SESSION 1

OPENING OF THE CONFERENCE

MONDAY: August 22, 1988
CHAIRMAN: M.W. First

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Melvin W. First

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WELCOME AND OBJECTIVES OF THE CONFERENCE

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It is a distinct pleasure for me to call the 20th DOE/NRC Nuclear Air Cleaning Conference to order and to introduce the first of sixteen sessions that will ultimately be included in the formal printed proceedings of this conference.

Twenty is a good round number on which to base a celebration and that is the reason for wanting to bring this particular air cleaning conference back to Boston where the first conference was held in 1950.

The year 1988 has its own special significance for nuclear air cleaning. It was in 1948, exactly 40 years ago, that the U.S. Atomic Energy Commission first realized that air and gas cleaning technology was becoming a matter of critical importance for the future health of nuclear development and awarded a two year research contract to Philip Drinker and Leslie Silverman, both professors at the Harvard School of Public Health, to pursue air cleaning research directed toward emission problems that were of major concern at that time. Two such matters were the release of unenriched uranium compounds from ore processing and chemical purification steps, and the release of beryllium dust from similar operations as well as from later beryllium fabrication steps. Therefore, the research agenda during this initial period was scarcely distinguishable from non-nuclear dust collection research except, of course, for the much higher efficiency standards that were demanded for the nuclear applications. The research effort during those first two years was equivalent to the full time work of about 1½ professionals. My own involvement with this program began just a few months before the expiration of the two year contract and consisted of preparing the final report of research findings. I was considered a suitable recruit for the task as I had just finished writing a doctoral research thesis conducted under the supervision of both Dr. Drinker and Dr. Silverman.

So rapidly were critical air and gas cleaning problems becoming recognized by the nuclear energy community and so dismal was the science of air and gas cleaning technology at that time that the Atomic Energy Commission did three things of importance in 1950 to improve the situation. First, it gave the Harvard School of Public Health substantial funding to undertake a serious research effort directed toward helping them solve numerous incipient disasters that involved releases of radioactive materials inside work places and to the environment. This was the origin of the Harvard Air Cleaning Laboratory. Second, the Atomic Energy Commission asked the Harvard Air Cleaning Laboratory to hold a conference on nuclear air and gas cleaning technology for AEC personnel and for AEC contractors to share whatever knowledge was then available. This was the first nuclear air cleaning conference that I mentioned earlier. The third thing AEC did was to request the Harvard Air Cleaning Laboratory to prepare a handbook on air cleaning that would gather together everything that might be of value for nuclear air cleaning applications. This handbook was published by the AEC in 1952 and was the principle reference source until the Nuclear Air Cleaning Handbook, authored by C.A. Burchsted and others at Oak Ridge National Laboratories, was published 17 years later. To give you some small idea of the state of knowledge about air and gas cleaning technology in 1950, I will simply tell you that the Handbook on Air Cleaning contained 89 pages. Of this number, less than one page was devoted to HEPA filters, then referred to as absolute filters. In 1969, the first edition of the Nuclear Air Cleaning Handbook contained 202 pages. The second edition, a few years later, contained 290 pages and the third edition, currently in preparation, will no doubt contain considerably more pages when it is published. Nonetheless, the 15 year period from 1950 to 1965 was a time of enormous productivity for the advancement of nuclear air and gas cleaning technology. Time does not permit me to dwell longer on the achievements during this period accept to say that the record is written large in the Proceedings of these air cleaning conferences.

The following 15 or so years saw a steady progression of technical improvement in nuclear air cleaning equipment but the introduction of little really new technology. This was a period of stabilization and consolidation for nuclear air cleaning technology. Currently, we seem to be well into a new period of exuberant creativity and innovation. New products are being introduced rapidly that are not only better but are different in kind. One example, the old familiar HEPA filter is being rapidly transformed, after 50 years of faithful service, into an ultra-low penetration air filter or ULPA filter and is in the process of losing all of its corrugated separators. These changes now call for new test protocols, different acceptance criteria, and modified nuclear codes and standards. During the course of this conference we will learn of other new and innovative developments in air cleaning for chemical processing, air cleaning plans for containment venting, and air cleaning needs for waste handling and plant decommissioning, among other topics. We will be offered a rich technical program over the next 3½ days by some 80 speakers. Therefore, it behooves me to cease my reminiscences and get on with the business at hand.

First, I wish to welcome each of you to my beloved Boston and to extend a special welcome to all who have come from other nations. I note that we have representatives from Belgium, Canada, Federal Republic of Germany, Finland, France, Holland, Hungary, Israel, Italy, Japan, People's Republic of China, Republic of China, Scotland, South Korea, Spain, Sweden, and the United Kingdom. We hope your visit to Boston and the United States will be not only profitable but extremely pleasurable.

We have every expectation that the Nuclear Regulatory Commission and the United States Department of Energy intend to continue their joint sponsorship of the Nuclear Air Cleaning Conferences. They have requested the industrial beneficiaries of these conferences to help them demonstrate to those who control their agency purse strings that the conferences are useful, needed, and productive for them, as well as for the agencies involved, by becoming identified as co-sponsors and by providing partial financial support. I am very pleased to report that a number of industry leaders have responded very positively to this suggestion and have identified themselves as an ad-hoc committee to explore the formation of an organization that will be interested in, and capable of, exercising a continuing cooperative effort to support future nuclear air conferences. This will be the subject of the Government-Industry session scheduled for 4:00 PM on Wednesday of this week. This is an open meeting and I urge everyone who has an interest in the continuation of the Nuclear Air Cleaning Conferences to attend. Presentations will be made by J. Louis Kovach from NUCON, Tom Allan, Flanders Filters, Fred Leckie, NCS Corporation, George Ello, Manville Corporation, and Ray Weidler, Duke Power Company. We are anxious to have for additional comment and input from all who attend. For this reason, the session will be the only event during that period. As it will be the last session of the day, their will be no barrier to continuing it beyond the assigned time slot, making it possible for everyone who wishes to have an input to the discussion to have that opportunity. I urge you to attend to become thoroughly informed about this issue and to spread the word to those not in attendance at this conference to help us to correct misinformation and misunderstanding regarding the current intentions of our long time government sponsors. It is important that we have everybody well informed. Since the idea of soliciting nuclear industry co-sponsorship first surfaced well over a year ago, the political future for nuclear energy has become much more obscure. We are meeting in a state governed by a presidential candidate who from his record gives little comfort to those interested in nuclear power and concerned about nuclear armament. His opponent has promised no tax increases, no reduction in military preparedness, and a reduction of our huge deficit. So I think we are pursuing the correct policy at the right time with respect to industry involvement in future air cleaning conferences.

There is no way this excellent program could have been assembled without the full participation of a program committee and there is no way this program could have been presented this week without their generous and willing assistance. Their names are listed in the program and I would like to express our collective appreciation for their fine efforts. We thank you very much.

INTRODUCTORY COMMENTS OF CHAIRMAN FIRST

Our first keynote address will be by Mr. William Russell, who is Regional Administrator, United States Nuclear Regulatory Commission. His topic is "People, Organization, and Nuclear Safety." Mr. Russell is Regional Administrator of Region I, a position he has had for approximately a year and a half. He also has had eight years of nuclear submarine experience and was Director of the Division of Human Factors, Safety Office of Nuclear Reactor Regulations. He is currently responsible for inspection, licencing, and enforcement for 31 nuclear power reactors, 18 research reactors, 5 nuclear fuel facilities, and over 3,000 medical, industrial, and research users of nuclear material. Region I covers all 11 Northeastern States and Washington D.C. and he is responsible for the agency's response to incidents and accidents in this whole area.

PEOPLE, ORGANIZATIONS AND NUCLEAR SAFETY

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Good morning. I as pleased to be here to address the 20th DOE/NRC Air Cleaning Conference. I see from your agenda that there are many technical topics addressing important issues related to routine and post-accident sampling and removal of radioactive material from air at nuclear power plants and other facilities. I believe, the research, system development and operating experience that you will be discussing over the next few days, can make an important contribution to public health and safety and can be an important factor in public acceptance of nuclear power in the United States. Improvements in containment air sampling and radioactivity measurement could significantly improve reactor coolant system leak detection capability and improvements in air cleaning systems and technology have the potential to significantly reduce both routine and postulated accident releases of radioactive materials. However, technology improvements whether they be in air cleaning systems or other reactor safety systems are not sufficient to gain public confidence in nuclear safety. The people side of the safety equation must be addressed and human performance must improve.

I believe that PEOPLE and ORGANIZATIONS play a key role in assuring NUCLEAR SAFETY. Let me give a few examples of how human error can adversely effect the reliability and availability of air sampling and cleaning systems. Errors in valve line ups have resulted in sampling the wrong air volume, sampling systems have been isolated, instruments improperly calibrated and automatic isolation features defeated. Failure to follow procedures in operation of sampling systems have frequently resulted in obtaining non-representative samples. Improper control of work has resulted in paint fumes and other contaminants reducing filter effectiveness of installed systems without corrective action to replace or test filter media. I have personally observed the face of roughing and HEPA filters that have been painted such that they have local blockage and high local flow rates. Other examples include improper isolation of spray additives such that post accident sprays would not be as effective at cleaning containment air, a utility performed maintenance and replaced charcoal filters on the wrong train of a standby gas treatment system, and improper manual activation of charcoal filter water deluge systems. I would encourage designers and researchers to consider the impacts of human performance on system performance and incorporate human factors in the design to minimize the potential for human error disabling the systems. Later this week results of NRC studies on control room habitability systems will be presented which further emphasize the importance of human factors in the operability of air cleaning systems.

I would like to expand the scope of my comments and discuss NUCLEAR SAFETY in general with a focus on PEOPLE and ORGANIZATIONS, but let me first set the stage by describing commercial nuclear power in the United States.

Currently there are 109 licensed operating reactors, located at 75 sites, comprising 81 different designs and operated by 54 different utilities. Individual plant designs have changed with age as a result of both utility modifications and NRC requirements. This combination of different designs and different operating organizations is such that it is very difficult to transfer operating experience between utilities or within a utility between plants. Each plant has its own learning curve and receives little benefit from the experience of other plants. Clearly, different designs, procedures, training and operating organizations makes effective exchange of information on human performance more difficult. Since all licensed plants must meet NRC's general design criteria and other aspects of the regulations related to design, and assuming that there are no breakthroughs in design safety of existing plants, I submit that significant improvement in the safety of current plants requires improvements in existing hardware reliability and human performance. Clearly, hardware reliability improvement is a maintenance and human performance issue. As evidence of this statement, I offer my earlier examples of human error impacting air cleaning systems operability and reliability. Expanding beyond air cleaning systems, plant owners must do a better job of maintaining all plant systems and diagnosing equipment problems before equipment failure occurs. I am impressed by the example the Japanese have set in improving nuclear power plant reliability through an aggressive and comprehensive maintenance and periodic inspection program and I am encouraged by recent United States industry efforts, through NUMARC* and INPO**, to provide maintenance guidelines and examples of good maintenance practices and programs. I am also encouraged by the commitment of United States utilities to perform self-assessments of their maintenance programs. I believe, however, that more needs to be done to accelerate the pace and comprehensiveness of maintenance improvements in United States nuclear power plants. Clearly, more needs to be done in the area of understanding the root causes of equipment failures and implementing effective corrective action and not simply repairing repetitive failures. This requires effective operational engineering support of both maintenance and surveillance testing activities.

Past human performance problems in nuclear safety have been well documented. The accidents at Three Mile Island and Chernobyl have significant root causes in human factors and in plant management. The failure of plant personnel to recognize the safety significance of their actions, procedures which were knowingly violated, a lack of awareness of plant conditions and status, and operators being misled by incorrect data and information were root causes of these accidents. The "operator culture" which led to the Peach Bottom shutdown of a year ago, equipment reliability and management deficiencies at the Pilgrim Station and the management problems at TVA also have their "root causes" in human performance failures.

What can utilities do to improve human safety performance, what can the NRC do to improve safety performance, and what can you do to improve human safety performance? In the first case, utilities must promote the concept of a

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* Nuclear Management and Resources Council, 1776 "I" Street, N.W., Washington, D.C. 20006

** Institute of Nuclear Power Operations, 1100 Circle 75 Parkway, Atlanta, Georgia 30339 "safety culture."*** Safety culture starts with personal dedication and accountability beginning at the top with senior corporate management. Senior management that fosters an attitude and safety consciousness in all personnel with responsibility for supervision, operation and maintenance of the nuclear power plant. It is formed by policies and administrative controls, which when implemented, ensure that correct practices are followed. Attributes of a good safety culture include clear lines of responsibility and accountability, sound procedures for which adherence is demanded, management which critically self-assessed their activities and training programs which emphasize the reasons behind the practices. With a positive safety culture, personnel understand the safety importance of their actions, and safety information is freely communicated, including an admission of errors such that others can benefit and future errors are avoided. The operators at a facility with a positive safety culture continually monitor plant status to confirm the availability of safety systems and are thereby prepared to act when systems depart from normal operation. I believe this safety culture concept is equivalent to seeking excellence in nuclear plant operation.

The facility's license, technical specifications and operating procedures define limits on process parameters (power, temperature pressure, etc.), places requirements on availability and operability of equipment and imposes other conditions which must be met during plant operation. These initial conditions are assumed in the safety analysis of the plant's response to various operational transients and design basis events. These initial conditions form what I call the "safe operating envelope" for a particular plant. Clearly, the objective during plant operation is to keep the plant within this "safe operating envelope" and to quickly identify those instances when it is not.

This is a significant human factor problem. Control rooms have hundreds of gauges, instruments, alarms and controls. Many more local gauge boards, alarms, valves and components are operated or monitored outside the control room. Reliable and prioritized alarms and other data for early fault identification must be provided. Utility efforts with respect to control room design reviews and implementation of safety parameter display systems have significantly improved human factors in some plants. However, not all utilities are aggressively pursuing these improvements. Further, these changes do not address important aspects of system status and operability. Plant management must develop effective administrative controls to ensure that procedures are strictly followed and that deviations require prior approval. Rigorous use of checklists, formalized watch turnover procedures and other measures to control the status of safety equipment are included in these administrative controls. An initiative that some licensees in Region I have undertaken in this area involves self-assessment of the readiness for restart from maintenance outages where systems and equipment have been manipulated and removed from service and a high potential exists for being improperly restored and therefore not available during operation.

*** International Atomic Energy Agency Report No. 75-INSAG-3, March 1988, "Basic Safety Principles for Nuclear Power Plants"

Since the Three Mile Island accident, NRC has emphasized the importance of symptom based emergency operating procedures. Our approach was to approve reactor vendor generic procedures and the plant specific process by which owners developed their own emergency procedures. In order to expedite implementations, NRC further made emergency procedure upgrade a post implementation review. Through NRC inspections over the last two years, NRC has identified a number of concerns with both technical and human factor aspects of emergency operating procedures. Recently, I required a New York utility to keep their plant shut down in part due to poor operator performance on emergency operating procedures. I believe that NRC's message is clear. Human factors and human performance must be considered in procedure development such that operators in time of stress are able to understand and effectively implement emergency operating procedures.

The final area I would like to mention for utility improvement is effective exchange of operating experience within a plant and between plants with emphasis on understanding and correcting the root cause of human error. I believe that utility participation and commitment to the industry's human performance evaluation system**** will help all utilities learn from and prevent human errors.

NRC has undertaken a wide variety of actions to improve human safety performance. The agency's systematic assessment of licensee performance is a periodic evaluation of each licensee's performance in specific functional areas in order to apply NRC inspection resources where needed. The SALP evaluation focuses upon human performance in operation, radiological controls, maintenance, surveillance and inservice testing, emergency preparedness, security and safeguards, assurance of quality, licensing, engineering support and training.

Perhaps the most important SALP functional area assessed is the one termed Assurance of Quality in which the NRC evaluates management performance. This area cuts across all other functional areas. It evaluates management's attitude and philosophy toward quality and their ability to self-assess their own performance and improve it. It does not address design issues but rather address the subject of workmanship, personal attitudes and management philosophy. It focuses on those management actions which ensure high quality of equipment and human performance and which send a clear message to all employees that the primary responsibility for quality rests with those who perform, not those who inspect, check or audit. NRC is shifting its inspection emphasis on quality towards performance based inspections and quality verification rather than paperwork reviews of quality assurance programs and documents.

What can you as experts in air cleaning do to improve reactor safety and human performance? First, I would encourage you to communicate your operating experience, research results and system design information as broadly as possible. Expand those involved to include more operators such that you can

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**** HPES Program is managed for the industry by INPO.

In closing, the future of nuclear industry in the United States clearly hinges upon improving safety performance and operational reliability of existing plants. Licensees must improve human performance by promoting a "safety culture" and seeking excellence. The NRC must ensure that plants are designed and operated in accordance with rigorous safety standards and when necessary that such standards are vigorously enforced. Designers, vendors, and the research community need to consider the system operator as part of the design process and should develop designs which minimize the potential for human error. Overall industry performance as measured by a variety of performance indicators is improving. There are fewer scrams of reactors, fewer safety system actuations, fewer significant events, lower personnel exposures, and reduced quantities of waste generation. The trends are in the right direction such that continued safe and reliable performance, with an absence of significant events and dedication to quality will achieve public confidence in nuclear safety.

DISCUSSION

FIRST: I was delighted to hear the many words of wisdom by Mr. Russell. His comment that we should take corrective measures not just report repetitive failures struck me as particularly appropriate because this has been the same message that Dade Moeller has been bringing to us at a number of Air Cleaning Conferences as a result of his analysis of licensee event reports (LERs).

I would like to ask you a question, Mr. Russell, which gets directly at this matter of human quality. The demand for higher and higher technically qualified people comes to the fore at our operating plants yet the plants operate for long periods of time, months and sometimes years, with nothing significant happening. When we get these more and more highly qualified people, how do we keep them from becoming so bored with a job that has no challenge for them that they remain sufficiently alert to be able to handle emergencies if they should arise?

RUSSELL: I think there are two aspects to insuring that operators are, in fact, maintaining their technical competence in their expertise and addressing the issue of boredom on watch. The first relates to the use of full scope simulators with a mandatory training program. In industry it is called "the continuous training program." In regulatory language it is called "the requalification program." We require demonstrated performance on emergencies and routine operations such as start ups and shut downs to keep the operators technically competent for operating the systems. In the areas of administrative controls (which I think is the key factor to address boredom), such activities as routine tours of the plant, checking operating equipment and machinery, and insuring that there is sufficient activity, even though that activity is repetitious, are important. The automatic data loggers, and the other information that goes into computer recording and logging systems, provide a situation where an individual feels he is no longer responsible for physically monitoring the equipment itself. So we are currently looking at those activities through administrative controls of watch turnover and relief, tours of the plant, checking operating equipment, and, in fact, encouraging utilities to use routine logs, where an operator is required to go out into the spaces and physically check the equipment.

MURTHY: Do you see any relationship between the ongoing debate on the greenhouse effect and the revival of the nuclear power industry?

RUSSELL: The agency has no position on the greenhouse effect. The agency's emphasis is to make sure that existing nuclear plants are operated safely so that public perception of nuclear safety becomes positive. A plant that operates well and reliably and produces power is a winning situation both for the utility that operates it and the regulator. I believe that the plant that operates well does so because of management attention to detail and the conduct of the operation. We generally find that such facilities not only perform well from the standpoint of megawatts produced but they also perform well on a safety basis. I think that is what it is going to take to make nuclear power revive. That is the reason for the emphasis by the NRC on poor performing plants and why we are diverting resources to those that are not performing as well as others.

KUGLER: I have been in the nuclear industry for about 20 years, but I only recently discovered that the Department of Defense has been implementing a system called the Integrated Logistics System for their weapons systems for the last two decades. That program addresses many of the concerns that you addressed relative to maintenance programs from design through root cause, through liability, through human factors, through mean time between failures, time to repair, hardware resources, tooling resources, people resources. Is there any consideration by the NRC to implement this proven system that has been used for two decades?

RUSSELL: What you are referring to is integrated logistics support, which includes original design concepts, through maintainability, and the entire life cycle cost for the system. That is a rather comprehensive approach. Currently, the approach is being looked at within the NRC. We are looking at a policy statement right now on maintenance activities. When you think about the problem of 109 facilities with 80-some different designs, you essentially have to end up developing an integrated logistic support system for each design. We are encouraging industry efforts through INPO activities and NUMARK activities to review their maintenance programs all the way from materials being stored in the warehouse through actual procedures for conducting maintenance, i.e., a technical repair standard. That activity is going ahead on a case by case basis and I hope that we will get to the point where the NRC starts to use the actual online availability of systems as an indicator and we start to measure system performance through an availability determination that includes not only mean time between failures but also mean time to repair. There is research work underway and we are looking into selecting a few key systems to start that type of an indicator approach.

KUGLER: I would like to make one comment, the indicator you use will lead the industry.

RUSSELL: We are aware of that.

<u>JOHNSON, J.:</u> How will the NRC deal with aging nuclear plants; especially with extending their operating life five to ten additional years?

<u>RUSSELL:</u> NRC has an extension aging research program with policy development underway on plant life extension. It is premature to describe or speculate on the outcome other than to state it is a high priority and the appropriate policy and rules are expected to be in place prior to expiration of Yankee Rowe's license in 1997.

<u>WEBER, L.D.:</u> A significant number of papers this week address containment venting technologies. Do you think containment vents have a role to play in fostering safety or public acceptance of nuclear power in the U.S.?

<u>RUSSELL:</u> NRC currently is reviewing venting as a possible aid in coping with severe accidents for boiling water reactors with Mark I containments. Venting has been approved for several boiling water reactors as part of their emergency procedures. I believe upgraded hardened vent paths, with appropriate controls, can reduce the risk of severe accidents.

<u>MEDDINGS:</u> What is your view on the use of automated process, logic-controlled plants as a means to reduce human operator errors?

<u>RUSSELL:</u> Rather than providing an individual response, I recommend you review the recent report on application of computer (automation) technology done by Principal Work Group No. 1, Committee for Safety of Nuclear Installations, OECD. This report is a survey of both regulator and industry views of participating members to OECD. This report was coordinated by W. Kennedy, US NRC, for the USA. I believe it was reported out of the working group at the September 1987 meeting in Paris.

<u>FIRST:</u> Thank you, Mr. Russell I think this has been a very seminal, thoughtful presentation.

INTRODUCTORY COMMENTS OF CHAIRMAN FIRST

Your program shows that the U.S. Nuclear Regulatory Commission Keynote address is to be presented by Mr. E.C. Baynard, III. He was not able to be present today. We have an excellent replacement, Mr. James Knight, Director of the Office of Safety Appraisals of the Department of Energy. Mr. Knight has been active in engineering and management related to nuclear technology for the past 25 years. He has participated in the commercial nuclear programs in the United States as both a licensee while at the National Bureau of Standards Reactor and, for almost two decades, as a regulator. From 1968 to 1986, Mr. Knight served with the regulatory staff of the Atomic Energy Commission and then the Nuclear Regulatory Commission. During this time, he directed many of the engineering and geo-science staffs engaged in the licensing of the majority of the nuclear power plants now operating. In 1986, Mr. Knight joined the geologic repository program at the Department of Energy as Director of siting, licensing, and quality assurance in the Office of Civilian Radioactive Waste Management. In February 1988, he was appointed Director of Nuclear Safety under the Assistant Secretary for Environment, Safety, and Health, the predecessor to his current position. As Director, Office of Safety Appraisals Mr. Knight is responsible for the conduct of the inspection and safety analysis review program for all DOE nuclear facilities.

BLUEPRINT FOR NUCLEAR SAFETY - A NONREGULATORY STRATEGY

James P. Knight
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U.S. Department of Energy

It is a pleasure to address this Conference as the keynote speaker for the Department of Energy. My presentation today centers on a key aspect of the continuing debate over the safety of nuclear plants in this country - and I pointedly include those of the Department of Energy - What can be done to improve the effectiveness of nuclear safety oversight? How can public confidence be bolstered, while avoiding burdensome and counterproductive regulatory prescriptions? Three Mile Island, WPPS, and now the Shoreham and Seabrook situations, have over time fueled the debate over needed institutional changes, including licensing reform, Price-Anderson, a single-Administrator NRC and the like. The issues facing the Department of Energy are similar in context, although different in origin: What can be done to reconstruct a credible and effective nuclear safety oversight program in a nonregulatory environment? This ha been and continues to be a daunting challenge, but one that provides an unprecedented opportunity. One of the designing an oversight program from the ground up, and in the process, availing ourselves of the institutional "lessons learned" of both the NRC and DOE. I would like to share what insights have come from, as well as to, this effort, and in the process, outline our "blueprint" for what is proving to be an effective nuclear safety program for the Department of Energy.

The Department of Energy operates a nuclear complex that now numbers over 250 facilities nationwide, many of which date back to the 1940s and 1950s. In 1985, Secretary Herrington moved to establish the Office of Environment, Safety and Health, give it needed resources and authorities, and begin extensive environmental protection and safety evaluations of all major DOE sites and facilities.

On the nuclear safety side this necessitates an integrated program that not only strengthens oversight but also builds DOE-wide technical capabilities and promotes safety performance. This has led up to focus our attention on three areas: 1) the DOE safety oversight system -- its resources, technical capabilities, and effectiveness; 2) the safety policy development and review; and 3) the Department's capabilities to foster technical inquisitiveness and overall excellence in safety performance. The essence of this approach is found in this last term -- performance. Performance that is results-oriented; founded on realized safety enhancements and risk reduction, not merely regulation for its own sake. Performance not merely in terms of hardware fixes, but also focusing on the human part of the safety equation.

For your benefit, I would like to address these one at a time.

I. Safety Oversight

The internal safety oversight system in DOE has been administered, historically, at three levels: by the line operating programs through contractor self-audits; by the Department's Operations Offices through their line management role; and by the Headquarters independent oversight function vested in the Office, of the Assistant Secretary for Environment, Safety and Health. Under this system, the contractor is held accountable to meet all DOE safety requirements as a condition of contract. This provision gives the Contracting Officer, i.e., the DOE Operations Office Manager, full responsibility to enforce safety requirements onsite, with the budgeting for necessary enhancements residing with DOE Headquarters Assistant Secretaries, who manage the DOE's major line programs, e.g., defense, nuclear energy, and energy research. Their role, in turn, provides independent assurance to the Secretary and the Under Secretary that an adequate ES&H program is being implemented.

The effectiveness of a nonregulatory system hinges on the checks and balances afforded it, both institutionally and through the support of management. The establishment of the Office of Environment, Safety and Health in 1985, the appointment of an external Advisory Committee on Nuclear Facility Safety this past February, and pending legislation which would establish a Defense Nuclear Facility Safety Board, are all designed to assure that an appropriate balance is struck. The National Academies of Science and Engineering went further in an October 1987 report to note that this balance cannot be successfully achieved unless specific measures were taken to increase safety resources, realign our headquarters safety organization and its oversight functions, and enhance DOE's technical safety credibility.

I might add that one of the largest challenges in making an oversight system of this kind of work is communication. Safety information, including any "bad news" that accompanies problems in any nuclear operation, must flow up to management where action can be taken. We have therefore added this to our "blueprint."

In response to these identified needs, DOE has taken a direct approach to establish safety functions that heretofore did not exist or were not being performed effectively. To strengthen oversight of field operations, the existing appraisal program has been strengthened by making the objectives of DOE appraisals less compliance or audit-oriented and more driven toward diagnostic evaluation of observed deficiencies. To increase awareness and response to facility safety issues, a program has been initiated similar to that of NRC to locate onsite residents at selected field operations. A performance evaluation program has been established to both collect and analyze data, and ultimately apply it to a program of annual systematic performance evaluations to gauge DOE-wide safety performance. These last two measures are expressly designed to facilitate the necessary communication and awareness noted previously.

From an institutional standpoint, however, perhaps the single most significant development has been with the decision making process for safety issues. Whereas, in the past, much of the decision making on facility operation, including restarts, unreviewed safety questions, operator qualification, and facility upgrades, had been the exclusive province of the line programs, our involvement has become not only expected, but required by the Under Secretary. Efforts are now being directed at policies and procedures which will institutionalize these gains.

A hard-learned lesson that we keep in mind in making these institutional changes is that they can be rendered useless unless management at all levels is safety conscious and demands a similar accountability from all workers. The instilling of a sound safety culture remains the toughest but perhaps the most important endeavor in our business.

II. Safety Assurance

The Department's approach to safety assurance is founded on policies and standards which clearly and consistently define and require acceptable safety performance and ensure a commitment to excellence beyond minimum requirements. Heretofore, DOE has employed the principle of comparability with NRC safety standards as a guiding principle in establishing departmental safety requirements. It is clear that comparability alone is neither an effective nor definitive criterion at the working level given the diversity of the DOE nuclear facilities. This past approach has tended to emphasize engineering and design, at the expense of human performance. What we believe is necessary are guiding principles that would strike an appropriate balance between both of these key parameters - design and human performance -- while assuring an independent self-sufficient basis for the Department's safety philosophy and standards.

Such a basis can be found in the Department's Nuclear Safety Objectives. The premise behind the Objectives is to provide a hierarchy of risk-based objectives, management principles, and performance criteria, which successively build upon each other and ultimately define DOE safety requirements and the corresponding benchmarks of the appraisal program. At its most fundamental, the risk-based objectives, like the NRC Safety Goals, define levels of safety to which DOE's safety analyses are to be compared and to which safety enhancements are to be directed. The qualitative objectives are further defined by corresponding quantitative objectives which provide risk-based "aiming points" for our safety enhancement programs.

Safety performance criteria provide the most definitive interpretation of how these safety objectives and principles can be achieved through the DOE safety assurance programs. Whereas, the safety objectives provide risk-based targets or goals for nuclear facility design and operations, the performance criteria provide mandatory elements of the Department's nuclear safety assurance program. These 40 some criteria are adapted from several sources, most notably the recently issued nuclear safety principles developed by the International Atomic Energy Agency. They range from a definition of management responsibilities for safety to human performance, citing, design, and accident management. Our aim is to use these criteria as a "living" set of objectives against which the adequacy of our Departmental safety orders and Appraisal criteria can be measured

"Defense-in-Depth" has long been attested to by the nuclear sector in both design and operations, but at least for DOE has not been defined effectively for implementation. As a key safety objective, we are proposing that DOE facilities employ defense-in-depth such that safety reliance is vested in multiple independent provisions or barriers, no one of which is relied upon excessively for overall facility safety. This objective offers not merely a design principle, but an institutional approach to safety assurance, as well. Accordingly, four levels of "defense" have been defined:

1) The first level of safety requires that a nuclear facility be soundly and conservatively built, tested, operated and maintained in accordance with stringent quality standards and engineering practices.

2) The second level provides measures to assure that foreseeable abnormal events or conditions will be detected and either arrested, or accommodated safely.

3) The third level supplements the first two through features that provide additional margins in the facility design to protect the public even in the event unlikely accidents do occur.

4) The fourth level recognizes that the first three levels of defense, notwithstanding, a residual risk remains for those severe accidents of very low probability that cannot be satisfied through design features. This entails consideration of emergency planning, and accident management procedures and training.

To interpret these principles for the workplace, we have gone further and have proposed quantitative guidelines for two dimensions of defense-in-depth that are to be given emphasis in safety assurance, prevention and mitigation. These guidelines address the probability of severe core damage and releases, mitigation of releases, and reviewing severe accident vulnerabilities. Overall, the application of this concept offers a means, particular for older facilities, to sustain needed safety margins where engineering design alone may not be sufficient.

Obviously, the implementation of the safety objectives, in general, and these defense-in-depth guidelines, specifically, will entail expanded use of probablistic risk assessments (PRAs) and other risk analysis techniques. All of the major DOE reactors are now in various stages of PRA development. We view a revised safety analysis using PRAs as the means by which the design basis of our older nuclear facilities can be verified against this defense-in-depth principle. PRA's will be used to identify dominant accident vulnerabilities, identify significant failure modes that warrant attention, and in many contexts where it is useful to have a realistic view of how and to what extent aspects of design and operations are important to risk. They will serve as a means to establish risk-based priorities and significance for operator training, emergency procedures, appraisal scope and frequency, maintenance and testing, and facility modifications. Recognizing the limited precision of PRAs, however, the emphasis will be on their use as an analytical tool, not a litmus test for safety acceptability.

III. Safety Excellence

Experience demonstrates that oversight alone and minimum compliance do not assure safety operation. Our strategy includes a goal to better develop and expand our in-house capabilities to address, evaluate, and achieve an in-depth understanding of the technical issues that confront us. It is clear that due to the age and unique design associated with our reactors and other nuclear facilities, this capability need to go beyond that in the commercial sector. Accordingly, our efforts have been directed at:

1) Increasing technical awareness among our contractors by a number of means including a recently signed INPO agreement, DOE contractor self-improvement programs oriented to the workplace safety enhancement, increased use of outside technical experts, training and qualification programs and peer technical review;

2) Promoting the use of state-of-the-art analytic techniques and programs not only for risk evaluations, but also operational applications. To this end, we have recently taken the initiative to establish a DOE risk assessment and applications program and will be working closely with PRA practitioners across DOE as a safety analyses are performed and applied.

