UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-237-SP
	)	50-249-SP
COMMONWEALTH EDISON COMPANY	)	
	)	
(Dresden Stations, Units 2 and 3	)	(Spent Fuel Pool Modification)

AFFIDAVIT OF RICHARD B. HUBBARD CONCERNING BOARD QUESTION 2 (UNRESOLVED SAFETY ISSUES)

STATE OF CALIFORNIA)
) ss.
COUNTY OF SANTA CLARA)

RICHARD B. HUBBARD deposes and says under oath as follows:

I. INTRODUCTION

1. My name is Richard B. Hubbard. I am a Professional Quality Engineer licensed by the State of California, a technical consultant, and a founder (in 1976) and vice president of MHB Technical Associates, a corporation engaged in the business of technical consulting on energy and environmental issues and having its principal office at 1723 Hamilton Avenue, San Jose, California 95125. I hold a B.S. in Electrical Engineering from the University of Arizona (1960) and an M.B.A. from the University of Santa Clara (1969). I have sixteen years' experience in nuclear power plant electronics, instrumentation, and controls, including eleven years' experience in responsible managerial positions in the Nuclear Instrumentation Department (1965-1971), Atomic Power Equipment Department (1971-1975), and

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Nuclear Energy Control and Instrumentation Department (1975-1976) of General Electric Company. I am a member of the IEEE Nuclear Power Engineering standards subcommittee responsible for the preparation of Quality Assurance standards for safety-related aspects of nuclear power facilities. I have testified on safety-related aspects of nuclear power facilities as an expert witness before Nuclear Regulatory Commission Atomic Safety and Licensing Boards; before (and at the request of) the NRC's Advisory Committee on Reactor Safeguards; before the Joint Committee on Atomic Energy of the United States Congress; and before various State legislative and administrative bodies. I have also provided technical consultation to the Swedish and West German governments concerning safety-related aspects of nuclear power plants. My experience and qualifications are further described in Attachment A, which is appended to this affidavit.

2. In addition to the training, experience, and qualifications summarized above, for the past four years I, along with my co-founders of MHB Technical Associates, have devoted nearly all of our professional attention to analyzing, evaluating, and consulting with regard to the technical, economic, and environmental aspects of unresolved safety-related issues concerning nuclear power plants, including (a) the more than 100 such issues which had been identified by the Nuclear Regulatory Commission even before the March 28, 1979, accident at Three Mile Island Unit 2("TMI-2") and (b) the additional unresolved safety issues which have been identified as a result of TMI-2 and the various inquiries undertaken into that accident.

II. STATEMENT OF ISSUES

3. On January 26, 1981, the Atomic Safety and Licensing Board responsible for reviewing the proposed Dresden 2 and 3

spent fuel storage expansion application propounded Board Question 2 as follows:

> "Based on a review and analysis of the various generic unresolved safety issues under continuing study, what relevance is there, if any, to the proposed spent fuel modification? Further, what is the potential health and safety implication of any relevant issues remaining unresolved."

The purpose of this affidavit is (a) to identify thirty-four (34) generic unresolved safety issues and generic TMI issues that I believe are relevant to Board Question 2, and (b) to describe briefly why the selected issues may impact public health and safety.

III. BACKGROUND

4. The operating license for Dresden 2 was issued by the NRC on December 22, 1969, and for Dresden 3 on January 12, 1971. Initial criticality of the two nuclear plants occurred on January 7, 1970, and January 31, 1971, respectively. For the decade since Dresden 2 and 3 licenses were issued, evidence of potential inadequacies of lightwater reactors, including spent fuel pool structures, systems, and components, has been accumulating in the form of generic unresolved safety issues. The term "unresolved safety issues" refers to deficiencies in nuclear plant equipment, operating procedures, or licensing which may -- by causing nuclear accidents, making them worse when they occur, or impeding a proper response to them by plant operators -- contribute to increasing the public health and safety risk inherent in the operation of nuclear plants.

5. In December 1972, the NRC's Advisory Committee on Reactor Safeguards ("ACRS") began publishing a series of lists, periodically updated, of generic problems which are of concern for lightwater reactors. A recent list, see Table 5-1, identifies a total of over 77 such problems, of which some 25 are considered unresolved, including such important and fundamental

TABLE 5-1

ACRS GENERIC ISSUES-RESOLUTION PENDING 3/

ACRS GENERIC ITEM	RELEVANT TO PWR	BWR	PRIORITY FOR RESOLUTION ACRS	NRC
GROUP II (Resolution Pending Since December 18, 1972)				
53. Turbine Missiles	X	X	A	A-37, A-32
54. Containment Sprays	X		B	C-10
55. Pressure Vessel Failure By Thermal Shock	X		A	A-11
56. Instruments to Detect (Severe) Fuel Failure	X	X	C	-
57. Excessive Vibration	X	X	B	-
58. Non-Random Multiple Failures	X	X	A	C-13
58A. Reactor Scram Systems	X	X	A	A-9
58B. Alternating Current Sources Onsite & Offsite	X	X	A	A-35, B-56, B-57
58C. Direct Current Systems	X	X	A	A-30
59. Behavior of Reactor Fuels Under Abnormal Conditions	X	X	A	B-22
60. BWR Recirculation Pump Overspeed During LOCA		X	B	B-68
61. Seismic Scram	X	X	C	D-1
62. ECCS Capability for Future Plants	X	X	A	D-2
GROUP II A: (Resolution Pending Since February 13, 1974)				
63. Ice Condenser Containments	X		B	B-54
64. Steam Generator Tube Leakage	X		A	A-3, A-4, A-5
65. ACRS/NRC Periodic 10-year Review	X	X	C	Policy
GROUP II B: (Resolution Pending Since March 12, 1975)				
66. Computer Reactor Protection System	X		B	A-19
67. BWR Mark III Containments		X	B	A-39, B-10
68. Stress Corrosion Cracking in BWR Piping		X	B	Policy
GROUP II C: (Resolution Pending Since April 16, 1976)				
69. Locking Out of ECCS Power Operated Valves	X	X	B	B-8
70. Design Features to Control Sabotage	X	X	A	A-29
71. Decontamination	X	X	B	A-15
72. Decommissioning	X	X	B	B-64
73. Vessel Support Structures	X		B	A-2
74. Water Hammer	X	X	A	A-1
75. BWR Mark I Containments		X	A	A-6, A-7, A-39
GROUP II D: (Resolution Pending Since February 24, 1977)				
76. Capability of Hermetic Seals	X	X	C	C-1
GROUP II E: (Resolution Pending Since November 15, 1977)				
77. Soil-structure Interaction	X	X	C	A-40, A-41

safety aspects as reactor containments, reactor pressure vessels, emergency core cooling system ("ECCS") components, piping, and electrical equipment.[1/] Ten of the ACRS' problems have been listed since 1972, but are still officially unresolved. While some 52 items have been declared "resolved", it is far from clear just what that means. According to the ACRS:[2/]

> "Resolved as used in the Generic Items reports refers to the following: In some cases an item has been resolved in an administrative sense, <u>recognizing that technical evaluation and satisfactory implementation are yet to be completed</u>. Anticipated Transients Without Scram represents an example of this category. In other instances, the resolution has been accomplished in a narrow or specific sense, <u>recognizing that further steps are desirable, as practical, or that different aspects of the problem require further investigation</u>. Examples are the possibility of improved methods of locating leaks in the primary system, and improved methods or augmented scope of in-service inspection of reactor pressure vessels." (Emphasis added.)

