risks of a seismic event causing a serious accident due to loss of shutdown heat removal capability range from six to fifteen times the estimated risk of core melt due to all causes for the PWR examined in WASH-1400. The Staff has proposed a course of action in terms of these estimated risks which implies a quantitative safety goal and a threshold risk level for certain kinds of action. Such safety goals and action levels may prove to be acceptable to the NRC after review and evaluation. However, we believe that these should not become de facto criteria without the benefit of proper consideration. Furthermore, in the absence of an evaluation of the uncertainties in and the expected value of the risk, the risk estimates presented may not be representative of the actual risk.

4. The risk estimates presented in the memorandum from Roger Mattson of August 8, 1980 are large enough, if accurate, to warrant considerable priority by the NRC and the affected utilities. In particular, efforts should be made to better quantify the risk on a plant specific basis in the next few months. Furthermore, we recommend that each affected licensee be asked to review his specific plant design and to take early remedial measures, as practical, if there is reason to suspect that any important aspect of the auxiliary feedwater system is likely not to perform its function during an earthquake similar to a safe shutdown earthquake for the plant.

5. We agree with the Staff that high priority should be given to resolution of this matter. We expect to continue to follow this subject closely.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 8, 1982

Honorable Victor Gilinsky
Commissioner
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Commissioner Gilinsky:

This is in response to your request for comments, dated December 3, 1981, regarding a letter from Professor Paul Jennings on proposed changes in seismic design methodology.

We believe that Professor Jennings' suggestions are of interest and merit careful consideration. We recommend that they be considered within the context of a broad review of the NRC Staff's current seismic design practices. The NRC Staff is planning to reactivate their reassessment of Appendix A to 10 CFR Part 100 in the near future. We believe that this would be an appropriate forum for these discussions and recommend that Professor Jennings be invited to participate. We note that the NRC Staff in licensing actions and in their past evaluations of seismic design practices has addressed some of the issues on which Professor Jennings has commented and has in some instances adopted approaches similar to what he has suggested. We expect to be involved in the planned reassessment of Appendix A to 10 CFR Part 100 and will further consider Professor Jennings' suggestions at that time.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 11, 1983

The Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Palladino:

SUBJECT: QUANTIFICATION OF SEISMIC DESIGN MARGINS

In its reports on several different nuclear power plants during the past few years, the ACRS has asked that aspects of the adequacy of the seismic design be examined. These plants include the North Anna, Davis Besse, Summer, Clinton, Sequoyah, Wolf Creek, Fermi 2, Clinch River, Midland, and Perry projects. Most of these cases introduced some site-specific or design-specific topics into the overall consideration of seismic safety. However, the general aim of the ACRS in this regard was that the seismic design be adequate. In its report on Perry, the ACRS stated its belief that it was important that there be considerable assurance that the combination of the seismic design basis and the procedures used to establish the design margin be such that the seismic risk represents an acceptably low contribution to the overall risk.

The issue of seismic safety is generic to all nuclear power plants. The seismic initiator has been included in some recent probabilistic risk assessments (PRA) and, although the uncertainties are considerable, earthquakes have been estimated to represent a significant contribution to risk. It appears possible to draw some tentative conclusions, including the following:

1. The NRC has seismic design criteria. However, the effectiveness of these criteria in controlling risk is not well quantified.

2. Current determination of the seismic contribution to risk involves considerable uncertainty; furthermore, there are several important contributors to this uncertainty.

3. The available evidence indicates that the seismic contribution to risk may be large enough that a careful evaluation of the matter is warranted.

4. There are some aspects of nuclear power design where considerable seismic safety margins have been identified. There are other parts of the nuclear power plant for which margins may exist, but they have not been identified with any certainty. There are also some areas of plant design that have not yet been examined for margins. Data on equipment fragility based on tests vary in quality, completeness, and usefulness.

The general issue of seismic safety margins was discussed with representatives of the NRC Staff at ACRS Subcommittee meetings on August 11 and October 21-22, 1982.

We believe that a possible approach to this issue would be for the Nuclear Regulatory Commission, as part of its evaluation of proposed safety goals, to include a probabilistic evaluation of the seismic risk to light water reactor (LWR) safety. This might include the following programs to be conducted during the next two to four years.

(a) Continued Research on Sources and Likelihood of Severe Earthquakes

The ongoing U.S. research program on geology, seismicity and geophysics should continue to receive strong support from the NRC. The recent reevaluation by the U.S. Geologic Survey of the possible causes and potential of a Charleston-like earthquake in the Eastern United States results in large part from such studies and gives strong support to the likely importance of a probabilistic examination of seismic design. The NRC research program in this area should attempt to develop an improved capability for estimating the severity of earthquake-induced ground motion which may have a return frequency in the range of 10^{-4} to 10^{-5} per year at reactor sites.

(b) Generic Studies of Seismic Risk (NRC and Industry)

These studies would involve analyses intended to better ascertain the likely margins beyond the design basis, as well as those design criteria and practices that tend to enhance or diminish such margins. This would be done for all representative systems, components, or structures whose integrity of function is required in order to achieve safe shutdown, given a severe earthquake.

In addition to assessing specific components, etc., a limited seismic PRA should be performed with particular attention to identifying potentially significant failure modes and to examining in detail the uncertainties and their susceptibility to quantification and reduction. Such a program should be carried out with as much cooperation as practical between the industry and the NRC and its contractors. As in the case of severe accident rulemaking, the industry should establish an appropriate generic studies group. We believe that significant industry involvement is essential if this effort is to be successful.

(c) Seismic Reviews of Specific Plants (Licensees)

Plant-specific seismic reviews would involve limited PRAs performed to gain more insight into the relative seismic contribution to risk and the uncertainties therein, to evaluate the seismic adequacy of the specific plants involved, and to benefit from review of plant-specific details not available from generic studies. For example, seismically induced systems interactions can best be examined in terms of the detailed design of individual plants. Similarly, the seismic behavior of small lines or other components, which are not currently evaluated systematically, would best be examined for actual plants. A few plants might be chosen for a limited seismic PRA partly based on special considerations such as population density and partly based on their potential for providing a reasonable representation of the design and construction approaches used by the vendors and architect-engineers.

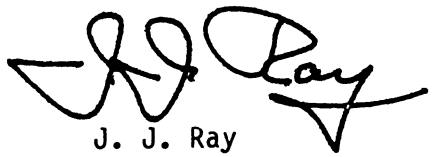
(d) Seismic Safety Research (NRC and Industry)

This program would be intended primarily to provide improved methodology for evaluating seismic risk and to obtain improved data on the seismic resistance capability of components (fragility). A beginning has been made on these matters in the Seismic Safety Margins Research Program (SSMRP) and the seismic risk studies included in the recent industry-sponsored PRAs for Zion and Indian Point. Since the fragility cannot be evaluated adequately for many components by analysis, a suitable experimental program should be undertaken. We believe that such a program should take advantage, to the extent possible, of assessment of damage in industrial facilities which have experienced severe earthquakes, of the Japanese program, and of qualification tests performed for other purposes. A systematic examination of the state of current knowledge should be made for all components of interest and a test program developed to provide generic information, if possible, for components that appear to lead to a greater vulnerability, separately or in combination. The experimental program should be given the benefit of insight from system studies and PRA studies undertaken in other sectors of the overall effort.

The proposed use of seismic probabilistic risk analysis would clearly be subject to large uncertainties in results, as well as to large differences of opinion among experts. Nevertheless, such studies, if performed carefully, can provide significant insight. Furthermore, if a probabilistic approach were used in design, some of the seismically induced failure modes predicted in recent PRAs, such as building-building interaction, could be avoided.

The ACRS believes it likely that the NRC will have need for a probabilistic evaluation of the seismic contribution to risk. If the NRC chooses to give sufficient priority to study of the matter and sets up a suitable schedule for decision making in this regard, the ACRS believes that such a generic approach could provide an appropriate substitute for examining certain seismic safety questions case by case.

Sincerely,



J. J. Ray
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 12, 1983

MEMORANDUM FOR: William J. Dircks
Executive Director for Operations

FROM: M. W. Libarkin, Acting
Executive Director, ACRS

SUBJECT: COMMENTS ON THE PILOT PROGRAM TO UTILIZE SEISMIC EXPERIENCE DATA FOR SEISMIC QUALIFICATION

During its 279th meeting, July 7-9, 1983, the Advisory Committee on Reactor Safeguards had the benefit of a briefing concerning the Pilot Program Report, "Program for the Development of an Alternate Approach to Seismic Equipment Qualification," by the Seismic Qualification Utilities Group (SQUG). This report was also discussed by the ACRS Subcommittee on the Qualification Program for Safety Related Equipment during a meeting in Washington, DC on March 15, 1983. This report addresses a preliminary investigation of the performance of certain classes of equipment in conventional power plants during selected California earthquakes, with the goal of utilizing these data to demonstrate the seismic adequacy of similar equipment installed in operating nuclear power plants.

The Committee was also informed that a Senior Seismic Review Panel (SSRP) has been formed to give the SQUG guidance concerning additional work required and how the data obtained would be utilized.

In its letter to Chairman Palladino on January 11, 1983 concerning seismic margins, the Committee recommended that, as part of the seismic safety research program, the NRC and industry should take advantage of the assessment of industrial facilities which have experienced severe earthquakes. The Committee believes that the program undertaken by the SQUG is in line with its recommendation and should be encouraged. However, the Committee also believes that more work is required to establish the operability of equipment during and after an earthquake and more data will be required to support conclusions drawn concerning the seismic resistance of the equipment investigated.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 6, 1983

Honorable Victor Gilinsky
Commissioner
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Commissioner Gilinsky:

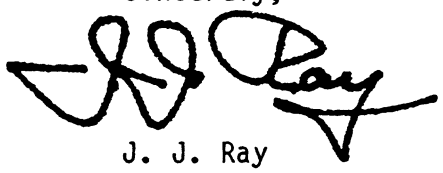
In your letter of July 8, 1983 you asked the ACRS whether the "tau effect" has a sufficiently sound scientific or engineering basis to justify its use to reduce seismic design standards that apply to nuclear power plants and whether the NRC should do further research on this question. In our review of the matter during the 281st ACRS meeting, August 31-September 1, 1983, we had the benefit of written reports from our consultants and from the NRC Staff (References 1-3) and of discussions with representatives of the NRC Staff.

The term "tau effect" was used in a report prepared by Dr. Nathan Newmark for the NRC Staff during the Diablo Canyon operating license review and symbolizes the time required for seismic waves having a horizontally propagating component to traverse the length of the foundation. The longer this "transit" time, the greater the foundation-averaging of this component of the seismic waves. Foundation-averaging can also occur because of randomization of some of the waves due to the effects of inhomogeneities. However, the "tau effect" is only one aspect of the overall kinematic-interaction effects that result from the kinematic restraints on ground motion imposed by a sufficiently rigid and extended foundation. For example, wave averaging is likely to be accompanied by increased rocking and torsional response.

We believe that, in general, there do not currently exist sufficient experimental data and adequate theoretical analysis to justify specific consideration of the "tau effect" routinely in the review of the seismic design of nuclear power plants. However, we would not rule out consideration of kinematic effects resulting from random spatial variations of ground motion in an assessment of seismic design margins when adequate ground motion and foundation/structural response models are used and when adequate data are available from sufficiently similar sites.

We believe that the general area of soil-strucure interaction warrants further research but would not single out the "tau effect" for specific emphasis.

Sincerely,



J. J. Ray
Chairman

References:

1. Letter from D. L. Bernreuter and James J. Johnson, Lawrence Livermore National Laboratory, to Dr. David Okrent, ACRS, Subject: LLNL Response to ACRS Questions on the Tau Effect, dated August 18, 1983
2. Letter from J. Enrique Luco, ACRS consultant, to Dr. R. Savio, ACRS, dated August 24, 1983 enclosing the subject report, "Comments on the Use of Kinematic Interaction Effects (Tau Effect) in Establishing Seismic Design Criteria"
3. Memo from W. J. Dircks, EDO, NRC, for Raymond F. Fraley, ACRS, Subject: ACRS Request for Staff Comments on the Use of the Tau Effect, dated August 30, 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 17, 1983

Honorable Victor Gilinsky
Commissioner
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Gilinsky:

SUBJECT: ACRS COMMENTS ON USE OF THE "TAU EFFECT" AT THE DIABLO CANYON NUCLEAR POWER PLANT

During its 282nd meeting, October 13-15, 1983, the Advisory Committee on Reactor Safeguards (ACRS) considered the questions you raised in your letters of September 8 and 28, 1983. A meeting of the ACRS Subcommittee on Extreme External Phenomena was held on October 12, 1983 to consider these questions. During its review, the Committee had the benefit of discussions with representatives of the Pacific Gas and Electric Company and the Joint Intervenors as well as with its own consultants and the NRC Staff.

In your letter of September 8, 1983 you asked for our opinion on, "whether it was appropriate to use the 'tau effect' at Diablo Canyon." In your letter of September 28, 1983, you asked, in particular, "whether the Committee believes the specific quantitative reductions applied to the Diablo Canyon spectrum are justified on the basis of scientific or engineering analysis."

The "specific quantitative reductions applied to the Diablo Canyon spectrum" were based on the recommendations of the NRC Staff's expert consultant, the late Dr. N. M. Newmark. Although Dr. Newmark's recommendations were based in part on engineering analyses that had been made by him and others in an attempt to understand and explain the results of observations and measurements of the behavior of structures subjected to seismic motions, the quantitative reductions recommended by him were based chiefly on his judgment and experience and only secondarily on the results of those analyses.

Although the state of the art has advanced considerably since Dr. Newmark made his recommendations, we do not believe that scientific or engineering analyses exist today that could be used to calculate the specific quantitative reductions in free-field seismic spectra that he recommended for the Diablo Canyon Nuclear Power Plant.

With regard to the question in your letter of September 8, 1983, the reduction attributed to the "tau effect" was only one element in a complex process leading to the development of design bases and criteria for the reevaluation of the Diablo Canyon Nuclear Power Plant for the Hosgri earthquake. Although each of these elements had been reviewed exhaustively by an

ACRS Subcommittee and by the full Committee, the Committee in its July 14, 1978 letter chose not to comment on or endorse the individual elements of the seismic design bases and criteria proposed by the NRC Staff. Instead, the Committee concluded only that "the use of the Staff approach leads to an acceptable level of safety in this instance." For the present Committee to do more would require essentially a new evaluation of the tau-effect issue, both to accommodate those current members who did not participate in the previous reviews and to refresh the memories of those who did. To perform such a review would require a major expenditure of time and resources by both the Committee and the NRC Staff. We do not propose to undertake such a review without further direction.

We do, however, call your attention to the penultimate paragraph in the ACRS letter of July 14, 1978 on the Diablo Canyon Nuclear Power Plant, which suggested "that the seismic design of Diablo Canyon be reevaluated in about ten years taking into account applicable new information."

Dr. Forrest J. Remick did not participate in the Committee's deliberations regarding this matter.

Sincerely,

J. J. Ray
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 18, 1984

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: QUANTIFICATION OF SEISMIC DESIGN MARGINS

On January 11, 1983, the ACRS wrote you regarding the quantification of seismic design margins and a possible program to evaluate the contribution to overall LWR risk that arises from earthquakes. We discussed and elaborated on the concerns and recommendations in that letter during our meeting with the Commission on November 17, 1983.

We proposed that a generic solution to our concerns about seismic margins and seismic risk could best be obtained by a four-part program involving both the NRC and the industry. In a memorandum to Commissioner Ahearne dated April 4, 1983, the EDO concurred that such an effort was timely. We understand that a task group has been established within the NRC Staff to consider and develop a program, but we have seen no reports of the activities of this group.

The ACRS Subcommittee on Extreme External Phenomena has held several meetings with NRR and RES Staff members and their consultants or contractors to discuss our concerns and to learn what was being done to address them. These meetings have been useful, but it does not appear that much progress has been made toward developing or implementing an integrated program on seismic margins.

Our ongoing review of the research program and budget for FY 1985 revealed a proposed project designated "Seismic Margins." There are other projects on seismic risk assessment and on seismic resistance (fragility) that would seem to be related to our concerns, but there does not seem to be a focused plan of research to address the problem.

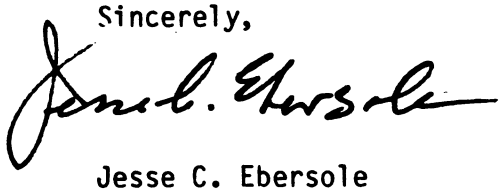
Of equal or greater importance is the fact that we have seen no viable plan for addressing the generic and plant-specific seismic PRAs recommended in our proposed program. We believe that seismic PRAs, properly done and evaluated, can contribute significantly to the resolution of our concerns.

A particular challenge will be how to obtain the cooperation of licensees in connection with the plant-specific seismic PRAs and that part of the program relating to fragilities. Since the resources required to perform a seismic PRA would be substantially reduced if a conventional PRA had already been done, or was being done, the proposed Integrated Safety Assessment Program might be considered as a medium for obtaining the limited number of seismic PRAs required.

The several programs described in our letter and discussed above will require the involvement of both NRR and RES within the NRC as well as individual utility licensees or an industry group, as appropriate. For this reason, the program has little chance of being initiated, much less completed, without strong centralized direction, clearly defined objectives, adequate funding in both RES and NRR and, above all, support and direction from the Commission.

The Committee or its appropriate Subcommittee is available to meet with the NRC Staff management to discuss approaches to addressing various aspects of this matter.

Sincerely,



Jesse C. Ebersole
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 16, 1984

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS COMMENTS ON THE STATUS OF USI A-46 (SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS)

During its 289th meeting, May 10-12, 1984, the Advisory Committee on Reactor Safeguards was briefed on the status of USI A-46 (Seismic Qualification of Equipment in Operating Plants) and its forthcoming resolution which includes the use of past earthquake experience data to establish the seismic ruggedness of certain classes of nuclear power plant equipment. The briefing included presentations by representatives of the Seismic Qualification Utilities Group (SQUG), the Senior Seismic Review and Advisory Panel (SSRAP), and the NRC Staff. A more detailed briefing was presented to the ACRS Subcommittee on the Qualification Program for Safety-Related Equipment during meetings in Washington, D. C., on March 29 and April 25, 1984.

In a letter to William J. Dircks, on July 12, 1983, the Committee acknowledged and encouraged the program undertaken by the SQUG. The Committee cautioned, however, that more work was required to establish the operability of equipment during and after an earthquake, and that more data will be required to support conclusions drawn concerning the seismic resistance of the equipment involved. We now find that USI A-46 is approaching resolution based on the SQUG/SSRAP efforts and recommend that the utility group program, its advisory panel and corresponding NRC Staff review work be expedited. We endorse this approach as a valuable contribution to the resolution of this issue.

At this time we have the following comments concerning Task 4 (Seismic Qualification of Equipment Using Seismic Experience Data) of the USI:

1. The problem description for USI A-46 as it appears in NUREG-0606 (Vol. 6, No. 1) indicates that the objective of this USI is to, "establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants.... Also, explicit guidelines will be established for use in requalifying equipment whose seismic qualification was found to be inadequate." The Committee has been informed that such guidelines are being developed. We intend to review them before commenting on the final resolution.

2. For the eight classes of equipment considered, we believe that the work to date by the SQUG and the NRC Staff has established the feasibility of using seismic experience data to verify seismic ruggedness. However, we continue to caution that more work is required to establish similarity rules or other guidelines for judging the operability of the full spectrum of seismically sensitive systems and equipment challenged during and after an earthquake. More data (e.g., more types and larger samples) may be required to support conclusions drawn concerning the seismic adequacy of the equipment involved. Particular attention should be given to the operability of control equipment (e.g., relays or instruments and their controllers) during an earthquake. The preponderance of relay control logic in the operating plants should be of special concern as it relates to "relay chatter" considerations. Attention should be directed also to other classes of safety-related equipment not covered by the SQUG program (e.g., HVAC, diesel generators, auxiliary steam turbines, piping and heat exchangers) and to the essential supporting services for such equipment including control air, electric power, and cooling water. It is not clear that sufficient seismic experience data exists for some of this equipment.

3. We were impressed by the professional character of the SSRAP independent assessment. We agree in principle with the conclusions stated in the SSRAP report, "Use of Past Earthquake Experience Data to Show Seismic Ruggedness of Certain Classes of Equipment in Nuclear Power Plants," dated February 1984. However, we emphasize the importance of observing carefully the numerous caveats, exclusions, and reservations stated in the report and note the proposed applicability of the report only to limited classes of equipment and equipment configurations, and only to post-earthquake operability. We reserve judgment concerning the proposed seismic motion bounds below which it is unnecessary to perform explicit seismic qualification to demonstrate functionality.

We understand that a generic letter on the proposed resolution and implementation of USI A-46 will be issued in the near future the under the provisions of 10 CFR 50.54f. We intend to review this letter during the public comment period.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

Fire Protection

Fluor Power Services (BOPSSAR)

F

Fire Protection

Fluor Power Services (BOPSSAR)

F

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 14, 1980

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: PROPOSED RULE MAKING ON FIRE PROTECTION - SECY-80-438A

Dear Dr. Ahearne:

During its 246th meeting, October 9-11, 1980, the Advisory Committee on Reactor Safeguards considered the proposed Rule on fire protection as submitted to the Commission with SECY-80-438A. Fire protection for nuclear plants has been considered by the ACRS and its subcommittees on a number of occasions since the Browns Ferry fire. This has involved Branch Technical Position (BTP) APCSB 9.5-1, Appendix A to that BTP, and a proposed Regulatory Guide based on that BTP.

The proposed Rule consists of a new Section 50.48 and a new Appendix R to 10 CFR Part 50. Appendix R provides detailed requirements for fire protection for those plants operating prior to January 1, 1979. Appendix R includes seventeen items from the much longer list in Appendix A of BTP 9.5-1. Some of these are further amplified or explained in Appendix R, and in three cases, more restrictive interpretations are proposed in the light of new information or new perceptions.

We have never disagreed with the need for greatly improved measures for fire protection or with the criteria embodied in BTP 9.5-1. Nor have we disagreed, in most cases, with the NRC Staff's interpretation of the criteria or with the implementation of the requirements of BTP 9.5-1.

However, we still do not believe that the extremely prescriptive requirements of BTP 9.5-1 provide the best means to ensure safety. This view was expressed by an ACRS Subcommittee in its review and disapproval of proposed Regulatory Guide 1.120. We are even more concerned about the proposal to incorporate the prescriptive criteria of BTP 9.5-1 into the Rules, either directly (for seventeen items) or by footnote reference.

We are aware of the several instances of technical disagreement between the NRC Staff and certain licensees regarding the implementation of the criteria in BTP 9.5-1. Although the adoption of the proposed Rule, including Appendix R, may accelerate the resolution of these disagreements by requiring strict compliance with the NRC Staff's position, we believe that the issuance of compliance orders would be more likely to provide optimized and integrated installation of these backfitted features for fire protection.

Although we do not favor the issuance of this Rule, if the Commission decides that its adoption is necessary or preferable, we urge strongly that the provisions of Sections 50.48(b) and 50.48(c) be adopted as proposed in the revised Rule.

Section 50.48(b) exempts from the provisions of Appendix R those fire protection features that have previously been accepted by the NRC Staff as meeting the provisions of Appendix A to BTP 9.5-1 as reflected in Staff fire protection Safety Evaluation Reports already issued, or those features accepted by the Staff in comprehensive fire protection Safety Evaluation Reports issued before Appendix A to BTP 9.5-1 was published in August 1976. We believe that such exemptions are justified subject to the NRC Staff's assurance that all such approved features provide protection equivalent to that prescribed in the BTP.

Section 50.48(c) provides a more relaxed, and substantially more realistic and reasonable, implementation schedule than that in the original version of the proposed Rule. The ACRS believes that this schedule is needed in order to assure that time is available for proper engineering design and proper installation of the required fire protection features. We believe that any additional risk to the public resulting from this delayed implementation will be small and will be more than offset by the decrease in risk resulting from properly engineered and installed fire protection features.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 20, 1977

United States Nuclear
Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON THE FLUOR PIONEER, INC. BALANCE OF PLANT STANDARD SAFETY ANALYSIS REPORT (BOPSSAR) AS APPLIED TO THE WESTINGHOUSE ELECTRIC CORPORATION RESAR-41 NSSS DESIGN

Dear Commissioners:

During its 207th meeting, July 14-15, 1977, the Advisory Committee on Reactor Safeguards completed its review of the application of Fluor Pioneer, Inc. for Preliminary Design Approval of its BOPSSAR, a standardized nuclear balance of plant (BOP) design that would interface with a single unit Westinghouse Electric Corporation, RESAR-41 pressurized-water reactor nuclear steam supply system (NSSS). A review of the RESAR-41 design was completed at the 185th meeting of the Committee and was discussed in its report of September 18, 1975. A Subcommittee meeting was held with representatives of Fluor Pioneer, Inc. and the Nuclear Regulatory Commission (NRC) Staff in Washington, DC on June 28, 1977. During its review the Committee had the benefit of discussions with the Applicant and the NRC Staff and of the documents listed.

The arrangement of BOPSSAR provides extensive physical separation of critical safety-related equipment to protect against common mode failures associated with fires or other operational contingencies. However, complete design details for BOPSSAR have not been developed and the concept has not yet been applied to a complete nuclear power plant design. Consequently, further review of the physical separation arrangement should be made prior to the Final Design Approval or when BOPSSAR is proposed for a nuclear power plant for which a construction permit is being sought.

The Committee believes that the safety-related interface requirements between BOPSSAR and RESAR-41 NSSS designs, and between BOPSSAR and the custom-designed site-related structures and components, are satisfactory for a Preliminary Design Approval, but expects the NRC Staff and the Applicant to examine them in greater depth when a construction permit application is received.

The coordination of interdependent instrumentation and controls of the NSSS and BOP will require attention at the time when BOPSSAR is used as a portion of a nuclear power plant license application. The Committee recommends that these matters be included in the NRC Staff's Standard Review Plan.

Although BOPSSAR and RESAR-41 include provisions for protection against industrial sabotage, the Committee believes that further steps can be taken beyond those provided. Before a construction permit referencing BOPSSAR and RESAR-41 is issued for a nuclear power plant, the Utility-Applicant should be required to demonstrate that acceptable provisions to preclude or to mitigate the consequences of industrial sabotage will be incorporated into the plant design.

The BOPSSAR design includes some provisions which anticipate the maintenance, inspection, and operational needs of the plant throughout its service life, including cleaning and decontamination of the primary coolant system, and eventual decommissioning. However, when BOPSSAR is used as a portion of a nuclear power plant license application the Committee believes that the NRC Staff and the Applicant should further review methods and procedures for removing accumulated contamination whereby maintenance and inspection programs and ultimate decommissioning can be more effectively and safely carried out.

Generic problems related to large water reactors are discussed in the Committee's report dated February 24, 1977. Those problems relevant to BOPSSAR and RESAR-41 should be dealt with appropriately by the NRC Staff and the Utility-Applicant as solutions are found.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above should be given further consideration during the plant licensing process and that, if due consideration is given to the foregoing, Preliminary Design Approval of BOPSSAR for use in conjunction with RESAR-41 can be granted in accord with the spirit and purposes set forth in the Commission's policy statement on standardization of nuclear power plants.

Sincerely yours,

M. Bender

M. Bender
Chairman

References

1. Fluor Pioneer, Inc. Balance of Plant Standard Safety Analysis Report, BOPSSAR, Volumes 1 through 8 and Amendments 1 through 10
2. Report to the Advisory Committee on Reactor Safeguards in the Matter of Fluor Pioneer, Inc. Balance of Plant Standard Safety Analysis Report June 3, 1977 by the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 16, 1979

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: REPORT ON THE FLUOR POWER SERVICES INCORPORATED BALANCE-OF-PLANT STANDARD SAFETY ANALYSIS REPORT (BOPSSAR) AS APPLIED TO THE BABCOCK-205 STANDARD NUCLEAR STEAM SUPPLY SYSTEM

Dear Dr. Hendrie:

During its 226th meeting, February 8-10, 1979, the Advisory Committee on Reactor Safeguards reviewed the application of Fluor Power Services Incorporated (formerly Fluor Pioneer Inc.) for Preliminary Design Approval of its BOPSSAR, a standard nuclear balance-of-plant (BOP) designed to interface with a single-unit Babcock and Wilcox Company Babcock-205 standard pressurized-water reactor nuclear steam supply system (NSSS). The Fluor BOPSSAR design has previously been reviewed by the Committee in relation to the Westinghouse RESAR-41 standard NSSS. The Committee's report on this review is dated July 20, 1977. The Committee's report on the Babcock and Wilcox Company Babcock-205 standard NSSS is dated August 18, 1977. The BOPSSAR design as applied to a Babcock-205 NSSS was considered at a Subcommittee meeting in Washington, D.C. on January 26, 1979. During its review, the Committee had the benefit of discussions with representatives of Fluor Power Services Inc. and the NRC Staff. The Committee also had the benefit of the documents listed.

The Committee believes that the BOP design proposed in BOPSSAR is acceptable for a Preliminary Design Approval. The Committee believes, however, that the several items mentioned in its report on BOPSSAR in relation to RESAR-41 are applicable also to the use of BOPSSAR with Babcock-205. These matters should be addressed by the NRC Staff and the Utility-Applicant, as appropriate, during the review of an application for a construction permit for a nuclear plant utilizing the BOPSSAR and Babcock-205 standard designs. These items include: separation of critical safety-related equipment; safety-related interface requirements; coordination of interdependent instrumentation and controls; provisions for protection against industrial sabotage; and provisions which anticipate the maintenance, inspection, and operational needs of the plant throughout its service life.

With regard to the generic items cited in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977, those items considered relevant to the use of BOPSSAR with a Babcock-205 NSSS are: II-2, 5B, 6, 9; IIA-4; IIC-1, 2, 3A, 3B, 4, 5, 6; IID-1, 2; IIE-1. These matters should be dealt with by the NRC Staff and the Applicant, as appropriate, when solutions are found.

The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the foregoing and to the recommendations in the Committee's report of August 18, 1977 on Babcock-205, Preliminary Design Approval can be granted for the use of BOPSSAR in conjunction with Babcock-205.

Sincerely yours,

Max W. Carbon
Chairman

References:

1. Fluor Power Services Inc. Balance of Plant Standard Safety Analysis Report, BOPSSAR, Volumes 1 through 11 and Amendments 1 through 20.
2. Report to the Advisory Committee on Reactor Safeguards in the Matter of Fluor Power Services Incorporated Reference Safety Analysis Report (BOPSSAR) as Related to BSAR-205, December 1978 by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.
3. Letter dated January 4, 1978 from W. W. Larkin, Fluor Pioneer Inc. to B. C. Rusche NRC containing information regarding tests to confirm the coefficient of friction between concrete and steel surfaces.

General Electric Company

Generic Items Relating to Light Water Reactors

G

General Electric Company

Generic Items Relating to Light Water Reactors

G

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 21, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: BOILING WATER REACTOR DESIGN, BWR/6

Dear Dr. Schlesinger:

At its 148th meeting, on August 10-12, 1972, the Advisory Committee on Reactor Safeguards completed its review of the design concept and expected performance of the new General Electric Company BWR/6 boiling water reactor product line. This concept was discussed at the 143rd ACRS meeting on March 2-4, 1972, and at Subcommittee meetings held in Washington, D. C., on March 21 and July 20, 1972. During its review, the Committee had the benefit of discussions with representatives of the General Electric Company and the AEC Regulatory Staff, and of the documents listed below.

The BWR/6 concept provides a 20 percent increase in thermal output over the previous product line of boiling water reactors with the same reactor vessel sizes. At the same time, changes in design provide reduction in the linear heat generation rates from 18.5 to 13.4 kw/ft, for the maximum, and from 7.1 to 6.0 kw/ft, for the average. The BWR/6 concept uses the same emergency core cooling systems as those in the previous designs.

Jet pumps have been made more compact, permitting an increase in the core size to accommodate a 10 percent increase in the number of fuel assemblies. Furthermore, the jet pumps have been improved to provide a 10 percent increase in flow. The overall fuel assembly size has been retained, but the fuel rods have been reduced in diameter and an 8x8 fuel assembly is used instead of the previous 7x7. The power density is to be increased 10 percent, from 51 to 56 kw/liter. The active fuel length has been increased from 144 to 148 inches.

Significant decreases in the local peaking factor are to be achieved through changes in the control rod blade thickness and fuel assembly channel wall thickness, leading to reduced and symmetric water gaps.

Further improvement is to be achieved through the use of one or more central water rods. The use of axially distributed gadolinia leads to a reduction in the axial peaking factor.

With these changes, the peak clad temperatures calculated by the applicant for the design basis loss-of-coolant accident (LOCA) are appreciably lower than those for previous designs. The lower peak clad temperatures and the increased thickness of the Zircaloy cladding provided in the BWR/6 concept combine to suggest that there would be few perforations in the event of a LOCA. Detailed evaluation of the stated ECCS performance remains to be accomplished.

Programs to demonstrate and evaluate equipment performance are in progress, including tests of control rods and jet pumps. Other tests to be completed are critical heat flux measurements and core spray and flooding heat transfer studies. Additional documentation for the nuclear, hydraulic, and mechanical design of the new configuration will be required to more thoroughly establish the appropriate safety margins and to provide adequate justification of the suitability of the proposed design. Design aspects related to possible reactivity transients also remain to be evaluated.

The Committee reiterates its previous comments concerning the need to study further means of preventing common mode failures from negating scram action and the design features to make tolerable the consequences of failure to scram during anticipated transients. Also resolution should be expedited of other problems relating to large water reactors, which have been identified by the Regulatory Staff and the ACRS and cited in previous reports.

The Committee believes that, subject to the above comments and to successful completion of the remaining confirmatory tests and analytical studies, the BWR/6 concept can be successfully engineered to provide an acceptable boiling water reactor design.

Sincerely yours,

C. P. Siess
Chairman

References Attached.

References - BWR/6

1. General Electric Topical Report NEDO 10569A BWR/6 Nuclear System from General Electric -- A Performance Description

2. General Electric Company Licensing Topical Report NEDO-10565, Testing of Cruciform Control Rods for BWR/6

3. General Electric Company Licensing Topical Report NEDO-10602, Jet Pump Testing for BWR/6

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 15, 1983

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: ACRS REPORT RELATED TO THE FINAL DESIGN APPROVAL OF THE GESSAR II BWR/6 NUCLEAR ISLAND DESIGN

Dear Dr. Palladino:

During its 278th meeting, June 9-11, 1983, the Advisory Committee on Reactor Safeguards completed a limited review of the reference design described in General Electric Standard Safety Analysis Report (GESSAR II) for a Final Design Approval (FDA). GESSAR II provides the safety information for a reference system consisting of a single BWR/6 Mark III nuclear steam supply system, with a rated core thermal power of 3579 MWt, and associated systems and structures, including the reactor building (the shield building and containment), fuel building, diesel generator buildings, control building, auxiliary building, and radwaste building. A Subcommittee meeting was held with representatives of the General Electric Company (the Applicant) and the Nuclear Regulatory Commission (NRC) Staff on April 21-22, 1983 in Santa Monica, California. The full Committee considered this matter during its 277th meeting, May 12-14, 1983. The Committee also had the benefit of the documents referenced.

The Committee has reviewed the GESSAR II FDA application in the context requested by the NRC Staff, that is, as being applicable only to those plants with a viable construction permit (CP) which referenced the Preliminary Design Approval (PDA) in their CP application. Our considerations to date have not developed to the point that we can endorse the use of GESSAR II beyond this scope.

The NRC Staff advised the ACRS that, if the proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation is approved containing its current Section VI on Standardization Policy, the Staff would shortly thereafter consider the GESSAR II design for an interim FDA under Section VI for future CP applications while awaiting completion of Commission action on severe accident policy. This aspect of Section VI is among the several aspects of the proposed Policy Statement which the ACRS does not favor. If the GESSAR II design (or some other proposed FDA) is brought to the ACRS for review in connection with a new plant prior to specific action by the Commission on severe accident policy, the Committee would expect to perform an ad hoc evaluation of such considerations as part of its own review.

It will be necessary for the General Electric Company to maintain vigilance over the interface boundaries between the standard nuclear island and the balance of plant. Efforts to ensure compatibility across these interfaces must be formalized and rigorously applied. The Committee remains interested in the methods used to maintain interface compatibility and wishes to be kept informed.

In spite of many past assurances that stress-assisted corrosion cracking of primary system components was well understood and would be avoided in the future, significant cracking is being observed in many currently operating BWRs. The materials being proposed for this application are supposedly much improved. However, because of actual operating experience to date, we believe that the applicant should maintain a vigorous research program to identify material and water chemistry changes that will control or eliminate the stress-assisted corrosion cracking problem and should closely monitor the performance of functional systems utilizing the newer materials. The Committee wishes to be kept informed.

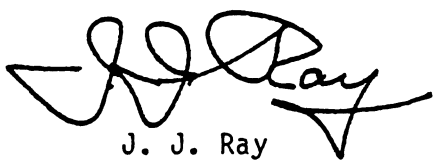
The Committee has in several prior operating license or construction permit reviews noted the importance of assuring that the seismic contribution to risk is acceptably low, with due allowance for lower frequency, more severe earthquakes than the safe shutdown earthquake. This recommendation applies equally well to the GESSAR II design.

The GESSAR II design does not include provisions for coping with the large amounts of hydrogen that would be generated in an accident involving severe core damage. It will be necessary for a utility applicant to make the appropriate provisions as part of the balance of plant.

At this point in the NRC Staff's review, no items remain as Outstanding Issues. There is a set of Confirmatory Issues that await additional documentation. We found no reason to believe that any of these issues will be especially difficult to complete and recommend that they be completed in a manner satisfactory to the NRC Staff.

The Committee believes that, subject to the comments noted above and approval of the balance-of-plant design, the GESSAR II design can be incorporated into those nuclear power plants with a valid CP that referenced the PDA in their CP application.

Sincerely,



J. J. Ray
Chairman

References:

1. General Electric Company Standard Safety Analysis Report, "GESSAR II, BWR/6 Nuclear Island Design," with Amendments through 14.

