SESSION I

INTRODUCTION TO THE NUCLEAR AIR CLEANING CONFERENCE: OBJECTIVES, NUCLEAR WASTE MANAGEMENT, RISK ASSESSMENT, AIR CLEANING FAILURES, AND STANDARDS FOR AIR AND GAS CLEANING INSTALLATIONS

Monday, August 7, 1978
CHAIRMAN: M. W. First

WELCOME

Melvin W. First
Harvard Air Cleaning Laboratory
Harvard School of Public Health
Boston, Massachusetts

My first pleasant duty is to welcome you on behalf of the U.S. Department of Energy and the Harvard Air Cleaning Laboratory, co-sponsors of this 15th Nuclear Air Cleaning Conference, and to express the wish that you will enjoy your stay in Boston. This conference started as the United States Energy and Development Administration Air Cleaning Conference and followed federal departmental reorganization by first becoming the 15th DOE Air Cleaning Conference and then the 15th DOE Nuclear Air Cleaning Conference when, much to our astonishment, we began to receive offers of papers on the subject of pulverized coal fly ash.

I hardly need tell you that a conference of this magnitude does not occur without the continuing efforts of many people. I know that I express your thanks as well as my own to the several members of the program committee of this conference who worked hard and well to put together the excellent groups of papers that we will listen to over the next several days and whose duties will not be completed until they have finished chairing the many technical sessions. The committee members are: W.L. Anderson, Clifford A. Burchsted, Harry L. Ettinger, Dade W. Moeller, Ronald R. Bellamy, John T. Collins, Humphrey Gilbert, Dallas Pence, Russ Brown, Jack C. Dempsey, Louis L. Kovach, and Richard D. Rivers.

Although this conference has always had a strong U.S. flavor because of the official sponsorship, we are delighted once again to greet and give a special welcome to those who join us from other countries. We have people here from Britain, from France, from West Germany, from Belgium, from Spain, from Sweden, from Saudi Arabia, from Australia, from Japan, and, although I have a hard time thinking of these people as from a foreign country, we have a fine delegation of our friends from Canada.

When I opened the Conference two years ago I expressed special concern that the opponents of nuclear energy were out-talking us by a considerable margin and reaching the small club groups as well as the national media very effectively with their strong "anti" messages. I am pleased to say I think we have done much better during the past two years to bring a positive message to the world about the benefits of nuclear energy as well as its safety and freedom from detrimental environmental impact. This is not to say we could not do much better, and I certainly hope we all will make a special effort to do so. Therefore, our current concerns have shifted to the topics we will discuss at this session. These are research and development in reactor safety, regulation to avoid accidents, and perhaps the most widely discussed topic of today, the matter of waste disposal. We are especially fortunate to have two keynote speakers of great stature to review the subject of nuclear research for us. The first speaker is from the Department of Energy, the second speaker from the Nuclear Regulatory Commission.

Our speaker from the Department of Energy is Mr. Alex Perge who received a chemical engineering degree from Case Institute of Technology in 1944 and then worked at K-25, the original Oak Ridge gaseous diffusion plant. He participated in A-bomb tests Able and Baker at Bikini and worked for General Electric at the Knowles Atomic Power Laboratory in fuel development and sodium purification systems for what eventually became the SIR and the Seawolfe submarine reactors. At Knowles Atomic Power Laboratory he was involved in the development of the spent fuel processing system that was used in the Savannah River plant and he worked for the Atomic Energy Commission in the design, construction, and start-up of the Savannah River reprocessing plants. In addition, he worked in Washington for the Atomic Energy Commission where he was with the Production Division's development programs. He also worked for the Operational Safety Division and was concerned with the safety of all of the AEC's materials processing operations. In 1971, he organized the Atomic Energy Commission's division of waste management and transportation and has followed along with that program through all of the organizational changes from AEC to ERDA to DOE. As of May 1 of this year, Mr. Perge became the Manager of the Special Staff for the Director of the Office of Nuclear Waste Management.

Our second keynote speaker is Mr. Saul Levine. He is a graduate of the U.S. Naval Academy and in addition has degrees from MIT in electronics and nuclear engineering. He spent his early career in the submarine service. He was project officer for the USS Enterprise under Admiral Rickover so we know he had a very rigorous instruction period. He was responsible for directing all technical, financial, production, and administrative aspects of the reactor plant prototypes and the production plants for the first nuclear powered aircraft carrier, the Enterprise. From 1958 to 1962 he was with the Polaris missle system under Admiral Rayborne where he managed design integration, installation testing, and performance evaluation of the Polaris navigation system. Next, he was Assistant Director for Reactor Technology in the Division of Reactor Licensing, United States Atomic Energy Commission, where he was responsible for directing the development of nuclear safety review techniques for nuclear reactors, requirements for safety research and development, and technical safety reviews for reactors of all types. From 1970 to 1972 he was Assistant Director, Division of Environmental Affairs of the Atomic Energy Commission where he managed programs related to environmental impact associated with AEC's programs and assisted in the establishment of requirements for the implementation of NEPA in the AEC. From 1972 to 1975, Mr. Levine was Project Staff Director of the Reactor Safety Study of the United States Atomic Energy Commission. With Professor Rasmussen of MIT he provided the principal technical and management direction of the study entitled, An Assessment of Accident Risks in U.S. Commercial Nuclear Reactor Plants, a study with which I know all of you here are well familiar. From 1973 to 1975 he was Special Assistant to the Director of the Division of Reactor Safety Research. Most recentlv he has been Deputy Director, and is presently Director of research of the United States Nuclear Regulatory Commission. In this assignment, he manages a research program to confirm assessments used by the Commission in regulating the commercial uses of nuclear energy, in particular, the areas of nuclear safeguards and environmental aspects.

NUCLEAR WASTE MANAGEMENT AT DOE

Alex F. Perge
Office of Nuclear Waste Management
Department of Energy

The Department of Energy (DOE) is an organization that came into existence 10 months ago in October 1977. Most of the people came from two agencies, the Energy Research and Development Administration and the Federal Energy Agency. However, functions and a smaller number of people came from several other agencies. The agency operates with several functional components that are headed by Assistant Secretaries. I'll list the ones that are involved in efforts such as yours.

The Director of Energy Research is responsible for all basic research and also has primary responsibility in the Department for waste management policy. The Director is responsible for four national laboratories; they are, Brookhaven National Laboratory, Argonne National Laboratory, Lawrence Berkeley Laboratory, and the Ames Laboratory.

The Assistant Secretary for Energy Technology is responsible for the programs concerned with development of energy technology. That is, the transition from basic research to useable technology. This Assistant Secretary is responsible for the development programs for: Fossil Fuels, Geothermal Energy, Solar Energy, Fusion Energy, Nuclear Energy, and the Nuclear Waste Management Program. This Assistant Secretary is also responsible for five laboratories; i.e., Pacific-Northwest Laboratory, Oak Ridge National Laboratory, Savannah River Laboratory, Idaho National Engineering Laboratory, and the Hanford Engineering Development Laboratory.

The Assistant Secretary for Resource Applications is responsible for the commercialization of energy technology after the technology has been demonstrated. Additionally, this Assistant Secretary is responsible for a number of major operations; e.g., the uranium enrichment program.

The Assistant Secretary for Defense Programs is responsible for the operations and development programs concerned with the national defense effort. This Assistant Secretary is responsible for three national laboratories; i.e., Los Alamos Scientific Laboratory, Lawrence Livermore Laboratory, and Sandia Laboratory.

The Assistant Secretary for Environment is responsible for programs concerned with health, safety, and environmental aspects of all of the DOE programs.

The Office of Nuclear Waste Management was separated from the Nuclear Technology Program on May 1, 1978, and made a separate entity reporting directly to the Assistant Secretary for Energy Technology. This Office is responsible for most of the Agency's programs of concern to this conference.

The present DOE programs for airborne wastes are supporting the development of technologies for both defense and commercial needs. Technologies are being developed for the concentration, recovery, immobilization, storage, and monitoring of airborne wastes from the defense programs fuel cycle or the nuclear power fuel cycle. Nuclear reactors, irradiated fuel storage, fuel reprocessing, and weapons-related activities all produce airborne radioactive wastes.

The DOE operations are varied by type and scale. There are production operations and laboratory operations. There are reactors, chemical plants, metallurgical operations and mechanical operations. These extend in scale from very small to extremely large. The nature of the gaseous and airborne waste streams from these operations vary. There are large volumes of dry, moist, and wet gases which vary in nature from inert to very reactive. The temperatures and pressures vary from very low to very high resulting from either normal conditions or accident conditions; and, of course, there is a variation from radioactive gases to radioactive particulates.

DOE is responsible for interim storage for some of these wastes and for the disposal for most of them. Of the wastes that have to be managed, a significant part are a result of treatment systems and devices for cleaning gases. The best treatment devices do not always yield the best waste forms for either storage or disposal.

The long term waste management objectives place minimal reliance on surveillance and maintenance. Thus, the concerns about the chemical, thermal and radiolytic degradation of wastes require technology for converting the wastes to forms acceptable for long term isolation.

The strategy of the DOE airborne radioactive waste management program is to increase the service life and reliability of filters; to reduce filter wastes; and in anticipation of regulatory actions that would require further reductions in airborne radioactive releases from defense program facilities, to develop improved technology for additional collection, fixation, and long-term management of gaseous wastes.

Available technology and practices are adequate to meet current health and safety standards. The program is aimed primarily at cost effective improvements, quality assurance, and the addition of new capability in areas where more restrictive standards seem likely to apply in the future.
The current activities and objectives are as follows:

1) Filter Service Life Extension:

   - To reduce costs and waste, iodine absorbents are being developed that have substantially longer service life and can be regenerated.

   - More durable filters are being developed for service in off-gas streams from radioactive processes that emit acid vapors or high temperature gases.

   - Several prefilter concepts are being developed for capturing radioactive particulates near their source. These concepts will extend the service life of HEPA filters and thus reduce the costs for their replacement and their disposal. Costs of concern are not just monetary but also radiation exposure.

2) Gas Monitoring:

   - Improved monitoring systems are being demonstrated for specific long lived airborne radioactive constituents in the off-gas of waste treatment processes, particularly iodine-129, carbon-14 and tritium.

   - Improved particle measurement technology is being developed for high temperature applications.

3) Gaseous Waste Treatment:

   - o Technology is being developed to remove tritium from air and water, to immobilize it and to store it safely.

   - o The state-of-the-art for limiting emissions of iodine-129, krypton-85, and carbon-14 is being assessed. Studies are being made on the adaptation of advanced technology for use at DOE sites.

4) Filter Quality Assurance:

   - o The operation of three filter test stations for testing HEPA filters prior to use in DOE facility filtration systems is continuing. The Hanford Station is being upgraded.

   - o Improved filter test methods are being developed for the DOE radioactive iodine and particle removal filtration systems. This work also supports the development of filtration standards, regulations, and test procedures.

Effective use of the manpower and resources allocated to airborne waste management is our objective. We need:

- o Matching of program tasks to both specific and general waste management needs; i.e., why is the technology required?

- o Early review of technical, legal, and economic feasibility of proposed and existing programs--in the context of process, environmental, regulatory, and storage/disposal constraints.

- o Establishment of a relationship to other development activities; i.e., other Federal, private, and international activities.

- o Application of common criteria for decisionmaking at both the development and application stages of airborne waste management.

To aid in this, we are setting up a lead organization at the Idaho Operations Office to integrate the airborne radioactive waste management program. This office will review the program elements, analyze their technologies and objectives for technical merit, range of applicability, and for consistency with DOE program goals.

Before closing I would like to mention two more points that may be of interest to you.

At the direction of the President, an Interagency Review Group, commonly called the IRG, was formed to review all aspects of radioactive waste management. Every Government agency that has any tie, in any way, to any part of waste management, is involved. The IRG is charged with reporting back to the President by October 1, 1978, with a comprehensive evaluation of nuclear waste management. The IRG will recommend to the President what has to be done, when it should be done, and who has to do what to get it done.

Even though we feel that we have been working closely with the Environmental Protection Agency and the Nuclear Regulatory Commission in the past, the IRG review has caused the three agencies to work closer together. This should be of help to the nuclear air cleaning program.

There is a significant involvement in both the program and the attendance at this meeting from the international scene; that is, non-U.S. We recognize that airborne waste management is of international concern. Though there are problems of various types in international activities, we, as the U.S., and the DOE as one of the concerned U.S. agencies, have been working toward international cooperation. The pace is slow but I believe there is progress.

I'd like to leave with a repetition of an earlier comment. Air cleaning systems produce wastes. The costs of those wastes--for treatment (if required) and disposal--will be increasing very significantly.

DISCUSSION

FIRST: When are we going to get that stuff buried?

PERGE: For most of it, not very soon. However, the IRG program will be presenting the President with a broadly based and detailed evaluation of the present programs, plans, and alternatives. This will allow the administration to give Congress a program for resolving the waste management problem. Congress has indicated a desire to do its part in resolving this matter and the administration's recommendations, including a waste plan, have been promised to Congress by the end of the year. I can only hope that this process turns out to be the start of finally resolving the waste management problem.

I might add that the IRG program includes involvement to some extent of just about every identifiable special interest group, both technical and nontechnical, government (federal & state) and industrial, etc.

RISK-ASSESSMENT TECHNIQUES AND THE REACTOR LICENSING PROCESS

Saul Levine
Director, Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

The principal subject of my talk is one that is of great interest to the nuclear community: the NRC's efforts to stabilize the reactor licensing process. I will start with a brief description of the Reactor Safety Study (WASH-1400), concentrating on the engineering aspects of the contribution to reactor accident risks. I will then go on to describe how we have applied the insights and techniques developed in this study to prepare a program, requested by Congress, to improve the safety of nuclear power plants. Finally, I will describe some new work we are just beginning on the application of risk-assessment techniques to stabilize the reactor licensing process.

An overview of the Reactor Safety Study is shown in Figure 1, and I am going to concentrate on step 3: fission-product source released from the containment. To define accidents in reactors, it is necessary to define the ways in which they can happen (i.e., accident sequences) and to assign probability values to those accidents. A set of processes is occurring within the reactor, and one must determine from those processes what fission products would be released from the containment. If we stop at this point, we can concentrate on the engineering insights that are applicable to reactor safety.

Figure 2 shows a typical simplified scheme that we use to define accident sequences. It starts with an initiating event, such as a pipe break or a transient, with which is associated a probability. What are the things that can work or fail that affect the course of events, given that initial failure? Of course, in the design-basis accident that is normally analyzed in the regulatory process one starts with the initial event, and all the engineered safety features that are provided are assumed to work. The result is a very small release of radioactivity with a probability that is the same as that of the pipe break. It is, of course, possible for any or all of these engineered safety features to fail. In fact, if the electrical power fails, none of the other systems can work. Therefore, the product $P_1 \times P_2$ gives a very large release of radioactivity: if there is a pipe break, no emergency core-cooling system, no fission-product removal, and no containment, a large amount of radioactivity will clearly be released. For instance, even if electrical power is available, the system can fail. There are then two alternatives: if the fission-product removal system works, there is one kind of release and probability, and if it does not work, there is a different kind of release and probability. These probabilities can be multiplied together if the conditional probabilities between the two events are taken into account.

BASIC SEVEN TASKS IN REACTOR SAFETY STUDY

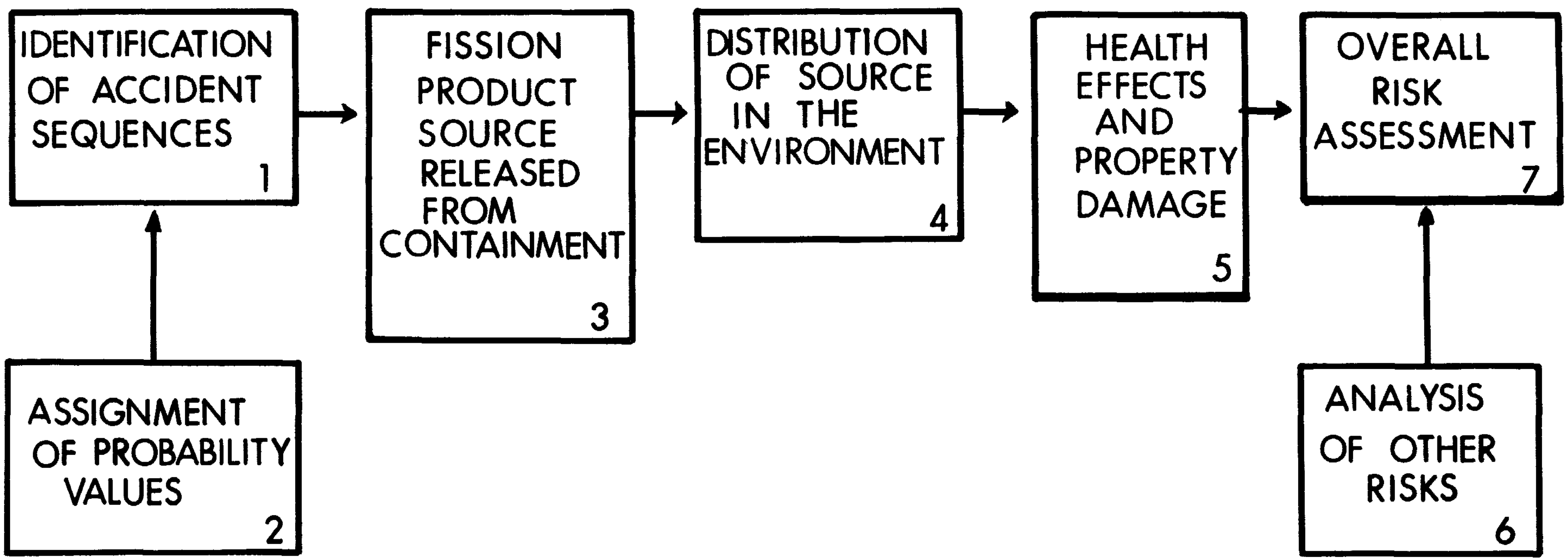


FIGURE 1

SIMPLIFIED EVENT TREE FOR A LOCA IN A TYPICAL NUCLEAR POWER PLANT

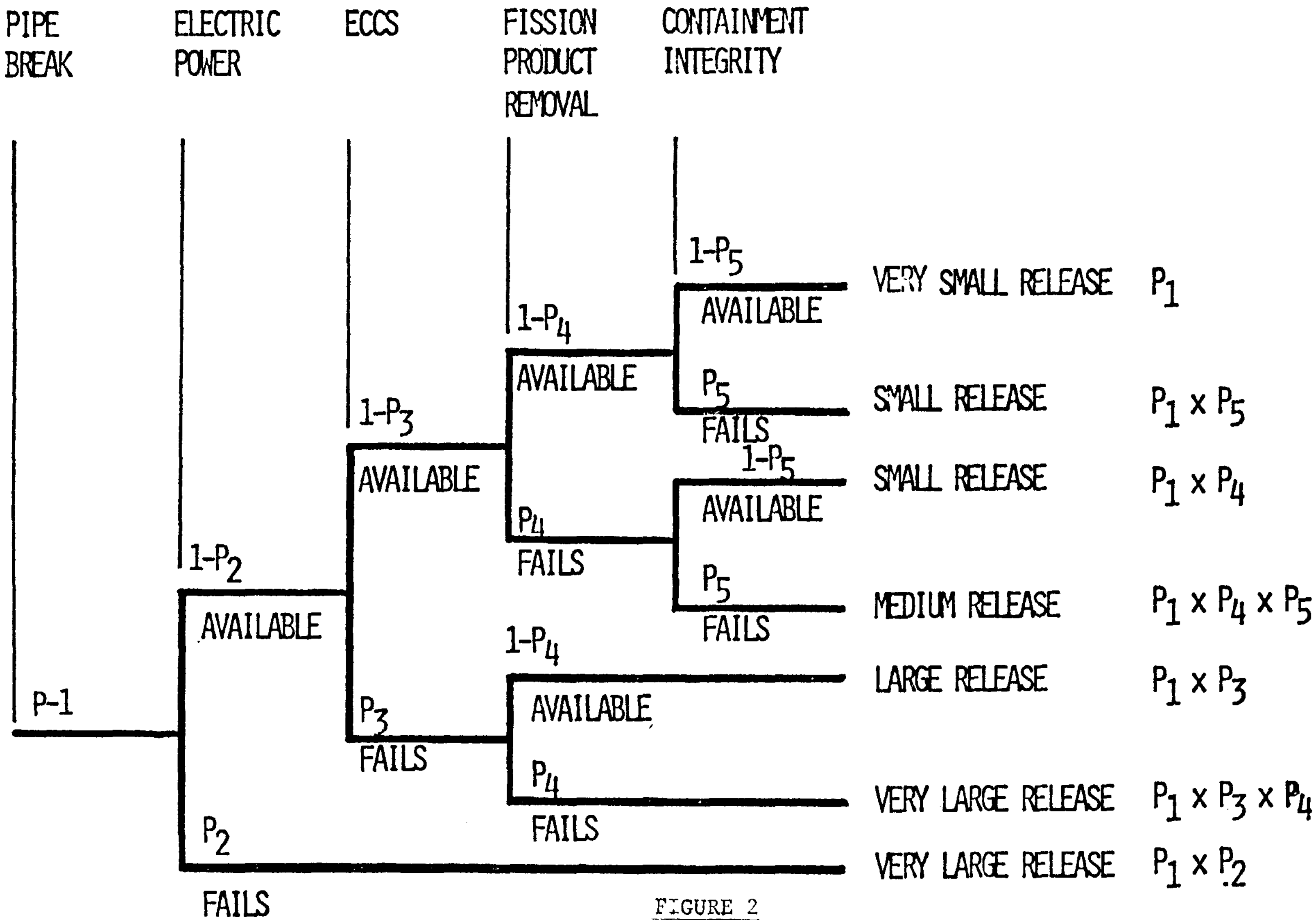


FIGURE 2

Figure 3 shows how the accident consequences can be categorized by size of release and then added within each category to get a histogram of consequences versus probability. This type of histogram was used to report the results of the Reactor Safety Study.