3) Increased support to pertinent applied research programs that can help identify and resolve safety technology issues unique to the Department's operations. Such research, coupled with an inquisitive and technically competent staff, represents an effective means to keep pace with the technology needs of the facilities.

These initiates address, but are not all inclusive of, the critical aspect of safety the National Academies of Science and Engineering called "technological vigilance." They observed that for effective safety performance, safety rules, goals, supervision, and evaluations of performance, are not enough. The safety culture -- the technical competence of workers, their motivation and sense of responsibility -- are equally important. This issue was probably one of the more challenging hurdles that the commercial nuclear industry had to face, and there is no reason to suspect that it will be any less difficult for the Department.

IV. Conclusions

I realize that to a group of technical practitioners like yourselves this perspective may sound long on philosophy and short on technical insights. However, the message I want to level with you today is that the responsibility for safety does not reside merely with the regulator, nor the utility president, although they certainly have an oversight and management role. Safety happens in the workplace, through the efforts, motivations, and technical inquisitiveness of professionals like yourselves and others who, in a real sense, "control" safety. In this field of nuclear air cleaning, you can have a substantial impact on safety performance. By your involvement in standards development, sharing of operational experience, and interaction of both workers and the public, you can, as a group, have a greater and more productive influence than any regulator or manager can. I challenge you as I and the Department's workforce have been challenged to accept this responsibility, to empower yourselves to be an active contributor to nuclear safety. To be critical of your operations, to foster an environment that dispels complacency and technical stagnancy. I firmly believe that the fate of nuclear power and operations like those of DOE rest not so much on its institutional prescriptions, but rather on the strength of you and other professionals.

DISCUSSION

FIRST: Thank you very much for that interesting and informative talk on DOE's future programs. I am sure you are all impressed with the fact that the philosophy of the two keynote addresses we have heard this morning are quite similar in that they seek to involve individuals and encourage responsibility and thinking at all echelons, as contrasted with depending on mechanisms and computers. I am encouraged greatly by this because I think that human involvement is the key.

SCHMIDT: I have been associated with the nuclear industry since 1942. I agree with your overall philosophy. I wish to take exception with your statement that safety cannot come from the outside, that it must come from the inside. I have been concerned for a number of years with a fact that there is a complete lack of performance standards for air monitoring from stack emissions as well as for onsite and offsite ambient air sampling. When I say a complete lack of standards, I mean with respect to particle size, flow calibration, flow measurement, and filter leakage. Those of us in this field look to the Department of Energy for setting standards and find it difficult to bring about when there has been no leadership from the top. Recently, as you may know, there has been a landmark example at the Waste Isolation Pilot Plant in New Mexico where some very expensive stack monitoring equipment was challenged by people who were concerned with public health and safety and it was thrown out as being unsatisfactory. I think that it should have come the other way around, with standards set from above by the Department of Energy.

KNIGHT: There are several things involved. First, I would like to touch on the philosophical aspects of the challenge. I do not believe that I did offer that there is not a need for guidance and a need for standards. From a more deeply philosophical point of view, even in the evolution of standards or the evolution of guidance, the mere existence of guidance does not guarantee safety. Its a useful tool, a step, but both the development of the standards and their implementation within the department are going to depend on the strength, the desires, and the technical competence of the individuals working on standards committees. I have long been a believer in national consensus standards, developed by people who sense a need and work to develop solutions within their professional community; bringing the message, as you have today, to the attention of people who need to focus on it and other issues but I cannot and will not concede that safety comes only from the individuals who are doing the job, not from the institution. What comes out of the institution, I believe, is a manifestation of their efforts.

INTRODUCTORY COMMENTS OF CHAIRMAN FIRST

Our next speaker is J. Louis Kovach who is president of Nuclear Consulting Services, Inc. The biography he has submitted is very short. He says he has thirty years in the nuclear and non-nuclear air cleaning business. I think most people here know Lou Kovach quite well from prior conferences as well as from other contacts. I will amplify his biography slightly and say that he has been a consultant to the Atomic Energy Commission as well as the current Nuclear Regulatory Commission and is an all-around expert on nuclear and non-nuclear air cleaning. Were it not for the fact that we are dealing with science and engineering on the cutting edge of current interest I would call him a Renaissance man. Perhaps that designation is an anachronism, but I like it. So, Lou, since you wanted to save more time for your presentation and less for my introduction, I will let you get on with it.

REVIEW OF CONTAINMENT VENT FILTER TECHNOLOGY

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I. ABSTRACT

The technology applied for the design and construction of containment vent filters is compiled and reviewed. The national positions leading to the selection of venting or method of filtration are extracted from position papers. Several areas of further information needs are identified.

II. INTRODUCTION

The first large nuclear reactors were built not for power generation, but for plutonium production. These reactors were not equipped with pressure containments. After the Windscale accident (1) air cleaning devices were installed in these production reactors to mitigate potential accident consequences. These systems are called "filtered confinements", such systems are in use at the N reactor in Hanford (2) and at the Savannah River Plant (3) in the U.S. At the same time, the non-power reactor fuel reprocessing technology forced the development of air cleaning systems capable of decontaminating air streams heavily contaminated by aerosols. The two major facilities in the U.S., Hanford and Savannah River use deep beds of curly fiber (4) and deep beds of segregated sized sand and gravel (5) (6) for this purpose.

When the construction of power reactors commenced in the U.S. and most other countries, these reactors were equipped with pressure containments and the design basis accidents (DBA) were assumed to be the governing conditions. The potential consequences of severe accidents which may or may not involve containment failures were covered in technical and policy studies such as WASH 740 (7) and WASH 1400 (8).

Discussion of venting of power reactor containments in case of DBA or severe accidents were rare (9) (10) (11) (12) and not a popular topic. Only some experimental reactors were equipped with vented containment (13) (14). Soon after the TMI-2 accident, it was realized that the accident exceeded the design basis and severe core damage occurred (15) (16) (17). This information rekindled interest in vented containments and further studies were made in many countries, particularly as the reevaluation of the past accident source term was completed.

While Sweden was, for a time being, the only country committed to filtered vented containment, if the post accident pressure within the containment exceeded 0.65 MPa (a preset value), (18) (19) other countries also evaluated the cost-benefit of such systems (20) (21) (22) (23).

After the Chernobyl accident with it's actual and political fallout, additional evaluations were performed and lead to the consideration of post accident venting in W. Germany, Finland, Switzerland, Italy and the U.S.A.

In the recent past, Canada, France and W. Germany announced designs and completed modification plans for post accident venting (24) (25) (26) (27). Many other countries evaluated post accident hydrogen recombiner improvements, enforced inerting requirements to ameliorate potential hydrogen explosion effects (28) (29), initiated extensive experimental work and process analysis for post accident venting (30) (31) (32).

The vented containment technology is based on the methodology which was developed for the early fuel reprocessing sites, with the additional benefit, in some cases, of using the thermal mass of the filtering media as a condenser also, as is the case in the Swedish design. In most cases, inert packing material (gravel, sand, etc.) is used as a primary filter in some cases plans are to follow by high efficiency particulate filters and solid adsorbents. The other alternative of using metal fiber filtration is also derived from the use of high efficiency stainless steel fiber moisture separators at the USDOE (ex USAEC) Savannah River Plant since 1963 (33) (34) (35). Therefore, it is worthwhile to review the extensive historical experience with gravel-sand bed filters and stainless steel demisters to see how they performed in arresting the release of particulate and vapor phase contaminants. These type of filters do not remove gases except by scrubbing action of condensing vapors or by coalescing liquid droplets, thus while high particulate filtration efficiency (99.99+%) can be achieved, the gaseous capture rate is relatively low. The adsorbents used in the SRP and Hanford confinement systems were designed only for elemental iodine removal in case of loss of coolant accidents (36) (37) and while improvements were made in adsorbent quality, they could not remove organic iodide forms at high efficiency in 2.5 cm (1.0 inch) impregnated carbon bed depths, from a 100% RH air stream at approximately 30 cm/sec (60 FPM) velocity.

The Steam Generating Heavy Water Reactor (SGHW) at Winfrith Heath UK also has a vented containment and extensive laboratory pilot scale experiments have been performed on it's containment venting design (10).

The analysis of installation design data and the history of performance of the sand bed and stainless steel mat filters has been reviewed recently (38) and is discussed here only to the extent of direct comparison to containment vent filter systems.

III. HISTORY OF DEEP BED FILTRATION OF PARTICULATES IN THE NUCLEAR INDUSTRY

1. Sand-Gravel Beds

Deep bed coke or slag filters have been used for aerosol removal in the chemical industry prior to World War II. Their first application for radioactive material filtration was initiated at the then "Hanford Works" of the USAEC in 1948, when high activity levels were detected (in particulate form) in the chemical processing ventilation stacks.

The size of a typical sand bed filter of those early days had the following typical size and capacity:

Air Flow	60,000 m3/hr	35,000 CFM
Filter Height	4,267 mm	14.0 ft
Filter Cross Section	26 X 26 m	85 X 85 ft
Pressure Drop	1900 Pa	8 in wg
Collection Efficiency (of total activity)	99.7%	
Life	(5 years) *	40+ years

The grading of the filtering media in the direction of flow was as follows (from bottom layer upward):

mm	Inches	Sand & Gravel Size	
305	12	1 inch to 3 inch	25 mm to 75 mm
305	12	0.5 inch to 2 inch	12 mm to 50 mm
305	12	0.5 inch to 4 mesh (US)	12 mm to 4.8 mm
305	12	4 mesh to 8 mesh (US)	4.8 mm to 2.4 mm
610	24	8 mesh to 20 mesh (US)	2.4 mm to 0.84 mm
915	36	20 mesh to 20 mesh (US)	0.84 mm to 0.3 mm
152	6	4 mesh to 8 mesh (US)	4.8 mm to 2.4 mm

The bottom gravel was supported on molded ceramic tile distributors. The operating velocity was approximately 2.5 cm/sec (5 fpm). The subsequent installation of similarly graded gravel-sand filters at SRP increased the capacity and the cross sectional area of these filters.

Air Flow	204,000 m^3/hr	120,000 cfm
Filter Height	2440 mm	8 ft
Filter Cross Section	30 X 70 m	100 ft X 240 ft
Pressure Drop	1194 Pa	4.8 inch H_2O
Collection Efficiency % (DOP)	99.95% ± 0.02	

* Estimated at time of construction, actual operational life of sand beds have been approximately 40 years.

Collection Efficiency % (of total activity)	99.95% *
Life	30+ years

The SRP sand-gravel filter packing gradiant is in the direction of flow (from bottom up):

Layer Thickness mm	Inches	Layer Particle Size	
305	12	1.25 inch to 3 inch	31.4 mm to 75 mm
305	12	0.63 inch to 1.25 inch	16.0 mm to 31.4 mm
305	12	0.25 inch to 0.63 inch	6.4 mm to 16.0 mm
152	6	0.25 inch to 8 mesh (US)	6.4 mm to 2.4 mm
305	12	8 mesh to 20 mesh	2.4 mm to 0.84 mm
915	36	20 mesh to 50 mesh	0.84 mm to 0.3 mm
152	6	0.25 inch to 8 mesh	6.4 mm to 2.4 mm

The designs of both the Hanford and Savannah River sand filters were in the up flow mode. Part of the reason for this was the shielding consideration. Later it was rationalized that Russian data indicated, that down flow sand beds tended to clog by caking and, therefore, the upflow is more advantageous (39). Dewatering of the beds can also be achieved by the upflow mode.

A study by Thomas and Yoder (40) of the aerosol filtration efficiency of uniform sand beds was reported in 1956 showing that the maximum penetration of two different sand beds (with the average particle sizes of 0.16 mm and 0.36 mm diameter) occurred in the 0.25 to 0.5 micron size Dioctyl phtalate (DOP) aerosol size range.

The penetration for the 0.1 to 0.9 micron aerosol size ranges decreased with decreasing air velocity (from 2.18 cm/sec to 0.109 cm/sec); as the aerosol particle size approached the 1.0 micron size the velocity effects were changing, (Figure 1).

Another significant observation was that while at the 0.1 micron DOP aerosol size, the flow direction did not have an effect on removal efficiency, for the 0.3 micron aerosol size, the downflow was two-fold more efficient, and for 0.8 micron aerosol size, the downflow mode was an order of magnitude more efficient than the upflow mode of operation (Figure 2).

Figure 3 shows input-output activity of one of the SRP gravel/sand beds, covering the time period of 1963 to 1969 (41). The data shows that the efficiency of the bed increased probably for two reasons, the compacting of the bed and the decrease of the interparticle space dimensions as filtered particle deposition took place.

* There were two significant penetrations when air distributor heads corroded, these events resulted in lower efficiency of short duration.

Figure 4 shows the movement of the activity peaks downward (opposite to flow direction) from the 2.4 mm X 0.84 mm (8 X 20 mesh) sand layer into the 4.8 mm X 2.4 mm (4 X 8 mesh) gravel layer during the period from 1957 to 1975 (41), which shows that filtrating efficiency is increasing due to filling up of void spaces without completely blocking the flow paths. Extension of the potential service life for such systems can be made by correlation of the void volume and the pressure drop changes in the various particle size packing (42). Interestingly, most of the design sizing studies were made on 0.59 to 0.3 mm (30 X 50 mesh) sand, while the life data indicates that the bulk of the particulates are removed (on an inlet activity basis) before this layer is reached.

A multistage filtration test rig, where conventional high efficiency particulate air (HEPA) filters follow sand bed filters was constructed at the Savannah River Plant (43). The data shows, that using 238 Pu as a tracer, an additional DF of 100 can be obtained with this HEPA filter addition. A reverse of this design, sand bed filter following conventional HEPA filter banks was also constructed at the Savannah River Laboratory (44). This sand filter was installed to provide additional protection against the release of radionuclides, (mainly 60C0, 238Pu, 239Pu, 244Cm and 252Cf) in case of an accident, such as fire or explosion that could breach the confinement provided by the HEPA filter banks alone.

Boehm et. al. (14) reported on the comparative evaluation of conventional HEPA filters and shallower sand bed filters than in use in the U.S. (550 mm versus ~ 2500 mm) for filtration of sodium oxide aerosols. In their experiments, the flow was downward through the narrow range graded sand bed. The report indicates in excess of ten-fold capacities for sodium oxide aerosol in the sand bed compared to the conventional HEPA filter at the same pressure drop through both type of filters. The HEPA filter was extensively damaged both by corrosion and by the exothermic reaction of the "unburnt" sodium, while no damage was observed on the basalt sand filter (Figure 5).

Gravel/sand beds have an additional advantage, which is shock attenuation. Shock waves which can easily damage modular HEPA filters, will have limited effect on the sand bed filters. Although only limited actual data is available on activity loss due to shock waves (45).

The other major advantage of the sand filters is the very large heat sink available in such beds. While neither the Hanford or SRP beds were designed to operate as condensers of potential steam-air mixtures, the capability to do so exists.

Kovach (11) proposed a PWR containment vent scheme for a combination system consisting of Heat Sink Sand Bed Filter and Carbon Bed in series. The design analysis for a 3000 CFM unit was based on using the coarse carbon bed as a delay bed similar to Kr-Xe delay, rather than the short life conventional radioiodine filtration bed models.

The first gravel/sand granular bed vented reactor containment was designed for the Zero Power Plutonium Reactor (ZPPR) (14) built at the National Reactor Test Station (NRTS) in Idaho.

The cross section of the containment with the gravel and sand bed roof is shown on Figure 6. This roof also serves as a heat sink and as the first and second stage post accident filter followed by a gallery of HEPA filters. The experimental model data indicated high removal efficiencies for the 760 mm (30 inch 0.84 X 0.3 mm (20 X 50 mesh) sand bed section as shown on Figure 7.

The potential nuclear application of the vertical cross flow sand bed filter has been reported by Goossens (46) where in model experiments with non radioactive contaminants, 1.5 mm (14 mesh) diameter sand in a 4.5 cm (1.75 inch) thick panel bed, removed flow ash at 99% efficiency.

Lapple (5) developed an activity removal efficiency equation for sand beds based on limited amount of experimental data.

$$= 1 - \exp\left(-KL^{0.5}\ V^{-0.33}\ D^{-1.33}\right),$$

where:

- : fractional collection efficiency on inlet radioactivity or mass basis (not particle count!).
- L : depth of filtering sand layer (ft).
- V : superficial velocity, based on empty bed cross section (FPM).
- D : average sand grain diameter (inch).
- K : proportionality constant (based on sand properties).

As an example, three sands tested at identical average grain diameter showed K values from 0.035 to 0.053. While attempts were made to correlate sand properties to establish precise parameters for K value estimation by measuring "roughness" factors (47), the only current reliable method is still the experimental determination of removal efficiency.

Sand beds in use in the U.S. reprocessing facilities are packed to a "loose" density which results in a void fraction of approximately 0.4. This practice makes proper evaluation of the sand/gravel filters difficult. While experiments have indicated that it is possible to compact (dense pack) sand beds, such practice also resulted insignificant increase in pressure drop (48), with an improvement in filter performance, particularly at higher velocities (710 cm/sec). Considering that "loose" packing is very difficult to reproduce, while a free fall type packing can result in reproducible dense packing, aerosol removal efficiency studies need to be properly performed for dense packed beds. When using gravel/sand bed filters for containment venting applications whether the bed pressure drop is 1200 Pa or 2400 Pa does not make much difference, while the overall size and predictable efficiency is very important.

2. Deep Fibrous Beds

The application data for packed deep fiberglass filters for reprocessing off-gas filtration was generated by Lapple (5) in 1948 while the sand filters were being built at Hanford. The small scale testing indicated favorable results and by 1950 (4) a full size system was in operation at the Hanford Works of the AEC.

The typical dimensions of the unit are:

Airflow	60,000 m^3/hour	35,000 scfm
Filter Depth	2772 mm	9 ft
Filter cross section	8.5 X 21.4 m	28 X 70 ft
Superficial Velocity	12.4 cm/sec	25 fpm
Pressure Drop	1245 Pa	5 inch wg
Collection efficiency (% total activity)	99.99	

The composition of the fibrous bed in the direction of flow.

Fiber Diameter	Bed Depth (Final Design)	
	mm	inch
200 micron	305	12
110 micron	305	12
30 micron	610	24
15 micron	305	12
7 micron	305	12
1 micron		0

The deep fibrous bed systems, while performing very well at the Hanford works, with life times similar to that of the sand bed filters did not become wide spread. There has been experience at Hanford of irrigating the fibrous filters to remove caked deposits from the inlet side which "regenerated" the fibrous bed plugged by ammonium nitrate. (49)

Juvinall (50) prepared a review of published information on sand bed filter use in nuclear applications.

IV. EXPERIENCE WITH SHALLOW FILTER APPLICATION IN THE NUCLEAR INDUSTRY

1. HEPA Filters

All currently produced fire resistant HEPA filters are constructed of 0.38 mm thick non woven media consisting of typically 1 to 4 micron glass fibers. This media is pleated into various shapes to result in a velocity of ~2.0 cm/sec through the media. While this type of construction has a very high efficiency for collecting aerosols, even in the difficult to filter 0.1 to 0.3 micron range, they are subject to damage by;

a) mechanical failure (puncture, pressure shock) (51) (52) (53) (54)

b) chemical failure (corrosion of frame or media) (55) (56) (57)

c) plugging by water droplets (58) (59) (60)

d) plugging by relatively low particulate loads (61) (62)

e) fire (while the media is non flammable, any burning material on the surface will melt the fiber glass fibers) (63) (64) (65)

Therefore, their application even for normal operation or loss of coolant type accidents, requires additional protection, i.e., moisture eliminators, prefilters and often redundancy of HEPA filter banks and even redundancy of full filter trains.

However, evaluation of existing air cleaning systems (not designed for FC venting) under simulated or actual accident conditions is a worthwhile approach because it shows the advantage and disadvantage if the potential application of various air cleaning compounds as FVC components.

2. The Containment Systems Experiments CSE (66)

Table 1 reproduces results from CSE runs A-13 through A-15 for iodine removal and Table 2 the results for Cesium removal.

In the test runs with 5 cm deep stainless steel fiber demister moisture separators, the distribution of the particulate iodine was 0.8 - 31.0% on the moisture separator and 22.0 to 28.3% on the HEPA filters. While the cesium distribution between these two components was 51.7% to 62.5% on the stainless steel moisture separator and 34.4 to 42.5% on HEPA filters.

Further analysis shows that CSE Run A-16 which consisted of releasing aerosol for 120 minutes while the filter train was operating, showed the highest particulate iodine collection on the demisters (in the other experiments, the same make as those used at Savannah River the filter train was started 30 to 40 minutes after the aerosol release).

The cesium removal efficiency of the moisture separator was 30 to 79% higher than that of the HEPA filter, when a moisture separator preceded the HEPA filter.

The HEPA filter by itself is more efficient for cesium aerosol filtration than the moisture separator, as shown by Experiment A-13 which used only a HEPA filter, but for the case of iodine, the same A-13 test without moisture separator, showed the lowest collection efficiency on the HEPA filter (8.8% of total iodine).

The CSE experimental series indicated that a very reliable, non-fragile fire resistant media (not even designed for particulate filtration) is in fact a very good particulate filter and by proper design modifications can be a safe high efficiency post accident particulate collection device.

3. The Savannah River Rod Drop Accident (35)

On 9 November 1970, an animony beryllium source rod was dropped from the charge machine at the SRP K reactor. The failure released ~80,000 Ci of radioactivity into the reactor process room. The K reactor, like the other production reactors at SRP, has a vented confinement. (3)

This venting exhaust system consists of five unshielded filter compartments of which at least four are on stream continuously. The filtering components are of 610 X 610 X 50 mm stainless steel fiber moisture separators followed by standard 610 X 610 X 295 mm HEPA filters and 25 mm bed depth pleated carbon adsorbers in 610 X 610 X 295 mm holding frames. The nominal air flow rating of the moisture separators is 2700 m^3/hr (1600 cfm), but at the time of the accident lower flows were in effect.

The distribution of activity among the operating confinement filters was:

Compartment	Initial Activity Ci	% of Total on Filter Trains	Activity Dist. on Moist. Sep.	% on HEPA
K-2	631	46	86	14
K-3	672	49	76	24
K-5	64	4.7	100	0
K-6	4	0.3	100	0

The overall removal efficiency of the confinement filter trains was 99.999% of the total inlet activity, and only 3 milicuries were released through the filter trains. The K-3 compartment initial external radiation reading was 70 R/hour at 2.5 cm (1 inch).

There are 20 moisture separators in each compartment and 82% of the total activity was retained by the moisture separators while only 18% reached the HEPA filters. This filtration took place in the "dry" mode, which is not the design mode of the moisture separators. (Note that the 12th AEC Air Cleaning Conference a report was presented which reversed the moisture separator: HEPA activity loading (67).

4. The German Kerforschung fur Kernenergie (KfK) Work (68) (69)

While there are differences in terminology in that the German workers consider these filters "deep bed", the configuration reported is not nearly the "deep" bed filter application as developed at Hanford. Therefore, the nomenclature used in this paper is "shallow" bed, on an arbitrary basis with <305 mm (12") bed depth as the cut off point for deep beds.

The metal fibers used by Dillman are stainless steel and consist of various fiber mat configurations for both prefilters and HEPA filters. The work on prefilters of this type has also been reported by Klein and Goosens. (70)

The prefilter stage consists of stainless steel (AISI TP 316, DIN No. 1.4401) fiber with diameters of 45065 micron, 22 micron and 4 micron diameters. The pressure drop of the filter through a 610 X 610 mm face area was 15 mm H_2O.

At a dust loading of 4.93 kg/m^2 filter face, the pressure drop rose to 30 mm H_2O and the dust loading was distributed as follows:

First part of pack	45-65 micron wire	4.8 kg/m^2
Second part of pack	22.4 micron wire	129 g/m^2

The test aerosol used was in the 1-10 micron range and the total removal efficiency was 97%.

It is obvious that for high particulate loadings of 1-10 micron aerosol diameter, the coarse wire size mat gives the high holding capacity and the finer wire size improves the efficiency. These studies were performed with dry aerosol, but the removal behaviors of these prefilters is similar to that of the "Moisture Separators" used in the U.S.

The use of smaller diameter stainless steel fiber resulted in a secondary, HEPA filter, stage which behaves similarly to glass fiber HEPA filters, without many of the disadvantages of glass fiber construction

Current development of 2 micron fibers reported by the Dillman, indicates, that 99.97 percent removal efficiency is obtainable at 25 cm/sec (50 fpm) face velocities.

The data for monodisperse aerosol removal efficiency versus mat superficial velocity needs to be fully developed on identical construction mat assemblies, to obtain direct comparison between glass fiber HEPA filters, and stainless steel fiber HEPA filters.

The pressure burst resistance test of these stainless steel mat filters in 610 X 610 X 290 mm size with 30 layers of 4 micron fibers of 300 g/m^2 density indicated no failure at 0.28 bar pressure drop. However, densification of the mats occurred which reduced the removal efficiency by an order of magnitude at 10 cm/sec (20FPM) velocity. No data was reported for these filters for particulate release under the air pressure shock event.

Figure 8 shows the folded filter design. Figure 9 shows the effect of an air pressure burst on the stainless filter and Table 3 shows the properties of the stainless filters tested.

V. VENTURI SCRUBBER - FILTER COMBINATIONS (71)

The first nuclear application of venturi-type scrubbers was for the Fast Flux Test Facility (FFTF). The FFTF scrubbed venting system is part of the Containment Margins System (CMS), and is designed to deal with very low probability events involving the release of primary system sodium, fuel and core debris into the reactor cavity. A system for venting and controlling excessive FFTF reactor containment pressure consists of a 30 inch diameter containment penetration line with two isolation valves located outside of containment. The isolation valves can be remotely operated from the control room and are equipped with key lock switches to prevent unauthorized operation. Downstream of the isolation valves is a combination scrubber/filter system. The scrubbed portion consists of a venturi scrubber utilizing water sprays (with a chemical additive to enhance removal of elemental iodine) to remove an estimated 90% of any particulate. The scrubbed gas then enters five cylindrical filters arranged in parallel composed of polyproplylene in a fibrous mat. The fibrous filter is estimated to remove about 99% of the remaining particles. Thus, the combined removal efficiency of the system is 99.9%. The effluent is then released to the stack, after being continuously monitored for gross radioactivity content. The system is designed as safety-related up to and including the outboard containment isolation valve, but is non-safety grade beyond that point.

VI. RISK REDUCTION EFFECTIVENESS OF FILTERED VENTED CONTAINMENT (FVC) (72)

The following potential source accidents were considered in the past (1)

1) Direct Bypass

   The initiating event causes containment to fail or to be bypassed. Examples include large earthquakes, stem generator tube ruptures, and check valve failures which cause primary system inventory to be released outside containment.

2) Failure to Isolate

The containment isolation system fails to provide a leak tight boundary.

3) Pre-core-melt Overpressure

Failure to remove heat from containment as fast as it is being produced in the core region causes the containment to fail by steam overpressurization. The emergency core cooling systems may subsequently fail because the pumps cavitate or because large structural deformation may damage the cooling lines.

4) In-vessel Steam Explosion

An explosive interaction between molten core materials and water in the lower plenum of the reactor pressure vessel destroys the vessel and causes containment to be breached by a missile.

5) Ex-vessel "Steam Spike"

Containment fails as a result of rapid pressurization by steam when the molten core penetrates the reactor vessel and is rapidly quenched by water in the reactor cavity or on the containment floor.

6) Hydrogen Burning

A widespread hydrogen deflagration, or a local detonation, causes containment failure at any time during the accident if airborne hydrogen concentrations are sufficiently high and flammability conditions are attained.

7) Long-term Overpressurization

Containment fails as a result of gradual overpressurization from steam and noncondensibles while the molten core is attacking the concrete basemat of the reactor cavity.

8) Thermal Degradation

Thermal radiation from the hot core materials in the reactor cavity and/or hot gases from the decomposition of concrete raises the containment structural temperature beyond the point where integrity can be maintained. Leakage paths may develop through containment penetrations seals.

9) Basemat Meltthrough

The hot core materials melt through the concrete basemat.

Some of these modes in the recent year have been found to be less likely (such as 4), for LWR reactors and in some of the cases FVC would not be a justifiable risk reduction. However, for cases 3, 7, 5, and 8 and the moderate H deflagration case (moderate 6), the FVC can mitigate the consequences of the severe accident. Naturally, the level of mitigation is also dependent on the containment design and on the population density.

As an example, the fraction of risk attributable to various containment failure modes for six different U.S. LWRs is shown on Table 4.

This tabulation shows that the relevant modes are 3, 5, 6, and 7 most of which can be mitigated by FVC.

Table 5 shows the relative public consequences for various accident types and various FVC system filtration efficiencies for a highly populated Northeastern USA site.

The table shows that in case of failure of ESF to operate even an inefficient (90%) FVC will reduce the relative early fatalities by two orders of magnitude. Thus, from some reactor types at specific sites the FVC application is beneficial.

At the same time, the potential negative effects of an FVC also have to be reviewed, as an example:

1) Activation through an inefficient vent may produce unnecessary releases (particularly large quantities of noble gases) where the vent is activated on a pressure signal (or automatically) but the containment would not have reached failure pressure.

2) In some accident scenarios, venting could reduce the containment noncondensible inventory, resulting in loss of net positive suction head for residual heat removal system pumps.

3) In the same scenario, inadvertent operation of the FVC can develop severe vacuum in the containment.

4) Venting for extended periods will deplete the necessary water level in the containment sump.

The potential vent volume rates per unit time postulated to date had a wide variation from 2500 - 300,000 CFM. The latter high values are not realistic, (11), (20), (24), (25).

Design considerations also vary whether the FVC air mover is solely the pressure differential between the containment and the ambient external pressure or if additional air moving devices are used to move gases and stem through the FVC when the containment failed.

Regardless of the type of FVC used, the following three criteria can be fulfilled:

a) Provide a heat sink
b) Have particulate filtration capability (aerosols)
c) Have vapor/gas clean up capability (iodine forms)

Protection from short lived noble gases at a realistic cost can not be achieved.

The design has to assure that proper locations are vented. As an example, for a BWR the optimum location is the wet well venting, but his mode prevents efficient forced filtration if the containment failed, therefore, dual FVC entry points may be needed. The case for PWR containment is simpler because any point of the containment can be vented.

The plugging of inlet lines, and channeling through filtering components have to be prevented.

In all cases, the main elements are either gravel/sand filters, metal fiber HEPA filters by themselves or in combination with other conventional air cleaning devices, such as venturi scrubbers, demisters, HEPA filters and iodine adsorbers. The components operate either in dry initial modes, or in liquid filled srubbers followed or incorporating other filtration modes. (71) (73)

The FVC technology is based on the development and application of these components from fuel reprocessing and vented confinement experiences from other facilities, and from the non nuclear pollution control industry. These systems have operated successfully for a number of years in their previous application fields. This available operational experience data should be carefully reviewed when designing any concept for FVC systems.

VII. THE SWEDISH POSITION

<u>Conclusions of the Swedish Nuclear Power Inspectorate</u> (74)

The Inspectorate concluded that, in order to achieve the objectives given by the requirements from the Radiation Protection Institute, more far-reaching measures than those discussed by the utilities in their status reports were needed. At the same time, the improved state of the art allows for finding alternative technical solutions to those chosen for Barsebaeck. In the light of those facts it should also be possible to take credit for prepared active efforts earlier after initiation of an event than after 24 hours which were stated for Barsebaeck.

The improvements that were considered to be necessary for those events in mind which are far beyond present design basis, consist of accident management procedures as well as hardware reinforcements and systems. The details and design that have to be worked out under the responsibilities of the utilities should be based on the following discussion.

The primary aim shall still be the prevention of core damages. In this respect a high quality in operation and maintenance is of high priority. Also the implementation of experiences gained from operation and incidents are of basic importance. Much attention should be paid to the onsite preparedness against accidents in order to gain profit from all possibilities to re-establish core cooling within due time if malfunctions should occur.

If a core accident nevertheless should occur, strategies must be prepared specific for each plant, to protect the containment and, as soon as possible, to reach a stable final state in the plant. In many cases, this would mean that the damage core should be recooled and covered with water in a pressureless primary system. If vessel melt-through has occurred it would mean core debris covered with water in the lower parts of pressureless containment. This should imply that flooding the containment with water above core level should be considered as part of the strategies. Important aims of the strategies are to create margins to overpressurization if more rapid pressure build up should occur, in order to gain time for establishing more permanent cooling and to create margins for uncertainties about the behavior of the radioactive substances in various stages of an accident.

It is particularly important that the containment barrier is intact at least during the first 10-20 hours after a core accident. Accordingly, necessary reinforcements should be made for those situations when the containments could be damaged as part of an initiating event or during the initial phase of a core accident. For BWR's protection should be implemented for the event when the pressure-suppression function is insufficient to limit containment pressure at a LOCA. This event could otherwise result in a damage containment barrier and a highly questionable reliability of the core cooling systems. Moreover, necessary improvements should be implemented for protection of the barrier against direct mechanical or thermal damages along with core melt-down progression in the containment and associated effects. Special attention should also be paid to situations when the containment is not properly isolated, so called by-pass situations. The protection against overpressurization in a longer time period, for example through malfunction of residual heat cooling or through gas production should be complete.