2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Final Design Approval of the GESSAR II BWR/6 Nuclear Island Design," NUREG-0979, dated April 1983.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 14, 1975

Honorable William A. Anders
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT (GESSAR-238)

Dear Mr. Anders:

At its 179th Meeting, March 6-8, 1975, the Advisory Committee on Reactor Safeguards completed a review of the General Electric Standard Safety Analysis Report (GESSAR). GESSAR-238 provides the safety information for a reference system consisting of a single BWR-6/Mark III nuclear system, with a rated core thermal power of 3579 MW(t), and of the associated systems including the reactor building (the shield building and containment), fuel building, auxiliary building, diesel generator buildings, control building, radwaste building,, and the off-gas system. Subcommittee meetings were held with representatives of the General Electric Company and the Nuclear Regulatory Commission (NRC) Staff on July 1, 1974, and September 11, 1974, in Washington, D. C., on November 9, 1974, in Bloomington, Minnesota, and on January 18, 1975, in Washington, D.C. The Committee also had the benefit of the documents listed below.

Site envelope parameters are included in GESSAR and application of GESSAR will require that specific site evaluations be made to confirm the acceptability of the site within the GESSAR design. The use of GESSAR for multiple reactor units at a single station will also require review of the safety-related components of plant duplication and layout.

Safety-related interfaces between the reference system and the balance of plant are specified in GESSAR. Since the utility-applicant is responsible for instituting the quality assurance programs necessary to assure that all safety-related interfaces have been identified and that all safety-related requirements are being fulfilled, the Committee will review these matters in more detail with the Applicants on a case-by-case basis. The Committee recommends that, during the design, procurement, construction and startup, timely and appropriate interdisciplinary system analyses be carried out to assure complete functional compatibility across each interface for an entire spectrum of anticipated operations and postulated design basis accident conditions.

The NRC Staff has identified 13 items requiring resolution prior to issuing their Preliminary Design Approval (PDA). The Committee believes that all of these matters should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed regarding the resolution of the following items:

1. Seismic capability of the offgas system.

2. Provisions to satisfy the single-failure criterion for the RHR system.

3. Additional requirements to be imposed if continuous venting of the containment is used.

4. Evaluation of the performance of the emergency core cooling systems using evaluation models meeting the requirements of 10 CFR 50.46, Appendix K.

The latest ACRS reports on nuclear generating stations utilizing the BWR-6/Mark III systems were the December 12, 1974 reports on the Allens Creek Nuclear Generating Station, Units 1 and 2, and the Perry Nuclear Power Plant, Units 1 and 2. In these reports, the ACRS has recommended that the ongoing R&D programs be used to fully resolve issues involving the Mark III containment design prior to completion of the affected portions of the plant. Further, additional generic matters, which include anticipated transients without scram (ATWS) and possible pump overspeed during a loss of coolant accident, should be dealt with appropriately by the NRC Staff. It is expected, that these items will be resolved in a manner satisfactory to the NRC Staff following Preliminary Design Approval (PDA) of GESSAR and prior to Final Design Approval (FDA). During this interim period, the Committee will continue to review these items on a case-by-case basis as well as through other appropriate ACRS Subcommittee meetings and full Committee meetings.

The Committee has not reviewed modifications which are expected to be made in the BWR/6 8x8 fuel. Such modifications and any other proposed changes will be reviewed when the appropriate documentation has been submitted and the improvements sought can be evaluated.

The introduction of new features in the instrumentation and control systems has been submitted through the specification of functional designs and design criteria which the NRC Staff has found to be adequate for the PDA. As in previous reports on related matters the Committee recommends that the NRC Staff determine the necessary environmental and reliability tests, including in situ tests where desirable for qualification of the new systems. In another matter relating to a periodic testing provision, the General Electric Company has committed to a study of the improvement of the testability of the automatic depressurization system. On all these issues involving instrumentation and control, the Committee will use the case-by-case basis to ascertain progress of the work until the GESSAR design has progressed to the stage where Final Design Approval is achieved.

The Committee will need to review the development and proof testing of the fast scram system, and the implementation of the proposed Reactor Manual Control System along with the provisions for ganged rod withdrawal.

The Committee believes that the General Electric Company and the NRC Staff should continue to review GESSAR for design changes that would further improve industrial security features.

The GESSAR design should include provisions which anticipate the maintenance, inspection, and operational needs of the plant throughout its service life, including cleaning and decontamination of the primary coolant system, and eventual decommissioning. In particular, the Committee believes that the NRC Staff and the General Electric Company should review methods and procedures for removing accumulations of radioactive contamination whereby maintenance and inspection programs can be more effectively and safely carried out.

The Committee believes that methods that seek to develop reference systems through standardization and through replication need to be coupled with ongoing programs that will permit changes which improve safety and which, when justified, would be implemented in a timely manner. Use of reference systems should lead to more efficient and effective licensing reviews. Programs such as GESSAR will contribute to this process. A transition period will be required in which the Committee would still give considerable attention to the items noted, on a case-by-case basis.

The Committee believes that, subject to the above comments and to successful completion of the R&D prgrams, GESSAR-238 can be successfully engineered to serve as a reference system.

Sincerely yours,

William Kerr
Chairman

References attached.

References

1. BWR/6 Standard Safety Analysis Report, Volume 1 through 7.

2. Amendments 1 through 28 to the Standard Safety Analysis Report.

3. General Electric Company letters and reports:

   a. July 31, 1973 letter forwarding proprietary information in support of the information made public in the safety analysis report.

   b. August 31, 1973 letter forwarding proprietary fuel data.

   c. September 28, 1973 letter forwarding proprietary information regarding core power distribution.

   d. December 28, 1973 letter regarding interfaces and electrical systems.

   e. November 6, 1974 letter regarding physics verification and number of safety/relief valves.

   f. February 19, 1974 letter regarding ATWS.

4. AEC/NRC Staff letters and reports:

   a. October 11, 1974 draft Safety Evaluation Report.

   b. November 13, 1974 Safety Evaluation Report.

   c. December 7, 1974 Supplement No. 1 to the Safety Evaluaion Report.

   d. January 30, 1975 letter regarding reevaluation of the high pressure drywell test.

   e. February 21, 1975 Supplement No. 2 to the Safety Evaluation Report.

   f. March 4, 1975 Supplement No. 3 to the Safety Evaluation Report.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 17, 1976

Honorable Marcus A. Rowden
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: GENERAL ELECTRIC STANDARD NUCLEAR STEAM SUPPLY SYSTEMS (GESSAR-238 NSSS and GESSAR-251)

Dear Mr. Rowden:

At its 200th meeting, December 9-11, 1976, the Advisory Committee on Reactor Safeguards completed its review of the applications by the General Electric Company for preliminary design approvals for the standardized nuclear steam supply systems described in the General Electric Standard Safety Analysis Reports, GESSAR-238 NSSS and GESSAR-251. These systems were also considered at the 199th meeting of the Committee, November 11-13, 1976. The review by the Committee was facilitated by the meeting of a Subcommittee with representatives of the General Electric Company and with members of the Nuclear Regulatory Commission (NRC) Staff on November 6 and 7, 1976, in Los Angeles, California. The Committee also had the benefit of the documents listed below.

These GESSAR systems consist of BWR/6 nuclear steam supply systems with thermal power ratings of 3579 MW for GESSAR-238 NSSS and 3800 MW for GESSAR-251. They do not include the containment structure, the reactor building, the fuel building, the auxiliary building, or other structures outside the nuclear systems, which are the responsibility of the utility-applicant.

GESSAR-238 NSSS and GESSAR-251 have essentially the same design, the principal differences arising from differences in power level. Many aspects of GESSAR-238 NSSS and GESSAR-251 are the same as those previously reviewed by the Committee in the General Electric Standard Safety Analysis Report for a nuclear island. The Committee's report on the GESSAR-238 Nuclear Island was issued on March 14, 1975.

The Committee and the Applicant discussed the possibility of damage to a heat exchanger of the Residual Heat Removal (RHR) system by overpressurization or by hydrodynamic forces that could conceivably result from valve malfunction or operator error. This hypothetical condition is associated with the steam condensing mode of operation of the RHR or when the Reactor Core Isolation Cooling system is in use. The Committee recommends that the NRC Staff review this problem, applying their usual

criteria of assuming malfunction and operator error, to determine whether such conditions can occur. The review should include particular study of the initial phase of steam entry, during which water initially present, or inadvertently accumulated, must be expelled from piping into and through the heat exchanger.

The Committee recommends that GESSAR-238 NSSS and GESSAR-251 incorporate appropriate systems to mitigate the consequences of an ATWS event.

Safety related interfaces between the reference system and the balance of plant are identified in these GESSAR's. It will be necessary for the NRC Staff to assure that all of the safety-related requirements are fulfilled when an application for a construction permit is filed. The Committee will review this matter in more detail when applications for construction permits referencing these systems are received.

The Committee recommends that, during the design, procurement, construction, and startup, timely and appropriate interdisciplinary system analyses be carried out to assure complete functional compatibility across each interface for an entire spectrum of anticipated operations and postulated design basis accident conditions.

Other generic problems relating to large water reactors are discussed in the Committee's report dated April 16, 1976. Those problems relevant to large boiling water reactors should be dealt with appropriately by the NRC Staff and General Electric Company as solutions are found. The relevant items are: II-3, 4, 5, 6, 7, 8, 9, 10, 11; IIA-2, 6; IIB-2, 4; IIC-1, 2, 3, 4, 6, 7.

The Committee believes that, subject to the above comments and to successful completion of the necessary R&D programs, GESSAR-238 NSSS and GESSAR-251 can be successfully engineered to serve as reference systems.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

References

1. 238 Nuclear Steam Supply System GESSAR and Amendments 1 through 4.

2. Report to the Advisory Committee on Reactor Safeguards in the Matter of General Electric Safety Analysis Report GESSAR-238 NSSS (Docket No. STN 50-550) Published: October 1976 by the U.S. Nuclear Regulatory Commission.

3. 251 General Electric Standard Safety Analysis Report (251 GESSAR) and Amendments 1 through 21.

4. Report to the Advisory Committee on Reactor Safeguards in the Matter of General Safety Analysis Report GESSAR-251 (Docket No. STN 50-531) Published: October 1976 by the U. S. Nuclear Regulatory Commission.

5. General Electric Company letter dated February 13, 1975 forwarding proprietary information regarding fuel assembly and core design.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 13, 1980

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: NRC ACCEPTANCE CRITERIA FOR THE MARK I CONTAINMENT LONG TERM PROGRAM

Dear Dr. Ahearne:

During its 238th meeting, February 7-9, 1980, the Advisory Committee on Reactor Safeguards reviewed the NRC Acceptance Criteria for the Mark I Containment Long Term Program. This matter was considered at ACRS Fluid Dynamics Subcommittee meetings held on May 23, 1978, November 28-30, 1978, September 13-14, 1979, and November 16, 1979. During its review, the Committee had the benefit of discussions with representatives of the NRC Staff and the Mark I Owners Group.

The NRC Acceptance Criteria for the Mark I Containment Long Term Program are intended to establish design basis loads that are appropriate for the anticipated life of each Mark I BWR facility and to restore the originally intended design safety margins to each Mark I containment system.

The Mark I program was initiated in 1975 in response to loss of coolant accident and safety relief valve (SRV) dynamic loads identified by the General Electric Company during the course of performing large scale testing for the Mark III pressure-suppression containment in 1972-1974. A period of reevaluation resulted in issuance of the Short Term Program Acceptance Criteria in December 1975 which established interim design bases for continued operation of the Mark I BWRs. The Acceptance Criteria for the Long Term Program have been developed from a program of small and full scale tests in two and three dimensional geometries.

The Mark I Owners submitted proposed loads in the "Mark I Containment Program Load Definition Report" in December 1978 and detailed the methods to be used in plant unique analyses in the "Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Applications Guide." Following review of the available information, the NRC Staff determined that certain changes and clarifications to the criteria proposed by the Mark I Owners were necessary. The NRC Staff technical requirements were delineated in the "NRC Acceptance Criteria for the Mark I Containment Long Term Program" issued in October 1979 and also in several additions to the acceptance criteria as discussed during the 238th ACRS meeting. The additions to the Acceptance Criteria were intended, in part, to alleviate some of the difficulties the Mark I Owners had in calculating credible structural responses to SRV actuations.

The Committee recognizes the thoroughness of the efforts taken by the NRC Staff and the Mark I Owners to resolve the generic Mark I issues and believes that the NRC Acceptance Criteria and additions, as proposed, provide a suitably conservative basis for performing the Long Term Mark I Containment structural response analyses. The Mark I Owners indicated that they continue to have significant difficulty in calculating credible structural responses to some SRV loads and they would like to continue to work with the Staff on a generic basis to resolve these difficulties. The NRC Staff would like to complete the generic Mark I program and resolve any remaining problems as they arise from the plant unique analyses. The Committee believes that the individual Mark I Owners can work with the Staff to resolve any additional difficulties that may arise from the plant unique analyses as modifications are being made to the containment structures.

The Committee believes that the Staff should assure the adequacy of the requirements for verification of the design, fabrication, and inservice inspection of the Mark I containment modifications and, in particular, the SRV discharge piping in the wetwell airspace. Further, in the interim period while the Mark I modifications are being performed, the Staff should investigate the potential for and consequences of a failure in the SRV discharge piping in the wetwell airspace for the existing designs. The Committee wishes to be kept informed on this matter.

The Committee believes that, with due consideration to the above items, the generic Mark I Long Term Program can be concluded and the modifications to the individual Mark I BWRs can be implemented on a reasonable schedule over the next 18 months.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. General Electric Company, "Mark I Containment Program Load Definition Report," Revision 0, NEDO 21888, December 1978.
2. General Electric Company, "Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Applications Guide," NEDO 24583, December 1978.
3. U.S. Nuclear Regulatory Commission, "NRC Acceptance Criteria for the Mark I Containment Long Term Program," October 1979, and additions included in the February 8, 1980, transcript of the 238th ACRS Meeting.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 14 1975

Lee V. Gossick
Executive Director for Operations

REVIEW OF GENERAL ELECTRIC MARK III CONTAINMENT DESIGN AND OTHER BWR CONTAINMENTS

In recent reviews of nuclear plants utilizing General Electric Mark III Containment, the Committee has been aware of a number of technical issues which have been left for future resolution as generic matters. These include post-LOCA pressure distribution, platform and pool loading during blowdown, suppression pool make-up requirements, functioning of the RHR systems with a single active failure, and response time testing, from sensor to actuator, of ESF's. The Staff has also identified other items that require resolution. Recently, the question of drywell pressure testing has been raised and there has been a change in the NRC Staff position on the basis for such testing. The Committee is also aware that some projects utilizing Mark III Containments are reaching the point at which resolution of some of these issues is necessary.

In addition, there is recent operating experience for other BWR containments regarding suppression pool behavior during primary system relief as well as experimental and analytical analyses of this phenomenon on which the Committee would like to be informed.

The Committee would like reports during its 181st meeting, May 8-10, 1975, on the status of the NRC Staff's review of BWR Containments.

W Kerr

W. Kerr
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

January 17, 1973

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON BOILING WATER REACTOR MARK III CONTAINMENT CONCEPT

Dear Dr. Schlesinger:

At its 153rd meeting, on January 11-13, 1973, the Advisory Committee on Reactor Safeguards completed its review of the General Electric Boiling Water Reactor Mark III Containment Concept. This design concept was discussed at a Subcommittee meeting held in Washington, D. C., on December 21, 1972. During its review, the Committee had the benefit of discussions with representatives of the General Electric Company and the AEC Regulatory Staff, and of the documents listed below.

The Mark III Containment Concept retains the pressure suppression pool, which is characteristic of boiling water reactor containments, and provides a larger volume than that of the torus-light bulb (Mark I) or the over-under (Mark II) designs. A cylindrical concrete drywell structure is provided within a cylindrical containment. The containment is a steel shell surrounded by a concrete shield building which may also serve as a second barrier. Alternatively, the containment may be of reinforced or prestressed concrete with a steel liner.

The General Electric Company has undertaken an experimental program, including a series of large scale confirmatory tests, to verify the performance of the new features of the pressure suppression design. An analytical program to determine the strength characteristics of the concrete drywell is also being undertaken.

The bases of the Committee's review were that Safety Guide 3 and Safety Guide 7 would be satisfied; that the drywell and containment would not be inerted; and that a BWR/6 nuclear system would be used - this system was the subject of a Committee report to you dated September 21, 1972.

The containment concept will utilize several systems for which the design criteria have not yet been established. The items include the vacuum relief system, the recirculating system between the drywell and the containment, and the system for coping with hydrogen. The Committee believes that the design criteria for these systems as well as for the overall containment design concept can be resolved in connection with the detailed review of a construction permit application.

The Committee believes that, subject to the above comments and to the successful completion of confirmatory tests and analytical studies, the Boiling Water Reactor Mark III Containment Concept can be engineered to provide an acceptable containment system.

Sincerely yours,

H. G. Mangelsdorf
Chairman

References:

1. General Electric Topical Report NEDO-10571 "General Electric Boiling Water Reactor Mark III Containment Concept"

2. Safety Evaluation by the Directorate of Licensing, U. S. Atomic Energy Commission in the Matter of General Electric Company Mark III Containment Concept dated October 5, 1972

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

February 12, 1974

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON GENERAL ELECTRIC 8X8 FUEL DESIGN FOR RELOAD USE

Dear Dr. Ray:

At its 166th meeting, February 7-9, 1974, the Advisory Committee on Reactor Safeguards completed a generic review of the design and expected performance of General Electric 8x8 fuel bundles to be employed in partial and full core reloads in boiling water reactors. These topics were discussed at the 165th ACRS meeting on January 10-12, 1974, and at Subcommittee meetings in Washington, D. C., on January 8, 1974, and in Denver, Colorado, on January 24, 1974. During its review the Committee had the benefit of discussions with representatives of the General Electric Company, the AEC Regulatory Staff, and of the documents listed below.

The General Electric 8x8 fuel assembly consists of 63 fuel rods plus one unfueled, water-filled, spacer-capture rod in a square 8x8 bundle array within a square channel box. The design of the fuel rods and fuel rod bundle in the 8x8 reload fuel assembly is, except for differences in the length of the fuel and gas plenum, the same as in the 8x8 fuel assembly used in the BWR/6 boiling water reactor concept referred to in the Committee's report of September 21, 1972. The 8x8 fuel bundles are interchangeable in General Electric boiling water reactors with the previously used 7x7 bundles.

In general, the thermal margins to fuel damage design bases are greater for 8x8 fuel than for 7x7 fuel. The design value of the linear heat generation rate for normal operation is 13.4 kw/ft for 8x8 fuel and 17.5 to 18.5 kw/ft for 7x7 fuel; specific power is slightly greater for the 8x8 fuel than for the 7x7 fuel. The General Electric Company believes the lower linear heat generation rate and slightly greater ratio of clad thickness to rod diameter should result in fewer failures in 8x8 fuel than in 7x7 fuel. Although the hydraulic resistance of 8x8 bundles is slightly greater than that of 7x7 bundles, the thermal-hydraulic performance of cores either partially or fully loaded with 8x8 assemblies is not degraded relative to cores loaded with 7x7 assemblies.

Since the U-235 enrichments for the individual fuel rods, the number and distribution of rods containing gadolinia, and the water-to-fuel ratio are similar in the 8x8 and 7x7 designs, the neutronic behavior of the two designs is not significantly different. The internally located water rod in the 8x8 design reduces rod-to-rod power peaking.

Since the neutronic and thermal-hydraulic characteristics of 8x8 and 7x7 fuel bundles are similar, their behavior under abnormal operational transients is expected to be comparable. The consequences of postulated control rod drop, fuel handling, and steam line break accidents are not expected to be significantly different for the two fuel designs.

The General Electric Company, several utility owner-operators of boiling water reactors, and the Regulatory Staff have performed analyses of the behavior of 8x8 fuel in several mixed and fully reloaded cores under transient and accident conditions. These analyses predict that, for boiling water reactors which have jet pumps, the peak clad temperatures during a postulated large-break LOCA using the Interim Acceptance Criteria are less for 8x8 than for 7x7 assemblies. However, for large postulated breaks in non-jet pump plants and for small and intermediate size breaks for all plants, the predicted peak clad temperatures are in the same range for both 8x8 and 7x7 fuel. Consequently, individual reviews by the Regulatory Staff of the expected performance of 8x8 reload fuel, including plant-specific system effects and any significant core fuel loading asymmetries, will be required for each reactor prior to operation with 8x8 fuel to determine limits on reactor operating conditions. The Committee wishes to be informed concerning the results of these reviews for the initial 8x8 fuel loadings in each of the several General Electric boiling water reactor product lines.

The Regulatory Staff plans to use the results of recently conducted spray-cooling and reflooding tests of a Zircaloy-clad bundle to verify analytical models for 8x8 fuel. The Committee wishes to be kept informed.

Experience and information exist both on the performance of fuel rods and bundles whose designs bracket the dimensions of the 8x8 fuel and on the performance of cores containing mixtures of assemblies with different numbers and sizes of fuel rods. Although mechanical tests have been performed on 8x8 fuel assemblies and components to demonstrate their integrity, additional tests to verify spacer grid strength have been requested by the Regulatory Staff. This matter should be resolved in a manner satisfactory to the Regulatory Staff.

The General Electric Company has planned a program of both pre- and post-irradiation examinations to monitor the performance of 8x8 fuel assemblies. The Committee endorses a comprehensive surveillance program and wishes to be kept informed.

The Regulatory Staff is currently reviewing the treatment of uncertainties in the establishment and monitoring of operating limits for boiling water reactors. The Committee wishes to be kept informed.

Re-evaluation of core operating limits will be necessary as a result of the recently promulgated Acceptance Criteria for Emergency Core Cooling Systems. The Committee wishes to be kept informed.

The Committee believes that, with due regard to the above comments, including the individual reload reviews by the Regulatory Staff which will address specific plant features, the General Electric 8x8 fuel assemblies are acceptable for use in the reload of General Electric boiling water reactors.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

References attached.

References

1. "Dresden 3 Nuclear Power Station, Second Reload License Submittal", General Electric Co., Nuclear Fuel Department, September 1973, and Supplement A, November 17, 1973, Supplement B, December 6, 1973; Supplement C, December 6, 1973; Supplement D, December 17, 1973; Supplement E, December 17, 1973; Supplement F, January 9, 1974; Supplement G, January 9, 1974; Supplement H, January 23, 1974.

2. "Nine Mile Point Unit 1 - Second Refueling", P. D. Raymond to A. Giambusso, September 14, 1973

"Nine Mile Point Unit 1 Safety Analysis for Type 5 and Type 6 Reload Fuel", Niagara Mohawk Power Corporation, October 15, 1973

"Nine Mile Point Unit 1, Part 1, Non-Proprietary Response and Part 2, Proprietary Response," January 15, 1974

3. "Monticello Nuclear Generating Plant, Permanent Plant Changes to Accommodate Equilibrium Core Scram Reactivity Insertion Characteristics", January 23, 1974

4. NEDO-20103, "General Design Information for General Electric Boiling Water Reactor Reload Fuel Commencing in Spring, '74", September 1973

5. H. E. Williamson and D. C. Ditmore, "Experience with BWR Fuel Through September 1971", NEDO-10505, May 1972

6. GEAP-4059, "Vibration of Fuel Rods in Parallel Flow", E. P. Quinn, July 1962

7. Letter, J. A. Hinds to V. Moore, February 4, 1974, "Proprietary Information, 8x8 Fuel Wear Tests"

Letter, J. A. Hinds to V. Moore, February 4, 1974, 8x8 Fuel Surveillance Program

8. NEDM-10735, "Densification Considerations in BWR Fuel Design and Performance", D. C. Ditmore and R. B. Elkins, December 1972, Supplement 2, "Response to AEC Questions, NEDM-10735 April 1973 (Proprietary), Supplement 2, "Response to AEC Questions, NEDM-10735 Supplement 1", May 1973 (Proprietary), Supplement 3, "Response to AEC Questions, NEDM-10735, Supplement 1, June 1973 (Proprietary), Supplement 4, "Responses to AEC Questions NEDM-10735", July 1973, (Proprietary), Supplement 5, "Densification Considerations in BWR Fuel", July 1973 (Proprietary), Supplements 6, 7, and 8, "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel".

References (cont'd)

9. NEDO-20181, Supplement 1, December 3, 1973 (Proprietary); GEGAP-III, "A Model for the Prediction of Pellet Conductance in BWR Fuel Rods"

10. "Technical Report on Densification of General Electric Reactor Fuels", U. S. Atomic Energy Commission, August 23, 1973, and "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels", December 14, 1973

11. "Sensitivity Study on BWR/6 Fuel Bundle Response to a Postulated LOCA", C. M. Moser and R. W. Griebe, December 1973

12. NEDE-10801, "Modeling the BWR/6 Loss-of-Coolant Accident" Core Spray and Bottom Flooding Heat Transfer Effectiveness", J. D. Duncan and J. E. Leonard, March 1973, and "Response to AEC Request for Additional Information on NEDE-10801", May 1973 (Proprietary)

13. NEDO-10993, "Core Spray and Bottom Flooding Effectiveness in the BWR/6", J. D. Duncan and J. E. Leonard, September 1973

14. "Core Thermal Analyses of a Stainless Steel Clad Heater Rod Bundle", C. M. Moser and R. W. Griebe, December 1973

15. "Status Report on the General Electric Company 8x8 Fuel Assembly", January 18, 1974, USAEC Regulatory Staff

16. "Technical Report on the General Electric Company 8x8 Fuel Assembly", February 5, 1974, USAEC Regulatory Staff

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

February 13, 1974

L. M. Muntzing
Director of Regulation

REVIEW OF PROPOSED RELOAD FUEL FOR NUCLEAR REACTORS

The recent review of the proposed GE 8x8 fuel element for use as a reload fuel in several boiling water reactors has reemphasized the need for the development of criteria and appropriate procedures for review of proposed reload cores in nuclear plants.

In view of the anticipated increase in the number and variety of reload cores that will be required in the near future, and the number of vendors who will be involved, the Committee recommends that this effort be augmented as required to provide appropriate procedures in timely fashion.

W. R. Stratton

W. R. Stratton
Chairman

cc: J. F. O'Leary, L
P. C. Bender, SECY
A. Giambusso, L
J. M. Hendrie, L

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

December 18, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS

Dear Dr. Schlesinger:

The Advisory Committee on Reactor Safeguards and the Regulatory Staff have identified a number of safety problems during recent years that are common to a specific type of light-water reactor or to all light-water reactors (LWRs). The "generic items" discussed herein have been cited in Committee reports pertaining to the construction or operation of LWRs; additional generic items not cited in the report have been identified by the Directorate of Licensing. The ACRS customarily has used a general paragraph to cover those generic items noted in its previous reports. This paragraph reads: "Other problems relating to large water reactors, which have been identified by the Regulatory Staff and the ACRS and cited in previous reports, should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed". The use of such a paragraph has permitted the ACRS to emphasize either new or particularly significant generic problems without listing in detail those problems cited in the past.

The Committee believes that many of the generic items cited in its reports have been resolved by actions of the applicants or decisions of the Regulatory Staff, in cooperation with the ACRS. Resolution of the remaining items are pending.

The Committee defines "resolved" to mean that a specific conclusion or policy decision has been reached by the Directorate of Licensing and the ACRS. Resolution of an item indicates that the Committee is satisfied in a generic sense; however, this does not mean that improvements should not be investigated and,

possibly, implemented. In fact, requirements may differ for specific plants because of such factors as site characteristics and construction authorization dates, especially as they apply to backfit requirements.

Group I of the attachment includes generic problems that have been resolved together with the specific action that resulted in the resolution. Group II includes those items for which resolution on a generic basis is still pending. The ACRS and the Regulatory Staff will continue to consider Group II items and their significance to safety on a case-by-case basis until generic resolution is reached. Formal actions, such as issuance of Regulations or Safety Guides, are anticipated for many of the Group II items.

The ACRS expects to report to you from time to time on the status of generic items.

Sincerely yours,

/s/

C. P. Siess
Chairman

Attachment:
Generic Items

GENERIC ITEMS

Group I - Resolved Generic Items

1. Net Positive Suction Head for ECCS Pumps:
   Covered by Safety Guide 1

2. Emergency Power:
   Covered by Safety Guides 6 and 9 and portions of IEEE-308 (1971)

3. Hydrogen Control After a Loss-of-Coolant Accident (LOCA):
   Covered by Safety Guide 7 and Supplement

4. Instrument Lines Penetrating Containment:
   Covered by Safety Guide 11 and Supplement

5. Strong Motion Seismic Instrumentation:
   Covered by Safety Guide 12

6. Fuel Pool Design Bases:
   Covered by Safety Guide 13

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles:
   Covered by Safety Guide 14

8. Protection Against Industrial Sabotage:
   Covered by Safety Guide 17

9. Vibration Monitoring of Reactor Internals and Primary System:
   Covered by Safety Guide 20

10. Inservice Inspection of Reactor Coolant Pressure Boundary:
    Covered by ASME Boiler and Pressure Vessel (BPV) Code, Section XI

11. Quality Assurance During Design, Construction, and Operation:
    Covered by 10 CFR 50, Appendix B; ASME BPV Code, Section III; ANSI N-45.2-1971; Safety Guide 28; and Proposed Standard ANS-3.2

12. Inspection of BWR Steam Lines Beyond Isolation Valves:
    Covered by ASME BPV Code, Section XI

13. Independent Check of Primary Sytem Stress Analysis:
    Covered by ASME BPV Code, Section III

14. Operational Stability of Jet Pumps:
    Tests and operating experience at Dresden 2 and 3 and other jet pump BWRs have satisfied the ACRS concerns for this generation plant.

15. Pressure Vessel Surveillance of Fluence and NDT Shift:
Covered by 10 CFR 50, Appendix A and proposed Appendix H; and ASTM Standard E-185-70

16. Nil Ductility Properties of Pressure Vessel Materials:
Covered by 10 CFR 50, Appendix A and proposed Appendix G; and ASME BPV Code, Section III

17. Operation of Reactor With Less Than All Loops in Service:
Covered by ACRS-Regulatory Staff position that manual resetting of several set points on the control room instruments under specific conditions and procedures is acceptable in taking one primary loop out of service. This position is based on the expectation that this mode of operation will be infrequent.

18. Criteria for Preoperational Testing:
Covered by Operational Guide for the Planning of Initial Startup Programs, December 7, 1970

19. Diesel Fuel Capacity:
Covered by ACRS-Regulatory Staff position requiring 7 days fuel

20. Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends:
Covered by ACRS-Regulatory Staff position cited in several letters that such a failure should have no unacceptable consequences. ACRS-Regulatory Staff position document will be prepared.

21. Operating One Plant While Other(s) is/are Under Construction:
Specific requirements have been established by ACRS-Regulatory Staff. Position will be prepared.

22. Seismic Design of Steam Lines:
Covered by Safety Guide 29

23. Quality Group Classifications for Pressure Retaining Components:
Covered by Safety Guide 26

24. Ultimate Heat Sink:
Covered by Safety Guide 27

25. Instrumentation to Detect Stresses in Containment Walls:
Covered by Safety Guide 18

Group II - Resolution Pending

1. Adequacy of Primary System Leak Detection and Location:

   ACRS-Regulatory Staff position requires systems in addition to sump measurements.

2. Positive Moderator Coefficient:

   One solution is use of fixed burnable poison in core.

3. Use of Sensitized Stainless Steel:

   ACRS-Regulatory Staff position is to minimize use.*

4. Protection Against Pipe Whip:

   Ultimate decision will depend on review of available information on pipe break probabilities.*

5. Turbine Missiles:

   Turbine failures for past 16 years have been evaluated and a statistical probability analysis has been completed.

6. Fixed In-Core Detectors on High Power PWRs:

   Some information is available.

7. Performance of Critical Components (pumps, cables, etc.) in Post-LOCA Environment:

   Substantial information available in topical reports. Evaluation is required to determine if all necessary information is on hand.*

8. Effective Operation of Containment Sprays in a LOCA:

   Extensive documentation in topical reports. Review and evaluation are required.

9. Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containments:

   Analyses made in topical reports. Evaluation required by ACRS-Regulatory Staff.

---

* A Safety Guide is in preparation.

10. Anticipated Transients Without Scram:

    Data provided by applicants. Evaluation is required to permit a decision to cover BWRs and PWRs operating and under construction.

11. Radwaste Management:

    10 CFR 50, Appendix I, covers in part: Formal rulemaking decision required to implement fully.

12. Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock:

    Safety Guide 2 covers current information. Ultimate position as to significance of thermal shock requires input of fracture mechanics data on irradiated steels from the Heavy Section Steel Technology Program.

13. Instruments to Detect Fuel Failures:

    Instrumentation exists to detect fuel failures: ACRS-Regulatory Staff believes progress is satisfactory; however, continuing work is required.

14. Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel:

    State-of-the-Art results appear promising. More work may be required prior to decision as to installation of equipment.

15. Common Mode Failures:

    Requirements for diverse components should be established.

16. Emergency Core Cooling System Capability:

    Need for improvement cited by Regulatory Staff and stated in ACRS report of January 7, 1972. Further studies and evaluations are in progress.

17. Behavior of Reactor Fuel Under Abnormal Conditions:

    This includes: flow blockage; partial melting of fuel assemblies as it affects reactor safety; and transient effects on fuel integrity. The PBF program will address some of these items.

18. Emergency Power for Two or More Reactors at the Same Site:

    Additional work is required on protection systems for multiple units.

19. Main Steam Isolation Valve Leakage of BWRs:

    A definitive position is required of ACRS-Regulatory Staff in the light of continuing experience.

20. Instrumentation to Follow the Course of an Accident:

    Some equipment exists; further analyses are required to establish equipment requirements.

21. BWR Recirculation Pump Overspeed During LOCA:

    Topical reports prepared. Decision required by ACRS-Regulatory Staff.

22. The Advisability of Seismic Scram:

    Further studies required to establish need.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

February 13, 1974

Honorable Dixy Lee Ray
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS

Dear Dr. Ray:

The Advisory Committee on Reactor Safeguards reported on the "Status of Generic Items Relating to Light-Water Reactors" in its letter of December 18, 1972. The Committee expected to report from time to time on the status of generic items. This is the second such report.

Group I of the attachment is a reiteration of the generic items considered resolved in the first report. Group IA includes those items resolved since that report, together with the specific action that resulted in the resolution. Group II includes those items included in the original report for which resolution on a generic basis is still pending. Group IIA includes generic items that were not included in the first report. The ACRS and the Regulatory Staff will continue to consider Group II and IIA items and their significance to safety on a case-by-case basis until generic resolution is reached. Formal actions such as issuance of Regulations or Regulatory Guides are anticipated for many of the Group II and IIA items.

The Committee reaffirms its position that "resolved" means that a specific conclusion or policy decision has been reached by the Directorate of Licensing and the ACRS. Resolution of an item indicates that the Committee is satisfied in a generic sense; however, this does not mean that improvements should not be investigated and, possibly, implemented; or that experience has been satisfactory. In fact, requirements may differ for specific plants because of such factors as site characteristics and

construction authorization dates, especially as they apply to backfit requirements.

The ACRS will continue to report from time to time on the status of generic items.

Sincerely yours,

W. R. Stratton

W. R. Stratton
Chairman

Attachments:
Generic Items
1) Group I
2) Group IA
3) Group II
4) Group IIA

GENERIC ITEMS

Group I - Resolved Generic Items

1. Net Positive Suction Head for ECCS Pumps:
   Covered by Regulatory Guide 1.1

2. Emergency Power:
   Covered by Regulatory Guides 1.6 and 1.9 and portions of IEEE-308 (1971)

3. Hydrogen Control After a Loss-of-Coolant Accident (LOCA):
   Covered by Regulatory Guide 1.7 and Supplement

4. Instrument Lines Penetrating Containment:
   Covered by Regulatory Guide 1.11 and Supplement

5. Strong Motion Seismic Instrumentation:
   Covered by Regulatory Guide 1.12

6. Fuel Storage Pool Design Bases:
   Covered by Regulatory Guide 1.13

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles:
   Covered by Regulatory Guide 1.14

8. Protection Against Industrial Sabotage:
   Covered by Regulatory Guide 1.17

9. Vibration Monitoring of Reactor Internals and Primary System:
   Covered by Regulatory Guide 1.20

10. Inservice Inspection of Reactor Coolant Pressure Boundary:
    Covered by ASME Boiler and Pressure Vessel (BPV) Code, Section XI

11. Quality Assurance During Design, Construction, and Operation:
    Covered by 10 CFR 50, Appendix B; ASME BPV Code, Section III; ANSI N-45.2-1971; Regulatory Guide 1.28, 1.33, 1.64; and Proposed Standard ANS-3.2

12. Inspection of BWR Steam Lines Beyond Isolation Valves:
    Covered by ASME BPV Code, Section XI

13. Independent Check of Primary System Stress Analysis:
    Covered by ASME BPV Code, Section III

14. Operational Stability of Jet Pumps:
Tests and operating experience at Dresden 2 and 3 and other jet pump BWRs have satisfied the ACRS concerns.

15. Pressure Vessel Surveillance of Fluence and NDT Shift:
Covered by 10 CFR 50, Appendix A and Appendix H; and ASTM Standard E-185-70

16. Nil Ductility Properties of Pressure Vessel Materials:
Covered by 10 CFR 50, Appendix A and Appendix G; ASME BPV Code, Section III; ACRS Pressure Vessel Report

17. Operation of Reactor With Less Than All Loops Inservice:
Covered by ACRS-Regulatory Staff position that manual resetting of several set points on the control room instruments under specific conditions and procedures is acceptable in taking one primary loop out of service. This position is based on the expectation that this mode of operation will be infrequent.

18. Criteria for Preoperational Testing:
Covered by Regulatory Guide 1.68

19. Diesel Fuel Capacity:
Covered by ACRS-Regulatory Staff position requiring 7 days fuel*

20. Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends:
Covered by ACRS-Regulatory Staff position cited in several letters that such a failure should have no unacceptable consequences.

21. Operating One Plant While Other(s) is/are Under Construction:
Specific requirements have been established by ACRS-Regulatory Staff. Position will be prepared.