Accordingly, even when an item has been declared "resolved", there is no assurance that a solution has in fact been implemented at <u>any</u> plant, let alone specifically for the Dresden spent fuel pool items. The ACRS recently suggested that the following three subjects be added to the list of unresolved safety issues: DC power supply reliability, single failure criterion, control system reliability.[4/] The ACRS letter is included as Attachment 2 of this affidavit. Finally, unlike the original licensing procedures for Dresden 2 and 3, the status of ACRS generic issues is described in a plant-specific evaluation in recent Staff SER's.

6. In the fall of 1976, several members of the NRC Staff identified 27 unresolved issues as "problems whose priority, progress, or resolution was in their opinion, unsatisfactory", and the NRC Commissioners directed the Staff to develop a program plan for the timely resolution of outstanding

generic issues. This process began in April 1977, when each of the four NRC Divisions reporting to the Office of Nuclear Reactor Regulation submitted a list of those generic issues the Division considered to warrant the highest priority. Proposals for 355 issues or "tasks" were received; after consolidation and elimination, a list of 133 issues was eventually developed and published in January 1978 as NUREG-0410.[5/] These 133 issues were classified, using a set of uniform criteria, into Categories A (warranting the highest priority attention), B, C, and D. Table 6-1 sets out the 41 highest-priority (Category A) issues.

7. After the issuance of NUREG-0410, the NRC also conducted a preliminary evaluation of the unresolved issues on a relative risk basis, in order to identify those issues having the greatest safety significance. The purpose of this evaluation was not to make an "absolute" risk determination, nor to decide that any issue presented an "acceptable" level of risk; rather, the object was to sort out the 133 NUREG-0410 issues on the basis of which ones had the greatest potential impact on safety. Four risk categories were developed, ranging from "high" to "none." Table 7-1 sets out the two highest priority issues - "high" risk and "low" risk items.[6/]

8. In January 1979, the NRC issued NUREG-0510 (its Report to Congress on unresolved safety issues), in which the 133 unresolved issues were once again reclassified. NUREG-0510 classified seventeen issues, consisting of 22 "tasks" to be worked on, as most important, and labeled them as the "unresolved safety issues" to be reported to Congress. [7/] Table 8-1 presents the reprioritized tabulation of issues.

TABLE 8-1
UNRESOLVED SAFETY ISSUES
AND
APPLICABLE GENERIC TASK NUMBERS[7/]

	UNRESOLVED SAFETY ISSUE:	NRC TASK NO:
1.	Water Hammer	A-1
2.	Asymmetric Blowdown Loads on the Reactor Coolant System	A-2
3.	Pressurized Water Reactor Steam Generator Tube Integrity	A-3, A-4, A-5
4.	BWR Mark I and Mark II Pressure Suppression Containments	A-6, A-7, A-8, A-39
5.	Anticipated Transients Without Scram	A-9
6.	BWR Nozzle Cracking	A-10
7.	Reactor Vessel Materials Toughness	A-11
8.	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	A-12
9.	System Interactions in Nuclear Power Plants	A-17
10.	Environmental Qualification of Safety-Related Electrical Equipment	A-24
11.	Reactor Vessel Pressure Transient Protection	A-26
12.	Residual Heat Removal Requirements	A-31
13.	Control of Heavy Loads Near Spent Fuel	A-36
14.	Seismic Design Criteria	A-40
15.	Pipe Cracks in Boiling Water Reactors	A-42
16.	Containment Emergency Sump Reliability	A-43
17.	Station Blackout	A-44

TABLE 6-1

CATEGORY A GENERIC ACTIVITIES

(Original Listing in NUREG-0410)

TASK NO:	TITLE:	RELEVANT TO BWR	RELEVANT TO PWR
A-1	WATER HAMMER	X	X
A-2	ASYMMETRIC BLOWDOWN LOADS ON THE REACTOR VESSEL	X	X
A-3	WESTINGHOUSE STEAM GENERATOR TUBE INTEGRITY		X
A-4	COMBUSTION ENGINEERING STEAM GENERATOR TUBE INTEGRITY		X
A-5	BABCOCK & WILCOX STEAM GENERATOR TUBE INTEGRITY		
A-6	MARK I SHORT-TERM PROGRAM	X	
A-7	MARK I LONG-TERM PROGRAM	X	
A-8	MARK II PROGRAM	X	X
A-9	ATWS	X	X
A-10	BWR NOZZLE CRACKING	X	
A-11	REACTOR VESSEL MATERIALS TOUGHNESS	X	X
A-12	FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS	X	X
A-13	SNUBBERS	X	X
A-14	FLAW DETECTION	X	X
A-15	DECONTAMINATION	X	X
A-16	STEAM EFFECTS ON BWR CORE SPRAY DISTRIBUTION	X	
A-17	SYSTEMS INTERACTION IN NUCLEAR POWER PLANTS	X	X
A-18	PIPE RUPTURE DESIGN CRITERIA	X	X
A-19	DIGITAL COMPUTER PROTECTION SYSTEMS	(PLANTS WITH DIGITAL COMPUTERS)	
A-20	IMPACTS OF COAL FUEL CYCLE	(ENVIRONMENTAL)	
A-21	MAIN STEAM LINE BREAK INSIDE CONTAINMENT		X
A-22	PWR MAIN STEAM LINE BREAK - CORE AND PRIMARY COOLANT BOUNDARY RESPONSE (MSLB OUTSIDE CONTAINMENT)		X
A-23	CONTAINMENT LEAK TESTING	X	X
A-24	QUALIFICATION OF CLASS IE SAFETY-RELATED EQUIPMENT	X	X
A-25	NONSAFETY LOADS ON CLASS IE POWER SOURCES	X	X
A-26	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION (OVERPRESSURE)		X
A-27	RELOAD APPLICATION GUIDE	X	X
A-28	INCREASE IN SPENT FUEL STORAGE CAPACITY	X	X
A-29	DESIGN FEATURES TO CONTROL SABOTAGE	X	X
A-30	ADEQUACY OF SAFETY-RELATED DC POWER SUPPLIES	X	X
A-31	RHR SHUTDOWN REQUIREMENTS	X	X
A-32	EVALUATION OF OVERALL EFFECTS OF MISSILES	X	X
A-33	NEPA REVIEWS OF ACCIDENT RISKS	(ENVIRONMENTAL)	
A-34	INSTRUMENTS FOR MONITORING RADIATION AND PROCESS VARIABLES DURING ACCIDENTS	X	X
A-35	ADEQUACY OF OFF-SITE POWER SYSTEMS	X	X
A-36	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	X	X
A-37	TURBINE MISSILES	X	X
A-38	TORNADO MISSILES	X	X
A-39	DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC	X	
A-40	SEISMIC DESIGN CRITERIA - SHORT-TERM PROGRAM	X	X
A-41	SEISMIC DESIGN CRITERIA - LONG-TERM PROGRAM	X	X