In order to protect the containments against overpressurization and to obtain a stable final state a controlled pressure relief should be possible. The relief devices should be designed to that they can be activated independent of operator actions and independent of other safety systems if the design pressure of the containment is exceeded considerably. Like other safety systems, the relief devices should be available for active use by the personnel. The relief equipment should be so designed that it, together with other measures for protecting the containment, with a high reliability will ensure that the release to the environment is kept below 0.1% of core inventory of radioactive substances, noble gases excepted.

The amount of activity should be related to a core of about 1800 MW thermal power. When assessing the 0.1% limit, some regard to the contribution to the environmental consequences from a specific substance could be taken. Measuring equipment should be available to make it possible to estimate the release in advance and to monitor the release afterwards.

VIII. THE SWEDISH FILTRA DESIGN (Barsebaeck)

The project history and the background experiment literature is described in the "Filtered Atmosphere Venting of Light Water Reactor Containments" Final Report generally referred to as "Filtra Final Report" (18). The basic design consideration was for the two ASEA ATOM BWRs at Barsebaeck. The system designed for 24 hour full automatic operability without any outside intervention consists of two main parts:

1. The vent lines connected to both reactor containments containing a rupture disk, which opens at a preset value before overpressurization damage can occur to the containment.

2. A gravel bed filter for the removal of particulate and condensable radioactivity in the steam-gas flow escaping from the containment after the rupture disc has opened.

The design basis for the Barsebaeck containment is 0.5 MPa. The rupture disc is set to open automatically at 0.65 MPa. (Data indicates that the 0.15 MPa overpressure on the containment would not cause leaks in the containment). A separate manually operated shut-off valve is installed downstream of the rupture disc to permit reisolation of the containment, if necessary. The 600 mm diameter vent line from each reactor containment is equipped with individual drain collector tanks. The venting line originates for each reactor containment above the condensation pool. Both the rupture disc and the manual isolation valve are located in the immediate vicinity of the containment. The leak tightness of the rupture disc is tested by isolating the shut-off valve and pressurizing the space between the rupture disc and the shut-off valve. Two 150 mm diameter lines with two normally closed valves in series run parallel to the large vent line. These small lines are connected to the dry well and to the top of the containment and are intended for manual venting of the containment.

The small lines permit manual containment depressurization even after an accident scenario which could flood the area of the inlet to the automatically opening line in the suppression pool.

The condensate drain tanks installed in the lines are of sufficient volume to receive and store all condensate obtained during the first 24 hours of venting following an accident.

The actual filtration plant consists of a gravel bed comprising of a 40 m high 20 m diameter 1 m wall thickness concrete cylinder of 10,000 m^3 volume filled with 15,000 tons of 25-35 mm diameter quarzite gravel. The main horizontal vent lines from the reactor containments turn vertically upward through the annulus of the gravel bed into a distributor section of small lines entering above the top of the gravel packing.

The entering steam condenses in a downflow mode on the colder gravel packing, and the condensate trickles downward ahead of the condensation zone. The 10,000 m^3 of gravel was conservatively sized to condense all of the steam flowing out of either reactor containment for a period of 24 hours after a severe accident.

The outlet is above the bottom of the gravel bed, leaving a 700 m^3 volume for condensate accumulation. This design permits free passage to the steam air mixture. The 700 m^3 space can be - if needed - attached to pumps to remove the radioactive condensate. The outlet line is equipped with a restriction to limit the gas velocity through the gravel bed. The normal exhaust line cross section is 0.01 m^2, there is a high volume standby outlet also, which would be activated for the ATWS sequence only.

The venting system and the gravel bed are inerted with nitrogen to prevent hydrogen burning (and also to prevent biological growth on the system while in standby mode).

The Filtra is instrumented for pressure, temperature, liquid level, radioactivity, gas flow rate and valve travel limits. Provisions exist for sampling at various locations. A dedicated facility with batteries, monitoring, and data logging is part of the system.

While the automatic activation of the FVC is triggered by the blowout of the rupture disc at 0.65 MPa, the considered manual initiation of venting is based on lower pressures thus the FILTRA operation is not necessarily automatic.

Manual venting is considered when any of these events occur:

1) Containment pressure reaches 0.45 MPa and continues to rise.

2) Pool temperature rises above 95°C.

3) Simultaneous high pressure and high activity signals occur in the containment.

4) High containment water level signal is observed.

The diagram of the Barsebaeck FILTRA arrangement is shown on Figure 10.

The releases to the atmosphere through the FILTRA venting system in fractions of the core inventory has been estimated as follows:

		Fraction Released
Noble gases	Xe, Kr	1.0
Elemental iodine	I^2	0
Organic Iodide	CH_3I etc.	<0.01

Particulates or attached to particulates	I^2, Cs, Rb Te, Sb, Sr Ru	$<-10^{-4}$

While numerous experiments were performed on various scale models, the DF the values for elemental and organic iodides may be somewhat ambitious. After 24 hours operation, when the bed is heated above water condensation temperatures, the physical adsorption on the relatively small surface area of the gravel (compared as an example to conventional adsorbents), will diminish and the major removal path will probably be reaction with cesium compounds dissolved in the water or deposited on the gravel rather than chemisorption on the bare gravel surface. Scrubbing of methyl iodide or other organic iodine forms is very difficult to postulate in such a system, and at the high temperature after the condensation stops, the possibility of chemical reaction with organic compounds which could deposit also on the gravel may be enhanced in the bed rather than arrested.

The particulate filtration tests performed during the pilot scale experiments indicate that a high efficiency will be obtained especially while condensation in the gravel bed takes place.

The aerosol simulant used in the particulate filtration experiments has been an Fe_2O_3-CaO mixture. How well such an aerosol mix represents the highly water soluble cesium salts occurring in the accident has not been discussed. The experiments were conducted with both dry air and mixtures of air and steam. The outlet sampling was stopped when mist (steam) breakthrough occurred. The efficiency of the particulate removal increased with increasing steam content and was greatly dependent on superficial velocity, reaching a minimum efficiency in the 40 cm/s (80 FPM) range as shown on Figure 11.

The Barsebaeck FILTRA was completed on 31 Oct. 1985. Estimate for current construction price is $20 million.

IX. THE SWEDISH MULTI VENTURI SCRUBBER SYSTEM (MVSS) (75) (76)

The MVSS is the selected design for the other ten Swedish nuclear power plants, consisting of 7 BWRs and 3 PWRs. The process functions of the MVSS, water scrubbing and packed bed filtration are integrated into a single unit which can be located in the vicinity of either BWR and PWR containment.

The MVSS system consists of the following major process units:

1) A system for automatic (or manual) pressure relief,
2) A venturi scrubber array,
3) A pool for iodine absorption,
4) A liquid droplet separator,

located in a stainless steel lined concrete pressure vessel.

The MVSS handles significantly lower mass flow rates than the Barsebaeck Filtra because the other Swedish BWRs which are equipped with MVSS are also equipped with unfiltered venting capability for the accident sequence of a LOCA involving the partial failure of the pressure suppression pool.

The main design parameters of the MVSS are:

Gas mass flow rate	0.1-13 kg/s	
Gas composition	Steam, N_2, H_2, O_2, (PWR)	
Gas temperature	70-150°C	
Rupture disc opening pressure	0.5-0.6 MPa	
Aerosol size distribution	log-normal og = 2	
Aerosol mass mean diameter	1.5 um	
	BWR	PWR
Total amounts of aerosols	90 kg	180 kg
Amount of radioactive aerosols	20 kg	20 kg
Min. required decontamination factor for aerosols and elementary iodine	100	500
Total decay power	400 kW	
Earth quake ground acc.	0.15 g	

Based on the above design parameters the sizing data are as follows:

Filter vessel	BWR	PWR
Design pressure	0.3 MPa + w.c.	0.4 MPa + w.c.
Total volume	250 m^3	400 m^3
Water volume	180 m^3	270 m^3
Inner diameter	7 m	7 m
Gravel bed volume	8 m^3	9 m^3

The pressure relief of the containment to the MVSS is similar to the Barsebaeck concept in that both automatic (at 0.65 MPa) and manual venting can be actuated. The vent line size is smaller.

The pressure relief pipe from the containment enters into the water filled chamber and at the bottom of the pool splits into horizontal distributors which are feeding the individual venturi pipes. Each exhaust pipe in the multiventuri unit has one venturi nozzle. The pressure drop across the venturi is determined by the height of the water between the pipe outlet and top level of the scrubber pool.

The gas acceleration in the venturi nozzle creates reduced pressure which pulls scrubbing pool water (solution) through holes into the nozzle. There a dense cloud of droplets acts as a "filter" for solid aerosol particles, (Figure 12).

The design calls for additives to the scrubbing solution to enhance elemental iodine removal. The additives are sodium hydroxide and sodium thiosulfate.

Water droplets entrained after passage through the pool are planned to be removed by a small gravel bed. (8 m^3 for BWR, 9 m^3 for PWR) before the scrubbed vent gas is passed to the station stack. The gravel bed is drained back into the pool.

These MVSS systems are descendents of air cleaning systems from electric arc melting of metals.

For all of the design studies, the test aerosol was quartz dust and the carrier gas was air. Correlation between these tests and the expected removal efficiency for highly water soluble aerosol constituents from steam air mixtures has not been published.

The model experiments were performed in single nozzle and in a four nozzle test rigs.

The experimental quartz aerosol decontamination factors were for

BWR conditions	PWR conditions
500 - 10,000	1,500 - 30,000

It is claimed that no significant effect on the overall decontamination factor could be detected from evaporation or from liquid entrainment. The entrainment was checked by use of copper salt. Based on the similarity of equipment purposefully built to generate submicron aerosols from salt solutions and the MVSS design the entrainment effects are not adequately studied.

Cost estimates for the nine units ordered (Oskarshamn 1 & 2 will share one unit) are $18 million.

All nine systems will be installed by end of 1988.

X. THE FRENCH POSITION (25) (77) (78)

The French studies, reviewing WASH 1400 indicated that instantaneous containment failure due to steam or hydrogen explosions is not realistic, while delayed (one day or later) containment failure has to be considered. The delayed pressurization could be caused as an example by core-concrete interaction. The conservative review of such accident sequence make it incompatible with current French emergency plans (evacuation within 5 km and controls within 10 km of the plant).

At the beginning of 1980 the French safety authorities and EdF concluded that substantial reduction in the amount of aerosol releases can be achieved for containment failure type accidents if a filtered venting system is available. This decision initiated the PITEAS series of studies for the design of filtered venting concept.

XI. THE FRENCH DESIGN (78) (79)

The commitment for FVC in France was taken prior to Chernobyl and the mitigation for PWRs included filtered venting, detection and treatment of "abnormal loss" of containment integrity faults and minimization of releases to the atmosphere in the event of base mat melt through.

The reference characteristics of the fluid to be treated are based on:

1) LOCA.
2) Loss of all electrical sources.
3) Transient.
4) Loss of feedwater supply to steam generators.
5) Loss of emergency feedwater to steam generators;

and for design purposes it is:

Pressure	: 5.0 BAR absolute
Maximum mass flow	: 3.5 kg/sec
Approximate volumetric flow	: 15,000 m^3/h
Steam content	: 29 percent
Air	: 33 percent
Carbon Dioxide	: 33 percent
Carbon Monoxide	: 5 percent
Temperature	: 140°C
Total Aerosol to be treated	: 5 kg
Maximum Aerosol Conc.	: 0.1 g/m^3
Aerosol Diameter, Average	: 1 micron

The system consists of:

1) Containment penetration, 350 mm diameter carbon steel pipe with two in series manual isolation valves as close to the containment as possible.

2) An orifice plate to permit gas depressurization and lowering of relative humidity (superheating).

3) The sand bed filter is 316 L stainless steel 4 mm wall thickness (torispheric heads 5 mm thick). The design pressure is 1.3 bar absolute, operating pressure 1.1 bar abs. The tank diameter is 7.30 m and the height is approximately 3.5 m. The empty weight is 12,000 kg and the filled weight is 92,000 kg.

4) System gas conditioning unit.

5) Connecting piping.

6) Radiation monitoring devices.

7) Release duct (400 mm) inside the main station duct.

To assure that the experimental design conditions are met, the gas distribution within the downflow sand bed is controlled by an inlet deflector plate followed by sheet of Kevlar screen 500 mm above the sand bed. The open area of the Kevlar screen is 29 m^2 the face area of the sand is 42 m^2.

The sand itself is 800 mm deep with an average diameter of 0.6 mm.

The sand bed is supported on a 0.3 mm mesh opening fiberglass lattice on a layer of 200 mm of expanded clay which is on a light weight concrete bottom supporting the collector network piping with 0.3 mm strainer hole openings.

The outlet gas collecting network is equipped with a torus type drain collector system. Piping to and from the vessel is connected by 300 mm long Kevlar flexible connectors. The outlet piping is 400 mm diameter.

The piping and the sandfilter are insulated (80 mm thick) and wrapped with stainless steel. The sand is dried for loading and kept dry by a continuous 500 m^3/h flow of HEPA filtered dry air purge.

The system is shown on Figure 13.

Extensive laboratory and pilot plant testing has been performed by the CEA to select sand particle size, depth and operating velocity.

The purification coefficiency (DF) of the test series is summarized in Figure 14. It is interesting to note that the carrier gas velocity increase in 0.7 - 1.2 mm sand sizes decreased the efficiency while in 0.5 mm sand particle size increased the purification efficiency.

Figure 15 shows the purification coefficient variation versus time. The initial condensation in the sand bed significantly reduces the purification efficiency and after reaching a minimum, the dry-out slowly improves the decontamination factor to above the initial DF.

This behavior is the reverse of that reported for the Swedish Filtra where the gravel beds were observed to perform better when wetted. The wetting problem probably can be resolved by adding a layer of water adsorbent material on top of the sand bed, where the heat of adsorption of the water vapor would prevent condensation in the sand layer.

The location of the sand bed filters on the roof of the auxiliary building near the control soon may require additional shielding requirements. The currently designed filter structures are not seismically qualified.

XII. THE GERMAN POSITION (26) (80) (81)

The completion of Phase B of the German Risk Study of Nuclear Power Plants indicated several deviations from Phase A of the same risk study which was originally derived from WASH 1400. On the basis of Phase B for core melt type accidents, the following information was derived:

1) It will take approximately four to five days before failure pressure of the inner steel containment vessel (40 mm thick, 6 bar design pressure) of a 1300 MWe PWR would be reached.

2) This longer time available would permit the implementation of measures to avoid overpressure failure such as filtered venting.

3) Steam explosions due to metal-water reactions can be excluded as containment failure causing mechanisms.

Out of the spectrum of relevant PWR and BWR accidents sequences those were selected which lead to the highest requirements for the design of the vent filters.

The venting system in case of slow pressure increases (no spikes are considered) lead to several advantages:

a) Lower releases to the environment because release path is through filters.

b) The deliberate venting can be stopped.

c) In case of inadvertent containment leakages, the quantity of uncontrolled leak can be decreased by filtered deliberate venting.

d) Containment pressure can be reduced before base mat melt through.

e) The containment atmosphere can be exchanged if such action is desirable.

It is reported that the German utilities' voluntary decision to install filtered containment vents on all BWRs and PWR was made on Dec. 1986.

The design loads on components (several German designs as discussed later have major components inside the containment) before and during venting are shown in the following.

Loads on components of a filter system
inside the containment before opening the flow path

	PWR's	BWR's (BL 69)
Pressure	equal to failure pressure of the containment	equal to failure pressure of the containment
Loads from LOCA's	as defined as LOCA loads	as defined as LOCA loads
Loads from a hydrogen burn	pressure increase 1-3 bar/sec	pressure increase 1-3 bar/sec
	pressure and temperature history not yet specified	pressure and temperature history not yet specified
Fire from oil, paints, insulation	to be considered only during the first day	no fires assumed for inerted containments

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Thermal hydraulic initial and boundary conditions
immediately after the opening of the flow path

	PWR's	BWR's
opening pressure	$\leqq$ test pressure	between design and test pressure
temperature	saturation temperature	saturation temperature
steam content	$\leqq$ 100%	$\leqq$ 100%
H_2/O_2 - mixture	no burning is assumed in the filter	no burning is assumed in the filter
Loads from water droplets	to be considered if condensate exist	to be considered if condensate exist
mass flow	mass flow equivalent to the decay heat at the moment of opening taking into account heat sinks and water injection and the required pressure reduction	mass flow equivalent to the decay heat at the moment of opening after exceeding the storage capability of the condensation pool

time for opening the flow path	2-3 days (depending on the heat transfer into the containment)	$\geq$ 4 hours
	No venting to be considered 7 days after the beginning of the accident	No venting to be considered 7 days after the beginning of the accident

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The design conditions for the performance of the German PWR and BWR filter vent units are as follows:

Loads from airborne products (<u>design</u> values)

	<u>PWR's</u>	<u>BWR's</u>
Mass of aerosols	40 kg	20 kg
Decay heat		
- from aerosols	2 kW	180 kW
- from iodine	5 kW (if retention within filter)	20 kW
Mass Efficiency		
- aerosols	99.9%	99.9%
- elem. iodine	90 %	
Fission products efficiency		
- elem. iodine	$1 \cdot 10^{-3}$	$2 \cdot 10^{-3}$
- org. iodide	$1 \cdot 10^{-3}$	$3 \cdot 10^{-4}$
- Cesium	$2 \cdot 10^{-4}$	$3 \cdot 10^{-2}$
- Tellurium	$3 \cdot 10^{-3}$	$1 \cdot 10^{-3}$
	of core inventory	of core inventory
Mass Flow	4.5 kg/sec	13.6 kg/sec
Operating Temperature	170°C	190°C
Design Pressure	6.3 bar abs.	11 bar abs.

Within these design criteria several types of Vent Filters are being installed in the FRG.

XIII. GERMAN ATMOSPHERIC CONTAINMENT VENTING (used mainly on PWRs)

The initial installations in the FRG were based on KfK LAF II developed stainsteel demister elements used as prefilters followed by a droplet separator and a HEPA grade stainless steel filter. (27) (68) (69) (82)

In the direction of flow, the density and fiber diameter of these elements is as follows:

a)	2.5 kg/m^2	30 um	Prefilter Section
	1.5 kg/m^2	22 um	
	1.5 kg/m^2	12 um	
	1.5 kg/m^2	8 um	
b)	inertial droplet separator		
c)	4.5 kg/m^2	2 um	HEPA Section

The filter systems using the atmospheric containment venting are located downstream of a control valve or a restricting orifice and only a rupture disk is located inside the containment.

A typical installation flow diagram is shown on Figure 16. System sizes up to 30,000 m^3/h have been constructed.

Those units were tested with submicron uranium aerosol and the reported DF is 15,000 - 25,000.

The typical dimensions of a 35 m^3 face area filter are:

Length	-	7 m
Width	-	2 m
Height	-	5.3 m

Because these units take advantage of the superheat generated by the full pressure differential between the containment pressure and the ambient environment, there is very limited condensation in the atmospheric venting system and it can be considered a dry filter.

XIV. GERMAN "SLIDING PRESSURE" FILTER UNITS (PWR)

The desire to reduce the size of the vent filter system and to hold the vent flow to more constant conditions, the throttle orifice was moved downstream of the filter unit in the design of the so call "sliding pressure" aerosol filters.

The filter is inside a stainless steel pressure vessel with a design pressure of 11 bar abs. and on operating pressure of 6.3 bar max.

The vessel height is 5 m and the diameter is 3 m.

A typical unit is shown on Figure 17.

The internal filter structure is similar in constituents to the atmospheric filter units, i.e., various fiber diameter stainless steel mats.

Pilot testing of the units showed a DF for submicron aerosols in excess of 1000.

The so called sliding pressure units are considered both for inside and outside the containment installation, as shown on Figures 18 & 19.

The latest installation schedule for FRG PWR containment vent filters is shown on Table 6.

XV. GERMAN MULTIVENTURI SCRUBBER - AEROSOL FILTER COMBINATION (BWR)

The filter unit is an 8 m high, 4 m diameter stainless steel 11 bar abs. design pressure vessel containing an array of multiventuri scrubbers. The venturi scrubbers are submerged in an 0.5% sodium hydroxide and 0.1% sodium thiosulfate solution. The venturi scrubber operation is as described for the Swedish MVSS units. The venturi array is installed in a star pattern and the top of the vessel holds the hexagon shaped bank of stainless steel demister, HEPA filter units of similar material as described for the atmospheric filters. The unit is shown on Figure 20. The scrubber solution volume is 27 m^3.

The installation flow diagram is shown on Figure 21. The unit is normally isolated and nitrogen filled. There is also a check valve to prevent backflow of oxygen which could occur if the containment cools.

The installation schedule for FRG BWRs is shown on Table 7.

XVI. THE CANADIAN DESIGN (21) (24)

The Canadian Emergency Filtered Air Discharge (EFAD) system design has been introduced first at the Darlington NGS of Ontario Hydro which has four CANDU PHW 850 MWe reactors.

It is designed to protect the plant containment from short term effects of LOCA and some scenarios of the early phases of a severe accident.

The design is more sophisticated than the filtered vented confinement of SRP and Hanford N reactor because it is fully redundant and has provisions for partial or full recycle of the vented air, but it is not a full fledged FVC.

The EFAD operates in conjunction with the Vacuum Building and is shown on Figures 22 & 23.

The concrete construction vacuum building is linked by pipework to the general plant containment covering not just the reactors, but also a fuel loading/unloading tunnel liking the four units. It has an internal volume of 10^5 m^3, equivalent to the plant containment internal volume. Under normal operation:

- The vacuum building is isolated from the plant containment by a valve in the closed position;
- Slight underpressure is maintained in the plant containment;
- A "vacuum" is maintained in the vacuum building (pressure 7 X 10^3 Pa).

In the event of a LOCA in any one of the reactors, the release of steam into the plant containment mechanically and automatically closes the containment isolating valves, the scram induces a vacuum in the vacuum building and isolates it from the external atmosphere, the valve linking the plant containment to the vacuum building opens and water spraying in the vacuum building cools and condenses the steam. In the event of a full clean break in the heavy water inlet pipe in the reactor vessel, this system enables the pressure in the plant containment to be brought below atmospheric pressure within less than one minute and kept subatmospheric for several hours.

The emergency filtered air discharge and containment system (EFAD) is used in a post-accident situation to maintain depressurization of the containment. It includes redundant air cleaning systems isolated by protection screens. It may take in air either from the general plant building itself or from a vacuum pump outlet of the vacuum building. The air filtered by the system may be directed to the plant stack or partly or totally recycled. Each air cleaning system consists of:

- A moisture separator to prevent downstream filters from being damaged;
- A conventional glass fiber HEPA filter retaining particles with a minimum efficiency of 99.97 percent for particles of diameter less than 0.3 um;
- An impregnated activated carbon iodine adsorber;
- A second HEPA filter;
- An air moving device.

Whereas the operation of the vacuum building is initially passive, the operation of the emergency filtered air discharge and containment system presupposes non-automatic action and the availability of power sources. The system has been designed in the context of a design basis accident and its suitability for handling accidents beyond the design basis has not been evaluated (in particular medium and long-term problems of water condensation and storage, inflammable gas combustion, efficiency and life of air cleaning component under severe accident loads). The long term pressure expectation following LOCA is shown on Figure 24.

XVII. THE FINNISH POSITION (31)

The Finnish regulations do not specifically call out methodology of severe accident mitigation.

In 1987 STUK published a more detailed Guide YVL 2.2 on the transient and accident analyses including severe accidents. The criteria for approval of the analyses concerning severe reactor accident, include i.a.

- burning or explosion of mixture of gases which can endanger containment integrity shall be highly unlikely
- containment integrity may also not be jeopardized by jet forces of missiles
- the long-term cooling of a damaged reactor's residues at the bottom of the containment shall be sufficiently effective in order to restrict the release of radioactive isotopes into the containment air space and to prevent reactor residues from penetrating the containment bottom
- the release of radioactive substances resulting from a severe reactor accident shall not be so extensive as to cause acute radiation damages for the inhabitants of the adjoining area
- for long-term radiation effects it shall be proved that a cesium release does not exceed 0.1% of the cesium inventory in the reactor and that
- other nuclides do not cause a higher long-term exposure than cesium release.

The two BWR units will be equipped with both filtered and unfiltered vents similar to the Swedish BWTs, however the actual filtering system selection has not been performed yet. It is scheduled for 1989.

The analysis of the Russian VVER (PWR) reactors located in a Westinghouse ice condenser resulted in an external cooling of the freestanding steel containment by a spray system rather than filtered venting. (83)

External cooling by a spray system has been chosen as a concept for further design and more detailed analysis because of the following reasons:

- The steel containment's strength against subatmospheric conditions is poor. A venting would endanger the containment integrity through venting of non-condensibles (= air). To avoid this risk, a complicated vacuum-breaker system should be installed.
- Irrespective of what the overpressure protection system is, the long-term residual heat removal shall be guaranteed.

Qualification of the existing containment internal spray system for severe reactor accident conditions would be difficult. Thus, in case of venting an entirely new decay heat removal system should be installed. The external spray system fulfills also this task.

- The external spray system prevents the radioactive releases from spreading to the environment and retains the radioactivity inside the containment.

- The required cooling rate is low, about 3 MW. This means, that the spray mass flow needed is relatively low and the maximum pressure in the containment will be limited to 1.7 - 1.9 bars, i.e. near design values.

VIII. THE ITALIAN POSITION (30) (84)

There are no currently installed FVC units. However, an extensive analysis is being performed on the utilization of the existing engineered safety systems and on gas processing systems in case of severe accidents. These are the reactor building purge system,
the standby gas treatment system, and
the augmented off-gas system.

This evaluation is unique in that the modification of existing air cleaning systems could be used for several accident sequences by themselves and in some cases the utilization of existing air cleaning systems downstream of a high thermal and aerosol load capable vent filters can significantly improve the overall decontamination factor of the combined systems.

XIX. THE U.S. POSITION (22)

While there are no installed FVC on US power reactors, there are currently several designs under consideration. These include gravel/sand bed filters, stainless steel media filters similar to German design, water filled scrubber heat sinks, (in some cases followed by conventional HEPA filters and iodine adsorbers).

Both the Chernobyl related political events and the latest revision of the source term indicate that for certain reactor types, there could be an FVC installed.

New sources indicate that LILCO is considering the use of the Swedish Barsebaeck Filtra design for its Shoreham plant and other utilities are evaluating filtered vented containment options to ease licensing problems.

The USNRC at one time postulated the use of FVC modifications for Mark 1 BWR containments but particularly based on the overcomplicated theoretically analyses in the Reactor Risk Reference Document (NUREG 1150) the benefit of such backfits is not considered cost effective by the NRC at the present time.

For reference several of the conceptual designs are discussed based on USNRC sponsored work at Sandia National Laboratory (85).

The Reactor Safety Study (8) determined that containment failure due to overpressurization represents the largest contributor to reactor risks. Subsequent studies (20) (86) (87) have reinforced the idea that containment venting could reduce reactor risk by reducing the probability of containment overpressurization. In April 1979, the USNRC initiated a program at Sandia National Laboratories to investigate filtered-vented containment concepts for light water reactors. That program has the following goals:

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

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

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

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

5. Quantitative analysis of values versus impacts.

Sandia's work on filtered-vented containment system design, development and evaluation during the first year of the program are described in Ref. (88) (89) (90) and summarized here. The following includes the description of the baseline pressurized (PWR) and boiling water reactors (BWR) analyzed, a summary of key accident scenarios and feasible venting strategies to mitigate them and a discussion of filtered-vented containment design options.

Baseline Reactors

The NRC sponsored Sandia study includes an investigation of the filtered-vented containment system design concepts for the following primary containment types: (1) large-dry pressurized water reactor (PWR) containment, (2) Mark I boiling water reactor (BWR) containment, (3) ice condenser PWR containment and (4) Mark III BWR containment. Preliminary analysis for category (1) and (2) above have been performed. Some characteristics of the large-dry PWR containment and the Mark I BWR containment are presented in Table 8.

The accidents selected for study represent best estimates of those accidents from the RSS that dominate risk to the public for each reactor containment type. Also included are accidents that may not dominate risk but provide and unusual challenge to the filtered-vented strategies capable of mitigating the effects of the accident.

PWR Accident Scenarios

A brief description of the accident scenarios selected from RSS for application to the PWR containment is given in Table 9.

Calculations of containment pressure vs time were made for the four listed accidents using the MARCH computer code. (91) The results of those calculations are presented in Figure 25.

The calculations indicate that a large pressure spike could occur if melt-through of the reactor vessel were to happen. The cause of the containment pressure spike varies, but combinations of the following are responsible:

1. Steam release from the primary system to the containment when the reactor vessel fails at high pressure (accidents TMLB' and S_2D).
2. Rapid steam formation caused by molten core interaction with water existing in the cavity at the time of reactor vessel failure (accident AB).
3. Rapid steam formation caused by flashing of some of the residual water in the primary loops when the reactor vessel fails, and by dumping the remainder of this residual water onto the molten core in the cavity (accidents TMLB' and S_2D).
4. Rapid steam formation caused by discharge of accumulator water at the time of reactor vessel failure and interaction of this water with the molten core in the cavity (accidents TMLB' and S_2D).
5. Deflagration of the hydrogen produced by Zircaloy-steam reaction, triggered by the interaction of the molten core with the concrete in the cavity (accidents AB and S_2D).

The magnitude and duration of the spike are subject to assumptions regarding the nature of core material interactions with water which may prove to be conservative. Further experiments are planned to investigate the phenomenology of steam spikes.

PWR Vent Strategy 1

In this strategy, containment internal pressure is vented at a low flow rate (24000 m^3/hr) when the containment pressure exceeds 6 bar. When the internal pressure falls below 6 bar the control valve would close. In this way, the containment internal pressure would be maintained at or below the containment design pressure. The advantages of this strategy are its simplicity and the minimum potential for adverse effects on engineered safety features (ESFs). This concept is similar to the German position for PWRs.

PWR Vent Strategy 2

Deliberate depressurization of the primary loop after most of the water has boiled off could be helpful during accidents initiated by transients or during small beak loss of coolant accidents (LOCAs). Deliberate depressurization of the reactor primary loop would require either automatic controls or operator judgement. This vent strategy has the disadvantage that an actuation error could cause a LOCA that otherwise would not have happened.

PWR Strategy 3

Anticipatory containment depressurization could prevent containment overpressurization by forecasting a core melt and venting containment in advance. During the interval between initiation of core melt and the failure of the reactor vessel lower head there is time to reduce the containment internal pressure to a level where subsequent pressure spikes would not exceed the containment failure pressure. Anticipatory venting could also reduce the magnitude of a hydrogen burn by removing hydrogen and oxygen from the containment.

Conditions used to initiate anticipatory venting might be; sustained low reactor vessel water level, high containment radiation levels, high reactor vessel temperature and high possibility of emergency core cooling (ECC) failure. Due to recirculation pump cavitation it might be necessary to place a booster pimp into the ECC recirculation inlet to meet the ECC vacuum breakers into the present containment boundary and to limit containment spray operation in order to counteract the possibility of a severe containment vacuum.

Anticipatory containment venting introduces greater potential for unnecessary radioactive release than other strategies because some accidents with incipient core melt might not threaten containment integrity. The anticipatory containment vent parameters (high radiation levels, high reactor pressure and temperature and low reactor water) might indicate incipient core melt, such as at Three Mile Island, and might signal the containment vent to open, whereas a full-scale core melting may not develop and no threat to the containment may occur. However, it is felt that the magnitude of such unnecessary radioactivity releases via the filtered-vented containment system should be small compared with uncontrolled release via a ruptured containment.

Figure 26 shows the effect on containment pressure vs time of implementing PWR vent strategy 2 and 3 on the TMLB' accident. It can be seen that the peak pressure is reduced below the containment failure pressure.

BWR Accident Scenarios

Four accidents were selected for the BWR from the RSS as posing moderate to high risk to the public should the primary containment fail. Those four accidents are described in Table 10.

The risk dominating accident sequences in the BWR (TC and TW) lead to primary containment overpressurization while the core is partially covered with water and hence not melted. Thus a primary requirement of the BWR filtered-vented containment system would be the prevention of containment overpressurization without degradation of the ECC function.

For the accident TQUV and AE where core melt down precedes containment overpressure a pressure spike occurs when the reactor vessel fails. The sharp pressure is due to:

1. Hydrogen release from the reactor vessel to the containment. this rapid containment pressurization can be prevented by the use of the automatic depressurization system (ADS).

2. Hydrogen formation caused by zirconium-steam reaction when the reactor vessel fails and the molten core falls into water.

Figure 27 presents the pressure vs time history of the four BWR accidents (TC, TW, TQUV, and AE).

BWR Vent Strategy 1

This strategy (low-volume containment pressure relief) is similar to PWR vent strategy 1 and requires approximately the same flow rate (24000 m^3/hr). Venting from the wetwell allows the suppression pool to be used as a filter for the drywell environment.

This low flow rate option would prevent accidents TW and TQUV from overpressuring the containment, but would not be adequate for AE and TC. Operation of this vent strategy during an accident with a failed suppression pool cooling system would result in a reduction of the NPSH below the design basis for the low pressure coolant recirculation (LPCR) pumps. Booster pumps could be incorporated in the LPCR system in order to increase the NPSH and prevent cavitation of the LPCR pumps. The LPCR pump inlet could be diverted from the suppression pool to another source (via existing cross-overs) such as the high pressure service water system (HPSW).