22. Seismic Design of Steam Lines:
Covered by Regulatory Guide 1.29

23. Quality Group Classifications for Pressure Retaining Components:
Covered by Regulatory Guide 1.26

24. Ultimate Heat Sink:
Covered by Regulatory Guide 1.27

25. Instrumentation to Detect Stresses in Containment Walls:
Covered by Regulatory Guide 1.18

*Regulatory Guide in preparation

Group IA - Generic Items Resolved Since December 18, 1972

1. Use of Sensitized Stainless Steel:
   Covered by Regulatory Guide 1.44

2. Primary system detection and location of leaks:
   Covered by Regulatory Guide 1.45

3. Protection Against Pipe Whip:
   Covered by Regulatory Guide 1.46

4. Anticipated Transients Without Scram:
   Covered by Regulatory Position Document, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors", WASH-1270, September, 1973.

5. ECCS capability of current and older plants:
   Covered by Rulemaking as a general policy decision, although acceptable detailed implementation remains to be developed. Docket RM-50-1, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors", December 28, 1973.

Group II - Resolution Pending

1. Positive Moderator Coefficient:
   One solution is use of fixed burnable poison in core. Reviews underway.

2. Turbine Missiles:
   Turbine failures for past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discusses the problem.*

3. Fixed In-Core Detectors on High Power PWRs:
   Some information is available. Further information is required.

4. Performance of Critical Components (pumps, cables, etc.) in Post-LOCA Environment:
   Substantial information available in topical reports. Evaluation is required to determine if all necessary information is on hand.*

5. Effective Operation of Containment Sprays in a LOCA:
   Extensive documentation in topical reports. Review and evaluation are required.

6. Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containments:
   Analyses made in topical reports. Evaluation required by ACRS-Regulatory Staff.

7. Radwaste Management:
   10 CFR Part 50, Appendix I covers in part.

8. Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock:
   Regulatory Guide 1.2 covers current information. Ultimate position as to significance of thermal shock requires input of fracture mechanics data on irradiated steels from the Heavy Section Steel Technology Program.

9. Instruments to Detect Fuel Failures:
   Instrumentation exists to detect fuel failures: ACRS-Regulatory Staff believes progress is satisfactory; however, continuing work is required.

10. Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel:
    State-of-the-Art results appear promising. More work may be required prior to decision as to installation of equipment.

*A Regulatory Guide is in preparation.

11. Common Mode Failures:
Requirements for diverse components should be established.

12. Emergency Core Cooling System Capability for future plants:
Need for improvement cited in ACRS report of January 7, 1972. Additional cooling capacity should be incorporated into future plant designs.

13. Behavior of Reactor Fuel Under Adnormal Conditions:
This includes: flow blockage; partial melting of fuel assemblies as it affects reactor safety; and transient effects on fuel integrity. The PBF program will address some of these items.

14. Emergency Power for Two or More Reactors at the Same Site:
Additional work is required on protection systems for multiple units.*

15. Main Steam Isolation Valve Leakage of BWRs:
A definitive position is required of ACRS-Regulatory Staff in the light of continuing experience. Note ACRS letters.*

16. Instrumentation to Follow the Course of an Accident:
Some equipment exists; further analyses are required to establish equipment requirements.*

17. BWR Recirculation Pump Overspeed During LOCA:
Topical reports prepared. Decision required by ACRS-Regulatory Staff.

18. The Advisability of Seismic Scram:
Further studies required to establish need.

*A Regulatory Guide is in preparation.

Group IIA - Resolution Pending - Items Added Since December 18, 1972

1. Pressure in containment following LOCA:
   Further criteria and methods are needed to better evaluate local dynamic pressures in a LOCA to establish more definitive design margins.

2. Control Rod Drop/Ejection Accident:
   Calculations indicate that the reactivity response differs from earlier values. New analyses are required. Regulatory Guides are being prepared for BWRs and PWRs.*

3. Fuel Densification:
   This is a facet of Item 13 in Group II, "Behavior of Reactor Fuel Under Abnormal Conditions," and Item 5 in Group IA, "ECCS Capability of Current and Older Plants."

4. Ice Condenser Containments:
   Additional analyses are required to establish response during a LOCA, and to establish design margins.

5. Rupture of High Pressure Lines Outside Containment:
   The possibility exists that failure of a high pressure line such as a steam pipe can prevent operation of critical safety components.*

6. PWR Pump Overspeed During a LOCA:
   Problem arises in similar manner to that of BWRs (Item 17 Group II).

7. Rod Sequence Control Systems:
   Further Evaluation Required to Establish conservatism in both BWRs and PWRs.

8. Isolation of Low Pressure From High Pressure Systems:
   Assurance Required that Low Pressure Systems cannot inadvertently be interconnected with a high pressure system leading to failure. There are potential interaction problems between Class 1 and Class 2 or Class 3 pressure connections.*

9. Steam Generator Tube Leakage:
   Concern primarily related to steam binding due to failure of a critical number or leakage of partially failed tubes during LOCA.*

10. Periodic Review of Older Reactors:
    To assure the adequacy of safety margins on a continuing basis.

11. Effluents from Light-Water Cooled Nuclear Power Reactors:
    To be covered by Formal Rulemaking. Acceptable detailed implementation remains to be developed. This is a facet of Item 7 in Group II, on "Radwaste Management".

*A Regulatory Guide is in preparation.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 12, 1975

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS

Dear Mr. Anders:

The Advisory Committee on Reactor Safeguards reported on the "Status of Generic Items Relating to Light-Water Reactors" in its letters of December 18, 1972 and February 13, 1974. The Committee expects to report from time to time on the status of generic items. This is the third such report. The Committee limits its definitions of generic items to those cited specifically in its letters pertaining to projects and related matters. It should not be considered an all-inclusive listing since the Nuclear Regulatory Commission Staff may have additional generic items.

Group I of the attachment is a reiteration of the generic items considered resolved in the first report. Group IA includes those items resolved between December 1972 and February 1974 while Group IB includes those items resolved since February 1974. Following each item is a brief statement of the specific action that resulted in the resolution. Group II lists those items included in the original report for which resolution on a generic basis is still pending. Group IIA includes generic items that were added in the second report. The ACRS and the NRC Staff will continue to consider the safety significance of Group II, IIA and IIB items on a case-by-case basis until generic resolution is reached. Formal actions such as issuance of Regulations or Regulatory Guides are anticipated for many of the Group II, IIA and IIB items.

The Committee reaffirms its position that "resolved" means that a specific conclusion or policy decision has been reached by the NRC Staff and the ACRS. Resolution of an item indicates that the Committee is satisfied in a generic sense; however, this does not

mean that improvements should not be investigated and, possibly, implemented, or that experience has been satisfactory. In fact, requirements may differ for specific plants because of such factors as site characteristics and construction authorization dates, especially as they apply to backfit requirements.

Sincerely,

W. Kerr
Chairman

Attachments:
Generic Items
1) Group I
2) Group IA
3) Group IB
4) Group II
5) Group IIA
6) Group IIB

GENERIC ITEMS

Group I - Resolved Generic Items

1. Net Positive Suction Head for ECCS Pumps: Covered by Regulatory Guide 1.1

2. Emergency Power: Covered by Regulatory Guides 1.6, 1.9, and 1.32 and portions of IEEE-308 (1971)

3. Hydrogen Control After a Loss-of-Coolant Accident (LOCA): Covered by Regulatory Guide 1.7 and Supplement

4. Instrument Lines Penetrating Containment: Covered by Regulatory Guide 1.11 and Supplement

5. Strong Motion Seismic Instrumentation: Covered by Regulatory Guide 1.12

6. Fuel Storage Pool Design Bases: Covered by Regulatory Guide 1.13

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles: Covered by Regulatory Guide 1.14

8. Protection Against Industrial Sabotage: Covered by Regulatory Guide 1.17

9. Vibration Monitoring of Reactor Internals and Primary System: Covered by Regulatory Guide 1.20

10. Inservice Inspection of Reactor Coolant Pressure Boundary: Covered by ASME Boiler and Pressure Vessel (BPV) Code, Section XI and Regulatory Guide 1.65

11. Quality Assurance During Design, Construction and Operation: Covered by 10 CFR 50, Appendix B; ASME BPV Code, Section III; ANSI N-45.2-1971, Regulatory Guides 1.28, 1.33, 1.64, 1.70.6 and Proposed Standard ANS-3.2

12. Inspection of BWR Steam Lines Beyond Isolation Valves; Covered by ASME BPV Code, Section XI

13. Independent Check of Primary System Stress Analysis: Covered by ASME BPV Code, Section III

14. Operational Stability of Jet Pumps: Test and operating experience at Dresden 2 and 3 and other jet pump BWRs have satisfied the ACRS concerns

15. Pressure Vessel Surveillance of Fluence and NDT Shift: Covered by 10 CFR 50, Appendix A and Appendix H; and ASTM Standard E-185

16. Nil Ductility Properties of Pressure Vessel Materials: Covered by 10 CFR 50, Appendix A and Appendix G; ASME BPV Code, Section III; ACRS Pressure Vessel Report

17. Operation of Reactor With Less Than All Loops In Service: Covered by ACRS-Regulatory Staff position that manual resetting of several set points on the control room instruments under specific conditions and procedures is acceptable in taking one primary loop out of service. This position is based on the expectation that this mode of operation will be infrequent.

18. Criteria for Preoperational Testing: Covered by Regulatory Guide 1.68

19. Diesel Fuel Capacity: Covered by ACRS-Regulatory Staff position requiring 7 days fuel

20. Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends: Covered by ACRS-Regulatory Staff position cited in several letters that such a failure should have no unacceptable consequences.

21. Operating One Plant While Other(s) is/are Under Construction: Specific requirements have been established by ACRS-Regulatory Staff. Position will be prepared.

22. Seismic Design of Steam Lines: Covered by Regulatory Guide 1.29

23. Quality Group Classifications for Pressure Retaining Components: Covered by Regulatory Guide 1.26

24. Ultimate Heat Sink: Covered by Regulatory Guide 1.27

25. Instrumentation to Detect Stresses in Containment Walls: Covered by Regulatory Guide 1.18

Group IA - Generic Items Resolved Since December 18, 1972

1. Use of Furnace Sensitized Stainless Steel: Covered by Regulatory Guide 1.44

2. Primary system detection and location of leaks: Covered by Regulatory Guide 1.45

3. Protection Against Pipe Whip: Covered by Regulatory Guide 1.46

4. Anticipated Transients Without Scram: Covered by Regulatory Position Document, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, September 1973

5. ECCS Capability of Current and Older Plants: Covered by Rulemaking as a general policy decision, although acceptable detailed implementation remains to be developed. Docket RM-50-1, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors," December 28, 1973

Group IB - Generic Items Resolved Since February 13, 1974

1. Positive Moderator Coefficient: PWRs presently have or expect to have zero or negative coefficients. Where some Technical Specifications allow a slightly positive coefficient, the accident and stability analyses take this into account. Burnable poison provisions have been designed into PWRs to reduce otherwise excessive positive coefficients to allowable values.

2. Fixed Incore Detectors on High Power PWRs: Fixed incore detectors are not required for PWRs since reviews of potential power distribution anomalies have not revealed a clear need for continuous incore monitoring.

3. Performance of Critical Components (pumps, cables, etc.) in post-LOCA Environment: Qualification requirements of critical components are now covered by Regulatory Guides 1.40, 1.63, 1.73 and 1.89 and IEEE Standards 382-1972, 383-1974, 317-1972, 323-1974.

4. Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containments: On designs prior to GE Mark III containment, resolution lies in surveillance and testing of vacuum relief valves. For Mark III containments, an additional requirement is that the design be capable of accommodating a bypass equivalent to one square foot for a given flow condition.

5. Emergency Power for Two or More Reactors at the Same Site: Resolved by issue of Regulatory Guide 1.81.

6. Effluents from Light-Water-Cooled-Nuclear Power Reactors: Resolved by issue of Appendix I to 10 CFR 50.

7. Control Rod Ejection Accident: Resolved for PWRs by Regulatory Guide 1.77.

GROUP II - Resolution Pending

1. Turbine Missiles: Turbine failures for past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discusses the problem.*

2. Effective Operation of Containment Sprays in a LOCA: Extensive documentation in topical reports. Review and evaluation are required.

3. Possible Failure of Pressure Vessel Post-LOCA By Thermal Shock: Regulatory Guide 1.2 covers current information. Ultimate position as to significance of thermal shock requires input of fracture mechanics data on irradiated steels from the Heavy Section Steel Technology Program.

4. Instruments to Detect Fuel Failures: Instrumentation exists to detect fuel failures. Continuing work is required.

5. Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel: State-of-the-Art results appear promising. More work may be required prior to decision as to installation of equipment.

6. Common Mode Failures: Requirements for diverse components should be established.

7. Behavior of Reactor Fuel Under Abnormal Conditions: This includes: flow blockage; partial melting of fuel assemblies as it affects reactor safety; and transient effects on fuel integrity. The PBF program will address some of these items.

*A Regulatory Guide is in preparation.

Group II Continued

8. BWR Recirculation Pump Overspeed During LOCA: Decision required by ACRS-Regulatory Staff.

9. The Advisability of Seismic Scram: Further studies required to establish need.

10. Emergency Core Cooling System Capability for Future Plants: Partially resolved by amendments to 10 CFR 50 [50.34(a)(4), 50.34(b)(4), 50.46, and Appendix K]. LOCA evaluation model complete. ACRS feels new cooling approaches should be explored.

11. Main Steam Isolation Valve Leakage of BWRs: Pending Regulatory Guide should resolve this issue.

12. Instrumentation to Follow the Course of an Accident: A Regulatory Guide to be issued should resolve the issue.

GROUP IIA - Resolution Pending - Items Added Since 12/18/72

1. Pressure in Containment Following LOCA: Further criteria and methods are needed to better evaluate local dynamic pressures in a LOCA to establish more definitive design margins.

2. Control Rod Drop Accident (BWRs): Calculations indicate that the reactivity response differs from earlier values. New analyses are required, including three-dimensional effects.

3. Fuel Densification: This is a facet of Item 7 in Group II, "Behavior of Reactor Fuel Under Abnormal Conditions," and Item 5 in Group IA, "ECCS Capability of Current and Older Plants."

4. Ice Condenser Containments: Additional analyses are required to establish response during a LOCA, and to establish design margins.

5. Rupture of High Pressure Lines Outside Containment: The possibility exists that failure of a high pressure line such as a steam pipe can prevent operation of critical safety components.

6. PWR Pump Overspeed During a LOCA: Problem arises in similar manner to that of BWRs (Item 8 Group II).

7. Rod Sequence Control Systems: Further evaluation required to establish conservatism in both BWRs and PWRs. Resolved for BWR-4 Design.

8. Isolation of Low Pressure From High Pressure Systems: Assurance required that low pressure systems cannot inadvertently be interconnected with a high pressure system leading to failure. There are potential interaction problems between Class 1 and Class 2 or Class 3 pressure connections.

9. Steam Generator Tube Leakage: Partially resolved by issue of Regulatory Guide 1.83 which addresses the concern from a preventative point of view.

10. ACRS/NRC Periodic 10-Year Review of All Power Reactors: A more effective, continuous alternative approach to periodic reviews is being proposed. Pending ACRS review, this item is still considered unresolved.

GROUP IIB - Resolution Pending - Items Added Since 2/13/74

1. Hybrid Reactor Protection System: Systems should be qualified for reliability, particularly through in situ tests and under various environmental conditions, prior to use in reactor system.

2. Qualification of new fuel geometries: The 16x16, 17x17 PWR and 8x8 BWR fuels should undergo testing to meet Item 7 in Group II and Item 3 in Group IIA.

3. Behavior of BWR Mark III Containments: Various aspects, including vent clearing, vent coolant interaction, pool swell, pool stratification, pressure loads and flow bypass should be resolved. This is an extension of Item 1 in Group IIA.

4. Seismic Category I Requirements for Auxiliary Systems: Systems such as radioactive offgas, letdown cooling and core auxiliary cooling should be examined to determine if it is necessary to design to Seismic Category I.

5. Stress Corrosion Cracking in BWR Piping: Several failures have occurred in operating BWRs. The ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution and extensive programs are underway by industry, ERDA, and NRC.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 9, 1976

Honorable William A. Anders
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Anders:

The ACRS has received your letter of January 13, 1976, which provides guidance to the Committee concerning the four generic issues on which ACRS review and comment have been requested by the Commission. The Committee is pleased to participate in these activities and will make every effort to provide its comments in accordance with the proposed schedule.

As you know, evaluation of the NRC regulatory and licensing programs in terms of the proposed national effort for long-term nuclear waste management will depend on the availability of information from several other agencies, particularly ERDA. If complete information is not available prior to the July 1976 suggested deadline, the ACRS will provide its comments on this subject in the form of an interim report.

It is possible that timely review by the Committee of these issues may require some delay in the review of several project applications tentatively scheduled for review during 1976. If this should be the case, the need for schedule changes will be discussed with the NRC staff in order to minimize the effects on project schedules to the extent practical.

The increased workload of generic reviews also makes it imperative to maintain the ACRS membership at or near its statutory level of 15 and to maintain the ACRS staff at the level necessary to provide required technical and management support. The Committee is giving urgent attention to its needs in light of the existing vacancy and will provide recommendations to you in the near future. In addition, the Committee urgently requests that four additional staff positions (3 professional, 1 secretarial) be authorized for the ACRS office to provide the increased level of staff support needed by the Committee to fulfill these obligations.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 16, 1976

Honorable Marcus A. Rowden
Acting Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS: REPORT NO. 4

Dear Mr. Rowden:

The Advisory Committee on Reactor Safeguards reported on the "Status of Generic Items Relating to Light-Water Reactors" in its letters of December 18, 1972, February 13, 1974, and March 12, 1975. This is the fourth such report. Since the Committee limits its definitions of generic items to those cited specifically in its letters pertaining to projects and related matters, the attached listing is not all-inclusive; the Nuclear Regulatory Commission Staff list has additional generic items.

Group I of the attachment is a reiteration of the generic items considered resolved at the time the Committee issued its first report. Group IA includes those items resolved between December 1972 and February 1974; Group IB includes those items resolved between February 1974 and March 1975; Group IC includes those items resolved since March 1975. Following each resolved item is a brief statement of the specific action that resulted in the resolution. Group II lists those items included in the original report for which resolution on a generic basis is still pending. Groups IIA and IIB include generic items that were added in the second and third reports; Group IIC includes those added in the present report. The ACRS and the NRC Staff will continue to consider the safety significance of Group II, IIA, IIB and IIC items on a case-by-case basis until generic resolution is reached. Formal actions such as issuance of Regulations or Regulatory Guides are anticipated for many of the Group II, IIA, IIB and IIC items.

The NRC Staff considers several of the items on the attached unresolved lists to be "resolved", based on specific positions taken by them; however, these "resolutions" do not meet the ACRS criteria for adequate documentation and for Committee approval so they are considered to be pending until formally discussed and accepted by the ACRS. Items in the above category are:

(1) II-2 "Effective Operation of Containment Sprays in a LOCA";

(2) II-4 "Instruments to Detect Fuel Failures";

(3) II-6 "Common Mode Failures";

(4) II-9 "The Advisability of Seismic Scram";

(5) IIA-1 "Pressure in Containment Following LOCA";

(6) IIA-2 "Control Rod Drop Accidents (BWR)";

(7) IIA-5 "Rupture of High Pressure Lines Outside Containment";

(8) IIA-8 "Isolation of Low Pressure from High Pressure Systems";

(9) IIB-3 "Behavior of BWR Mark III Containments."

Owing to questions raised concerning the scope and intent of various generic issues, the Committee has incorporated into the attachments a brief description for all items unresolved now or at the time of the third report.

"Resolved" as used in the Generic Items reports refers to the following: In some cases an item has been resolved in an administrative sense, recognizing that technical evaluation and satisfactory implementation are yet to be completed. Anticipated Transients Without Scram represents an example of this category. In other instances, the resolution has been accomplished in a narrow or specific sense, recognizing that further steps

are desirable as practical or that different aspects of the problem require further investigation. Examples are the possibility of improved methods of location of leaks in the primary system, and of improved methods or augmented scope to inservice inspection of reactor pressure vessels.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

Attachments:
1) Group I
2) Group IA
3) Group IB
4) Group IC
5) Group II
6) Group IIA
7) Group IIB
8) Group IIC

GENERIC ITEMS

Group I - Resolved Generic Items

1. Net Positive Suction Head for ECCS Pumps: Covered by Regulatory Guide 1.1.

2. Emergency Power: Covered by Regulatory Guides 1.6, 1.9, and 1.32 and portions of IEEE-308 (1971).

3. Hydrogen Control After a Loss-of-Coolant Accident (LOCA): ACRS concurred in proposed Staff position, covered by NRC Standard Review Plan for Nuclear Power Plants.

4. Instrument Lines Penetrating Containment: Covered by Regulatory Guide 1.11 and Supplement.

5. Strong Motion Seismic Instrumentation: Covered by Regulatory Guide 1.12.

6. Fuel Storage Pool Design Bases: Covered by Regulatory Guide 1.13.

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles: Covered by Regulatory Guide 1.14.

8. Protection Against Industrial Sabotage: Covered by Regulatory Guide 1.17.

9. Vibration Monitoring of Reactor Internals and Primary System: Covered by Regulatory Guide 1.20.

10. Inservice Inspection of Reactor Coolant Pressure Boundary: Covered by ASME Boiler and Pressure Vessel (BPV) Code, Section XI and Regulatory Guide 1.65.

11. Quality Assurance During Design, Construction and Operation: Covered by 10 CFR 50, Appendix B; ASME BPV Code, Section III; ANSI N-45.2-1971, Regulatory Guides 1.28, 1.33, 1.64, 1.70.6 and Proposed Standard ANS-3.2.

12. Inspection of BWR Steam Lines Beyond Isolation Valves: Covered by ASME BPV Code, Section XI.

13. Independent Check of Primary System Stress Analysis: Covered by ASME BPV Code, Section III.

14. Operational Stability of Jet Pumps: Test and operating experience at Dresden 2 and 3 and other jet pump BWRs have satisfied the ACRS concerns.

Group I Continued

15. Pressure Vessel Surveillance of Fluence and NDT Shift: Covered by 10 CFR 50, Appendix A and Appendix H; and ASTM Standard E-185.

16. Nil Ductility Properties of Pressure Vessel Materials: Covered by 10 CFR 50, Appendix A and Appendix G; ASME BPV Code, Section III; ACRS Pressure Vessel Report.

17. Operation of Reactor With Less Than All Loops In Service: Covered by ACRS-Regulatory Staff position that manual resetting of several set points on the control room instruments under specific conditions and procedures is acceptable in taking one primary loop out of service. This position is based on the expectation that this mode of operation will be infrequent.

18. Criteria for Preoperational Testing: Covered by Regulatory Guide 1.68.

19. Diesel Fuel Capacity: Covered by ACRS-Regulatory Staff position requiring 7 days fuel.

20. Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends: Covered by ACRS-Regulatory Staff position cited in several letters that such a failure should have no unacceptable consequences.

21. Operating One Plant While Other(s) is/are Under Construction: Specific requirements have been established by ACRS-Regulatory Staff. Position will be prepared.

22. Seismic Design of Steam Lines: Covered by Regulatory Guide 1.29.

23. Quality Group Classifications for Pressure Retaining Components: Covered by Regulatory Guide 1.26.

24. Ultimate Heat Sink: Covered by Regulatory Guide 1.27.

25. Instrumentation to Detect Stresses in Containment Walls: Covered by Regulatory Guide 1.18.

Group IA - Generic Items Resolved Since December 18, 1972

1. Use of Furnace Sensitized Stainless Steel: Covered by Regulatory Guide 1.44.

2. Primary System Detection and Location of Leaks: Covered by Regulatory Guide 1.45.

3. Protection Against Pipe Whip: Covered by Regulatory Guide 1.46.

4. Anticipated Transients Without Scram: Covered by Regulatory Position Document, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, September 1973.

5. ECCS Capability of Current and Older Plants: Covered by Rulemaking as a general policy decision, although acceptable detailed implementation remains to be developed. Docket RM-50-1, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors," December 28, 1973.

Group IB - Generic Items Resolved Since February 13, 1974

1. Positive Moderator Coefficient: PWRs presently have or expect to have zero or negative coefficients. Where some Technical Specifications allow a slightly positive coefficient, the accident and stability analyses take this into account. Burnable poison provisions have been designed into PWRs to reduce otherwise excessive positive coefficients to allowable values.

2. Fixed Incore Detectors on High Power PWRs: Fixed incore detectors are not required for PWRs since reviews of potential power distribution anomalies have not revealed a clear need for continuous incore monitoring.

3. Performance of Critical Components (pumps, cables, etc.) in post-LOCA Environment: Qualification requirements of critical components are now covered by Regulatory Guides 1.40, 1.63, 1.73 and 1.89 and IEEE Standards 382-1972, 383-1974, 317-1972, 323-1974.

4. Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containments: On designs prior to GE Mark III containment, resolution lies in surveillance and testing of vacuum relief valves. For Mark III containments, an additional requirement is that the design be capable of accommodating a bypass equivalent to one square foot for a given flow condition.

5. Emergency Power for Two or More Reactors at the Same Site: Resolved by issue of Regulatory Guide 1.81.

6. Effluents from Light-Water-Cooled-Nuclear Power Reactors: Resolved by issue of Appendix I to 10 CFR 50.

7. Control Rod Ejection Accident: Resolved for PWRs by Regulatory Guide 1.77.

Group IC - Generic Items Resolved Since March 12, 1975

1. Main Steam Isolation Valve Leakage of BWR's: Covered by Regulatory Guide 1.96.

2. Fuel Densification: Covered by 10 CFR 50 Appendix K plus case-by-case review of vendor fuel models.

3. Rod Sequence Control Systems: Covered by NRC Staff Review and Approval of NEDO-10527 and Presentation to ACRS.

4. Seismic Category I Requirements for Auxiliary Systems: Covered by Regulatory Guides 1.26 and 1.29.

IC-1 - MAIN STEAM ISOLATION VALVE LEAKAGE OF BWRs

The BWR main steam isolation valve (MSIV) leakage problem relates to a loss of coolant accident condition where radioactivity levels are postulated to exist in the coolant. The anticipated leakage levels in this class of valve may result in excessive releases of activity to the environment. Possible solutions include another valve in series to decrease leakage to acceptable levels, sealing mechanisms such as water seals to trap and retain radioactivity, or another type of MSIV capable of meeting the low levels of leakage. This issue has been resolved through issuance of Regulatory Guide 1.96.

IC-2 - FUEL DENSIFICATION

Fuel Densification is a facet of Behavior of Reactor Fuel Under Abnormal Conditions (II-7) and ECCS Capability of Current and Older Plants (IA-5). The densification of fuel changes fuel-cladding gaps, contained sensible heat, and fuel temperatures, leading to unpredictable fuel behavior throughout life. Newer fuel has been modified in seveal ways to minimize densification, particularly with regard to increasing initial densities of fuel pellets. Existing results indicate that these changes have eliminated densification as a problem. This issue is considered resolved on the joint bases of conformance to Appendix K of 10 CFR 50 and case-by-case reviews of vendor fuel models.

IC-3 - ROD SEQUENCE CONTROL SYSTEMS

Rod sequence control systems or rod pattern control systems are designed to prevent a control rod pattern to exist where a control rod accident could result in peak fuel enthalpies in excess of 280 calories/gram for the entire range of plant operations and core exposure. The problem is one of inadequate experience or analytic evaluation to confirm the design conservatism. Topical reports have been submitted and NRC reviews are complete, so the issue is considered resolved.

IC-4 - SEISMIC CATEGORY I REQUIREMENTS FOR AUXILIARY SYSTEMS

Various auxiliary systems provide continuous or intermittent functions insofar as control of radioactivity, safe plant operation including startup and shutdown, or essential safety actions under accident conditions. These systems are designed to various codes and standards and may or may not be seismic Category I. These auxiliary systems have been evaluated on the basis of such factors as potential release of retained radioactivity, disruption of operation, and failure to provide vital safety functions to determine what components, if any, need to be designed and constructed to meet seismic category requirements. These factors have been incorporated into Regulatory Guides 1.26 and 1.29.

Group II - Resolution Pending

1. Turbine Missiles: Turbine failures for past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discusses the problems.*

2. Effective Operation of Containment Sprays in a LOCA: Extensive documentation in topical reports. Review and evaluation are required.

3. Possible Failure of Pressure Vessel Post-LOCA By Thermal Shock: Regulatory Guide 1.2 covers current information. Ultimate position as to significance of thermal shock requires input of fracture mechanics data on irradiated steels from the Heavy Section Steel Technology Program.

4. Instruments to Detect Fuel Failures: Instrumentation exists to detect fuel failures. Continuing work is required..

5. Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel: State-of-the-Art results appear promising. More work may be required prior to decision as to installation of equipment.

6. Common Mode Failures: Requirements for diverse components should be established.

7. Behavior of Reactor Fuel Under Abnormal Conditions: This includes: flow blockage; partial melting of fuel assemblies as it affects reactor safety; and transient effects on fuel integrity. The PBF program will address some of these items.

8. BWR Recirculation Pump Overspeed During LOCA: Decision required by ACRS-Regulatory Staff.

9. The Advisability of Seismic Scram: Further studies required to establish need.

10. Emergency Core Cooling System Capability for Future Plants: Partially resolved by amendments to 10 CFR 50 [50.34(a)(4), 50.34(b)(4), 50.46, and Appendix K]. LOCA evaluation model complete. ACRS feels new cooling approaches should be explored.

11. Instrumentation to Follow the Course of an Accident: A Regulatory Guide to be issued should resolve the issue.

---

*Regulatory Guide is in preparation.

II-1 - TURBINE MISSILES

Turbine failures for the past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discuses the problem.

Three issues require answers to resolve the turbine missile problem: (1) The first relates to the appropriate failure probability value; based on historical failures the probability is about 10^{-4}. Industry predicts a much lower failure probability based on improvements in materials and design. To date the ACRS has accepted the more conservative value; (2) The second issue is strongly dependent on turbine orientation with respect to critical safety structures. Strike probabilities from high angle missiles are acceptably low for single units and may be acceptable for multi-unit plants, depending on plant layout; however, lower angle missiles with non-optimum (tangential) turbine orientation have unacceptably high strike probabilities; (3) The third issue is one of penetration and damage of structures housed in the containment. The limited experimental data pertaining to penetration of large irregularly shaped missiles are not sufficient to determine structural response to impingement of turbine disc segments. Most missile penetration formulas are not relevant to this case. Some experiments with irregular missiles might resolve this issue, particularly for older plants with non-optimum turbine orientations.

II-2 - EFFECTIVE OPERATION OF CONTAINMENT SPRAYS IN A LOCA

Review and evaluation are required of the variety of experiments which have been conducted on the effectiveness of various containment sprays on the removal and retention of airborne radioactive materials anticipated to be present within containment following a LOCA. Such review should consider adequacy or definition of the physical and chemical forms of the anticipated airborne radionuclides, and quality or evaluative tests of the removal efficiencies of various sprays under the conditions of temperature, pressure, and radiation doses expected to exist under LOCA conditions. A desirable extension might be benefit/risk analyses of the use of sprays containing chemicals (such as NaOH) which have the potential for damaging equipment within containment. Studies using other spray additives, such as hydrazine, have been conducted. If compounds, such as this, have distinct advantages, action should be taken to encourage their use.

II-3 - POSSIBLE FAILURE OF PRESSURE VESSEL POST-LOCA BY THERMAL SHOCK

Earlier nuclear reactor pressure vessels subjected to fluences of $1\text{-}4 \times 10^{19}$ nvt, which are anticipated in the last 20 years of a 40-year life, may suffer severe radiation damage denoted by pronounced shift in impact transition temperature at the inner surface. There will be a damage gradient which decreases sharply, so that the properties halfway through the wall are essentially those of the as-fabricated material. If a LOCA occurs near end-of-life, the injection of cold water on the region of degraded properties may initiate and propagate a crack because of high local stresses near the surface. Analytic procedures indicate the stresses drop rapidly with distance through the wall so the flaw should not propagate beyond some limiting point. The lack of experimental evidence and the relative width of the error band in the analytic results are such that some experiments are required to validate the analytic model. These are planned under the HSST program.

II-4 - INSTRUMENTS TO DETECT FUEL FAILURES

In the event of local fuel overheating that leads to clad failure and to melting of sections of one or more irradiated fuel elements, relatively large amounts of fission products could be rapidly released to the reactor coolant. Although the course of events subsequent to such fuel melting may not have serious safety significance, there has, as yet, been no experiemental research program to investigate such situations and there is little applicable reactor experience. Also, the subsequent course of events may depend, in part, on the initiating cause of fuel overheating. Early warning and timely response may avert an incident becoming an accident.

Instrumentation related to such diagnostic purposes is being used on most power reactors. A study to define the appropriate minimum requirements and judicious responses to various signals is needed.

II-5 - MONITORING FOR EXCESSIVE VIBRATION OR LOOSE PARTS INSIDE THE PRESSURE VESSEL

Loose parts monitoring can provide early warning of potential mechanical problems or failures within the pressure vessel and throughout the primary coolant circuit. Reactor vendors have developed monitoring systems; however, general requirements remain to be established.

II-6 - NON-RANDOM MULTIPLE FAILURES (FORMERLY "COMMON MODE FAILURE")

The term "common mode failures" has, in many instances come to mean multiple failures of identical components exposed to identical or nearly identical conditions or environments, and the use of diversity in components has been proposed or required to avoid such failures. The concern of the ACRS is better expressed by the term "non-random multiple failures", which is intended to include not only the type of "common mode failure" discussed above but other types of multiple failures for which the consequences and probabilities cannot be predicted by application of the single-failure criterion. Examples include the use of the same sensors or components for both control and protection systems (a resolved matter); sequential multiple failures due to a "domino effect", and simultaneous multiple failures due to a single fault. Since designs usually do not knowingly incorporate features susceptible to such failures, techniques and criteria need to be developed to detect and avoid them in all systems important to safety.

II-7 - BEHAVIOR OF REACTOR FUEL UNDER ABNORMAL CONDITIONS

The Behavior of Reactor Fuel under Abnormal Conditions is still considered unresolved due to the limited experimental data available. Partial melting of fuel assemblies due to flow blockage might lead to autocatalytic effects leading to more extensive fuel failure, pressure pulses, etc. Similar behavior might occur in the case of reactivity transients. The ACRS encourages analytic modeling but believes appropriate experimental data are necessary. It is anticipated that tests in the Power Burst Facility (PBF) should supply much of the required data.

II-8 - BWR PUMP OVERSPEED DURING A LOCA

It is possible for a BWR recirculation pump to overspeed if a large break occurs at the appropriate position in specific piping. Conservative estimates indicate substantial overspeed and possible failure of components with the generation of missiles. The problem is being approached analytically and experimentally with scaled pumps. The reliabiity of such protective measures as the use of decouplers between pump and motor is under study.

II-9 - THE ADVISABILITY OF SEISMIC SCRAM

The ACRS has recommended that studies be made of techniques for seismic scram and of the potential safety advantages and potential disadvantages of prompt reactor scram in the event of strong seismic motion, say more than one-half the safe shutdown earthquake. Various suitable techniques have been identified and exist, but thus far only limited studies have been reported on the pros and cons of seismic scram. The principal potential advantage identified arises from the greatly improved coolability of a core in the unlikely event of a seismically induced LOCA, should scram precede the LOCA by several seconds. A principal reason given in opposition to seismic scram relates to a stated interest in keeping power stations on the line to provide power offsite should a severe earthquake occur.

II-10 - ECCS CAPABILITY FOR FUTURE PLANTS

The ACRS has placed considerable emphasis on ECCS safety R&D so that the extent of the conservatism in the ECCS licensing requirements could be made more precise. With more experimental data a realistic and quantitative appraisal of ECCS systems would lead to valid judgments on the changes in licensing which could be put on a firm basis.

Parallel approaches that seek to improve the reliability of ECCS systems, to improve the monitoring of low power peaking, and to improve those fuel assembly designs which lower peaking factors are encouraged. Further, changes in plant design which improve the reflooding of the reactor core should be sought and evaluated.

R&D efforts on analysis of core blowdown and reflood should be increased and combined with the results of the standard problems and the associated experiments. Improved analytical methods would provide a basis for optimized ECCS.

II-11 - INSTRUMENTS TO FOLLOW THE COURSE OF AN ACCIDENT

Instrumentation for determining the nature and the course of potentially serious accidents, on a time scale that will permit appropriate emergency action, should be provided at power plants and appropriate calibration methods and calculated bases for interpreting instrument responses should be available. The diversity of the installed instrumentation should be adequate to assure that assessments can be made of accidents and appropriate action taken if they follow the predicted course. In addition, the range of the instrumentation should be adequate to cover various potential radioactivity releases, particularly the upper range limits so as to fully encompass the spectra of all possible accidents.

Group IIA - Resolution Pending - Items Since December 18, 1972

1. Pressure in Containment Following LOCA: Further criteria and methods are needed to better evaluate local dynamic pressures in a LOCA to establish more definitive design margins.

2. Control Rod Drop Accident (BWRs): Calculations indicate that the reactivity response differs from earlier values. New analyses are required, including three-dimensional effects.

3. Ice Condenser Containments: Additional analyses are required to establish response during a LOCA, and to establish design margins.

4. Rupture of High Pressure Lines Outside Containment: The possibility exists that failure of a high pressure line such as a steam pipe can prevent operation of critical safety components.

5. PWR Pump Overspeed During a LOCA: Problem arises in similar manner to that of BWRs (Item 8 Group II).

6. Isolation of Low Pressure From High Pressure Systems: Assurance required that low pressure systems cannot inadvertently be interconnected with a high pressure system leading to failure. There are potential interaction problems between Class 1 and Class 2 or Class 3 pressure connections.

7. Steam Generator Tube Leakage: Partially resolved by issue of Regulatory Guide 1.83 which addresses the concern from a preventative point of view.

8. ACRS/NRC Periodic 10-Year Review of all Power Reactors: A more effective, continuous alternative approach to periodic reviews is being proposed. Pending ACRS review, this item is still considered unresolved.

IIA-1 - PRESSURE IN CONTAINMENTS FOLLOWING LOCA

A potential problem is the overpressurization of subcompartments following a LOCA. A suitably conservative analytic pressure model should be developed to determine that adequate safety margins exist with regard to subcompartment loads.