A. TASKS A-42, A-43, & A-44 WERE ADDED TO THE LIST IN NUREG-0510.

B. TASKS A-45 A-46, A-47, & A-48 WERE ADDED TO THE LIST IN DEC. 80

TABLE 7-1

RISK-BASED CATEGORIZATION - NRC GENERIC ISSUES [6]

- CATEGORY I: Potential High Risk Items

Group A:

TASK:	DESCRIPTION:
A-9	ATWS
A-6, A-7, A-8, A-39, B-55	Mark I, II programs, SRV pool dynamic loads and temperature limits, Target Rock Valve Reliability
A-17	Systems Interactions
A-40	Seismic Design Criteria
A-29	Design Features to Control Sabotage
A-10	BWR Nozzle Cracking
A-24	Qualification of Class IE Safety-Related Equipment

Group B:

B-57	Station Blackout Requirements
B-63	Isolation of Low Pressure Systems Connected to RCPB
B-30	Design Basis Floods and Probability
B-34	Occupational Radiation Exposure Reduction

Group C:

C-3	Insulation Usage Within Containment (sump blockage)

- CATEGORY II: Potential Low Risk Items

Group A:

A-3, A-4, A-5	W, B&W, CE Steam Generator Tube Integrity
A-1	Water Hammer
A-12	Fracture Toughness of SG/RCP Supports
A-2	Asymmetric Blowdown Loads
A-30	DC Power Supplies
A-15	Decontamination

Group B:

B-64	Decommissioning of Reactors (or parts of NSSS)

9. Also in January 1979, the NRC's Director of Nuclear Reactor Regulation (Harold Denton) and a Staff Steering Committee further reassessed the unresolved issues as part of an attempt to redirect the NRC's manpower toward the highest-priority issues.[8/] They used a system of assigning "point values" to each issue, based on a set of standardized criteria. Each of the issues was assigned a point value (ranging from 230 to 0). As a result, the top twenty issues or "tasks" (all those having from 160 to 230 points) were selected for the priority assignment of manpower, and each NRC Division was ordered to "commit resources to (them) as necessary to assure that these tasks are completed in a timely fashion." The 20 tasks selected for the priority assignment of manpower were the remaining uncompleted so-called "unresolved safety issues" and issue B-6. In addition, NRC management, at their discretion, could assign manpower to the next 24 highest priority tasks.[9/]

10. The rescheduling of the resolution of unresolved issues was once again presented by the NRC in the 1980 status report on the issues.[10/] In addition, at the end of 1980, four issues were added to the list of unresolved issues by the NRC Commissioners (see Attachment 3). The most recent summary of the issues was presented in a February 13, 1981 status report. [11/]

11. The NRC has now issued documents which provide the Staff's recommended resolution of a few of the unresolved safety issues. However, even if the NRC's resolutions prove acceptable, implementation of the resolution on operating plants such as Dresden 2 and 3 may not occur until the mid-1980's, as the NRC recently candidly acknowledged: [12/]

> "The definition of what constitutes completion of an unresolved safety issue (USI) has recently been expanded to include the implementation of the technical resolution. This is in acknowledgement of the fact that real safety benefits occur only after the implementation has taken place. Important elements of this implementation phase are:

1. The provision of a public comment period following the issuance of a draft NUREG report incorporating the Staff's technical resolution followed by a discussion and disposition of the comments received in a final NUREG report.

2. The provision for incorporation of the technical resolution into the NRC's regulations, standard review plan, regulatory guides, or other NRC official guidance or requirements as appropriate.

3. The provision for application of the technical resolution to individual operating plants in the form of hardware or design changes, technical specification change, and/or change to the operating procedures as appropriate.

Since the River Bend licensing decision in 1977, the NRC has been on notice of the need to report in the SER on the plant-specific implementation status of generic issues. 13/

12. The listing and re-listing of unresolved nuclear safety issues described in the preceding all occurred before the March 28, 1979, accident at Three Mile Island, Unit 2 ("TMI-2"). A perennial problem with such lists, which has arisen with some frequency over the history of the United States nuclear power program, is that in many cases the accidents which occur are not the ones which have been analyzed in the licensing process (though after the fact, the accident initiators or major contributors may well be found to have been somewhere on someone's list of safety concerns). This was true, for example, of the accident at Commonwealth Edison's Dresden Unit 2 in 1970 and of the similar accident at Edison's Dresden Unit 3 in 1971; 14/ it was true of the Browns Ferry Niclear Plant fire in 1975; 15/ and it was true in multiple respects of the TMI-2 accident.

13. The TMI-2 accident spawned at least nine different inquiries, including those of the NRC Special Inquiry Group 16/ and the President's Commission on the Accident at Three Mile Island (the "Kemeny Commission"). 17/ It is generally accepted that the TMI-2 accident identified numerous safety-related areas

of serious weakness and deficiency in the design, construction, operation, licensing, and regulation of nuclear plants in the United States. The accident also led to still more re-evaluation of unresolved safety issues. After the accident, Harold Denton, Director of the NRC's Office of Nuclear Reactor Regulation, briefed the NRC Commissioners (in document SECY-79-344) on the Staff's plans to continue work on the pre-TMI unresolved safety issues. Mr. Denton also anticipated (correctly) that TMI would result in expanding the scope of some of those existing issues, as well as in identifying new unresolved safety issues.

14. In May 1980, the NRC issued the TMI Action Plan (NUREG-0660).[18/] The plan was divided into five general categories: Operational safety; siting and design; emergency preparedness and radiation effects; practices and procedures; and NRC policy, organization and management. It included 176 different "tasks" -- all safety-related -- of which 58 fell in the category of siting and design. As had been done before the TMI-2 accident with regard to the NRC's list of unresolved safety issues, the 176 "tasks" identified in the TMI Action Plan were given priority rankings on a "point value" basis. This priority ranking is shown in Tables 1 and B.2 of NUREG-0660; Table B.1 of NUREG-0660 shows the "point" system which was used. From Table B.1 one can determine that, though others may also fall in this group, any Action Plan "task" with a point value greater than 160 is necessarily considered by the NRC to have high safety significance.

15. The TMI items that the NRC has approved for implementation at operating reactors, such as Dresden 2 and 3, are included in NUREG-0737, which was issued in November, 1980. [19/]
In addition, on December 18, 1980, the NRC Commissioners revised its earlier Statement of Policy to provide further clarification and guidance concerning the procedures for assessing TMI issues in individual licensing proceedings.[20/]

It is difficult to summarize the importance and priorities of the specific TMI tasks without addressing each one in turn. However, that is not the purpose of this affidavit. The descriptions in Section IV will, however, place the TMI tasks I have selected in perspective and will provide the reasons why the selected issues appear to be priority items for the Dresden 2 and 3 spent fuel storage expansion.