BWR Vent Strategy 2

During the TC accident it is possible to continue high pressure coolant injection (HPCI) and prevent a total core melt down as long as water is available. Containment venting with mass flow rate equal to the rate of steam formation (as a result of HPCI) would create a steady flow process into the primary and out to the suppression pool then into the wetwell and out the containment vent.

This steady state situation would be achieved with a vent rate of 240,000 m^3/hr at the containment internal pressure of 6.8 bar.

Success for this venting strategy during the TC accident depends upon the restoration of the reactor protection system within 3 hours or the availability of an external water source (such as the high pressure service water) to supply the HPCI system indefinitely.

BWR Vent Strategy 3

This strategy (anticipatory venting) is similar to the PWR vent strategy 3. It would be effective in preventing drywell failure due to pressure spikes except when the suppression pool is saturated at the onset of wetwell venting. Suppression pool saturation would slow containment depressurization because of boiling from the pool.

Filtered-Vented Containment System Designs

PWR Design Options

Five filtered, atmospheric vented design options and a filtered, contained design option for the PWR under study were formulated. There options represent successively higher levels of fission product removal from the containment vent gas stream.

PWR vent-filter design option 1 is shown schematically in Figure 28. This is the most simple of all options in that it consists of a gravel chamber as the only filter component. The gas stream is vented through a valve manifold in an existing penetration in the concrete containment vessel into a vent line of approximately 1.0 m diameter. The filter element is a buried gravel bed 20 m long X 10 m deep X 40 m wide for the low flow (24000 m^3/hr) vent strategy. The dimensions of the bed would be proportionally larger to accommodate the vent strategy 3 (150,000 m^3/hr). The filtered noncondensible gas stream would then discharge to the atmosphere via a tall stack. Recent experiments with crushed gravel suggest that gravel beds of sufficient height will remove submicron particles without excessive pressure drop. (92) The pressure drop across the bed is designed to be less than 0.7 bar.

The claimed advantages of option 1 are its simplicity, low cost, and that it requires no electric power, disadvantages are the limited amount of performance data with large scale systems and an unknown decontamination factor that is sensitive to varying particle size and gas velocity.

Vent-filter design option 2 is based on a system being developed at Hanford Engineering Development Laboratory. (73) This option is shown in Figure 29 and consists of a gravel bed submerged in an alkaline water pool. This option has the capability to condense steam, which option 1 has only to a limited degree. Estimated fission product removal efficiencies are: 98% particles, 98% I_2, 50% CH_3I, 0% Xe and 0% Kr. In this option a provision for recirculation of the filtered containment exhaust and long term heat removal from the suppression pool has been made. There is even less large scale data available for this option.

Design option 3 is shown schematically in Figure 30 in both the passive and recirculation mode. This option consists of a BWR type suppression pool shown in figure 31 and a sand-gravel filter shown in Figure 32. Suppression pools are a tested and proven method of cooling and condensing gas streams. Suppression pools require less volume than crushed rock for the same heat load and provide a solution to a long term heat exchangers in the wetwell. In this option the toroidal shell has a volume of 8500 m^3 of which 50% is chemically treated water. The 4250 m^3 of water will condense all the steam generated during the TMLB', AB and S_2D accidents. The 4250 m^3 air space allows for the condensate storage. The entire torus and all piping is located below grade in a concrete lined pit. in order to maintain a 1.3 m submergence over the downcomer outlets, a spillway is located to allow for condensate carry-over into the air space. The pressure drop across the suppression pool is designed to be 0.13 bar. This pressure drop should present no problems because the driving pressure (containment internal pressure) will be on the order of 5.0 bar. The piping from the containment to the suppression pool would have to be capable of transmitting a peak flow rate of 420,000 m^3/hr. The existing purge penetrations would satisfy these requirements.

The sand-gravel filter shown in Figure 32 consists of a large buried concrete vault filled with alternate layers of gravel and sand. The approximate dimensions of the vault are 36 m long by 36 m wide by 15 m deep. a drain network and integral condensate storage tank are provided to store the contaminated condensate. The structure was designed to handle a flow of 84000 m^3/hr at a pressure drop of 0.04 bar maximum. A space is provided in the chamber to accommodate a hydrogen ignition source. The so-called hydrogen burn chamber/space is overlaid by a gravel layer; this layer serves as a flame arrestor and heat sink for the combustion gases. Total fission product removal efficiencies for option 3 are estimated to be: 99.98% particles, 98% I_2, 50% Ch_3I, 0% Xe, and 0% Kr.

Design option 4 consists of the toroidal suppression pool and sand-gravel filter of option 3 plus a zeolite-charcoal filter consists of a water shaped tank about 5 m thick and 12 m in diameter. The wafer is fabricated of 304 stainless steel and is gas/water tight. the water is filled with a top layer of 10 cm layer of triethylenediamine (TEDA) impregnated charcoal. These two layers are followed by a layer of HEPA filters to trap charcoal and other particulate. The layers of filter media could be separated by packed fiber. The zeolite carbon component is designed to be submerged in a 250 m^3 water tank. The water tank would provide passive cooling of the fission product decay heat (during TMLB' accident) from the wafer. The estimated total fission product removal efficiencies for option 4 are: 99.98% particles, 99.95% I_2, 99.90% CH_3I, 0% Xe and 0% Kr.

Design option 5 is essentially the same as option 4 except xenon holdup is provided for. This requires a thick layer of adsorbent (1.7 m thick) between the TEDA charcoal and the HEPA filter trays. This option is shown schematically in Figure 34. The estimated total fission product removal efficiencies for option 5 are: 99.98% particles, 99.98% I_2, 99.98% CH_3I, 98% Xe and 10% Kr.

Design option 6 is a completely contained (no vent to the atmosphere) system. This option is presented in Figure 36. The main features of the system include a toroidal suppression pool and a hydrogen burning area plus a large (30,000 m^3) second containment building. At this volume, the design pressure of the second containment would have to be about 2.8 bar. The hydrogen carried over from the first containment building would have to be the second containment due to hydrogen burning. This option has the potential of holding up all fission products from the damaged reactor. The main disadvantage of this option is the high cost of the second containment building and the difficulty of finding space for this size structure at many existing reactor sites.

BWR Design Option

The design options for the baseline BWR are similar to the PWR options except there is no need of a suppression pool since the BWR Mark I has a suppression pool in the primary containment. The option 1 gravel bed would be somewhat larger because it is designed to the heat loads of accident TC.

Consequence Evaluation of the Design Options

An evaluation of the public health consequences using the CORRAL and CRAC computer codes for the TMLB' accident was made. The calculations were made by using the RSS fission product transport and consequence models and the fission product removal efficiencies of the individual design options. Furthermore, it was assumed that the containment vessel would be completely failed if there were no FVCS and the filtered-vented containment design options would operate at their predicted efficiencies and prevent containment failure. Weather and population profiles specific to a densely populated Northeast site were used. The results of those calculations are shown in Figure 36 and indicate that FVC systems can lower risk for several accident scenarios.

The U.S. position regarding post severe accident filtered containment venting is ambiguous. The USNRC funded design concept evaluations were somewhat removed from technological realism. The risks of inadvertent venting, the reisolatability of the containment after venting and the evaluation of any and all alternatives is occupying the regulatory realm.

There are no industry generated standards (ANSI, ANS, ASME) for component or system requirements. (Not only for containment vent filter systems but for any ventilation, air cleaning, damper, duct, etc. components under severe accident conditions.)

There are no clear regulatory positions regarding filtered containment venting even though one utility (LILCO) proposed a Barsebaeck Filtra type system with battery power for 48 hour operation capability for the isolation valve operation. Other utilities have also considered the use of one or another European based vent filter systems.

There is a current research program, the Advanced Containment Experiments (ACE) program, managed by EPRI. The program objectives are:

1) Provide a comparative experimental basis for various filtration techniques.

2) Provide data for modeling the transport of radioiodine species;

3) investigate fission product releases from core concrete interactions; and

4) develop and validate computer codes.

The work of primary interest to filtered venting is that related to providing experimental data for various filtration techniques. This work will be conducted at the Hanford Engineering Laboratory. The data are then to be used to compare the merits of several filter concepts. Specifically, efficiencies of the following filter concepts will be evaluated experimentally:

1) dry sand/gravel beds;

2) deep pool scrubbers;

3) submerged gravel scrubbers;

4) combinations of pools and gravel scrubbers; and

5) combined Venturi pools.

The first phase of the work is to consist of 10 tests using five filter types at two water temperatures. Data on efficiency versus particle size will be collected. Aerosols of CsOH, CsI and MnO in a gas flow of about 0.1 M^3/S with steam heating to simulate decay heat are to be used. The second phase consists of separate effects tests to evaluate the effects on filter efficiency of:

1) pool depth;

2) decay heat;

3) the ratio of noncondensible gas to steam;

4) volatile iodine species; and

5) design specific parameters.

From the program outline, it appears that the project is prejudiced toward liquid scrubber included designs.

In general, it is currently unpredictable which of the following three events will take place in the U.S.

1) Installation of a containment vent filter.

2) Generation of the research results on how to build such a filter.

3) The regulatory position on how to rebuild the filter.

XX. THE SWISS POSITION (32)

The Swiss Federal Nuclear Safety Inspectorate after evaluating the severe accident consequences established that filtered venting prevents overpressure failure of the containment in some of the severe accident scenarios. Based on the scenarios it is required that an FVC be

1) Capable to vent the steam production corresponding to the decay heat level of 1.0% of the thermal reactor power. For PWRs with large dry containment this may be reduced to 0.5% if the reduced venting capacity is adequate to handle accident sequences with slow pressure buildup.

2) The capacity of the FVC is specified for the pressure level maintenance.

   non inerted containment
      the lesser of 1/2 failure pressure or the design pressure (H_2 combustion to be considered)

   inerted containment
      the lesser of 2/3 failure pressure or the test pressure

3) To prevent long-term land contamination a

      DF of 1000 for aerosols

4) To limit thyroid doses a DF of 100 for elementary iodine.

The retention factors of 3 and 4 have to be demonstrated both at 10% and 100% of the system design flow rate.

The additional requirements and criteria to be considered for the design are:

1) <u>Design and maintenance</u>

   a) The design specifications of the containment have to be applied to the venting system up to and including the second isolation valve.

   b) The remaining sections for the venting system ahead of an eventual throttling device, have to be designed according to safety class 4 (Swiss design rule R-06) and SSe (earthquake). The design pressure should be 1.5 times the nominal relief pressure specified above.

   c) Conservative consideration of the temperatures to be expected during operation, the possibility of a local accumulation of hydrogen gas has to be considered.

   d) The system should operate with variable pressure in the containment and with variable flow rates.

   e) For BWR Mark I containments, the system should still work after flooding the drywell and filling the wetwell.

   f) The flow rate through the device should be adjustable.

   g) The filtered steam/air mixture has to be vented to the atmosphere via the plant stack or another suitable line.

h) Possibility of the periodical test and check of the equipment.

i) Measures to prevent degradation of the filter equipment during standby.

j) With respect to the expected maximal load on the filters consisting of radioactive and non-radioactive material, an amount of 150 kg of aerosols is postulated to go into the vent system. A large part of this aerosol mass consists of inactive material.

2) Radiation protection and handling of radioactive materials

a) Manual operation of the vent system should not result in intolerable doses to the plant personnel (planning value for emergency conditions: individual dose smaller than 10 rem).

b) At the boundary of the NPP, direct radiation from accumulated radioactivity in the venting system should not cause a significant increase of the accident caused dose rate by direct radiation from the remaining parts of the NPP. The venting system has to be provided with an appropriate shielding for the protection of the plant personnel and the public if necessary. For sites with more than one plant this statement has to be applied in the same sense with respect to the non-affected plants.

c) Handling of the retained materials after the accident, prevention of an eventual long-term re-release of that material.

3) Operation of the filtered venting system

a) The system should operate in a flexible and predominantly passive manner.

b) Operation of the system should be possible from the control room, but also manually from nearby the system.

c) The system should be permanently ready to operate (no assembly work).

d) Instrumentation for the operation of the venting system, and a specialized instrumentation to observe and record the radioactive materials released to the environment have to be installed.

e) Possibility to replenish and change the water in case of water scrubbing devices.

f) Consideration of dynamic loads (e.g. condensation shocks).

g) Handling and safe storage of a possibly considerable amount of water which could be condensed in the device.

4) Independence of the filtered venting system, eventual implications for the existing plant

a) No detrimental influence on the normal plant operation.

b) No detrimental influence on the other safety functions and systems, especially the isolation of the containment should not be impaired (in particular with respect to design base accidents).

c) The instrumentation of the venting system has to work in stand alone mode for 100 hours.

d) The availability of the system must be assured even in case of failure of electric equipment not pertaining to the vent system.

The Swiss position is highly commendable even if one does not agree with the specific individual requirements in that, the decision was clearly made and the designer or operators know what to design for.

XXI. REVIEW OF EXISTING DATA AND OF VARIOUS DESIGNS

While there is considerable amount of data available for typical components of potential FVC systems, most of the data were obtained either by operating air cleaning systems for various fuel reprocessing facilities or from short duration laboratory or pilot plant simulations. Very little data is available regarding the long term behavior of the postulated systems under post severe accident venting conditions.

Hydrogen Burn Effects

Several of the proposed designs provide empty spaces for hydrogen burn expansion (20) (22) (90) while experimental data indicates (93) (94) that packing (large surface to volume ratios) greatly decreases the possibility of the hydrogen ignition and not only the effects of the detonation. Another factor that is also ignored regarding hydrogen detonation effects is that the design stress of materials to contain detonations is considerably higher than normal pressure retention applications because of the extremely short pressure exposure time., Values of design stress approximately 160 percent of the normal curve have been found completely satisfactory. (95) The effects of FVC inerting and/or the use of diluents needs also to be better understood. Current data shows that CO_2 may be the best inert diluent to prevent hydrogen explosion rather than nitrogen. (95)

The minimal effect of hydrogen deflagration or detonation has been confirmed only for the Swedish Filtra design. (45) (98) These tests were performed with H_2-Air mixtures, and the diluent effect of steam which can also slow down the hydrogen burn was not considered.

While deep gravel beds can be expected to well withstand the effects of hydrogen detonation, very little is known of the shock wave effects on small particle size sand filter sections which are unrestrained. The movement of the sand particles may release trapped activity and/or may result in partial channeling of wet sand beds.

Review of hydrogen detonation in BWR noble gas delay beds containing activated carbon has shown no particle degradation but any particulate activity transfer to the gas phase was not observable due to masking effects of the partitioning of the adsorbed noble gas activity into the gas phase.

The temperature rise in a packed bed flow system due to hydrogen burn is not significant. As an example, 20,000 kg carbon bed temperature rose only from 57°F to 85°F due to hydrogen detonation within the carbon. (94)

Radioiodine Removal

The design experiments performed for the FILTRA project indicated the following iodine retention effects on packed granular beds. (96)

A) 80% overall removal efficiency at $\sim 3X10^{-5}$ g/min input
65% overall removal efficiency at $\sim 1X10^{-4}$ g/min input
17% overall removal efficiency at $\sim 2X10^{-3}$ g/min input

at 98-100% RH, at 2.4 cm/sec velocity through a 1500 mm sand bed, under equilibrium conditions (after breakthrough of iodine occurred).

B) Changing the velocity from 2.4 cm/sec to 0.6 cm/sec at $2X10^{-3}$ g/min input increased the iodine removal efficiency from 17% to 34%.

C) Increase of relative humidity from 30% RH to ~100% RH, under identical conditions, decreased iodine removal efficiency 70% to 60%.

D) Increase of temperature from 21°C to 100°C, under identical conditions, decreased iodine removal efficiency from a stable 70% to a still decreasing 50%. (No equilibrium achieved.)

E) A threefold increase in sand diameter from 1.4 X 2.0 mm to 4.2 X 5.6 mm decreased iodine removal efficiency, under identical conditions from 65% to 40%.

Deposited iodine repartitioning releases of an initially wet but later dried out sand bed have not been performed. However, under wet sand conditions, extensive conversion to water soluble iodide forms were observed.

Only relatively short time experiments were reported for the removal efficiency test of packed gravel bed used at FILTRA (~5 hours) (96) while some other data indicates that iodine deposited in particulate form with "aging" can be transformed to gas (or vapor) phase form and migrate in the direction of flow. Table 11 shows such iodine form transformation over 96 hours. (97) Therefore, iodine removal efficiency tests should be conducted for the proposed duration time of the venting and not be abbreviated in model experiments. For iodine removal efficiency of sand beds there is only one incompletely described experiment available in the literature (other than the FILTRA studies) but which indicates only limited iodine removal efficiency both in the wetted and in the unwetted section of the sand bed. (98)

In the absence of better gravel/sand bed radioiodine decontamination data, the use of known, well established radioiodine adsorbents is necessary for the removal and retention of vapor phase or potentially partitioning into the vapor phase after initial deposition radioiodine forms.

When using adsorbents which could be carbon based or silver treated non-carbonaceous materials, several critical parameters need to be considered:

a) the stability of the impregnant for methyl iodide under the temperature, flow conditions (99) (100) (101),

b) the prevention of reaching iodine desorption temperature (or ignition conditions for carbonaceous adsorbents) (102) (103),

c) the experience showing that silver impregnated adsorbents can act as hydrogen-oxygen recombiner catalysts and initiate hydrogen explosion (104),

d) that none of these adsorbents work well in once through condensing (>100% RH) systems. (105)

For carbonaceous adsorbent, much larger particle size (4-8 mm) than used in conventional short term adsorbers, should be considered. The use of larger particles in deeper beds permits wider distribution of iodine and thus decay heat, (106) (107) and increases the ignition temperature. (102) (103)

The heat of adsorption of water will initially superheat the adsorbents and water equilibrium is not obtained for several hours as shown by Figure 37 from the CSE experiments (66). This superheat will lower the relative humidity in the adsorbent and its initial iodine decontamination performance will be much higher than after reaching 100% RH equilibrium.

<u>Condensate Flow Direction</u>

Most of the packed granular bed filter operating experience has been obtained with upflow beds, where condensation trickles back in the direction of flow. The FILTRA design is the first major nuclear industry application of a downflow system where condensation flow precedes the condensing zone of the packed bed, and the condensate dissolved salts deposited on the gravel surface can dry out as the condensation zone proceeds further down. The re-entrainment effects of a dried out gravel bed, if investigated for the FILTRA design have not been reported.

Submicron Aerosol Generation

The venturi nozzle creates conditions identical to the well known Laskin nozzle, which is used to generate submicron aerosols. Any downstream filtration component should be evaluated on its performance under the aerosol loads generated by the multiventuri scrubbers.

Selection of Proof Test Aerosol

In several of the design experiments leading to the construction of full scale systems, the aerosol selected was water insoluble, non-hygroscopic material (quartz, iron oxide, calcium oxide, etc). It is expected that the aerosol removal efficiency of several systems will be significantly different for water soluble salts than for water insoluble particulates.

Overall System Design

The flow volume for containment pressurization is not very large (large flows are postulated only for filtering already failed containments) and, therefore, the use of the conventional air cleaning components, demisters, pressure wave resistant (possibly stainless steel construction) HEPA filters and properly designed radioiodine adsorber should be considered. The added cost of these elements on top of the heat sink and major bulk filtering sand/gravel beds is less than 5% added cost.

The automatically activated FVC flows should always have manually operated block valves to isolate venting if automatic actuation was accidental or no further venting is advantageous.

The placement of small volume non grave/sand containing system (stainless steel fiber, etc.) or even small gravel/sand bed combinations in sublevel, water cooled enclosures for easily controlled temperature operation is a possible size reduction method.

The use of external suppression pools followed by gravel/sand beds or other demister type filters needs to be evaluated.

The use of irrigated demister type (regenerable while operating) primary filtration elements requiring only external water source by gravity can also be a cost effective size reduction for FVC systems.

The conventional gravel/sand beds were designed for the long term (40+ years) low velocity operation. There is limited but promising data available showing that the packed granular beds can be operated at high velocity for short term duration, which is the case of containment venting application (47) (107).

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TABLE 1

Iodine Distribution in Loop

Location	Percent of Total Iodine Found in Loop[a] A-13	A-14	A-15	A-16	A-17
HEX	--	0.3	--	9.3	4.0
Demister	--	3.3	--	31.0	0.8
Prefilter	0.2	0.1	0.02	1.3	--
HEPA Filter	8.8	22.0	22.0	28.3	27.0
1st Charcoal Bed 1/2[b]	68.0	42.0	39.0	16.0	41.0
2/2	9.2	19.0	16.0	6.1	13.0
2nd Charcoal Bed 1/2	5.9	0.7	7.9	4.0	5.8
2/2	4.2	0.3	6.2	2.5	4.1
3rd Charcoal Bed 1/2	2.3	8.3	5.4	1.2	2.7
2/2	1.3	4.2	3.9	0.8	2.1

(a) At end of experiment

(b) 1/2 refers to first half of bed thickness; 2/2 refers to second half

TABLE 2

Cesium Distribution in Loop

Location	Percent of Total Cesium in Loop[a] A-13	A-14	A-15	A-16	A-17
HEX	---	2.5	---	11.0	5.2
Demister	---	62.5	---	51.7	52.0
Prefilter	0.7	0.02	1.5	2.9	---
HEPA Filter	99.3	35.0	98.5	34.4	42.8

(a) At end of experiment

Table 3 The tested fiber materials

Sign	Material	Fiber Ø	Porosity	Weight
class 2	AISI 302 1.4300	100 - 200 µm	-	-
class 1	AISI 302 1.4300	45 - 65 µm	-	-
WB 22	AISI 316 1.4401	22 µm	99 %	300 g/m^2
WB 12	AISI 316 1.4401	12 µm	99 %	300 g/m^2
WB 08	AISI 316 1.4401	8 µm	99 %	300 g/m^2
WB 04	AISI 316 1.4401	4 µm	99 %	300 g/m^2
WB 02	AISI 316 1.4401	2 µm	99 %	75 g/m^2
Fiberfrax Ph Fine	Al_2O_3, SiO_2	1 - 8 µm	98 %	-

Table 4 Fraction of Risk* Attributable to Various Containment Failure Modes

Containment Failure Mode	Peach Bottom (Unit 2)	Grand Gulf (Unit 1)	Calvert Cliffs (Unit 2)	Surry (Unit 1)	Sequoyah (Unit 1)	Oconee (Unit 3)
1. Direct Bypass	c**	c	c	.20	.18	.15
2. Failure to Isolate	c	c	c	c	c	c
3. Pre-Core-Melt Overpressure	.71	.98	.01	.07	c	c
4. In-Vessel Steam Explosion	c	c	c	.01	.01	c
5. Ex-Vessel "Steam Spike"	.01	c	.54	.02	c	c
6. Hydrogen Burning	c	.01	.44	.61	.76	.84
7. Long-Term Overpressure	.28	c	.01	.10	.06	c
8. Thermal Degradation	c	c	c	c	c	c
9. Basemat Penetration	c	c	c	c	c	c

*Risk measure used here is population dose per year.
**The symbol c is used to represent contributions of less than .01.

Table 5 Relative Public Consequences* for Various Accident Types and Various FVCS Filtration Efficiencies for a Highly Populated Northeastern USA Site

Accident Type	Description	FVC System Penetration**	Relative Early Fatalities	Relative Early Injuries	Relative Latent Fatalities	Relative Population Dose	Relative Property Damage
7b	Severe core damage, engineered safety features (ESFs) inoperative, containment over-pressurization	No FVCS (Containment fails)	1.0	1.0	1.0	1.0	1.0
		.1	.01	.04	.32	.39	.11
		.05	.002	.02	.19	.24	.05
		.01	.0003	.01	.05	.07	.007
2a	Severe core damage, ESFs operative, containment isolation failure	Not Applicable	.0002	.005	.07	.09	.01
9a	Severe core damage, ESFs operative, basemat melt-through	Not Applicable	<.0001	<.0001	.0002	.0003	.002

*The figures in this chart are normalized with respect to the results for accident Type 7b with no FVCS.
**Filter penetration figures apply to all fission product species except noble gases, which are assumed to be unaffected by the FVC system.

Table 6

Plant		MW_e	Planning and Design	Venting System with Sliding Pressure Filter	Venting System with Atmospheric Filter	Filter Type	Completion Piping	Completion Filter
Obrigheim,	KWO	357	YES	x, outside cont.		in Discussion	exists	1989
Stade,	KKS	662	YES		x	in Discussion	1989/90	1989/90
Biblis "A",	KWB-A	1204	YES	x, inside cont.		in Discussion	1989/90	1989/90
Biblis "B",	KWB-B	1300	YES	x, inside cont.		in Discussion	1989/90	1989/90
Unterweser,	KKU	1300	YES		x	in Discussion	1989/90	1989/90
Neckarwestheim,	GKN 1	855	YES	x, outside cont.		in Discussion	1989	1989
Grafenrheinfeld,	KKG/BAG	1300	YES	x, outside cont.		in Discussion	1989	1989
Grohnde,	KWG	1365	YES	x, outside cont.		in Discussion	1989/90	1989/90
Philippsburg,	KKP 2	1362	YES	x, outside cont.		in Discussion	1988	1989
Brokdorf,	KBR	1365	Completed		x	MF	1987	1987
Neckarwestheim,	GKN 2	1314	Completed		x	MF	1987	1987
Isar Block 2,	KKI 2	1370	Completed		x	MF	1987	1987
Emsland,	KKE	1314	Completed		x	MF	1987	1987
Mühlheim-Kärlich		1300	YES	x		in Discussion	1989/90	1989/90

MF = Metal Fibre Filter for Aerosols

Implementation Program of Containment Venting Systems for German PWRs

Table 7

GERMAN NPS WITH BWR	Würgassen (KWW)	Brunsbüttel (KKB)	Isar 1 (KKI 1)	Philippsburg 1 (KKP 1)	Krümmel (KKK)	Grundremmingen B (KRB B)	Grundremmingen C (KRB C)
CAPACITY (MWel, brutto)	672	806	907	900	1316	1310	1310
IN OPERATION SINCE	1972	1976	1977	1979	1983	1984	1984
CONTAINMENT DESIGN	69	69	69	69	69	72	72
APPLICATION FOR LICENSING	October 1987	June 1987	February 1988	February 1988	May 1987	not determined	not determined
REALIZED OR PLANNED DATE FOR COMMISSIONING OF ALTERED CONTAINMENT VENTING SYSTEM	June 1989	March 1988	March 1989	February 1989	February 1988	1989/1990	1989/1990

HEW 1988 — BWR FILTERED CONTAINMENT VENTING SYSTEM

Implementation Program

Table 8 Characteristics of the Baseline Reactors.

Reactor	PWR	BWR
Thermal Power	3025 MW	3293 MW
Containment	Steel-lined, reinforced concrete domed cylinder	Mark I drywell/wetwell, inerted to less than 5% O_2 (molar)
Containment Cooling	(1) Containment air coolers, 112 MW max. (2) Containment sprays, 20,000 ℓ/min max.	Suppression pool circulated through heat exchanger cooled by HPSW. 82 MW max. cooling
ECC Water Sources	(1) 4 accumulators pressurized to 45 bar (abs), 7.9×10^4 ℓ. (2) RWST, 1.3×10^6 ℓ.	(1) Suppression pool, 3.9×10^6 ℓ (2) CST, 5.7×10^5 ℓ.
High Pressure ECC	HPI system, injects from RWST, 4700 ℓ/min max.	HPCI system, powered by reactor steam, injects from CST or suppression pool, 19,000 ℓ/min max. Can be supplemented by RCIC.
Low Pressure ECC	LPI system, injects from RWST, recirculates from recirculation sump, 23,000 ℓ/min max.	(1) LPCI system, injects and recirculates from suppression pool, 1.5×10^5 ℓ/min max. cross tie with HPSW system allows injection of river water into reactor vessel. Some water can be diverted to containment sprays. (2) CSI system, injects from CST or suppression pool, recirculates from suppresion pool, 47,000 ℓ/min max.
Primary System Depressurization	Manual, through S/R valves. Requires ac power.	ADS. Requires dc power.

Table 9 PWR Accident Scenarios

RSS Accident Notation	PWR Accident Sequences	Estimated Contribution to Reactor Risk
TMLB'	Loss of offsite and onsite ac power for at least 3 hours. Failure of power conversion system and auxiliary feedwater system.	High
S_2D	Small LOCA with failure of ECC injection and recirculation.	High
S_2G	Small LOCA with failure of containment heat removal.	Moderate
AB	Large LOCA with loss of offsite and onsite ac power.	Small

Table 10 BWR Accident Scenarios

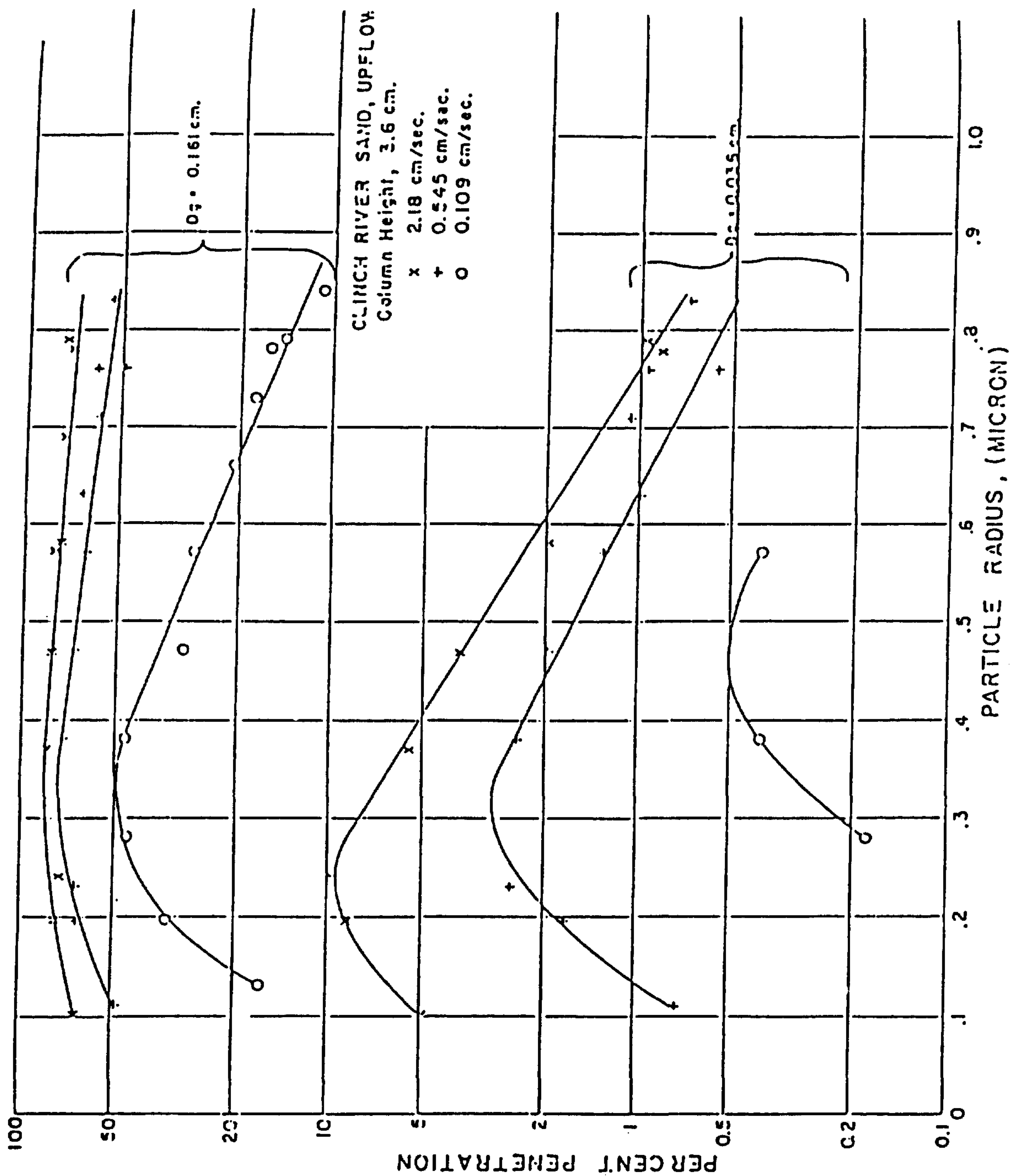
RSS Accident Notation	BWR Accident Sequences	Risk
TW	Transient initiating event with failure of suppression pool cooling.	High
TC	Transient initiating event with failure of reactor protection system.	High
TQUV	Transient initiating event with failure of feedwater and ECC availability.	Moderate
AE	Large LOCA with failure of ECC injection.	Moderate

Table 11

Distribution of ^{133}I in components of the May Packs

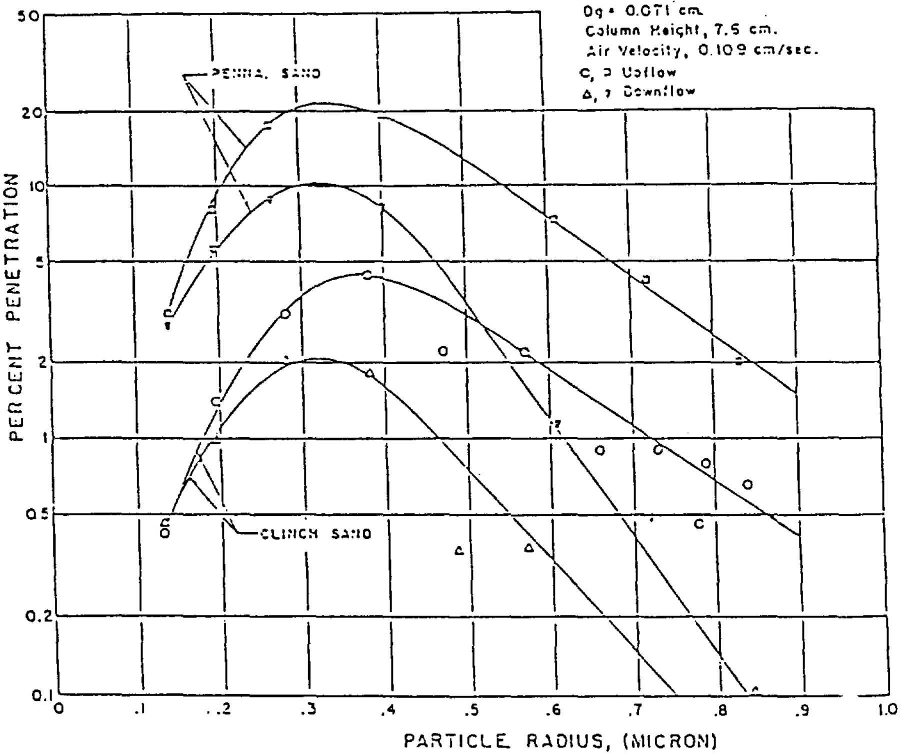
(The other iodine nuclides ^{131}I and ^{135}I behaved identically).