IIA-2 - CONTROL ROD DROP ACCIDENT (BWRs)

Some uncertainties have arisen in previous calculations of this postulated accident, including the choice of negative reactivity insertion rate due to scram and the potential differences between a two dimensional and a three dimensional calculation. Particularly for the latter point, more precise theoretical comparisons may be required to resolve the matter although probabilistic considerationss may be relevant.

IIA-3 - ICE CONDENSER CONTAINMENTS

The ice condenser containments have substantially smaller volume on the assumption that the ice will condense the steam during a LOCA, thus preventing system overpressurization. The rate of condensation is critical in the initial stages of the blowdown and is influenced by interaction of vapor with the ice. If the current analyses prove that the condensation model is suitably conservative, the problem may be resolved.

IIA-4 - RUPTURE OF HIGH PRESSURE LINES OUTSIDE CONTAINMENT

The functional or structural integrity of components required for safe shutdown and the maintenance of cold shutdown may be endangered by a failure of high pressure piping at certain locations outside of containment. Specific design and single failure criteria need to be developed to resolve this problem.

IIA-5 - PWR PUMP OVERSPEED DURING A LOCA

It is possible for a PWR primary coolant pump to overspeed if a large break occurs at the appropriate position in specific piping. Conservative estimates indicate substantial overspeed and possible failure of components such as flywheels with the generation of missiles. The problem is being approached analytically and experimentally with scaled pumps. The reliability of such protective measures as electrical braking of the pump motor is under study.

IIA-6 - ISOLATION OF LOW PRESSURE FROM HIGH PRESSURE SYSTEMS

Assurance is required that low pressure systems cannot inadvertently be interconnected with a high pressure system leading to a failure of the low pressure systems due to overpressurization. Both Class 2 systems required for ECCS and Class 3 systems used for many auxiliary coolant systems provide vital functions during normal and accident conditions. Any conditions leading to interconnection could trigger a LOCA. It is recognized that these systems must be interconnected because of the functions they meet. Therefore, particular attention is required in valve reliability and reliability of valve actuating circuits to assure that these valves will not open while the primary system is at pressure.

IIA-7 - STEAM GENERATOR TUBE LEAKAGE

Normally the steam generator is not a critical component during a LOCA-ECCS. However, a special case exists where the steam generator tubes have been degraded due to corrosion, wastage, etc. If the shock loads imposed by the LOCA cause a critical number of tubes to fail, say by a double-ended (guillotine) break the inflow from the secondary side can cause choking of flow during ECCS, preventing adequate cooling of the core. The critical number of tubes is relatively small. A position such as one specifying a statistically significant level of nondestructive examination (NDE) might resolve this issue. The purpose of NDE would be to confirm that damage is not excessive on the assumption that damaged tubes may fail catastrophically.

IIA-8 - PERIODIC (10-YEAR) REVIEW OF ALL POWER REACTORS

In its report of June 14, 1966, the ACRS recommended that periodic comprehensive reviews be conducted of operating licensed power reactors by the NRC Staff. These reviews would be preceded by a comprehensive report by the operator which evaluated the past experience and the safety of future operation of the plant.

The NRC Staff has maintained a continuing review of the safety of operating plants. In particular, as generic matters of potential safety significance arise, the appropriate operating reactors are asked to assess the relevance of the matter to each particular reactor. This is a necessary but different aspect of the continuing surveillance and review of the safety of operating reactors than was envisaged by the ACRS in its recommendation of June 1966.

The Committee continues to believe both approaches are desirable and awaits the development of a program of periodic comprehensive reviews.

Group IIB - Resolution Pending - Items Added Since February 13, 1974

1. Hybrid Reactor Protection System: Systems should be qualified for reliability, particularly through in situ tests and under various environmental conditions, prior to use in reactor system.

2. Qualification of new fuel geometries: The 16x16, 17x17 PWR and 8x8 BWR fuels should undergo testing to meet Item 2 in Group IC and Item 7 in Group II.

3. Behavior of BWR Mark III Containments: Various aspects, including vent clearing, vent coolant interaction, pool swell, pool stratification, pressure loads and flow bypass should be resolved. This is an extension of Item 1 in Group IIA.

4. Stress Corrosion Cracking in BWR Piping: Several failures have occurred in operating BWRs. The ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution and extensive programs are underway by industry, ERDA, and NRC.

IIB-1 - HYBRID REACTOR PROTECTION SYSTEMS

The proposed systems would contain some types of components and subsystems not previously used for reactor protection. It is necessary that the required system reliability, both during normal operation and under postulated abnormal conditions, be established through an appropriate combination of tests and analyses.

IIB-2 - QUALIFICATION OF NEW FUEL GEOMETRIES

New fuels proposed for both BWRs and PWRs include the 8x8 (BWR) and 16x16 and 17x17 PWR fuels. The Committee recognizes that these fuels are intended to operate at power densities lower than earlier fuel designs. However, testing programs are considered necessary to establish their densification behavior (IC-2) as well as their behavior under abnormal conditions (II-7). Appropriate experimental programs should be developed dealing with flow blockage, behavior of fuel after partial melting and fuel response under transient conditions. It is anticipated that the solution of this item will include a synthesis of PBF data, experiments on earlier fuel types, behavior of fuel in commercial reactors, and confirmatory experiments on these fuel designs.

IIB-3 - BEHAVIOR OF BWR MARK III CONTAINMENT

The BWR Mark III Containment differs in many respects from the Mark I and II designs. Various aspects such as vent clearing, vent/coolant interaction, pool swell, pool stratification, pressure loads, and flow bypass must be evaluated and approved; ongoing experimental tests should develop much of the necessary data to confirm the conservatism in design.

IIB-4 - STRESS CORROSION CRACKING IN BWR PIPING

Several failures have occurred in operating BWRs. An ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution, and extensive programs are underway by Industry, ERDA and NRC.

The austenitic stainless steels are commonly used as piping material in many of the smaller BWR lines. A combination of weld sensitization, residual stresses, superposed loads, and oxygen equal to or greater than 0.2 ppm in the BWR coolant can lead to cracking, initiating on the inner surface and propagating through the wall. In most cases there will be a leak well before pipe failure so there is adequate warning; however, one can postulate a LOCA caused by a guillotine break with minimal prior warning. Current efforts are to minimize stress corrosion by using other materials.

Group IIC - Resolution Pending - Items Added Since March 12, 1975

1. Locking Out of ECCS Power Operated Valves: The Committee suggests that further attention be given to procedures involving locking out electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS.

2. Fire Protection: The Committee recommends review of design features intended to prevent the occurrence of damaging fires and to minimize the consequences to safety-related equipment should a fire occur.

3. Design Features to Control Sabotage: Attention should be given to aspects of design that could improve plant security.

4. Decontamination and Decommissioning of Reactors: Specific plans should be developed, including definitive codes and standards covering plant decommissioning. Also experience should be gained in reactor decontamination so that such information is available when needed.

5. Vessel Support Structures: Questions that have arisen concerning the loads on pressure vessel support structures due to certain postulated loss-of-coolant accidents should be resolved.

6. Water Hammer: Several cases of water slugging or water hammer have occurred in both PWRs and BWRs. Corrective measures should be taken to minimize such events.

7. Maintenance and Inspection of Plants: Provisions should be included in the design of future plants which anticipate the maintenance, inspection and operational needs of the plant throughout its service life.

8. Behavior of BWR Mark I Containments: Various aspects relevant to the BWR Mark I Containment should be resolved. Included are such items as relief valve restraint, control of local dynamic loads in the torus, vent clearing and establishment of torus water temperature limits during a LOCA. This is an extension of Item 1 in Group IIA.

IIC-1 - LOCKING OUT OF ECCS POWER-OPERATED VALVES

The physical locking out of electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS has been required, based on the assumption that a spurious electrical signal at an inopportune time could activate the valves to the adverse position; e.g., closed rather than open, or opened rather than closed. While such an event has a finite probability another probability exists that the valves might be adversely positioned due to operator error.

The ACRS believes the matter should be studied using a systems approach, and considering such items as (1) the evaluation of the probability of a spurious signal; (2) time required to reactivate the valve operator; (3) status of signal lights when the ciruit breaker is open; (4) can the valve be locked out in an improper position due to a faulty indicator; (5) are there other designs improving reliability without lock-out; (6) what are the advantages and disadvantages of corrective action by an alert operator in case of incorrect positioning vis-a-vis a system with power locked out.

IIC-2 - FIRE PROTECTION

The incidence of fires having the potential of damage to safety-related equipment has been sufficiently large to warrant concern. Measures should be taken to review design features intended to prevent the occurrences of damaging fires and to minimize the consequences to safety-related equipment should a fire occur. Where feasible, modifications should be made in older plants in addition to incorporating such changes into plants now being designed. Particular attention should be devoted to regions where vital systems are in near juxtaposition since an event such as a fire could affect redundant and diverse engineered safety functions.

IIC-3 - DESIGN FEATURES TO CONTROL SABOTAGE

Considerable attention has been devoted to control of industrial sabotage of nuclear power plants, particularly with regard to control of unauthorized access, and potential modes of sabotage by individuals or groups external to the operating organization. The ACRS believes that deliberate attention should be given to aspects of design that could improve plant security. With the emphasis being placed on standardized plant designs, it becomes especially important to introduce design measures that could protect against industrial sabotage, or mitigate the consequences thereof.

IIC-4 - DECONTAMINATION AND DECOMMISSIONING OF REACTORS

The Committee believes that well developed plans, confirmed by appropriate experiments where necessary, should be available to cover such events as the decontamination of primary reactor systems and the decommissioning of reactors. At this time the information on full scale decontamination is quite limited, particularly for BWRs. Examples of potential problems include such items as handling of decontamination solutions, potential hideout of radioactive products, enhanced corrosion and crud formation following decontamination, and the possible imcompatibility of the different alloys in the pressure boundary to the decontamination solutions.

Experience is limited with regard to decommissioning operations including rules for dismantling and for mothballing. Definitive plans and standards should be developed covering such items as adequacy of action, problems in restitution of site, mutual responsibility of State and Federal Government, etc.

IIC-5 - VESSEL SUPPORT STRUCTURES

A possible consequence of the instantaneous double-ended pipe break postulated to occur in certain large pipes of PWRs is the asymmetric loading of the reactor pressure vessel support structures. The magnitude and effects of such loads on the pressure vessel should be determined to establish if such loads adversely affect the predicted course of a LOCA. If analysis indicates that the results are unacceptable, appropriate corrective action should be taken. A potential effect is pressure vessel movement due to blowdown jet forces at the location of the rupture, transient differential pressures in the annular region between the vessel and the shield, and transient differential pressures across the core barrel within the reactor vessel.

IIC-6 - WATER HAMMER

Several instances of water slugging or water hammer have occurred in both BWRs and PWRs due to causes such as the trapping of water between two valves. This slug of water is accelerated by steam or water once the valves are opened. The stored energy is sufficient to damage piping, bend or break pipe restraints, and damage support structures. Water hammer may occur due to flow instabilities in steam generators in conjunction with water flowing into the feedwater inlets, resulting in comparable damage.

Corrective measures should be taken to minimize such occurrences after completion of analytic and experimental studies directed to an understanding of the causes.

IIC-7 - MAINTENANCE AND INSPECTION OF PLANTS

Experience with older plants has verified that appropriate modifications in piping layout, with respect of walls and structures, type of insulation used, and weld joint design, to cite some obvious items, lead to improved maintenance, more reliable inservice inspections, and a better meeting of the operational needs of the plant throughout its service life, including decontamination and eventual decommissioning. An additional benefit is the reduction in personnel exposures in plants making them more amenable to maintenance and inspection. Appropriate changes should be considered in future designs to meet these criteria.

IIC-8 - BEHAVIOR OF BWR MARK I CONTAINMENTS

Recent tests on the BWR Mark I Containment design revealed phenomena not anticipated on the basis of earlier tests where pressure loads were imposed by insertion of air. Specific problems somewhat comparable to those under review for the Mark III Containment, include relief valve discharge pipe restraints in the torus, local dynamic loads on the torus, vent clearing, and influence of torus temperature on the LOCA.

Ongoing experiments are expected to develop the necessary data to confirm the adequacy of the existing design or to permit necessary modifications.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 24, 1977

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS: REPORT NO. 5

Dear Mr. Rowden:

The Advisory Committee on Reactor Safeguards has previously reported on the "Status of Generic Items Relating to Light-Water Reactors" in its letters of December 18, 1972, February 13, 1974, March 12, 1975 and April 16, 1976. Since the Committee limits its definition of generic items to those cited specifically in its letters pertaining to projects and related matters, the attached listing is not all-inclusive; the Nuclear Regulatory Commission Staff list has additional generic items.

Group I of the attachment is a reiteration of the generic items considered resolved at the time the Committee issued its first report. Group IA includes those items resolved between December 1972 and February 1974; Group IB includes those items resolved between February 1974 and March 1975; Group IC includes those items resolved between March 1975 and April 1976; Group ID includes those items resolved since April 1976. Following each resolved item is a brief statement of the specific action that resulted in the resolution. Group II lists those items included in the original report for which resolution on a generic basis is still pending. Groups IIA, IIB and IIC include generic items that were added in the second, third and fourth reports; Group IID includes those added in the present report. The ACRS and the NRC Staff will continue to consider the safety significance of Group II, IIA, IIB, IIC and IID items on a case-by-case basis until generic resolution is reached. Formal actions such as issuance of Regulations or Regulatory Guides are anticipated for many of the Group II, IIA, IIB, IIC and IID items.

The April 16, 1976, ACRS letter cited nine items that the NRC Staff considers to be "resolved", based on specific positions taken by the Staff. However, these "resolutions" did not meet the ACRS criteria for adequate documentation and Committee approval and were therefore considered to be pending until formally discussed and accepted by the ACRS. Items in this category were:

(1) "Effective Operation of Containment Sprays in a LOCA";
(2) "Instruments to Detect Fuel Failures";
(3) "Common Mode Failures";
(4) "The Advisability of Seismic Scram";
(5) "Pressure in Containment Following LOCA";
(6) "Control Rod Drop Accidents (BWR)";
(7) "Rupture of High Pressure Lines Outside Containment";
(8) "Isolation of Low Pressure from High Pressure Systems";
(9) "Behavior of BWR Mark III Containments."

Since the April 16, 1976 ACRS letter, the NRC Staff has provided documents acceptable to the Committee for items 2 and 5, above. Two other items which have been resolved since the Report of April 16, 1976 are also listed in Group ID.

Owing to questions raised concerning the scope and intent of various generic issues, the Committee has incorporated into the attachments a brief description for all items unresolved now or in earlier reports.

With regard to the status of generic issues, as they apply to each plant, the ACRS understands that the NRC Staff will address the status of the issues in the applicable Safety Evaluation Report.

"Resolved" as used in the Generic Items reports refers to the following: In some cases an item has been resolved in an administrative sense, recognizing that technical evaluation and satisfactory implementation are yet to be completed. Anticipated Transients Without Scram represents an example of this category. In other instances, the resolution has been accomplished in a narrow or specific sense, recognizing that further steps are desirable as practical or that different aspects of the problem require further investigation. Examples are the possibility of improved methods of locating leaks in the primary system, and of improved methods or augmented scope to inservice inspection of reactor pressure vessels.

Sincerely yours,

M. Bender
Chairman

Attachments:
(1) Group I; (2) Group IA; (3) Group IB; (4) Group IC;
(5) Group ID; (6) Group II; (7) Group IIA; (8) Group IIB;
(9) Group IIC; (10) Group IID

GENERIC ITEMS

Group I - Resolved Generic Items

1. Net Positive Suction Head for ECCS Pumps: Covered by Regulatory Guide 1.1.

2. Emergency Power: Covered by Regulatory Guides 1.6, 1.9, and 1.32 and portions of IEEE-308 (1971).

3. Hydrogen Control After a Loss-of-Coolant Accident (LOCA): ACRS concurred in proposed Staff position, covered by NRC Standard Review Plan for Nuclear Power Plants.

4. Instrument Lines Penetrating Containment: Covered by Regulatory Guide 1.11 and Supplement.

5. Strong Motion Seismic Instrumentation: Covered by Regulatory Guide 1.12.

6. Fuel Storage Pool Design Bases: Covered by Regulatory Guide 1.13.

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles: Covered by Regulatory Guide 1.14.

8. Protection Against Industrial Sabotage: Covered by Regulatory Guide 1.17.

9. Vibration Monitoring of Reactor Internals and Primary System: Covered by Regulatory Guide 1.20.

10. Inservice Inspection of Reactor Coolant Pressure Boundary: Covered by ASME Boiler and Pressure Vessel (BPV) Code, Section XI and Regulatory Guide 1.65.

11. Quality Assurance During Design, Construction and Operation: Covered by 10 CFR 50, Appendix B; ASME BPV Code, Section III; ANSI N-45.2-1971, Regulatory Guides 1.28, 1.33, 1.64, 1.70.6 and Proposed Standard ANS-3.2.

12. Inspection of BWR Steam Lines Beyond Isolation Valves: Covered by ASME BPV Code, Section XI.

13. Independent Check of Primary System Stress Analysis: Covered by ASME BPV Code, Section III.

14. Operational Stability of Jet Pumps: Test and operating experience at Dresden 2 and 3 and other jet pump BWRs have satisfied the ACRS concerns.

Group I Continued

15. Pressure Vessel Surveillance of Fluence and NDT Shift: Covered by 10 CFR 50, Appendix A and Appendix H; and ASTM Standard E-185.

16. Nil Ductility Properties of Pressure Vessel Materials: Covered by 10 CFR 50, Appendix A and Appendix G; ASME BPV Code, Section III; "Report on the Integrity of Reactor Vessels for Light-Water Power Reactors," (WASH-1285) by the Advisory Committee on Reactor Safeguards dated January 1974.

17. Operation of Reactor With Less Than All Loops In Service: Covered by ACRS-Regulatory Staff position that manual resetting of several set points on the control room instruments under specific conditions and procedures is acceptable in taking one primary loop out of service. This position is based on the expectation that this mode of operation will be infrequent. Cited in Standard Review Plan Appendix 7-A, Branch Technical Position EICSB 12.

18. Criteria for Preoperational Testing: Covered by Regulatory Guide 1.68.

19. Diesel Fuel Capacity: Covered by ACRS-Regulatory Staff position requiring 7 days fuel.

20. Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends: Covered by ACRS-Regulatory Staff position cited in several letters that such a failure should have no unacceptable consequences.

21. Operating One Plant While Other(s) is/are Under Construction: Specific requirements have been established by ACRS-Regulatory Staff. Covered in Regulatory Guide 1.17, 1.70 Section 13.6.2; 1.101; ANSI N 18.17 and Standard Review Plan 13.3 Appendix A and 13.6.

22. Seismic Design of Steam Lines: Covered by Regulatory Guide 1.29.

23. Quality Group Classifications for Pressure Retaining Components: Covered by Regulatory Guide 1.26.

24. Ultimate Heat Sink: Covered by Regulatory Guide 1.27.

25. Instrumentation to Detect Stresses in Containment Walls: Covered by Regulatory Guide 1.18.

Group IA - Generic Items Resolved Since December 18, 1972

1. Use of Furnace Sensitized Stainless Steel: Covered by Regulatory Guide 1.44.

2. Primary System Detection and Location of Leaks: Covered by Regulatory Guide 1.45.

3. Protection Against Pipe Whip: Covered by Regulatory Guide 1.46.

4. Anticipated Transients Without Scram: Covered by Regulatory Position Document, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, September 1973.

5. ECCS Capability of Current and Older Plants: Covered by Rulemaking as a general policy decision, although acceptable detailed implementation remains to be developed. Docket RM-50-1, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors," December 28, 1973.

Group IB - Generic Items Resolved Since February 13, 1974

1. Positive Moderator Coefficient: PWRs presently have or expect to have zero or negative coefficients. Where some Technical Specifications allow a slightly positive coefficient, the accident and stability analyses take this into account. Burnable poison provisions have been designed into PWRs to reduce otherwise excessive positive coefficients to allowable values.

2. Fixed Incore Detectors on High Power PWRs: Fixed incore detectors are not required for PWRs since reviews of potential power distribution anomalies have not revealed a clear need for continuous incore monitoring.

3. Performance of Critical Components (pumps, cables, etc.) in post-LOCA Environment: Qualification requirements of critical components are now covered by Regulatory Guides 1.40, 1.63, 1.73 and 1.89 and IEEE Standards 382-1972, 383-1974, 317-1972, 323-1974.

4. Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containments: On designs prior to GE Mark III containment, resolution lies in surveillance and testing of vacuum relief valves. For Mark III containments, an additional requirement is that the design be capable of accommodating a bypass equivalent to one square foot for a given flow condition.

5. Emergency Power for Two or More Reactors at the Same Site: Resolved by issue of Regulatory Guide 1.81.

6. Effluents from Light-Water-Cooled-Nuclear Power Reactors: Resolved by issue of Appendix I to 10 CFR 50.

7. Control Rod Ejection Accident: Resolved for PWRs by Regulatory Guide 1.77.

Group IC - Generic Items Resolved Since March 12, 1975

1. Main Steam Isolation Valve Leakage of BWR's: Covered by Regulatory Guide 1.96.

2. Fuel Densification: Covered by 10 CFR 50 Appendix K plus case-by-case review of vendor fuel models.

3. Rod Sequence Control Systems: Covered by NRC Staff Review and Approval of NEDO-10527 and Presentation to ACRS.

4. Seismic Category I Requirements for Auxiliary Systems: Covered by Regulatory Guides 1.26 and 1.29.

Group ID - Generic Items Resolved Since April 16, 1976

1. Instruments to Detect (limited) Fuel Failures - NRC document, "Fuel Failure Detection in Operating Reactors", B. L. Siegel and H. H. Hagen, June, 1976 resolves issue for limited fuel failures, but not for severe failures (See II-4).

2. Instrumentation to Follow the Course of an Accident" Regulatory Guide 1.97 Revision 1 resolves ACRS concerns.

3. Pressure in Containment Following LOCA-NRC document, "Containment Subcompartment Analysis" September 1976.

4. Fire Protection. Resolved by Branch Technical Position 9.5.1, and Regulatory Guide 1.120.

Group II - Resolution Pending

1. Turbine Missiles: Turbine failures for past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discusses the problems.*

2. Effective Operation of Containment Sprays in a LOCA: Extensive documentation in topical reports. Review and evaluation are required.

3. Possible Failure of Pressure Vessel Post-LOCA By Thermal Shock: Regulatory Guide 1.2 covers current information. Ultimate position as to significance of thermal shock requires input of fracture mechanics data on irradiated steels from the Heavy Section Steel Technology Program.

**4. Instruments to detect (severe) fuel failures - NRC document, "Fuel Failure Detection in Operating Reactors", B. L. Siegel and H. H. Hagen. Item ID covers limited failures. More work is required for the severe failure case to establish instrumentation criteria.

5. Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel: State-of-the-Art results appear promising. More work may be required prior to decision as to installation of equipment.

6. Common Mode Failures - Requirements for diverse components should be established. An example of a partial solution to the problem of non-random, multiple failures is WASH 1270, "Technical Report on Anticipated Transients Without Scram" (Item IA-4).

7. Behavior of Reactor Fuel Under Abnormal Conditions: This includes: flow blockage; partial melting of fuel assemblies as it affects reactor safety; and transient effects on fuel integrity. The PBF program will address some of these items.

8. BWR Recirculation Pump Overspeed During LOCA: Decision required by ACRS-Regulatory Staff.

9. The Advisability of Seismic Scram: Further studies required to establish need.

10. Emergency Core Cooling System Capability for Future Plants: Partially resolved by amendments to 10 CFR 50 [50.34(a)(4), 50.34(b)(4), 50.46, and Appendix K]. LOCA evaluation model complete. ACRS feels new cooling approaches should be explored.

---

* Regulatory Guide is in preparation.

** Identified in the Committee's Report of April 16, 1976 as "Instruments to Detect Fuel Failures."

II-1 - TURBINE MISSILES

Turbine failures for the past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discuses the problem.

Three issues require answers to resolve the turbine missile problem: (1) The first relates to the appropriate failure probability value; based on historical failures the probability is about 10^{-4}. Industry predicts a much lower failure probability based on improvements in materials and design. To date the ACRS has accepted the more conservative value; (2) The second issue is strongly dependent on turbine orientation with respect to critical safety structures. Strike probabilities from high angle missiles are acceptably low for single units and may be acceptable for multi-unit plants, depending on plant layout; however, lower angle missiles with non-optimum (tangential) turbine orientation have unacceptably high strike probabilities; (3) The third issue is one of penetration and damage of structures housed in the containment. The limited experimental data pertaining to penetration of large irregularly shaped missiles are not sufficient to determine structural response to impingement of turbine disc segments. Most missile penetration formulas are not relevant to this case. Some experiments with irregular missiles might resolve this issue, particularly for older plants with non-optimum turbine orientations.

II-2 - EFFECTIVE OPERATION OF CONTAINMENT SPRAYS IN A LOCA

Review and evaluation are required of the variety of experiments which have been conducted on the effectiveness of various containment sprays on the removal and retention of airborne radioactive materials anticipated to be present within containment following a LOCA. Such review should consider adequacy of definition of the physical and chemical forms of the anticipated airborne radionuclides, and quality of evaluative tests of the removal efficiencies of various sprays under the conditions of temperature, pressure, and radiation doses expected to exist under LOCA conditions. A desirable extension might be analyses of the use of sprays containing chemicals (such as NaOH) which have the potential for damaging equipment within containment. Studies using other spray additives, such as hydrazine, have been conducted. If compounds, such as this, have distinct advantages, insofar as minimizing equipment damage in the event of inadvertent actuation, action should be taken to encourage their use.

II-3 - POSSIBLE FAILURE OF PRESSURE VESSEL POST-LOCA BY THERMAL SHOCK

Earlier nuclear reactor pressure vessels subjected to fluences of $1\text{-}4 \times 10^{19}$ nvt, which are anticipated in the last 20 years of a 40-year life, may suffer severe radiation damage denoted by a pronounced shift in impact transition temperature at the inner surface. There will be a damage gradient which decreases sharply, so that the properties halfway through the wall are essentially those of the as-fabricated material. If a LOCA occurs near end-of-life, the injection of cold water on the region of degraded properties may initiate and propagate a crack because of high local stresses near the surface. Analytic procedures indicate the stresses drop rapidly with distance through the wall so the flaw should not propagate beyond some limiting point. The lack of experimental evidence and the relative width of the error band in the analytic results are such that some experiments are required to validate the analytic model. These are planned under the HSST program.

II-4 - INSTRUMENTS TO DETECT (SEVERE) FUEL FAILURES

In the event of substantial fuel failure, including the possibility of fuel melt, large amounts of fission products could be rapidly released to the reactor coolant and possibly to the environment. Instrumentation capable of early warning and timely response may avert an incident becoming an accident.

Instrumentation related to such diagnostic purposes for limited fuel failure is being used on most power reactors. (See Item ID-1). Further work is required to establish criteria for similar instrumentation for severe fuel failures.

II-5 - MONITORING FOR EXCESSIVE VIBRATION OR LOOSE PARTS INSIDE THE PRESSURE VESSEL

Loose parts monitoring can provide early warning of potential mechanical problems or failures within the pressure vessel and throughout the primary coolant circuit. Reactor vendors have developed monitoring systems; however, general requirements remain to be established.

II-6 - NON-RANDOM MULTIPLE FAILURES (FORMERLY "COMMON MODE FAILURE")

The term "common mode failures" has, in many instances come to mean multiple failures of identical components exposed to identical or nearly identical conditions or environments, and the use of diversity in components has been proposed or required to avoid such failures. The concern of the ACRS is better expressed by the term "non-random multiple failures", which is intended to include not only the type of "common mode failure" discussed above but other types of multiple failures for which the consequences and probabilities cannot be predicted by application of the single-failure criterion. Examples include the use of the same sensors or components for both control and protection systems (a resolved matter); sequential multiple failures due to a "domino effect", and simultaneous multiple failures due to a single fault. Since designs usually do not knowingly incorporate features susceptible to such failures, techniques and criteria need to be developed to detect and avoid them in all systems important to safety.

II-7 - BEHAVIOR OF REACTOR FUEL UNDER ABNORMAL CONDITIONS

The Behavior of Reactor Fuel under Abnormal Conditions is still considered unresolved due to the limited experimental data available. Partial melting of fuel assemblies due to flow blockage might lead to autocatalytic effects leading to more extensive fuel failure, pressure pulses, etc. Similar behavior might occur in the case of reactivity transients. The ACRS encourages analytic modeling but believes appropriate experimental data are necessary. It is anticipated that tests in the Power Burst Facility (PBF) should supply much of the required data.

II-8 - BWR PUMP OVERSPEED DURING A LOCA

It is possible for a BWR recirculation pump to overspeed if a large break occurs at the appropriate position in specific piping. Conservative estimates indicate substantial overspeed and possible failure of components with the generation of missiles. The problem is being approached analytically and experimentally with scaled pumps. The reliabiity of such protective measures as the use of decouplers between pump and motor is under study.

II-9 - THE ADVISABILITY OF SEISMIC SCRAM

The ACRS has recommended that studies be made of techniques for seismic scram and of the potential safety advantages and potential disadvantages of prompt reactor scram in the event of strong seismic motion, say more than one-half the safe shutdown earthquake. Various suitable techniques have been identified and exist, but thus far only limited studies have been reported on the pros and cons of seismic scram. The principal potential advantage identified arises from the greatly improved coolability of a core in the unlikely event of a seismically induced LOCA, should scram precede the LOCA by several seconds. A principal reason given in opposition to seismic scram relates to a stated interest in keeping power stations on the line to provide power offsite should a severe earthquake occur.

II-10 - ECCS CAPABILITY FOR FUTURE PLANTS

The ACRS has placed considerable emphasis on ECCS safety R&D so that the extent of the conservatism in the ECCS licensing requirements could be made more precise. With more experimental data a realistic and quantitative appraisal of ECC systems would lead to valid judgments on the changes in licensing which could be put on a firm basis.

Parallel approaches that seek to improve the reliability of ECC systems, to improve the monitoring of low power peaking, and to improve those fuel assembly designs which lower peaking factors, are encouraged. Further, changes in plant design which improve the reflooding of the reactor core should be sought and evaluated.

R&D efforts on analysis of core blowdown and reflood should be increased and combined with the results of the standard problems and the associated experiments. Improved analytical methods would provide a basis for optimized ECCS.

Group IIA - Resolution Pending - Items Since December 18, 1972

1. Control Rod Drop Accident (BWRs): Calculations indicated that the reactivity response differs from earlier values. New analyses are required including three dimensional effects.

2. Ice Condenser Containments: Additional analyses are required to establish response during a LOCA, and to establish design margins.

3. Rupture of High Pressure Lines Outside Containment: The possibility exists that failure of a high pressure line such as a steam pipe can prevent operation of critical safety components.

4. PWR Pump Overspeed During a LOCA: Problem arises in similar manner to that of BWRs (Item 8 Group II).

5. Isolation of Low Pressure From High Pressure Systems: Assurance required that low pressure systems cannot inadvertently be interconnected with a high pressure system leading to failure. There are potential interaction problems between Class 1 and Class 2 or Class 3 pressure connections.

6. Steam Generator Tube Leakage: Partially resolved by issuance of Regulatory Guide 1.83 which addresses the concern from a preventative point of view.

7. ACRS/NRC Periodic 10-Year Review of all Power Reactors: A more effective, continuous alternative approach to periodic reviews is being proposed. Pending ACRS review, this item is still considered unresolved.

IIA-1 - CONTROL ROD DROP ACCIDENT (BWRs)

Some uncertainties have arisen in previous calculations of this postulated accident, including the choice of negative reactivity insertion rate due to scram and the potential differences between a two dimensional and a three dimensional calculation. Particularly for the latter point, more precise theoretical comparisons may be required to resolve the matter although probabilistic considerationss may be relevant.

IIA-2 - ICE CONDENSER CONTAINMENTS

The ice condenser containments have substantially smaller volume on the assumption that the ice will condense the steam during a LOCA, thus preventing system overpressurization. The rate of condensation is critical in the initial stages of the blowdown and is influenced by interaction of vapor with the ice. If the current analyses prove that the condensation model is suitably conservative, the problem may be resolved.

IIA-3 - RUPTURE OF HIGH PRESSURE LINES OUTSIDE CONTAINMENT

The functional or structural integrity of components required for safe shutdown and the maintenance of cold shutdown may be endangered by a failure of high pressure piping at certain locations outside of containment. Specific design and single failure criteria need to be developed to resolve this problem.

IIA-4 - PWR PUMP OVERSPEED DURING A LOCA

It is possible for a PWR primary coolant pump to overspeed if a large break occurs at the appropriate position in specific piping. Conservative estimates indicate substantial overspeed and possible failure of components such as flywheels with the generation of missiles. The problem is being approached analytically and experimentally with scaled pumps. The reliability of such protective measures as electrical braking of the pump motor is under study.

IIA-5 - ISOLATION OF LOW PRESSURE FROM HIGH PRESSURE SYSTEMS

Assurance is required that low pressure systems cannot inadvertently be interconnected with a high pressure system leading to a failure of the low pressure systems due to overpressurization. Both Class 2 systems required for ECCS and Class 3 systems used for many auxiliary coolant systems provide vital functions during normal and accident conditions. Any conditions leading to interconnection could trigger a LOCA. It is recognized that these systems must be interconnected because of the functions they perform. Therefore, particular attention is required to valve reliability and reliability of valve actuating circuits to assure that these valves will not open while the primary system is at pressure.

IIA-6 - STEAM GENERATOR TUBE LEAKAGE

Normally the steam generator is not a critical component during a LOCA-ECCS. However, a special case exists where the steam generator tubes have been degraded due to corrosion, wastage, etc. If the shock loads imposed by the LOCA cause a critical number of tubes to fail, say by a double-ended (guillotine) break, the inflow from the secondary side can cause choking of flow during ECCS, preventing adequate cooling of the core. The critical number of tubes is relatively small. A position such as one specifying a statistically significant level of nondestructive examination (NDE) might resolve this issue. The purpose of NDE would be to confirm that damage is not excessive; such examinations should minimize the possibility of catastrophic failure of a significant number of tubes.

IIA-7 - PERIODIC (10-YEAR) REVIEW OF ALL POWER REACTORS

In its report of June 14, 1966, the ACRS recommended that periodic comprehensive reviews be conducted of operating licensed power reactors by the NRC Staff. These reviews would be preceded by a comprehensive report by the operator which evaluated the past experience and the safety of future operation of the plant.

The NRC Staff has maintained a continuing review of the safety of operating plants. In particular, as generic matters of potential safety significance arise, the appropriate operating reactors are asked to assess the relevance of the matter to each particular reactor. This is a necessary but different aspect of the continuing surveillance and review of the safety of operating reactors than was envisaged by the ACRS in its recommendation of June 1966.

The Committee continues to believe both approaches are desirable and awaits the development of a program of periodic comprehensive reviews.

Group IIB - Resolution Pending - Items Added Since February 13, 1974

**1. Computer Reactor Protection System: Systems should be qualified for reliability, particularly through in situ tests and under various environmental conditions, prior to use in reactor system.

2. Qualification of new fuel geometries: The 16x16 and 17x17 PWR, and 8x8 BWR fuels should undergo testing to meet Item 2 in Group IC and Item 7 in Group II.

3. Behavior of BWR Mark III Containments: Various aspects, including vent clearing, vent/coolant interaction, pool swell, pool stratification, pressure loads and flow bypass should be resolved. This is an extension of Item 3 in Group ID.

4. Stress Corrosion Cracking in BWR Piping: Several failures have occurred in operating BWRs. The ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution and extensive programs are underway by industry, ERDA, and NRC.

** Identified in the ACRS Report of April 16, 1976 as "Hybrid Reactor Protection System."

IIB-1 - COMPUTER REACTOR PROTECTION SYSTEMS**

The proposed systems would contain some types of components and subsystems not previously used for reactor protection. It is necessary that the required system reliability, both during normal operation and under postulated abnormal conditions, be established through an appropriate combination of tests and analyses. While the issue originated with the B&W Hybrid concept it is equally applicable to the proposed CE and W computer reactor protection systems.

** Identified in the ACRS Report of April 16, 1976 as "Hybrid Reactor Protection System."

IIB-2 - QUALIFICATION OF NEW FUEL GEOMETRIES

New fuels proposed for both BWRs and PWRs include the 8x8 (BWR), and 16x16 and 17x17 PWR fuels. The Committee recognizes that these fuels are intended to operate at power densities lower than earlier fuel designs. However, testing programs are considered necessary to establish their densification behavior (IC-2) as well as their behavior under abnormal conditions (II-7). Appropriate experimental programs should be developed dealing with flow blockage, behavior of fuel after partial melting and fuel response under transient conditions. It is anticipated that the solution of this item will include a synthesis of Power Burst Facility data, experiments on earlier fuel types, behavior of fuel in commercial reactors, and confirmatory experiments on these fuel designs.

IIB-3 - BEHAVIOR OF BWR MARK III CONTAINMENT

The BWR Mark III Containment differs in many respects from the Mark I and II designs. Various aspects such as vent clearing, vent/coolant interaction, pool swell, pool stratification, pressure loads, and flow bypass must be evaluated and approved; ongoing experimental tests should develop much of the necessary data to confirm the conservatism in design.

IIB-4 - STRESS CORROSION CRACKING IN BWR PIPING

Several failures have occurred in operating BWRs. An ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution, and extensive programs are underway by Industry, ERDA and NRC.

The austenitic stainless steels are commonly used as piping material in many of the smaller BWR lines. A combination of weld sensitization, residual stresses, superposed loads, and oxygen equal to or greater than 0.2 ppm in the BWR coolant can lead to cracking, initiating on the inner surface and propagating through the wall. In most cases there will be a leak well before pipe failure so there is adequate warning; however, one can postulate a LOCA caused by a guillotine break with minimal prior warning. Current efforts are to minimize stress corrosion by using other materials.

Group IIC - Resolution Pending - Items Added Since March 12, 1975

1. Locking Out of ECCS Power Operated Valves: The Committee suggests that further attention be given to procedures involving locking out electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS.

2. Design Features to Control Sabotage: Attention should be given to aspects of design that could improve plant security.

3. Decontamination and Decommissioning of Reactors: Specific plans should be developed, including definitive codes and standards covering plant decommissioning. Also experience should be gained in reactor decontamination so that such information is available when needed.