16. The history of the NRC's handling of unresolved nuclear safety issues, briefly summarized in the preceding paragraphs, is not calculated to inspire confidence that those issues will be resolved in an adequate and timely fashion, or that solutions to the many issues which directly affect the safety of the expanded spent fuel storage facility will be adequately implemented at Dresden. In Part IV of this Affidavit, I discuss these problems in greater detail, with regard to thirty-four of the safety issues which the NRC itself has identified as (a) known, (b) unresolved, and (c) of high priority.

IV. DISCUSSION OF SELECTED ISSUES

17. Pursuant to the conference call of March 13, 1981, between Applicant, Staff, Intervenor, and the Board, Intervenor is briefly outlining in the following its proposed list of generic unresolved safety issues which should be addressed in response to Board Question 2. Also provided is a brief explanation of how these issues are relevant to the proposed spent fuel pool modification.

18. The issues identified herein are directly related to the public health and safety and to the issues being addressed in this proceeding. All of the selected issues are directly related to the function of safety-related structures, systems, and components. Enclosure 1 of NRC witness Belke's testimony on Contention 2 provides a listing of those items to which the CECO QA system applies and which have been designated by CECO as Class I (equipment, material, systems

and structures which can have a first order effect on nuclear safety). Included in this listing are the spent fuel pool, spent fuel storage facilities, storage equipment which includes the spent fuel storage racks and tube assemblies, emergency electrical power and instrument control air systems, area monitoring system, and the primary containment inerting system. The required systems for maintaining the safety of the spent fuel pool are further described in the testimony of CECO witness Adams on Contentions 1 and 4.

19. Based on a review of the following U.S. Nuclear Regulatory Commission documents: "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," NUREG-0410 (January 1978); "Generic Task Problem Description," NUREG-0471 (June 1978); " Identification of Unresolved Safety Issues Relating to Nuclear Power Plants," NUREG-0510 (January 1979); "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants," NUREG-0649 (February 1980); US NRC letter, Dircks to Chilk,December 24, 1980, entitled "SECY-80-325, Special Report to Congress Identifying Unresolved Safety Issues;" the following generic issues are relevant to the proposed spent fuel pool modification.

20. Task A-17, Systems Interaction In Nuclear Power Plants. The issue arises because the design and analysis of spent fuel systems and storage racks is assigned to teams with functional engineering specialties -- such as civil, electrical, mechanical, or nuclear. The question is whether the work of these functional specialties is sufficiently integrated in their design and analysis activities to enable them to identify adverse interactions between and among systems. Such adverse events might occur, for example, because designers did not assure that redundancy and independence of safety equipment were provided under all conditions of operation required, which might happen if the functional teams were not adequately coordinated. Task A-17 is a Category A issue, a "high-risk" issue, and an ACRS issue.

21. Task A-24, Qualification Of Class 1E Safety-Related Equipment. The equipment for the Dresden 2 and 3 spent fuel pools was purchased and installed prior to the first issuances of the IEEE Standard 323-1971 which provided environmental qualification requirements for Class 1E equipment. A number of aspects of equipment qualification are being pursued at this time by the NRC Staff on a generic basis to achieve a uniform implementation of the requirements established in the subsequent revision of the Standard (IEEE 323-1974). Task A-24 is one of these activities. It involves the development of Staff positions to form the basis for licensing reviews of equipment qualification programs.

22. Task A-28, Increase In Spent Fuel Pool Storage Capacity. This task involves the development of consistent and formalized acceptance criteria regarding the use of high density storage racks in existing spent fuel storage pools. Revisions of current NRC guidelines are being developed that incorporate insights gained in the case-by-case reviews of numerous past applications for increased spent fuel storage pool capacity. This task involves documenting and formalizing the acceptance criteria currently being used by the NRC for the review of applications for increased spent fuel storage capacity at nuclear power plants and applying the knowledge gained to the Dresden 2 and 3 proposed expansion.

23. Task A-29, Design Features to Control Sabotage. The Dresden spent fuel pools are accessible on a controlled basis during plant operation. The objective of this task is to identify and evaluate possible plant design variations which could improve the inherent sabotage resistance of the pools. For current plants high assurance of protection against industrial sabotage is achieved by the physical security measures required by 10 CFR 73.55 rather than by design measures.

24. Task A-30, DC Power Supplies. This generic task

originated from a letter to the NRC's Advisory Committee on Reactor Safeguards from one of its consultants that questioned the reliability of DC power supplies at nuclear power stations. If all sources of DC power were lost, continued cooling of the reactor core and spent fuel pool cannot be assured. The ACRS in 1980 again expressed concern about this issue (see Attachment 2).

25. Task A-34, Instruments For Monitoring Radiation and Process Variables During Accidents. Contention 4 in this proceeding addresses accident monitoring instrumentation including those required following an accident. The purpose of this task is to develop criteria and guidelines to be used by licensees and staff reviewers to support implementation of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," which was extensively revised following the TMI accident. The revised criteria and guidelines provide specific guidance on functional and operational capabilities required of the various classes of instruments.

26. Task A-36, Control Of Heavy Loads Near Spent Fuel. Contention 6 of this proceeding addresses spent fuel handling accidents which is the subject of this task. NUREG-0612 which was issued in 1980 resolved this task. However, the implementation at Dresden, in accordance with letters issued December 22, 1980, and revised by letter on February 2, 1981, remains to be determined.

27. Task A-40, Seismic Design Criteria. There are a number of plants, such as Dresden 2 and 3, with operating licenses issued before the NRC's current seismic regulations and regulatory guides were in place. For this reason, re-review of seismic design of the existing spent fuel equipment is necessary to assure that the old designs do not present an undue public risk.

28. Task A-42, Pipe Cracks In Boiling Water Reactors. Pipe cracking has occurred in the heat affected zones of welds in primary system piping in BWR's since the mid-1960's. These cracks have occurred mainly in Type 304 stainless steel that is being used in most operating BWR's. The major problem is recognized to be intergranular stress corrosion cracking (IGSCC). Revision 1 of NUREG-0313 sets forth the NRC staff's revised guidelines for reducing the IGSCC susceptibility of BWR piping. The guidelines describe a number of preventive and corrective measures acceptable to the NRC, including guidelines for: (1) corrosion resistant materials for installation in BWR piping, (2) methods of testing, (3) processing techniques, (4) augmented in-service inspection, and (5) leak detection. Resolution for Dresden 2 and 3 remains to be determined.

29. Task A-44, Station Blackout. Electric power for safety systems is supplied by redundant and independent divisions. Each of these electrical divisions includes an offsite alternating current (A.C.) source, an onsite A.C.source (diesel generators), and a direct current (D.C.) source. The unlikely, but possible loss of all A.C. power (that is, the loss of A.C. power from the offsite source and from the onsite source) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor would be dependent on the availability of systems which do not require A.C. power supplies, and on the ability to restore A.C. power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event and that the consequences of this event may be unacceptable. In ALAB-603, the appeal board ruled that in some cases station blackout must be considered a design basis event.