In making a risk assessment it is necessary to determine where to start. The most important factor is to prevent melting of the fuel because most of the radioactivity resides within the fuel. There are, however, only two ways to melt the fuel: there are loss-of-coolant events in which there is a loss of coolant that is not restored, and there are transient events in which the fuel is overpowered or the coolant flow is reduced to the point where the fuel melts.

The potential accidents analyzed in the Reactor Safety Study can be summarized as follows:

A. Potential accidents that could involve the reactor:

   1. Event trees for events involving many systems

      a. Large loss-of-coolant accident (LOCA): breaks >6 inches in equivalent diameter

      b. Small LOCA 1: breaks 2 to 6 inches in equivalent diameter

      c. Small LOCA 2: breaks 0.5 to 2 inches in equivalent diameter

      d. Reactor-vessel rupture

      e. Transient events

   2. External forces--earthquakes, tornadoes, floods, aircraft impacts, turbine missiles, tidal waves.

   3. Sabotage.

B. Noncore accidents--spent-fuel storage pool and shipping casks, waste storage tanks, refueling operations.

We can see from the above that event trees are drawn for three different kinds of LOCA (large, small 1, and small 2) because they all have different combinations of systems to operate in case of need. In our study we considered all of these as carefully as we could except for sabotage because we did not know how to estimate the probability of successful sabotage.

Figure 4 shows the fundamental structure of a pressurized water reactor (PWR), and Figure 5 shows a reactor--a pressurized water reactor or a boiling water reactor--in which the main coolant pipes are broken. It is undergoing a LOCA. This simplified drawing shows the heart of the analysis. The emergency core-cooling water is starting to be pumped into the core to prevent core melting. What melts the core in the absence of water is the decay heat in the core.

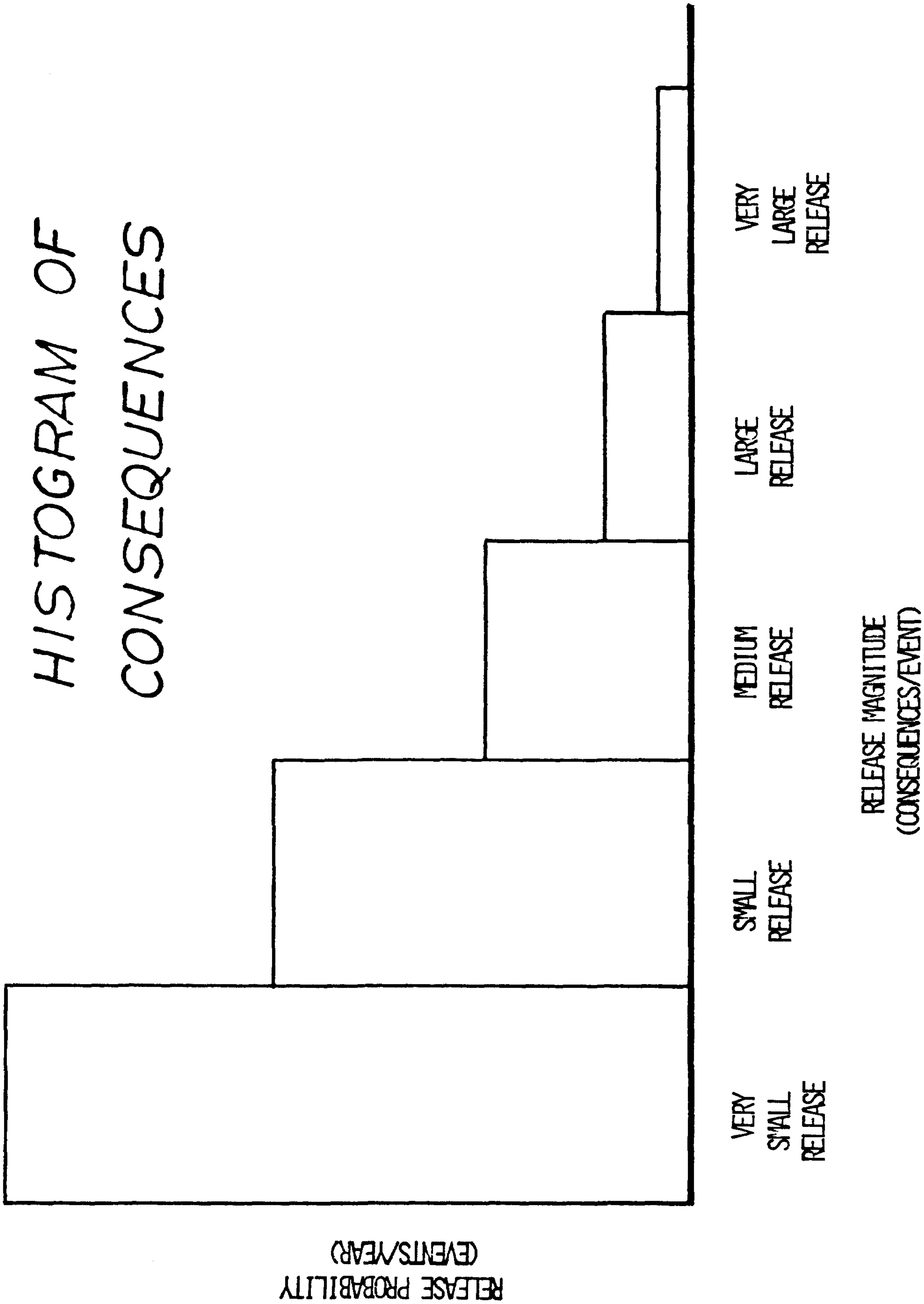

HISTOGRAM OF CONSEQUENCES
VERY SMALL RELEASE
SMALL RELEASE
MEDIUM RELEASE
LARGE RELEASE
VERY LARGE RELEASE
RELEASE MAGNITUDE (CONSEQUENCES/EVENT)
RELEASE PROBABILITY (EVENTS/YEAR)

FIGURE 3

TYPICAL PRESSURIZED WATER REACTOR

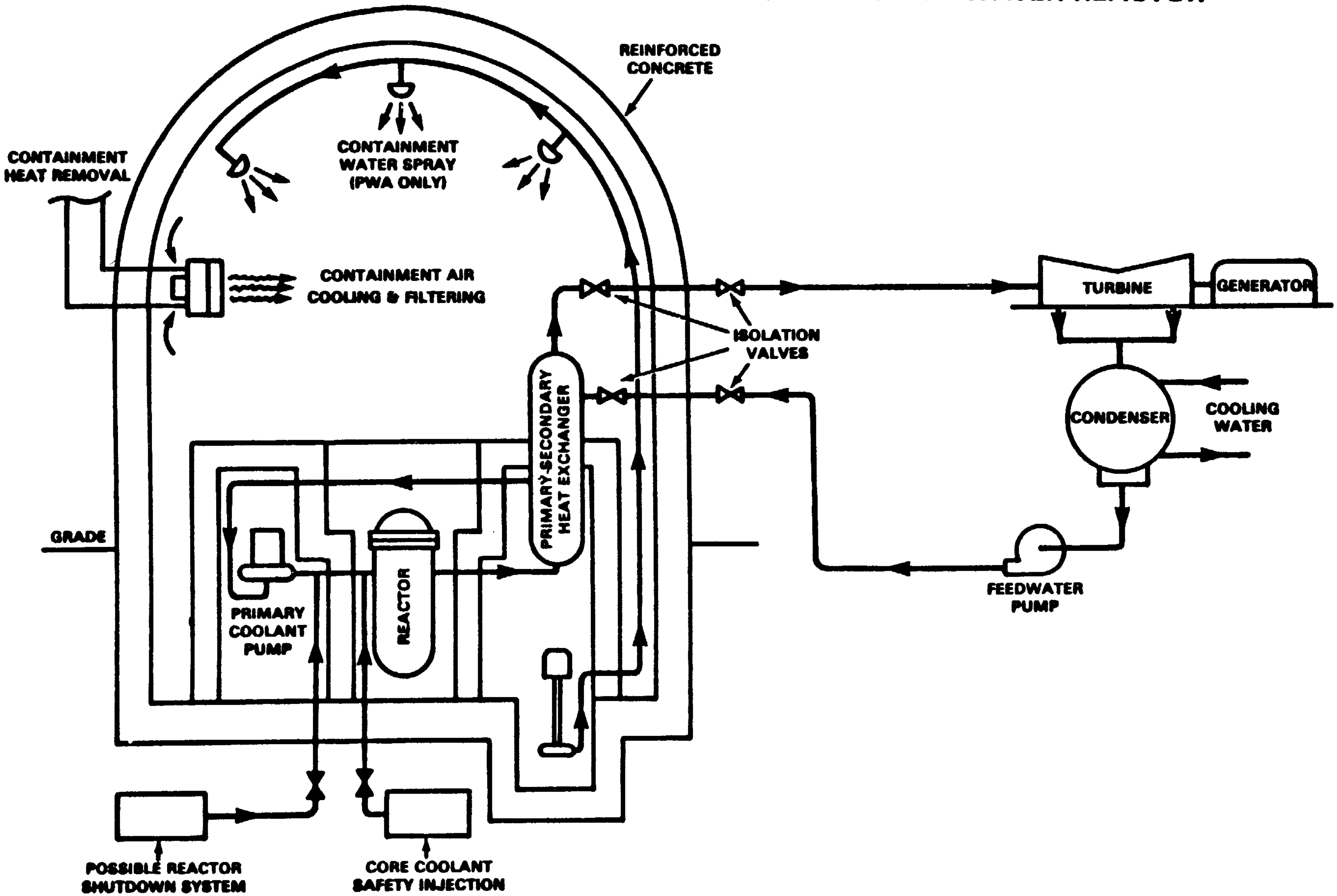
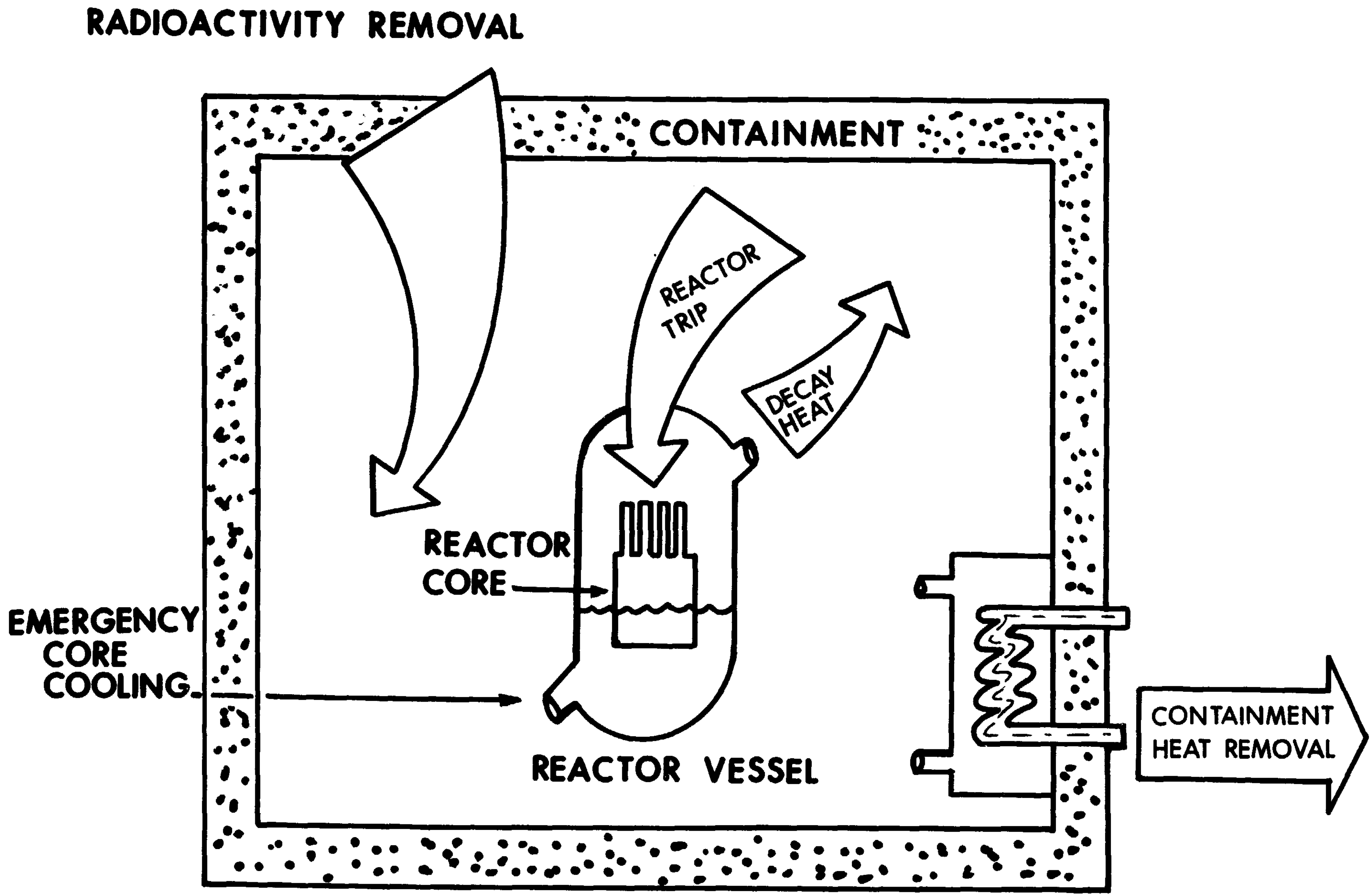


FIGURE 4



FUNCTIONS OF ENGINEERED SAFETY FEATURES

FIGURE 5

The emergency core-cooling water is removing the decay heat and transferring it into the containment. The containment is sized to withstand the pressure and temperature associated with the blowdown of the stored energy in the primary coolant water. Constantly pumping decay heat into the containment will overpressure the containment or overheat it until it ruptures. A heat-removal system is therefore necessary to remove the decay heat from the containment. It is not enough to have the emergency core-cooling system work; heat must be removed from the containment because the containment is essentially an insulated box of steel and concrete. Obviously, if the heat is not removed from the containment, the containment will rupture, at about 120 psi, or twice its design pressure. The temperature will be in the neighborhood of 400°F and the emergency core-cooling system will then fail because it will be in the recirculation mode, where it is sucking water from the containment sump; this will in fact occur at 120 psi and 400°F, and when the pressure is relieved, the water will boil, the pump will cavitate and burn out, and the core will melt. It is possible to have an accident, therefore, in which first the containment ruptures and then the core melts--which is a rather bad accident. Reactor trip is also necessary to shut down thereafter in most accidents because when the core is reflooded, it will go critical again and generate power, which will defeat our purpose. And, of course, there are the radioactivity-removal system sprays and/or filters and the containment itself. These are not systems--these are simply functions to be performed.

Figure 6 is an event tree that is perfectly general, that shows all the functions with all possible combinations of success or failure. After going through some functional logic we can arrive at Figure 7, which shows not a symmetrical tree but a highly skewed tree that takes into account the dependencies and functional failures; hence the size of the tree has been reduced enormously, which is very important. We now have a tree that defines the functions and relationships among them. To quantify this tree we put systems along the top line and then construct fault trees for the systems to define the probability of failure. Figure 8 shows that for the functions along the top of the tree there are sets of systems that must perform those functions and that there are logical interrelationships among those systems, among the functions in fact, that must be taken into account. Taking into account the logic of the functional tree and the logical interrelationships with the systems, it is possible to draw another tree (Figure 9) that has the systems along the top and a correct representation of their interrelationships. It is important to know that, if one had not performed this removal of interrelationships in the correct way, the tree in Figure 9 would have contained 30,000 accident sequences. The fact that it has been reduced to 38 sequences means that there are "what if" questions about common-mode failures between systems that do not need answers.

Figure 9 has one of two outcomes. Either the core does not melt, such as in the design-basis accident, or enough things fail and the core does melt. It is necessary to define the relationship between the molten core and the containment-failure modes because the containment failure mode determines the release to the environment.