4. Vessel Support Structures: Questions that have arisen concerning the loads on pressure vessel support structures due to certain postulated loss-of-coolant accidents should be resolved.

5. Water Hammer: Several cases of water slugging or water hammer have occurred in both PWRs and BWRs. Corrective measures should be taken to minimize such events.

6. Maintenance and Inspection of Plants: Provisions should be included in the design of future plants which anticipate the maintenance, inspection and operational needs of the plant throughout its service life.

7. Behavior of BWR Mark I Containments: Various aspects relevant to the BWR Mark I Containment should be resolved. Included are such items as relief valve restraint, control of local dynamic loads in the torus, vent clearing and establishment of torus water temperature limits during a LOCA. This is an extension of Item 3 in Group ID.

IIC-1 - LOCKING OUT OF ECCS POWER-OPERATED VALVES

The physical locking out of electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS has been required, based on the assumption that a spurious electrical signal at an inopportune time could activate the valves to the adverse position; e.g., closed rather than open, or open rather than closed. While such an event has a finite probability another probability exists that the valves might be adversely positioned due to operator error.

The ACRS believes the matter should be studied using a systems approach, and considering such items as (1) the evaluation of the probability of a spurious signal; (2) time required to reactivate the valve operator; (3) status of signal lights when the ciruit breaker is open; (4) the possibility of locking out in an improper position due to a faulty indicator; (5) other designs with improved reliability without lock-out; (6) the advantages and disadvantages of corrective action by an alert operator in case of incorrect positioning vis-a-vis a system with power locked out.

IIC-2 - DESIGN FEATURES TO CONTROL SABOTAGE

Considerable attention has been devoted to control of industrial sabotage of nuclear power plants, particularly with regard to control of unauthorized access, and potential modes of sabotage by individuals or groups external to the operating organization. The ACRS believes that deliberate attention should be given to aspects of design that could improve plant security. With the emphasis being placed on standardized plant designs, it becomes especially important to introduce design measures that could protect against industrial sabotage, or mitigate the consequences thereof.

IIC-3 - DECONTAMINATION AND DECOMMISSIONING OF REACTORS

The Committee believes that well developed plans, confirmed by appropriate experiments where necessary, should be available to cover such events as the decontamination of primary reactor systems and the decommissioning of reactors. At this time the information on full scale decontamination is quite limited, particularly for BWRs. Examples of potential problems include such items as handling of decontamination solutions, potential hideout of radioactive products, enhanced corrosion and crud formation following decontamination, and the possible imcompatibility of the different alloys in the pressure boundary to the decontamination solutions.

Experience is limited with regard to decommissioning operations including rules for dismantling and for mothballing. Definitive plans and standards should be developed covering such items as adequacy of action, problems in restitution of site, mutual responsibility of State and Federal Government, etc.

IIC-4 - VESSEL SUPPORT STRUCTURES

A possible consequence of the instantaneous double-ended pipe break postulated to occur in certain large pipes of PWRs is the asymmetric loading of the reactor pressure vessel support structures. The magnitude and effects of such loads on the pressure vessel should be determined to establish if such loads adversely affect the predicted course of a LOCA. If analysis indicates that the results are unacceptable, appropriate corrective action should be taken. A potential effect is pressure vessel movement due to blowdown jet forces at the location of the rupture, transient differential pressures in the annular region between the vessel and the shield, and transient differential pressures across the core barrel within the reactor vessel.

IIC-4 - VESSEL SUPPORT STRUCTURES

A possible consequence of the instantaneous double-ended pipe break postulated to occur in certain large pipes of PWRs is the asymmetric loading of the reactor pressure vessel support structures. The magnitude and effects of such loads on the pressure vessel should be determined to establish if such loads adversely affect the predicted course of a LOCA. If analysis indicates that the results are unacceptable, appropriate corrective action should be taken. A potential effect is pressure vessel movement due to blowdown jet forces at the location of the rupture, transient differential pressures in the annular region between the vessel and the shield, and transient differential pressures across the core barrel within the reactor vessel.

IIC-5 - WATER HAMMER

Several instances of water slugging or water hammer have occurred in both BWRs and PWRs due to causes such as the trapping of water between two valves. This slug of water is accelerated by steam or water once the valves are opened. The stored energy is sufficient to damage piping, bend or break pipe restraints, and damage support structures. Water hammer may occur due to flow instabilities in steam generators in conjunction with water flowing into the feedwater inlets, resulting in comparable damage.

Corrective measures should be taken to minimize such occurrences after completion of analytic and experimental studies directed to an understanding of the causes.

IIC-6 - MAINTENANCE AND INSPECTION OF PLANTS

Experience with older plants has verified that appropriate modifications in piping layout, with respect to walls and structures, type of insulation used, and weld joint design, to cite some obvious items, lead to improved maintenance, more reliable inservice inspections, and a better meeting of the operational needs of the plant throughout its service life, including decontamination and eventual decommissioning. An additional benefit is the reduction in personnel exposures in plants making them more amenable to maintenance and inspection. Appropriate changes should be considered in future designs to meet these criteria.

IIC-7 - BEHAVIOR OF BWR MARK I CONTAINMENTS

Recent tests on the BWR Mark I Containment design revealed phenomena not anticipated on the basis of earlier tests where pressure loads were imposed by insertion of air. Specific problems somewhat comparable to those under review for the Mark III Containment, include relief valve discharge pipe restraints in the torus, local dynamic loads on the torus, vent clearing, and influence of torus temperature on the LOCA.

Ongoing experiments are expected to develop the necessary data to confirm the adequacy of the existing design or to permit necessary modifications.

Group IID - RESOLUTION PENDING - Items added since April 16, 1976

1. Safety related interfaces between reactor island and balance-of-plant: The nuclear steam suppliers and some architect-engineers have submitted standardized plant designs. The Committee wishes to be sure that adequate attention is devoted to the interface between the reactor island and balance-of-plant to minimize problems during design and construction. The development and use of interdisciplinary system analyses is an aspect of this problem.

2. Assurance of continuous long-term capability of hermetic seals on instrumentation and electrical equipment: The integrity of seals during post-accident conditions may be critical in controlling such an accident. The Committee believes appropriate test and maintenance procedures should be developed to assure long-term reliability.

IID-1 - SAFETY RELATED INTERFACES BETWEEN REACTOR ISLAND AND BALANCE-OF-PLANT

Questions have been raised concerning both standardized balance-of-plant and nuclear steam supply systems on the one hand and custom-designed site-related structures and components on the other hand. The depth of detail required at the stage of Preliminary Design Approval may not be adequate for construction approval. Procedures for instituting quality assurance programs covering design, procurement, construction, and startup with emphasis on timely and appropriate interdisciplinary system analyses to assure functional compatibility across the interfaces as well as for other systems are necessary to assure functional compatibility for the postulated design basis accident conditions.

IID-2 - ASSURANCE OF CONTINUOUS LONG-TERM CAPABILITY OF HERMETIC SEALS ON INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Certain classes of instrumentation incorporate hermetic seals. When safety related components within containment must function during post-LOCA accident conditions, their operability is sensitive to the ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging. The damage processes may fall within Item IB-3, "Performance of Critical Components in Post-LOCA Environment"; however, a special case requiring evaluation has to do with personnel errors in the maintenance of such equipment since such errors could lead to the loss of effective hermetic seals.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 11, 1977

Mr. Lee V. Gossick
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: SHEARON HARRIS NUCLEAR PLANT, INQUIRY REGARDING RESOLUTION OF ACRS GENERIC ITEMS

Dear Mr. Gossick:

The ACRS has been informed by the NRC Staff that, during the Shearon Harris pre-hearing conference on June 19, 1977, Atomic Safety and Licensing Board Member Dr. J. V. Leeds, Jr. requested guidance from the ACRS regarding which items in the Committee list of Generic Items for Light-Water Reactors must be resolved prior to the issuance of a construction permit, and which must be resolved after construction permit issuance, but prior to issuance of an operating license.

The Unresolved Generic Items listed by the ACRS have the following characteristics:

a) They are items of concern to the ACRS for which neither the ultimate solution nor its implementation for reactors in various stages of licensing, construction or operation have yet been determined.

b) They are applicable not only to a given plant or license application but also to a class of plants or, in some cases, to all light-water reactors.

In the ACRS review of a particular application, it may be decided that certain of the Generic Items should be resolved prior to issuance of a construction permit or, more likely, prior to operation of the plant. In such cases, a recommendation to this effect is made specifically in the body of the ACRS letter.

Those Generic Items referred to, and now listed explicitly, in the penultimate paragraph of the ACRS letter, are intended to be considered generically, outside the scope of the particular licensing action. It is the intent that, when solutions are found, a determination will be made by the NRC Staff and the ACRS as to their implementation on all plants for which they are applicable and necessary.

Sincerely yours,

M. Bender
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 15, 1977

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS: REPORT NO. 6

Dear Dr. Hendrie:

The Advisory Committee on Reactor Safeguards has previously reported on the "Status of Generic Items Relating to Light-Water Reactors" in its letters of December 18, 1972, February 13, 1974, March 12, 1975, April 16, 1976 and February 24, 1977. Since the Committee limits its definition of generic items to those cited specifically in its letters pertaining to projects and related matters, the attached listing is not all-inclusive; the Nuclear Regulatory Commission Staff has additional generic items.

Groups I through ID of the attachments are a reiteration of the generic items considered resolved at the time the Committee issued its Report No. 5 on February 24, 1977. Group IE includes those items resolved since February 1977. Following each resolved item is a brief statement of the specific action that resulted in the resolution. Groups II through IID include items previously listed as those for which resolution on a generic basis is still pending. Group IIE includes those added in the present report. The ACRS and the NRC Staff will continue to consider the safety significance of items in Groups II through IIE on a case-by-case basis until generic resolution is reached. Formal actions, such as issuance of Regulations or Regulatory Guides, are anticipated for many of these items.

Owing to questions raised concerning the scope and intent of various generic issues, the Committee has incorporated into the attachments a brief description for all unresolved items cited in this report.

With regard to the status of generic issues, as they apply to each plant, the NRC Staff addresses the status of the pertinent issues in the applicable Safety Evaluation Report. The ACRS identifies those that it believes relevant in its reports on individual projects.

The ACRS has received requests concerning the priorities to be placed on the resolution of outstanding generic issues. Such priorities are shown in Table 1, attached.

"Resolved" as used in the Generic Items reports refers to the following: In some cases an item has been resolved in an administrative sense, recognizing that technical evaluation and satisfactory implementation are yet to be completed. Anticipated Transients Without Scram represents an example of this category. In other instances, the resolution has been accomplished in a narrow or specific sense, recognizing that further steps are desirable, as practical, or that different aspects of the problem require further investigation. Examples are the possibility of improved methods of locating leaks in the primary system, and of improved methods or augmented scope to inservice inspection of reactor pressure vessels.

Sincerely yours,

M. Bender
Chairman

Attachments:
(1) Group I; (2) Group IA; (3) Group IB; (4) Group IC; (5) Group ID; (6) Group IE; (7) Group II; (8) Group IIA; (9) Group IIB; (10) Group IIC; (11) Group IID; (12) Group IIE; and (13) Table 1, Priorities For Resolution of ACRS Generic Items.

GENERIC ITEMS

Group I - Resolved Generic Items

1. Net Positive Suction Head for ECCS Pumps: Covered by Regulatory Guide 1.1.

2. Emergency Power: Covered by Regulatory Guides 1.6, 1.9, and 1.32 and portions of IEEE-308 (1971).

3. Hydrogen Control After a Loss-of-Coolant Accident (LOCA): ACRS concurred in proposed Staff position, covered by NRC Standard Review Plan for Nuclear Power Plants.

4. Instrument Lines Penetrating Containment: Covered by Regulatory Guide 1.11 and Supplement.

5. Strong Motion Seismic Instrumentation: Covered by Regulatory Guide 1.12.

6. Fuel Storage Pool Design Bases: Covered by Regulatory Guide 1.13.

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles: Covered by Regulatory Guide 1.14.

8. Protection Against Industrial Sabotage: Covered by Regulatory Guide 1.17.

9. Vibration Monitoring of Reactor Internals and Primary System: Covered by Regulatory Guide 1.20.

10. Inservice Inspection of Reactor Coolant Pressure Boundary: Covered by ASME Boiler and Pressure Vessel (BPV) Code, Section XI and Regulatory Guide 1.65.

11. Quality Assurance During Design, Construction and Operation: Covered by 10 CFR 50, Appendix B; ASME BPV Code, Section III; ANSI N-45.2-1971, Regulatory Guides 1.28, 1.33, 1.64, 1.70.6 and Proposed Standard ANS-3.2.

12. Inspection of BWR Steam Lines Beyond Isolation Valves: Covered by ASME BPV Code, Section XI.

13. Independent Check of Primary System Stress Analysis: Covered by ASME BPV Code, Section III.

14. Operational Stability of Jet Pumps: Test and operating experience at Dresden 2 and 3 and other jet pump BWRs have satisfied the ACRS concerns.

Group I Continued

15. Pressure Vessel Surveillance of Fluence and NDT Shift: Covered by 10 CFR 50, Appendix A and Appendix H; and ASTM Standard E-185.

16. Nil Ductility Properties of Pressure Vessel Materials: Covered by 10 CFR 50, Appendix A and Appendix G; ASME BPV Code, Section III; "Report on the Integrity of Reactor Vessels for Light-Water Power Reactors," (WASH-1285) by the Advisory Committee on Reactor Safeguards dated January 1974.

17. Operation of Reactor With Less Than All Loops In Service: Covered by ACRS-Regulatory Staff position that manual resetting of several set points on the control room instruments under specific conditions and procedures is acceptable in taking one primary loop out of service. This position is based on the expectation that this mode of operation will be infrequent. Cited in Standard Review Plan Appendix 7-A, Branch Technical Position EICSB 12.

18. Criteria for Preoperational Testing: Covered by Regulatory Guide 1.68.

19. Diesel Fuel Capacity: Covered by ACRS-Regulatory Staff position requiring 7 days fuel (Standard Review Plan 9.5.4).

20. Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends: Covered by ACRS-Regulatory Staff position cited in several letters that such a failure should have no unacceptable consequences.

21. Operating One Plant While Other(s) is/are Under Construction: Specific requirements have been established by ACRS-Regulatory Staff. Covered in Regulatory Guide 1.17, 1.70 Section 13.6.2; 1.101; ANSI N 18.17 and Standard Review Plan 13.3 Appendix A and 13.6.

22. Seismic Design of Steam Lines: Covered by Regulatory Guide 1.29.

23. Quality Group Classifications for Pressure Retaining Components: Covered by Regulatory Guide 1.26.

24. Ultimate Heat Sink: Covered by Regulatory Guide 1.27.

25. Instrumentation to Detect Stresses in Containment Walls: Covered by Regulatory Guide 1.18.

Group IA - Generic Items Resolved Since December 18, 1972

1. Use of Furnace Sensitized Stainless Steel: Covered by Regulatory Guide 1.44.

2. Primary System Detection and Location of Leaks: Covered by Regulatory Guide 1.45.

3. Protection Against Pipe Whip: Covered by Regulatory Guide 1.46.

4. Anticipated Transients Without Scram: Covered by Regulatory Position Document, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, September 1973.

5. ECCS Capability of Current and Older Plants: Covered by Rulemaking as a general policy decision, although acceptable detailed implementation remains to be developed. Docket RM-50-1, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors," December 28, 1973.

Group IB - Generic Items Resolved Since February 13, 1974

1. Positive Moderator Coefficient: PWRs presently have or expect to have zero or negative coefficients. Where some Technical Specifications allow a slightly positive coefficient, the accident and stability analyses take this into account. Burnable poison provisions have been designed into PWRs to reduce otherwise excessive positive coefficients to allowable values.

2. Fixed Incore Detectors on High Power PWRs: Fixed incore detectors are not required for PWRs since reviews of potential power distribution anomalies have not revealed a clear need for continuous incore monitoring.

3. Performance of Critical Components (pumps, cables, etc.) in post-LOCA Environment: Qualification requirements of critical components are now covered by Regulatory Guides 1.40, 1.63, 1.73 and 1.89 and IEEE Standards 382-1972, 383-1974, 317-1972, 323-1974.

4. Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containments: On designs prior to GE Mark III containment, resolution lies in surveillance and testing of vacuum relief valves. For Mark III containments, an additional requirement is that the design be capable of accommodating a bypass equivalent to one square foot for a given flow condition.

5. Emergency Power for Two or More Reactors at the Same Site: Resolved by issue of Regulatory Guide 1.81.

6. Effluents from Light-Water-Cooled-Nuclear Power Reactors: Resolved by issue of Appendix I to 10 CFR 50.

7. Control Rod Ejection Accident: Resolved for PWRs by Regulatory Guide 1.77.

Group IC - Generic Items Resolved Since March 12, 1975

1. Main Steam Isolation Valve Leakage of BWR's: Covered by Regulatory Guide 1.96.

2. Fuel Densification: Covered by 10 CFR 50 Appendix K plus case-by-case review of vendor fuel models.

3. Rod Sequence Control Systems: Covered by NRC Staff Review and Approval of NEDO-10527 and Presentation to ACRS.

4. Seismic Category I Requirements for Auxiliary Systems: Covered by Regulatory Guides 1.26 and 1.29.

Group ID - Generic Items Resolved Since April 16, 1976

1. Instruments to Detect (limited) Fuel Failures - NRC document, "Fuel Failure Detection in Operating Reactors," B. L. Siegel and H. H. Hagen, June, 1976 resolves issue for limited fuel failures, but not for severe failures (See II-4).

2. "Instrumentation to Follow the Course of an Accident" Regulatory Guide 1.97 Revision 1 resolves ACRS concerns.

3. Pressure in Containment Following LOCA - NRC document, "Containment Subcompartment Analysis" September 1976.

4. Fire Protection. Resolved by Branch Technical Position 9.5.1, and Regulatory Guide 1.120.

Group IE - Generic Items Resolved Since February 24, 1977

1. Control Rod Drop Accident (BWRs): Resolved through NRC review and documentation establishing such an event as not having severe consequences (Memorandum for M. Bender, Chairman ACRS, from Denwood F. Ross, Jr., Assistant Director for Reactor Safety, DSS, dated February 11, 1977.)

2. Rupture of High Pressure Lines Outside Containment: Resolved by positions in Standard Review Plan 3.6.1 and 3.6.2.

3. Isolation of Low Pressure from High Pressure Systems: Resolved by positions in Standard Review Plan 5.4.7.

Group II - Resolution Pending

1. Turbine Missiles: Turbine failures for past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discusses the problems.*

2. Effective Operation of Containment Sprays in a LOCA: Extensive documentation in topical reports. Review and evaluation are required.

3. Possible Failure of Pressure Vessel Post-LOCA By Thermal Shock: Regulatory Guide 1.2 covers current information. Ultimate position as to significance of thermal shock requires input of fracture mechanics data from the Heavy Section Steel Technology Program.

**4. Instruments to detect (severe) fuel failures - NRC document, "Fuel Failure Detection in Operating Reactors," B. L. Siegel and H. H. Hagen. Item ID covers limited failures. More work is required for the severe failure case to establish instrumentation criteria.

#5A. Monitoring for Loose Parts Inside the Reactor Pressure Vessel: State-of-the-Art results appear promising and some equipment has been installed.

#5B. Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel: Neutron Noise Analysis has been successful in detecting vibration of some components, however, additional work may be required concerning systems for detecting vibration in other components within the Reactor Pressure Vessel.

#6. Common Mode Failures: This heading covers a multiplicity of diverse components for which requirements should be established. Due to their diversity the ACRS feels that specific items should be separated into subsets under the general heading of common mode failures;

- 6A - Reactor Scram Systems
- 6B - Alternating Current Sources onsite and offsite
- 6C - Direct Current Systems

The above items are easily identified, other specific items may be added to this listing in the future.

---

*Regulatory Guide is in preparation.

**Identified in the Committee's Report of April 16, 1976 as "Instruments to Detect Fuel Failures."

#These are a separation of items included under the same numbers in previous reports.

Group II Continued

7. Behavior of Reactor Fuel Under Abnormal Conditions: This includes: flow blockage; partial melting of fuel assemblies as it affects reactor safety; and transient effects on fuel integrity. The PBF program will address some of these items.

8. BWR Recirculation Pump Overspeed During LOCA: Decision required by ACRS-NRC Staff.

9. The Advisability of Seismic Scram: Further studies required to establish need.

10. Emergency Core Cooling System Capability for Future Plants: Partially resolved by amendments to 10 CFR 50 [50.34(a)(4), 50.34(b)(4), 50.46, and Appendix K]. LOCA evaluation model complete. ACRS feels new cooling approaches should be explored.

II-1 - Turbine Missiles

Turbine failures for the past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discuses the problem.

Three issues require answers to resolve the turbine missile problem: (1) The first relates to the appropriate failure probability value; based on historical failures the probability is about 10^{-4}. Industry predicts a much lower failure probability based on improvements in materials and design. To date the ACRS has accepted the more conservative value; (2) The second issue is strongly dependent on turbine orientation with respect to critical safety structures. Strike probabilities from high angle missiles are acceptably low for single units and may be acceptable for multi-unit plants, depending on plant layout; however, lower angle missiles with non-optimum (tangential) turbine orientation have unacceptably high strike probabilities; (3) The third issue is one of penetration and damage of structures housed in the containment. The limited experimental data pertaining to penetration of large irregularly shaped missiles are not sufficient to determine structural response to impingement of turbine disc segments. Most missile penetration formulas are not relevant to this case. Some experiments with irregular missiles might resolve this issue, particularly for older plants with non-optimum turbine orientations.

II-2 - Effective Operation Of Containment Sprays In a LOCA

Review and evaluation are required of the variety of experiments which have been conducted on the effectiveness of various containment sprays on the removal and retention of airborne radioactive materials anticipated to be present within containment following a LOCA. Such review should consider adequacy of definition of the physical and chemical forms of the anticipated airborne radionuclides, and quality of evaluative tests of the removal efficiencies of various sprays under the conditions of temperature, pressure, and radiation doses expected to exist under LOCA conditions. A desirable extension might be analyses of the use of sprays containing chemicals (such as NaOH) which have the potential for damaging equipment within containment. Studies using other spray additives, such as hydrazine, have been conducted. If compounds, such as this, have distinct advantages, insofar as minimizing equipment damage in the event of inadvertent actuation, action should be taken to encourage their use.

II-3 - Possible Failure Of Pressure Vessel Post-LOCA By Thermal Shock

Earlier nuclear reactor pressure vessels subjected to fluences of $1\text{-}4 \times 10^{19}$ nvt, which are anticipated in the last 20 years of a 40-year life, may suffer severe radiation damage denoted by a pronounced shift in impact transition temperature at the inner surface. There will be a damage gradient which decreases sharply, so that the properties halfway through the wall are essentially those of the as-fabricated material. If a LOCA occurs near end-of-life, the injection of cold water on the region of degraded properties may initiate and propagate a crack because of high local stresses near the surface. Analytic procedures indicate the stresses drop rapidly with distance through the wall so the flaw should not propagate beyond some limiting point. The lack of experimental evidence and the relative width of the error band in the analytic results are such that some experiments are required to validate the analytic model. These are planned under the HSST program.

II-4 - Instruments To Detect (Severe) Fuel Failures

In the event of substantial fuel failure, including the possibility of fuel melt, large amounts of fission products could be rapidly released to the reactor coolant and possibly to the environment. Instrumentation capable of early warning and timely response may avert an incident becoming an accident.

Instrumentation related to such diagnostic purposes for limited fuel failure is being used on most power reactors. (See Item ID-1.) Further work is required to establish criteria for similar instrumentation for severe fuel failures.

II-5A - Monitoring For Loose Parts Inside The Pressure Vessel

Loose parts monitoring can provide early warning of potential mechanical problems or failures within the pressure vessel and throughout the primary coolant circuit. Reactor vendors nave developed monitoring systems; however, requirements remain to be established.

II-5B - Monitoring For Excessive Vibration Inside The Reactor Pressure Vessel

Neutron noise analysis can detect vibration within specific components such as the core barrel. The detection of vibration in other reactor pressure vessel components is less well established.

II-6 - Non-Random Multiple Failures (Formerly "Common Mode Failure")

The term "common mode failures" has, in many instances, come to mean multiple failures of identical components exposed to identical or nearly identical conditions or environments, and the use of diversity in components has been proposed or required to avoid such failures. The concern of the ACRS is better expressed by the term "non-random multiple failures," which is intended to include not only the type of "common mode failure" discussed above but other types of multiple failures for which the consequences and probabilities cannot be predicted by application of the single-failure criterion. Examples include the use of the same sensors or components for both control and protection systems (a resolved matter); sequential multiple failures due to a "domino effect," and simultaneous multiple failures due to a single fault. Since designs usually do not knowingly incorporate features susceptible to such failures, techniques and criteria need to be developed to detect and avoid them in all systems important to safety. The following is a partial listing of systems whose common mode failure has been cited by the ACRS as a matter of safety concern:

II-6A - Scram Systems
II-6B - Alternating Current Sources
II-6C - Direct Current Sources

Other items may be added to this listing in the future.

II-7 - Behavior Of Reactor Fuel Under Abnormal Conditions

The behavior of reactor fuel under abnormal conditions is still considered unresolved due to the limited experimental data available. Partial melting of fuel assemblies due to flow blockage might lead to autocatalytic effects leading to more extensive fuel failure, pressure pulses, etc. Similar behavior might occur in the case of reactivity transients. The ACRS encourages analytic modeling but believes appropriate experimental data are necessary. It is anticipated that tests in the Power Burst Facility (PBF) should supply much of the required data.

II-8 - BWR Pump Overspeed During A LOCA

It is possible for a BWR recirculation pump to overspeed if a large break occurs at the appropriate position in specific piping. Conservative estimates indicate substantial overspeed and possible failure of components, with the generation of missiles. The problem is being approached analytically and experimentally with scaled pumps. The reliability of such protective measures as the use of decouplers between pump and motor is under study.

II-9 - The Advisability Of Seismic Scram

The ACRS has recommended that studies be made of techniques for seismic scram and of the potential safety advantages and potential disadvantages of prompt reactor scram in the event of strong seismic motion, say more than one-half the safe shutdown earthquake. Various suitable techniques have been identified and exist, but thus far only limited studies have been reported on the pros and cons of seismic scram. The principal potential advantage identified arises from the greatly improved coolability of a core in the unlikely event of a seismically induced LOCA, should scram precede the LOCA by several seconds. A principal reason given in opposition to seismic scram relates to a stated interest in keeping power stations on the line to provide power offsite should a severe earthquake occur.

II-10 - ECCS Capability For Future Plants

The ACRS has placed considerable emphasis on ECCS safety R&D so that the extent of the conservatism in the ECCS licensing requirements could be made more precise. With more experimental data a realistic and quantitative appraisal of ECC systems would lead to valid judgments on the changes in licensing which could be put on a firm basis.

Parallel approaches that seek to improve the reliability of ECC systems, to improve the monitoring of low power peaking, and to improve those fuel assembly designs which lower peaking factors, are encouraged. Further, changes in plant design which improve the reflooding of the reactor core should be sought and evaluated.

R&D efforts on analysis of core blowdown and reflood should be increased and combined with the results of the standard problems and the associated experiments. Improved analytical methods would provide a basis for optimized ECCS.

Group IIA - Resolution Pending - Items Since December 18, 1972

1. Ice Condenser Containments: Additional analyses are required to establish response during a LOCA, and to establish design margins.

2. PWR Pump Overspeed During a LOCA: Problem arises in similar manner to that of BWRs (Item 8 Group II).

3. Steam Generator Tube Leakage: Partially resolved by issuance of Regulatory Guide 1.83 which addresses the concern from a preventative point of view.

4. ACRS/NRC Periodic 10-Year Review of all Power Reactors: A more effective, continuous alternative approach to periodic reviews is being proposed. Pending ACRS review, this item is still considered unresolved.

IIA-1 - Ice Condenser Containments

The ice condenser containments have substantially smaller volume on the assumption that the ice will condense the steam during a LOCA, thus preventing system overpressurization. The rate of condensation is critical in the initial stages of the blowdown and is influenced by interaction of vapor with the ice. If the current analyses prove that the condensation model is suitably conservative, the problem may be resolved.

IIA-2 - PWR Pump Overspeed During a LOCA

It is possible for a PWR primary coolant pump to overspeed if a large break occurs at the appropriate position in specific piping. Conservative estimates indicate substantial overspeed and possible failure of components such as flywheels with the generation of missiles. The problem is being approached analytically and experimentally with scaled pumps. The reliability of such protective measures as electrical braking of the pump motor is under study.

IIA-3 - Steam Generator Tube Leakage

Normally the steam generator is not a critical component during a LOCA-ECCS. However, a special case exists where the steam generator tubes have been degraded due to corrosion, wastage, etc. If the shock loads imposed by the LOCA cause a critical number of tubes to fail, say by a double-ended (guillotine) break, the inflow from the secondary side can cause choking of flow during ECCS, preventing adequate cooling of tne core. The critical number of tubes is relatively small. A position, such as one specifying a statistically significant level of nondestructive examination (NDE), might resolve this issue. The purpose of NDE would be to confirm that damage is not excessive; such examinations should minimize the possibility of catastrophic failure of a significant number of tubes.

IIA-4 - Periodic (10-Year) Review Of All Power Reactors

In its report of June 14, 1966, the ACRS recommended that periodic comprehensive reviews be conducted of operating licensed power reactors by the NRC Staff. These reviews would be preceded by a comprehensive report by the operator which evaluated the past experience and the safety of future operation of the plant.

The NRC Staff has maintained a continuing review of the safety of operating plants. In particular, as generic matters of potential safety significance arise, the appropriate operating reactors are asked to assess the relevance of the matter to each particular reactor. This is a necessary but different aspect of the continuing surveillance and review of the safety of operating reactors than was envisaged by the ACRS in its recommendation of June 1966.

The Committee continues to believe both approaches are desirable and awaits the development of a program of periodic comprehensive reviews.

Group IIB - Resolution Pending - Items Added Since February 13, 1974

*1. Computer Reactor Protection System: Systems should be qualified for reliability, particularly through in situ tests and under various environmental conditions, prior to use in reactor system.

2. Qualification of new fuel geometries: The 16x16 and 17x17 PWR, and 8x8 BWR fuels should undergo testing to meet Item 2 in Group IC and Item 7 in Group II.

3. Behavior of BWR Mark III Containments: Various aspects, including vent clearing, vent/coolant interaction, pool swell, pool stratification, pressure loads and flow bypass should be resolved. This is an extension of Item 3 in Group ID.

4. Stress Corrosion Cracking in BWR Piping: Several failures have occurred in operating BWRs. The ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution and extensive programs are underway by industry, ERDA, and NRC.

---

* Identified in the ACRS Report of April 16, 1976 as "Hybrid Reactor Protection System."

IIB-1 - Computer Reactor Protection Systems*

The proposed systems would contain some types of components and subsystems not previously used for reactor protection. It is necessary that the required system reliability, both during normal operation and under postulated abnormal conditions, be established through an appropriate combination of tests and analyses. While the issue originated with the B&W Hybrid concept it is equally applicable to the proposed CE and W computer reactor protection systems.

---

* Identified in the ACRS Report of April 16, 1976 as "Hybrid Reactor Protection System."

IIB-2 - Qualification Of New Fuel Geometries

New fuels proposed for both BWRs and PWRs include the 8x8 (BWR), and 16x16 and 17x17 PWR fuels. The Committee recognizes tnat these fuels are intended to operate at power densities lower tnan earlier fuel designs. However, testing programs are considered necessary to establish their densification behavior (IC-2) as well as their behavior under abnormal conditions (II-7). Appropriate experimental programs should be developed dealing with flow blockage, behavior of fuel after partial melting, and fuel response under transient conditions. It is anticipated that the solution of this item will include a synthesis of Power Burst Facility data, experiments on earlier fuel types, behavior of fuel in commercial reactors, and confirmatory experiments on these fuel designs.

IIB-3 - Behavior Of BWR Mark III Containment

The BWR Mark III Containment differs in many respects from the Mark I and II designs. Various aspects such as vent clearing, vent/coolant interaction, pool swell, pool stratification, pressure loads, and flow bypass must be evaluated and approved; ongoing experimental tests should develop much of the necessary data to confirm the conservatism in design.

IIB-4 - Stress Corrosion Cracking In BWR Piping

Several failures have occurred in operating BWRs. An ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution, and extensive programs are underway by Industry, ERDA and NRC.

The austenitic stainless steels are commonly used as piping material in many of the smaller BWR lines. A combination of weld sensitization, residual stresses, superposed loads, and oxygen equal to or greater than 0.2 ppm in the BWR coolant can lead to cracking, initiating on the inner surface and propagating through the wall. In most cases there will be a leak well before pipe failure so there is adequate warning; however, one can postulate a LOCA caused by a guillotine break with minimal prior warning. Current efforts are to minimize stress corrosion by using other materials.

Group IIC - Resolution Pending - Items Added Since March 12, 1975

1. Locking Out of ECCS Power Operated Valves: The Committee suggests that further attention be given to procedures involving locking out electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS.

2. Design Features to Control Sabotage: Attention should be given to aspects of design that could improve plant security.

*3A. Decontamination of Reactors: As experience is gained in reactor decontamination it should be factored into future plants to optimize control of radioactivity levels.

*3B. Decommissioning of Reactors: Specific plans should be developed, including definitive codes and standards to cover the ultimate decommissioning of plants.

4. Vessel Support Structures: Questions that have arisen concerning the loads on pressure vessel support structures due to certain postulated loss-of-coolant accidents should be resolved.

5. Water Hammer: Several cases of water slugging or water hammer have occurred in both PWRs and BWRs. Corrective measures should be taken to minimize such events.

6. Maintenance and Inspection of Plants: Provisions should be included in the design of future plants which anticipate the maintenance, inspection and operational needs of the plant throughout its service life.

7. Behavior of BWR Mark I Containments: Various aspects relevant to the BWR Mark I Containment should be resolved. Included are such items as relief valve restraint, control of local dynamic loads in the torus, vent clearing and establishment of torus water temperature limits during a LOCA. This is an extension of Item 3 in Group ID.

---

*This is a further separation of the issues identified as IIC-3 in previous reports.

IIC-1 - Locking Out Of ECCS Power-Operated Valves

The physical locking out of electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS has been required, based on the assumption that a spurious electrical signal at an inopportune time could activate the valves to the adverse position; e.g., closed rather than open, or open rather than closed. While such an event has a finite probability another probability exists that the valves might be adversely positioned due to operator error.

The ACRS believes the matter should be studied using a systems approach, and considering such items as: (1) the evaluation of the probability of a spurious signal; (2) time required to reactivate the valve operator; (3) status of signal lights when the circuit breaker is open; (4) the possibility of locking out in an improper position due to a faulty indicator; (5) other designs with improved reliability without lock-out; (6) the advantages and disadvantages of corrective action by an alert operator in case of incorrect positioning vis-a-vis a system with power locked out.

IIC-2 - Design Features To Control Sabotage

Considerable attention has been devoted to control of industrial sabotage of nuclear power plants, particularly with regard to control of unauthorized access, and potential modes of sabotage by individuals or groups external to the operating organization. The ACRS believes that deliberate attention should be given to aspects of design that could improve plant security. With the emphasis being placed on standardized plant designs, it becomes especially important to introduce design measures that could protect against industrial sabotage, or mitigate the consequences thereof.

IIC-3A - Decontamination Of Reactors

The Committee believes that well developed plans, confirmed by appropriate experiments when necessary, should be available for the decontamination of primary reactor systems. At this time the information on full scale decontamination is limited. Examples of potential problems include such items as handling of decontamination solutions, potential hideout of radioactive products, enhanced corrosion and crud formation following decontamination, and the possible incompatibility of the different alloys in the pressure boundary with the decontamination solutions

IIC-3B - Decommissioning Of Reactors

Experience is limited with regard to decommissioning operations, and particularly with rules for dismantling and for mothballing. Definitive plans and standards should be developed covering such items as adequacy of action, problems in restitution of site, mutual responsibility of State and Federal Government, etc.

IIC-4 - Vessel Support Structures

A possible consequence of the instantaneous double-ended pipe break postulated to occur in certain large pipes of PWRs is the asymmetric loading of the reactor pressure vessel support structures. The magnitude and effects of such loads on the pressure vessel should be determined to establish if such loads adversely affect the predicted course of a LOCA. If analysis indicates that the results are unacceptable, appropriate corrective action should be taken. A potential effect is pressure vessel movement due to blowdown jet forces at the location of the rupture, transient differential pressures in the annular region between the vessel and the shield, and transient differential pressures across the core barrel within the reactor vessel.

IIC-6 - Maintenance And Inspection Of Plants

Experience with older plants has verified that appropriate modifications in piping layout, with respect to walls and structures, type of insulation used, and weld joint design, to cite some obvious items, lead to improved maintenance, more reliable inservice inspections, and a better meeting of the operational needs of the plant throughout its service life, including decontamination and eventual decommissioning. An additional benefit is the reduction in personnel exposures in plants, making them more amenable to maintenance and inspection. Appropriate changes should be considered in future designs to meet these criteria.

IIC-7 - Behavior Of BWR Mark I Containments

Recent tests on the BWR Mark I Containment design revealed phenomena not anticipated on the basis of earlier tests where pressure loads were imposed by insertion of air. Specific problems somewnat comparable to those under review for the Mark III Containment, include relief valve discharge, pipe restraints in the torus, local dynamic loads on the torus, vent clearing, and influence of torus temperature on the LOCA.

Ongoing experiments are expected to develop the necessary data to confirm the adequacy of the existing design or to permit necessary modifications.

Group IID - Resolution Pending - Items added since April 16, 1976

1. Safety related interfaces between reactor island and balance-of-plant: The nuclear steam suppliers and some architect-engineers have submitted standardized plant designs. The Committee wishes to be sure that adequate attention is devoted to the interface between the reactor island and balance-of-plant to minimize problems during design and construction. The development and use of interdisciplinary system analyses is an aspect of this problem.

2. Assurance of continuous long-term capability of hermetic seals on instrumentation and electrical equipment: The integrity of seals during post-accident conditions may be critical in controlling such an accident. The Committee believes appropriate test and maintenance procedures should be developed to assure long-term reliability.