30. Task A-46, Seismic Qualification Of Equipment In Operating Plants. The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change as

described in paragraph 27. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this unresolved safety issue 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. This guidance will concern equipment required to safely shutdown the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions such as spent fuel pool cooling. The NRC has yet to prepare a plan and schedule for the resolution of this task.

31. Task A-47, Safety Implication Of Control Systems. Contention 6 of this proceeding involves this issue which concerns the potential for accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident under consideration and would be in addition to any control system failure that may have initiated the event. Although it is generally believed that control system failures are not likely to result in loss of safety functions which could lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed to support this belief. In addition, the NRC has yet to prepare a plan and schedule for the resolution of this task.

32. Task A-48, Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment. Postulated reactor accidents which result in a degraded or melted core can result in generation and release to the containment and potentially to the spent fuel area through the vessel head vent system of large quantities of hydrogen. The hydrogen is formed from the reaction of the zirconium fuel cladding with steam at high

temperatures and/or by radiolysis of water. Experience gained from the TMI-2 accident indicates that the NRC may require more specific design provisions for handling larger hydrogen releases than currently required by the regulations particularly for smaller, low pressure containment designs such as Dresden's Mark I containment. This issue will investigate in part various means to cope with large releases of hydrogen to the containment such as inerting of the containment (as at Dresden) or controlled burning. The potential effects of proposed hydrogen control measures on safety including the effects of hydrogen burns on safety related equipment will also be investigated. No plan and schedule for resolution of this issue has been prepared by the NRC.

33. Task B-34, Occupational Radiation Exposure Reduction. Contention 5 of this proceeding addresses this issue which in the NRC's risk based categorization was determined to be a "high-risk" item. This task involves the development of additional criteria and guidelines to provide the basis for the NRC to review the spent fuel design and operations to support full implementation of the NRC's regulatory requirement that radiation exposures should be maintained as low as is reasonably achievable.

34. Task B-67, Effluent And Process Monitoring Instrumentation. This issue relates to Contention 4 in this proceeding. The task involves reviewing gaseous and liquid effluent monitoring systems for old operating plants, such as Dresden 2 and 3, to determine their effectiveness in meeting the effluent release limits of 10 CFR Parts 20 and 50.

35. Based on a review of U.S. NRC "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, Vol. 1 and Vol. 2 (May, 1980) and U.S. NRC, "Clarification of TMI Action Plan Requirements," NUREG-0737 (November, 1980), the following additional issues appear to be relevant to this spent fuel pool modification:

36. Item I.D.1, Control-Room Design Reviews. The ability of the operator to control the spent fuel pool systems during and following accidents is relevant to the issues covered by Contention 6, and parts of Contention 4 in this proceeding. The objective of this item is to improve the ability of nuclear power plant control-room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them. The NRC has requested the Dresden licensee to perform a detailed control room design review to identify and correct deficiencies. This review will include an assessment of control room layout, the adequacy of the information provided, the arrangement and identification of important controls and instrumentation displays, the usefulness of the audio and visual alarm systems, the information recording and recall capability, lighting, and other considerations of human factors that have an impact on operator effectiveness.

37. Item III.D.3.4., Control Room Habitability. The NRC will follow a two-step approach to assure that workers are adequately protected from radioactivity, radiation, and other hazards, and that the control room can be used in the event of an emergency. First, NRC will require all old facilities, such as Dresden 2 and 3, that have not been reviewed for conformance to Regulatory Guides 1.78 and 1.95 and Standard Review Plan Sections 2.2.1, 2.2.2, 2.2.3, and 6.4 to do the evaluations and establish a schedule for necessary modifications. Then, NRC will examine and evaluate other sources and pathways of radioactivity and radiation that may lead to control room habitability problems. This is an extension to the task described in paragraph 36.

38. Item II.B.1, Reactor-Coolant-System Vent. The NRC has required (a) the installation of high-point reactor coolant system and reactor vessel head vents in the spent fuel pool area that are remotely operable from the control room; (b) analysis of loss-of-coolant accidents initiated by a break in the vent pipe; and (c) analyses demonstrating that direct

venting of noncondensable gases with perhaps a high hydrogen concentration limit does not result in violation of combustible gas concentration limits in the containment structure. The vents are to provide the ability to deal effectively with the unexpected presence of noncondensable gases in the reactor vessel and primary coolant system, particularly in quantities that could interfere with coolant flow and distribution, by establishing a safe vent path (Also see paragraph 32).

39. Item II.B.2., Plant Shielding. Plant shielding is necessary to provide access to vital areas and protect safety equipment for postaccident operation. The NRC has required (a) a radiation and shielding design review of spaces around systems in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by radiation during operation following an accident and (b) implementation of identified plant modifications that will permit access to vital areas and protect safety equipment. Therefore, the spent fuel pool portion of the Dresden shielding evaluation should be reviewed as part of Contention 6 in this proceeding.

40. Item II.B.3, Post-Accident Sampling. In an issue related to the plant shielding presented in paragraph 39, the NRC has required the Dresden licensee to conduct (a) a review of the reactor coolant and containment atmosphere sampling systems and the radiological spectrum and chemical analysis facilities; (b) describe implementation of modifications necessary to permit personnel to obtain samples within 1 hour after an accident, to analyze samples within 2 hours for radioactive noble gases, iodines, cesiums, and nonvolatile isotopes, to analyze samples within 1 hour for boron, and to analyze for chlorides within a shift; and (c) prepare procedures for analyzing these samples with existing equipment. The adequacy of the review as it relates to access to the spent fuel area should be addressed.

41. Item II.B.7, Analysis of Hydrogen Control. The Dresden containment is inerted to prevent the structure being

overpressurized during a severe accident. It may be appropriate to use features and procedures other than inerting to cope with the generation of hydrogen and particularly to enable increased in-containment access for maintenance personnel during plant operation.

42. Item II.F.1, Additional Accident-Monitoring Instrumentation. The objective of this task is to provide instrumentation to monitor plant variables and systems during and following an accident. Indications of plant variables and status of systems important to safety are required by the plant operator (licensee) during accident situations. Requirements for additional accident monitoring instrumentation were submitted to operating reactor licensees in NRC letters dated September 13, and October 30, 1979 (Also see paragraph 25.).

43. Item I.F.1., Expand QA Test, and Item I.F.2., Develop More Detailed QA Criteria. The NRC is developing more detailed criteria for various aspects of quality assurance for design, construction, and operations. The existing NRC criteria, criteria formed after the Dresden project, are general and allow broad interpretation. Detailed guidance is needed to clarify NRC requirements for the QA function in design, construction, and operations. In addition, the NRC is developing guidance for licensees to expand their QA lists to cover equipment important to safety and rank the equipment in order of its importance to safety. The results of the Interim Reliability Evaluation Program (IREP) and the systems interactions tasks will be used to establish the importance of equipment as it relates to safety. These issues are relevant to Contentions 2 and 3 in this proceeding.

44. Item II.C.3., Systems Interaction. The purpose of this item is to coordinate and expand ongoing NRC work on systems interaction (Unresolved Safety Issue A-17) so as to incorporate it into an integrated plan for addressing the broader question of system reliability in conjunction with IREP and other efforts. Both analytical techniques, such as failure modes and effects analysis, event-trees, and fault-trees and physical

techniques, such as system walkdowns, are in the process of being implemented.