ILLUSTRATIVE EVENT TREE FOR LOCA FUNCTIONS

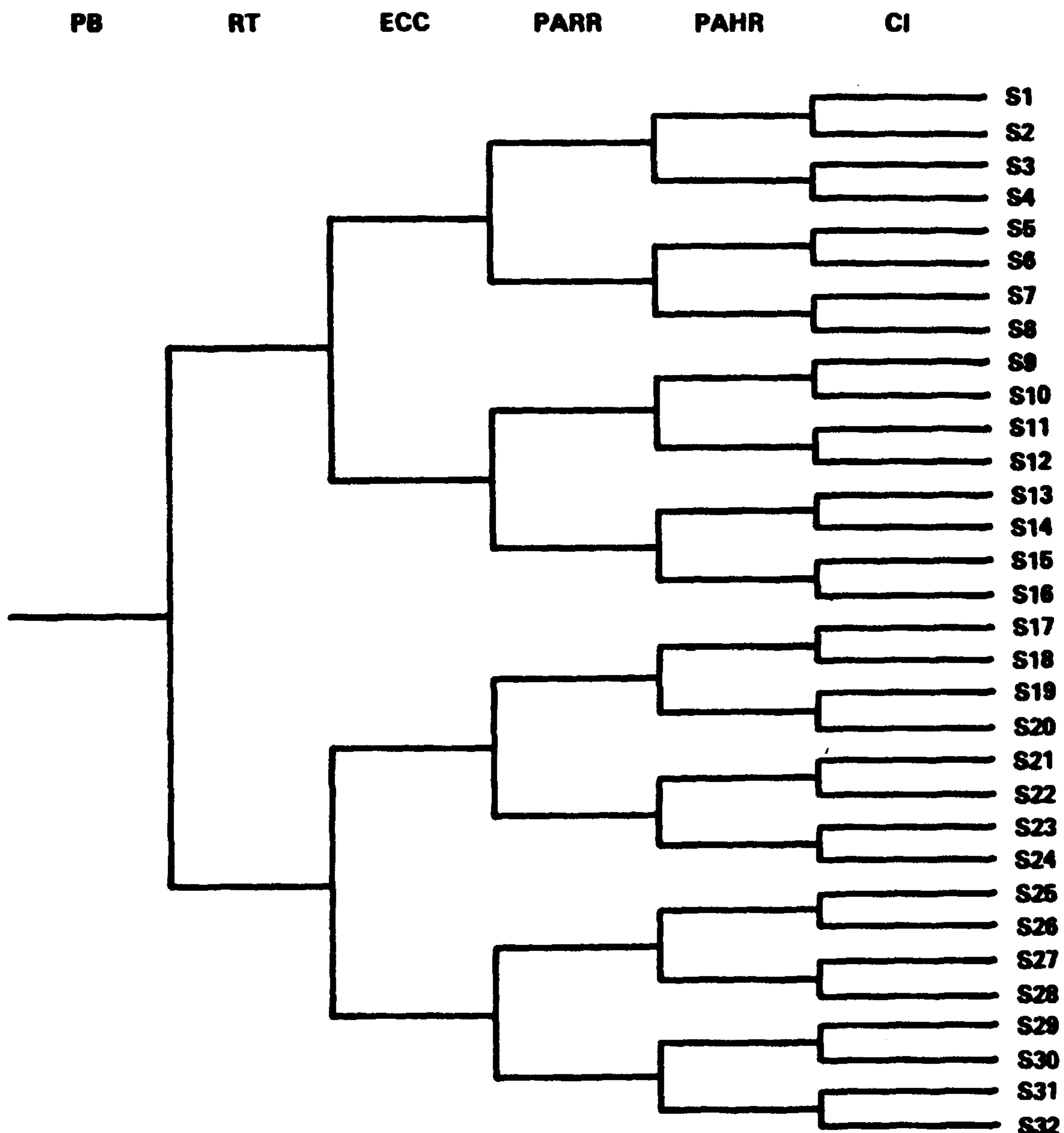


FIGURE 6

FUNCTIONAL LOCAL EVENT TREE SHOWING INTERRELATIONSHIPS WITH RT

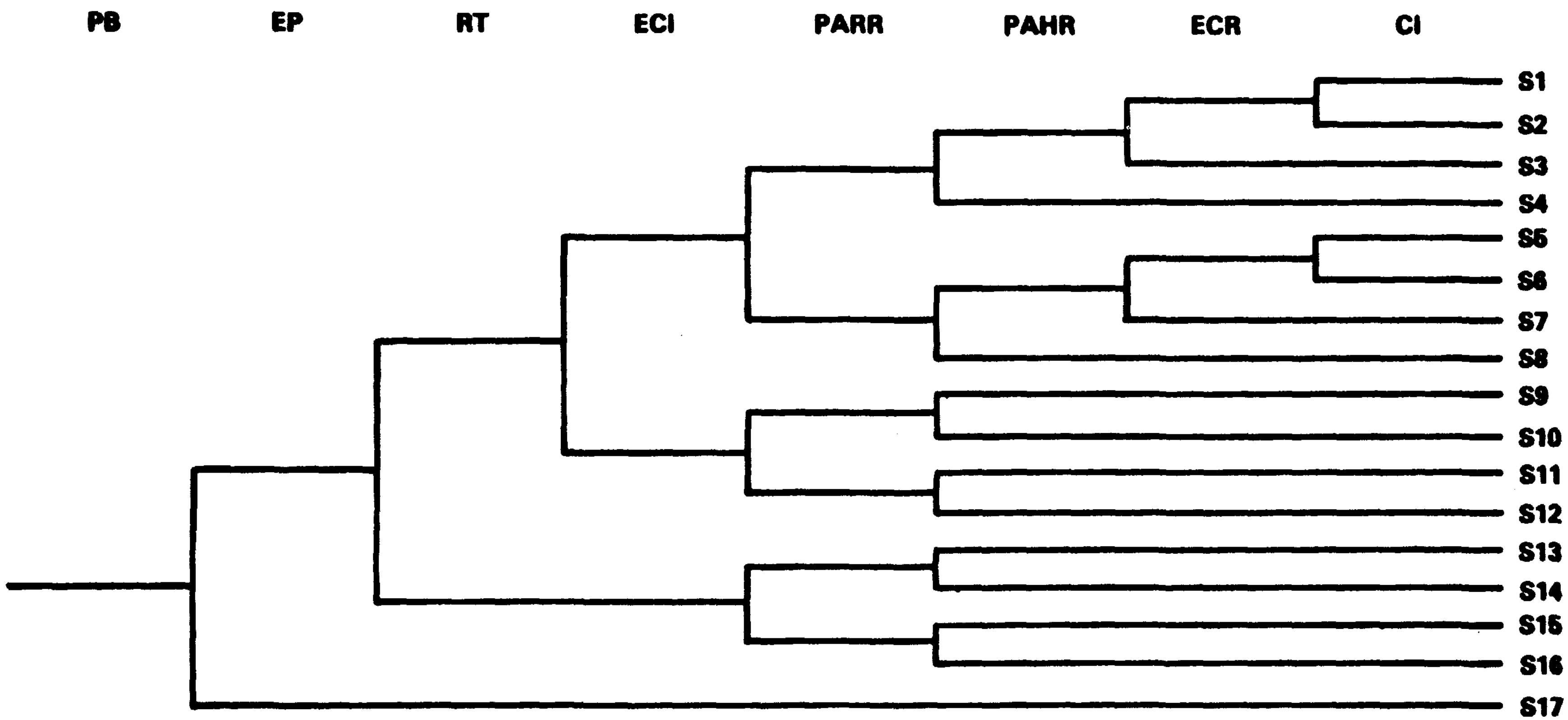


FIGURE 7

ESF FUNCTIONS TO ESF SYSTEM INTERRELATIONSHIPS

	RT	ECI	PARR	PAHR		ECR	
PWR LARGE LOCA >6" DIAM.		ACC AND LPIS	CSIS OR CSRS + SHA	CSRS AND CHRS		LPRS	CSRS AND CHRS
PWR SMALL LOCA 2"-6" DIAM. BREAK	RPS	ACC AND HPIS	SAME	SAME		LPRS AND HPRS	CSRS AND CHRS
PWR SMALL LOCA ½"-2" DIAM. BREAK	SAME	HPIS AND AFW	SAME	CSRS AND CHRS	CSIS	SAME	

FIGURE 8

PWR LARGE LOCA EVENT TREE

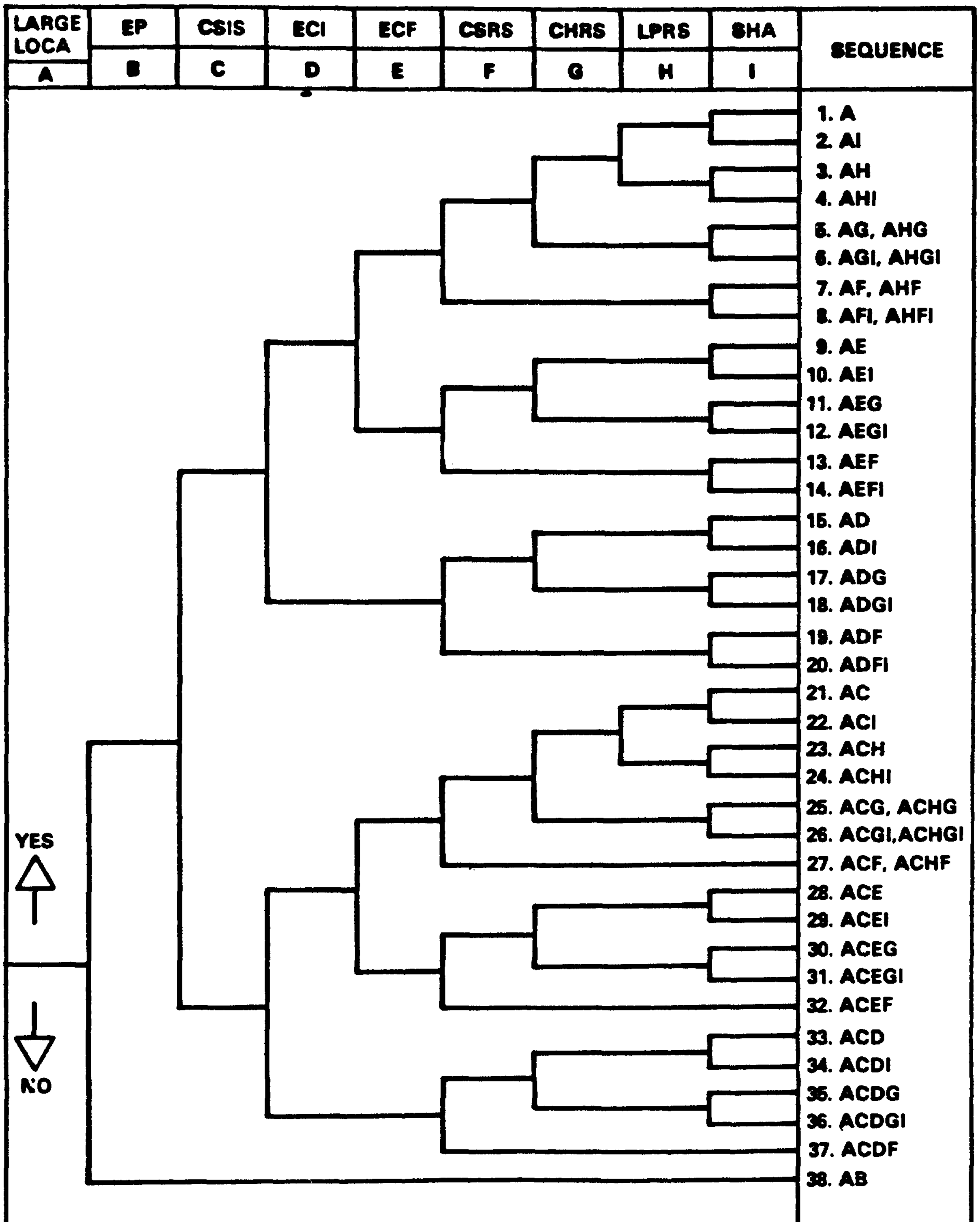


FIGURE 9

PWR CONTAINMENT EVENT TREE

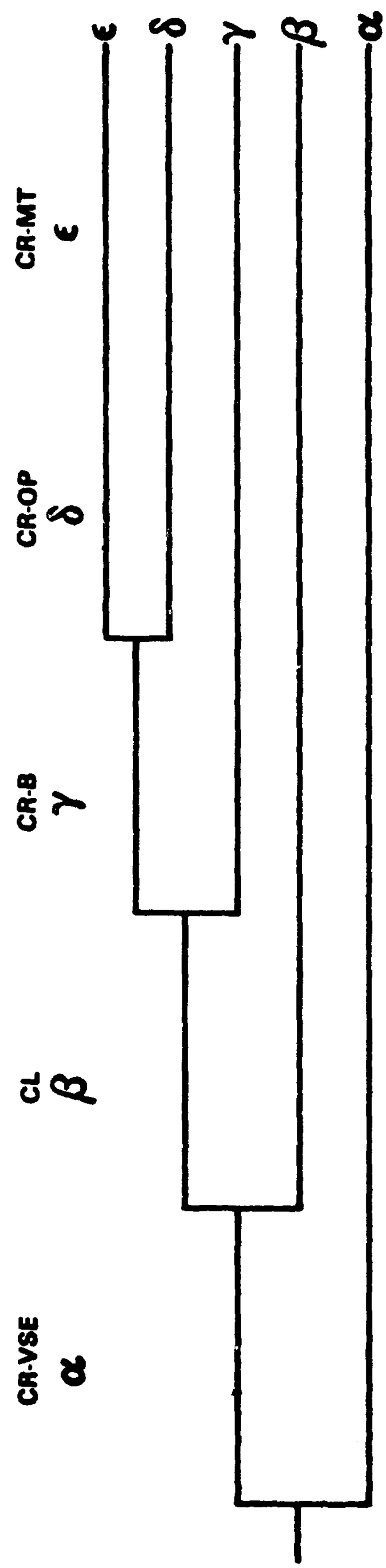


FIGURE 10

Figure 10 shows a containment event tree for which the inputs are the accident sequences from the last tree that resulted in containment rupture or in core melt. The various modes in which the containment can fail are examined. The containment can rupture because of a steam explosion in the reactor vessel. This would have to be a very large explosion, large enough to rip the head off the reactor vessel and blow it out through the top of the containment. That is a low-probability event, obviously. Containment failure can also result from failure to isolate it, in which case there would have to be a large enough hole so that the containment cannot be overpressurized. Another cause of containment failure is combustion and explosion of hydrogen generated by the reaction between stainless steel and water. If none of these things happen, the containment can rupture simply through failure of the heat-removal system. And, of course, there is one more mode of containment failure: the core will melt through the bottom of the reactor vessel, through the containment, and into the ground. These are the ways in which the containment can fail; each has a different probability and vastly different consequences.

Figure 11 indicates that we have to couple together the LOCA tree, or the transient tree, with the containment event tree to get the complete set of accident sequences. The form of the containment event tree changes depending on the a priori conditions of the LOCA tree.

From the large LOCA tree in the PWR combined with the containment event tree, there are 150 accident sequences possible (Figure 12). These are broken down in two ways: (1) by the size of the radioactivity release (1, 2, 3, 4, etc.) and (2) by probability. Probabilities are assigned to these sequences to determine the probability of each of the releases. Clearly, only a few sequences, anywhere from one to four or five, determine the probability of occurrence of any release. The probabilities of the other sequences are so much smaller than these few that they do not contribute. Thus a general case of 30,000 accident sequences in the large LOCA tree has reduced to 20. The same exercise can be done for each event tree that was considered in the study: the small LOCA tree and the transient tree. Figure 13 shows the top of the large LOCA tree, the two small LOCA trees, the reactor-vessel-rupture tree, and the transient tree. There are 80 sequences out of a possible total of 130,000. In fact, only two or three sequences dominate the risk.

Figure 14 shows the generalized form of the accident sequences. Each sequence has the probability of some initiating event x, the probability of some engineered safety system failing. In most cases this is a single system, but in a few cases it is more than one. Examples are the probability of a given containment-failure mode, the probability of a particular weather condition, and the probability of a particular population distribution being exposed. For a typical sequence the probability of a pipe break is 10^{-3}, the probability of a system failure is 10^{-2}, and the probability of a containment-failure mode is 10^{-1}. From the viewpoint of seeking to improve safety, it is obvious that the largest consequences come from the containment-failure mode in which the containment ruptures above the