IID-1 - Safety Related Interfaces Between Reactor Island And Balance-Of-Plant

Questions have been raised concerning both standardized balance-of-plant and nuclear steam supply systems on the one hand and custom-designed site-related structures and components on the other hand. The depth of detail required at the stage of Preliminary Design Approval may not be adequate for construction approval. Procedures for instituting quality assurance programs covering design, procurement, construction, and startup with emphasis on timely and appropriate interdisciplinary system analyses to assure functional compatibility across the interfaces as well as for other systems, are necessary to assure functional compatibility for the postulated design basis accident conditions.

IID-2 - Assurance Of Continuous Long-Term Capability Of Hermetic Seals On Instrumentation And Electrical Equipment

Certain classes of instrumentation incorporate hermetic seals. When safety related components within containment must function during post-LOCA accident conditions, their operability is sensitive to the ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging. The damage processes may fall within Item IB-3, "Performance of Critical Components in Post-LOCA Environment"; however, a special case requiring evaluation has to do with personnel errors in the maintenance of such equipment since such errors could lead to the loss of effective hermetic seals.

Group IIE - Resolution Pending - Items Added Since February 24, 1977

1. Soil-Structure Interactions: Several matters related to soil-structure interaction and the appropriate seismic response spectrum for use at foundation levels of nuclear plants are under review and reevaluation.

IIE-1 - Soil Structure Interactions

Ongoing studies by the NRC and the industry are reviewing and re-evaluating matters related to soil-structure interaction and to the appropriate seismic response spectrum to be used at the foundation level of a nuclear power plant. These reviews may lead to a modification of current criteria used in the seismic design of foundation structures.

Table 1 - Priorities For Resolution Of ACRS Generic Items

GENERIC ITEMS		RELEVANT TO PWR	RELEVANT TO BWR	PRIORITY FOR RESOLUTION 1/ ACRS	PRIORITY FOR RESOLUTION 1/ NRC
II-1	Turbine Missiles	X	X	A	A-37*
II-2	Containment Sprays	X		B	C-10
II-3	Pressure Vessel Failure By Thermal Shock	X		A	A-11
II-4	Instruments to Detect (Severe) Fuel Failure	X	X	C	-
II-5A	Excessive Vibration	X	X	B	-
II-5B	Loose Parts Monitoring	X	X	B	B-60

1/ The ACRS has adopted and uses the categorizations developed by the NRC Staff and paraphrased below:

A. Those items judged to warrant priority attention in terms of manpower and/or funds to attain early resolution. These items include those, the resolution of which could (1) provide a significant increase in assurance of the health and safety of the public, or (2) have a significant impact upon the reactor licensing process.

B. Those items judged to be important in assuring the health and safety of the public but for which early resolution is not required or which have a lesser safety significance than Category A matters.

(Continued on Page 2)

*The numerals have no significance regarding the priority for resolution but are included to identify the NRC program plans related to each item.

GENERIC ITEMS		RELEVANT TO PWR	RELEVANT TO BWR	PRIORITY FOR RESOLUTION 1/ ACRS	PRIORITY FOR RESOLUTION 1/ NRC
II-6	Non-Random Multiple Failures	X	X	A	C-13
6A	Reactor Scram Systems	X	X	A	A-9
6B	Alternating Current Sources Onsite & Offsite	X	X	A	A-35 B-56 B-57
6C	Direct Current Systems	X	X	A	A-30
II-7	Behavior of Reactor Fuels Under Abnormal Conditions	X	X	A	B-22
II-8	BWR Recirculation Pump Overspeed During LOCA		X	B	B-68
II-9	Seismic Scram	X	X	C	D-1
II-10	ECCS Capability for Future Plants	X	X	A	D-2

Footnote 1 Continued:

C. Those items judged to have little direct or immediate safety significance but which could lead to improved understanding of particular technical issues or refinements in the licensing process.

D. Those items judged not to warrant the expenditure of manpower or funds because little or no importance to safety or improvements to the licensing process can be attributed to the item.

GENERIC ITEMS		RELEVANT TO PWR	RELEVANT TO BWR	PRIORITY FOR RESOLUTION 1/ ACRS	PRIORITY FOR RESOLUTION 1/ NRC
IIA-1	Ice Condenser Containments	X		B	B-54
IIA-2	PWR Pump Overspeed During a LOCA	X		B	B-68
IIA-3	Steam Generator Tube Leakage	X		A	A-3 A-4 A-5
IIA-4	ACRS/NRC Periodic 10-Year Review	X	X	C	Policy*
IIB-1	Computer Reactor Protection System	X		B	A-19
IIB-2	Qualification of New Fuel Geometry	X	X	C	B-22
IIB-3	BWR Mark III Containments		X	B	A-39 B-10
IIB-4	Stress Corrosion Cracking in BWR Piping		X	B	Policy*
IIC-1	Locking Out of ECCS Power Operated Valves	X	X	B	B-8

*The resolution of this item is to be effected through administrative means rather than by a specific technical activity.

GENERIC ITEMS		RELEVANT TO PWR	RELEVANT TO BWR	PRIORITY FOR RESOLUTION 1/ ACRS	PRIORITY FOR RESOLUTION 1/ NRC
IIC-2	Design Features to Control Sabotage	X	X	A	A-29
IIC-3A	Decontamination	X	X	B	A-15
IIC-3B	Decommissioning	X	X	B	B-64
IIC-4	Vessel Support Structures	X		B	A-2
IIC-5	Water Hammer	X	X	A	A-1
IIC-6	Maintenance and Inspection	X	X	B	B-34
IIC-7	BWR Mark I Containments		X	A	A-6 A-7 A-39
IID-1	Interfaces	X	X	A	Policy A-17
IID-2	Capability of Hermetic Seals	X	X	C	C-1
IIE-1	Soil-structure Interaction	X	X	C	A-40 A-41

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 21, 1979

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS: REPORT NO. 7

Dear Dr. Hendrie:

The Advisory Committee on Reactor Safeguards has previously reported on the "Status of Generic Items Relating to Light-Water Reactors" in its letters of December 18, 1972, February 13, 1974, March 12, 1975, April 16, 1976, February 24, 1977 and November 15, 1977. Since the Committee limits its definition of generic items to those cited specifically in its letters pertaining to projects and related matters, the attached listing is not all-inclusive; the Nuclear Regulatory Commission Staff has additional generic items.

In an effort to simplify referencing, the Committee has revised the numbering system for its generic items. (Attachment 4 cross-references this numbering system with that in Report No. 6.) Items 1 through 48 in Attachment 1 are a reiteration of the generic items considered resolved at the time the Committee issued Report No. 6, on November 15, 1977. Items 49 through 52 are those items resolved since November 1977. Following each resolved item is a brief statement of the specific action that resulted in resolution. Items 53 through 77 listed in Attachment 2 are those items previously listed for which resolution on a generic basis is still pending. The ACRS and the NRC Staff will continue to consider the safety significance of these items on a case-by-case basis until generic resolution is reached. Formal actions, such as issuance of Regulations or Regulatory Guides, are anticipated for many of these items.

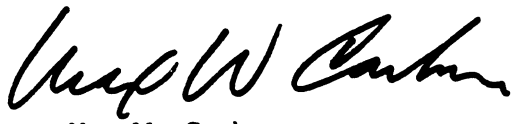
Owing to questions raised concerning the scope and intent of various generic issues, the Committee has included, in Attachment 3, a brief description for all unresolved items cited in this report.

With regard to the status of generic issues, as they apply to each plant, the NRC Staff addresses the status of the pertinent issues in the applicable Safety Evaluation Report. The ACRS identifies those that it believes relevant in its reports on individual projects.

"Resolved" as used in the Generic Items reports refers to the following: In some cases an item has been resolved in an administrative sense, recognizing that technical evaluation and satisfactory implementation are yet to be completed. Anticipated Transients Without Scram represents an example of this category. In other instances, the resolution has been accomplished in a narrow or specific sense, recognizing that further steps are desirable, as practical, or that different aspects of the problem require further investigation. Examples are the possibility of improved methods of locating leaks in the primary system, and of improved methods or augmented scope of in-service inspection of reactor pressure vessels.

The ACRS expects to report to you from time to time on the status of generic items.

Sincerely yours,



Max W. Carbon
Chairman

Attachments:
1. Resolved Generic Items
2. Unresolved Generic Items
3. Descriptions of the Unresolved Generic Items
4. Cross-reference of Numbering System between the Present and Previous Report.

GENERIC ITEMS

Resolved Generic Items

1. Net Positive Suction Head for ECCS Pumps: Covered by Regulatory Guide 1.1.

2. Emergency Power: Covered by Regulatory Guides 1.6, 1.9, and 1.32 and portions of IEEE-308 (1971).

3. Hydrogen Control After a Loss-of-Coolant Accident (LOCA): ACRS concurred in proposed Staff position, covered by NRC Standard Review Plan for Nuclear Power Plants.

4. Instrument Lines Penetrating Containment: Covered by Regulatory Guide 1.11 and Supplement.

5. Strong Motion Seismic Instrumentation: Covered by Regulatory Guide 1.12.

6. Fuel Storage Pool Design Bases: Covered by Regulatory Guide 1.13.

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles: Covered by Regulatory Guide 1.14.

8. Protection Against Industrial Sabotage: Covered by Regulatory Guide 1.17.

9. Vibration Monitoring of Reactor Internals and Primary System: Covered by Regulatory Guide 1.20.

10. In-service Inspection of Reactor Coolant Pressure Boundary: Covered by ASME Boiler and Pressure Vessel (BPV) Code, Section XI and Regulatory Guide 1.65.

11. Quality Assurance During Design, Construction and Operation: Covered by 10 CFR 50, Appendix B; ASME BPV Code, Section III; ANSI N-45.2-1971, Regulatory Guides 1.28, 1.33, 1.64, 1.70.6 and Proposed Standard ANS-3.2.

12. Inspection of BWR Steam Lines Beyond Isolation Valves: Covered by ASME BPV Code, Section XI.

13. Independent Check of Primary System Stress Analysis: Covered by ASME BPV Code, Section III.

14. Operational Stability of Jet Pumps: Test and operating experience at Dresden 2 and 3 and other jet pump BWRs have satisfied the ACRS concerns.

Attachment 1

15. Pressure Vessel Surveillance of Fluence and NDT Shift: Covered by 10 CFR 50, Appendix A and Appendix H; and ASTM Standard E-185.

16. Nil Ductility Properties of Pressure Vessel Materials: Covered by 10 CFR 50, Appendix A and Appendix G; ASME BPV Code, Section III; "Report on the Integrity of Reactor Vessels for Light-Water Power Reactors," (WASH-1285) by the Advisory Committee on Reactor Safeguards dated January 1974.

17. Operation of Reactor With Less Than All Loops In Service: Covered by ACRS-Regulatory Staff position that manual resetting of several set points on the control room instruments under specific conditions and procedures is acceptable in taking one primary loop out of service. This position is based on the expectation that this mode of operation will be infrequent. Cited in Standard Review Plan Appendix 7-A, Branch Technical Position EICSB 12.

18. Criteria for Preoperational Testing: Covered by Regulatory Guide 1.68.

19. Diesel Fuel Capacity: Covered by ACRS-Regulatory Staff position requiring 7 days fuel (Standard Review Plan 9.5.4).

20. Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends: Covered by ACRS-Regulatory Staff position cited in several letters that such a failure should have no unacceptable consequences.

21. Operating One Plant While Other(s) is/are Under Construction: Specific requirements have been established by ACRS-Regulatory Staff. Covered in Regulatory Guide 1.17, 1.70 Section 13.6.2; 1.101; ANSI N 18.17 and Standard Review Plan 13.3 Appendix A and 13.6.

22. Seismic Design of Steam Lines: Covered by Regulatory Guide 1.29.

23. Quality Group Classifications for Pressure Retaining Components: Covered by Regulatory Guide 1.26.

24. Ultimate Heat Sink: Covered by Regulatory Guide 1.27.

25. Instrumentation to Detect Stresses in Containment Walls: Covered by Regulatory Guide 1.18.

26. Use of Furnace Sensitized Stainless Steel: Covered by Regulatory Guide 1.44.

27. Primary System Detection and Location of Leaks: Covered by Regulatory Guide 1.45.

28. Protection Against Pipe Whip: Covered by Regulatory Guide 1.46.

29. Anticipated Transients Without Scram: Covered by Regulatory Position Document, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, September 1973.

30. ECCS Capability of Current and Older Plants: Covered by Rulemaking as a general policy decision, although acceptable detailed implementation remains to be developed. Docket RM-50-1, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors," December 28, 1973.

31. Positive Moderator Coefficient: PWRs presently have or expect to have zero or negative coefficients. Where some Technical Specifications allow a slightly positive coefficient, the accident and stability analyses take this into account. Burnable poison provisions have been designed into PWRs to reduce otherwise excessive positive coefficients to allowable values.

32. Fixed Incore Detectors on High Power PWRs: Fixed incore detectors are not required for PWRs since reviews of potential power distribution anomalies have not revealed a clear need for continuous incore monitoring.

33. Performance of Critical Components (pumps, cables, etc.) in post-LOCA Environment: Qualification requirements of critical components are now covered by Regulatory Guides 1.40, 1.63, 1.73 and 1.89 and IEEE Standards 382-1972, 383-1974, 317-1972, 323-1974.

34. Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containments: On designs prior to GE Mark III containment, resolution lies in surveillance and testing of vacuum relief valves. For Mark III containments, an additional requirement is that the design be capable of accommodating a bypass equivalent to one square foot for a given flow condition.

35. Emergency Power for Two or More Reactors at the Same Site: Resolved by issue of Regulatory Guide 1.81.

36. Effluents from Light-Water-Cooled-Nuclear Power Reactors: Resolved by issue of Appendix I to 10 CFR 50.

37. Control Rod Ejection Accident: Resolved for PWRs by Regulatory Guide 1.77

38. Main Steam Isolation Valve Leakage of BWR's: Covered by Regulatory Guide 1.96.

39. Fuel Densification: Covered by 10 CFR 50 Appendix K plus case-by-case review of vendor fuel models.

40. Rod Sequence Control Systems: Covered by NRC Staff Review and Approval of NEDO-10527 and Presentation to ACRS.

41. Seismic Category I Requirements for Auxiliary Systems: Covered by Regulatory Guides 1.26 and 1.29.

42. Instruments to Detect (limited) Fuel Failures - NRC document, "Fuel Failure Detection in Operating Reactors," B.L. Siegel and H. H. Hagen, June, 1976 resolves issue for limited fuel failures, but not for severe failures (See Item 56).

43. "Instrumentation to Follow the Course of an Accident" Regulatory Guide 1.97 Revision 1 resolves ACRS concerns.

44. Pressure in Containment Following LOCA - NRC document, "Containment Subcompartment Analysis" September 1976.

45. Fire Protection. Resolved by Branch Technical Position 9.5.1, and Regulatory Guide 1.120.

46. Control Rod Drop Accident (BWRs): Resolved through NRC review and documentation establishing such an event as not having severe consequences (memorandum for M. Bender, Chairman ACRS, from Denwood F. Ross, Jr., Assistant Director for Reactor Safety, DSS, dated February 11, 1977.)

47. Rupture of High Pressure Lines Outside Containment: Resolved by positions in Standard Review Plan 3.6.1 and 3.6.2.

48. Isolation of Low Pressure from High Pressure Systems: Resolved by positions in Standard Review Plan 5.4.7.

49. Monitoring For Loose Parts Inside The Reactor Pressure Vessel: Resolved by Staff position to be documented in Regulatory Guide 1.133.

50. Qualification of New Fuel Geometries: Resolved by position in Standard Review Plan 4.2, Revision 1.

51. Maintenance and Inspection of Plants: Resolved by the amount of Staff attention and industry involvement documented in Memorandum for Larry P. Crocker, Technical Assistant to the Director, DPM, from William E. Kreger, Acting Assistant Director for Site Analysis, DSE subject: Resolution of ACRS Generic Item II C-6 dated February 28, 1979.

52. Safety Related Interfaces Between Reactor Island and Balance of Plant: Resolved by position in Standard Review Plan 1.8.

Resolution Pending

53. Turbine Missiles: Turbine failures for past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discusses the problems. (1)

54. Effective Operation of Containment Sprays in a LOCA: Extensive documentation in topical reports. Review and evaluation required.

55. Possible Failure of Pressure Vessel Post-LOCA By Thermal Shock: Regulatory Guide 1.2 covers current information. Ultimate position as to significance of thermal shock requires input of fracture mechanics data from the Heavy Section Steel Technology Program.

56. Instruments to Detect (Severe) Fuel Failures - NRC document, "Fuel Failure Detection in Operating Reactors," B. L. Siegel and H. H. Hagen. Item 42 covers limited failures. More work is required for the severe failure case to establish instrumentation criteria. (2)

57. Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel: Neutron Noise Analysis has been successful in detecting vibration of some components; however, additional work may be required concerning systems for detecting vibration in other components within the Reactor Pressure Vessel.

58. Non-Random Multiple Failures: This heading covers a multiplicity of diverse components for which requirements should be established. Due to their diversity, the ACRS feels that specific items should be separated into subsets under the general heading of non-random multiple failures;

    58A - Reactor Scram Systems
    58B - Alternating Current Sources onsite and offsite
    58C - Direct Current Sources

    The above items are easily identified, other specific items may be added to this listing in the future.

---

(1) Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missles," will be modified to cover both low and high trajectory missiles.

(2) Identified in the Committee's Report of April 16, 1976 as "Instruments to Detect Fuel Failures."

Attachment 2

59. Behavior of Reactor Fuel Under Abnormal Conditions: This includes: flow blockage; partial melting of fuel assemblies as it affects reactor safety; and transient effects on fuel integrity. The PBF program will address some of these items.

60. BWR Recirculation Pump and PWR Primary Coolant Pump Overspeed During LOCA: Decision required by ACRS-NRC Staff. (3)

61. The Advisability of Seismic Scram: Further studies required to establish need.

62. Emergency Core Cooling System Capability for Future Plants: Partially resolved by amendments to 10 CFR 50 [50.34(a)(4), 50.34(b)(4), 50.46, and Appendix K]. LOCA evaluation model complete. ACRS feels new cooling approaches should be explored.

63. Ice Condenser Containments: Additional analyses are required to establish response during a LOCA, and to establish design margins.

64. Steam Generator Tube Leakage: Partially resolved by issuance of Regulatory Guide 1.83 which addresses the concern from a preventative point of view.

65. ACRS/NRC Periodic 10-Year Review of all Power Reactors: A more effective, continuous alternative approach to periodic reviews is being proposed. Pending ACRS review, this item is still considered unresolved.

66. Computer Reactor Protection System: Systems should be qualified for reliability, particularly through in situ tests and under various environmental conditions, prior to use in reactor system. (4)

67. Behavior of BWR Mark II Containments: Various aspects, including vent clearing, vent/coolant interaction, pool swell, pool stratification, pressure loads and flow bypass should be resolved. This is an extension of Item 44.

---

(3) Item 60 combines two previous items which dealt with PWR and BWR pump overspeed separately.

(4) Identified in the ACRS Report of April 16, 1976 as "Hybrid Reactor Protection System."

68. Stress Corrosion Cracking in BWR Piping: Several failures have occurred in operating BWRs. The ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution and extensive programs are underway by industry, ERDA, and NRC.

69. Locking Out of ECCS Power Operated Valves: The Committee suggests that further attention be given to procedures involving locking out electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS.

70. Design Features to Control Sabotage: Attention should be given to aspects of design that could improve plant security.

71. Decontamination of Reactors: As experience is gained in reactor decontamination it should be factored into future plants to optimize control of radioactivity levels.

72. Decommissioning of Reactors: Specific plans should be developed, including definitive codes and standards to cover the ultimate decommissioning of plants.

73. Vessel Support Structures: Questions that have arisen concerning the loads on pressure vessel support structures due to certain postulated loss-of-coolant accidents should be resolved.

74. Water Hammer: Several cases of water slugging or water hammer have occurred in both PWRs and BWRs. Corrective measures should be taken to minimize such events.

75. Behavior of BWR Mark I Containments: Various aspects relevant to the BWR Mark I Containment should be resolved. Included are such items as relief valve restraint, control of local dynamic loads in the torus, vent clearing and establishment of torus water temperature limits during a LOCA. This is an extension of Item 44.

76. Assurance of Continuous Long-term Capability of Hermetic Seals on Instrumentation and Electrical Equipment: The integrity of seals during post-accident conditions may be critical in controlling such an accident. The Committee believes appropriate test and maintenance procedures should be developed to assure long-term reliability.

77. Soil-Structure Interactions: Several matters related to soil-structure interaction and the appropriate seismic response spectrum for use at foundation levels of nuclear plants are under review and reevaluation.

53 Turbine Missiles

Turbine failures for the past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discuses the problem.

Three issues require answers to resolve the turbine missile problem: (1) The first relates to the appropriate failure probability value; based on historical failures the probability is about 10^{-4}/turbine-year. Industry predicts a much lower failure probability based on improvements in materials and design. To date the ACRS has accepted the more conservative value; (2) The second issue is strongly dependent on turbine orientation with respect to critical safety structures. Strike probabilities from high angle missiles are acceptably low for single units and may be acceptable for multi-unit plants, depending on plant layout; however, lower angle missiles with non-optimum (tangential) turbine orientation have unacceptably high strike probabilities; (3) The third issue is one of penetration and damage of structures housed in the containment. The limited experimental data pertaining to penetration of large irregularly shaped missiles are not sufficient to determine structural response to impingement of turbine disc segments. Most missile penetration formulas are not relevant to this case. The EPRI turbine missile impact experiments might resolve this issue, particularly for older plants with non-optimum turbine orientations.

Attachment 3

54 - Effective Operation of Containment Sprays in a LOCA

Review and evaluation are required of the variety of experiments which have been conducted on the effectiveness of various containment sprays for the removal and retention of airborne radioactive materials anticipated to be present within containment following a LOCA. Such review should consider adequacy of definition of the physical and chemical forms of the anticipated airborne radionuclides, and quality of evaluative tests of the removal efficiencies of various sprays under the conditions of temperature, pressure, and radiation doses expected to exist under LOCA conditions. A desirable extension might be analyses of the use of sprays containing chemicals (such as NaOH) which have the potential for damaging equipment within containment. Studies using other spray additives, such as hydrazine, have been conducted. If compounds, such as this, have distinct advantages, insofar as minimizing equipment damage in the event of inadvertent actuation, action should be taken to encourage their use.

55 - Possible Failure of Pressure Vessel Post-LOCA By Thermal Shock

Earlier nuclear reactor pressure vessels subjected to fluences of $1\text{-}4 \times 10^{19}$ nvt, which are anticipated in the last 20 years of a 40-year life, may suffer severe radiation damage denoted by a pronounced shift in impact transition temperature at the inner surface. There will be a damage gradient which decreases sharply, so that the properties halfway through the wall are essentially those of the as-fabricated material. If a LOCA occurs near end-of-life, the injection of cold water on the region of degraded properties may initiate and propagate a crack because of high local stresses near the surface. Analytic procedures indicate the stresses drop rapidly with distance through the wall so the flaw should not propagate beyond some limiting point. The lack of experimental evidence and the relative width of the error band in the analytic results are such that some experiments are required to validate the analytic model. These are under way in the HSST program.

56 - Instruments To Detect (Severe) Fuel Failures

In the event of substantial fuel failure, including the possibility of fuel melt, large amounts of fission products could be rapidly released to the reactor coolant and possibly to the environment. Instrumentation capable of early warning and timely response may avert an incident becoming an accident.

Instrumentation related to such diagnostic purposes for limited fuel failure is being used on most power reactors (see Item 42). Further work is required to establish criteria for similar instrumentation for severe fuel failures.

57 - Monitoring For Excessive Vibration Inside The Reactor Pressure Vessel

Neutron noise analysis can detect vibration of specific components such as the core barrel. The detection of vibration in other reactor pressure vessel components is less well established.

58 - Non-Random Multiple Failures (Formerly "Common Mode Failure")

The term "common mode failures" has, in many instances, come to mean multiple failures of identical components exposed to identical or nearly identical conditions or environments, and the use of diversity in components has been proposed or required to avoid such failures. The concern of the ACRS is better expressed by the term "non-random multiple failures," which is intended to include not only the type of "common mode failure" discussed above but other types of multiple failures for which the consequences and probabilities cannot be predicted by application of the single-failure criterion. Examples include the use of the same sensors or components for both control and protection systems (a resolved matter); sequential multiple failures due to a "domino effect," and simultaneous multiple failures due to a single fault. Since designs usually do not knowingly incorporate features susceptible to such failures, techniques and criteria need to be developed to detect and avoid them in all systems important to safety. The following is a partial listing of systems whose common mode failure has been cited by the ACRS as a matter of safety concern:

58A - Scram Systems
58B - Alternating Current Sources
58C - Direct Current Sources

Other items may be added to this listing in the future.

59 - Behavior Of Reactor Fuel Under Abnormal Conditions

The behavior of reactor fuel under abnormal conditions is still considered unresolved due to the limited experimental data available. Partial melting of fuel assemblies due to flow blockage might lead to autocatalytic effects leading to more extensive fuel failure, pressure pulses, etc. Similar behavior might occur in the case of reactivity transients. The ACRS encourages analytic modeling but believes appropriate experimental data are necessary. It is anticipated that tests in the Power Burst Facility (PBF) should supply much of the required data.

60 - BWR and PWR Pump Overspeed During A LOCA

It is possible for a BWR recirculation pump or a PWR primary coolant pump to overspeed if a large break occurs at the appropriate position in specific piping. Conservative estimates indicate substantial overspeed and possible failure of components, with the generation of missiles. The problem is being approached analytically and experimentally with scaled pumps. The reliability of such protective measures as the use of decouplers between pump and motor is under study for BWRs. In PWRs the reliability of such protective measures as electrical braking of the pump motor is under study.

61 - The Advisability Of Seismic Scram

The ACRS has recommended that studies be made of techniques for seismic scram and of the potential safety advantages and potential disadvantages of prompt reactor scram in the event of strong seismic motion, say more than one-half the safe shutdown earthquake. Various suitable techniques have been identified and exist, but thus far only limited studies have been reported on the pros and cons of seismic scram. The principal potential advantage identified arises from the greatly improved coolability of a core in the unlikely event of a seismically induced LOCA, should scram precede the LOCA by several seconds. A principal reason given in opposition to seismic scram relates to a stated interest in keeping power stations on the line to provide power offsite should a severe earthquake occur.

62 - ECCS Capability For Future Plants

The ACRS has placed considerable emphasis on ECCS safety R&D so that the extent of the conservatism in the ECCS licensing requirements could be made more precise. With more experimental data a realistic and quantitative appraisal of ECC systems would lead to valid judgments on changes in licensing which could be put on a firm basis.

Parallel approaches that seek to improve the reliability of ECC systems, to improve the monitoring of low power peaking, and to improve those fuel assembly designs by achieving lower peaking factors, are encouraged. Further, changes in plant design which improve the reflooding of the reactor core should be sought and evaluated.

R&D efforts on analysis of core blowdown and reflood should be pursued and combined with the results of the standard problems and the associated experiments. Improved analytical methods would provide a basis for optimized ECCS.

63 - Ice Condenser Containments

The ice condenser containments have substantially smaller volume on the assumption that the ice will condense the steam during a LOCA, thus preventing system overpressurization. The rate of condensation is critical in the initial stages of the blowdown and is influenced by interaction of vapor with the ice. If the current analyses prove that the condensation model is suitably conservative, the problem may be resolved.

64 - Steam Generator Tube Leakage

Normally the steam generator is not a critical component during a LOCA-ECCS. However, a special case exists where the steam generator tubes have been degraded due to corrosion, wastage, etc. If the shock loads imposed by the LOCA cause a critical number of tubes to fail, say by a double-ended (guillotine) break, the inflow from the secondary side can cause choking of flow during ECC, preventing adequate cooling of the core. The critical number of tubes is relatively small. A position, such as one specifying a statistically significant level of nondestructive examination (NDE), might resolve this issue. The purpose of NDE would be to confirm that damage is not excessive; such examinations should minimize the possibility of catastrophic failure of a significant number of tubes.

65 - Periodic (10-Year) Review Of All Power Reactors

In its report of June 14, 1966, the ACRS recommended that periodic comprehensive reviews be conducted of operating licensed power reactors by the NRC Staff. These reviews would be preceded by a comprehensive report by the operator which evaluate the past experience and the safety of future operation of the plant.

The NRC Staff has maintained a continuing review of the safety of operating plants. In particular, as generic matters of potential safety significance arise, the appropriate operating reactors are asked to assess the relevance of the matter to each particular reactor. This is a necessary but different aspect of the continuing surveillance and review of the safety of operating reactors than was envisaged by the ACRS in its recommendation of June 1966.

The Committee continues to believe both approaches are desirable and awaits the development of a program of periodic comprehensive reviews.

66 - Computer Reactor Protection Systems*

The proposed systems would contain some types of components and subsystems not previously used for reactor protection. It is necessary that the required system reliability, both during normal operation and under postulated abnormal conditions, be established through an appropriate combination of tests and analyses. While the issue originated with the B&W Hybrid concept it is equally applicable to the proposed CE and W computer reactor protection systems.

---

* Identified in the ACRS Report of April 16, 1976 as "Hybrid Reactor Protection System."

67 - Behavior Of BWR Mark III Containments

The BWR Mark III Containment differs in many respects from the Mark I and II designs. Various aspects such as vent clearing, vent/coolant interaction, pool swell, pool stratification, pressure loads, and flow bypass must be evaluated and approved; ongoing experimental tests should develop much of the necessary data to confirm the conservatism in design.

68 - Stress Corrosion Cracking in BWR Piping

Several failures have occurred in operating BWRs. An ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution, and extensive programs are underway by Industry, ERDA and NRC.

The austenitic stainless steels are commonly used as piping material in many BWR lines. A combination of weld sensitization, residual stresses, superposed loads, and oxygen equal to or greater than 0.2 ppm in the BWR coolant can lead to cracking, initiating on the inner surface and propagating through the wall. In most cases there will be a leak well before pipe failure so there is adequate warning; however, one can postulate a LOCA caused by a guillotine break with minimal prior warning. Current efforts are to minimize stress corrosion by using other materials.

69 - Locking Out Of ECCS Power Operated Valves

The physical locking out of electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS has been required, based on the assumption that a spurious electrical signal at an inopportune time could activate the valves to the adverse position; e.g., closed rather than open, or open rather than closed. While such an event has a finite probability another probability exists that the valves might be adversely positioned due to operator error.

The ACRS believes the matter should be studied using a systems approach, and considering such items as: (1) the evaluation of the probability of a spurious signal; (2) time required to reactivate the valve operator; (3) status of signal lights when the circuit breaker is open; (4) the possibility of locking out in an improper position due to a faulty indicator; (5) other designs with improved reliability without lock-out; (6) the advantages and disadvantages of corrective action by an alert operator in case of incorrect positioning vis-a-vis a system with power locked out.

70 - Design Features To Control Sabotage

Considerable attention has been devoted to control of industrial sabotage of nuclear power plants, particularly with regard to control of unauthorized access, and potential modes of sabotage by individuals or groups external to the operating organization. The ACRS believes that deliberate attention should be given to aspects of design that could improve plant security. With the emphasis being placed on standardized plant designs, it becomes especially important to introduce design measures that could protect against industrial sabotage, or mitigate the consequences thereof.

71 - Decontamination Of Reactors

The Committee believes that well developed plans, confirmed by appropriate experiments when necessary, should be available for the decontamination of primary reactor systems. At this time the information on full scale decontamination is limited. Examples of potential problems include such items as handling of decontamination solutions, potential hideout of radioactive products, enhanced corrosion and crud formation following decontamination, and the possible incompatibility of the different alloys in the pressure boundary with the decontamination solutions.

72 - Decommissioning of Reactors

Experience is limited with regard to decommissioning operations, and particularly with rules for dismantling and for mothballing. Definitive plans and standards should be developed covering such items as adequacy of action, problems in restitution of site, mutual responsibility of State and Federal Government, etc.

73 - Vessel Support Structures

A possible consequence of the instantaneous double-ended pipe break postulated to occur in certain large pipes of PWRs is the asymmetric loading of the reactor pressure vessel support structures. The magnitude and effects of such loads on the pressure vessel should be determined to establish if such loads adversely affect the predicted course of a LOCA. If analysis indicates that the results are unacceptable, appropriate corrective action should be taken. A potential effect is pressure vessel movement due to blowdown jet forces at the location of the rupture, transient differential pressures in the annular region between the vessel and the shield, and transient differential pressures across the core barrel within the reactor vessel.

74 - Water Hammer

Several instances of water slugging or water hammer have occurred in both BWRs and PWRs due to causes such as the trapping of water between two valves. This slug of water is accelerated by steam or water once the valves are opened. The stored energy is sufficient to damage piping, bend or break pipe restraints, and damage support structures. Water hammer may occur due to flow instabilities in steam generators in conjunction with water flowing into the feedwater inlets, resulting in comparable damage.

Corrective measures should be taken to minimize such occurrences after completion of analytic and experimental studies directed to an understanding of the causes.

75 - Behavior Of BWR Mark I Containments

Recent tests on the BWR Mark I Containment design revealed phenomena not anticipated on the basis of earlier tests where pressure loads were imposed by insertion of air. Specific problems, somewhat comparable to those under review for the Mark III Containment, include relief valve discharge, pipe restraints in the torus, local dynamic loads on the torus, vent clearing, and influence of torus temperature on the LOCA.

Ongoing experiments are expected to develop the necessary data to confirm the adequacy of the existing design or to permit necessary modifications.

76 - Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

Certain classes of instrumentation incorporate hermetic seals. When safety related components within containment must function during post-LOCA accident conditions, their operability is sensitive to the ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging. The damage processes may fall within Item 33, "Performance of Critical Components in Post-LOCA Environment"; however, a special case requiring evaluation has to do with personnel errors in the maintenance of such equipment since such errors could lead to the loss of effective hermetic seals.

77 - Soil-Structure Interactions

Ongoing studies by the NRC and the industry are reviewing and re-evaluating matters related to soil-structure interaction and to the appropriate seismic response spectrum to be used at the foundation level of a nuclear power plant. These reviews may lead to a modification of current criteria used in the seismic design of foundation structures.

Cross-Reference of Numbering System
Between Present Report and the Previous Report*

Present	Previous
1	I-1
2	I-2
3	I-3
4	I-4
5	I-5
6	I-6
7	I-7
8	I-8
9	I-9
10	I-10
11	I-11
	I-12
13	I-13
14	I-14
15	I-15
16	I-16
17	I-17
18	I-18
19	I-19
20	I-20
21	I-21
22	I-22
23	I-23
24	I-24
25	I-25
26	IA-1
27	IA-2
28	IA-3
29	IA-4
30	IA-5
31	IB-1
32	IB-2
33	IB-3
34	IB-4
35	IB-5
36	IB-6
37	IB-7
38	IC-1
39	IC-2
40	IC-3
41	IC-4
42	ID-1
43	ID-2
44	ID-3
45	ID-4
46	IE-1
47	IE-2
48	IE-3
49	II-5A
50	IIB-2
51	IIC-6
52	IID-1
53	II-1
54	II-2
55	II-3
56	II-4
49	II-5A
57	II-5B
58	II-6
59	II-7
60	II-8 & IIA-2
61	II-9
62	II-10
63	IIA-1
60	IIA-2
64	IIA-3
65	IIA-4
66	IIB-1
50	IIB-2
67	IIB-3
68	IIB-4
69	IIC-1
70	IIC-2
71	IIC-3A
72	IIC-3B
73	IIC-4
74	IIC-5
51	IIC
75	IIC-7
52	IID-1
76	IID-2
77	IIE-1

Status of Generic Items Relating to Light-Water Reactor Rpt. No. 6, 11/15/77.

Attachment 4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 12, 1980

Honorable John F. Ahearne
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: NEW UNRESOLVED SAFETY ISSUES

Dear Dr. Ahearne:

During its 244th meeting, August 7-9, 1980, the ACRS discussed with the NRC Staff their selection of new Unresolved Safety Issues.

We agree that the items suggested by the Staff deserve the priority of study that they will receive if they are classified as Unresolved Safety Issues. In addition, we believe the following should be added to the list.

1. DC Power Supply Reliability - This issue is currently being addressed and may be resolved in the near future, but it should be carried as unresolved until resolution is clearly achieved.

2. Single Failure Criterion - Many current safety evaluations use the single failure criterion as a measure of reliability. Its inadequacy is widely recognized. It should be replaced, where feasible, with criteria that consider the possible contributions to risk of multiple failures.

3. Control System Reliability - Recent experience has indicated that more attention must be given to reactor control system reliability. Most safety analyses in the past have given minimum attention to control system reliability based partly on the assumption that failure of the system makes it unavailable and ignores the fact that this failure may actually produce an unsafe mode of reactor behavior. This problem should receive further study to determine appropriate reliability standards for control systems. Appropriate reliability of nonsafety system information displayed for use of the reactor operator is a related important issue.

We believe there are two potential problems with the Staff's method of choosing candidate items for the Unresolved Safety Issues list. First, because of the manner by which items must be sponsored by specific units of the Staff, the procedure may tend to miss important problems which are complex and not yet clearly defined. Second, the possibility that a problem may be resolved in six

months does not mean that it will be resolved and should not be grounds for its exclusion from the list. Assignment of such an item to Unresolved Safety Issues status may make its resolution more probable.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

Reference:

U.S. Nuclear Regulatory Commission Staff Paper, "Special Report to Congress Identifying New Unresolved Safety Issues," SECY-80-325, dated July 9, 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 17, 1981

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS

Dear Dr. Hendrie:

Starting more than a decade ago, the Advisory Committee on Reactor Safeguards developed a list of generic items related to light-water reactors. This grew into a list which was last reported in our letter of March 21, 1979. The work of the Staff to resolve these, as well as items from their list of generic items, has steadily become more formal, stemming more recently from the requirements of Section 210 of the Energy Reorganization Act of 1974 as amended.

The Committee believes the Staff's Task Action Plans for Category A generic activities and the program for selecting and resolving Unresolved Safety Issues (USIs) have now progressed to the point where it would be advantageous to abolish our separate list of generic items and work with the Staff's list. The feasibility of this plan was discussed with the Staff at a Subcommittee meeting on March 11, 1981; agreement was reached that such a change would be advantageous.