45. Item II.F.5., Classification of Instrumentation, Control and Electrical Equipment. The NRC, in conjunction with IEEE, has prepared a standard that provides a classification approach for determining the applicability of design criteria and design requirements for nuclear power plant systems, based on the level of their importance to safety. The standard sets forth criteria for determining the level of importance to safety of the instrumentation, control, and electrical portions of nuclear power plant systems. Methods are provided to determine the design basis for each of these systems and to determine the degree of applicability of the requirements of other standards to each of these systems, with such determination to be based on the level of importance to safety of each system. This item is relevant to Contentions 2 and 6 in this proceeding.

46. Based on a view of Chairman Carbon's letter to NRC Chairman Hendrie on March 21, 1979; and ACRS Chairman Plesset's letter to NRC Chairman Ahearne on August 12, 1980, these following generic issues have been determined to be relevant to the proposed fuel pool notification:

47. Item 58, Non-Random Multiple Failures. The issue of non-random multiple failures is relevant to Contention 6. In the past, 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; 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 (Also see paragraph 20.).

48. Item 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 initial findings of the NRC's evaluation of older plants as part of the NRC's Systematic Evaluation Program (SEP) for the eleven oldest reactors, including Dresden 2, should be addressed to the extent that they apply to the spent fuel pool systems.

49. Item 68, Stress Corrosion Cracking in BWR Piping. 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 (Also see paragraph 28.).

50. Item 70, Design Features To Control Sabotage. As discussed in paragraph 23, 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, however, proposed that deliberate attention should be given to aspects of design that could improve plant security.

51. Item 77, Soil Structure Interactions. Ongoing studies by the NRC 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 (Also see paragraph 27.).

52. New Item 1, DC Power Supply Reliability. For details, see paragraph 24.

53. New Item 2, Single Failure Criterion. The NRC's safety evaluation for the spent fuel expansion uses 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.

54. New Item 3, Control System Reliability. According to the ACRS, recent experience has indicated that more attention must be given to control system reliability. Past NRC safety analyses 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. 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. (Also see paragraph 31.).

V. CONCLUSIONS

55. Based on the foregoing discussion and background information, I conclude that the thirty-four (34) generic unresolved safety issues and generic TMI issues identified in this affidavit are appropriate issues to be addressed in response to Board Question 2. Further, I conclude that these unresolved issues may be important to public health and safety, both singularly and cumulatively. Finally, I conclude that lists of generic safety issues have existed for many years. What is needed now are decisions by the NRC, and a timely implementation of the selected solution by Commonwealth Edison.

I have read the foregoing and swear that it is true and accurate to the best of my knowledge.

Richard B. Hubbard

Richard B. Hubbard

Subscribed and sworn to before
me this 17th day of March, 1981

Linda L. Roberson

Notary Public

My commission expires: 8-29-83

OFFICIAL SEAL
LINDA L ROBERSON
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My comm. expires AUG 29, 1983

LIST OF REFERENCES

1. Letter from M Bender, ACRS Chairman, to J. Hendrie, NRC Chairman, Nov. 17, 1977, titled "Status of Generic Items Relating to Light-Water Reactors: Report No. 6." (An updated report was issued March 21, 1979).

2. Bender letter, _supra_ note 1, p.2.

3. Letter from M. Carbon, ACRS Chairman, to J. Hendrie, NRC Chairman, March 21, 1979, titled "Status of Generic Items Relating to Light-Water Reactors: Report No. 7."

4. Letter from M. Plesset, ACRS Chairman, to J. Ahearne, NRC Chairman, August 12, 1980, titled "New Unresolved Safety Issues."

5. NUREG-0410, _NRC Program For The Resolution Of Generic Issues Related To Nuclear Power Plants_ (USNRC, Washington, D.C., January 1978.)

6. NUREG-0510, _Identification of Unresolved Safety Issues Relating to Nuclear Power Plants, Report to Congress_ (USNRC, Washington, D .C., January 1979). Appendix C for risk-based tabulation.

7. _Ibid_., 6.

8. See NRC Document SECY-79-76 (January 30, 1979), a memorandum from Harold R. Denton, Director, Office of Nuclear Reactor Regulation, to the NRC Commissioners.

9. Memorandum from Harold R. Denton to Roger S. Boyd, _et al_., January 23, 1979 (attached to SECY-79-76, _supra_ note 8) pp. 1-2 and Enclosure 1.

10. NUREG-649, _Task Action Plans For Unresolved Safety Issues Related To Nuclear Power Plants_ (USNRC, Washington, D.C., February, 1980.).

11. NUREG-0606, Vol. 3, No. 1, _Unresolved Safety Issues Summary_ USNRC, Washington, D.C., February 13, 1981).

12. _Ibid_., 11.

13. _Gulf States Utilities Co._ (River Bend Station, Units 1 & 2), ALAB-444, 6 N.R.C. 760, 767-68 (1977).

14. Kendall, _et al_., _The Risks of Nuclear Power Reactors_, (Union of Concerned Scientists, Cambridge, Massachusetts, August, 1977. See Appendix B.)

15. Browns Ferry Nuclear Plant Fire, Hearings Before the Joint Committee on Atomic Energy (US Gov't Printing Office, Washington, D.C., 1975).

16. NUREG/CR-1250, Three Mile Island, A Report To The Commissioners And To The Public, NRC Special Inquiry Group (USNRC, Washington, D.C., January 1980.)

17. The Need for Change: The Legacy of TMI, Report of the President's Commission on the Accident at Three Mile Island (U.S. Gov't Printing Office, Washington, D.C., October 1979)

18. NUREG-0660, NRC Action Plan Developed As A Result Of The TMI-2 Accident, Vols. I, II (USNRC, Washington, D.C., May 1980).

19. NUREG-0737, Clarification of TMI Action Plan Requirements, (USNRC, Washington, D.C., November, 1980.)

20. Statement of Policy: Further Commission Guidance For Power Reactor Operating Licenses, NRC Commissioners Memorandum and Order CLI-80-42, December 18, 1980.