	% of total					
Time	14:25 15:25	15:30 16:30	17:10 18:10	18:25 19:25	19:47 20:47	4 days later
Copper screen	57	59	53	52	46	7
Glass Fibre	15	3	1	1	1	6
Charcoal paper	20	28	28	36	45	64
Charcoal column granules	8	11	17	11	7	22



-DOP aerosol penetration through Clinch River Sand.

Figure 1



·DOP aerosol penetration Pennsylvania and Clinch River Sand (20-30 mesh sand, D_3 = 0.071 cm).

Figure 2

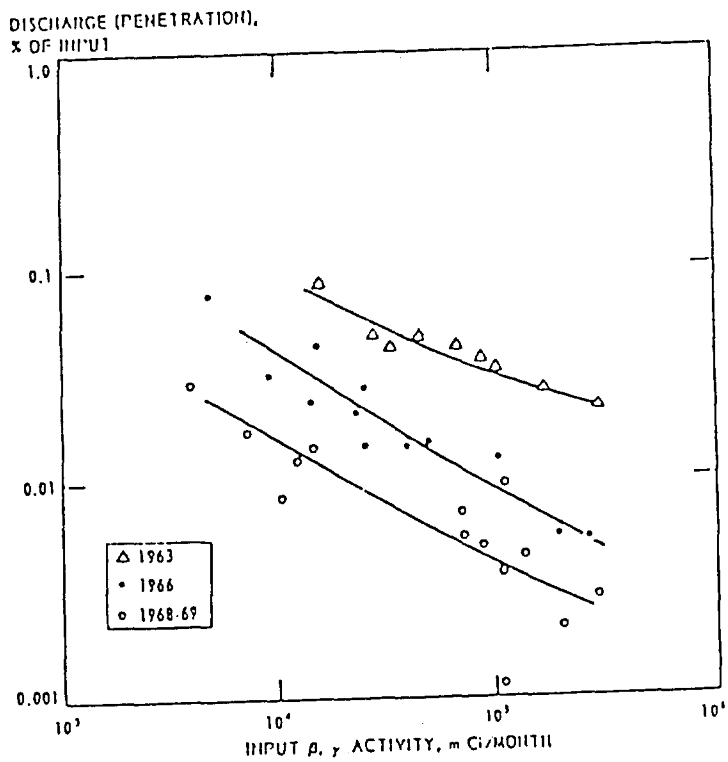


FIGURE 3. FILTER PERFORMANCE AT SRP FROM 1963 to 1969.

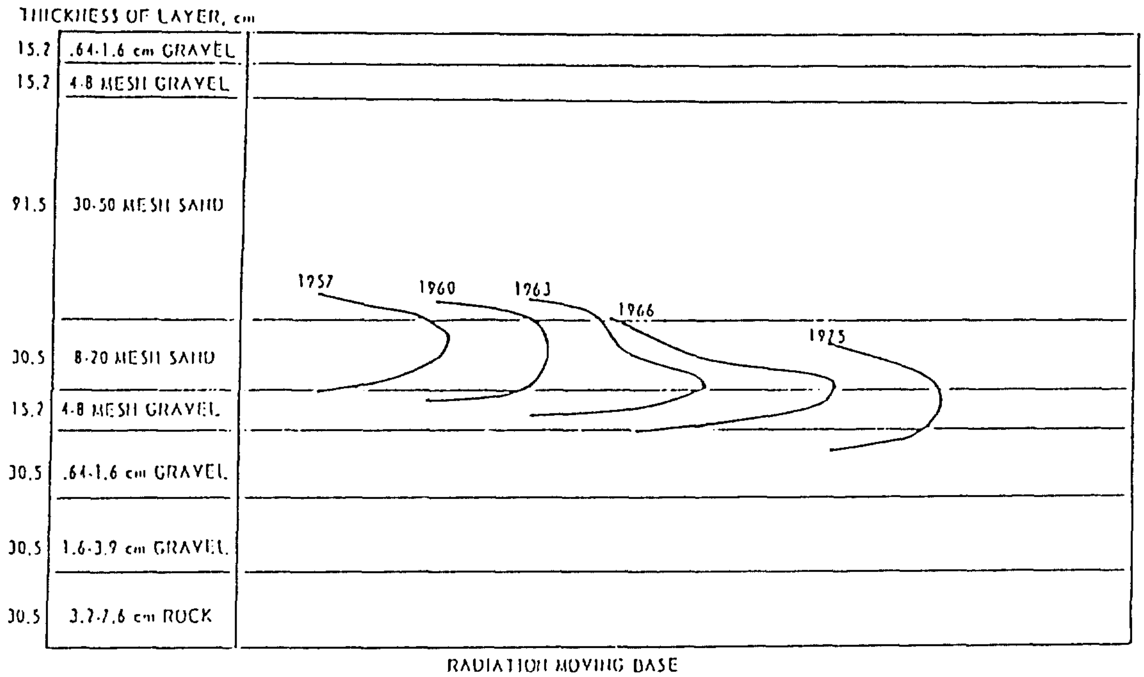


FIGURE 4 RELATIVE DEPTHS OF RADIATION PEAKS IN SAVANNAH RIVER PLANT SAND BEDS FROM 1957 to 1975.

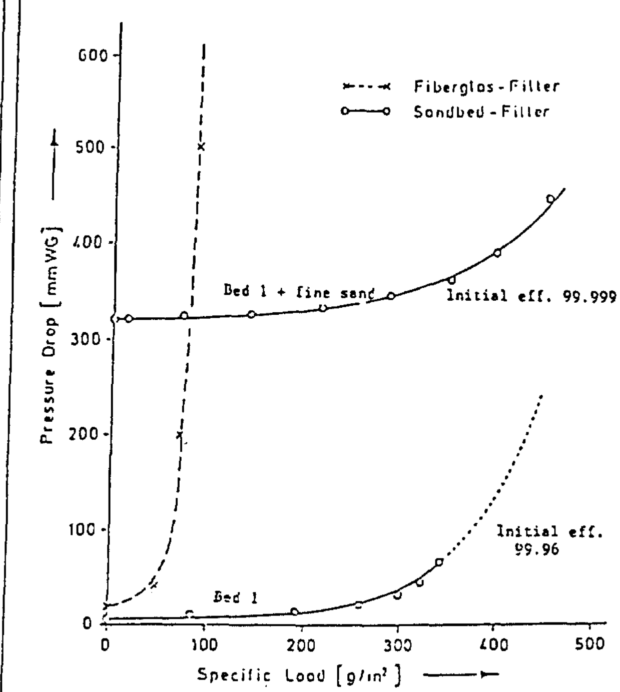


Figure 5	Pressure Drop of different Filters during Load with Sodium-oxide-Aerosols

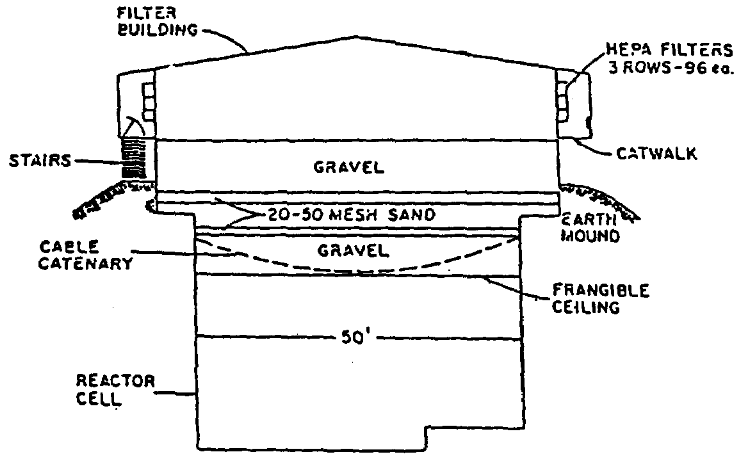


Figure 6 ZPPR Cross Section Sketch of Filtered Vented Confinement

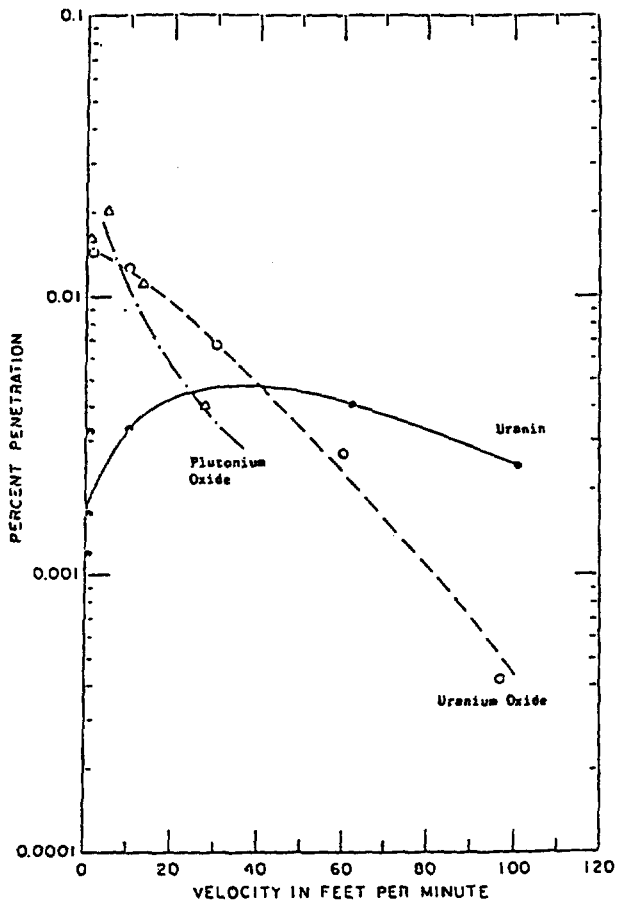


Figure 7 Composite of Uranium, Plutonium, and Uranine Penetration through nominal 30 inches of sand

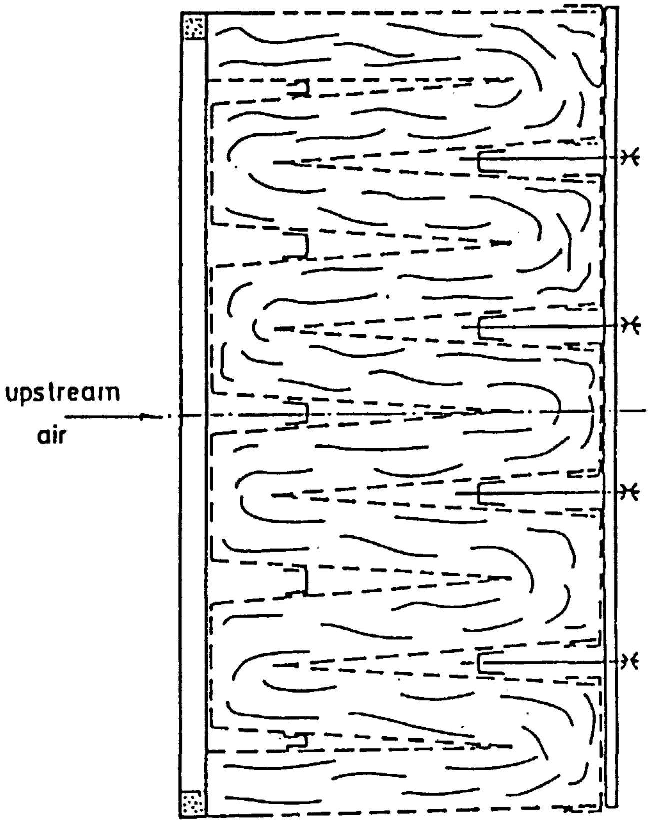


FIG. 8 FOLDED FILTER CELL TYPE FOR METAL FIBER FILTERS DEVELOPED AT KfK.

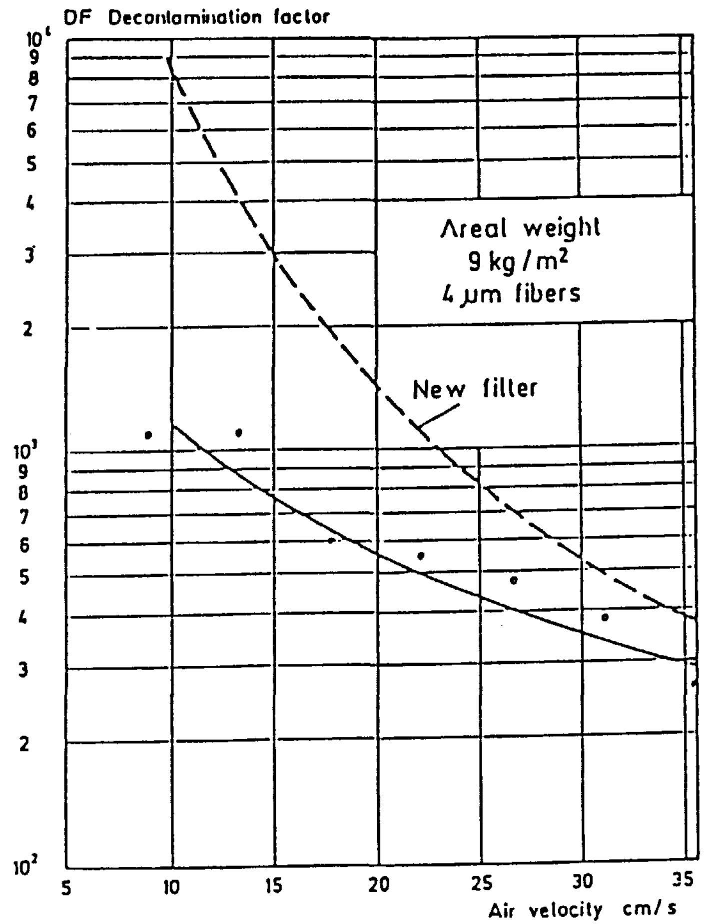


FIG. 9 DECONTAMINATION FACTOR OF A 4 µm STAINLESS STEEL FILTER AFTER A 28 kPa PRESSURE BURST AIR VELOCITY DURING PRESSURE BURST: 2 m/s

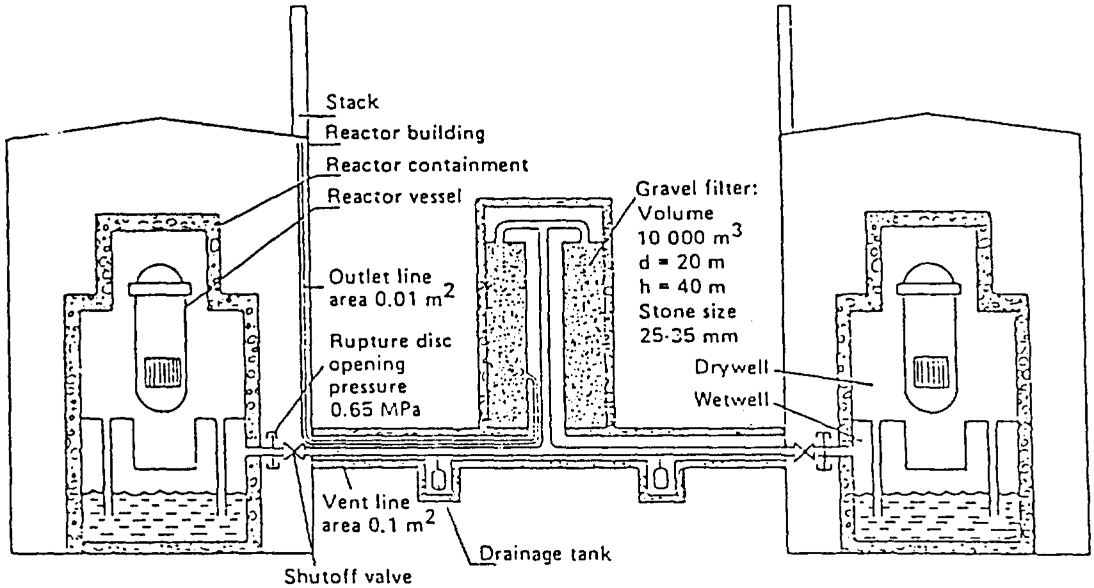


Figure. 10 Schematic drawing – filtered venting of reactor containment. (FILTRA)

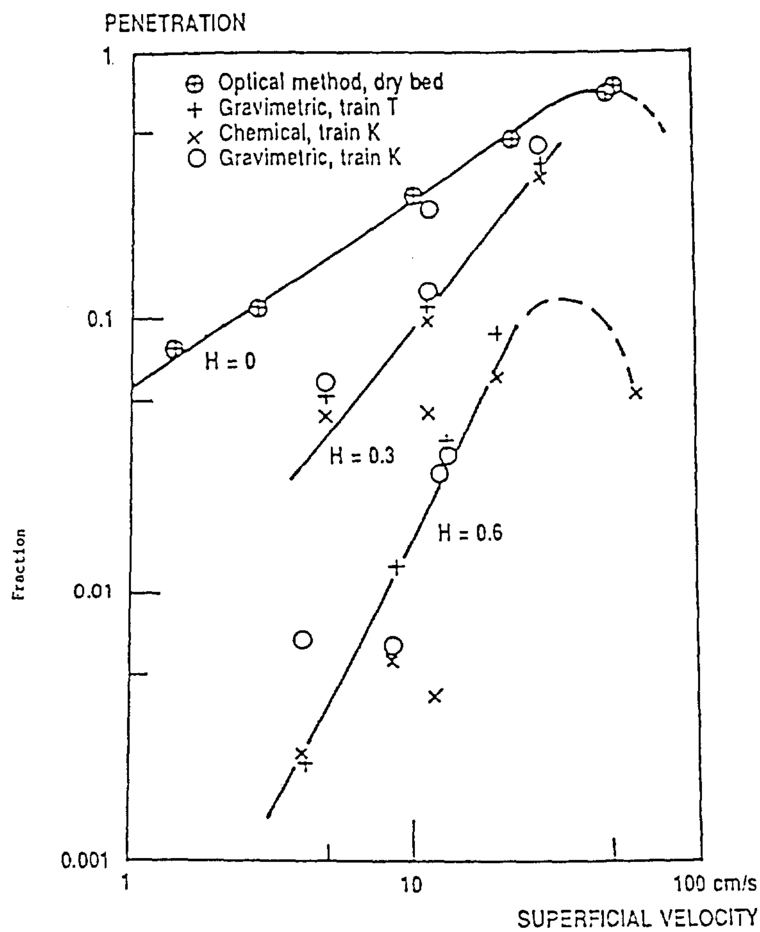
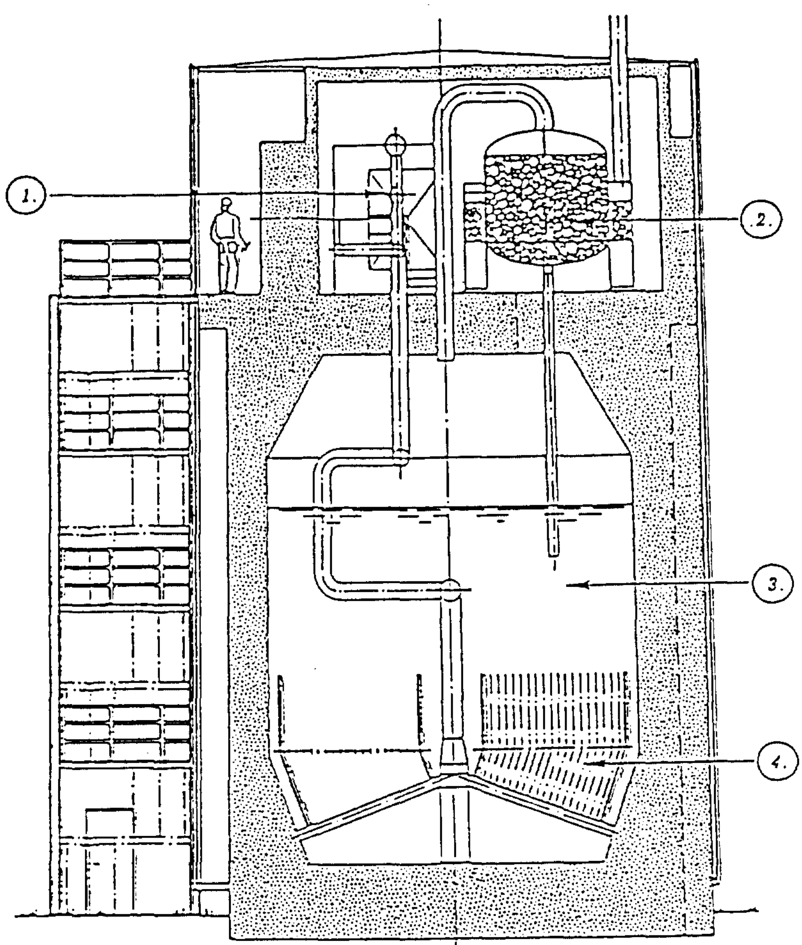


Figure 11

Fractional penetration for test aerosol from different measurement methods at varying steam content H (parts by volume).



FILTRA-MVSS
1. Pressure relief system for the reactor containment.
2. Moisture separation system.
3. Pool.
4. Venturi scrubber system.

Figure 12 The Swedish MVSS Filter Concept

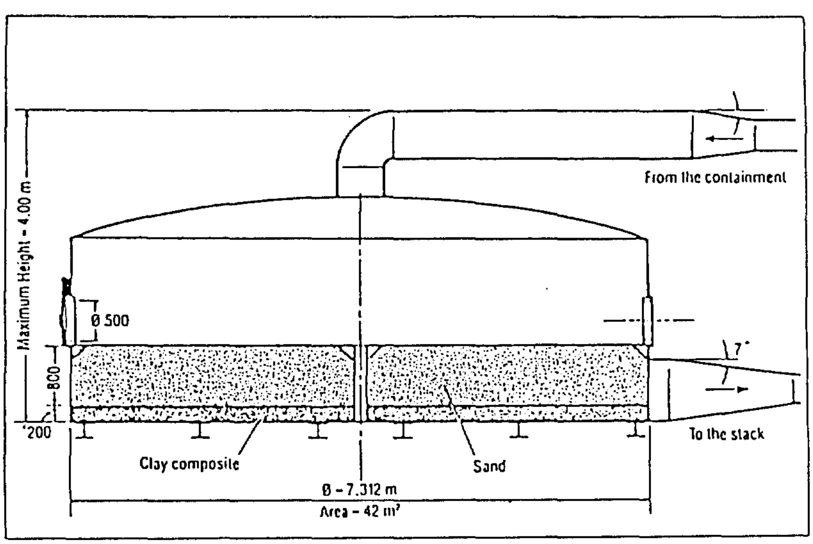


Figure 13 Sand filter for venting of French PWR containments

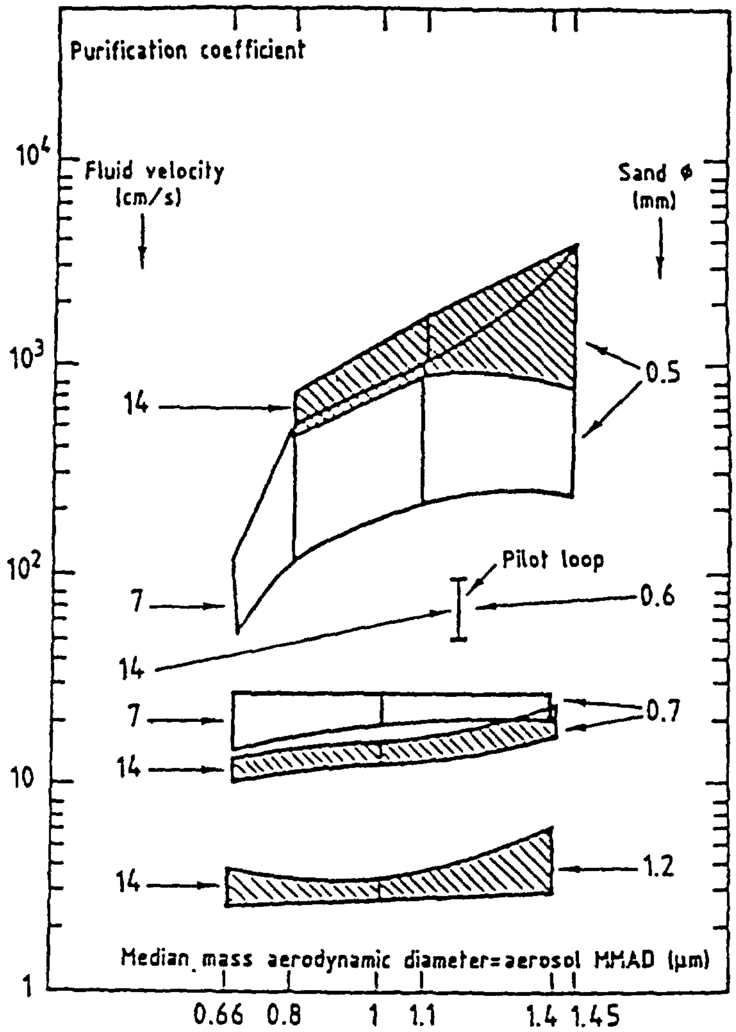


Figure 14 - *Purification coefficient - Test under steady-state operating conditions*

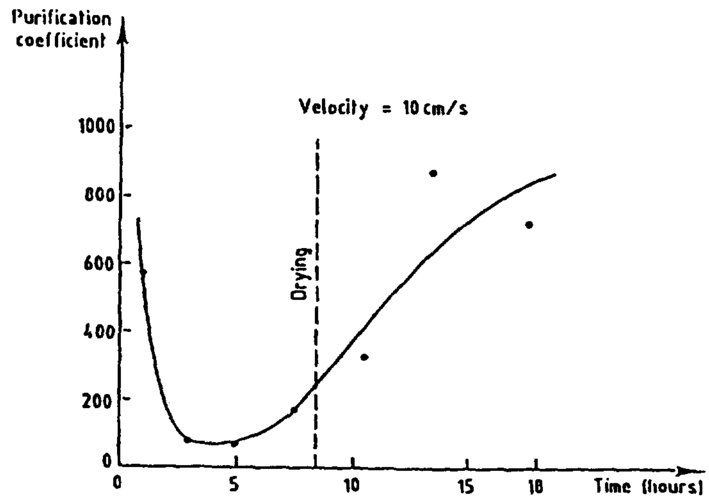
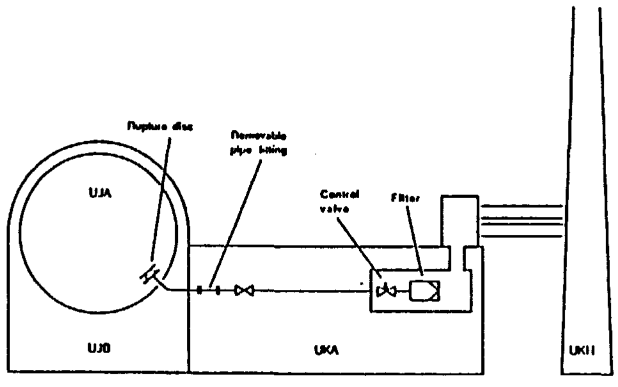


Figure 15 - *Purification coefficient for aerosols of diameter 2.15µm (EMAMD) and gases with a water content of 32%*



UJA : Primary containment
UJB : Annulus
UKA : Reactor auxiliary building
UKH : Vent stack

CONTAINMENT
Design pressure : 5.3 bar
Test pressure : ca. 6.74 bar
Free volume : ca 70 000 m^3

PWR 1300 MW CONTAINMENT VENTING
FLOW DIAGRAM AND ARRANGEMENT : ATMOSPHERIC FILTER (with control valve)

UB KWU - U 8 223 / 10.05.88

Figure 16

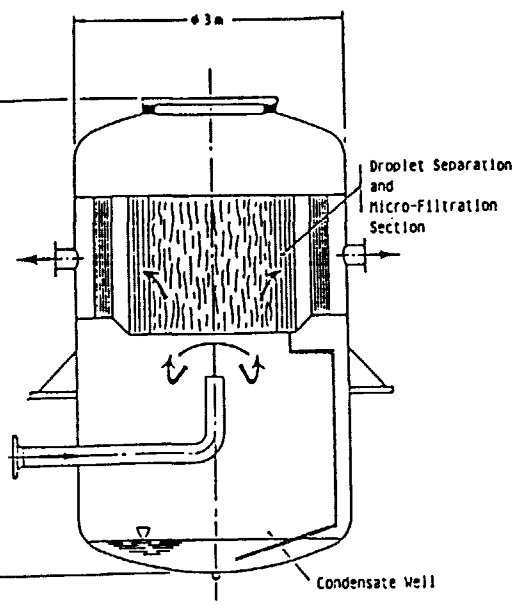
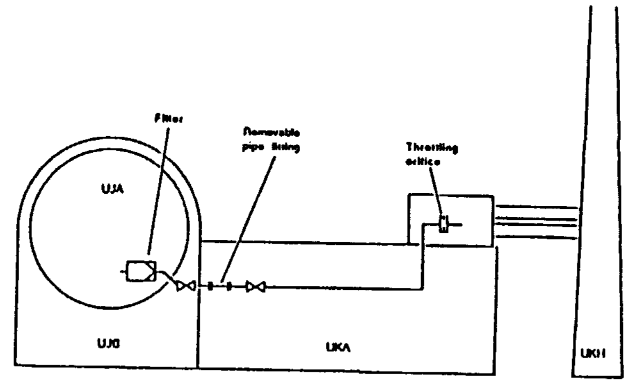


Figure 17
Sliding Pressure Filter Unit, Type Siemens



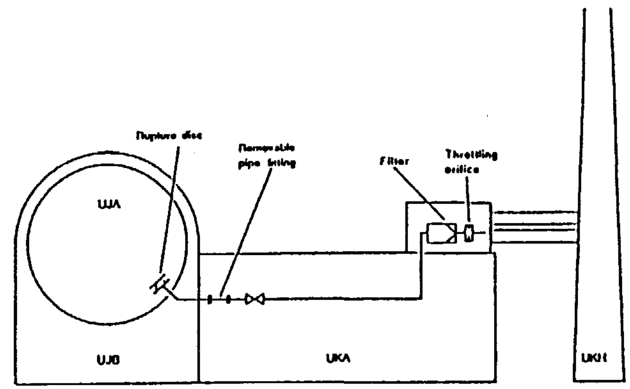
UJA : Primary containment
UJB : Annulus
UKA : Reactor auxiliary building
UKH : Vent stack

CONTAINMENT
Design pressure : 5.3 bar
Test pressure : ca. 6.74 bar
Free volume : ca 70 000 m^3

PWR 1300 MW CONTAINMENT VENTING
FLOW DIAGRAM AND ARRANGEMENT : SLIDING PRESSURE FILTER INSIDE CONTAINMENT

UB KWU - U 8 223 / 10.05.88

Figure 18



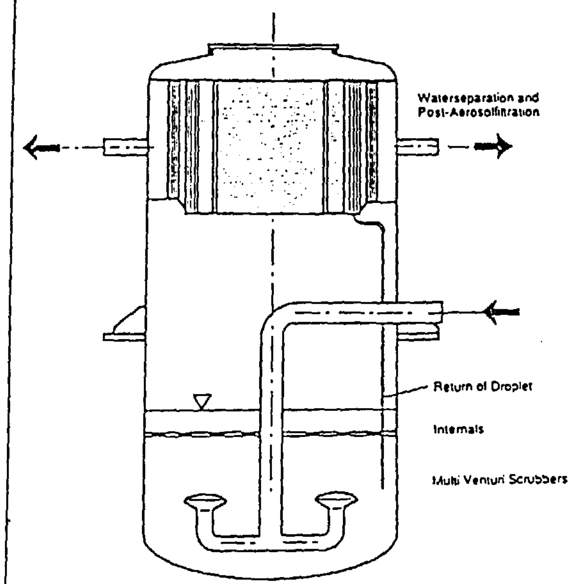
UJA : Primary containment
UJB : Annulus
UKA : Reactor auxiliary building
UKH : Vent stack