The procedures the Staff has developed for handling USIs adequately allows for ACRS input to the selection of new issues, for following progress, and for reviewing the resolution of generic items. We understand that the Staff plans to enlarge the scope of their program to prioritize their list of remaining generic items (other than USIs) and lay out programs to resolve these items as resources become available. Our own list is adequately covered by the Staff's list and our priorities closely parallel theirs.

We will no longer maintain a separate list of generic items or reference items from such a list in our reports on applications for Construction Permits or Operating Licenses. We will, from time to time, recommend additional items for the list of generic issues maintained by the Staff, and expect to review the resolutions of such issues as they are achieved

or proposed. We will continue to advise the Commission regarding the identification and resolution of USIs.

Sincerely,

J. Carson Mark
Chairman

References:

1. NUREG-0606, "Unresolved Safety Issues Summary, Aqua Book," Vol. 3, No. 1, dated February 13, 1981.
2. NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants," dated February 1980.
3. Letter to J. Hendrie, Chairman, USNRC, from M. W. Carbon, Chairman, ACRS, Subject: Status of Generic Items Relating to Light-Water Reactors: Report No. 7, dated March 21, 1979.
4. NUREG-0371, "Task Action Plans for Generic Activities, Category A," dated November 1978.
5. NUREG-0471, "Generic Task Problem Descriptions, Category B, C, and D Tasks," dated June 1978.
6. NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," dated January 1, 1978.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

January 11, 1983

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE PROPOSED GENERIC SAFETY ISSUES PRIORITIZATION METHODOLOGY

During its 273rd meeting, January 6-8, 1983, the ACRS reviewed the methodology described in NUREG-0933 that is proposed to aid in establishing priorities for resolving generic safety issues. This matter was also discussed by the ACRS Subcommittee on Generic Items during the Subcommittee meetings on December 8, 1982 and January 5, 1983.

The approach described in NUREG-0933 seems logical. However, very large uncertainties in the application of the approach exist. These uncertainties are primarily in: probabilities related to a sequence of events associated with a specific safety issue and their consequences, and projected methods for and costs of resolving the safety issue (staff resources, research, licensee resources, implementation time, etc.). If this approach is to be used responsibly to allocate NRC and licensee/applicant resources, the data on which judgments are made must be adequately validated. In addition, it is essential that internal peer review be employed effectively at appropriate stages, including the initial application of the value-impact score formula described in NUREG-0933.

Because of the above mentioned uncertainties, we believe that considerations other than those treated in the approach described in NUREG-0933 may have a major influence on the final priority grading of many of the generic safety issues. We will continue to review and comment on the adequacy of the methodology and its application to the individual generic safety issues at appropriate intervals.

Sincerely,

J. J. Ray
Chairman

Reference:
U.S. Nuclear Regulatory Commission, "A Prioritization of Generic Safety Issues," NUREG-0933, Revision 0 (Draft), dated November 10, 1982.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 20, 1983

MEMORANDUM FOR: W. J. Dircks
Executive Director for Operations

FROM: R. F. Fraley, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: ACRS COMMENTS ON THE PRIORITIZATION OF GENERIC ISSUES

During November 1982, the NRC Staff requested that the ACRS review the following with regard to the prioritization of generic issues:

- Adequacy of the numerical methodology, described in Draft NUREG-0933, that was used in prioritizing generic issues.
- Adequacy of the application of the methodology to individual generic issues.

During its 273rd meeting, January 6-8, 1983, the ACRS completed its review of the adequacy of the numerical methodology and transmitted its comments to Chairman Palladino in a letter dated January 11, 1983.

During its 278th meeting, June 9-11, 1983, the ACRS performed a partial review of the adequacy of the application of the methodology to individual generic issues. Results of this partial review are contained in the following attachments:

- Attachment 1 lists those items for which the ACRS agrees with the priority rankings proposed by the NRC Staff.
- Attachment 2 includes a list of items for which the ACRS agrees with the priority rankings proposed by the NRC Staff, but has comments.
- Attachment 3 contains a list of items for which the ACRS disagrees with the NRC Staff's proposed priority rankings along with the reasons therefor.

The ACRS will continue its review of the adequacy of the proposed priority rankings for the remaining generic issues and will provide additional comments as they become available.

It is requested that the NRC Staff provide written responses to the ACRS comments identified in Attachments 2 and 3. The ACRS may review the adequacy of the NRC Staff's responses during one of its future meetings.

Attachments: As Stated

June 20, 1983

ATTACHMENT 1

LIST OF ITEMS FOR WHICH THE

ACRS AGREES WITH THE PRIORITY RANKINGS PROPOSED

BY THE NRC STAFF

New Generic Issues	Title
2	Failure of Protective Devices on Essential Equipment
3	Set Point Drift in Instrumentation
5	Design Check and Audit of Balance-of-Plant Equipment
6	Separation of Control Rod from Its Drive and BWR High Rod Worth Events
11	Turbine Disc Cracking
12	BWR Jet Pump Integrity
14	PWR Pipe Cracks
15	Radiation Effects on Reactor Vessel Supports
16	BWR Main Steam Isolation Valve Leakage Control Systems
19	Safety Implications of Nonsafety Instrument and Control Power Supply Bus
20	Effects of Electromagnetic Pulse on Nuclear Plant Systems

ATTACHMENT 1 (Cont'd)

New Generic Issues (Cont'd)	Title
26	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power
28	Pressurized Thermal Shock
29	Bolting Degradation or Failure in Nuclear Power Plants

Task Action Plan Items	Title
A-14	Flaw Detection
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning
A-19	Digital Computer Protection System
A-20	Impacts of the Coal Fuel Cycle
A-22	PWR Main Steamline Break - Core, Reactor Vessel, and Containment Building Response
A-25	Non-Safety Loads on Class IE Power Sources
A-28	Increase in Spent Fuel Pool Storage Capacity
A-32	Missile Effects
A-33	NEPA Review of Accident Risks
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents

ATTACHMENT 1 (Cont'd)

Task Action Plan Items (Cont'd)	Title
A-35	Adequacy of Offsite Power Systems
A-37	Turbine Missiles
A-38	Tornado Missiles
B-3	Event Categorization
B-5(a)	Ductility of Two Way Slabs and Shells
B-7	Secondary Accident Consequence Modeling
B-9	Electrical Cable Penetrations of Containment
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA
B-21	Core Physics
B-22	LWR Fuel
B-23	LMFBR Fuel
B-24	Seismic Qualification of Electrical Mechanical Components
B-25	Piping Benchmark Problems
B-28	Radionuclide/Sediment Transport Program
B-29	Effectiveness of Ultimate Heat Sinks
B-31	Dam Failure Model
B-33	Dose Assessment Methodology
B-34	Occupational Radiation Exposure Reduction
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors

ATTACHMENT 1 (Cont'd)

Task Action Plan Items (Cont'd)	Title
B-37	Chemical Discharges to Receiving Waters
B-38	Reconnaissance Level Investigations
B-39	Transmission Lines
B-40	Effects of Power Plant Entrainment on Plankton
B-41	Impacts on Fisheries
B-42	Socioeconomic Environmental Impacts
B-43	Value of Aerial Photographs for Site Evaluation
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants
B-45	Need for Power - Energy Conservation
B-48	BWR CRD Mechanical Failure (Collet Housing)
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components
B-55	Improved Reliability of Target Rock Safety-Relief Valves
B-56	Diesel Reliability
B-57	Station Blackout
B-58	Passive Mechanical Failures
B-60	Loose-Parts Monitoring System

ATTACHMENT 1 (Cont'd)

Task Action Plan Items (Cont'd)	Title
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions
B-64	Decommissioning of Reactors
B-67	Effluent and Process Monitoring Instrumentation
B-68	Pump Overspeed During LOCA
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps
B-71	Incident Response
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure
C-8	Main-Steamline Leakage Control Systems
C-11	Assessment of Failure and Reliability of Pumps and Valves
C-12	Primary System Vibration Assessment
C-13	Non-Random Failures
C-14	Storm Surge Model for Coastal Sites
C-15	NUREG Report for Liquids Tank Failure Analysis

ATTACHMENT 1 (Cont'd)

Task Action Plan Items (Cont'd)	Title
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes
D-3	Control Rod Drop Accident

TMI Action Plan Items	
I.F.1	Expand QA List
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities
I.F.2(6)	Increase the Size of Licensees' QA Staff
I.F.2(7)	Clarify that the QA Program is a Condition of the Construction Permit and Operating License
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation
I.F.2(11)	Define Role of QA in Design and Analysis Activities
II.A.2	Site Evaluation of Existing Facilities
II.B.5(1)	Behavior of Severely Damaged Fuel
II.B.5(2)	Behavior of Core Melt

ATTACHMENT 1 (Cont'd)

TMI Action Plan Items (Cont'd)	Title
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities
II.B.7	Analysis of Hydrogen Control
II.B.8	Rulemaking Proceeding on Degraded-Core Accidents
II.C.1	Interim Reliability Evaluation Program
II.C.2	Continuation of Interim Reliability Evaluation Program
II.C.3	Systems Interaction
II.C.4	Reliability Engineering
II.D.2	Research on Relief and Safety Valve Requirements
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide
II.E.3.2	Systems Reliability
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements
II.E.3.4	Alternate Concepts Research
II.E.3.5	Regulatory Guide
II.E.4.3	Integrity Check
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Valve
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability
II.E.4.4(4)	Evaluate Purging and Venting During Normal Operation
II.E.4.4(5)	Issue Modified Purging and Venting Requirement

ATTACHMENT 1 (Cont'd)

TMI Action Plan Items (Cont'd)	Title
II.F.3	Instruments for Monitoring Accident Conditions
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure
II.H.3	Evaluate and Feed Back Information Obtained from TMI-2
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections
II.J.1.2	Modify Existing Vendor Inspection Program
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers
II.J.2.1	Reorient Construction Inspection Program
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program
II.J.2.3	Assign Resident Inspectors to All Construction Sites
II.J.3.1	Organization and Staffing to Oversee Design and Construction
II.J.3.2	Issue Regulatory Guide
II.J.4.1	Revise Deficiency Reporting Requirements
III.A.1.3(1)	Workers
III.A.1.3(2)	Public

ATTACHMENT 1 (Cont'd)

TMI Action Plan Items (Cont'd)	Title
III.A.3.1(1)	Define NRC Role in Emergency Situations
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610
III.A.3.1(4)	Prepare Commission Paper
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices
III.A.3.2	Improve Operations Centers
III.A.3.3(1)	Install Direct Dedicated Telephone Lines
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems
III.A.3.4	Nuclear Data Link
III.A.3.5	Training, Drills, and Tests
III.A.3.6	Interaction of NRC and Other Agencies
III.B.1	Transfer of Responsibilities to FEMA
III.B.2(1)	The Licensing Process
III.B.2(2)	Federal Guidance
III.D.1.2	Radioactive Gas Management
III.D.1.3(1)	Decide Whether Licensees Should Perform Studies and Make Modifications
III.D.1.3(2)	Review and Revise SRP
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems

ATTACHMENT 1 (Cont'd)

TMI Action Plan Items (Cont'd)	Title
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber
III.D.2.4(1)	Study Feasibility of Environmental Monitors
III.D.2.4(2)	Place 50 TLDs Around Each Site
III.D.2.6	Independent Radiological Measurements
III.D.3.1	Radiation Protection Plans
III.D.3.2(1)	Amend 10 CFR 20
III.D.3.2(2)	Issue a Regulatory Guide
III.D.3.2(3)	Develop Standard Performance Criteria
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators
III.D.3.5(1)	Develop Format for Data to be Collected by Utilities Regarding Total Radiation Exposure to Workers
III.D.3.5(2)	Investigative Methods of Obtaining Employee Health Data by Nonlegislative Means
III.D.3.5(3)	Revise 10 CFR 20
IV.C.1	Extend Lessons Learned from TMI-2 to Other NRC Programs
IV.E.1	Expand Research on Quantification of Safety Decision-Making
IV.E.2	Plan for Early Resolution of Safety Issues
IV.E.3	Plan for Resolving Issues at the CP Stage
IV.E.4	Resolve Generic Issues by Rulemaking
IV.E.5	Assess Currently Operating Reactors
IV.H.1	NRC Participation in the Radiation Policy Council

ATTACHMENT 2

LIST OF ITEMS FOR WHICH THE ACRS
AGREES WITH THE PRIORITY RANKINGS
PROPOSED BY THE NRC STAFF, BUT WITH COMMENTS

Issue No: 7

Title: Failures Due to Flow-Induced Vibrations

Proposed NRC Staff Priority: DROP

ACRS Comments: The Staff makes an adequate case for the types of failures they consider. However, they do not appear to have considered the problem of flow-induced breaking loose of flow deflectors of the sort that the Office for Analysis and Evaluation of Operational Data (AEOD) has brought out.

Issue No: 21

Title: Vibration Qualification of Equipment

Proposed NRC Staff Priority: Covered in USI A-46

ACRS Comments: The scope appears to be too narrow. A review of the dynamic loads to be included should be performed. Specifically, flow-induced vibrations should be evaluated and valve dynamic loads under faulted conditions (i.e.: rapid closure of main steam isolation valves under main steamline break) should be included. It should be noted that the scope of USI A-46 does not appear to include this issue as indicated by NUREG-0933.

ATTACHMENT 2 (Cont'd)

Issue No: 23

Title: Reactor Coolant Pump Seal Failures

Proposed NRC Staff Priority: HIGH

ACRS Comments: Coolant pump seal failure is a part of the small LOCA generic category. The utilities need to improve their ability to forstall these by design, and by better failure symptoms to signal for corrective maintenance actions. This does not appear to be a matter for NRC Staff work other than surveillance of licensee progress. Seal failures are not of more concern than LOCAs from power operated relief valve leaks or small line failures. The rate of coolant loss is the principal issue. Suggest that minor level of Staff effort be assigned.

Issue No: A-13

Title: Snubber Operability Assurance

Proposed NRC Staff Priority: RESOLVED

ACRS Comments: The Staff has imposed inspection requirements on the licensed plants. Effectiveness of these measures is not yet known. What frequency of snubber malfunction is tolerable? Does the test and inspection requirements satisfy the reliability requirements? To resolve this issue would require a study of failure trends over a period of time. Should be assigned to the Institute of Nuclear Power Operations (INPO) for reporting. Does not need Staff resources. This is only resolved in the sense that the NRC Staff believes that their requirements will make the failure rates OK. Should be listed as "resolved with qualifications".

ATTACHMENT 2 (Cont'd)

Issue No: A-18

Title: Pipe Rupture Design Criteria

Proposed NRC Staff Priority: DROP

ACRS Comments: This was never on the ACRS generic list but, because of the concern for pipe whip restraint problems of reliability and inspection access, it deserves attention. The Staff is ready to take a position on Westinghouse PWR primary piping that would eliminate double-ended pipe breaks as a design basis. To be useful, this action would have to extend to all PWRs and should address other piping systems which represent the bulk of the problem.

Not worthy of effort unless the results can be made available within a couple of years. Existing Westinghouse PWRs would probably not be altered by a change in criteria. The problem needs attention mainly because the case for reliable pipe whip restraints is weak and it would be better to build the safety argument on ductile inelastic response of piping systems. (The matter of most importance is the mode of rupture and potential locations. Not all restraints need be eliminated.)

Issue No: A-21

Title: Main Steamline Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification

Proposed NRC Staff Priority: LOW

ACRS Comments: The NRC Staff has codified the interim criteria (NUREG-0588) in the new Rule, 10 CFR 50.49. The environmental conditions inside containment appear to be based on successful operation of isolation devices such as the Main Steam Isolation Valves (MSIVs), turbine stop valves, and control valves to preclude the blowdown of more than one steam generator inside containment. The reliability of these valves to perform their isolation function should be evaluated. The ACRS will follow implementation.

ATTACHMENT 2 (Cont'd)

Issue No: A-23

Title: Containment Leak Testing

Proposed NRC Staff Priority: REGULATORY IMPACT ISSUE

ACRS Comments: The ACRS agrees with the Staff's proposed priority, but only within the strict context of the issue as described, not with the broader context of the title of the issue.

Issue No: B-1

Title: Environmental Technical Specifications

Proposed NRC Staff Priority: ENVIRONMENTAL ISSUE (RESOLVED)

ACRS Comments: The ACRS agrees with the Staff's proposed priority. However, it urges that, in revising these Specifications, the NRC Staff attempt to minimize the accompanying work load on the utilities. Data that are not necessary should not be required. Consideration should also be given to changing the title of this issue. The current title could imply that it pertains to the environmental qualification of safety-related nuclear power plant equipment.

Issue No: B-5(b)

Title: Buckling Behavior of Steel Containments

Proposed NRC Staff Priority: MEDIUM

ACRS Comments: If buckling could lead to early failure of the containment in some of the core-melt scenarios, the consequences could be changed more than in the analyses limited only to design basis accidents. This might justify a higher priority. However, research is now under way and probably cannot be accelerated much.

ATTACHMENT 2 (Cont'd)

Issue No: B-6

Title: Loads, Load Combinations, and Stress Limits

Proposed NRC Staff Priority: HIGH

ACRS Comments: This continues to be a matter of controversy between the Staff and licensees. It is a broader aspect of the issue A-18, "Pipe Rupture Design Criteria," applying to all structures. This has to be dealt with probabilistically. The work to date is confusing and the requirements lack consistency. Needs Staff work but should address all types of structures (piping, containments, supports, equipment, instrumentation, and controls) as influenced by structural loads.

Issue No: B-16

Title: Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

Proposed NRC Staff Priority: Covered in A-18 which has a priority ranking of DROP.

ACRS Comments: Requirements for these piping failures have been in place for a long time but questions have been raised about the interpretation and applicability of the requirements to older plants. The issue needs clarification. The Interim Reliability Evaluation Program (IREP) studies should indicate what is needed. No current basis exists for judging need for priority attention.

ATTACHMENT 2 (Cont'd)

Issue No: B-26

Title: Structural Integrity of Containment Penetrations

Proposed NRC Staff Priority: MEDIUM

ACRS Comments: If the research on penetration integrity in severe accidents shows that the penetrations are weak spots and containment is breached at lower pressure than gross failure, the consequence estimates in the Staff's analyses may be increased significantly. The cost of a fix might also be increased greatly. For these reasons, it is hard to prioritize the narrow issue defined here. MEDIUM may be OK for now, but the more general (and probably more important) issue of containment and penetration is clearly of HIGH priority.

Issue No: B-27

Title: Implementation and Use of Subsection NF of the ASME Code

Proposed NRC Staff Priority: LICENSING ISSUE

ACRS Comments: This requirement for structural supports is intended to assure adequacy of materials on which Code vessels are supported. This needs follow-up by the Office of Inspection and Enforcement (IE) and should be covered in the Interim Reliability Evaluation Program studies.

Issue No: B-47

Title: Inservice Inspection of Support Classes 1, 2, 3 and MC Components

Proposed NRC Staff Priority: DROP

ACRS Comments: This issue is not clear. The need for inspection depends on the safety concerns. The Staff needs to clarify the issue.

ATTACHMENT 2 (Cont'd)

Issue No: C-1

Title: Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

Proposed NRC Staff Priority: RESOLVED

ACRS Comments: This problem appears to be resolved by the codification of NUREG-0588 and the Division of Operating Reactor (DOR) Guidelines. Requirements have been established and are being implemented. Apparently, during the equipment qualification evaluation, maintenance procedures are reviewed. However, details of the NRC Staff requirements are not known. In addition, this issue appears to be limited to moisture ingress through damaged hermetic seals. Other areas, such as moisture ingress through conduits should be evaluated.

Issue No: C-7

Title: PWR System Piping

Proposed NRC Staff Priority: RESOLVED

ACRS Comments: The interpretation of this issue is unclear. BWR piping systems have problems but PWRs have no identifiable difficulties unless they are in pressure-letdown circuits.

Issue No: C-10

Title: Effective Operation of Containment Sprays in a LOCA

Proposed NRC Staff Priority: RESOLVED

ACRS Comments: It is not clear that the evaluation of damage to equipment by inadvertent actuation has been fully resolved. The documents cited as resolving this issue (SRP 6.5.2 and ANSI/ANS 56.5-1979) do not address damage to equipment. It should be noted also that chemical addition systems for containment spray are considered optional.

ATTACHMENT 2 (Cont'd)

Issue No: III.D.1.4

Title: Radwaste System Design Features to Aid in Accident Recovery and Decontamination

Proposed NRC Staff Priority: DROP

ACRS Comments: Experience shows that radwaste systems in existing nuclear power plants are inadequate to meet post-accident decontamination requirements. Although this situation does not justify backfitting such systems on existing plants, the ACRS believes that this issue should be "flagged" for reconsideration if and when applications for construction permits are forthcoming.

ATTACHMENT 3

LIST OF ITEMS FOR WHICH THE ACRS DISAGREES WITH THE PRIORITY RANKINGS PROPOSED BY THE NRC STAFF

Issue No: B-32

Title: Ice Effects on Safety-Related Water Supplies

Proposed NRC Staff Priority: LICENSING ISSUE

ACRS Recommendation: MEDIUM

Reasons: The ACRS believes that interference with supply of cooling water to safety-related equipment could decrease the reliability of ultimate heat sink to adequately cool the core. An evaluation is required to determine what contribution a reduction in ultimate heat sink reliability makes to overall core melt.

Issue No: I.F.2(1)

Title: Assure the Independence of the Organization Performing the Checking Function

Proposed NRC Staff Priority: LOW

ACRS Recommendation: MEDIUM or HIGH

Reasons: Why is this issue ranked LOW when significant IE resources are being expended on related QA initiatives, particularly those looking at designated representatives and third party audits?

ATTACHMENT 3 (Cont'd)

Issue No:	I.F.2(4)
Title:	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment
Proposed NRC Staff Priority:	LOW
ACRS Recommendation:	MEDIUM or HIGH
Reasons:	Why is this issue ranked LOW when resources are being expended in this area? This issue is being addressed by the study required under P.L. 97.415, Section 13b(1). The Commission is required to report to the Congress with the results of that study by April 1984. The Staff has scheduled study completion accordingly. IE also has ongoing research related to a graded QA approach to assuring plant quality. In addition, the ACRS has expressed its interest in this general area (Ref. NUREG-0963, Section 2.3).

Issue No:	I.F.2(5)
Title:	Establish Qualification Requirements for QA and QC Personnel
Proposed NRC Staff Priority:	LOW
ACRS Recommendation:	MEDIUM or HIGH
Reasons:	Why is this issue ranked LOW when significant IE resources are being expended on related QA initiatives, particularly those related to the qualification and certification of QA/QC personnel?

ATTACHMENT 3 (Cont'd)

Issue No:	II.A.1
Title:	Siting Policy Reformulation
Proposed NRC Staff Priority:	MEDIUM
ACRS Recommendation:	LOW
Reasons:	Because of the lack of applications for construction permits for new nuclear power plants, and the existence of a regulatory basis for reviewing and approving sites for such plants, the ACRS recommends that consideration be given to downgrade the priority of this issue from MEDIUM to LOW.

Issue No:	II.B.5(3)
Title:	Effects of Hydrogen Burning and Explosions on Containment Structure
Proposed NRC Staff Priority:	MEDIUM
ACRS Recommendation:	HIGH
Reasons:	Why is this ranked MEDIUM in view of the fact that extensive and expensive research on this subject is either in progress or planned?

ATTACHMENT 3 (Cont'd)

Issue No:	III.D.2.1
Title:	Radiological Monitoring of Effluents
Proposed NRC Staff Priority:	LOW
ACRS Recommendation:	MEDIUM or HIGH
Reasons:	The ACRS is concerned that the NRC regulations currently do not require that monitors for airborne effluents be in duplicate. As a result, when such monitors are out of service, little or no information on release to the environment would be available. Although frequently both low-range (for routine releases) and high-range (for accidental releases) monitors are available in the same effluent line, the low-range units would provide no useful information for high-range releases. In addition, the Subcommittee believes that the high-range instruments should contain the capabilities for specific radionuclide analyses.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 15, 1983

MEMORANDUM FOR: W. J. Dircks
Executive Director for Operations

FROM: M. W. Libarkin, Acting Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: ADDITIONAL ACRS COMMENTS ON THE PRIORITIZATION OF GENERIC ISSUES

During its 278th meeting, June 9-11, 1983, the ACRS reviewed the adequacy of the NRC Staff's assignment of priority rankings to a group of generic issues, and transmitted its comments to you through a memorandum from R. F. Fraley, dated June 20, 1983.

During its 279th meeting, July 7-9, 1983, the ACRS reviewed the proposed priority rankings of an additional group of remaining generic issues, and its comments are contained in the following attachments:

- Attachment 1 lists those items for which the ACRS agrees with the priority rankings proposed by the NRC Staff.
- Attachment 2 includes a list of items for which the ACRS agrees with the priority rankings proposed by the NRC Staff, but has comments.
- Attachment 3 contains a list of items for which the ACRS disagrees with the NRC Staff's proposed priority rankings along with the reasons therefor.

It is requested that the NRC Staff provide written responses to the ACRS comments identified in Attachments 2 and 3. Cognizant ACRS Subcommittees or the ACRS may review the NRC Staff's responses.

Please note that in the June 20, 1983 memorandum from R. F. Fraley to W. J. Dircks, Subject: ACRS Comments on the Prioritization of Generic Issues, the following issues had inadvertently been included in Attachment 1, the list of items for which the ACRS indicated agreement with the NRC Staff's assignment of priority:

II.J.3.1 Organization and Staffing to Oversee Design and Construction

III.A.3.4 Nuclear Data Link

The ACRS actually disagrees with the priority rankings proposed by the NRC Staff for these issues, and the Committee's recommendations and the reasons therefor are contained in Attachment 3 to this memorandum.

The ACRS will continue its review of the proposed priority rankings for the remaining generic issues and will provide additional comments as they become available.

Attachments: As Stated

ATTACHMENT 1

LIST OF ITEMS FOR WHICH THE ACRS

AGREES WITH THE PRIORITY RANKINGS

PROPOSED BY THE NRC STAFF

New Generic Issues	Title
8	Inadvertent Actuation of Safety Injection in PWRs
9	Reevaluation of Reactor Coolant Pump Trip Criteria
10	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges
13	Small-Break LOCA from Extended Overheating of Pressurizer Heaters
18	Steam-Line Break With Consequential Small LOCA
24	Automatic Emergency Core Cooling System Switch to Recirculation
27	Manual Vs Automated Actions

Task Action Plan Items	
A-16	Steam Effects on BWR core Spray Distribution
A-27	Reload Applications
A-30	Adequacy of Safety-Related DC Power Supplies
B-10	Behavior of BWR Mark III Containments
B-11	Subcompartment Standard Problems
B-13	Marviken Test Data Evaluation
B-15	CONTEMPT Computer Code Maintenance
B-17	Criteria for Safety-Related Operator Actions
B-18	Vortex Suppression Requirements for Containment Sumps
B-19	Thermal-Hydraulic Stability
B-20	Standard Problem Analysis

ATTACHMENT 1 (Cont'd)

Task Action Plan Items (Cont'd)	
B-52	Fuel Assembly Seismic and LOCA Responses
B-54	Ice Condenser Containments
B-59	N-1 Loop Operation in BWRs and PWRs
B-61	Allowable ECCS Equipment Outage Periods
B-69	ECCS Leakage Ex-Containment
C-3	Insulation Usage Within Containment
C-4	Statistical Methods for ECCS Analysis
C-5	Decay Heat Update
C-6	LOCA Heat Sources

TMI Action Plan Items	
I.A.1.4	Long-Term Upgrading
I.A.2.2	Training and Qualifications of Operations Personnel
I.A.2.4	NRR Participation in Inspector Training
I.A.2.6(1)	Revise Regulatory Guide 1.8
I.A.2.6(2)	Staff Review of NRR 80-117
I.A.2.6(3)	Revise 10 CFR 55
I.A.2.6(4)	Operator Workshops
I.A.2.6(6)	Nuclear Power Fundamentals
I.A.2.7	Accreditation of Training Institutions
I.A.3.3	Requirements for Operator Fitness
I.A.3.4	Licensing of Additional Operations Personnel
I.A.3.5	Establish Statement of Understanding with INPO and DOE

ATTACHMENT 1 (Cont'd)

TMI Action Plan Items (Cont'd)	
I.A.4.1(1)	Short-Term Study of Training Simulators
I.A.4.1(2)	Interim Changes in Training Simulators
I.A.4.2(1)	Research on Training Simulators
I.A.4.2(2)	Upgrade Training Simulator Standards
I.A.4.2(4)	Review Simulators for Conformance to Criteria
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator
I.A.4.4	Feasibility Study of NRC Engineering Computer
I.C.1(4)	Confirmatory Analyses of Selected Transients
I.D.5(2)	Plant Status and Post-Accident Monitoring
I.D.5(3)	On-Line Reactor Surveillance System
I.D.6	Technology Transfer Conference
I.G.2	Scope of Test Program
II.E.2.1	Reliance on ECCS
II.E.2.2	Research on Small-Break LOCAs and Anomalous Transients
IV.A.1	Seek Legislative Authority
IV.A.2	Revise Enforcement Policy
IV.D.1	NRC Staff Training
IV.F.1	Increase IE Scrutiny of the Power-Ascension Test Program
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants
IV.G.1	Develop a Public Agenda for Rule Making
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules
IV.G.3	Improve Rule Making Procedures
IV.G.4	Study Alternatives for Improved Rule Making Process

ATTACHMENT 2

LIST OF ITEMS FOR WHICH THE ACRS

AGREES WITH THE PRIORITY RANKINGS

PROPOSED BY THE NRC STAFF, BUT WITH COMMENTS

Issue No: 22

Title: Inadvertent Boron Dilution Events

Proposed NRC Staff Priority: DROP

ACRS Comments: Considering the "worst" of the designs, it should be ascertained that the probability of a radiation injury or fatality to plant personnel working in the vicinity of the core is acceptably low.

Issue No: B-8

Title: Locking Out of ECCS Power Operated Valves

Proposed NRC Staff Priority: DROP

ACRS Comments: The NRC Staff suggests dropping this item since no significant accident initiators have been identified to date. However, the Staff has not examined any plant PRAs in this regard. Although the ACRS agrees with the proposed priority ranking for now, it plans to explore the significance of this item vis-a-vis available PRAs. In the event of significant findings, the ACRS may request a reopening of this item.

ATTACHMENT 2 (Cont'd)

Issue No: B-53

Title: Load-Break Switch

Proposed NRC Staff Priority: REGULATORY IMPACT ISSUE
(Possible Resolution Identified for Evaluation)

ACRS Comments: The question as to whether the grid or the plant main generator should be used as the preferred source of AC power should be examined in the near future either under this activity or under a new activity.

Issue No: I.A.3.2

Title: Operator Licensing Program Changes

Proposed NRC Staff Priority: RESOLVED

ACRS Comments: The ACRS agrees that the specific actions called for under this item have been adequately addressed. However, significant operator licensing program changes are still under way and necessary. The appropriate content and form of licensing examinations are in question. The qualifications required for license examiners need to be specified.

ATTACHMENT 3

LIST OF ITEMS FOR WHICH THE ACRS

DISAGREES WITH THE PRIORITY RANKINGS

PROPOSED BY THE NRC STAFF

Issue No: A-29

Title: Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Proposed NRC Staff Priority: MEDIUM

ACRS Recommendation: HIGH

Reasons: The ACRS believes that the topic of interest is the reduction of nuclear power plant vulnerability to industrial sabotage. Industrial sabotage at nuclear power plants is a sensitive issue which receives significant public attention. A-29 should not be restricted in scope so as to address only design changes to nuclear power plants. The Staff needs to consider sabotage in the broadest possible terms and to ensure that the opportunities and likelihood for sabotage are as low as reasonably achievable. The ACRS recognizes the difficulties inherent in using probabilistic techniques to determine public risk due to acts of sabotage. "Resolution" of this issue most probably will consist of applying serious, broad, and continuing attention to the matter. The ACRS believes that the scope of A-29 should be expanded and a priority ranking of HIGH should be assigned.

Issue No: A-41

Title: Long-Term Seismic Program

Proposed NRC Staff Priority: MEDIUM

ACRS Recommendation: HIGH

Reasons: This program should be augmented or a new activity established as per the ACRS letter of January 11, 1983, to Chairman Palladino, related to the Quantification of Seismic Safety Margins.

ATTACHMENT 3 (Cont'd)

Issue No:	B-30
Title:	Design-Basis Floods and Probability
Proposed NRC Staff Priority:	LICENSING ISSUE
ACRS Recommendation:	HIGH
Reasons:	The NRC currently lacks a quantitative basis for evaluating this event.

Issue No:	B-50
Title:	Post-Operating Basis Earthquake Inspection
Proposed NRC Staff Priority:	REGULATORY IMPACT ISSUE
ACRS Recommendation:	MEDIUM
Reasons:	Some advanced planning as to what would be required before plant startup should be done to avert unnecessarily long and expensive shutdowns and to assure that the proper actions would be taken.

Issue No:	D-1
Title:	Advisability of a Seismic Scram
Proposed NRC Staff Priority:	DROP
ACRS Recommendation:	MEDIUM
Reasons:	The NRC Staff has not adequately evaluated this issue nor obtained an understanding as to the basis for this requirement by the Japanese. Similarly, the British are currently reviewing the advisability of a seismic scram.

ATTACHMENT 3 (Cont'd)

Issue No: D-2

Title: Emergency Core Cooling System Capability for Future Plants

Proposed NRC Staff Priority: Covered in USI A-45

ACRS Recommendation: MEDIUM

Reasons: The focus of USI A-45 is on current-generation plants, not future plants as Item D-2 implies. In addition, based on its review and discussion with the cognizant NRC Staff, the ACRS believes that this item is not covered in USI A-45. As the ACRS has noted in a recent report to the Commission on the NRC Safety Research Program Budget for FY 1985 and 1986, the current Appendix K requirements have influenced, and will continue to influence the design and operation of ECC systems in a manner that could be deleterious to the overall concerns of ECC.

The ACRS believes that the NRC Staff should carry this as a separate generic issue, with a MEDIUM priority, and explore it in the context of the ongoing RES effort to revise Appendix K of 10 CFR Part 50.

Issue No: I.A.2.5

Title: Plant Drills

Proposed NRC Staff Priority: LOW

ACRS Recommendation: MEDIUM or HIGH

Reasons: The ACRS questions the technical basis for the assumptions made to compute the risk reduction potential associated with this issue. Because of uncertainties in the assumptions made, coupled with the fact that the Value/Impact Score indicated that the ranking of this issue would be LOW to MEDIUM, the conservative approach would be to assign a MEDIUM priority to this issue (at least until better information is obtained). Also, matters related to simulator fidelity and validity of training programs are being pursued with high priority under several human factors generic issues. This suggests that a higher priority might be warranted.

ATTACHMENT 3 (Cont'd)

Issue No:	I.A.2.6(5)
Title:	Develop Inspection Procedures For Training Program
Proposed NRC Staff Priority:	RESOLVED
ACRS Recommendation:	MEDIUM
Reasons:	Why is this considered RESOLVED when RES has a research program under way that should provide some criteria for assisting in training effectiveness evaluation? In addition, Staff efforts are under way to respond to Public Law 97-425. The Staff's proposed package in response to the Act includes a new training and qualification rule, a new training regulatory guide, and revised NRC inspection modules. Since the training regulatory guide and inspection modules (not yet developed) will provide detailed guidance to the industry and inspectors, and since the Staff intends to recommend 2-5 years for Industry to comply with the new training rule/guidance, a MEDIUM priority seems appropriate. Finally, one of the 23 tasks contained in the Human Factors Program Plan also deals with this specific topic. Since extensive Staff resources are being expended in this area, how can this issue be considered RESOLVED?

Issue No:	I.A.4.2(3)
Title:	Regulatory Guide on Training Simulators
Proposed NRC Staff Priority:	RESOLVED
ACRS Recommendation:	MEDIUM or HIGH
Reasons:	While Regulatory Guide 1.149, "Nuclear Power Plant Simulators For Use in Operator Training" was published to endorse the revised ANSI/ANS 3.5-1981, additional training simulator changes still need to be made. Regulatory Guide 1.8 should be evaluated and revised as necessary in light of the ongoing simulator research. The Staff plans to make another revision to Regulatory Guide 1.149. This revision will continue to endorse ANSI/ANS 3.5-1981 (with some exceptions) but will require nuclear power plants to have a plant referenced simulator or other facility proposed device acceptable to the NRC. Continued expenditure of Staff resources in this area appears appropriate.

ATTACHMENT 3 (Cont'd)

Issue No: I.B.1.1

Title: Organization and Management of Long-Term Improvements, including:

I.B.1.1(1) Prepare Draft Criteria

I.B.1.1(2) Prepare Commission Paper

I.B.1.1(3) Issue Requirements for the Upgrading of Management and Technical Resources

I.B.1.1(4) Review Responses to Determine Acceptability

I.B.1.1(6) Prepare Revisions to Regulatory Guides 1.33 and 1.8

I.B.1.1(7) Issue Regulatory Guides 1.33 and 1.8

Proposed NRC Staff Priority: MEDIUM

ACRS Recommendation: HIGH

Reasons: The ACRS believes that improvements to a utility's management and organization will reap benefits other than human-error rate reduction. These other benefits (e.g., improved productivity, improved plant quality, reduced absenteeism and turnover rate, etc.) should be taken into account when determining the priority for these items. The ACRS believes that such an analysis would indicate that these items should receive a HIGH priority. The ACRS would also like to caution the Staff that human-error rates vary greatly with managerial systems (probably much more than 0-20%). Human factors experts should be actively involved in evaluating the assumptions made to arrive at the total risk reduction.

ATTACHMENT 3 (Cont'd)

Issue No:	I.B.1.1(5)
Title:	Review Implementation of the Upgrading Activities
Proposed NRC Staff Priority:	RESOLVED
ACRS Recommendation:	HIGH
Reasons:	The Staff states that, since the Office of Inspection and Enforcement (IE) routinely develops and issues inspection procedures which address new or revised regulations and requirements, this item is considered as RESOLVED. Since utilities will be required to submit a new proposed organization and management plan which will be reviewed by the NRC (including a site review), and since the IE Staff will perform annual assessments to assure each utility is satisfactorily meeting NRC management and organization requirements (as identified in the initial NRR plant review), it seems that this issue should remain open until after the first IE audit subsequent to NRC approval of each utility's organization and management plan. The priority assigned to this item should be commensurate with other items in I.B.1.1.