LIST OF ATTACHMENTS

Attachment	Description
1.	Qualifications of Richard B. Hubbard
2.	New Unresolved Safety Issues, ACRS, August 12, 1980, Letter to NRC Chairman
3.	Dircks Letter of December 24, 1980

ATTACHMENT 1

Qualifications of
Richard B. Hubbard

PROFESSIONAL QUALIFICATIONS OF RICHARD B. HUBBARD

RICHARD B. HUBBARD
MHB Technical Associates
1723 Hamilton Avenue
Suite K
San Jose, California 95125
(408) 266-2716

EXPERIENCE:

9/76 - PRESENT

Vice-President - MHB Technical Associates, San Jose, California. Founder, and Vice-President of technical consulting firm. Specialists in independent energy assessments for government agencies, particularly technical and economic evaluation of nuclear power facilities. Consultant in this capacity to Oklahoma and Illinois Attorney Generals, Minnesota Pollution Control Agency, German Ministry for Research and Technology, Governor of Colorado, Swedish Energy Commission, Swedish Nuclear Inspectorate, and the U.S. Department of Energy. Also provided studies and testimony for various public interest groups including the Center for Law in the Public Interest, Los Angeles; Public Law Utility Group, Baton Rouge, Louisiana; Friends of the Earth (FOE), Italy; and the Union of Concerned Scientists, Cambridge, Massachusetts. Provided testimony to the U.S. Senate/House Joint Committee on Atomic Energy, the U.S. House Committee on Interior and Insular Affairs, the California Assembly, Land Use, and Energy Committee, the Advisory Committee on Reactor Safeguards, and the Atomic Safety and Licensing Board. Performed comprehensive risk analysis of the accident probabilities and consequences at the Barseback Nuclear Plant for the Swedish Energy Commission and edited, as well as contributed to, the Union of Concerned Scientist's technical review of the NRC's Reactor Safety Study (WASH 1400).

2/76 - 9/76

Consultant, Project Survival, Palo Alto, California. Volunteer work on Nuclear Safeguards Initiative campaigns in California, Oregon, Washington, Arizona, and Colorado. Numerous presentations on nuclear power and alternative energy options to civic, government, and college groups. Also resource person for public service presentations on radio and television.

5/75 - 1/76

Manager - Quality Assurance Section, Nuclear Energy Control and Instrumentation Department, General Electric Company, San Jose, California.
Report to the Department General Manager. Develop and implement quality plans, programs, methods, and equipment which assure that products produced by the Department meet quality requirements as defined in NRC regulation 10 CFR 50, Appendix B, ASME Boiler and Pressure Vessel Code, customer contracts, and GE Corporate policies and procedures. Product areas include radiation sensors, reactor vessel internals, fuel handling and servicing tools, nuclear plant control and protection instrumentation systems, and nuclear steam supply and Balance of Plant control room panels. Responsible for approximately 45 exempt personnel, 22 non-exempt personnel, and 129 hourly personnel with an expense budget of nearly 4 million dollars and equipment investment budget of approximately 1.2 million dollars.

11/71 - 5/75

Manager - Quality Assurance Subsection, Manufacturing Section of Atomic Power Equipment Department, General Electric Company, San Jose, California.
Report to the Manager of Manufacturing. Same functional and product responsibilities as in Engagement #1, except at a lower organizational report level. Developed a quality system which received NRC certification in 1975. The system was also successfully surveyed for ASME "N" and "NPT" symbol authorization in 1972 and 1975, plus ASME "U" and "S" symbol authorizations in 1975. Responsible for from 23 to 39 exempt personnel, 7 to 14 non-exempt personnel, and 53 to 97 hourly personnel.

3/70 - 11/71

Manager - Application Engineering Subsection, Nuclear Instrumentation Department, General Electric Company, San Jose, California.
Responsible for the post order technical interface with architect engineers and power plant owners to define and schedule the instrumentation and control systems for the Nuclear Steam Supply and Balance of Plant portion of nuclear power generating stations. Responsibilities included preparation of the plant instrument list with approximate location, review of interface drawings to define functional design requirements, and release of functional requirements for detailed equipment designs. Personnel supervised included 17 engineers and 5 non-exempt personnel.

12/69 - 3/70

Chairman - Equipment Room Task Force, Nuclear Instrumentation Department, General Electric Company, San Jose, California. Responsible for a special task force reporting to the Department General Manager to define methods to improve the quality and reduce the installation time and cost of nuclear power plant control rooms. Study resulted in the conception of a factory-fabricated control room consisting of signal conditioning and operator control panels mounted on modular floor sections which are completely assembled in the factory and thoroughly tested for proper operation of interacting devices. Personnel supervised included 10 exempt personnel.

12/65 - 12/69

Manager - Proposal Engineering Subsection, Nuclear Instrumentation Department, General Electric Company, San Jose, California. Responsible for the application of instrumentation systems for nuclear power reactors during the proposal and pre-order period. Responsible for technical review of bid specifications, preparation of technical bid clarifications and exceptions, definition of material list for cost estimating, and the "as sold" review of contracts prior to turnover to Application Engineering. Personnel supervised varied from 2 to 9 engineers.

8/64 - 12/65

Sales Engineer, Nuclear Electronics Business Section of Atomic Power Equipment Department, General Electric Company, San Jose, California.
Responsible for the bid review, contract negotiation, and sale of instrumentation systems and components for nuclear power plants, test reactors, and radiation hot cells. Also responsible for industrial sales of radiation sensing systems for measurement of chemical properties, level, and density.

10/61 - 8/64

Application Engineer, Low Voltage Switchgear Department, General Electric Company, Philadelphia, Pennsylvania.
Responsible for the application and design of advanced diode and silicon-controlled rectifier constant voltage DC power systems and variable voltage DC power systems for industrial applications. Designed, followed manufacturing and personally tested an advanced SCR power supply for product introduction at the Iron and Steel Show. Project Engineer for a DC power system for an aluminum pot line sold to Anaconda beginning at the 161KV switchyard and encompassing all the equipment to convert the power to 700 volts DC at 160,000 amperes.

9/60 - 10/61

GE Rotational Training Program

Four 3-month assignments on the GE Rotational Training Program for college technical graduates as follows:

a. Installation and Service Eng. - Detroit, Michigan.
Installation and startup testing of the world's largest automated hot strip steel mill.

b. Tester - Industry Control - Roanoke, Virginia.
Factory testing of control panels for control of steel, paper, pulp, and utility mills and power plants.

c. Engineer - Light Military Electronics - Johnson City, New York.
Design of ground support equipment for testing the auto pilots on the F-105.

d. Sales Engineer - Morrison, Illinois.
Sale of appliance controls including range timers and refrigerator cold controls.

EDUCATION:

Bachelor of Science Electrical Engineering, University of Arizona, 1960.

Master of Business Administration, University of Santa Clara, 1969.

PROFESSIONAL AFFILIATION:

Registered Quality Engineer, License No. QU805, State of California.

Member of Subcommittee 8 of the Nuclear Power Engineering Committee of the IEEE Power Engineering Society responsible for the preparation and revision of the following 4 national Q.A. Standards:

a. IEEE 498 (ANSI N45.2.16): Supplementary Requirements for the Calibration and Control of Measuring and Test Equipment used in the Construction and Maintenance of Nuclear Power Generating Stations.

PROFESSIONAL AFFILIATION: (Contd)

b. IEEE 336 (ANSI N45.2.4): Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment during the Construction of Nuclear Power Generating Stations.

c. IEEE 467 (ANSI 45.2.14): Quality Assurance Program Requirements for the Design and Manufacture of Class IE Instrumentation and electric Equipment for Nuclear Power Generating Stations.

d. IEEE Draft: Requirements for Replacement Parts for Class IE Equipment Replacement Parts for Nuclear Power Generating Stations.