LINKING OF ACCIDENT AND CONTAINMENT EVENT TREES

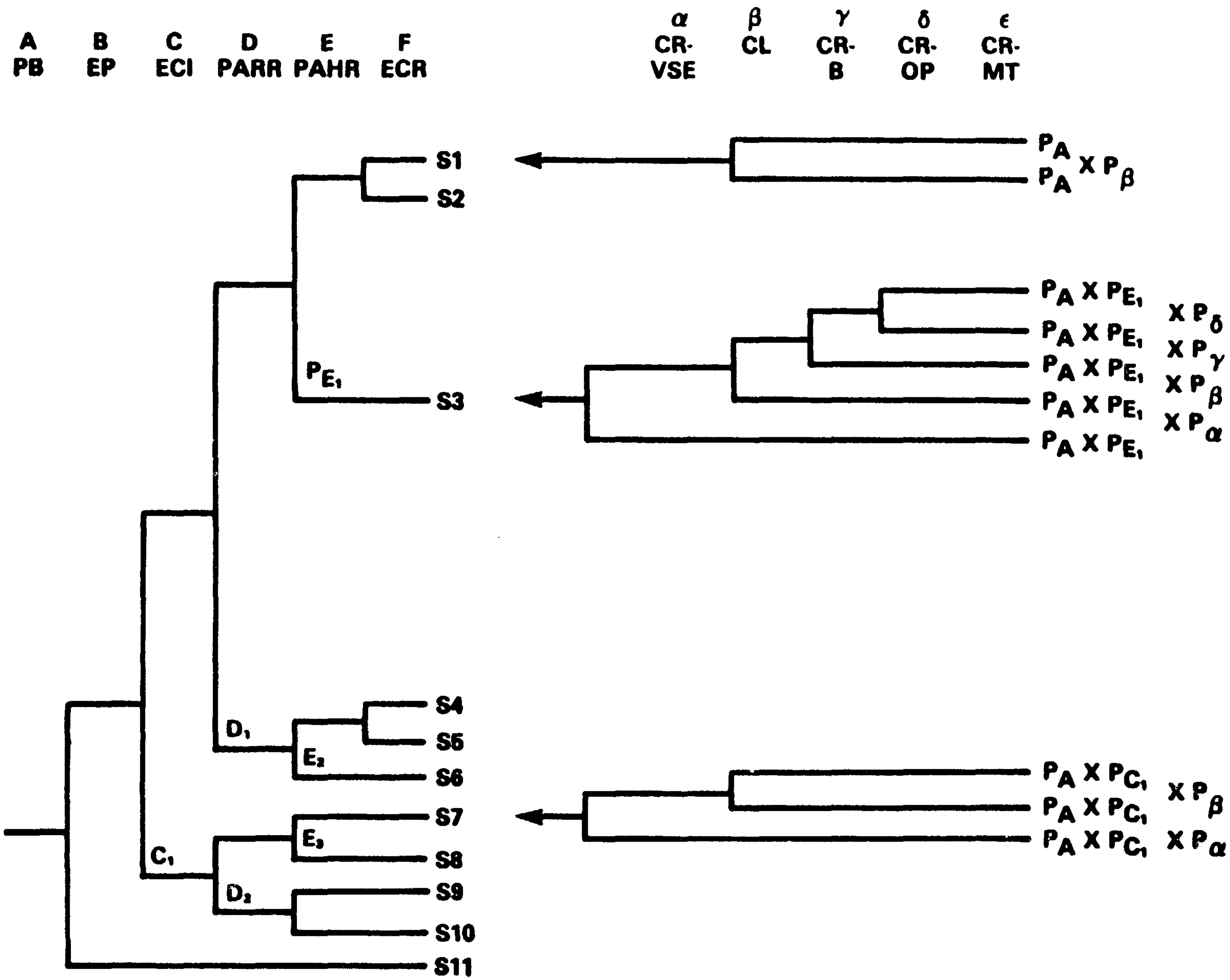


FIGURE 11

PWR LARGE LOCA ACCIDENT SEQUENCES VS. RELEASE CATEGORIES

Core Melt | No Core Melt

CONSEQUENCE CATEGORIES								
1	2	3	4	5	6	7	8	9
DOMINANT LARGE LOCA ACCIDENT SEQUENCES WITH POINT ESTIMATES								
AB-α 1×10^{-11}	AB-γ 1×10^{-10}	AD-α 2×10^{-8}	ACD-β 3×10^{-11}	AD-β 9×10^{-9}	ACHF-ε 1×10^{-7}	AD-ε 2×10^{-6}	A-β 5×10^{-7}	A 1×10^{-4}
AF-α 1×10^{-10}	AHF-δ 2×10^{-10}	AII-α 1.3×10^{-8}		AH-β 6×10^{-9}		AH-ε 1×10^{-6}		
ACD-α 5×10^{-11}		AF-δ 1×10^{-8}						
AG-α 9×10^{-11}		AG-δ 9×10^{-9}						
OTHER LARGE LOCA ACCIDENT SEQUENCES								
ACDGI-α	ADF-β	AHG-α	ACDGI-β	AHI-β	ADF-ε	ANG-δ	AI-β	AI
AHFI-α	AHFI-δ	AHGI-α	ADG-β	AHG-β	ACHGI-ε	AHGI-δ	AC-β	AC
ACHF-α	ACHF-δ	ADF-α	ACDI-β	AHGI-β	AHFI-ε	AHCI-ε	ACI-β	ACI
ACDI-α	ACHF-γ	ADFI-α	ACDG-β	ADI-β	ADFI-ε	ACH-ε		
ACDG-α	ACDF-γ	ACH-α	ADGI-β	ACH-β	ACDF-ε	ACHI-ε		
AGI-α	ACEF-γ	ACHI-α	ACE-β	ACHI-β	ACDGI-ε	ACHG-δ		
AFI-α	AHFI-β	ACHG-α	ACEI-β	ACHG-β	AEF-ε	ACHG-ε		
ACG-α	ADFI-β	ACHGI-α	ACEG-β	AE-β	AEFI-ε	ACHGI-ε		
ACGI-α	ACHF-β	ACI-δ	ACEGI-β	AEI-β	ACEF-ε	ACDI-ε		
ACF-α	ACDF-β	AFI-δ	AEG-β		ACEGI-ε	ACDG-δ		
ACDF-α	AHI-γ	ACG-δ	AEGI-β		AB-ε	ACDG-ε		
ACEI-α	AHFI-γ	ACGI-δ			AHF-ε	ADG-δ		
ACEG-α	AB-δ	ACF-δ				ADGI-δ		
ACEGI-α	AEF-β	AHI-α				AHG-ε		
ACEF-α	AEFI-β	ADGI-α				ADI-ε		
ACE-α	ACEF-β	ADI-α				ADG-ε		
AHF-α	AEF-δ	ADG-α				ACD-ε		
	AEFI-δ	AE-α				ADGI-ε		
	ACEF-δ	AEI-α				AHI-ε		
	AB-β	AEF-α				AE-ε		
	AHF-β	AEFI-α				AEI-ε		
		AEC-α				ACE-ε		
		AEGI-α				ACEI-ε		
						ACEG-ε		
						ACEG-δ		
						ACEGI-δ		
						ACHGI-δ		
						AEG-δ		
						AEGI-δ		
						AEG-ε		
						AEGI-ε		

FIGURE 12

PWR DOMINANT ACCIDENT SEQUENCES VS. RELEASE CATEGORIES

RELEASE CATEGORIES	Core Melt							No Core Melt	
	1	2	3	4	5	6	7	8	9
LARGE LOCA A	AB-α $1x10^{-11}$ AF-α $1x10^{-10}$ ACD-α $5x10^{-11}$ AC-α $9x10^{-11}$	AB-γ $1x10^{-10}$ AHF-δ $2x10^{-10}$	AD-α $2x10^{-8}$ AH-α $1.3x10^{-8}$ AF-δ $1x10^{-8}$ AC-δ $9x10^{-9}$	ACD-β $3x10^{-11}$	AD-β $9x10^{-9}$ AH-β $6x10^{-9}$	ACHF-ε $1x10^{-7}$	AD-ε $2x10^{-6}$ AH-ε $1x10^{-6}$	A-β $5x10^{-7}$	A $1x10^{-4}$
A Probabilities	$2x10^{-9}$	$1x10^{-8}$	$1x10^{-7}$	$1x10^{-8}$	$2x10^{-8}$	$7x10^{-7}$	$5x10^{-6}$	$1x10^{-5}$	$1x10^{-4}$
SMALL LOCA S_1	S_1B-α $3x10^{-11}$ S_1CD-α $7x10^{-11}$ S_1F-α $3x10^{-10}$ S_1G-α $2x10^{-10}$	S_1B-γ $2x10^{-10}$ S_1HF-δ $6x10^{-10}$ S_1B-β $5x10^{-11}$	S_1D-α $3x10^{-8}$ S_1H-α $4x10^{-8}$ S_1F-δ $3x10^{-8}$ S_1G-δ $2x10^{-8}$	S_1CD-β $1x10^{-10}$	S_1H-β $2x10^{-8}$ S_1D-β $3x10^{-8}$	S_1DF-ε $3x10^{-7}$	S_1D-ε $6x10^{-6}$ S_1H-ε $3x10^{-6}$	S_1-β $2x10^{-6}$	S_1 $3x10^{-4}$
S_1 Probabilities	$3x10^{-9}$	$2x10^{-9}$	$2x10^{-7}$	$3x10^{-8}$	$9x10^{-8}$	$6x10^{-7}$	$9x10^{-6}$	$3x10^{-5}$	$3x10^{-4}$
SMALL LOCA S_2	S_2B-α $1x10^{-10}$ S_2F-α $1x10^{-9}$ S_2CD-α $1x10^{-10}$ S_2G-α $9x10^{-10}$	S_2B-γ $1x10^{-9}$ S_2HF-δ $2x10^{-9}$	S_2D-α $1x10^{-7}$ S_2H-α $1x10^{-7}$ S_2F-δ $1x10^{-7}$ S_2C-δ $2x10^{-8}$ S_2G-δ $9x10^{-8}$	S_2DG-β $1x10^{-11}$	S_2D-β $9x10^{-8}$ S_2H-β $6x10^{-8}$	S_2B-ε $1x10^{-8}$ S_2CD-ε $2x10^{-8}$ S_2HF-ε $2x10^{-9}$	S_2D-ε $8x10^{-6}$ S_2H-ε $6x10^{-6}$		
S_2 Probabilities	$7x10^{-9}$	$4x10^{-7}$	$3x10^{-6}$	$4x10^{-7}$	$2x10^{-7}$	$2x10^{-6}$	$2x10^{-5}$		
REACTOR VESSEL RUPTURE - R	RC-α $2x10^{-12}$	RC-γ $2x10^{-11}$ RF-δ $1x10^{-11}$ RC-β $1x10^{-11}$	R-α $1x10^{-9}$				R-ε $1x10^{-7}$		
R Probabilities	$2x10^{-11}$	$1x10^{-10}$	$1x10^{-9}$	$2x10^{-10}$	$1x10^{-9}$	$9x10^{-9}$	$1x10^{-7}$		
INTERFACING SYSTEMS LOCA (CHECK VALVE) - V		V $4x10^{-6}$							
V Probabilities	$4x10^{-7}$	$4x10^{-6}$	$4x10^{-7}$						
TRANSIENT EVENT - T	TMLB-α $3x10^{-8}$	TMLB-γ $4x10^{-7}$ TMLB-δ $1x10^{-7}$	TML-α $3x10^{-8}$ TKQ-α $3x10^{-8}$ TKMQ-α $1x10^{-8}$		TML-β $6x10^{-10}$ TKQ-β $5x10^{-10}$	TLMB-ε $3x10^{-6}$	TML-ε $3x10^{-6}$ TKQ-ε $2x10^{-6}$ TKMQ-ε $9x10^{-7}$		
T Probabilities	$9x10^{-8}$	$5x10^{-7}$	$2x10^{-7}$	$6x10^{-9}$	$4x10^{-7}$	$4x10^{-6}$	$8x10^{-6}$		
(Σ) SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORY									
MEDIAN (50% VALUE)	$7x10^{-7}$	$8x10^{-6}$	$6x10^{-6}$	$5x10^{-7}$	$1x10^{-6}$	$1x10^{-5}$	$6x10^{-5}$	$4x10^{-5}$	$4x10^{-4}$

FIGURE 13

SUMMARY OF ACCIDENTS INVOLVING CORE

RELEASE CATEGORY	PROBABILITY (Yr^{-1})	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	WARNING TIME FOR EVACUATION (Hr)	ELEVATION OF RELEASE (Meters)	FRACTION OF CORE INVENTORY RELEASED							
						Xe-Kr	Org-I	I-Br	Cs-Rb	Te	Ba-Sr	Ru	La
PWR 1	7×10^{-7}	1.5	0.5	1.5	25	0.8	6×10^{-3}	0.6	0.4	0.4	0.05	0.4	3×10^{-3}
PWR 2	5×10^{-6}	2.5	0.5	1.5	0	0.9	7×10^{-3}	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR 3	6×10^{-6}	2.0	1.0	1.5	0	0.8	6×10^{-3}	0.2	0.2	0.3	0.02	0.02	3×10^{-3}
PWR 4	5×10^{-7}	2.5	3.0	1.5	0	0.5	2×10^{-3}	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR 5	1×10^{-6}	2.5	4.0	1.5	0	0.2	2×10^{-3}	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR 6	1×10^{-5}	12.0	10.0	1.5	0	0.2	2×10^{-3}	8×10^{-4}	7×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR 7	6×10^{-5}	10.0	10.0	1.5	0	5×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR 8	4×10^{-5}	0.5	0.5	N/A	0	2×10^{-3}	5×10^{-6}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR 9	4×10^{-4}	0.5	0.5	N/A	0	3×10^{-6}	7×10^{-9}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0

FIGURE 14

ground and releases a large cloud of airborne radioactivity. Let us see now whether or not the probability can be reduced significantly, say by an order of magnitude.

The three highest release categories, which in fact are the categories that determine the total risk, are 1, 2, and 3. The other categories are so small that there are extremely small consequences from them. The accident sequences that determine the probability of all three of these release categories--and whose probability determines the total probability of reactor accident risks in the PWR--are TMLB$'\gamma$, TLMB$'\delta$, and $S_2C\ \delta$ (see Figure 13). The Greek letters denote the probability of failure of the containment from overpressure by failure of the containment heat-removal system and the probability of containment failure from hydrogen burning.

This suggests that if in fact those probabilities can be reduced, say by venting the containment through a filter so that it does not fail with that probability, the probabilities of these three categories and the overall risk can be reduced.

Figure 14 demonstrates this point in terms of radioactivity releases. The fraction of core inventory released is shown by fission-product release groups for the three highest categories: PWR 1, 2, and 3. One can see how large these three groups are in comparison with all the others. Nothing can be done to reduce the release fractions, but efforts can certainly be made to reduce the probabilities, which would then reduce the risks.

I am now going to talk about the second subject: improved safety research. In the Fiscal Year 1978 NRC Authorization Act, Congress asked the NRC to produce a plan for research to improve the safety of reactors. A wide range of sources were consulted for suggestions for improved safety, and a research review group was formed consisting of about 40 consultants, including antinuclear people. The most prolific source of suggestions was the Advisory Committee on Reactor Safeguards, although the NRC staff also made numerous suggestions. A report issued by the American Physical Society's (APS) Study Group on Reactor Safety contained a number of suggestions, as did a study sponsored by the Ford Foundation. Other suggestions were contained in the U.S. Atomic Energy Commission's ECCS Acceptance Criteria, written at the end of the 2-year ECCS hearing, and in a fairly comprehensive report by Environmental Quality Laboratories. In all there were over 200 suggestions, which were grouped into 16 research topics. To decide which of these 16 research topics are the most important, a set of criteria was set up, including the following: the breadth of support from this group, the risk-reduction potential (from the viewpoint of the Reactor Safety Study), applicability to existing and future reactors, applicability to BWRs and PWRs, and implementation cost. For lack of time the value-impact analysis was qualitative only.

The results are summarized in Figure 15, which shows the four evaluation criteria and the five topics that were selected. Vented containment, for instance, had high support, high risk-reduction potential, high applicability, and medium cost. (Medium cost was

	BREATH OF SUPPORT	RISK REDUCTION	APPLICABILITY	COST
VENTED CONTAINMENT	H	H	H	M
DECAY HEAT REMOVAL	H	H	M-H	M
ECCS SYSTEMS	H	M	M	L-M
ACCIDENT RESPONSE	H	H-M	H	L
SEISMIC DECOUPLING	M	H-M	L-M	M-H

FIGURE 15

defined as between $10 to $15 million per plant.) Decay-heat removal had high support, high risk-reduction potential, less clear applicability, and lower cost. One of the recommendations that was featured prominently in the APS report and the ECCS Acceptance Criteria was research on emergency core-cooling systems. It is well known how much effort has been and is still going into emergency core-cooling systems. Interestingly enough, though there is high support, there is only a moderate risk-reduction potential, moderate applicability, and large to medium costs. The risk-reduction potential is only moderate because it was established in the Reactor Safety Study that LOCA sequences are not among the three principal contributors to risk. Another topic selected for research is improved accident response within the plant, i.e., principally by the operators and by providing them with more analytical information than they now have. It turns out that the operators and maintenance personnel contribute significantly to accident risks. This topic had high support, high to medium risk-reduction potential, high applicability, and very low cost. There were a number of suggestions for decoupling seismic forces from the reactor plant, such as floating it in a pond or putting it in a muddy environment. This topic had moderate support, high to medium cost, and low general applicability (it would be applicable only to future plants and probably only in selected sites). The highest-priority topic and the one that seems to have the largest potential for risk reduction is the vented containment: if the core is melting, the containment is opened to the atmosphere through filters to prevent the large uncontrolled release of gaseous activity that occurs when the containment ruptures by itself. It is proposed to define a conceptual system configuration to determine feasibility, sizing, and cost and to perform a quantitative value-impact analysis. Concepts such as a separate building where the containment is vented through a pool of water, sand and gravel filters, or charcoal filters, will be explored. There is much information about the various concepts, and it is not sure that any physical research will be required. If it is, then the NRC will do it. Also studied will be some of the other containment improvements that can be made.

We have given a specific example of the application of risk-assessment techniques to improve the safety of nuclear power plants. There are many other examples of applications that have been made in solving specific licensing problems. The principal problem is associated with stabilizing the licensing process; that is, finding stopping places in the NRC review process.

Many believe that a major advance in licensing stability would be achieved if criteria for acceptable levels of risk were established. Presumably they feel that such criteria could provide a more rational basis for decisionmaking, both within the regulatory process and on a more broadly applicable societal basis. There are very real questions about the utility of such criteria in societal applications. Quantitative levels of acceptable risk have not been established in the United States regarding any human endeavor. There are some examples in other countries, but that does not change the problem of implementing such procedures in the United States. Furthermore, there is a body of opinion among social scientists, with some supporting evidence, that participation by divergent elements in an open and local, and I emphasize local, decisionmaking process is the only way of determining societally acceptable risk levels. This viewpoint will not sit well with those of us in the physical sciences who have the traditional belief that somewhere there are some decisionmakers who will respond well to a fairly rational determination of acceptable levels of risk. It is likely that neither of these viewpoints is entirely correct.

If there is one place where criteria for acceptable risk levels would, on the surface, appear to be useful it is in the reactor licensing process. There are many examples of the difficulty involved in deciding how far to follow the path of a particular accident sequence in the review process. Clearly, there is some point at which the combined probability of the events postulated is so low as to make the consideration of additional events unnecessary. However, even there, if such criteria were available, one would have to face the problem of allocating portions of the allowable risks to the various safety features of the plant in order for the criteria to be usable. This is risk allocation, which is a whole separate and complicated subject of its own. The establishment of such risk allocations would be a formidable task, and even if feasible, would represent and additional limitation on freedom in design. Furthermore, it is not clear that such allocations would be more useful than other approaches. For instance, present NRC licensing activity already uses what is essentially a qualitative level of acceptable risk. The plants that are licensed have to meet NRC regulations and guidance embodied in standard review plans and regulatory guides. Moreover, the risk assessment in the Reactor Safety Study has in effect measured the accident risks in plants that the staff has found acceptable. Using this as a point of departure, in many cases a simple straightforward analysis can demonstrate that a particular accident sequence would or would not contribute to the overall accident risk defined in the Reactor Safety Study. The stopping places in licensing analysis can be found by this approach. The NRC's probabilistic analysis staff has demonstrated that in many cases matters that have been of concern to the regulatory staff and to the Advisory Committee on Reactor Safeguards have not in fact been significant contributors to the overall accident risk and could be ignored.

It is therefore clear that the quantitative risk-assessment techniques can be used to great advantage in stabilizing the licensing process without defining quantitative criteria for acceptable levels of risk. We are now just beginning to apply risk-assessment techniques to a whole variety of problems, and this should help significantly in stabilizing the licensing process. This is being done in close coordination between the Office of Nuclear Regulatory Research and the Office of Nuclear Reactor Regulation. I will just list five topics that we are looking at now. There are outstanding generic issues on reactor safety, 133 of them. We have reviewed these already, and it seems that perhaps 10 to 20 may significantly affect risk whereas the others do not matter very much. This has yet to be finished, promulgated, published, and accepted, but that is where we are today. We are beginning to look at the standard review plan, which guides individual reviewers and their review of reactors from a risk-assessment viewpoint to determine which items considered may not be significant to risk. We are going to review those to find which are important to safety and which are not. We will look at technical specifications to find out which items are not significant to risk. There is also the systematic evaluation plan for evaluating the eleven oldest reactors in the country to find out what to do with them. And we are going to review them from a risk-assessment viewpoint to find out which items must be looked at and which items need not be looked at. I am very excited about this new work that we are just starting. It shows that the techniques of the Reactor Safety Study have come of age and have achieved broad acceptance both in the scientific community and in the NRC, and I hope we can do important things with it.

DISCUSSION

WILHELM: Could you give us some of the random conditions for a filter system of a vented containment with respect to temperature, pressure difference, dose, and relative humidity?

LEVINE: I am not able to define these conditions at this time since the study I mentioned has not yet been done. Obviously, a spectrum of conditions (i.e., flow rates, pressures, etc.) will have to be examined to optimize the system design.