CONTAINMENT
Design pressure : 5.3 bar
Test pressure : ca. 6.74 bar
Free volume : ca 70 000 m^3

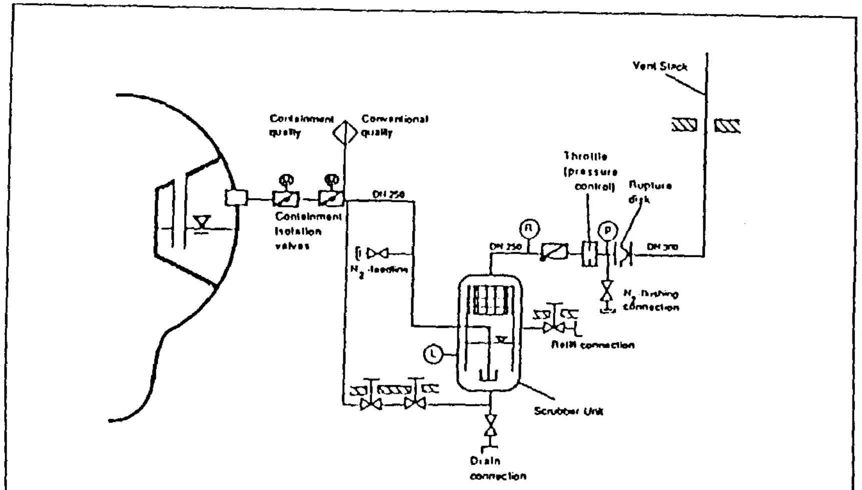
PWR 1300 MW CONTAINMENT VENTING
FLOW DIAGRAM AND ARRANGEMENT : SLIDING PRESSURE FILTER OUTSIDE CONTAINMENT

UB KWU - U 8 223 / 10.05.88

Figure 19



HEW 1988	BWR FILTERED CONTAINMENT VENTING SYSTEM	
Venturi Scrubber Unit Figure 20		Original KWU



HEW 1988	BWR FILTERED CONTAINMENT VENTING SYSTEM	
Sketch of Flow Diagramm Figure 21		Original: KWU

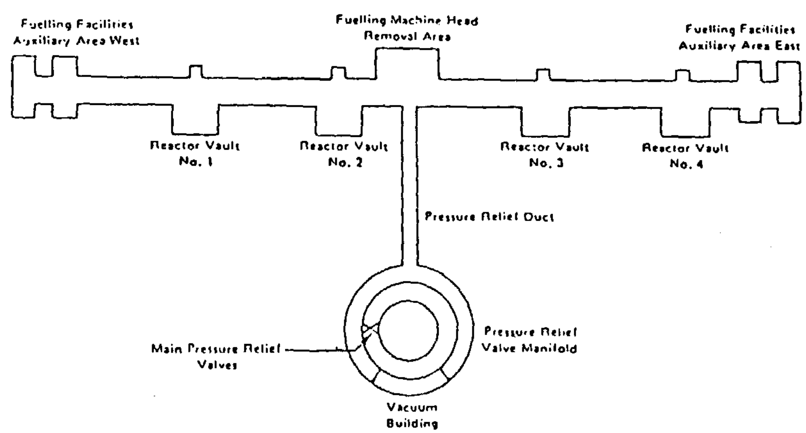


FIGURE 22
CANDU Containment Envelope AT DARLINGTON NPP

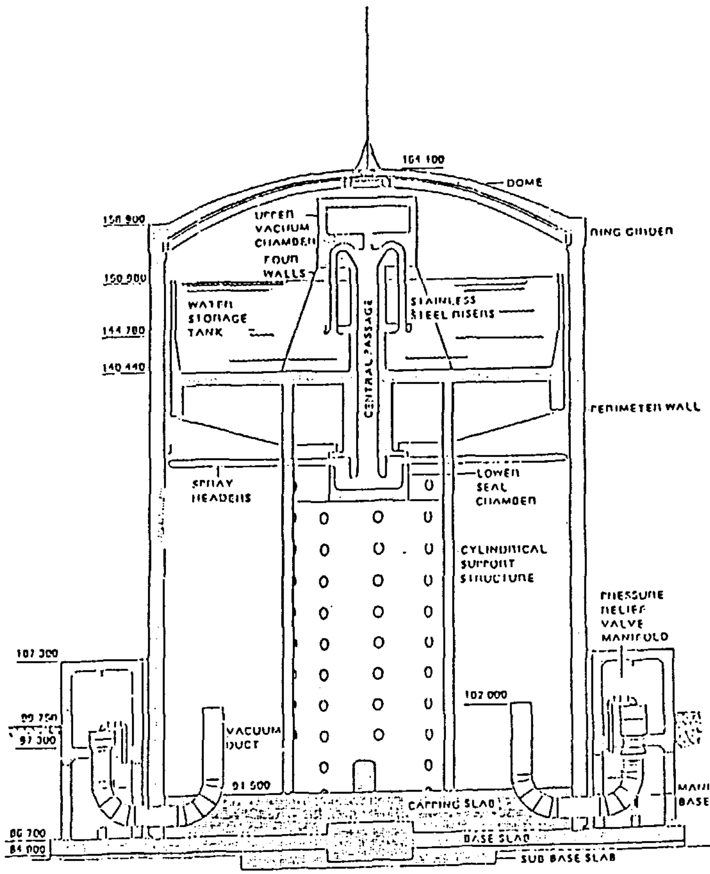


FIGURE 23
Candu Vacuum Building Cross Section at Darlington NPP

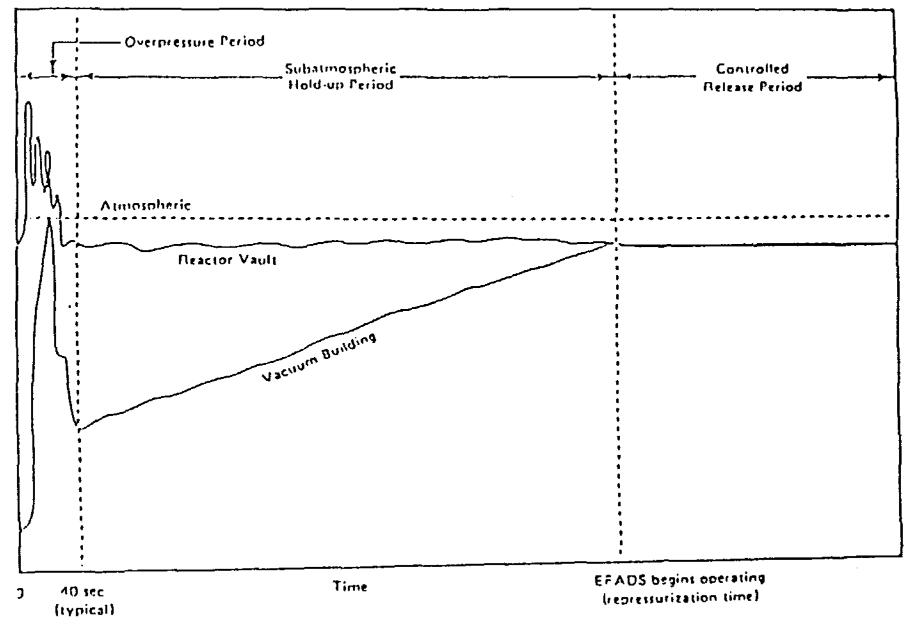


FIGURE 24
Long term containment atmosphere pressure following LOCA with Vacuum Building and EFAD. CANDU Darlington NPP

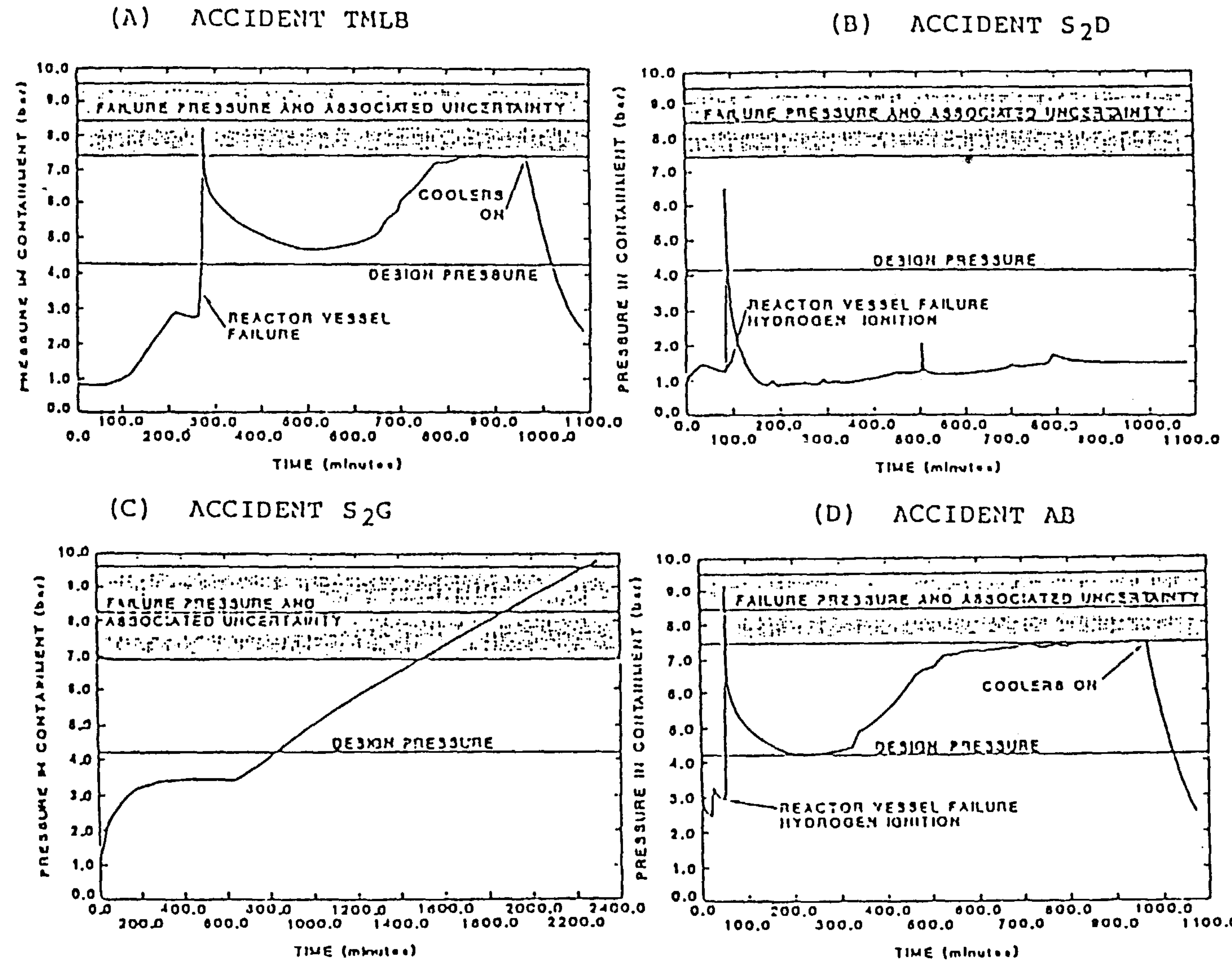
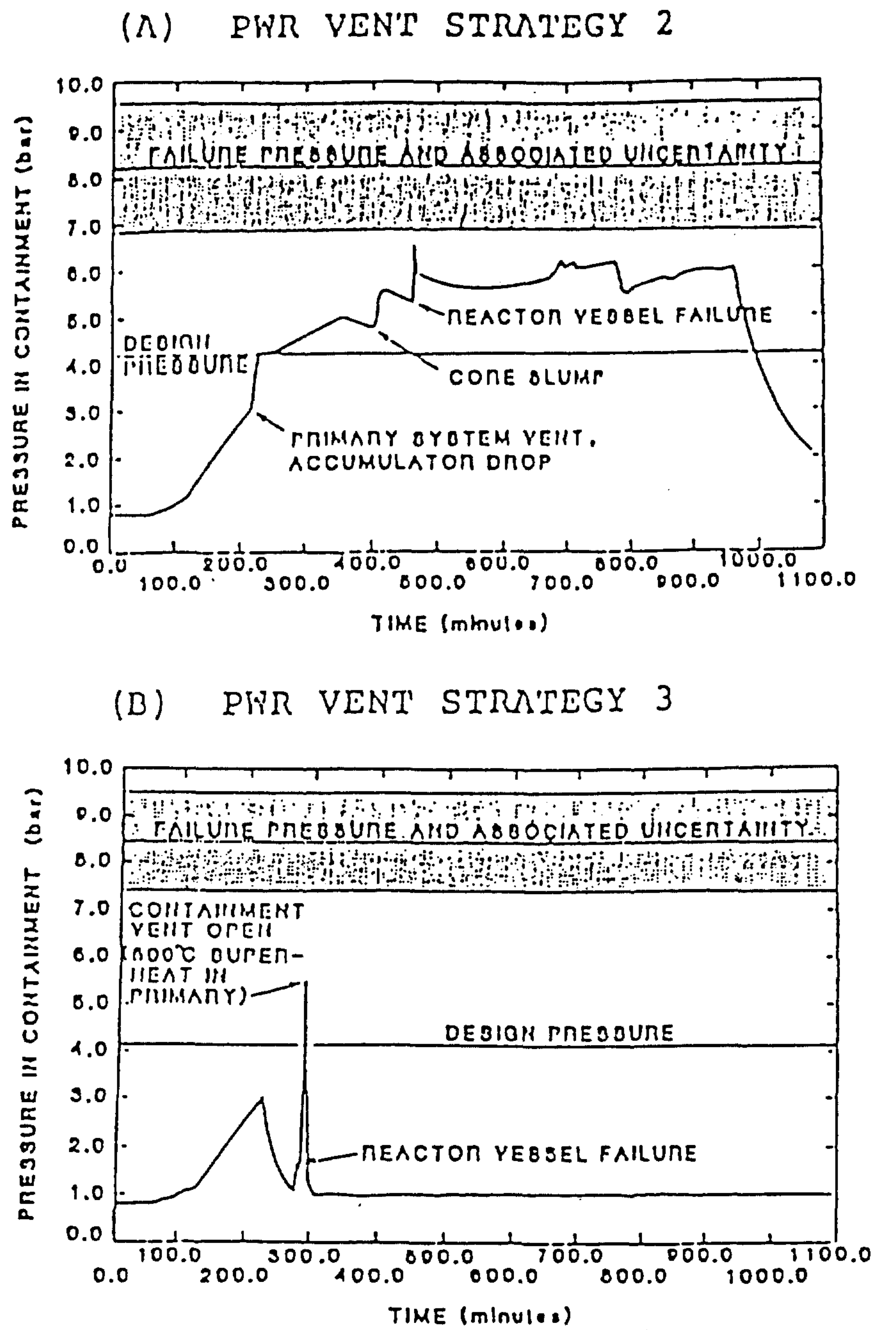


Figure 25 MARCH Code Calculations of Containment Pressure Versus Time for Four Hypothesized Accidents in the Baseline PWR Plant



MARCH Code Calculations of Containment Pressure Versus Time for the Accident TMLB' in the Baseline PWR with Different Venting Strategies.

Figure 26

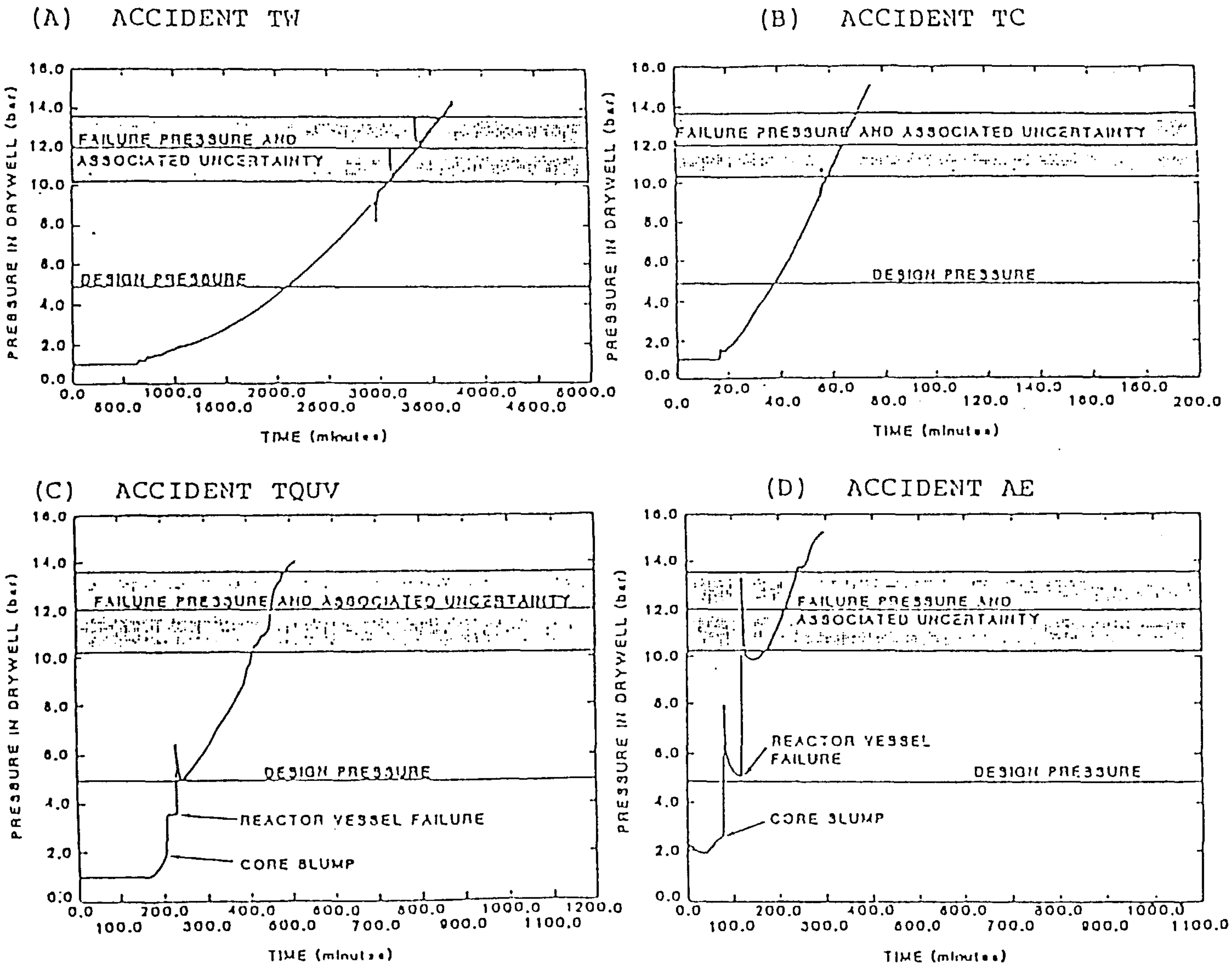


Figure 27 MARCH Code Calculations of Containment Pressure Versus Time for Four Hypothesized Accidents in the Baseline BWR Plant

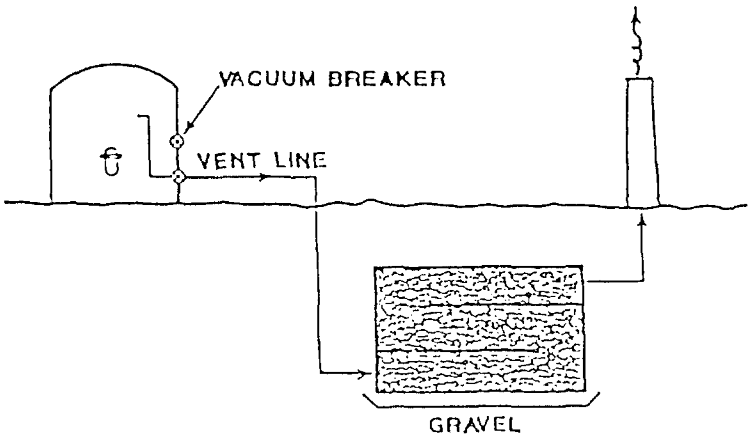


Figure 28 PWR Design Option 1

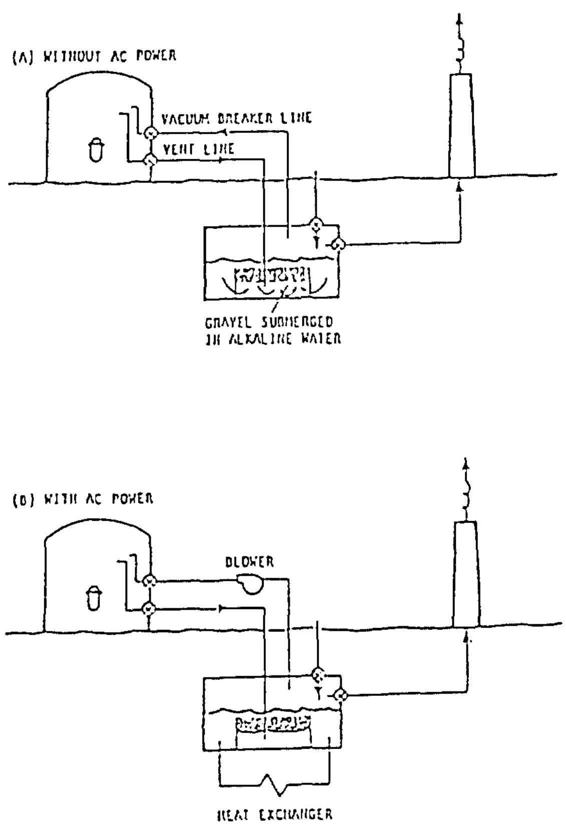


Figure 29 PWR Design Option 2

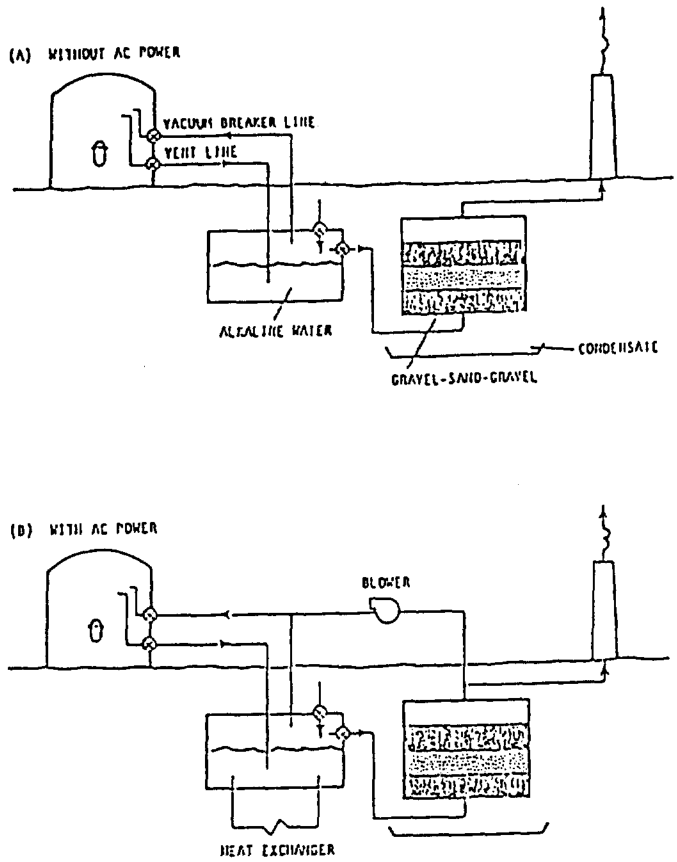


Figure 30 PWR Design Option 3

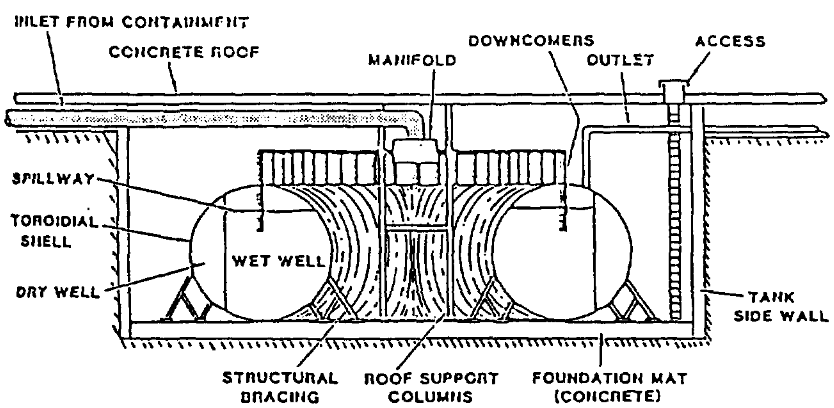


Figure 31 Suppression Pool Section

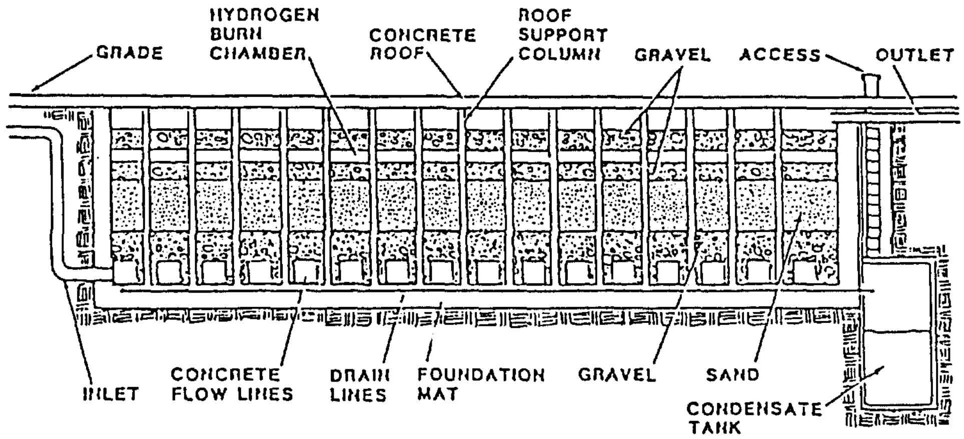


Figure 32 Sand-Gravel Filter Section

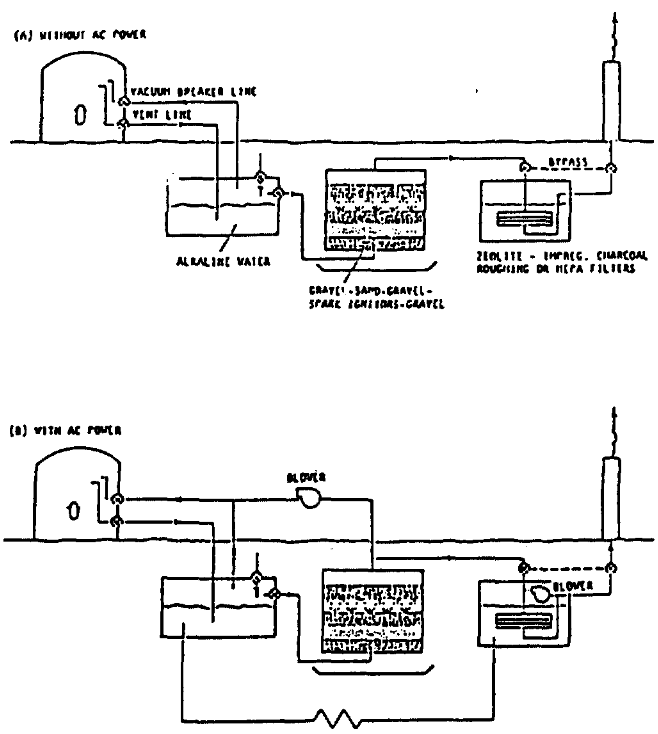


Figure 33 Design Option 4

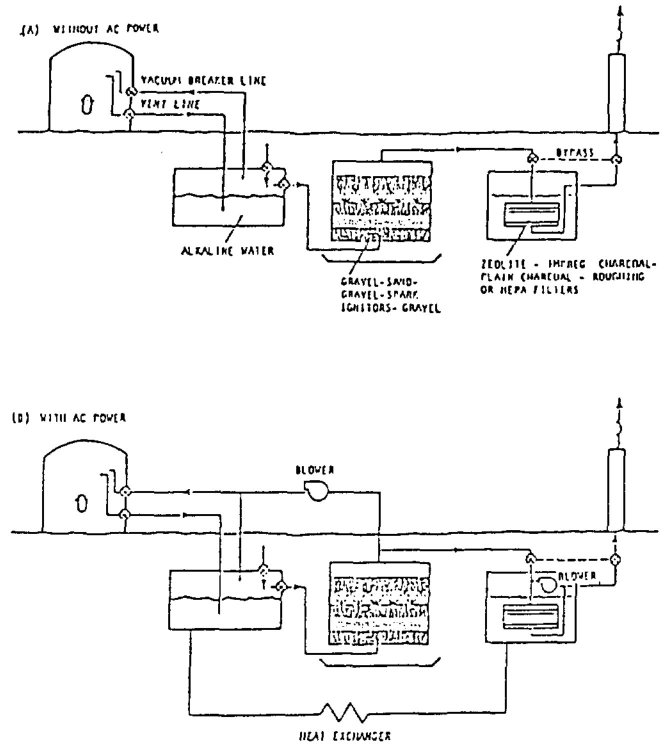


Figure 34 Design Option 5

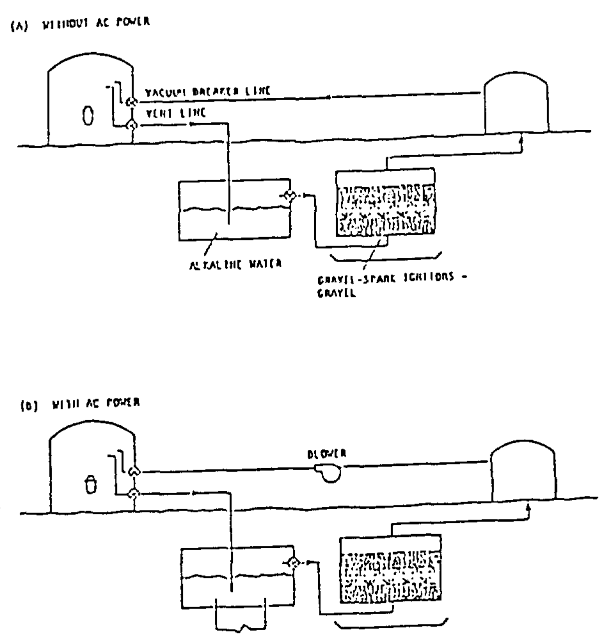


Figure 35 Design Option 6

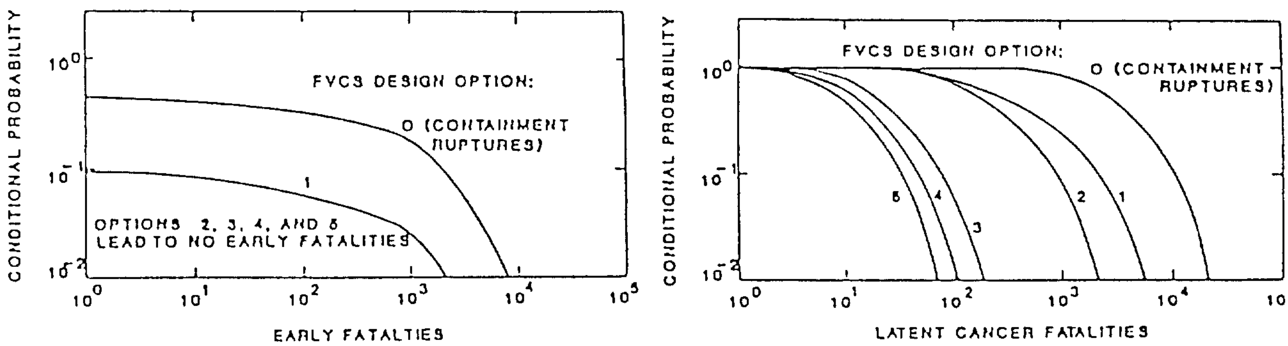
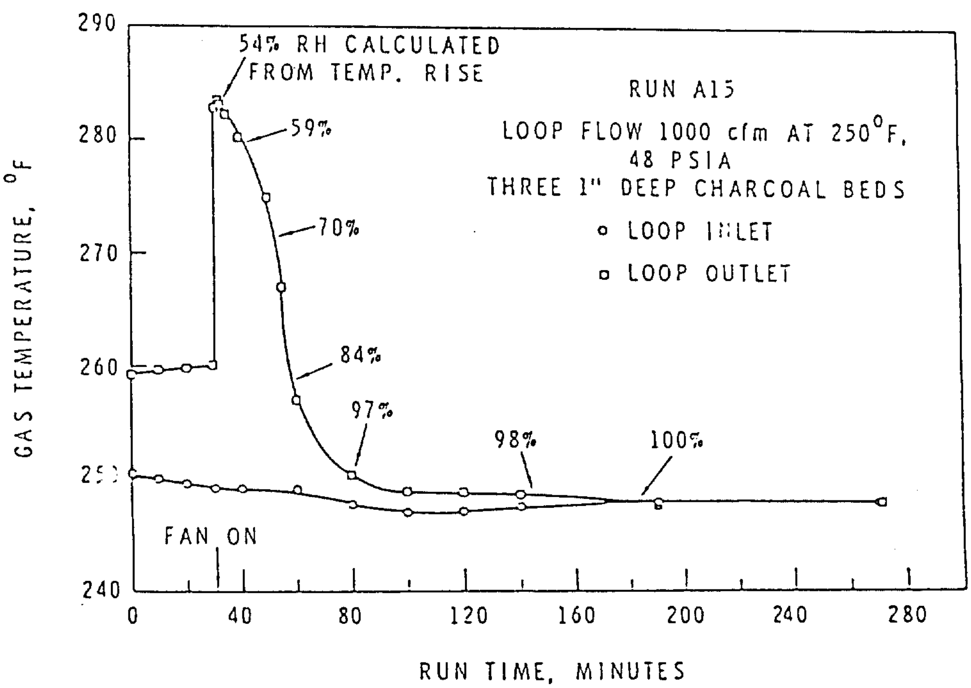


Figure 36 Probability of Early Fatalities and Latent Cancers for TMLB'



Velocity: 20 cm/sec

Bed Depth: 7.5 cm

HEATING DUE TO MOISTURE ADSORPTION ON CHARCOAL BEDS, RUN A15

FIGURE 37

DISCUSSION

FIRST: Thank you for an excellent review of a very difficult topic to cover briefly.