PERSONAL DATA:

Birth Date: 7/08/37
Married; three children
Health: Excellent

PUBLICATIONS AND TESTIMONY:

1. In-Core System Provides Continuous Flux Map of Reactor Cores, R.B. Hubbard and C.E. Foreman, Power, November, 1967.

2. Quality Assurance: Providing It, Proving It, R.B. Hubbard, Power, May, 1972.

3. Testimony of R.B. Hubbard, D.G. Bridenbaugh, and G.C. Minor before the United States Congress, Joint Committee on Atomic Energy, February 18, 1976, Washington, DC. (Published by the Union of Concerned Scientists, Cambridge, Massachusetts.) Excerpts from testimony published in Quote Without Comment, Chemtech, May, 1976.

4. Testimony of R.B. Hubbard, D.G. Bridenbaugh, and G.C. Minor to the California State Assembly Committee on Resources, Land Use, and Energy, Sacramento, California, March 8, 1976.

5. Testimony of R. B. Hubbard and G.C. Minor before California State Senate Committee on Public Utilities, Transit, and Energy, Sacramento, California, March 23, 1976.

6. Testimony or R.B. Hubbard and G.C. Minor, Judicial Hearings Regarding Grafenrheinfeld Nuclear Plant, March 16 & 17, 1977, Wurzburg, Germany.

PUBLICATIONS AND TESTIMONY: (Contd)

7. Testimony of R.B. Hubbard to United States House of Representatives, Subcommittee on Energy and the Environment, June 30, 1977, Washington, DC, entitled, Effectiveness of NRC Regulations - Modifications to Diablo Canyon Nuclear Units.

8. Testimony of R.B. Hubbard to the Advisory Committee on Reactor Safeguards, August 12, 1977, Washington, DC, entitled, Risk Uncertainty Due to Deficiencies in Diablo Canyon Quality Assurance Program and Failure to Implement Current NRC Practices.

9. The Risks of Nuclear Power Reactors: A Review of the NRC Reactor Safety Study WASH-1400, Kendall, et al, edited by R.B. Hubbard and G.C. Minor for the Union of Concerned Scientists, August, 1977.

10. Swedish Reactor Safety Study: Barsebäck Risk Assessment, MHB Technical Associates, January 1978 (Published by Swedish Department of Industry as Document DSI 1978:1).

11. Testimony of R.B. Hubbard before the Energy Facility Siting Council, March 31, 1978, in the matter of Pebble Springs Nuclear Power Plant, Risk Assessment: Pebble Springs Nuclear Plant, Portland, Oregon.

12. Presentation by R.B. Hubbard before the Federal Ministry for Research and Technology (BMFT), August 31 and September 1, 1978, Meeting on Reactor Safety Research, Risk Analysis, Bonn, Germany.

13. Testimony by R.B. Hubbard, D.G. Bridenbaugh, and G.C. Minor before the Atomic Safety and Licensing Board, September 25, 1978, in the matter of the Black Fox Nuclear Power Station Construction Permit hearings, Tulsa, Oklahoma.

14. Testimony of R.B. Hubbard before the Atomic Safety and Licensing Board, November 17, 1978, in the matter of Diablo Canyon Nuclear Power Plant Operating License Hearings, Operating Basis Earthquake and Seismic Reanalysis of Structures, Systems, and Components, Avila Beach, California.

15. Testimony of R.B. Hubbard and D.G. Bridenbaugh before the Louisiana Public Service Commission, November 19, 1978, Nuclear Plant and Power Generation Costs, Baton Rouge, Louisiana.

16. Testimony of R.B. Hubbard before the California Legislature, Subcommittee on Energy, Los Angeles, April 12, 1979.

PUBLICATIONS AND TESTIMONY: (Contd)

17. Testimony of R.B. Hubbard and G.C. Minor before the Federal Trade Commission, on behalf of the Union of Concerned Scientists, Standards and Certification Proposed Rule 16 CFR Part 457, May 18, 1979.

18. ALO-62, Improving the Safety of LWR Power Plants, MHB Technical Associates, prepared for U.S. Department of Energy, Sandia National Laboratories, September, 1979, available from NTIS.

19. Testimony by R.B. Hubbard before the Arizona State Legislature, Special Interim House Committee on Atomic Energy, Overview of Nuclear Safety, Phoenix, AZ, September 20, 1979.

20. "The Role of the Technical Consultant," Practising Law Institute program on "Nuclear Litigation," New York City and Chicago, November, 1979. Available from PLI, New York City.

21. Uncertainty in Nuclear Risk Assessment Methodology, MHB Technical Associates, January, 1980, prepared for and available from the Swedish Nuclear Power Inspectorate, Stockholm, Sweden.

22. Italian Reactor Safety Study: Caorso Risk Assessment, MHB Technical Associates, March, 1980, prepared for and available from Friends of the Earth, Rome, Italy.

23. Development of Study Plans: Safety Assessment of Monticello and Prairie Island Nuclear Stations, MHB Technical Associates, August, 1980, prepared for and available from the Minnesota Pollution Control Agency.

24. Affidavit of Richard B. Hubbard and Gregory C. Minor before the Illinois Commerce Commission, In the Matter of an Investigation of the Plant Construction Program of the Commonwealth Edison Company, prepared for the League of Woman Voters of Rockford, Illinois, November 12, 1980, ICC Case No. 78-0646.

25. Systems Interaction and Single Failure Criterion, MHB Technical Associates, November, 1980, prepared for and available from the Swedish Nuclear Power Inspectorate, Stockholm, Sweden.

ATTACHMENT 2

New Unresolved Safety Issues, ACRS, August 12, 1980, letter to NRC Chairman.

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

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month 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.

ATTACHMENT 3

Letter, S. Chilk to W. Dircks, subject: SECY-80-325, Special Report to Congress Identifying Unresolved Safety Issues (Commissioner Action Item), December 24, 1980.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C.

OFFICE OF THE
SECRETARY

December 24, 1980

Cornell
Rehm
Michelson
Shapar
Dooly

MEMORANDUM FOR: William J. Dircks, Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: SECY-80-325 - SPECIAL REPORT TO CONGRESS IDENTIFYING UNRESOLVED SAFETY ISSUES (COMMISSIONER ACTION ITEM)

This is to advise you that the Commission (with all Commissioners concurring, except Commissioner Gilinsky as noted below) has made the following decisions with respect to the subject staff paper:

1. The Commission has approved <u>only</u> these new Unresolved Safety Issues (USI's):

   - Shutdown Decay Heat Removal Requirements
   - Safety Implication of Control Systems (including steam generator and reactor overfill transients)
   - Seismic Qualification of Equipment in Operating Plants
   - Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

   Commissioner Gilinsky would have preferred the addition of the following as a USI: Steam Line Break with Small LOCA. The other Commissioners did not agree.

2. The 1980 Annual Report should include a brief discussion of each of the new USIs. (MPA)

3. In the future, the staff is requested to use the Office of Policy Evaluation's (OPE's) proposed screening and selection criteria for making final decisions on candidate issues identified for further study and proposal of candidate USI issues. For ready reference, see attached memorandum of November 25, 1980 from the Director, OPE to Chairman Ahearne).

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