WILHELM: You may be interested to learn that we calculated the conditions for such a vented containment filter system and the figures are rather shocking. The flow, calculated for a German 1300 megawatt electric pressurized water reactor, is 10,000 cubic feet/minute through the filter system. This is a problem when considering the use of sand bed filters. The temperature rises to 300^{o} in the filter system and the amount of hydrogen in the containment amounts of some 10,000 cubic meters of hydrogen. With hydrogen present, you can't use silver zeolites for the adsorption of iodine because the iodine would react to form hydrogen iodide and desorb. Doses may rise to 10^{11} rad on the adsorbent and the filter material, depending, of course, on the size of the filter system. I wish to mention these figures only to show the audience what kind of challenge it is to build such a system.

LEVINE: I can't quarrel with your numbers although I suspect your flow rate may be rather high. It turns out that to prevent a typical U.S. PWR containment from rupturing due to overpressure, requires a four or five inch hole. Now, you can't get adequate flowrates through a four inch hole to accomodate your flow rate, so I suspect there is some difference in the assumptions on which you made your calculations. You're certainly right about the challenge that would occur to a filter system, but there are filter systems in the United States that can handle very large flowrates and do so effectively. They're classified, unfortunately, so I can't talk much about them. It's a challenge, but I think it can be done.

KOVACH: I disagree with Mr. Wilhelm's analysis of the vented containment. If the venting is started early enough after a LOCA, a smaller than 10,000 CFM filter is required. We have presented such an analysis at the 14th Air Cleaning Conference. The quantities that were shown were a few thousand CFM, one to three, to prevent an overpressurization of the reactor. At that point, you have no hydrogen.

LEVINE: I agree with you.

FIRST: In view of the fact that Saul Levine has given us such an excellent review of how accident trees were used for making the assessments that he's covered so well, I've taken the privilege of slightly altering our program by asking Dade Moeller to give his talk next instead of waiting for the paper that was to go in between. The reason I've asked him to do this is that Saul Levine identified accidents sequences and talked about engineering safety system failure. Dr. Moeller's paper, entitled Review of Failures in Nuclear Air Cleaning Systems, fits very neatly with the papers we've just heard in terms of the practical applications. Dr. Moeller has spoken on this topic at previous meetings and the information he has brought to us has been not only excellent but quite startling, and I'm sure you will find what he has to say today equally interesting and important.

REVIEW OF FAILURES IN NUCLEAR AIR CLEANING SYSTEMS (1975 - 1978)*

Dade W. Moeller, Chairman
Department of Environmental Health Sciences
Harvard School of Public Health
Boston, Massachusetts

Abstract

During the period from January 1, 1975, through June 30, 1978, over 9,000 Licensee Event Reports (LERs) pertaining to the operation of commercial light water nuclear power plants in the U. S. were reported to the Nuclear Regulatory Commission. Of these reports, over 1,200 (approximately 13%) pertained to failures in air monitoring, ventilating and cleaning systems. For BWR installations, over half of the reported events related to failures in equipment for monitoring the performance of air cleaning systems as contrasted to failures in the systems themselves. In PWR installations, failures in monitoring equipment amounted to about 32% of the total. Reported problem areas in BWR installations included the primary containment and standby gas treatment and off-gas systems, as well as the High Pressure Coolant Injection and Reactor Core Isolation Systems. For PWR installations, reported problem areas included primary containment and associated spray systems and waste processing equipment. Although data on reported failures in power reactor installations can be interpreted in a variety of ways, one message is clear. There is a need for research on the development of more reliable equipment for sampling and monitoring air systems. Equipment that provides inaccurate data on the performance of such systems can lead to as many problems as inadequacies in the systems themselves.

I. Introduction

At the 13th Air Cleaning Conference in 1974, this author presented a paper(1) in which an analysis was performed of 55 failures that had been reported in nuclear air cleaning systems during the time period from 1966 through 1974. Of these failures, 28 had occurred in commercial nuclear power plants. Since that time, there has been a large increase in the number of commercial nuclear power plants in operation. In fact, during the period from January 1, 1975, through approximately June 30, 1978 (the time period covered by this study), over 9,000 additional Licensee Event Reports were submitted,(2)(3)(4)(5)(6) of which over 1,200 pertain to failures in nuclear air systems. This paper presents a summary of observations made on the basis of an indepth review of these newer items. In evaluating this report, it should be noted that it differs in two basic ways from the earlier paper. First, this analysis was confined to events occurring within commercial nuclear power plants; second, it was broadened to include reports on air monitoring and ventilating, as well as air cleaning systems.

*This work was supported in part by the U.S. Department of Energy under contract EY 76 S 02-3049

Table I. Overall tabulation of licensee event reports

	Power Reactors		All Events		Air Cleaning Events*	
Year	Reactor Type	Number Operating**	Total Number	Number per Reactor	Total Number	Number per Reactor
1975	BWR	20	1169	58	121	6.0
	PWR	27	1097	41	141	5.2
1976	BWR	23	1253	54	227	9.9
	PWR	30	1264	42	131	4.4
1977	BWR	24	1190	50	152	6.3
	PWR	36	1678	47	210	5.8
1978	BWR	24	637***	53****	125***	10.4****
	PWR	41	1076***	52****	120***	5.9****

* Includes events pertaining to air monitoring, ventilating, and cleaning

** As of July 1 of the given year

*** Through approximately June 30, 1978

****Projected through December 31, 1978

As may be noted in Table I, there was a yearly average of 50 to 58 LERs per operating commercial Boiling Water Reactor (BWR) and 41 to 52 LERs per operating commercial Pressurized Water Reactor (PWR) during the three and one-half year period covered by this study. It should be observed, however, that these averages were calculated using a simplistic approach. That is, they were computed by dividing the total number of LERs for each year by the number of reactors operating as of July 1, of the same year. For BWR plants, approximately 10% to 20% of the reported events pertain to failures in air systems; for PWR plants, the range is from 10% to 13%. The higher average number of events pertaining to air systems in BWRs is to be expected since there are more opportunities for radioactive airborne releases from such facilities and the number of associated air monitoring, ventilating and cleaning systems is greater.

II. Review of Specific Failures

Details of LERs pertaining to air monitoring, ventilating and cleaning units within specific reactor systems for BWRs and PWRs are summarized in Tables II and III, respectively.

Problems in BWRs

As may be noted, the reported events for BWRs relate to the commonly expected areas such as primary and secondary containment, and standby gas treatment and off-gas systems, as well as to the less expected areas such as the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems. Of particular note is the large number of reported events associated with the equipment designed to sample and monitor the performance of ventilating and air cleaning systems. In fact, an analysis of the data shows that 51% of all reported failures were in the equipment installed to monitor the performance of the air cleaning and ventilating systems. Further analyses show that, of the failures in air sampling and monitoring equipment in primary containment, 41% occurred as a result of deficiencies in the air sampling portion of the system, as contrasted to failures in the detector or analysis unit itself. Of the failures within the detectors and analyzers, 31% were with hydrogen and/or oxygen analyzers, 35% were with pressure sensors, 26% were with radiation monitors (gaseous and particulate), and about 9% were with temperature monitors. The relatively high frequency of failures in hydrogen and oxygen analyzers is of special significance in view of the importance of such monitors to warn of explosive mixtures within various BWR systems.

With respect to specific failures in air monitoring, ventilating and cleaning systems in BWRs, the following appear worthy of note.

Violation of Single Failure Criterion

During operation of one plant in 1975, it was discovered that an auto initiation signal for one standby gas treatment unit automatically closed the inlet valve to the second unit. As a result, if the inlet valve for the initiating unit failed, no inlet valve for either unit would be open. This proved to be a violation of the single failure criterion and the logic system was modified to remove this deficiency. This same problem was discovered at a second operating BWR in 1976 and again reported as an LER.

Table II. Summary of licensee event reports

Air Monitoring, Ventilating and Cleaning Systems

(Boiling Water Reactors, 1975 - 1978)

System	Component	Nature of Problem	Number of Events 1975	1976	1977	1978*
Primary containment	Atmospheric sampling and monitoring	Failure in sampling system	7	36	14	26
		Failure of detector or analysis unit	17	46	24	33
	Diluting, inerting, or ventilating system	Deficiencies in nitrogen purge or ventilating system	19	31	11	10
		Deficiencies in filter system	0	0	0	2
	Torus	Failure of vacuum breakers	10	10	9	3
		Improper water level or indicator malfunction	9	19	0	3
	Containment spray system	Deficiencies in operation of valves	2	8	0	0
		Failure of supporting equipment	2	4	1	1

*Through approximately June 30, 1978

System	Component	Nature of Problem	Number of Events 1975	1976	1977	1978*
Reactor cooling system	Coolant purification system	Excessive airborne release	0	2	2	0
		Hydrogen explosion	1	0	0	0
	Leak detection system	Failure of particulate sampler	0	0	1	0
	Injection and isolation system	Isolation of high pressure coolant injection system	1	6	13	0
		Isolation of reactor core isolation cooling system	1	3	1	0
Secondary containment (Reactor building)	Atmospheric sampling and monitoring system	Failure in sampling system	0	3	1	1
		Failure of radioactive gas monitor	6	5	7	8
	Diluting and ventilating system	Failure of blowers, isolation valves, or dampers, or cooling water flow	7	4	5	0

*Through approximately June 30, 1978

System	Component	Nature of Problem	Number of Events 1975	1976	1977	1978*
Standby gas treatment system	Air flow system	Failure of blowers, low flow, train overheated, excessive moisture	14	8	8	10
	Filter system	Adsorbers depleted or absent, filters plugged	2	2	2	2
	Fire protection system	Flooding of charcoal filters	0	1	0	3
Off-gas system	Sampling and monitoring system	Failure in sampling system	5	5	5	3
		Failure of detector or analysis unit	2	4	11	3
	Air flow system	Failure of drain line, leaks in line	2	4	8	2
	Filter system	Excessive pressure drop (plugged filter)	0	0	1	0
		Fires or explosions	2	5	2	1
	Combustible gas control	Excessive hydrogen concentration	1	2	2	1

* Through approximately June 30, 1978

System	Component	Nature of Problem	Number of Events 1975	1976	1977	1978*
Control room or building	Sampling and monitoring system	Failure in sampling system	0	1	0	0
		Failure of detector or analysis unit	1	4	1	0
	Ventilating system	Failure in emergency ventilating system	2	1	2	5
Turbine room	Atmospheric monitoring system	Failure of radiation monitor	0	0	1	0
	Airborne radioactive release control	Excessive airborne release	0	4	0	0
	Ventilating system	Lack of interlocks on supply and exhaust fans	1	0	0	0
Main stack discharge system	Sampling and monitoring system	Failure in sampling system	4	7	7	2
		Failure of detector or analysis unit	3	2	12	4
	Air flow system	Inadequate flow	0	0	1	2
		TOTALS	121	227	152	125

* Through approximately June 30, 1978

Isolation of HPCI and RCIC Systems

An interesting set of failures has been the isolation of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems that has occurred as a result of inadequacies in the ventilating systems within several BWRs. One such event occurred in 1975, nine in 1976, and eleven in 1977. The basis of the problem is that the areas through which the piping from the HPCI and RCIC systems passes are equipped with temperature sensors which are designed to isolate these systems in case there is a steam leak in the lines. If, however, there is a malfunction in the ventilating systems for these areas, or a sudden change in the outdoor temperature which leads to the sensor indicating a steam leak, the two systems are automatically isolated. Correction of the problem appears to be to increase the capacity of the air ventilation systems for the affected areas.

Deficiencies in Control Room Ventilating System

In 1977, a worker was distracted while filling the caustic and acid tanks in the makeup demineralizer water room. The tanks overflowed and the acid and caustic interacted with each other and the concrete floor producing fumes. Since the exhaust fan in the room was inoperable, the fumes seeped into the Control Room ventilation system. Since the Control Room ventilation system was limited to a maximum of 10% outside (makeup) air, the operators shut off the ventilation system and transferred to the portable supplied air packs. Without the ventilation system on, the Control Room heated up and possible instrument failures were feared. As a result, the plant was put on Emergency Alert until the situation could be rectified.

Fires and Explosions in Off-Gas Systems

Fires and explosions continue to be a problem in BWR off-gas systems. There were two such events in 1975, five in 1976, two in 1977 and one has occurred in 1978. In 1975, a closed isolation valve in the off-gas system at one BWR forced off-gas from the steam jet air ejector through a loop-seal drainline from the holdup line to a sump and back to the dilution fans prior to being discharged through the elevated release point. The sump became pressurized and an explosion occurred when a health physicist removed a manhole cover to the sump and turned on a sampler to check for air contamination. Two people were injured as a result of this event. Later inspection showed that the control room valve position indicating lights and control switch showed the closed valve to be open. Errors in the electrical wiring to the valve were corrected.

In another event in 1975, catalyst pellets from the recombiners were somehow dislodged and transported, perhaps by system flushing, into the preheaters, a pressure valve, and two low point drains in an off-gas system. Later the pellets ignited and caused an explosion.

In 1976, a buildup of ice in the upper portion of the exhaust stack at a BWR resulted in backpressure and the accumulation of hydrogen in the off-gas building. Later, an explosion completely demolished the building. Corrective action included heat tracing and insulation of the upper portion of the elevated discharge pipe. In another case during the same year, an explosion occurred in the stack filter house at a BWR. This resulted from the improper positioning of a demister that permitted moisture to collect and freeze on a HEPA filter, causing a decrease in the flow rate. Pressure buildup resulted in the unseating of a number of off-gas loop seals, which permitted both airborne radioactive materials and hydrogen gas to enter the stack filter house. Corrective actions included proper positioning of the demister and other measures.

In another sequence of events in 1977, a welder's torch set off a hydrogen explosion in an off-gas delay line. Although the line was designed to withstand an explosion, the pressure wave caused water to be removed from the loop seals provided to draw condensation from the gaseous mixture in the off-gas pipe. Since the seals were not refilled, hydrogen built up in two unventilated rooms at the base of the plant stack. Later when a sump pump in the base of the stack was activated, the explosive mixture in the two rooms was ignited.

In early 1978, temperature transients were noted in six charcoal beds in an off-gas system. Because a drainline in the off-gas preheater was plugged, dilution steam condensed on the recombiner catalyst, preventing the recombination of the hydrogen and oxygen passing through the system. Ignition of the mix downstream of the recombiner apparently ignited the charcoal. Nitrogen purge was used to cool the charcoal and extinguish the fire. The drain line was unplugged and the beds returned to service.

Hydrogen Explosion in Acid Day Tank

In 1975, a hydrogen explosion occurred in the condensate demineralizer regeneration system acid day tank at a BWR power plant. The hydrogen was formed when moisture from the atmosphere interacted with concentrated sulfuric acid in the tank due to depletion of the dessicant in a vent line. The explosion, believed to have been ignited by a spark from a nearby welding operation, blew the top off the tank and broke the vent and fill piping. As a result of this event, acid was deposited on nearby equipment, cable trays and the floor.

Failures in Recombiners

The importance of recombiners in preventing the accumulation of explosive mixtures in off-gas and SBGT systems is well known. Proper operation of such units, however, is important from other aspects. For example, while attempting to return a recombiner system mechanical compressor to service after maintenance at one BWR in 1978, the operator left open the inlet-sump drain valve. This allowed radioactive gas to escape from the system to a ventilated sump and then through the vent stack. The resulting release of radioactive material was 4.7 times the Technical Specification limit for a short period of time.

Browns Ferry Fire

As is well known, the use of a candle for testing for penetration leakage in the wall between the cable spreading room and the reactor building in Browns Ferry, Unit 1, in March, 1975, led to a fire in the cable spreading room. As a result of this experience, many improvements and changes in fire protection procedures at nuclear power plants are now in effect. In addition to terminating the use of open flame for testing air leaks, recommendations resulting from this event include the requirement that (a) control and power cables for a ventilation system that is important in fire control should not be routed through areas the system must ventilate in case of fire; (b) ventilation designs should be provided with the capability of isolating fires by use of cutout valves and dampers; (c) capability for the control of ventilation systems to deal with fire and smoke should be provided, but such provisions should be compatible with requirements for the containment of radioactive materials.

Fire Protection Systems

Because of the long recognized possibility of fires in standby gas treatment systems, equipment has been installed to deluge the charcoal with water in case of a fire. In one instance in 1976 and three in 1978, shorts in the electrical wiring have caused this equipment to actuate unnecessarily and flood the charcoal, thereby requiring complete replacement of the adsorbent. This is particularly noteworthy inasmuch as equipment designed to prevent or correct problems has actually been a source of failure itself. Similar problems have occurred in diesel generator rooms where defective smoke detectors have led to the discharge of CO_2 fire protection systems.

Problems in PWRs

The reported events for PWRs relate to the commonly expected problems associated with primary containment and its associated spray system, as well as to the perhaps less expected problems associated with waste processing systems. Again, there is a large number of events related to equipment designed to sample and monitor the performance of ventilating and air cleaning systems. In this case, however, the percentage of such failures in terms of all reported failures was only 32%, as contrasted to 51% for BWRs. The percentage of air sampling and monitoring failures within the primary containment due to the sampling portion of the system was about 42%, essentially the same as the value (41%) for BWRs. Of the failures within the detectors and analyzers themselves, the data show that 61% were with gaseous and particulate radiation monitors, 30% were with pressure sensors, and 9% were with hydrogen and oxygen analyzers. There were no reported failures of temperature monitors.

With respect to specific failures in air monitoring, ventilating and cleaning systems in PWRs, the following appear to be worthy of note.