OSLINGER: In Canada, one of the prime requirement of our post accident venting system is to facilitate offsite emergency procedures in the same sense that you recommend. We realize the risk of having an evacuation when you do not need one. For some scenarios the consequences are worse than if you did not have a venting system. We want to avoid evacuating three million people (particulary when you don't need to) with associated human accidents and economic losses. So, we have spent as much on optimizing our design to avoid taking unnecessary offsite actions, as we are spending on the filter part of it. Is this the international perspective as well?

KOVACH: Basically, the international perspective is the same. I have difficulty justifying containment venting from a health effects standpoint. I can justify it from a protection standpoint. We have to look at all the potential applications, including the psychological aspect of saying that even if we had a very severe accident we would not release more than "x" percent in most accidents. You cannot claim that a vented containment would work in all potential accidents modes as there are some hypothetical accidents that could fail the containment. There are very remote possibilities that the vented containment would not save you. In many cases it would. That is why I believe we must not only look at health effects but also at psychological effects, land recovery, etc.

STROM: Concerning the choice of test aerosols for vent filters: at the time of the FILTRA tests, it was believed that the core would give off 1000 kg of iron and 100 kg Cs aerosol particles. Opinions have since changed. I think it is very difficult to predict containment aerosol composition, especially after a long period at high temperature, when settling and revaporization have reformed this core aerosol. Also, other materials in the containment, like paint, electrtical insulation, etc. might have become gas-borne.

INTRODUCTORY COMMENTS OF CHAIRMAN FIRST

The next technical paper, is entitled "Concepts and Operating Results of Minimizing Radioactive Emissions for the Wackersdorf Reprocessing Plant." Reprocessing is a subject that we in the United States are very much dependent upon our Asiatic and European colleagues for developing these days and we are very grateful for it. Our speakers are two, Dr. Juergen Furrer, who is Senior Research Engineer with the Laboratory for Aerosol Physics and Filter Technology of the Laboratory for Nuclear Research in Karlsruhe and Dr. Walter Weinlaender who is a member of the Executive Board of DWK, concerned with design and construction of the reprocessing plant at Wackersdorf. He is with the Institute of Hot Chemistry with the nuclear research facility at Karlsruhe and is project engineer and project leader for the Gorelevan plant and deputy managing director for the reprocessing plant at Karlsruhe. Dr. Furrer is going to give the paper.

CONCEPTS AND OPERATING RESULTS OF MINIMIZING RADIOACTIVE EMISSIONS FOR THE WACKERSDORF REPROCESSING PLANT

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Abstract

In a time of growing concern over environmental impacts from any industrial plant, the ideal goal of zero release has to be approached. This applies to the industrial reprocessing plant in Wackersdorf (WAW), too. Regarding gaseous emissions, remarkable efforts have been made in Germany to improve the offgas cleaning systems. Significant results and operation experience from KfK, WAK and PAMELA are presented.

The design of the offgas trains for WAW is based on these results. It fulfils operational needs as well as the requirements imposed by the German licensing authorities to minimize as far as possible any impacts from emissions.

Offgas systems in reprocessing plants have to treat dry and humid gas streams and to separate aerosols and gaseous materials. A minimization of emissions will be achieved by improving the cleaning units of the offgas lines and by decreasing the total amount of offgas or of special constituents. Both measurements are presented, and the decontamination factors achieved are described. In this context, details are given of the behavior of some nuclides such as ruthenium and iodine. Finally, the emissions expected of WAW are indicated.

I. Introduction

The safe management of radioactive wastes generated in nuclear power plants and, above all, the orderly disposal of them are of paramount importance to the peaceful use of nuclear energy. The safe management of wastes from nuclear power plants under an integrated waste management concept continues to be the prerequisite of nuclear power plant construction and operation in the Federal Republic of Germany. The integrated waste management concept confirmed on September 28, 1979 by a resolution adopted by the Federal Chancellor and the State Minister Presidents provides for implementation of waste management in general by on-site and off-site intermediate storage followed by reprocessing of the spent fuel elements, utilization of the radioactive residues, and conditioning as well as disposal of the radioactive wastes /1/.

The emission levels which Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen (DWK) applied for to be applicable to vent air and liquid effluent discharges have been examined by experts appointed by the authorities and do in no case imply that the limits laid down in the German Radiation Protection Regulation are exceeded. Likewise, aspects of minimizing the radiation exposure habe been taken into account.

The industrial-scale Wackersdorf reprocessing plant (WAW) has been designed for a maximum throughput of 500 metric tons of spent nuclear fuels having undergone burnups of up to 55 GWd/te HM. The site is at Wackersdorf in Bavaria. The premises are 1.1 km^2 in surface area. The fuel element receipt storage building, the central workshop, the module test bench, and the guard building at the entrance gate have been erected so far. Active operation is planned to start in 1996 provided that all licenses will have been granted in time.

II. Criteria to Be Met by Offgas Systems

Reprocessing spent fuel elements implies the exposure of the fuel matrix and its dissolution in boiling nitric acid. Valuable uraninum and plutonium is separated from fission products employing the PUREX process. The resulting high level fission solution is vitrified.

As a consequence of the ALARA principle, which is applied in one way or the other in all countries engaged in the nuclear field, offgas systems must meet very stringent criteria.

Offgas must be stripped of all liquid and solid aerosols, all noxious gases, organic compounds, and water vapor because of its Tritium content. Different offgases arise as a consequence of the flowsheet used. They can be classified in three categories on the basis of their main characteristics:

- Dissolver offgas containing medium amounts of contaminants (aerosols, I-129, H-3, Kr-85, C-14) and arising in small flow rates (approx. 2 E 2 Std.m^3/h) for the Wackersdorf reprocessing plant (WAW).
- Vessel offgas containing small amounts of contaminants (aerosols, I-129) and arising in large flow rates (approx. 5 E 3 Std.m^3/h).
- Vitrification offgas containing large amount of contaminants (aerosols, Ru-106, Tc-99) and arising in small flow rates (approx. 1 E 2 Std.m^3/h).

Additional offgas streams to be purified arise from secondary processes and as exhaust air of buildings. However, they contribute little to the emissions from a plant and, for this reason, will not be considered any further in this paper. The application filed for WAW /2/ contains maximum levels of annual activity releases as listed in Table 1.

Nuclides	Bq/a	Ci/a
α-aerosols	1.4 E 9	0.038
β-aerosols	5.4 E10	1.5
Iodine-129	1.85 E 9	0.05
Krypton-85	1.6 E17	4.32 E6
Tritium	1.5 E15	4.05 E4
Carbon-14	1.3 E13	3.5 E2

Table 1: Proposed annual emissions of radioactive nuclides from the Wackersdorf reprocessing plant.

In accordance with the conditions in the German Radiation Protection Ordinance, these limits must not only be observed, but in fact must be underrun as far as is technically feasible.

Demands of this kind have given rise to thinking about ways and means of reducing emissions even further. This minimization must start at the source terms and then proceed to improvements of removal facilities. For this purpose, the source terms had to be measured in various parts of the plant.

III. Offgas Measurements

III.1 Dissolver Offgas

In the shearing and dissolution steps, the protective zircaloy claddings of the fuel rods are opened and the fuel matrix is destroyed.

In addition to the produced shearing dust and solution aerosols, the dissolution step also sets free to such volatile nuclides as Kr-85, I-129, H-3 and C-14.

To determine the aerosol source terms in the dissolution cycle of WAK /3/, samples of the dissolver offgas were taken upstream of the HEPA filter stage and the source terms downstream of the dissolver were calculated (Table 2).

WAK components		dissolver	NO_2 scrubber	gas scrubber
source term (mg/m^3)	shearing	3.5	–	0.35
	dissolution	5–20	–	0.5–2.0

PASSAT components	dissolver	NO_2 scrubber	HEME
source term: shearing (mg/m^3)	11	2.5	10^{-3}
decontamination factor		4.4	1800

Table 2: Behavior of shear and dissolver aerosols in WAK and of simulated shear aerosols in PASSAT.

The aerosol concentrations in the upstream gas of the HEPA filter stage were determined over a number of dissolutions steps and found to vary beween 0.35 and 2 mg/m^3. The average value during several dissolution cycles amounted 0.4 mg/m^3. Assuming a decontamination factor (DF) of 10 for the offgas precleaning the source terms downstream of the dissolver can be calculated to 3.5 to 20 mg/m^3 and an average value of 4 mg/m^3.

The greatest discharge arose when feed solution was heated up at the beginning of the dissolution step. Increasing bubble formation produced more and more aerosols, which were released into the offgas train as a consequence of the growing vapor volume.

A remarkable feature was the relatively small amount of shear dust release of 3.5 mg/m^3, which can be explained as being due to the use of a pin-shear.

For comparison with the WAK plant, the dissolution cycle was studied with simulated shear dust and liquid aerosols, respectively, in the PASSAT dissolver offgas test rig (Fig. 1) /4/.

The PASSAT facility was built by KfK in accordance with the latest state of the art and can be considered to be a prototype of WAW dissolver offgas system. One major point investigated in the program is the improvement of mist elimination by means of a high-efficiency mist eliminator capable for regeneration (HEME) /4/.

The shear aerosols were simulated by lead oxide (PbO), which is physically and chemically similar to UO_2. The maximum of the frequency distribution of the particle diameter was around 0.4 µm.

For comparable source terms, the aerosol concentration in the PASSAT facility, upstream of the HEPA filter system, was roughly two orders of magnitude below the WAK level (Table 2). This finding demonstrates the high efficiency of HEME in retaining shear aerosols and liquid aerosols with high DFs (1800 and 500, respectively).

III.2 Vessel Offgas

In the process steps of extraction and waste treatment, aerosols are released and carried away with the offgas whenever solutions are moved. The vessel offgas systems installed for the removal of those aerosols are made up first of several subcollectors with gas coolers. This design has been chosen for flowsheet reasons and because of safety. As a consequence of complexity typical for operating plants it is not possible in WAK to obtain any data on single source terms.

aerosol source	aerosol source terms downstream of airlift vessel and of the electrochemical components, respectively
airlift: HA-column driving air : 1.5 std.m³/h "HAW" transfer: 0.8 m³/h U-content : <3 mg/l airlift: C-column driving air : 3.7 std.m³/h 1CU transfer : 2.1 m³/h U-content : 74 g/l	<0.3 mg/std.m³ (U content: not detectable) (value not representative, no salt content) 24–35 mg/std.m³ (U-content: 25–35%)
electrolytic U-IV production cell scavenging air : 84 std.m³/h U-content : 285 g/l electrolytic reoxidation cell (ROXI) current : 2700 A scavenging air : 158 std.m³/h U-content : 15 g/l	24 mg/std.m³ (U content: 40%) ≤6 mg/std.m³

Table 3: Aerosol source terms of airlifts and electrochemical components of UEZ/TEKO.

In order to find such information, use was made of the uranium extraction test rig (UEZ) in TEKO /5/, which employs depleted uranium. Specific measurements were carried out in this way on airlifts and electrochemical equipment, the results of which are summarized in Table 3.

To complete our findings, and also for parameter studies of source terms and retention components, a vessel offgas test rig (BEATE) /6/ was built by KfK.

That test rig allowed to investigate the aerosol source terms of a solution by
- scavenging it with air,
- agitating it with air and pulsator,
- transferring it by means of airlifts and jets.

The highest source terms were found for processes in the course of which air passes through liquids, such as air sparging, airlifting, and in electrochemical equipment. Those source terms were between 5 and 50 mg/m^3, which was on the same order of magnitude as in the dissolver of WAK, due to identical mechanisms of aerosol generation. Small contributions were made by jets, by pulsators and by the air scavenging mode of operation. Only blowing out the pulsator again resulted in detectable aerosol formation.

A factor of similar importance as aerosol generation is aerosol behavior in the downstream pipes and in the offgas condensor. The DFs are between 7 and 10, depending on the aerosol sources. Increasing humidity of the carrier gas causes the DFs to rise as a result of the increase in aerosol size and the entrainment effect of the condensing vapor. As a result of the excellent removal characteristics, the aerosol concentration downstream of the condenser was far below 10 mg/m^3.

III.3 Vitrification Offgas

HLWC vitrification in WAW will be carried out on the basis of the PAMELA process /7/. The HLWC is fed directly into the melter at a pool temperature of approx. 1100 - 1150 ^{o}C. This means that the steps of evaporation, denitration, calcination, and vitrification are performed in one process area.

Aerosols and highly volatile compounds, such as ruthenium and technetium, are generated in the process. Valid data about the discharge of aerosols from these high temperature processes were obtained for the first time in the PAMELA vitrification plant. For the melter, a DF of 65 was found for the β-activity (Table 4).

On the basis of the waste oxide content of the low enriched waste concentrate (LEWC), this gives rise to a calculated aerosol source term of 35 g/h and, related to the melter offgas, a concentration of 300 mg/m^3.

Intensive gas scrubbing by means of a jet scrubber reduced this term to < 10 mg/m^3. Including HEME, a DF of > 3 E 6 was achieved in the wet part of the offgas system, and an offgas load, respectively, upstream of the HEPA stage of < 0.1 μg/m^3.

Throughput	
LEWC from Eurochemic	20.7 m^3
waste oxide content	2632 kg
vitrified α-activity	3.16 E 14 Bq
vitrified β-activity	8.65 E 16 Bq
Results	
aerosols source term of ceramic melter	300 mg/m^3
decontamination factor of ceramic melter	65
decontamination factor melter to HEME	$>3 \times 10^6$
calculated mass concentration upstream of HEPA filter	$<0.1\ \mu g/m^3$

Table 4: Aerosol Source term and decontamination factors of the PAMELA vitrification plant (LEWC Campaign I).

III.4 Specific Nuclides

III.4.1 Iodine-129

The long halflife of iodine, and its specific enrichment in the human thyroid, necessitates high iodine removal factors from the off-gases of reprocessing plants. Iodine, because of its high volatility and the easy reducibility of its compounds, is released into the off-gas from all vessels, pieces of equipment and pipes once it has spread throughout the process. This release occurs immediately or after some delay, depending on existing physico-chemical conditions.

For all these reasons it is recommended, and has been tested in WAK, to strip the iodine from the fuel solution as quantitatively as possible in the dissolver offgas by boiling under reflux and then remove it from the dissolver offgas by special iodine filters. In WAK operation, these filters have a DF > 1 E 3.

III.4.2 Ruthenium-106

Ruthenium may form highly volatile compounds under high temperature and oxidating conditions e.g. in the vitrification of process solutions. In the literature, the formation of ruthenium tetroxide and ruthenium nitrosyl complexes is described /8/. These compounds are retained quite effectively in condensers and scrubbers. In addition, there are ruthenium dioxide aerosols, which are easy to remove.

In WAK, a ruthenium DF > 1 E 5 was measured for the dissolver and a DF >1 E 4 for the 1W evaporator. The overall DF (feed to stack) for ruthenium is 1 E 10 (Table 5). No ruthenium was detected in WAK on special RuO_4 filters (grade 40 silicagel) downstream of the HEPA filter stage. Consequently, the volatile RuO_4 fraction in the emission of ruthenium can be taken to be below 1 %.

WAK

components	Ru
dissolver	$>10^5$
1W evaporator	$>10^4$
HEPA filter (DOG)	$>10^4$
overall (feed solution to stack)	$>10^{10}$

PAMELA

components	Ru	Tc	β
ceramic melter	8.5	2.2	65
wet offgas cleaning system	300	400	50
offgas system (melter-jet scrubber)	2600	870	3100
overall	–	–	$>10^{13}$

Table 5: Ru-DF's in WAK and Ru-, Tc-, total β-DF's in PAMELA.

In the PAMELA vitrification plant, a ruthenium DF of 8.5 was found for the melter. A comparision with the β-aerosol release indicated a source term eight times higher.

For the complete train, including the melter, wet scrubber, condenser, and jet scrubber, a DF of 2600 was found, which was on the same order of magnitude as for the β-aerosols. The overall DF for the throughput of β-emitters versus offgas release was > 1 E 13.

These results clearly show that ruthenium-106 is no problem nuclide in reprocessing and waste treatment and that no special ruthenium filters are required in the offgas system, because ruthenium behaves as a β-aerosol in the filter train.

III.4.3 Technetium-99

As a result of its relatively long halflife of 2.1 E 5 years, special attention is being devoted to the retention of Tc-99. In the vitrification step, this nuclide produces volatile compounds, such as $CsTcO_4$ /9/.

In PAMELA, a DF of 2.2 was found for Tc in the melter. This means a release from the melter almost thirty times higher than that of other β-emitters, airborne only as aerosols (Table 5).

In the wet part of the offgas train, a DF of 400 is achieved for Tc (good solubility of alkaliper-technetate), which is eight times higher than the DF for the total β-aerosols. This fact is due to the efficient removal of Tc in the scrubbers. The very high overall DF for all β-emitters (> 1 E 13) shows that the emission of Tc is well under control.

IV. Removal Efficiencies of Components

The experience accumulated in the PASSAT, WAK and PAMELA plants was used in choosing the appropriate components and systems for WAW (Fig. 1):

- As a rule, condensers are the first components in a system. Their DFs depend especially on the fraction of condensable offgas and are in the range of 7-100.

- Gas scrubbers are state-of-the art systems. The more intensive the scrubbing process, i.e., the higher the energy input, and the more effective the downstream demister, the higher will be the DF. This is confirmed by a special design, the jet scrubber in the PAMELA plant, with a DF in excess of 30.

- HEMEs that can be regenerated are designed as fiber packs. They have become indispensable components in our present systems. High decontamination factors (> 2000 for PAMELA) extend the service lives of the HEPA filters, thus reducing the arisings of solid medium level waste. The elimination of mist carrying low salt loads (secondary aerosols) prevents generation of penetrating aerosols in the sub-micron range which are formed after heating the offgas effectively in front of the HEPA filters. In this way, HEMEs directly contribute to raise the overall DF of the system.

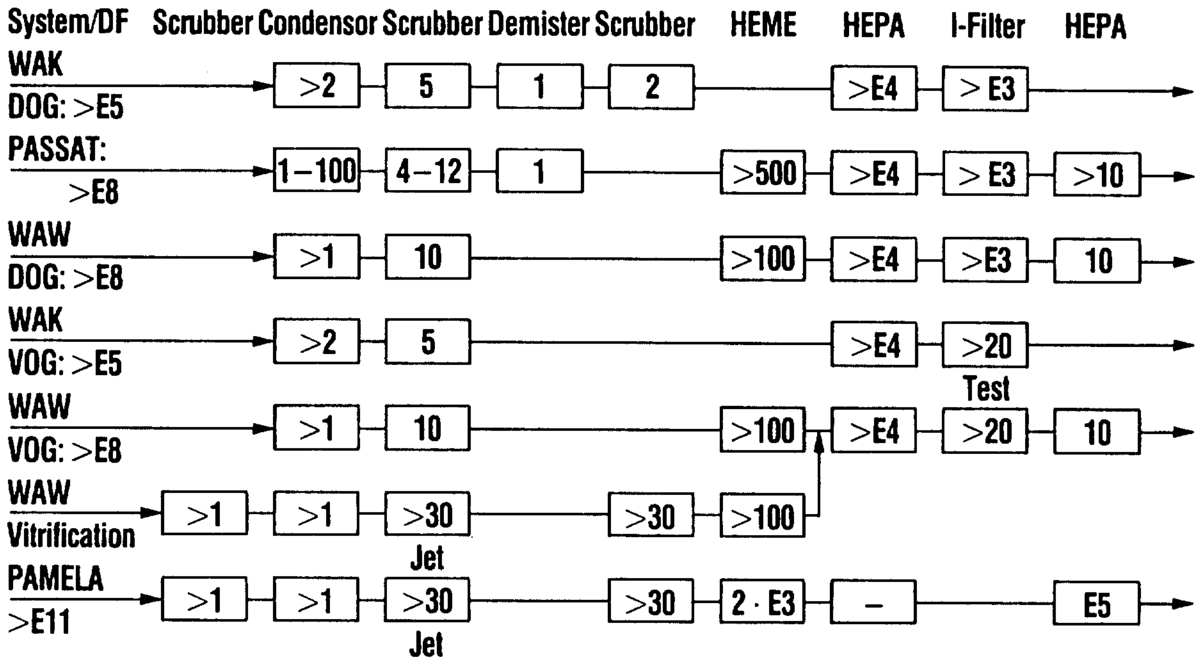

Fig. 1: Basic Flow Diagram for the Offgas-Systems and DF's of PASSAT, PAMELA, WAK and WAW.

- Although HEPA filters now represent the state of the art, specific adaptations to process conditions, such as
  . the use of acid resistant and heat resistant ahhesives and sealing materials,
  . design features ensuring contamination -free filter replacement,
  . continuous monitoring of tight fit,

have continuously improved the DF > 1 E 4 in PASSAT, PAMELA and WAK.

- Two-stage HEPA filters clearly add to the plant DF. In addition, the second stage is expected to exercise a barrier function to α-aerosols.

  In PASSAT and in PAMELA, the two-stage concept has been confirmed by DFs > 1 E 6 resp. > 1 E 5.

- Iodine filters in the dissolver offgas stream are proven components attaining DFs > 1 E 3 /10/.

- Iodine filters for the vessel offgas are under development. Their use in WAW has become necessary as a result of the iodine emission limit having been lowered from 200 to 50 mCi/a. Test filters are being investigated in WAK. A modified version of the proven adsorption material used for the dissolver offgas has been found to attain DFs > 40. Further optimization work is being focused on extending the service lives of filters.

V. Complete Offgas Purification Systems

V.1 Description of Offgas Systems

The DFs attained by individual components are not sufficient to characterize their combined purification efficiency. What is decisive is the overall concept and the arrangement and matching of compounds in the system.

- WAK:
The dissolver offgas and vessel offgas systems of WAK, which were designed and built already some time ago, basically consist of coolers, scrubbers and HEPA filters (Fig. 1). A DF of > 10^5 for aerosols is attained in both systems.

Iodine retention in the dissolver offgas has been improved by two iodine filter installed in line. In this way, the first iodine filter is loaded to its full capacity, while the second iodine filter serves as a safety filter.

- PASSAT:
The PASSAT test rig is based on experience accumulated in WAK and, in addition, takes into account the most recent findings in the elimination of very fine mist. For this reason, a wave plate separator and a HEME has been installed between the scrubber column and the HEPA filter stage.

The offgas components have been improved both in design and with respect to remote operation. A continuous system of monitoring the tight fit of the filter element sealings has been introduced. Improved mist retention and the two-stage HEPA filter system have helped to achieve a system DF of more than 1 E 8 for the aerosols generated.

-PAMELA:
For the offgas system of the PAMELA (Fig.1), PASSAT experience was exploited for aerosol retention. Because of the high aerosol discharges from the melter, a wet scrubber, based on experiences accumulated in the KfK vitrification test rig /11/, was installed upstream of the condenser and a jet scrubber downstream of it.

A very effective system was found to be the combination of a jet scrubber and an NO_x scrubber; the aerosols intensely wetted in the jet scrubber were retained with DFs > 30 each in both systems.

A HEME confirmed the concept of very fine mist retention for the first time in hot operation with a DF > 2000. The system offgas-DF was found to be > 1 E 11.

V.2 WAW Concept

The experience and the findings outlined above were used in designing the WAW offgas systems. These systems will be briefly outlined below.

V.2.1 Dissolver Offgas

The dissolver offgas system is based on the proven concepts used in the PASSAT facility, WAK, and PAMELA (Fig.1). The dissolver is followed by a reflux condenser, a scrubber column, HEME, HEPA stage I, two iodine filters installed in line, and HEPA stage II. An aerosol DF > 1 E 8 is expected in the light of experience with other plants.

V.2.2 Vessel Offgas

The vessel offgas system contains the same components for aerosol retention and, for the first time, iodine filters (Fig. 1). The expected aerosol DF is > 1 E 8.

V.2.3 Vitrification Offgas

The offgas of the vitrification process is treated in accordance with the PAMELA system and is added to the vessel offgas stream upstream of the HEPA and iodine filter stage. The same DFs are expected as for the PAMELA offgas system.

VI. WAW Emissions

VI.1 Aerosols

Emissions primarily are proportional to the offgas streams, their loads and specific activities, and inversely proportional to the DF of the offgas system.

Extensive measurements described earlier indicate an expected offgas load < 10 mg/m^3 downstream of the condenser for all offgas sources. Because of the high aerosol load of the vitrification offgas, this level will be reached only downstream of the additional jet scrubber. For all systems, however, this point is followed by another offgas scrubbing station as well as wet and dry aerosol removal stages with decontamination factors > 1 E 8.

Hence, the expected emission levels from normal operating conditions are calculated to be

< 1.85 E 9 Bq/a (< 50 mCi/a) for β-aerosols, and

< 3.7 E 7 Bq/a (< 1 mCi/a) for α-aerosols.

These expected levels are clearly below the permissible annual activity releases.

VI.2 Iodine

Basically the same iodine concept as in WAK will be used in WAW, i.e., stripping iodine from the fuel solution as completely as possible and retaining it on iodine filters in the dissolver offgas system (DF $\geq$ 1 E 3.)

The residual iodine is expected to be distributed in reprocessing process similarly to WAK: 50 % of the residual iodine will be released in the vessel offgas and largely removed in iodine filters (DF $\geq$ 40) /12/. The remaining iodine stays in waste streams and is solidified. In the light of this experience, an iodine emission of < 1.1 E 9 Bq/a (< 30 mCi/a) is expected.

VI.3 Tritium, Krypton-85, Carbon-14

These nuclides arise mainly in fuel dissolution and, hence, in the dissolver offgas. Because of their small contributions to the radiological burden in the environment, no additional retention components would be required.

Tritium scrubbing in the first extraction cycle and recycling of the acid causes the tritium to remain in the head end and to be eliminated from the process largely as tritiated water.

In line with a request made by the licencing authority, a pilot plant will be built to demonstrate the retention of Krypton from the dissolver offgas of WAW. Preliminary concepts have been drafted of the methods eligible for this purpose. A decision about the process was made in 1988 by comparision of the three processes for separation of Krypton-85. It was decided to take the selective absorption process by Freon-12 washing at subambient pressure which had been developed by KfK /13/14/15/.

VII. Final Remarks

The concept of minimizing radioactive emissions from a reprocessing plant is a very important requirement in the Federal Republic of Germany. It is for this reason that major research and development efforts have been, and will be made to reduce the generation of radioactive aerosols and improve the removal efficiencies of offgas purification systems.

The results of these studies will be used in the design of the WAW-offgas systems; they indicate that dose commitments may be expected in the environment, which are very clearly below the limits specified in the German Radiation Protection Regulations.

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INTRODUCTORY COMMENTS OF CHAIRMAN FIRST

Our next presentation is entitled "Species Available for Release for a Spectrum of Containments," and we have two authors. One is B.R. Ross, Deputy Director for Research of the United States Regulatory Commission and R.S. Denning, Battelle Columbus Laboratory. The presentation will be made by Mr. Ross, who is Deputy Director for Research of the United States Regulatory Commission. He has been 21 years with the Nuclear Regulatory Commission, twelve years at the National Reactor Testing Station at Idaho, and with General Dynamics, Fort Worth, in reactor operations. He has a Doctor of Engineering degree in Nuclear Engineering.

AIRBORNE CONCENTRATIONS OF RADIOACTIVE MATERIALS IN SEVERE ACCIDENTS

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Abstract

Radioactive materials would be released to the containment building of a commercial nuclear reactor during each of the stages of a severe accident. Results of analyses of two accident sequences are used to illustrate the magnitudes of these sources of radioactive materials, the resulting airborne mass concentrations, the characteristics of the airborne aerosols, the potential for vapor forms of radioactive materials, the effectiveness of engineered safety features in reducing airborne concentrations, and the release of radioactive materials to the environment. Ability to predict transport and deposition of radioactive materials is important to assessing the performance of containment safety features in severe accidents and in the development of accident management procedures to reduce the consequences of severe accidents.

Background

The purpose of this paper is to describe the quantitites and characteristics of radioactive materials released to the containment building of a light water reactor during a severe accident. The current state of understanding of the release and transport of radioactive species is based on extensive experimentation, model development and accident analyses performed under the support of the U.S. Nuclear Regulatory Commission, the Electric Power Research Institute, and other countries with cooperative severe accident research programs. Much of this effort has occurred since the accident at Three Mile Island.

Results of analyses performed with the NRC's Source Term Code Package[1] are presented in this paper for severe accident sequences in two reactors: Sequoyah,[2] a pressurized water reactor (PWR) with an ice condenser containment, and Peach Bottom,[3] a boiling water reactor (BWR) with a Mark I containment design. The release and transport of radioactive material depends on the conditions associated with each different severe accident sequence as well as the design of the plant. The results that are presented should be considered only as illustrative examples.

A large variety of radionuclides are potentially available for release in an accident. Of these, approximately 54 contribute most of the exposure and must be included in the analyses. Which are the most important radionuclides depends on the conditions in a particular accident sequence and the dose effect of interest. For example, cesium-137 is typically found to be the largest contributor to latent health effects whereas iodine-131, iodine-133, and tellurium-132 are typically major contributors to early fatalities. Computer codes that predict environmental transport of radioactive material and the health consequences to exposed individuals must include all of the dominant radionuclides. In analyzing the release and transport of radionuclides within containment, however, it is possible to combine radionuclides into groups with common chemical properties. This simplification does involve some approximation, however, since radioactive daughters may transport differently from the parent. Table 1 shows the nine elemental groups used in the Source Term Code Package.

Characteristics of Example Sequences

Sequoyah - TB

The Sequoyah plant is a pressurized water reactor with an ice condenser containment design. The accident sequence used as an example in this paper involves station blackout (loss of all ac power) with resulting assumed failure of reactor coolant pump seals.(2) Because of the loss of power, the hydrogen ignition system and fan recirculation system (that circulates air from the upper containment compartment to the lower containment compartment and back through the ice bed) do not operate. Containment failure was predicted to occur shortly after vessel meltthrough as the result of hydrogen burning in the upper compartment region. A summary of key event times is provided in Table 2.

Peach Bottom - TBUX

The Peach Bottom plant is a boiling water reactor with a Mark I containment design. A station blackout sequence with loss of both ac and dc power was selected as the example sequence.(3) Common mode failure of the station batteries following loss of offsite power results in inability to start the diesel generators, early boiloff of water in the vessel and reactor meltdown with the vessel at elevated pressure. Failure of the containment is assumed to occur in the reactor wetwell without leading to loss of water in the torus. Transport of radioactive material through the enclosing reactor building and refueling bay is also considered in the calculations. Key event times are provided in Table 3.

Sources of Radionuclides

Three types of radioactive materials are present within the reactor coolant system that can be released in an acident: fission products, actinides and activation products. The contribution from the activation products in a severe accident is small and is typically ignored. Table 4 shows characteristic inventories of fission products and actinides for the two reactors as calculated by the ORIGEN code. The inventories are shown in kilograms. Not all of the mass of fission product elements is radioactive; stable daughters are included. In addition to the radioactive materials that can be released to the containment, non-radioactive material, such as tin from the Zircaloy cladding, will be released which contributes to the aerosol loading in the containment.