ATTACHMENT 3 (Cont'd)

Issue No:	I.C.9
Title:	Long-Term Program Plan for Upgrading of Procedures
Proposed NRC Staff Priority:	MEDIUM
ACRS Recommendation:	HIGH
Reasons:	The Staff's analysis concludes that resolution of this item might result in a total reduction in public risk of 5×10^{4} man-rem. This is the safety importance at which an item would change from a MEDIUM to a HIGH priority. This risk reduction is based on a uniform 30% improvement in human error, including maintenance, through the dominant accident sequences. This 30% improvement includes improvements due to upgraded Emergency Operating Procedures (EOPs) which are no longer being considered as part of I.C.9. The Staff assumes that 70% of the 30% improvement will result from upgrading EOPs and the remaining 30% will result from upgrading other procedures (e.g., normal, abnormal, maintenance, etc.). In view of the high safety importance associated with this item and the ACRS' belief that more than 30% of the total benefits derived from upgrading procedures (including EOPs) will be accrued by upgrading normal, abnormal, and maintenance procedures (excluding EOPs), the ACRS believes that this item should be assigned a HIGH priority. It is also important for the NRC to recognize that improving job design (e.g. procedures, hardware, etc.) is as important as modifying people (e.g., training) in reducing human errors at nuclear power plants.

ATTACHMENT 3 (Cont'd)

Issue No: I.D

Title: Control Room Design, including:

I.D.3	Safety System Status Monitoring
I.D.4	Control Room Design Standard
I.D.5(5)	Disturbance Analysis Systems

Proposed NRC Staff Priority: MEDIUM

ACRS Recommendation: HIGH

Reasons: There is overwhelming evidence that poorly designed control rooms contribute to operator error. Emphasis on the machine side of the man-machine interface (to reduce human error) is as important as the selection and training of plant personnel. In addition, the weak link in reactor operations appears to involve diagnosis of the root cause of a plant's upset condition. Diagnosis involves cognitive skills such as judgment, problem solving, and decision making. Control room operators need all the help they can get in a time of upset plant conditions.

ATTACHMENT 3 (Cont'd)

Issue No:	I.D.5(1)
Title:	Operator-Process Communication
Proposed NRC Staff Priority:	RESOLVED
ACRS Recommendation:	HIGH
Reasons:	While RES has issued a Research Information Letter (RIL-124) that provides recommendations for future action related to the operator-machine interface in reactor control rooms, this item should not be considered as RESOLVED until either those actions are carried out or they are deemed unnecessary. The ACRS believes that this item should be assigned a HIGH priority similar to other items under general topic I.D, Control Room Design.

Issue No:	I.D.5(4)
Title:	Process Monitoring Instrumentation
Proposed NRC Staff Priority:	RESOLVED
ACRS Recommendation:	HIGH
Reasons:	NRC has been evaluating number of systems (e.g., liquid level monitoring) at the LOCA experimental facilities at ORNL and INEL. While this work is almost completed, this item should remain open until research results are documented and regulatory guidance has been provided to the nuclear power industry. The ACRS believes that completion of this ongoing work should be given a HIGH priority.

ATTACHMENT 3 (Cont'd)

Issue No:	II.E.2.3
Title:	Uncertainties in Performance Predictions
Proposed NRC Staff Priority:	LOW
ACRS Recommendation:	MEDIUM
Reasons:	Higher priority should be given to the evaluation of uncertainties for small breaks.

Issue No:	II.E.6.1
Title:	Test Adequacy Study
Proposed NRC Staff Priority:	MEDIUM
ACRS Recommendation:	HIGH
Reasons:	Disagree with the reliability initially assigned to new valves operating under adverse or design conditions. Also, the Staff's analysis assumes that an improved testing program would reduce valve failures by 5 percent. Can this number be justified? Improved inservice testing of valves should make use of base-line data obtained from the valve in its new condition, e.g., a strip chart recording taken as the valve was cycled would show the approach to inoperability better than the leak rate tests or periodic cycling that are now used. In addition, effort should be made to develop dynamic tests for both new and inservice valves to assure their operability under design loads.

ATTACHMENT 3 (Cont'd)

Issue No:	II.J.3.1
Title:	Organization and Staffing to Oversee Design and Construction
Proposed NRC Staff Priority:	MEDIUM
ACRS Recommendation:	HIGH
Reasons:	See comments related to I.B.1.1 (Page 5, Attachment 3)

Issue No:	III.A.3.4
Title:	Nuclear Data Link
Proposed NRC Staff Priority:	MEDIUM
ACRS Recommendation:	DROP
Reasons:	The ACRS believes that the proposed Nuclear Data Link should not be implemented.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 16, 1983

MEMORANDUM FOR: W. J. Dircks
Executive Director for Operations

FROM: R. F. Fraley, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: ACRS COMMENTS ON THE PRIORITIZATION OF GENERIC SAFETY ISSUES

During its 280th meeting, August 4-6, 1983, the ACRS completed its review of the adequacy of the NRC Staff's assignment of priority rankings to individual generic issues and its comments are contained in the following attachments. Partial lists of ACRS comments on the related matter were transmitted to you on two previous occasions through a memorandum from R. F. Fraley, dated June 20, 1983 and also through a memorandum from M. W. Libarkin, dated July 15, 1983:

- Attachment 1 lists those items for which the ACRS agrees with the priority rankings proposed by the NRC Staff.
- Attachment 2 includes a list of items for which the ACRS agrees with the priority rankings proposed by the NRC Staff, but has comments.
- Attachment 3 contains a list of items for which the ACRS disagrees with the NRC Staff's proposed priority rankings along with the reasons therefor.

It is requested that the NRC Staff provide written responses to the ACRS comments identified in Attachments 2 and 3. Cognizant ACRS Subcommittees and/or the ACRS may review the NRC Staff's responses.

Please note that those items designated as Unresolved Safety Issues were not reviewed by the ACRS at this time; these items will be reviewed by the appropriate ACRS Subcommittees and/or the ACRS when their resolutions become available.

In addition, the following items were reviewed by the ACRS Subcommittee on Reactor Radiological Effects during its meeting on April 29, 1983, but action was deferred owing to lack of sufficient information and/or other reasons (See Attachment 4). These issues will be reviewed by the cognizant Subcommittee and the ACRS when sufficient information becomes available:

ITEM NO	TITLE
1	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems, and for Normal Ventilation Systems
B-65	Iodine Spiking
B-66	Control Room Infiltration Measurements
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants
III.D.2.3(2)	Discriminate Between Sites and Plants that Require Consideration of Liquid Pathway Interdiction Techniques
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction
III.D.2.3(4)	Prepare a Summary Assessment

Also note that the following were not reviewed by the ACRS since they are not considered to be nuclear-safety-related issues:

ITEM NO	TITLE
B-2	Forecasting Electricity Demand
B-45	Need for Power - Energy Conservation

ITEM NO	TITLE
III.C.1(1)	Review Publicly Available Documents
III.C.1(2)	Recommend Publication of Additional Information
III.C.1(3)	Program of Seminars for News Media Personnel
III.C.2(1)	Develop Policy and Procedures for Dealing with Briefing Requests
III.C.2(2)	Provide Training for Members of the Technical Staff

Attachments: As Stated

ATTACHMENT 1

LIST OF ITEMS FOR WHICH THE ACRS AGREES WITH THE PRIORITY RANKINGS PROPOSED BY THE NRC STAFF

Item No.	Title
17	Loss of Offsite Power Subsequent to LOCA
B-12	Containment Cooling Requirements (Non-LOCA)
I.B.1.3(1)	Require Licensees to Place Plant in Safest Shutdown Cooling Following a Loss of Safety Function Due to Personnel Error
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling
I.B.2.2	Resident Inspector at Operating Reactors
I.B.2.3	Regional Evaluations
I.B.2.4	Overview of Licensee Performance
I.E.1	Office for Analysis and Evaluation of Operational Data
I.E.2	Program Office Operational Data Evaluation
I.E.3	Operational Safety Data Analysis
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs
I.E.5	Nuclear Plant Reliability Data System
I.E.6	Reporting Requirements
I.E.7	Foreign Sources
I.E.8	Human Error Rate Analysis
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment
III.D.2.5	Offsite Dose Calculational Manual

ATTACHMENT 2

LIST OF ITEMS FOR WHICH THE ACRS
AGREES WITH THE PRIORITY RANKINGS
PROPOSED BY THE NRC STAFF, BUT WITH COMMENTS

Issue No:	25
Title:	Automatic Air Header Dump on BWR Scram System
Proposed NRC Staff Priority:	RESOLVED
ACRS Comments:	The ACRS is satisfied with the specific actions taken by the NRC Staff to resolve this issue for the current generation of plants. The ACRS, in its review of future plants, will consider whether fundamental changes should be required to achieve higher system reliability.

Issue No:	II.E.5.1
Title:	Design Evaluation
Proposed NRC Staff Priority:	Resolution Available
ACRS Comments:	The ACRS agrees with the NRC Staff's approach to resolution of this issue, i.e., to continue the effort to improve B&W response to feedwater transients within a broader program of analysis, experimentation, and field trials of modifications. The ACRS wishes to be kept informed of the operational experience gained from the modifications already approved by the Staff for Midland, which are similar to those proposed for other B&W plants that may yield data before the Midland Units. Since the resolution of II.E.5 is closely associated with those of USI A-47 and A-49, the cognizant ACRS Subcommittees and the ACRS will be following any advances made in conjunction with those issues by means of periodic reviews. Also, the ACRS commends the recent agreement by the NRC, the B&W Owners Group, the B&W Company, and EPRI to initiate a joint experimental program on B&W reactors; however, it reserves judgment on the adequacy of the program until its scope has been better defined.

ATTACHMENT 2 (Cont'd)

Issue No:	II.E.5.2
Title:	B&W Reactor Transient Response Task Force
Proposed NRC Staff Priority:	Resolution Available
ACRS Comments:	See comments made on II.E.5.1

Issue No:	II.F.4
Title:	Study of Control and Protective Action Design Requirements
Proposed NRC Staff Priority:	DROP
ACRS Comments:	The ACRS is satisfied with the proposed NRC Staff actions on this issue. The Electrical Systems Subcommittee will examine related issues in their review of USI A-47.

ATTACHMENT 3

LIST OF ITEMS FOR WHICH THE ACRS
DISAGREES WITH THE PRIORITY RANKINGS
PROPOSED BY THE NRC STAFF

Issue No:	4
Title:	End-of-Life and Maintenance Criteria
Proposed NRC Staff Priority:	RESOLVED
ACRS Comments:	There should be some discussion of the benefits and limitations of "type" testing - especially as regards assembly after repair of type-tested items. The matter of ensuring that equipment will properly perform under emergency environmental and load conditions should be discussed with respect to testing. Consideration should be given to the range of conditions which might be obtained within a normally "mild" environment such as drenching from fire extinguishing processes or steaming and condensation from steam leaks. A particular concern is the matter of ensuring proper valve performance after years of cycling tests at essentially no-load conditions. The ACRS believes that maintenance is an important issue and the Staff should pay continued attention to this matter. The ACRS plans to establish a Subcommittee on Maintenance to look into maintenance-related issues.

ATTACHMENT 3 (Cont'd)

Issue No:	B-4
Title:	ECCS Reliability
Proposed NRC Staff Priorhty:	Covered in II.E.3.2 which in turn is indicated to be incorporated into USI A-45.
ACRS Comments:	NUREG-0933 states that this issue is covered under the TMI-Action Plan item II.E.3.2, "System Reliability," which in turn has been incorporated into USI A-45, "Shutdown Decay Heat Removal Requirements." Based on its review, the ACRS believes that the assignment of Issue B-4 to USI A-45 is appropriate only to the extent that Issue B-4 deals with ECCS reliability in small-break accidents. In these scenarios, reactor pressure vessel inventory is maintained over a long period of time via high pressure makeup systems. The ECC systems are currently being treated within this context in USI A-45. To the extent that Issue B-4 is expected to deal with broader issues, such as large break LOCAs, it should be carried out as a separate generic issue with a MEDIUM priority ranking.

ATTACHMENT 3 (Cont'd)

Issue No:	B-63
Title:	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary
Proposed NRC Staff Priority:	RESOLVED
ACRS Comments:	Simple redundancy does not preclude common mode failures. Leak testing does not necessarily reveal strength margins or interlock vulnerabilities. It is the quality of the test data with respect to operation under emergency conditions and the reliability of the interlock systems that is in question. In addition, we are not aware of any provisions made to detect the failure of one train of a redundant system given that the second train performs as expected. Although the overpressurization of low pressure systems connected to the secondary side of a PWR is not addressed by this item, the ACRS assumes it is being reviewed in conjunction with pipe breaks outside of containment. This should be verified.

ATTACHMENT 3 (Cont'd)

Issue No: I.B.2.1

Title: Revise OIE Inspection Program, including:

I.B.2.1(1), Verify the Adequacy of Management and Procedural Controls and Staff Discipline.

I.B.2.1(2), Verify that Systems Required to be Operable are Properly Aligned.

I.B.2.1(3), Follow-up on Completed Maintenance Work Orders to Assure Proper Testing and Return to Service.

I.B.2.1(4), Observe Surveillance Tests to Determine Whether Test Instruments are Properly Calibrated.

I.B.2.1(5), Verify that Licensees are Complying with Technical Specifications.

I.B.2.1(6) Observe Routine Maintenance.

I.B.2.1(7) Inspect Terminal Boards, Panels, and Instrument Racks for Unauthorized Jumpers and Bypasses.

Proposed NRC Staff Priority: LICENSING ISSUE - RESOLVED

ACRS Comments: Onsite audits should also include verification of:

- Procedures under which systems must be operated and protected when non-aligned.
- The effectiveness and reliability of equipment isolation and return-to-service procedures including evaluation of locks, seals, and tag-out methods.

Of particular concern is simultaneous, improper maintenance of redundant equipment, leading to common mode failure.

ATTACHMENT 3 (Cont'd)

Issue No:	IV.B.1
Title:	Revised Practices for Issuance of Instructions and Information to Licensees
Proposed NRC Staff Priority:	LICENSING ISSUE - RESOLVED
ACRS Comments:	Information distributed to a utility by a vendor must, of course, be acted upon if it affects safety-related equipment; however, there is no requirement that the utility acknowledge receipt of the information. In light of the Salem incident, this shortcoming should be addressed.

ATTACHMENT 4

COMMENTS BY THE ACRS SUBCOMMITTEE ON REACTOR RADIOLOGICAL EFFECTS

Issue 1 - Failures in Air Monitoring, Air Cleaning and Ventilating Systems (Staff proposes DROP)

Discussion of this issue during the Subcommittee meeting on April 29, 1983 revealed that the Subcommittee had not been clear in defining the problem. As we see it, it involves two subareas:

a. Environmental Control for the Protection of Equipment, and

b. Control Room Habitability.

Issues B-36 and B-66 are related to control room habitability and should be grouped under item b. of this issue as retitled above. In terms of detail, we believe that control room leakage should be evaluated under conditions of positive as well as negative pressure, and that the safety implications of air monitors that fail up-scale as well as down-scale should be assessed.

Once the above reorientation is accomplished and priorities have been assigned, we will be pleased to review these issues further.

B-65 - Iodine Spiking (Previously classified as an environmental issue)

The Staff informed the Subcommittee during the meeting on April 29, 1983 that this issue had been incorrectly classified and should consist of two subparts: one dealing with the model for evaluating design basis accidents, which is a licensing issue; the other is a safety issue, dealing with standard technical specification limits on iodine.

We concur with the Staff that the second part is a safety issue and suggest that it be assigned an appropriate regulatory priority (e.g., MEDIUM priority).

III.D.2.3 - Liquid Pathway Radiological Control (Staff claims that possible resolution has been identified for this issue)

The Subcommittee was unable to evaluate this issue due to a lack of information on the criteria to be used as guidance in evaluating liquid releases, techniques for assessing the loss of a major societal resource that might accompany a major liquid release, specific interdictive measures that are to be evaluated, and methods for assessing the effectiveness of such measures in reducing population doses. We will be pleased to consider this issue and the Staff's resolution when answers to these questions can be provided.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 7, 1983

Mr. William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

SUBJECT: ACRS REVIEW OF NRC STAFF RESOLUTION FOR UNRESOLVED SAFETY ISSUES A-1, "WATER HAMMER," AND A-43, "CONTAINMENT EMERGENCY SUMP PERFORMANCE"

During its 281st meeting, August 31-September 1, 1983, the Advisory Committee on Reactor Safeguards reviewed the NRC Staff resolution for Unresolved Safety Issues (USI's) A-1, "Water Hammer," and A-43, "Containment Emergency Sump Performance." These issues were discussed at Subcommittee meetings held on December 2-3, 1981 and May 10, 1983. The Committee also had the benefit of the referenced documents.

While we generally concur with the proposed resolution for USI's A-1 and A-43, we wish to bring the following specific points to your attention:

USI A-1

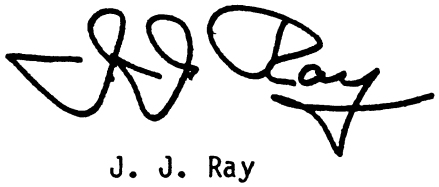
In the discussion of USI A-1, it became apparent that there may not have been sufficient attention given to the complicating effects of water hammer occurring in the course of abnormal events initiated by other causes. The scope of A-1 has been confined to evaluation of events initiated by water hammer, and the development of means to reduce the frequency and consequence of the types of events that actually have been experienced. We accept that scope for A-1, but believe that the more general question deserves attention. We would like to have a report from the NRC Staff describing the extent to which existing risk analyses have accounted for water hammer phenomena which could be postulated to render important preventive or mitigative systems inoperative or to otherwise worsen the course of transients or accidents which might threaten the public health and safety.

USI A-43

The NRC Staff review has indicated there is potential for pump failure if particulates are ingested in water-cooled bearings or seals. However, no specific review requirements for this item appear in the A-43 resolution document (NUREG-0869). We recommend that the final resolution of USI A-43 include explicit consideration of the consequences of particulate ingestion on pump performance as well as other possible effects, such as the plugging of various system's components.

We believe that the estimates of probability of pipe breaks used in the A-43 analysis are reasonable values to assume given the present state of understanding of the nature of pipe failures and recent information about the extent of cracking in certain LWR primary piping systems. Although we agree it was properly applied in this case, we believe that the probabilistic and fracture mechanics arguments concerning leak before break, which were developed in support of a proposed resolution of the asymmetric loads issue, should not be too quickly nor too generally applied to other issues without very thorough analysis and review.

Sincerely,



J. J. Ray
Chairman

References:

1. U. S. Nuclear Regulatory Commission, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants - Technical Findings Relevant to Unresolved Safety Issue A-1,"USNRC Report NUREG-0927, dated May 1983
2. U. S. Nuclear Regulatory Commission, "Value-Impact Analysis for USI A-1, 'Water Hammer,'" USNRC Report NUREG-0993, dated May 1983
3. U. S. Nuclear Regulatory Commission, "Unresolved Safety Issue A-1, 'Water Hammer,'" USNRC Report SECY-83-188, dated May 16, 1983
4. U. S. Nuclear Regulatory Commission, "USI A-43, Resolution Positions," USNRC Report NUREG-0869, dated April 1983
5. U. S. Nuclear Regulatory Commission, "Containment Emergency Sump Performance - Technical Findings Related to Unresolved Safety Issue A-43," USNRC Report NUREG-0897, dated April 1983
6. U. S. Nuclear Regulatory Commission, "Unresolved Safety Issue A-43, 'Containment Emergency Sump Performance,'" USNRC Report SECY-83-149 dated April 27, 1983
7. Memorandum from A. W. Serkiz, NRC, to K. Kniel, NRC, "Report on the A-43 Public Meetings Held on June 1-2, 1983," dated June 8, 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 18, 1983

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS COMMENTS ON THE NRC INTEGRATED PROGRAM FOR THE RESOLUTION OF UNRESOLVED SAFETY ISSUES A-3, A-4 AND A-5 RELATING TO PWR STEAM GENERATORS

During its 282nd meeting, October 13-15, 1983, the Advisory Committee on Reactor Safeguards heard a report from its Metal Components and Decay Heat Removal Systems Subcommittees and presentations from the NRC Staff regarding a proposed Integrated Program for the Resolution of USIs A-3, A-4 and A-5 relating to PWR steam generator tube integrity. During our review, we had the benefit of discussions with representatives of the Electric Power Research Institute, the Steam Generator Owners Group, and the NRC Staff. The Committee also had the benefit of the document referenced.

The Staff has proposed a list of improved practices for maintaining the integrity of steam generator tubing. We believe the proposals have considerable merit and that their implementation will be of benefit to the nuclear power industry. However, the argument for their implementation lies not in a significant reduction of risk to health and safety of the public, e.g., core melt frequency, but in a reduced rate of challenge to safety systems and in economic benefits to the consumers of electric power.

There is, thus, a question of whether the Commission wishes to impose on the industry new regulations whose main effect would be to assure the wider use of sound operating practice. We believe the proposals should not be implemented as new regulatory requirements but instead as recommended industry actions.

Sincerely,

J. J. Ray,
Chairman

Reference:

1. U.S. Nuclear Regulatory Commission, "NRC Integrated Program for the Resolution of Unresovled Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," NUREG-0844, For Comment, dated July 1983.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 22, 1983

Mr. William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

SUBJECT: GENERIC ISSUE B-10, HYDRODYNAMIC LOADS ON MARK III CONTAINMENT STRUCTURES

During its 283rd meeting, November 17-19, 1983, the ACRS heard a report from the NRC Staff on the proposed final resolution of Generic Issue B-10. This issue is concerned with the definition of hydrodynamic loads that may be experienced by Mark III containment structures during large break loss of coolant accidents. A subcommittee of the ACRS had previously reviewed work on this issue during its meetings on September 24-25, 1981, January 22, 1982, and July 29-30, 1982. In addition, we had the benefit of a draft of NUREG-0978, "Technical Evaluation Report on Mark III LOCA-Related Hydrodynamic Load Definition." The ACRS had also, within the past two years, reviewed and favorably commented on the definitions of hydrodynamic loads for three specific plants which incorporated containments of the Mark III design; namely, Grand Gulf Unit 1, Clinton Unit 1, and Perry Unit 1.

We believe the assumptions and methodology presented in draft NUREG-0978 are an appropriate basis for the generic definition of LOCA-related loads for Mark III containments. We believe that the loads defined by these means will be sufficiently conservative to provide assurance of the integrity and reliability of essential containment systems during a LOCA.

While there are some indications of large conservatisms in the load definitions, we believe it would be inappropriate to reduce such conservatisms unless supported by additional experimental and analytical work. It is our understanding that there is now little incentive for such further work because the conservative loads can be satisfactorily accommodated by the existing structures and designs for Mark III containments.

Sincerely,

J. J. Ray
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 15, 1984

MEMORANDUM FOR: William J. Dircks
Executive Director for Operations

FROM: R. F. Fraley, Executive Director, ACRS

SUBJECT: FIRST SET OF ACRS COMMENTS ON THE PRIORITIZATION OF THE REMAINING GENERIC ISSUES

During 1983, the ACRS reviewed the adequacy of the NRC Staff's assignment of priority rankings to individual generic issues and transmitted its comments to you on three different occasions through memoranda from me, dated June 20, 1983, from M. W. Libarkin, dated July 15, 1983, and again from me, dated August 16, 1983. Since then, the NRC Staff had assigned priority rankings to several of the remaining generic issues and they were sent to the ACRS for review.

During its 289th meeting, May 10-12, 1984, the ACRS reviewed the adequacy of the proposed priority rankings to a group of generic issues identified in the attached Table 1, and its comments are contained in the following attachements:

- Attachment 1 lists those items for which the ACRS agrees with the priority rankings proposed by the NRC Staff.
- Attachment 2 includes a list of items for which the ACRS agrees with the priority rankings proposed by the NRC Staff, but has comments.
- Attachment 3 contains a list of items for which the ACRS disagrees with the NRC Staff's proposed priority rankings along with the reasons therefor.

The following items are still being considered by the ACRS and comments will be included in its next report on this matter:

35. Degradation of Internal Appurtenances in LWRs

36. Loss of Service Water (Calvert Cliffs Unit 1)

58. Inadvertent Containment Flooding

80. Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments

It is requested that the NRC Staff provide written responses to the ACRS comments identified in Attachments 2 and 3. Cognizant ACRS Subcommittees or the ACRS may review the NRC Staff's responses.

The ACRS will continue its review of the adequacy of the proposed priority rankings for the remaining generic issues when they become available.

Attachments: As Stated

TABLE 1

LIST OF GENERIC ISSUES REVIEWED BY THE ACRS DURING THE 289TH, MAY 10-12, 1984, ACRS MEETING

GENERIC ISSUE NO.	TITLE	PRIORITY RANKING PROPOSED BY THE NRC STAFF
34	RCS Leak	DROP
40	Breaks in the BWR Scram System	Resolution Available
43	Contamination of Instrument Air Lines	DROP
45	Inoperability of Instrumentation Due to Extreme Cold Weather	Resolution Available
48	Limiting Conditions for Operation (LCO) for Class IE Vital Instrument Buses in Operating Reactors	Resolution Available
50	Reactor Vessel Level Instrumentation in BWRs	Resolution Identified
51	Proposed Requirements for Improving Reliability of Open Cycle Service Water Systems	MEDIUM
61	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	MEDIUM
65	Probability of Core Melt Due to Component Cooling Water System Failure	HIGH
69	Make-up Nozzle Cracking	Resolution Identified
75	Generic Implications of ATWS at the Salem Nuclear Plant	Parts of this issue had been resolved. The remaining parts had been classified as "Licensing Issue"
77	Flooding of Safety Equipment Compartments by Back-Flow Through Floor Drains	HIGH
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	MEDIUM
82	Beyond Design Basis Accidents in Spent Fuel Pools	MEDIUM

ATTACHMENT 1

LIST OF ITEMS FOR WHICH THE ACRS AGREES WITH THE PRIORITY RANKINGS PROPOSED BY THE NRC STAFF

GENERIC ISSUE NO.	TITLE
45	Inoperability of Instrumentation Due to Extreme Cold Weather
48	Limiting Conditions for Operation (LCO) for Class IE Vital Instrument Buses in Operating Reactors
50	Reactor Vessel Level Instrumentation in BWRs
61	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments
69	Make-up Nozzle Cracking
75	Generic Implications of ATWS at the Salem Nuclear Plant
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown

ATTACHMENT 2

LIST OF ITEMS FOR WHICH THE ACRS
AGREES WITH THE PRIORITY RANKINGS
PROPOSED BY THE NRC STAFF, BUT WITH COMMENTS

Issue No: 51

Title: Proposed Requirements for Improving Reliability of Open Cycle Service Water Systems

Proposed NRC Staff Priority: MEDIUM

ACRS Comments: As consideration is given to the rather gradual fouling mechanisms considered, attention should also be given to the more critical issue of "sudden" fouling of components due to release of accumulated organisms and other loosely bound materials.

Issue No: 65

Title: Probability of Core Melt Due to Component Cooling Water System Failures

Proposed NRC Staff Priority: HIGH

ACRS Comments: The Component Cooling Water (CCW) system is a treated water, closed system. There are no "standard" designs for the CCW system. Many of the functions of the CCW system have been handled by open cycle service water systems, although service water is a low-pressure, raw-water system. It is suggested that some consideration be given to maintaining the function of the CCW system via service water interties for emergency use. Trade-offs such as use of raw water (chloride) contamination of systems vs. preventing seal failures in an emergency should be explored.

Issue No: 77

Title: Flooding of Safety Equipment Compartments by Back-Flow Through Floor Drains

Proposed NRC Staff Priority: HIGH

ACRS Comments: As adverse interactions due to internal flooding are explored, it would seem appropriate to consider external event flooding in the accident scenarios. Consideration should also be given to loss of concrete inter-pour seals. The inadequacies of such seals, as well as floor drains, may be masked by pumping systems installed during the construction phase for draw-down of the site water table and are then left in continuous operation. The seals and drains are thus not verified and external flood designs based on performance of these and other design features to maintain a "dry" environment will not be adequate (a case in point is Browns Ferry).

Issue No: 82

Title: Beyond Design Basis Accidents in Spent Fuel Pools

Proposed NRC Staff Priority: MEDIUM

ACRS Comments: The priority of "MEDIUM" is the absolute maximum that could possibly be tolerated for this issue. We would find a priority of "LOW" quite acceptable.

It should be noted that the assigned priority applies only to BWR pools; because pools in PWRs are generally below grade, the staff analyst was unable to come up with any reasonably high probability of an accident beyond the design bases.

For pools in BWRs, accidents were considered involving loss of cooling, draining by siphoning or as the result of a cask drop, and draining and simultaneous opening to the atmosphere as a result of tornado missile or earthquake. As a result of very pessimistic assumptions, one on top of the other, the total probability came out to be about 3.7E-6 per reactor-year, with consequences of about 18 man-rem per reactor-year.

In view of the low consequences, even with very conservative assumptions, and in view of the fact that the Staff does not seem to have an approach about how to resolve this issue, we have reached the conclusion stated above that a priority of "MEDIUM" is acceptable if it does not mean that resources will be diverted from work toward the resolution of issues rated "HIGH," or even from most of the issues we have seen that are rated "MEDIUM."

ATTACHMENT 3

LIST OF ITEMS FOR WHICH THE ACRS
DISAGREES WITH THE PRIORITY RANKINGS
PROPOSED BY THE NRC STAFF

Issue No: 34

Title: RCS Leak

Proposed NRC Staff Priority: DROP

ACRS Comments: Before this issue is dropped, we recommend ensuring the chain of events that resulted in a small LOCA at Robinson, caused by a spurious safety injection, represents a set of circumstances with acceptable consequences at other plants (i.e., how typical is Robinson?).

Issue No: 40

Title: Breaks in the BWR Scram System

Proposed NRC Staff Priority: Resolution Available

ACRS Comments: NRC and the General Electric Company (GE) are pursuing resolution of this issue by consideration of leak-before-break, environmental qualification of affected equipment, and reliance on operator action to identify and ultimately isolate a scram system leak (break). We understand that GE will be submitting information for NRC review that supports this resolution path in the near future.

Given the above situation, we are unable to effectively comment on the proposed resolution of this issue at this time. We recommend that upon completion of its review, the NRC Staff provide their conclusions to the ACRS for review and comment.

Issue No: 43

Title: Contamination of Instrument Air Lines

Proposed NRC Staff Priority: DROP

ACRS Comments: We disagree with the NRC Staff's proposal to drop this issue. However, based on the information provided by the NRC Staff at the April 26, 1984 ACRS Subcommittee meeting on the Air Systems, we now understand that the NRC Staff plans to reevaluate the priority for this issue, after receiving the results of the study of air systems currently under way within the Office for Analysis and Evaluation of Operational Data (AEOD). We concur in this approach and look forward to receipt of the resulting reassessment.

When this reevaluation is made, we would encourage the NRC Staff to consider revising the scope of this issue; including in their analyses the impact of less severe, more frequent, accidents; factoring into the assessment the associated collective occupational, as well as public, doses; and including other economic impacts such as potentially increased plant availability and reduced maintenance costs associated with the proposed backfits.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 11, 1984

Mr. William J. Dircks
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Dircks:

SUBJECT: EPRI CATEGORIZATION OF GENERIC SAFETY AND LICENSING ISSUES

In a memorandum to Mr. R. F. Fraley dated April 20, 1984, Mr. Harold Denton, Director, Office of Nuclear Reactor Regulation, requested ACRS review and comment on the correctness of the proposed categorization of generic safety and licensing issues, identifying which issues impact future LWR standardized plant designs. This categorization results from a cooperative effort between the NRC Staff and the Electric Power Research Institute (EPRI) and is part of the EPRI LWR Standardized Plant Design Evaluation Program.

In our review of this matter during the 293rd meeting, September 6-8, 1984, we had the benefit of discussions with representatives of the NRC, EPRI, and the Atomic Industrial Forum (AIF), and of the documents referenced. We also had the benefit of a subcommittee meeting held on September 5, 1984.

We believe that the general approach of attempting to categorize safety and licensing issues into groups applicable or inapplicable to new LWR designs, as EPRI now envisages such designs, should be useful in several ways. It helps to provide a mechanism for identifying those issues warranting attention for resolution and is leading to a useful exchange of information and ideas between the NRC Staff and industry. We support continuation of this effort.

EPRI and the AIF plan to attempt to develop proposed means for resolution of the issues EPRI has identified as unresolved and applicable to new LWR designs. The EPRI program is intended to lead to accepted means of resolving currently identified issues and minimizing the chance of backfit for future LWRs. If the ambitious EPRI plan for the resolution of such issues for future LWRs is to receive adequately informed review and evaluation by the NRC Staff, a very considerable allocation of NRC resources will be required in addition to the current NRC effort on unresolved safety issues and high priority generic issues focused on plants in operation and under construction.

We wish to note a few other caveats with regard to the program:

. Many previous issues have been "resolved" by modifications to plant designs which were required to compensate for fundamental

weaknesses in the overall design approach; thus, reviewing future designs against only the outstanding issues without considering how previous issues were resolved would not disclose undesirable design compromises which should not be perpetuated.

- It is to be expected that the new LWR designs contemplated by EPRI may introduce new safety issues.

- Care is needed in tracking all portions of each issue as they are categorized, resolved, dropped, etc. to be sure that significant matters do not get misplaced or lost.

- It will be important to ascertain that issues previously treated as resolved in terms of existing designs and existing Standard Review Plan requirements do not reemerge in terms either of new design considerations or changed views of safety requirements when one goes beyond the old standard design basis accidents and the single failure criterion.

- Some matters now categorized as licensing issues may remain subject to considerable uncertainty. It is possible that research and future experience will lead to changes in requirements.

- The NRC Staff noted that an archive of all issues will be maintained by the NRC, including those that have been dropped or resolved. Such an archive should be useful, and proponents of new designs should review the archive to assure that a change in design does not restore a formerly resolved issue to one of significance.

We have not reviewed in detail the categorizations developed up to this time. The ACRS expects to examine this matter when this work is nearer to completion.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

References:

1. U. S. Nuclear Regulatory Commission, "A Prioritization of Generic Safety Issues," USNRC Report NUREG-0933, dated December 1983
2. S. Levy Incorporated, Draft Interim Report, SLI-8410, "LWR Standardized Plant Design Evaluation," dated April 1984 and Revised June 1984
3. Memorandum from Harold R. Denton, Director, Office of Nuclear Reactor Regulation, to Raymond F. Fraley, Executive Director, ACRS, Subject: Generic Safety and Licensing Issues - A Categorization to Identify Which Issues Impact LWR Standardized Future Plant Design dated April 20, 1984

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 15, 1984

Mr. William J. Dircks
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Dircks:

SUBJECT: ACRS COMMENTS ON IDENTIFICATION OF NEW UNRESOLVED SAFETY ISSUES

During its 294th meeting, October 11-13, 1984, the ACRS considered the comments provided by its various Subcommittees based on their review of the following categories of items, as requested by the NRC Staff in the memorandum from T. P. Speis to R. F. Fraley, dated June 8, 1984.

1. Adequacy of the NRC Staff's evaluation of 25 HIGH-priority generic issues and the resulting conclusion that these issues do not have the potential for designation as new Unresolved Safety Issues (USIs).

2. Existing information on the generic issues that are yet to be assigned priorities.

3. Adequacy of the updated information on certain generic issues that were considered previously by the NRC Staff for potential designation as new USIs and the associated NRC Staff's conclusions.

Our comments on the above mentioned items are as follows:

- We agree that none of the 25 remaining HIGH-priority generic issues should be designated as USIs. We previously agreed with the designation of Generic Issue No. 23, "Reactor Coolant Pump Seal Failures," as a USI, as noted in our report dated August 14, 1984.

- We prefer to defer review of the generic issues in Item 2 until they have been assigned priorities by the NRC Staff. We note, however, that delay in the process of reviewing an item may be equivalent to assigning it a low priority, since work on its resolution is effectively delayed. For this reason, we wish to obtain the NRC Staff's schedules for review of these remaining items and for resolution of those items that have already been assigned HIGH or MEDIUM priorities.

- With two possible exceptions, we agree with the NRC Staff's conclusions regarding the issues previously considered. The exceptions are those designated "Steam Generator Transients and Accidents" and "Piping and Use of Highly Combustible Gases in Vital Areas." These issues will be reviewed further by an ACRS Subcommittee and the full Committee and comments will be provided later.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

Reference:
Memoranda from Themis P. Speis, Director, Division of Safety Technology, NRC, to Raymond F. Fraley, Executive Director, ACRS, transmitting the document on "Indentification of New Unresolved Safety Issues," dated June 8, 1984 and June 29, 1984.

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE.

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG-1125, Volume 4
Generic Subjects A-G

2. TITLE AND SUBTITLE
A Compilation of Reports of the Advisory Committee on Reactor Safeguards, 1957-1984, Volume 4, Part 2: ACRS Reports on Generic Subjects (Accident Analysis - Generic Items)

3. LEAVE BLANK

4. DATE REPORT COMPLETED
MONTH: February YEAR: 1985

5. AUTHOR(S)

6. DATE REPORT ISSUED
MONTH: April YEAR: 1985

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Advisory Committee on Reactor Safeguards
US Nuclear Regulatory Commission
Washington, DC 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Same as above

11a. TYPE OF REPORT
Compilation

b. PERIOD COVERED (Inclusive dates)
September 1957-December 1984

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This six-volume compilation contains over 1000 reports prepared by the Advisory Committee on Reactor Safeguards from September 1957 through December 1984. The reports are divided into two groups: Part 1: ACRS Reports on Project Reviews, and Part 2: ACRS Reports on Generic Subjects. Part 1 contains ACRS reports alphabetized by project name and within project name by chronological order. Part 2 categorizes the reports by the most appropriate generic subject area and within subject area by chronological order.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS
Nuclear Reactors
Nuclear Reactor Safety
Nuclear Reactor Sites
Safety Engineering
Safety Research
Reactor Operations

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page) Unclassified
(This report) Unclassified

17. NUMBER OF PAGES

18. PRICE