Table III. Summary of licensee event reports

Air Monitoring, Ventilating and Cleaning Systems

(Pressurized Water Reactors, 1975 - 1978)

System	Component	Nature of Problem	Number of Events 1975	1976	1977	1978*
Primary containment	Atmospheric sampling and monitoring	Failure of sampling system	8	18	27	17
		Failure of detector or analysis unit	17	21	29	29
	Diluting or ventilating system	Deficiencies in purge (ventilation) system	23	26	35	11
		Deficiencies in filter system	4	1	7	6
		Deficiencies in air cooling system	28	17	28	15
	Emergency combustible gas control	Failure of recombiners	0	1	0	1
	Containment spray system	Spray system not available	11	5	13	8
		Performance degraded	6	7	12	6
	Containment isolation system	Vacuum breaker inoperable	2	2	3	0

*Through approximately June 30, 1978

System	Component	Nature of Problem	Number of Events 1975	1976	1977	1978*
Reactor cooling system (primary)	Pumps, valves, pressurizer, sensing lines, etc.	Excessive airborne release	10	7	12	5
Control room	Normal ventilating system	Failure of dampers or fans, or design error	0	2	7	4
		Failure of heaters, coolers, or compressors	0	2	1	2
	Emergency ventilating system	Failure of dampers or fans, or design error	1	0	3	1
		Leak in header	0	0	0	1
	Emergency sampling and monitoring system	Failure of chlorine or radiation detector	0	1	2	1
Enclosure building	Ventilating system	Water in off-gas pipe	1	0	0	0
Fuel storage building	Ventilating system	Degraded charcoal filter	0	0	0	1
Auxiliary building	Sampling and monitoring system	Failure of radiation monitor	0	0	0	3
	Ventilating system	Failure of dampers, or fans, or loss of power	3	0	4	1
		Filter plugging or heater failure	0	1	1	0

*Through approximately June 30, 1978

System	Component	Nature of Problem	Number of Events 1975	1976	1977	1978*
Diesel generator room	Ventilating system	Failure of dampers or fans, or design error	3	2	2	0
		Failure of air cooler	0	1	0	0
Turbine building	Atmospheric sampling and monitoring	Failure of radiation monitor	0	0	1	0
Switch gear room	Ventilating system	Failure of air cooler	0	1	0	0
Waste processing system	Waste gas decay tank	Excessive airborne release	12	8	5	0
	Waste gas surge tank	Excessive airborne release	3	1	1	0
	Waste gas vent header	Failure of radiation monitor on loop seal	0	0	2	0
		Excessive airborne release	3	1	5	3
	Liquid waste system	Excessive airborne release	3	1	1	0
Main stack discharge system	Sampling and monitoring system	Failure of sampling system	0	2	2	1
		Failure of radiation monitor	3	2	6	2
	Air flow system	Failure of exhaust fan	0	1	1	0
		TOTALS	141	131	210	120

*Through approximately June 30, 1978

Excessive Airborne Releases

Normal procedures call for the venting of the mixed-bed demineralizer, using primary makeup water that contains no radioactive gases. Because of personnel error, venting at one PWR in 1975 was done with the demineralizer connected to the reactor coolant system. This resulted in the release of radioactive gases from the cooling system to the auxiliary building through a loose manhole cover on the equipment drain tank. As a result of this mishap, in which an estimated 63 curies of radioactive gases were released, the operating procedures have been upgraded and the manhole cover has been tightened.(3)

Ice Condenser Pressure Suppression System

Effective operation of the ice condenser system requires that an adequate amount of ice be maintained in the containment at all times. To accomplish this task, the ice bays are separated from the rest of containment by a panorama of doors which are designed to easily open to permit any steam released within containment to enter the ice chambers and be condensed. Initial operation of these systems has revealed some problems with ice forming on the doors and hinge mechanisms, through condensation on the outside and bottom of the door frame. Since lake water is used to cool the containment atmosphere in the specific plant in which this problem occurred, it is postulated that the higher ambient temperature during summer months led to the condensation of vapor from the atmosphere on the cold door frames. From there, the water seeped into the insulation where it subsequently froze. To alleviate the problem, the floor cooling system controls have been adjusted to increase the floor temperature, the soaked insulation has been replaced and periodic inspections of the doors have been instituted.(3)

Failures in Containment Spray Systems

One of the engineered safety features for dealing with a postulated loss of coolant accident in a PWR is the containment spray system. Initially, such a system will take water from the Raw Water Storage Tank (RWST) and pump it through the containment sprays. After being sprayed, the water collects in sumps at the bottom of containment. When the supply in the RWST has been exhausted, the spray system continues by recirculating the water from the sumps. As the spray cools the containment, however, the containment pressure decreases. Calculations performed in 1977 showed that this pressure decrease could lead to a reduction in the Net Positive Suction Head (NPSH) and result in cavitation in the recirculating pumps. Since proper performance of the pumps is essential to long term cooling of containment and the pumps could be damaged if operated for a period of time without water, this situation called for a thorough analysis of the implications of the NPSH problem to the overall performance of containment spray systems. The problem was solved by showing that NPSH would not be reduced to the point that the pumps would be damaged.

Because of pumps being locked out of service, valves misaligned, or loss of power, there were 37 events reported over the three and one half year period of this study in which the containment spray system was unavailable. Of these, over 90% were due to human error. In addition, there were 31 events in which the performance of this system was degraded. In the Reactor Safety Study(7), it was estimated that there were about three chances in one thousand that both of the duplicate spray systems would be unavailable because the valves were not properly realigned after testing or the sensors that activate the systems were not properly calibrated. During the period covered by this study, there were approximately 113 reactor-years of operations. Assuming the containment spray systems in each PWR nuclear power plant were tested monthly, this means a total of about 1350 tests were conducted. Using these data, one can calculate that the 37 reports of unavailability over this period represent a probability of about 28 per thousand that one of the two containment spray systems would be unavailable. The probability that both systems would be unavailable would be about one in one thousand, which is substantially in agreement with the estimate given in the Reactor Safety Study.

Containment Purging Reduced to Alleviate Airborne Releases

Excessive airborne releases at one plant in 1977 led to a decision to reduce the frequency of containment purging. A factor entering into this decision was that the plant had 36 inch diameter purge lines and the NRC prefers not to permit continuous purging unless smaller 8 inch lines have been installed. As a result of the reduction in the frequency of purging, the plant's airborne releases were reduced. At the same time, however, this led to a reduction in the frequency with which containment could be entered for visual inspections of safety related equipment, such as piping, snubbers, etc. This is a good example of the interactions of various systems within nuclear power reactors and highlights the importance of good air cleaning equipment for the safe operation of such facilities.

Auxiliary Building Exhaust Fan Causes Reactor Trip

Another example of systems interactions was the instance in which an operator, in anticipation of taking an air flow reading, started the second auxiliary building ventilation fan. Because the discharge damper on the fan was leaking, back pressure from the operating fan caused the second idle fan to be rotating backwards. When it was switched on, a large starting current was demanded. This led, for unknown reasons, to tripping of the MCC-6 supply breaker, instead of the fan breaker, and caused a reactor trip and safety injection. The discharge damper was adjusted and the breakers examined.

Impact of Faulty Sensors

A review of the reported events revealed a number of instances in which failure of a sensor led to difficulties. For example, because of the failure of a temperature sensor in the air intake tunnel, the deluge fire protection system at one PWR plant activated. This led, in turn, to the loss of the building ventilation system for a period of two and one-half hours. In another situation, the annulus emergency ventilation system was found to be inoperable. The problem was traced to an incorrect reading from a pressure sensor. The atmospheric pressure tap had been taped over during painting of the auxiliary building outside walls and had not been removed. In a related incident at another PWR power plant, the performance of the containment spray system was degraded because tape was left on the spray nozzles following painting of containment.

During the performance of preventive maintenance to verify operation of the deluge system on a shield building ventilation system, a heat gun was used to elevate the temperature of a heat detector. Activation of the detector, however, locked out train A of the shield building exhaust and recirculation fans, disabling that train. Since train B had not been tested, this violated the Technical Specifications. The procedure was inadequate in that it did not include evaluation of the consequences of activating the heat sensor.

In another occurrence at the same plant, a diesel generator was taken out of service concurrently with the outage of safeguards train A special zone ventilation fan. Subsequently, the special ventilation fan in safeguards train B was tested, with the operating personnel failing to recognize that a safeguards train A component was out of service prior to placing the safeguards train B emergency power out of service.

Violation of Single Failure Criterion

During a review of the electrical circuitry associated with the containment ventilation isolation valves, it was found that the single failure criterion could not be satisfied for a postulated short circuit or foreign voltage imposition in the control circuitry for these valves. All four valves were controlled from the same electrical circuit. The same condition existed for the three containment pressure relief valves. Modifications were made to provide independent circuitry to one supply and one exhaust valve and both pressure relief valves outside containment.

Failure of Hydrogen Recombiners

Although not frequent, there were several reports of failures of emergency recombiners during the period of this study. For example, during the semiannual operational check at one plant in 1976, it was found that a hydrogen recombiner could be loaded only to 43KW. The resulting heater temperature was 170°F below the Technical Specification limit. Preliminary inspection indicated that one phase of the heater was grounded. A similar failure was reported in 1978.

Ventilation and Instrument Performance

The importance of ventilation as a cooling source was well illustrated by an event in a PWR power plant in 1975 wherein a critical instrumentation bus grid was lost due to high ambient temperature. The cause was determined to be excessive ambient temperature during a high load on the inverter. The system was redesigned to improve local ventilation. In another plant in 1977, the air ejector radiation monitor blower tripped off. The cause was an overheated condition in the monitor cabinet due to poor ventilation.

During a refueling outage at another plant, condensation built up and moisture shorted out the detector tube for the component cooling heat exchanger radiation monitor. The source was condensation buildup on the service water piping in the auxiliary building. Although the detector tube was replaced and the piping in the area of the monitor was subsequently insulated, the new detector tube failed, apparently because moisture had entered the detector tube chamber prior to installation of the isulation.

Ventilation Systems and Containment Pressure

In 1977, a shift in the wind over a lake led to an alteration in the water flow and a sudden change in the temperature of the cooling water flow to the circulating and service water pumps in a PWR plant. This resulted in an increase in the temperature and pressure within containment and caused two of the four containment high pressure signals to alarm, even though the actual pressure increase was only about 1 psi. To correct the situation, the containment was vented. In a contrasting event, a decrease in the outside ambient temperature and a concurrent decrease in the component cooling water temperature serving the containment recirculation fans at another PWR plant caused the containment temperature to drop below 100°F, which was in violation of the Technical Specifications. The fans were secured and service water to the component heat exchangers was throttled back to raise the component cooling water temperature.

Problems with Waste Gas Processing Systems

Although the number is decreasing, the frequency of excessive airborne releases from waste gas decay tanks in PWRs appears to be high. Twelve gases were reported in 1975, eight in 1976 and five in 1977. In addition, there were 22 reports of excessive airborne releases from other components within plant waste processing systems. When one considers that a typical PWR pressurized waste gas decay tank may contain a considerable radionuclide inventory (upwards of one third that in the charcoal beds of a BWR off-gas system), it may be that more attention to developing procedures for avoiding these releases is warranted.

III. Commentary

Although there is a variety of ways in which the data from this study can be interpreted, one message is clear. There is a need for research on the development of more reliable equipment for sampling and monitoring air systems. Equipment that provides inaccurate data on the performance of such systems can lead to as many problems as inadequacies in the systems themselves.

These analyses have also shown that LER data can be used to gain a better understanding of the various inputs required for studies of the risks associated with the operation of nuclear power plants. One example was the observed frequency for the unavailability of containment spray systems within PWRs. The data from this study support the estimates used in the Reactor Safety Study. There are undoubtedly other instances in which LER data can be used to confirm or improve failure rate probability estimates for other safety systems.

As this author has pointed out in a previous report,[1] a review and analysis of LERs pertaining to air monitoring, ventilating and cleaning systems is a difficult task. One of the major reasons for this is that LERs pertaining to this subject area are not easy to extract from the totality of reported events. In addition, there are many variations in the way in which individual nuclear plant operators report LERs, as well as in the key words selected for recording them in the data bank. Events are classified under a variety of titles and frequently the titles are misleading from the standpoint of the air cleaning implications of the event. Many licensees, for example, appear to use the words, "reactor building," "shield building," and "containment" interchangeably. Others appear to take a similar approach to the use of the terms, "air cooling," "ventilation," and "purging," while others do not distinguish between systems for normal operation versus those for emergency situations.

Further compounding the problem is the fact that the indexes are not designed to be of maximum help to air cleaning specialists. For example, the index to the 1976 list of LERs for PWRs[5] contains only one item under the subject of "air cleaning"; the corresponding list for BWRs[4] contains only two items under this category. The reports for 1975 contain no items within this category.[2][3] Since the use and analyses of these data can yield a multitude of benefits in terms of improved nuclear air monitoring, ventilating and cleaning systems, it would appear that the Nuclear Regulatory Commission should be encouraged to conduct a study to determine and implement approaches to improve the methods by which LERs pertaining to this subject area are logged into the system.

IV. References

1. Moeller, D.W., "Performance of air cleaning systems in nuclear facilities," Proceedings of the 13th AEC Air Cleaning Conference, U.S. Atomic Energy Commission, Report CONF-740807, 1:10-28 (1975).

2. Scott, R.L. and Gallaher, R.B., "Annotated bibliography of safety-related occurrences in boiling-water nuclear power plants as reported in 1975," Report ORNL/NUREG/NSIC-126, Nuclear Safety Information Center, Oak Ridge National Laboratory, Oak Ridge, TN (1976).

3. Scott, R.L. and Gallaher, R.B., "Annotated bibliography of safety-related occurrences in pressurized-water nuclear power plants as reported in 1975," Report ORNL/NUREG/NSIC-127, Nuclear Safety Information Center, Oak Ridge National Laboratory, Oak Ridge, TN (1976).

4. Scott, R.L. and Gallaher, R.B., "Annotated bibliography of safety-related occurrences in boiling-water nuclear power plants as reported in 1976," Report ORNL/NUREG/NSCI-137, Nuclear Safety Information Center, Oak Ridge National Laboratory, Oak Ridge, TN (1977).

5. Scott, R.L. and Gallaher, R.B., "Annotated bibliography of safety-related occurrences in pressurized-water nuclear power plants as reported in 1976," Report ORNL/NUREG/NSIC-138, Nuclear Safety Information Center, Oak Ridge National Laboratory, Oak Ridge, TN (1977).

6. Private communication, Nuclear Safety Information Center, Oak Ridge National Laboratory, Oak Ridge, TN (1978).

7. Nuclear Regulatory Commission, "Reactor safety study: an assessment of accident risks in U.S. commercial nuclear power plants," Report WASH-1400 (NUREG-75-014), National Technical Information Service, Springfield, Virginia (1975).

DISCUSSION

GIBSON: What is being done about improving the reliability of air monitoring and sampling systems?

MOELLER: One step that has recently been taken is a reaffirmation of the American National Standards Institute Guide to "Sampling Airborne Radioactive Materials in Nuclear Facilities" (BSR N13.1-1969). On the basis of the data reported here today, however, I believe much more needs to be done. In particular, DOE and/or NRC need to sponsor a research program specifically directed to the types of failures in such systems as recorded in the LER's.

FIRST: In addition to engineering and careful design, we badly need standards which will point the way and avoid the kinds of accidents we have been hearing about. The next two speakers will describe the very active program of nuclear standards development which had been supported by the Atomic Energy Commission and ERDA and is now supported by the Department of Energy. Its importance is well recognized.

PROGRESS IN STANDARDS FOR NUCLEAR AIR AND GAS TREATMENT*

C. A. Burchsted
Nuclear Standards Office
Oak Ridge National Laboratory
Oak Ridge, Tennessee

Abstract

Standardization in nuclear air and gas treatment spans a period of more than 25 years, starting with military specifications for HEPA filters and filter media, and now progressing to the development of a formal code analogous to the ASME Boiler and Pressure Vessel Code. Whereas the current standard for components and installation of nuclear air cleaning systems is limited to safety related facilities for nuclear power plants, the proposed code will cover all types of critical ventilation and air and gas treatment installations for all types of nuclear facilities.

Introduction

Standards for ventilation and offgas-treatment in nuclear facilities span a period of at least 25 years, progressing from a standard specification for the basic high efficiency particulate air (HEPA) filter and filter medium to the present effort to develop a formal Code for the technology. The next paper of this conference will review the present status of the Code program. The purpose of this paper is to review the background that has led up to that Code.

Early Standards

The earliest standards in this field were military specifications MIL-F-51068 and MIL-F-51079 which covered requirements for HEPA filters and HEPA filter medium, respectively. To avoid the trap of distinguishing between a standard and a specification, these documents can be categorized as "standard specifications" -- that is, documented minimum requirements for performance, construction, and testing of the basic component that is at the heart of nearly every nuclear exhaust and process-offgas treatment system. Although these standards were developed by the Army's chemical warfare service to describe what was in the early 1950's a primarily military component, they have undergone considerable modification over the years to meet the needs of the nuclear industry. We are now at the point of issuing the fifth revision of MIL-F-51068 and the third revision of MIL-F-51079. MIL-STD-282, which provided the standard test for establishing the primary performance characteristic of the HEPA filter (the monodisperse dioctyl phthalate [DOP] particle-removal-efficiency test) was published concurrently with these standards. A new edition of that standard will be issued in the near future to reflect modifications of the test introduced by the nuclear industry.

About the time that the military standards were being issued, Underwriters Laboratories (UL), at the urging of Humphrey Gilbert of the U. S. Atomic Energy

*Research sponsored by the Division of Nuclear Power Development, U. S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

Commission (AEC), undertook the development of a series of tests to define and measure the minimum fire and hot air resistance of HEPA filters. This culminated in UL-586, Safety Standard for HEPA Filters, which has since been adopted as an American National Standard (ANSI/UL 586). With publication of UL-586 and a series of follow-on meetings at UL headquarters in Chicago aimed at unifying critical requirements of HEPA filters, a core of individuals concerned with standardization in the nuclear air cleaning field began to take form. This small group, which has been in large measure shepherded by Mr. Gilbert, has been largely responsible for the development standards for the nuclear air cleaning industry and remains central to its standards to this day. Gilbert seldom served directly on standards-writing work groups, but was often the prime mover in getting the standards started and in goading the work groups on to completion. This group included representatives of several AEC [now Department of Energy (DOE)] contractors, the Naval Research Laboratory, all of the HEPA filter manufacturers, and later the major manufacturers of nuclear grade activated carbon. It was, and still is, well balanced between producers and users in the industry.

During the 1960's the core group was called upon to work with the American Association for Contamination Control (AACC, since merged with the Institute of Environmental Sciences, or IES) to develop standards relating to clean rooms and clean air devices. Although this may appear anomalous, the AEC, as the nation's single largest user and operator of clean rooms at that time, had an important stake in the development of effective standards for critical aspects of those facilities. The AACC effort crystallized the core group of nuclear standards personnel, and it was to this group that the Nuclear Technical Advisory Board (NTAB, now the Nuclear Standards Management Board, or NSMB of the American National Standards Institute) turned to provide the expertise for the development of standards for the air cleaning facilities of commercial nuclear power plants. The closest thing to a system standard at that time was an AEC report, ORNL-NSIC-65,[1] and its emphasis was on the needs of AEC research reactors and laboratory facilities. The original charter of the committee initated by NTAB, Nuclear Standards Committee N45-8, was to prepare a standard for boiling water reactor standby gas treatment systems. From the first meeting in the summer of 1971 it was obvious that this scope was much too narrow. The committee quickly broadened the scope to include all engineered safety feature (ESF) air cleaning systems for all nuclear power plants, this with the approval of NTAB. It was decided that the most effective approach to this project was to develop minimum specifications for each of the critical components of an ESF air cleaning system -- filters, adsorbers, demisters, housings, fans, ducts, dampers, etc. -- and procedures for the tests needed to evaluate the acceptability and performance of those components after they were installed. These became, in a sense, the basic building blocks of the standards. From this effort came the two standards which are basic to the nuclear air cleaning industry today, ANSI N509, *Standard for Nuclear Power Plant Air Cleaning Units and Components*, and ANSI N510, *Standard for Testing of Nuclear Air Cleaning Systems* (published 1976 and 1975, respectively).

Current Standards Activities

The earliest standards efforts stemmed from the need of users to define their requirements for critical components to industry, and the need of industry to have those requirements defined clearly, uniformly, and without equivocation so that competition among manufacturers could be on an equal footing. As the nuclear industry developed, other participants came into the picture -- architect-engineers (A-E's), utilities, nongovernment laboratories, consultants, and now, increasingly, the general public. There arose the need for more than simply specifying components and equipment items. There arose the need for defining minimum requirements for and performance of integrated systems made up of these basic building blocks for specific applications in the interest of public safety. This took place, and is taking place, not only in air cleaning, but throughout the nuclear industry. It may be noted, however, that this branch of the nuclear industry appears to have anticipated the trend, being among the earliest in the nuclear standards "game".

The first response to the expanded interest in nuclear standards was seen in the AEC Safety Guides, now the Regulatory Guides (RG) of the Nuclear Regulatory Commission (NRC). The guide of major interest to us has been RG 1.52, *Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption System Units of Light-Water-Cooled Nuclear Power Plants* (initial issue June 1973). This guide has undergone one revision and a second is in progress. Other guides dealing with non-ESF air cleaning systems are also coming. The Regulatory Guides, strictly speaking, are not standards but only recommendations of the NRC as to what they consider minimum requirements. Insofar as possible, NRC desires that the Regulatory Guides be little more than documents that invoke national standards developed under the voluntary consensus system. The guides were developed to fill the vacuum that existed because no suitable consensus standards were available; the guides may be phased out as suitable standards are developed.

Expansion of interest in nuclear standards has also led to the present effort under the American Nuclear Society (ANS) to develop a family of system standards covering requirements specific to certain types of air cleaning systems for each major category of reactor, for specific types of fuel manufacturing and processing facilities, hot cells, and so on. These will not duplicate the requirements for basic building blocks of such systems, and specified in ANSI N509 and N510, but will provide the framework for invoking the appropriate portions of those standards for specific applications. Although none of the ANS standards have been published to date, most are in one phase or another of the consensus process that leads to publication (see Appendix).

The most important current effort in the nuclear air cleaning standards arena is the development of a formal Nuclear Air and Gas Treatment (NA&GT) Code. Development of this Code, which has been assigned to the American Society of Mechanical Engineers (ASME), is an outgrowth of the NTAB-NSMB N45-8 activity. In the fall of 1976, N45-8 was transferred to sponsorship of ASME and became the ASME Committee on Nuclear Air and Gas Treatment (CONAGT). Although the committee's first efforts have been aimed at updating and correcting certain deficiencies of the current ANSI N509 and N510, its charter was the development of a code analogous to the ASME Boiler and Pressure Vessel Code which will eventually replace those standards. Whereas N509 was limited, by scope, to ESF systems of nuclear power generation plants, the proposed Code will cover all essential ventilation, air cleaning, and process-offgas treatment equipment for all types of nuclear facilities. The writing of the Code will not be an easy job and will involve inputs from a great number of technical and professional societies, trade associations, and governmental organizations. Mr. J. F. Fish, Chairman of CONAGT, will review the present status of the Code effort in the next paper.

As did ANSI N509, the NA&GT Code will draw upon existing technology. Wherever possible, requirements of the Code will be defined by reference to documents of other organizations. There will be no restating or inventing of requirements that have been previously, and perhaps better stated in existing standards. It will be recognized that components, procedures, and certain functional guides are basic building blocks of the industry, and that their requirements have been adequately defined by various professional or technical societies (e.g., ASME, IEEE, ASTM) or trade associations (e.g., SMACNA, AMCA, ARI). The function of the Code will be to tie these together into a comprehensive whole, to eliminate (by exception) portions of the reference documents that are irrelevant or inadequate, and to supplement them as necessary to alleviate any shortcomings. Once invoked, together with any exceptions and/or supplementary requirements, these documents will assume the mandatory and legal status of the Code when applied within the framework of the Code. On the other hand, they will not have Code status when applied outside of the context of the Code, and therefore their usefulness and applicability in other industrial contexts will not be diminished. A list of organizations and relevant documents that could be considered for Code investment is given in the Appendix.

Codes vs Standards

Much fruitless effort has been expended in technical committees over the years in trying to make a distinction between what is a code and what is a standard. The following definitions from Webster may serve to avoid this trap in the furtherance of the current Code effort:(2)

- Standard - something that is established by authority or general consent as a model or example to be followed;

- Code - a set of rules of procedure and standards...designed to secure uniformity and protect the public interest. A set of rules for, or standards of professional practice set up by an organized group and...commonly having the force of law in a particular jurisdiction.

That is, a code is of itself a standard. But it is more. Whereas a standard is a model to be followed and implies some choice in its following, a code is mandatory and carries the force of law when invoked in the statutes of a political jurisdiction. Furthermore, it contains rules for professional practice and has the express purpose of protecting the public interest.

The stake of the public in nuclear air and gas treatment is substantially greater than in industrial ventilation and air pollution control. Much of the concern expressed by members of the public who fear nuclear energy can be reduced to a fear of the gaseous release of radioactivity. The gaseous release is the most probable mechanism by which the public would be exposed to radioactivity, both from the "expectable" malfunction that may occur in the normal operation of a nuclear plant and in the event of a "maximum credible accident". The equipment and systems covered by the proposed NA&GT Code constitute the single most important mechanism for countering any gaseous release. The importance of the program, therefore, is clear.

On the other hand, there has been a certain suspicion of standards in general, and codes in particular, on the part of industry over the years. Some manufacturers of equipment that has proven highly competent and reliable in the climate of industry have tended toward a *laissez faire* attitude in adapting those items to the requirements of the nuclear industry -- the products have withstood the test of time, why change now? Some feared that the restrictive requirements of the nuclear submarine program might be imposed on them, and openly resisted standardization efforts. These attitudes have been largely dispelled in this day, but there still remains a belief on the part of some scientists, engineers, and managers that the imposition of mandatory standards, in a free society, may be stifling and perhaps would inhibit or prevent the exercise of proper technical judgment in particular situations calling for such judgment. There is also the feeling that standardization necessitates compromise and reduces technical judgment to the lowest common denominator of the standardizers; since, in our society and system of voluntary standards, anyone has the right to be heard in matters of standardization, this is considered to mean reduction to the lowest common denominator of the general public. But this is not true.

The consensus process by which codes and standards are reviewed, and eventually reach the stage of approval and publication, not only enables every voice to be heard, it also provides the means for resolving disputes and rejecting the unsound or irrational viewpoint or objection. Consensus does not mean unanimity. It is unanimity that reduces technology to a lowest common denominator, not consensus. Consensus does require compromise, but it is informed compromise. The typical standard first goes through a series of prescribed reviews and evaluations by the writing group and by whatever subcommittees and committees the sponsoring organization deems necessary to achieve the required level of competence and exposition. At that stage it can be adopted and published as an authorized standard of the sponsoring organization. To become an American National (ANSI) Standard, it must go through an additional, and broader series of reviews, including review by the general public under the auspices of ANSI, to ensure that all interested parties have had the opportunity to assess its implications or to appeal from its proposed requirements. This, briefly, is the consensus process. It provides for competent peer review, for all points of view to be exercised and ensures that elements that may have been overlooked by the experts are given due consideration.

Every supplier to the nuclear air and gas treatment industry has its own standards for controlling the work that he does. The mechanism of consensus and ANSI standards reduces the possibility that those supplier standards will bend to market-place pressures or the expediency of an immediate situation. There is concern in the private sector, however, that such national standards can become regulatory and cannot be changed when technical judgment indicates that change is needed. The public sector, on the other hand, is sometimes suspicious of the motives behind the technical judgments of industry and needs some way to have confidence that such judgments are based on the public interest. Standards, as noted earlier, even national standards, have an aura of choice; they imply an ability to deviate from agreed upon guidelines. Codes, on the other hand, are more rigid; when invoked they have the force of law. Although developed by the same competent people who develop standards, they necessarily are subject to the additional public review of an ANSI standard. Finally, they are developed and maintained in the traditional voluntary manner, not by lawmakers or regulatory agencies. The force of law comes only after the code has been developed to the satisfaction of all participant parties and has been invoked in the statues of a political or regulatory jurisdiction.

The proposed NA&GT Code is a logical and evolutionary development of the previous standards efforts and of the concerns of both industry and the public. It provides a fundamental and necessary framework which, if properly understood and applied, should allay the fears of both the public and private sectors. There may still be the feeling on the part of individual engineers and scientists that, although the objectives of the Code are sound, he is somehow the victim of a system that will inevitably stifle his individuality. To that engineer or scientist, let it be said that no amount of standardization can hope to provide for all contingencies that will be met in its application. The Code can ensure that competent design and day-to-day processes and functions of the system are carried out in the best possible manner, but beyond that lies an illimitable area for individual judgment and action. The requirements for original and creative thought and the exercise of technical judgment and responsibility remain undiminished. The writing and review of this Code will involve the donation of a great deal of time and effort over the next few years by a large number of people -- it is hoped that the members of this audience will respond appropriately when called upon.

References

1. C. A. Burchsted and A. B. Fuller, *Design, Construction and Testing of High-Efficiency Air Filtration Systems for Nuclear Application,* USAEC Report ORNL-NSIC-65, Oak Ridge National Laboratory, Jan. 1970.

2. *Webster's Third New International Dictionary,* G. and C. Meriam Co., Springfield, MA, 1959.

APPENDIX

STANDARDS ORGANIZATIONS AND DOCUMENTS RELEVANT TO THE PROPOSED ASME NUCLEAR AIR AND GAS TREATMENT CODE

American National Standards Institute (ANSI)

American Society of Mechanical Engineers (ASME)

ANSI N509	Nuclear Power Plant Air Cleaning Units and Components	(P)
ANSI N510	Testing of Nuclear Air Cleaning Systems	(P)
ANSI N45.2	Quality Assurance Programs for Nuclear Power Plants	(P)
ANSI N45.2.1	Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants	(R)
ANSI N45.2.2	Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants	(P)
ANSI N45.2.6	Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Facilities	(R)
ASME III	Boiler and Pressure Vessel Code, Nuclear Power Plant Components	(P)
III-ND	Class 3 Components	(P)
V	Nondestructive Examination	(P)
IX	Welding and Brazing Qualifications	(P)

American Nuclear Society (ANS)

ANSI N101.6	Concrete Radiation Shields	(P)
ANSI N101.2	Protective Coatings (Paints) for Light-Water Nuclear Reactor Containment Facilities	(P)
ANSI N101.4	Quality Assurance for Protective Coatings Applied to Nuclear Facilities	(P)
ANSI N512	Protective Coatings (Paints) for the Nuclear Industry	(P)
ANSI N202	Radioactive Gas Waste System for the Stationary Gas-Cooled Reactor Plant	(D)
ANSI N657	Gas-Cooled Reactor Plant Containment Atmospheric Clean-up System	(D)
ANSI N720	Gaseous Radioactive Waste Processing Systems for Light-Water Reactor Plants	(R)
ANSI N275	Containment Hydrogen Control	(D)
ANSI N276	Boiling Water Reactor Containment Ventilation Systems	(R)
ANSI N277	Pressurized Water Reactor Containment Ventilation Systems	(R)
ANSI N189	Safety-Related Ventilation Systems	(D)
ANSI N290	Design, Construction, and Operation of Ventilation Systems for Mixed Oxide (UO_2-PuO_2) Fuel Fabrication Plants	(R)
ANSI N303	Guide for Control of Gasborne Radioactive Materials at Nuclear Fuel Reprocessing Facilities	(R)

Institute of Electrical and Electronic Engineers (IEEE)

ANSI C50.20	Test Code for Polyphase Induction Motors and Generators	(P)
ANSI N41.7	Guide for Seismic Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations	(P)

(P) = Published, current edition; (D) = Under development; (R) = Under review.

ANSI C2	National Electrical Safety Code	(P)
IEEE 279	Criteria for Protection Systems for Nuclear Power Generating Stations	(P)
IEEE 323	Qualifying Class IE Equipment for Nuclear Power Generating Stations	(P)
IEEE 334	Type Tests of Continuous Duty Class IE Motors for Nuclear Power Generating Stations	(P)
IEEE 336	Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	(P)
IEEE 338	Periodic Testing of Nuclear Power Generating Station Safety Systems	(P)
IEEE 344	Siesmic Qualification of Class I Electric Equipment for Nuclear Power Generating Station	(P)
IEEE 383	Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations	(P)
IEEE 384	Criteria for Separation of Class IE Equipment and Circuits	(P)
IEEE 415	Planning of Pre-operational Testing Programs for Class IE Power Systems for Nuclear Power Generating Stations	(P)
IEEE 420	Guide for Class IE Control Switchboards for Nuclear Power Generating Stations	(P)

<u>Institute of Environmental Sciences (IES)</u>

IES/AACC CS 1	HEPA Filters	(P)
IES/AACC CS 8	High-Efficiency Gas-Phase Adsorber Cells	(P)

<u>American Society of Heating, Refrigeration, and Air Conditioning Engineers (ASHRAE)</u>

ASHRAE 52	Method of Testing Air Cleaning Devices Used in General Ventilation for Removing Particulate Matter	(P)
ASHRAE 37	Methods of Testing and Rating Unitary Air Conditioning and Heat Pump Equipment	(P)
ASHRAE 68P	Method of Testing Sound Power Radiated Into Ducts from Air Moving Devices	(P)
ASHRAE 62	Natural and Mechanical Ventilation	(P)

<u>Underwriters Laboratories (UL)</u>

UL 586	Safety Standard for High Efficiency Particulate Air Filters	(P)
UL 900	Safety Standard for Air Filter Units	(P)

<u>Air Moving and Conditioning Association (AMCA)</u>

AMCA 99	Standards Handbook	(P)
AMCA 201	Fan Application Manual - Fans and Systems	(P)
AMCA 202	Fan Application Manual - Troubleshooting	(P)
AMCA 203	Fan Application Manual - Field Performance Measurements	(P)
AMCA 210	Test Code for Air Moving Devices	(P)
AMCA 300	Test Code for Sound Rating	(P)
AMCA 500	Test Methods for Louvers, Dampers, and Shutters	(P)

<u>American Welding Society (AWS)</u>

AWS D 1.1	Structural Welding Code	(P)

National Fire Protection Association (NFPA)

NFPA 90A	Installation of Air Conditioning and Ventilating Systems	(P)
NFPA 90B	Installation of Warm Air Heating and Air Conditioning Systems	(P)
NFPA 91	Installation of Blower and Exhaust Systems	(P)

Air Conditioning and Refrigerating Institute (ARI)

ARI 410	Standard for Forced Circulation Air Cooling and Air Heating Coils	(P)
ARI 680	Standard for Air Filter Equipment	(P)

National Electrical Manufacturers Association (NEMA)

IS 1.1	Enclosures for Industrial Controls and Systems	(P)
IS 2.2	Specification Guide for Industrial Motor Drive Systems	(P)
MG 1	Motors and Generators	(P)
MG 2	Safety Standard for Construction and Guide for Selection, Installation, and Use of Motors and Generators	(P)

American Society for Testing and Materials (ASTM)

ASTM A36	Specification for Structural Steel	(P)
ASTM A123	Specification for Zinc Hot-Galvanized Coatings on Products Fabricated from Rolled, Pressed, and Forged Steel Shapes, Plates, Bar, and Strip	(P)
ASTM A164	Specification for Electrodeposited Coating of Zinc on Steel	(P)
ASTM A167	Specification for Stainless and Heat-Resisting Chromium-Nickel Steel Plate, Sheet, and Strip	(P)
ASTM A283	Low and Intermediate Tensile Strength Carbon Steel Plates of Structural Quality	(P)
ASTM A525	Specification for Steel Sheet, Zinc-Coated (Galvanized) by the Hot-Dip Process, General Requirements	(P)
ASTM A526	Specification for Steel Sheet, Zinc-Coated (Galvanized) by the Hot-Dip Process, Commercial Quality	(P)
ASTM A527	Specification for Steel Sheet, Zinc-Coated (Galvanized) by the Hot-Dip Process, Lock Forming Quality	(P)
ASTM A570	Hot-Rolled Carbon Steel Sheet and Strip, Structural Quality	(P)
ASTM A606	Specification for Steel Sheet and Strip, Hot-Rolled and Cold-Rolled, High-Strength, Low-Alloy with Improved Corrosion Resistance	(P)
ASTM A607	Specification for Steel Sheet and Strip, Hot-Rolled and Cold-Rolled, High-Strength, Low-Alloy Columbium and/or Vanadium	(P)
ASTM A666	Specification for Austenitic Stainless Steel, Sheet, Strip, Plate and Flat Bar for Structural Applications	(P)
ASTM D2854	Specification for Test for Apparent Density of Activated Carbon	(P)
ASTM D2862	Specification for Test for Particle Size Distribution of Granulated Activated Carbon	(P)
ASTM E11	Specification for Wire Cloth Sieves for Testing Purposes	(P)
ASTM D2866	Test for Total Ash Content of Activated Carbon	(P)
ASTM D2867	Test for Moisture in Activated Carbon	(P)

ASTM Dxxxx	Test for Radioiodine Testing of Nuclear Grade Gas-Phase Adsorbents	(R)
ASTM Dxxxx	Test for pH of Activated Carbon	(R)
ASTM Dxxxx	Test for Ball Pan Hardness of Activated Carbon	(R)
ASTM Dxxxx	Analysis for Potassium and Iodine Impregnants in Activated Carbon	(R)
ASTM Dxxxx	Analysis for Tiethylenediamine (TEDA) Impregnant in Activated Carbon	(R)
ASTM A245	Specification for Flat-Rolled Carbon Steel Sheets	(P)
ASTM A479	Specification of Stainless and Heat-Resisting Steel Bars and Shapes	(P)
ASTM A499	Specification for Hot-Rolled Carbon Steel Bars and Shapes	(P)
ASTM A500	Specification for Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes	(P)
ASTM D1056	Specification for Sponge and Expanded Cellular Rubber Products	(P)
ASTM D3467	Method of Test for Carbon Tetrachloride Activity of Activated Carbon	(P)
ASTM D3466	Method of Test for Ignition Temperature of Activated Carbon	(P)

Sheet Metal and Air Conditioning Contractors National Association (SMACNA)

SMACNA	Round Industrial Duct Construction Standards	(P)
SMACNA	Rectangular Industrial Duct Construction Standards	(D)
SMACNA	High Velocity Duct Construction Standards	(P)
SMACNA	Manual for the Adjustment and Balancing of Air Distribution Systems	(P)

American Conference of Governmental Industrial Hygienists (ACGIH)

ACGIH	Industrial Ventilation	(P)

Industrial Perforators Association (IPA)

IPA	Perforating Industry Standards and Practices Manual	(P)

U. S. Department of Defense (DOD)

MIL-F-51068	Filter, Particulate, High Efficiency, Fire-Resistant	(P)
MIL-F-51079	Filter Medium, Fire Resistant, High Efficiency	(P)
MIL-STD-282	Filter Units, Protective Clothing Gas Mask Components, and Related Products, Performance Test Methods	(P)

REPORT ON ANSI/ASME NUCLEAR AIR AND GAS TREATMENT STANDARDS FOR NUCLEAR POWER PLANTS

James F. Fish
Nuclear Environmental Systems
American Air Filter Co., Inc.
Louisville, Ky.

Abstract

Original "N" Committee, N45-8, has completed and published through the approved American National Standards Institute process two Standards, N-509 and N-510. This committee has been dissolved and replaced by ASME Committee on Nuclear Air and Gas Treatment with expanded scope to cover not only air cleaning, but thermal treatment equipment. Current efforts are directed to produce "Code" documents rather than "Standards" type publications. This report summarizes changed scope, current organization and sub-committee coverage areas.

I. Introduction

Proceedings of the Twelfth AEC Air Cleaning Conference at Oak Ridge, Tennessee August 28-31, 1972 contain an initial report by Leckie and Thompson on ANSI N45-8 Nuclear Gas Systems Treatment Standards.

Over the intervening six years, much has been accomplished. The organizational foremat has been changed and the scope and direction of the committee's activity has been greatly expanded.

II. Voluntary Standards Procedures in U.S., particularly for Nuclear Equipment.

Under the procedure controlling voluntary standards in the U.S., the American National Standards Institute recognizes three methods leading to a National Standard:

1) The ANSI Committee method; in the nuclear area, the N Committee.

2) The accredited organization method.

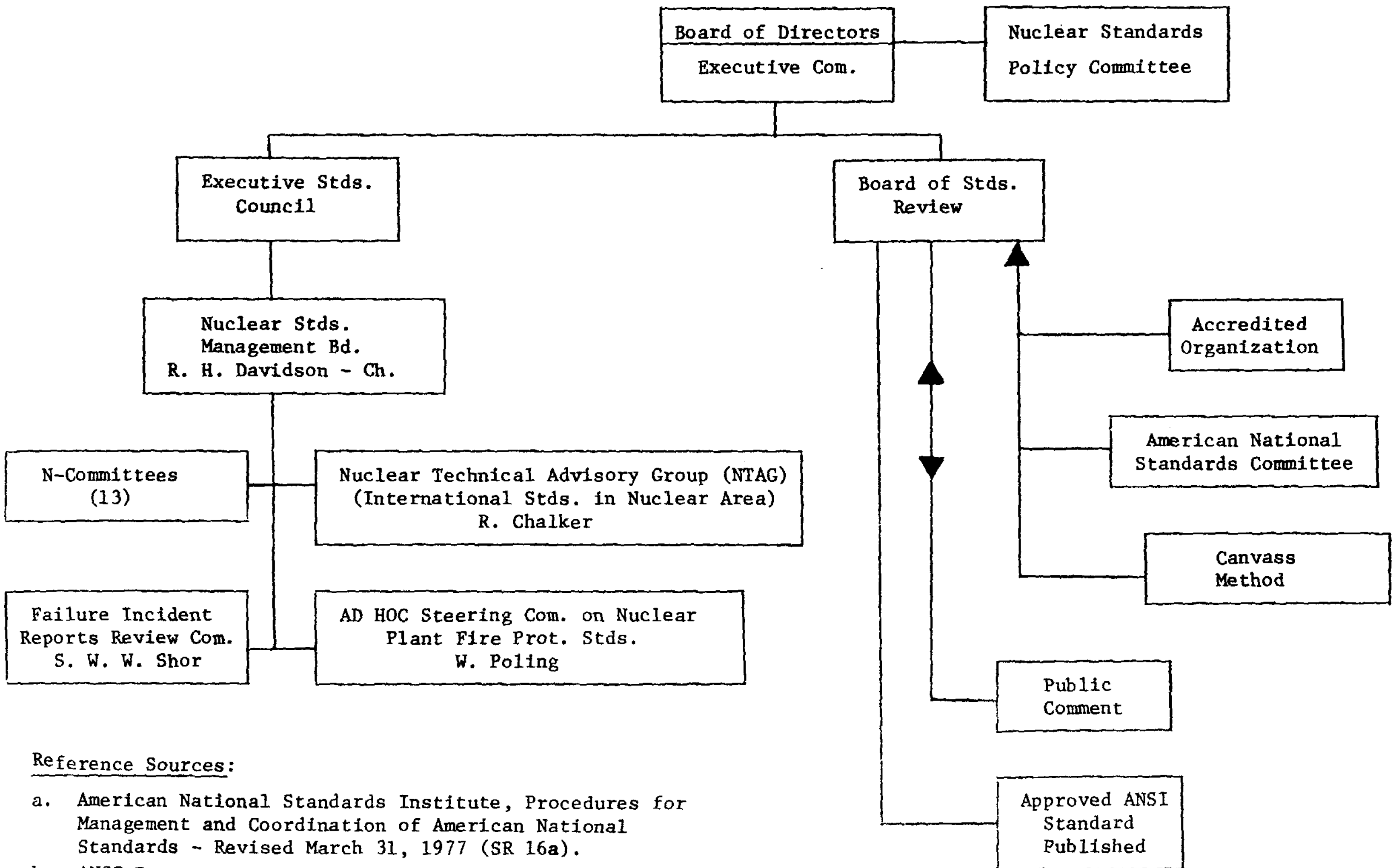
3) The canvass method.

The ANSI organizational structure applicable to Nuclear activity is shown in Table 1.

In 1971, this activity was organized under the "N" Committee method and assigned to American Society of Mechanical Engineers as the secretariat organization. Other "N" Committees covering other areas are assigned to other secretariates for standards action. Overall responsibility for such assignments was and is under American National Standards Institute.

AMERICAN NATIONAL STANDARDS INSTITUTE

NUCLEAR STANDARDS ACTIVITY ONLY



Reference Sources:

a. American National Standards Institute, Procedures for Management and Coordination of American National Standards - Revised March 31, 1977 (SR 16a).

b. ANSI Progress Report 1977 (PR25)

c. Personal Communications

TABLE 1

For some time, national voluntary standards activity has been receiving a considerable amount of political attention. The basic question is "Should the Federal Government take over such activity?" under the bannoer of all concerned? ANSI activity has been carefully controlled and is documented to insure that all interested parties have full input access opportunity to any standard before any document is issued. To this end, the approved ANSI procedures are available in printed form with each required step documented in great detail.

When a secretariat professional organization has demonstrated structure in full accordance with these procedures, it becomes an "accredited organization". This means that the organization's procedures are also documented in detail, committees are balanced representing all interests, and ballot procedures are established to insure that all interests have full right of input, question and objection to any proposed document. When any draft standard is forwarded to ANSI's Board of Standards Review, it is also held open for comment by the general public for a rather extended period. All comments received require detailed consideration by the originating Committee and written response back to the originator of the comment or question.

III. Nuclear Air Cleaning Standards Activity to Present

N45-8

In June 1973, AEC first published Regulatory Guide 1.52 covering requirements for Engineered Safety Feature Air Cleaning Systems. ANSI Standards were not recognized as they did not, at that time, exist outside of Committee.

In 1975, the N45-8 Committee document ANSI N510 covering "Testing of Nuclear Air Cleaning Systems" was published following in 1976 by ANSI N509, "Nuclear Power Plant Air Cleaning Units and Components."

Therefore, in July 1976 and March 1978 when Revisions 1 and 2 to 1.52 were issued, N509 and N510 were extensively referenced, a practice continued with Regulatory Guide 1.140 for normal ventilation system filters first issued in March 1978. When satisfactory standards are available, it is the practice of NRC to reference them to the extent practicable.

Meanwhile, the American Society of Mechanical Engineers, Nuclear Codes and Standards Supervisory Committee had achieved ANSI "accredited organization" status. The N45-8 organization was changed from "N" Committee status to "Accredited Organization" Status under this Committee after accreditation was achieved, see Table 2. The N45-8 Committee was then dissolved.

CONAGT

The new name is Committee on Nuclear Air and Gas Treatment, CONAGT, with approved scope as follows:

"1. To Develop, review, maintain, and coordinate Codes and Standards for design, fabrication, installation, testing, and inspection of equipment for gas treatment for Nuclear Power Plants. As used herein "gas treatment" includes both HVAC and gas processing.

AMERICAN SOCIETY OF MECHANICAL ENGINEERS

NUCLEAR CODES AND STANDARDS ACCREDITED ORGANIZATION

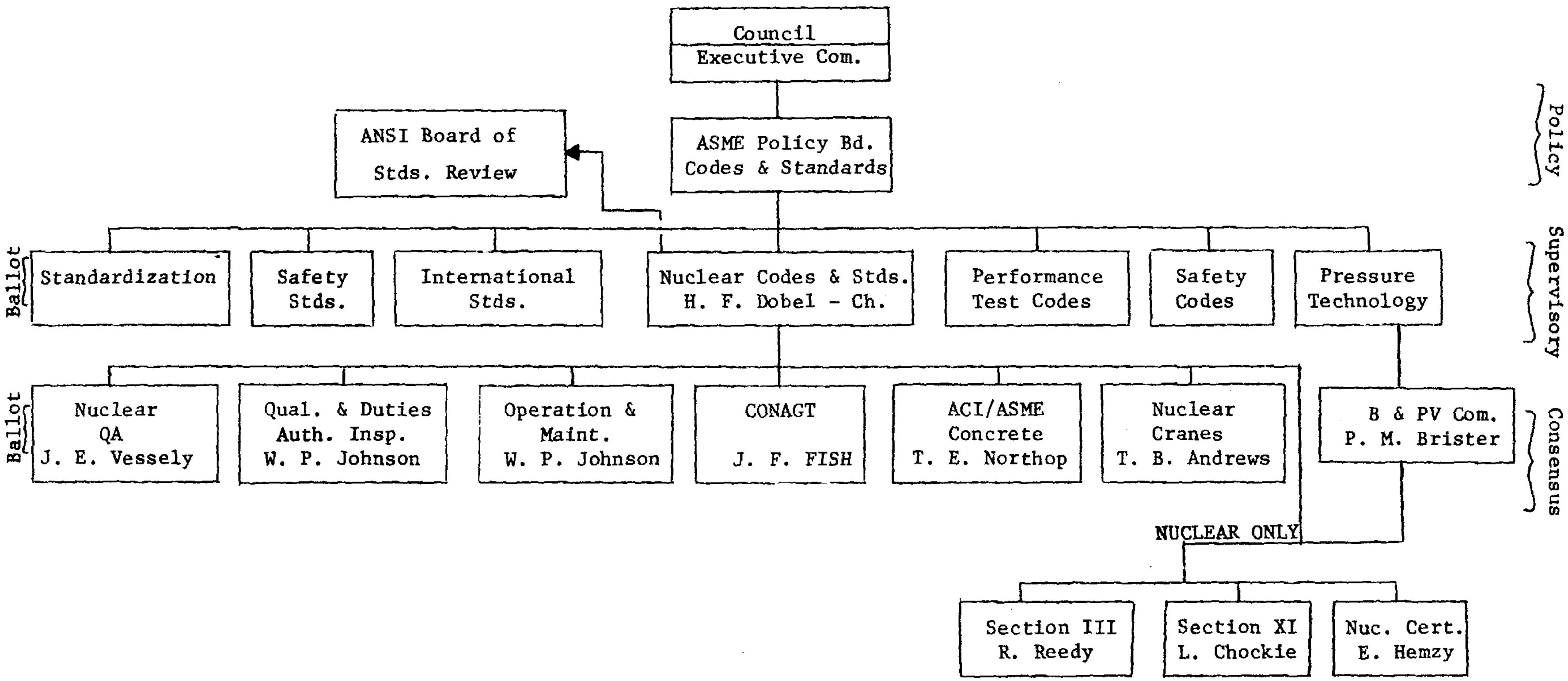


TABLE 2

JUNE 1978

(a) HVAC - the moving and conditioning of air which is supplied exhausted, or recirculated into and from an enclosed space to maintain prescribe ambient conditions. These conditions include pressure, temperature, humidity, and contaminants.

(b) Gas Processing - the transportation of gas and the separation, isolation, and disposal of its constituents by physical, mechanical, chemical, delay, electrical and thermodynamic means.

2. Codes and Standards developed by this Committee will be supervised by the ASME Nuclear Codes and Standards Committee.
3. This Committee will establish recommendations for ASME representation on Committees of organizations other than ASME developing interfacing Codes or Standards.
4. This Committee will develop recommendations for ASME positions on interfacing or referenced Standards.
5. This Committee will develop Codes and Standards in accordance with the Committee Procedures for Nuclear Projects approved by the American National Standards Institute under the Accredited Organization Method."

The original scope, limited to air cleaning, has now been greatly expanded.

Codes vs. Standards

At approximately the time this was taking place, pressure was being exerted to change over to a Code foremat rather than a Standard foremat. The Code concept goes back basically to the ASME Boiler and Pressure Vessel activity, firmly established by many years of operating experience. The reason for this change is that Codes are legally enforceable documents which can be adopted by political governing bodies to have the force of law. They specify minimum enforceable requirements. Standards on the other hand tend to be somewhat looser and do not lend themselves, in most cases, to legal status.

Organization

With the Committee scope broadened, it became obvious that a more complex organization, including additional breadth of knowledge, was required. Table 3 shows how the CONAGT Committee is currently organized within the ASME structure.

The Executive Committee under Dr. Wittke brings policy and or other considerations to the attention of the main committee for appropriate consideration. Because this CONAGT activity interfaces with the work of so many other Committees and organizations, the Coordinating Committee under Dr. Burchsted has its hands full attempting to stay abreast of developments and arranging CONAGT comment and input pertinent to other documents.

The breakdown of the Equipment Subcommittee under Mr. Miller is particularly interesting as it indicates the diversity of equipment now included under the Committee's surveillance. Subgroups cover: Fans, Dampers and Valves, Air Cleaning Equipment, Structures, Refrigeration Equipment, Conditioning Equipment and Gas Processing Equipment. The application of such items to nuclear plants involves many considerations not encountered in conventional industrial installations.

AMERICAN SOCIETY OF MECHANICAL ENGINEERS

NUCLEAR CODES AND STANDARDS ACCREDITED ORGANIZATION

COMMITTEE ON NUCLEAR AIR AND GAS TREATMENT

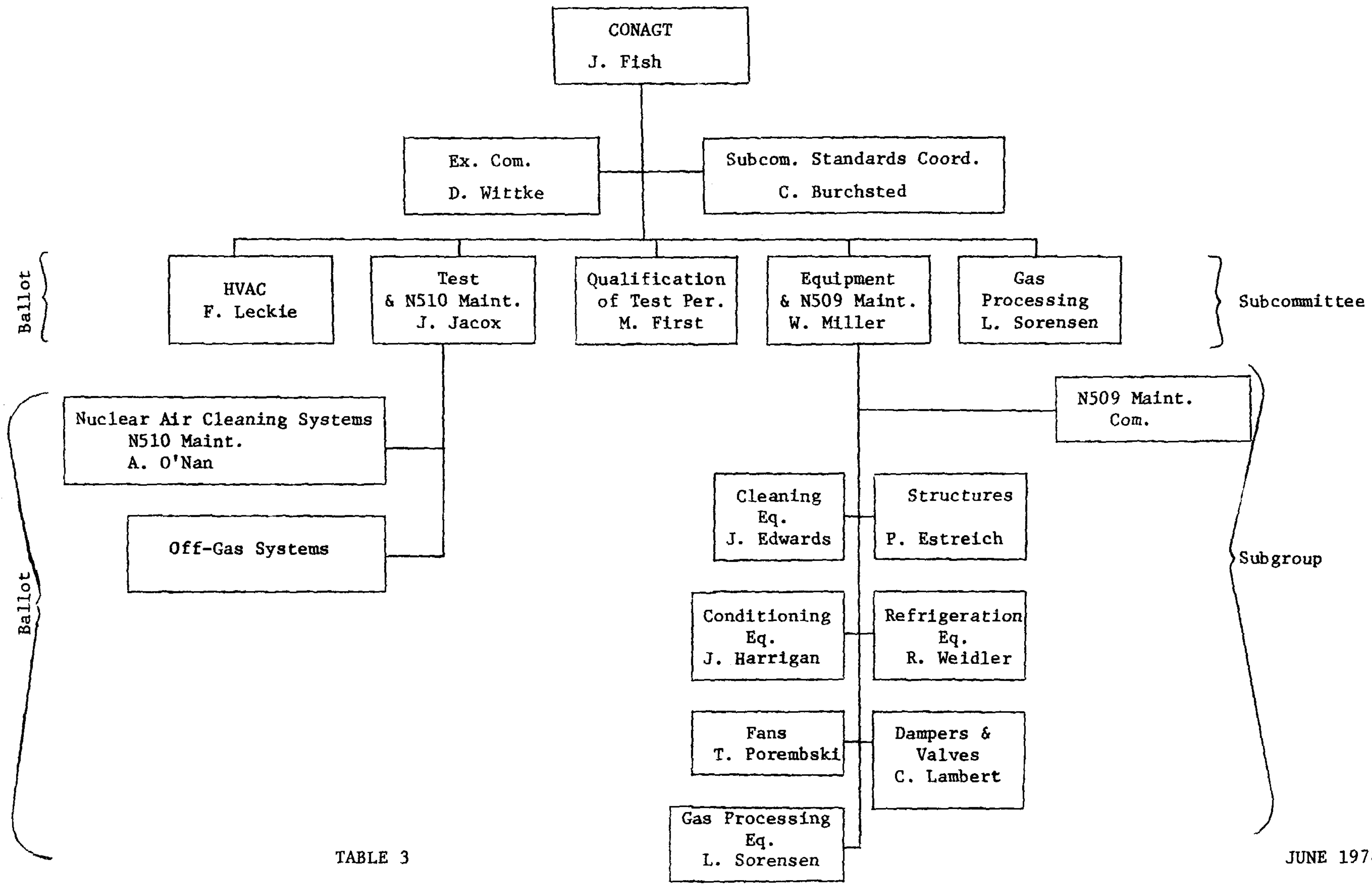


TABLE 3

JUNE 1978

The Testing Subcommittee's responsibilities under Mr. Jacox are currently directed towards air cleaning systems as initially covered in N510, but they are in the process of expanding their work to include off-gas system tests and will extend coverage to critical areas of other equipment as required.

The Qualification of Test Personnel Subcommittee under Dr. First is a most critical area and will eventually set forth requirements for both Lab and Field Test Personnel. Such tests are most important to protection of our overall environment. Therefore, it is considered prudent to carefully delineate the minimum required ability of persons engaged in such test activity.

Revisions to N509 and N510

As few words of man are so perfectly set forth as to require no revision, experience with both N509 and N510 has indicated that some changes are desirable. The problem, however, was that the N45-8 committee that produced these standards was no longer functioning. Also, of course, both documents will eventually be incorporated into Code documents with updating and inquiry procedures (code cases) being formalized. This will take some extended period of time.

Consequently, the Chairman of the Subcommittee on Equipment formed an ad hoc committee of his Sub Group Chairman to revise N509. The Chairman of the Subcommittee on Testing likewise formed an N510 ad hoc maintenance committee. These Committees will handle revisions to existing Standards pending issuance of Codes.

Both of these groups are in the process of completing their final drafts of Revision 1 to be balloted this year and if approved will go to Nuclear Codes and Standards and thence to ANSI Board of Standards Review, with formal issue estimated in early 1979.

Overall efforts are first directed at critical Engineering Safety Feature equipment and then to other non-ESF but critical items of hardware falling under the CONAGT scope.

When adequate standards exist or when other organizations are producing specific item requirements, we are very happy to reference them and not attempt to reinvent the wheel!! ASTM, for example, particularly the D28 Committee under Dr. Burchsted and specifically the D28.04 group under Mr. Rivers, have developed key input reference documents.

Summary

Codes and Standards are sometimes viewed as a painful reality. On the other hand, technical personnel use such references every day of their lives. Try to think for a moment what life would be like if there were not standards for all sorts of things we live with every day.

Now, almost eight years since the first get together at Engineering Society Headquarters in New York City, it requires effort to think of Nuclear Air Cleaning Equipment before Standards were drawn up, documented and enforced. Where would we be without codes and standards? With that thought, I leave you all to consider alternatives!!