Stage 1. Core Uncovery and Fuel Heatup

The initial stages that could precede a severe accident have been well studied and the methods of analysis have been validated against experiment. The postulated accidents require multiple equipment failures or operator errors, and go beyond what is generally referred to as a design basis accident. If the safety systems do not function properly and the reactor vessel water level drops below the top of the fuel, the uncovered portions of the fuel will heat up. If the reactor protection system has functioned, the heat source due to fissioning will stop, but decay heat from the fission products continues. For some sequences, it is assumed that the protection system does not work. At about 1800 °F (1000 °C) the reaction between steam and the Zircaloy cladding begins to become significant. The cladding oxidizes, heat is liberated (along with the decay heat) accelerating the rate of fuel heatup, and hydrogen is produced. These aspects of the accident have also been well studied in experiments and the rate of fuel heatup can be predicted with confidence.

Clad ballooning and rupture can occur in this stage of the accident resulting in the release of radionuclides that have migrated to the cladding gap, such as a few percent of the core inventories of the noble gases and iodine.

Stage 2. Fuel Degradation and Radionuclide Release, In-vessel

The melting point of Zircaloy is about 3360 °F (1850 °C). At this point cladding begins to melt and "candling" occurs. (Zircaloy drips down the fuel rods.) In addition, some dissolution of the fuel into the molten Zircaloy also occurs. Thus, at a temperature substantially below the melting point of uranium dioxide, 5140 °F (2840°C), liquification and relocation of some fuel occurs. As the geometry of the core becomes more distorted, ability to predict hydraulics, temperatures and mass redistribution of the fuel becomes more uncertain.

As the fuel further heats up, and the cladding fails, fission product vapors are released from the fuel rods to the neighboring coolant channels. A number of experimental studies have been performed to determine the rate of release of fission products from fuel at elevated temperature. The tests have included simulant fuels with surrogate fission products, irradiated fuels taken from reactors and test out-of-pile, fuels with minimal irradiation tested in-pile, and pre-irradiated fuels tested in-pile. The models (for example, CORSOR) currently in use for predicting fission product release in core meltdown accidents are primarily empirical in nature. More mechanistic models are under development, but have not yet received broad review or use. At intermediate temperatures the fission products are released according to their relative volatilities. At very high temperatures the uranium dioxide matrix can be stripped from the surface of the fuel, liberating fission products in proportion to their inventories. By the time the in-vessel melting period is completed, it is expected that a large fraction of the volatile fission products in the noble gas, iodine and cesium groups will have been released from the fuel. Predicted releases from the fuel are shown in Table 5 for this phase of the accident. It has been found experimentally that tellurium reacts with unoxidized cladding. Thus the amount of tellurium released to the channel depends on the degree of oxidation of the cladding. The less volatile species are released in smaller fractions. The predicted fractional release from fuel for the less volatile elements is highly uncertain, not only because of uncertainty regarding the chemistry and mass transport of the fission products, but also because of uncertainty about the temperature history of the fuel and the characteristics of the environment.

Stage 3. Transport in the Reactor Coolant System

As the vapors of fission products and structural materials are transported from the hot zone from which they are released (the core), the less volatile materials would condense to form aerosols. Depending on the environmental conditions, the vapors of the volatile fission products may condense on aerosols, react with aerosols, condense on cooler metal surfaces, react with surfaces, or remain as vapors. Regardless of the chemical form in which the fission products have been released from the fuel, the variety of elements may interact among themselves and become new chemical forms.

The chemical form of iodine has received a great deal of attention over the past decade. In the Reactor Safety Study it was assumed that iodine would transport through the reactor coolant system as I_2. As a result, it was believed that material retained on the surface of the coolant system would be quickly released and no retention was assumed in those calculations. Experimentation and analysis now indicate that the principal chemical form of iodine within the reactor coolant system would be CsI, but that some of the iodine could be in a more volatile form. The chemical form of iodine remains an issue, however. The transport of aerosols within the reactor coolant system is a dominant factor in determining the extent to which fission products are retained within the system. Computer models such as TRAP-MELT are used to predict the extent of deposition. The less volatile fission products transport almost entirely in the aerosol form. Depending on the environmental conditions, the volatile fission products may also transport on aerosols. As indicated in Table 5, current methods of analysis predict considerable deposition of aerosols in the coolant system for a variety of accident sequences.

If the vessel is pressurized during core meltdown, when vessel failure occurs airborne material will be rapidly released to the reactor cavity or pedestal region. Some deposited aerosols could also be resuspended at this time.

Stage 4. Release from Fuel Ex-Vessel

If the reactor vessel is under pressure at the time of vessel failure, core debris that is in molten from could be ejected from the vessel, atomized, and dispersed around the containment. The potential for high-pressure melt ejection has been demonstrated in experiments at Sandia Laboratories. The dispersal of fine fuel fragments and the reaction of the containment atmosphere with the fuel to release additional fission products represent an additional release term of radionuclides to the containment atmosphere. For example, ruthenium, which is not predicted to be released from the fuel in large quantities in the reducing environment of the reactor coolant system, could be much more volatile if exposed to the oxygen of the containment atmosphere.

If molten core debris accumulates on concrete in the containment, a vigorous interaction occurs in which the concrete is decomposed and the decomposition products are dissolved by the core debris pool. As the interaction progresses, gases from the concrete, steam and carbon dioxide, sparge the melt and oxidize the melt constituents. In addition, the gases act as a carrier of fission products from the molten core debris to the containment atmosphere. In the NRC analyses, considerable quantities of the barium, strontium, tellurium, and lanthanum elemental groups are predicted to be released for some accident sequences and types of concrete as indicated in Table 6. The results are quite sensitive to thermal-hydraulic, mass transport, and chemistry assumptions in the analysis and to the plant design.

Containment Conditions

The range of thermal-hydraulic conditions that can exist in a severe accident is broad. The two example sequences involve complete core meltdown and failure of the containment. Arrested sequences and sequences in which containment safety features operate could involve more moderate conditions. In the boiling water reactor sequence the drywell temperature late in the accident is predicted to reach 1800° F (1000° C). In the PWR sequence, very high temperatures 2700° F (1500° C) are predicted during hydrogen burns but these temperature spikes decrease rapidly.

A very high radiation environment is also predicted in the containment. Peak dose rates of approximately 10^6 R/hr gamma and 10^7 R/hr beta are predicted.

Sequoyah - TB

Figure 1 illustrates the time dependent behavior of the airborne mass in the lower compartment of containment. The sharp increase that occurs between 350 and 360 minutes is associated with core slumping and the sweepout of material from vessel by rapid steam production. The second large peak occurs during the period of rapid aerosol generation from 80 to 120 minutes following the start of core concrete attack. The third peak occurs when the reactor cavity boils dry at approximately 800 minutes. At this time the rate of concrete attack increases, releasing more aerosols, and the scrubbing effect of the overlying pool of water is removed. The average radius of the aerosols is also shown in Figure 1. The effect of aerosol growth in the reactor coolant system is evident in the larger radius of particles in the containment in the early stages of the accident. During the periods of rapid aerosol generation during core concrete attack, the particle size again increases but then decreases as the airborne mass decreases.

Figure 2 shows the airborne mass concentration in the upper compartment and mass leaked to the environment. The calculated decontamination factor for the ice bed varies from 3 to 7 over the time of the accident. As a result of deposition in the lower compartment and the ice bed, the mass concentration in the upper compartment is approximately an order of magnitude smaller than in the lower compartment. The initial rapid increase in leaked mass occurs at the time of containment failure and consists of material released from the fuel in-vessel. Most of the leaked mass arises from aerosols generated during core concrete attack, however. The ultimate distribution of the radionuclides is shown in Table 7. Note that a large fraction of the radionuclides released from the fuel in-vessel is predicted to be retained on reactor coolant system surfaces and that a large fraction is also deposited in the ice condenser.

Peach Bottom - TBUX

Figure 3 illustrates the airborne mass in the drywell as a function of time for the boiling water reactor sequence. During the period of core melting in-vessel, the pathway by which radioactive material reaches the drywell is via the safety relief lines, through the suppression pool, to the wetwell and through the vacuum breakers to the drywell. After vessel failure at 201 minutes, hot core debris begins to attack concrete in the drywell and the mass loading in the drywell increases dramatically. The peak source rate of 1.3 kg/s occurs 2 hours after the start of core concrete attack. The greater release of material during core concrete attack in the BWR analysis than in the PWR analysis is a result of the assumed type of concrete (limestone) and the greater quantity of unreacted zirconium.

After vessel failure the flow path in the containment is from the drywell down the downcomers, through the suppression pool and into the wetwell vapor space. The mass loading of the wetwell vapor space is illustrated in Figure 4. The decontamination factor for scrubbing by the suppression pool ranges from 70 to 200 during the in-vessel period of release and from 20 to 150 during the period of high release ex-vessel. The point of release for the relief line is deeper than for the downcomers which partially explains the higher DFs. The aerosol size distribution in the wetwell is also shown in Figure 4. The peak in aerosol size corresponds to a low flow condition. Even though the aerosol size in the drywell increases at the time of high mass loading (330 min), the size in the wetwell decreases. The pool DF is high at this time and effectively removes the larger aerosols from the distribution.

The airborne mass in the refueling bay and integral release from the refueling bay to the environment are shown in Figure 5. The ultimate distribution of radionuclides at the completion of the accident is tabulated in Table 8. The majority of the radionuclides released from the fuel either winds up on RCS surfaces or in the suppression pool.

Potential for Late Releases

Over an extended period following vessel failure volatile radionuclides that had been deposited on surfaces in the vessel may be evolved and released from the vessel. The extent of revaporization and release that would occur depends on the degree to which surfaces heat up, the chemical reactions that occur with the surface, flow patterns within the vessel and whether air ingress to the vessel occurs. Analyses performed by different authors span the entire spectrum from virtually no revaporization to complete revaporization of deposited iodine and cesium. Revaporized radioactive material is likely to transport as a vapor within the heated reactor coolant system but to condense or interact with aerosols when introduced into the cooler containment environment.

It is also possible that iodine dissolved as CsI salt in the BWR suppression pool or PWR containment sump can undergo hydrolysis reactions to form elemental iodine or interact with organics to form volatile organic iodides. Recent analyses at Oak Ridge National Laboratory indicated that the potential for the production of organic iodides is probably limited to less than 1%. The release of elemental iodine from water pools is, however, very sensitive to the pH of the pool and the magnitude of the radiation environment.

For pH in the neighborhood of six or lower, a major fraction of dissolved iodine could eventually be released from the pool. This is a high priority area for further research.

Significance to Reactor Safety

The understanding of the release and transport of radioactive materials within containment buildings in severe accidents has developed rapidly since the Three Mile Island accident. Processes occurring within the containment can have a major effect on the magnitude of the release of radionuclides to the environment, as illustrated by the examples in this paper. Thus improved understanding of containment processes is important to developing a more accurate assessment of risk to the public. In addition, the NRC is investigating the value of different accident management strategies that could be used to minimize the consequences of a severe accident should one occur. Thus it is important to understand accident progression and conditions in the containment in order to evaluate the effect of containment safety feature operation in developing appropriate operator actions.

References

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2. Denning, R. S., et al, "Radionuclide Release Calculations for Selected Severe Accident Scenarios", NUREG/CR-4624, BMI 2139, Vol. 2, July 1986.

3. Leonard, M. T., et al, "Supplemental Radionuclide Release Calculations for Selected Severe Accident Scenarios", NUREG/CR-5062, BMI 2160, to be published.

Table 1. Radionuclide Groups

Group	Elements
1	Xe, Kr
2	I, Br
3	Cs, Rb
4	Te, Sb, Se
5	Sr
6	Ru, Rh, Pd, Mo, Tc
7	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y
8	Ce, Pu, Np
9	Ba

Table 2. Timing of Key Events - Sequoyah TB

Event	Time, minutes
Core Uncovery	236.6
Start Melt	327.1
Core Slump	354.4
Core Collapse	356.0
Hydrogen Burn	373.2
Hydrogen Burn/Bottom Head Failure	373.6
Hydrogen Burn/Containment Failure	373.6
Hydrogen Burn/Start Concrete Attack	373.6
Hydrogen Burn	497.6
Hydrogen Burn	509.6
Corium Layers Invert	528.1
End Calculation	973.6

Table 3. Timing of Key Events - Peach Bottom TBUX

Event	Time, minutes
Core uncovery	66.7
Start melt	134.2
Core slump	167.7
Core collapse	168.7
Bottom head dryout	178.6
Bottom head failure	201.1
Start concrete attack	202.2
Corium layers invert	339.2
Wetwell failure/secondary containment failure	349.2
Hydrogen burn	349.9
End calculation	802.3

Table 4. Initial Inventories of Radionuclides and Structural Materials

Fission Products			Actinides/Structural		
Element	Mass (kg)		Element	Mass (kg)	
	Sequoyah	Peach Bottom		Sequoyah	Peach Bottom
Kr	17.0	25.7	U	89,000	140,500
Rb	18.7	23.3	Pu	596	743
Sr	60.9	62.7	Np	33	41.2
Y	29.1	36.2	Mn	--	432
Zr	227	267	Fe	8,690	5,130
Nb	3.5	4.3	Cr	--	4,140
Mo	197	--	Ni	--	2,560
Tc	47.2	58.8	Zr	23,100	65,500
Ru	132	172	Sn	332	1,050
Rh	26.6	33.2	Gd	--	287
Pd	66.8	83.2	Ag	2,290	--
Te	31.7	34.9	In	421	--
I	15.2	16.6	Cd	144	--
Xe	330	387			
Cs	166	207			
Ba	77.7	105			
La	79.2	98.3			
Ce	167	208			
Pr	64.5	80.4			
Nd	217	271			
Pm	9.2	11.5			
Sm	43.2	53.8			
Eu	11.3	14.1			

Table 5. Masses of Radionuclide Released From Fuel and Retained on RCS (by Group)

Group	Sequoyah - TB		Peach Bottom - TQUX	
	Released From Fuel (KG)	Retained on RCS Surfaces (KG)	Released From Fuel (KG)	Retained on RCS Surfaces (KG)
I	14.7	10.1	15.5	4.1
CS	178.9	134.4	215.4	133.5
TE	26.6	24.3	11.2	10.8
SR	.0	.0	.1	.0
RU	.0	.0	.0	.0
LA	.0	.0	.0	.0
NG	336.5	.0	385.4	.0
CE	.0	.0	.0	.0
BA	.9	.7	1.7	1.4

Table 6. Release During Core-Contrete Attack

Elemental Group	Mass Released (KG) Sequoyah - TB	Peach Bottom - TBUX
NG	0	0
I	0.081	1.1
Cs	1.4	15.0
Te	1.1	14.1
Sr	2.0	47.3
Ru	1.6×10^{-3}	8.1×10^{-4}
La	1.1	27.2
Ce	1.0	62.6
Ba	1.6	55.2

Table 7. Distribution of Fisson Products by Group - Sequoyah TB
(Fraction of Initial Inventory)

Species	RCS	Cavity Water	Melt	Lower Compartment	Ice Bed	Upper Compartment	Environment
I	0.66	2.5×10^{-2}	0	8.0×10^{-2}	0.21	2.9×10^{-3}	2.0×10^{-2}
Cs	0.73	2.4×10^{-2}	0	5.8×10^{-2}	0.18	2.3×10^{-3}	1.7×10^{-2}
Te	0.76	3.9×10^{-2}	8.6×10^{-2}	2.2×10^{-2}	7.9×10^{-2}	9.4×10^{-4}	7.8×10^{-3}
Sr	4.8×10^{-4}	0.14	0.83	4.8×10^{-3}	2.2×10^{-2}	2.1×10^{-4}	6.3×10^{-3}
Ru	8.0×10^{-7}	6.5×10^{-7}	1.0	4.1×10^{-7}	2.6×10^{-6}	2.7×10^{-8}	2.2×10^{-7}
La	7.6×10^{-8}	7.2×10^{-3}	0.99	2.9×10^{-4}	1.0×10^{-3}	1.1×10^{-5}	2.8×10^{-4}
Ce	0	5.4×10^{-3}	0.99	1.9×10^{-4}	8.2×10^{-4}	8.0×10^{-6}	2.3×10^{-4}
Ba	8.6×10^{-3}	7.8×10^{-2}	0.89	3.5×10^{-3}	1.5×10^{-2}	1.6×10^{-4}	4.0×10^{-3}

Table 8. Distribution of Fisson Products by Group - Peach Bottom TBUX
(Fraction of Initial Inventory)

Species	RCS	Melt	Drywell	Supression Pool	Wetwell	Reactor Building	Refueling Bay	Environment
I	2.4×10^{-1}	0.0	5.4×10^{-2}	6.8×10^{-1}	1.2×10^{-2}	1.8×10^{-3}	1.2×10^{-4}	2.6×10^{-3}
Cs	5.8×10^{-1}	0.0	4.5×10^{-2}	3.6×10^{-1}	7.9×10^{-3}	1.2×10^{-3}	7.6×10^{-5}	1.7×10^{-3}
Te	3.1×10^{-1}	2.7×10^{-1}	1.2×10^{-1}	2.8×10^{-1}	7.8×10^{-4}	1.8×10^{-2}	2.0×10^{-4}	4.3×10^{-3}
Sr	6.9×10^{-4}	2.4×10^{-1}	3.2×10^{-1}	4.3×10^{-1}	2.8×10^{-4}	4.2×10^{-3}	1.1×10^{-4}	1.7×10^{-3}
Ru	1.2×10^{-6}	1.0	1.5×10^{-7}	1.0×10^{-6}	8.0×10^{-4}	3.5×10^{-7}	2.8×10^{-9}	1.1×10^{-7}
La	9.8×10^{-8}	9.7×10^{-1}	1.3×10^{-2}	1.9×10^{-2}	1.5×10^{-5}	1.8×10^{-4}	5.2×10^{-6}	8.0×10^{-5}
Ce	0.0	9.4×10^{-1}	2.5×10^{-2}	3.7×10^{-2}	1.8×10^{-5}	3.7×10^{-4}	1.0×10^{-5}	1.5×10^{-4}
Ba	1.3×10^{-2}	4.6×10^{-1}	2.0×10^{-1}	3.2×10^{-1}	2.2×10^{-4}	3.8×10^{-3}	8.4×10^{-5}	1.2×10^{-3}

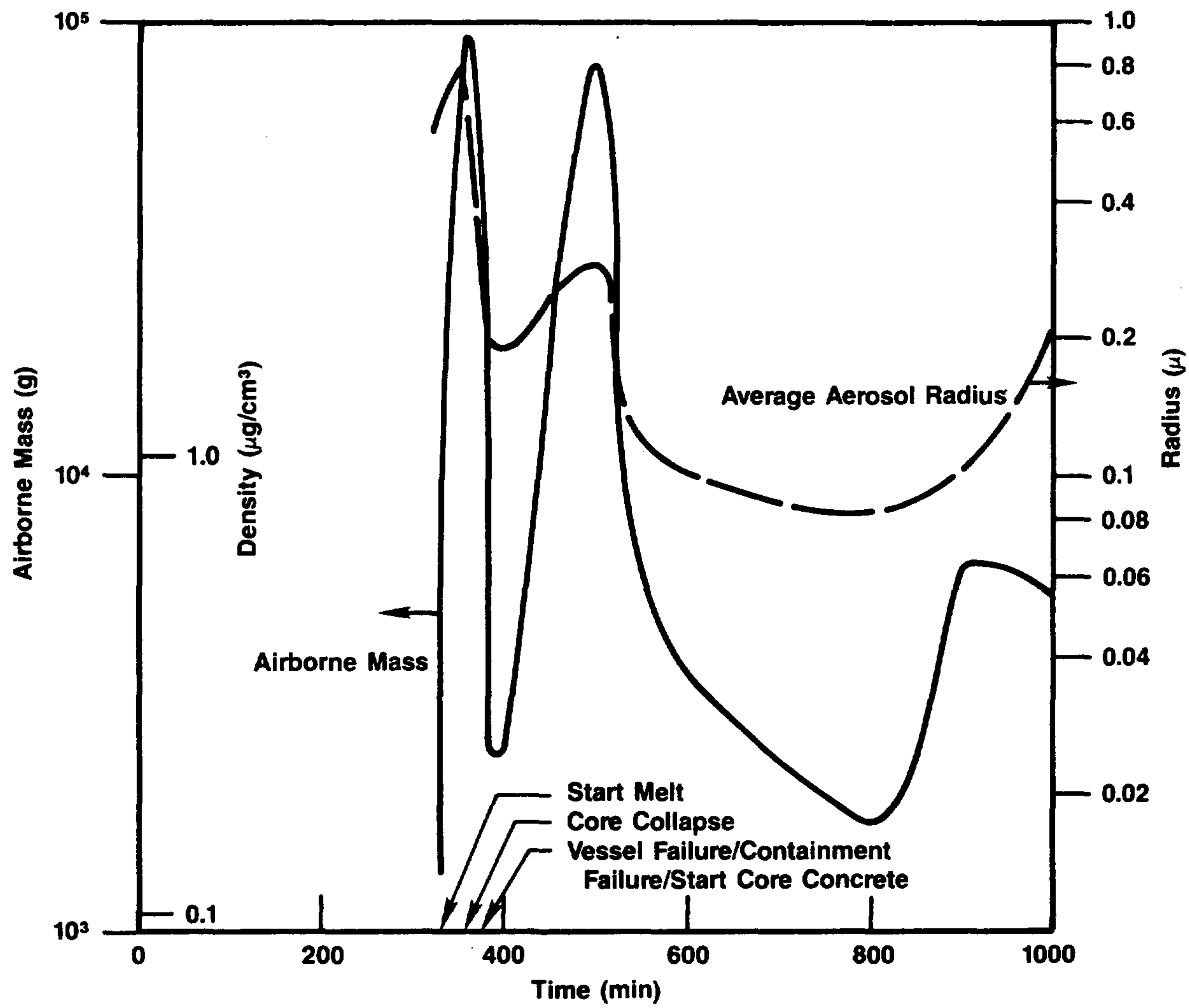

Figure 1 Airborne mass and aerosol size in lower compartment— Sequoyah TB.

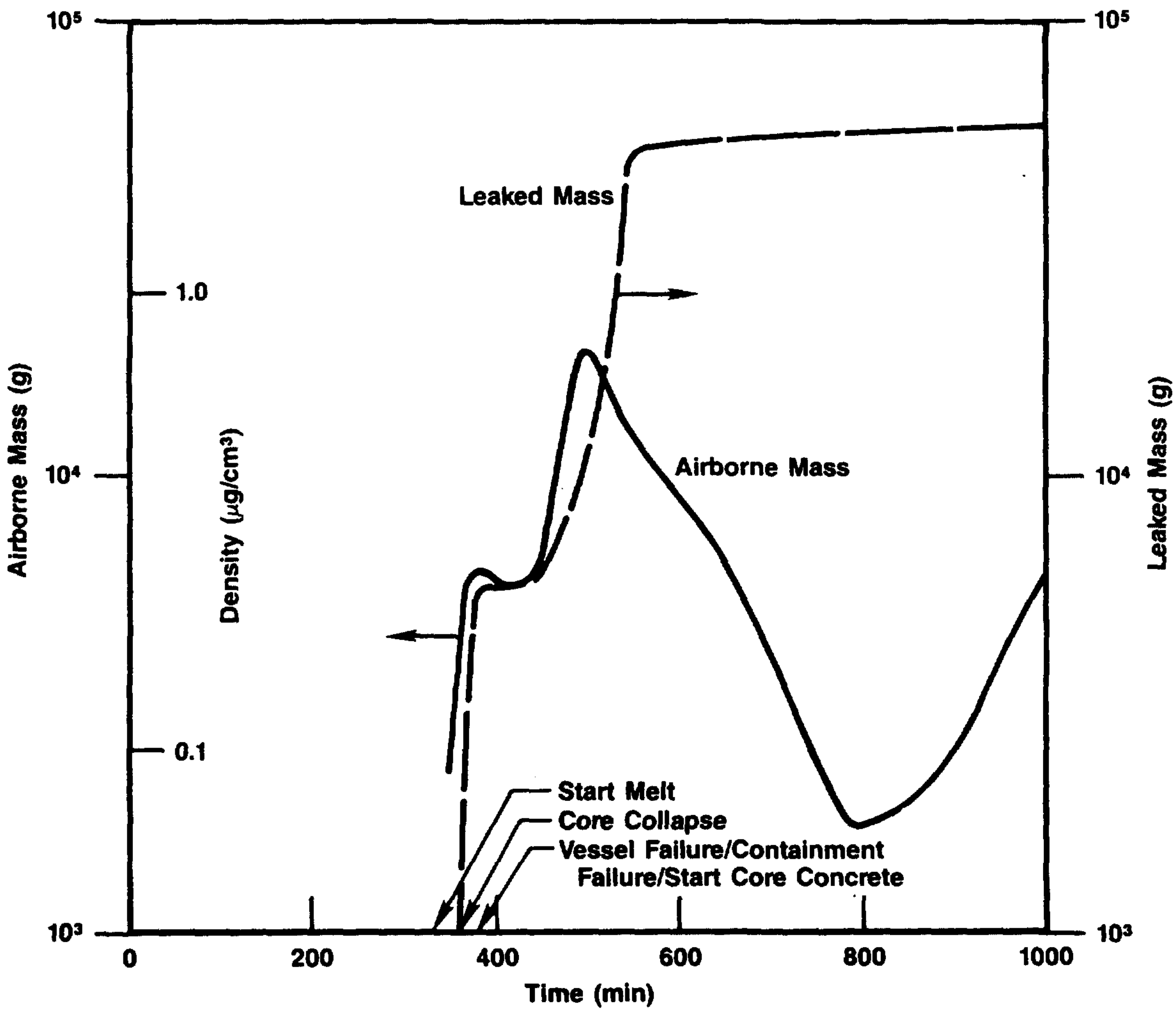

Figure 2 Airborne mass and leaked mass from upper compartment— Sequoyah TB.

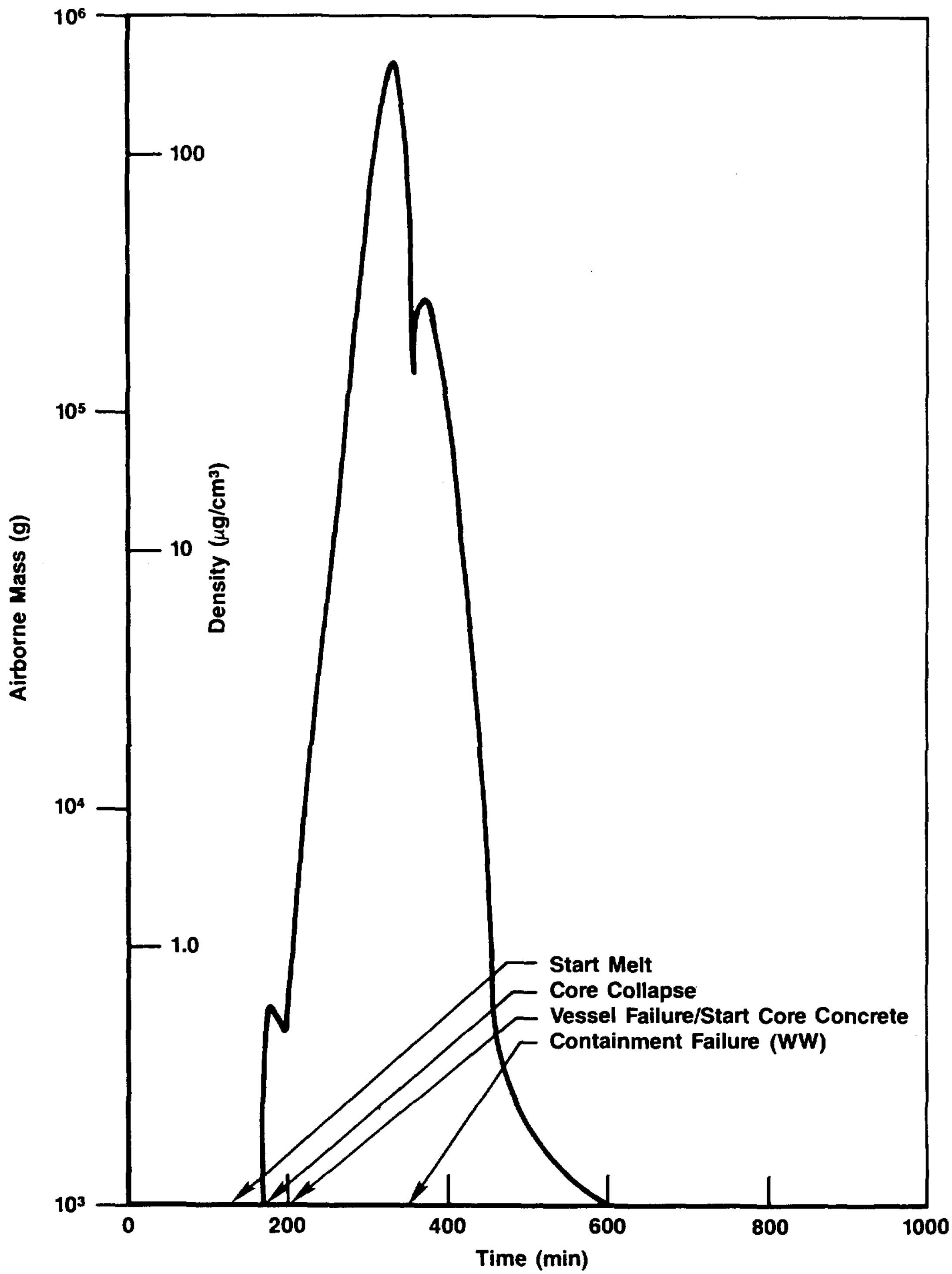

Figure 3 Airborne mass in drywell—Peach Bottom TBUX.

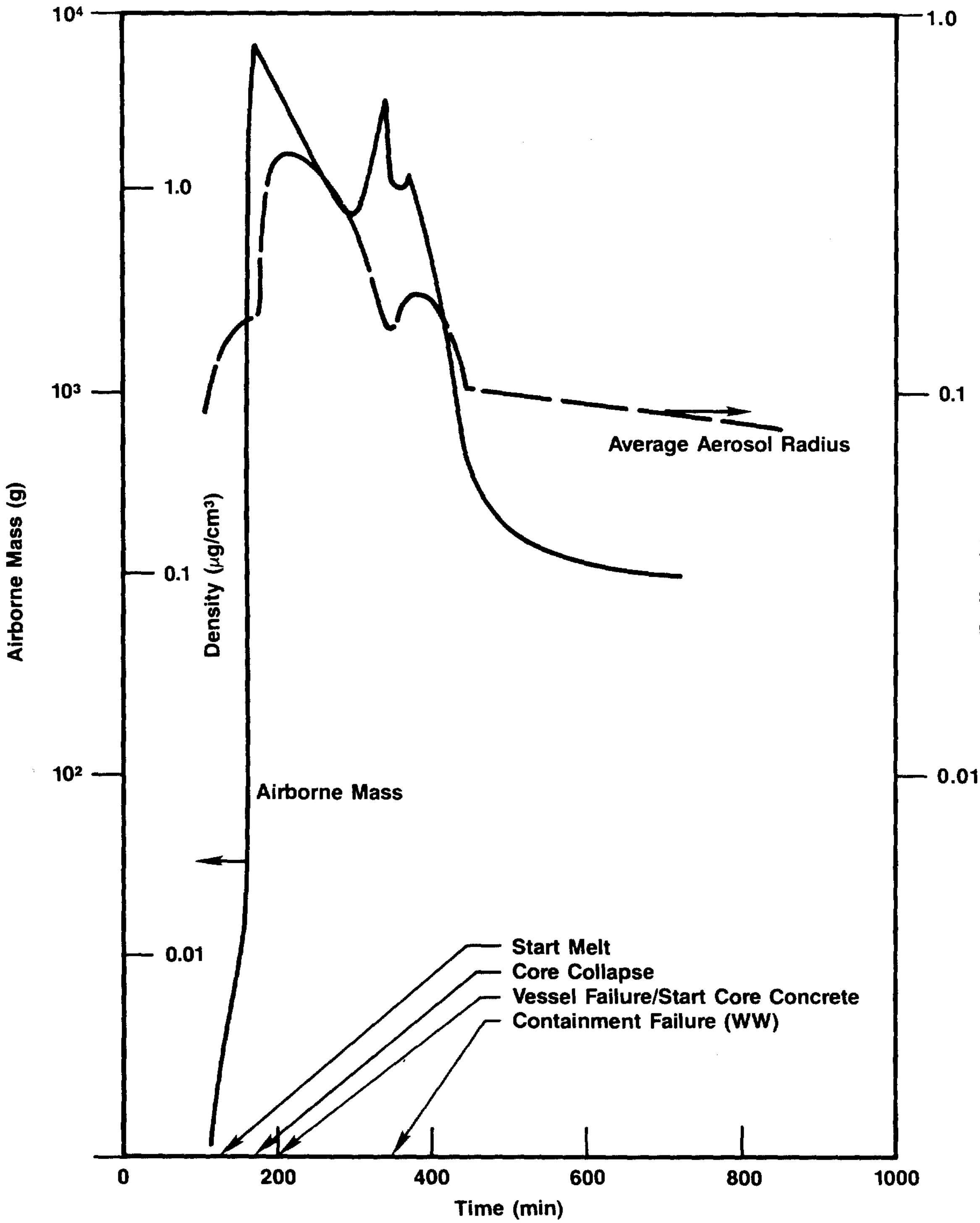

Figure 4 Airborne mass and particle size in wetwell—Peach Bottom TBUX.

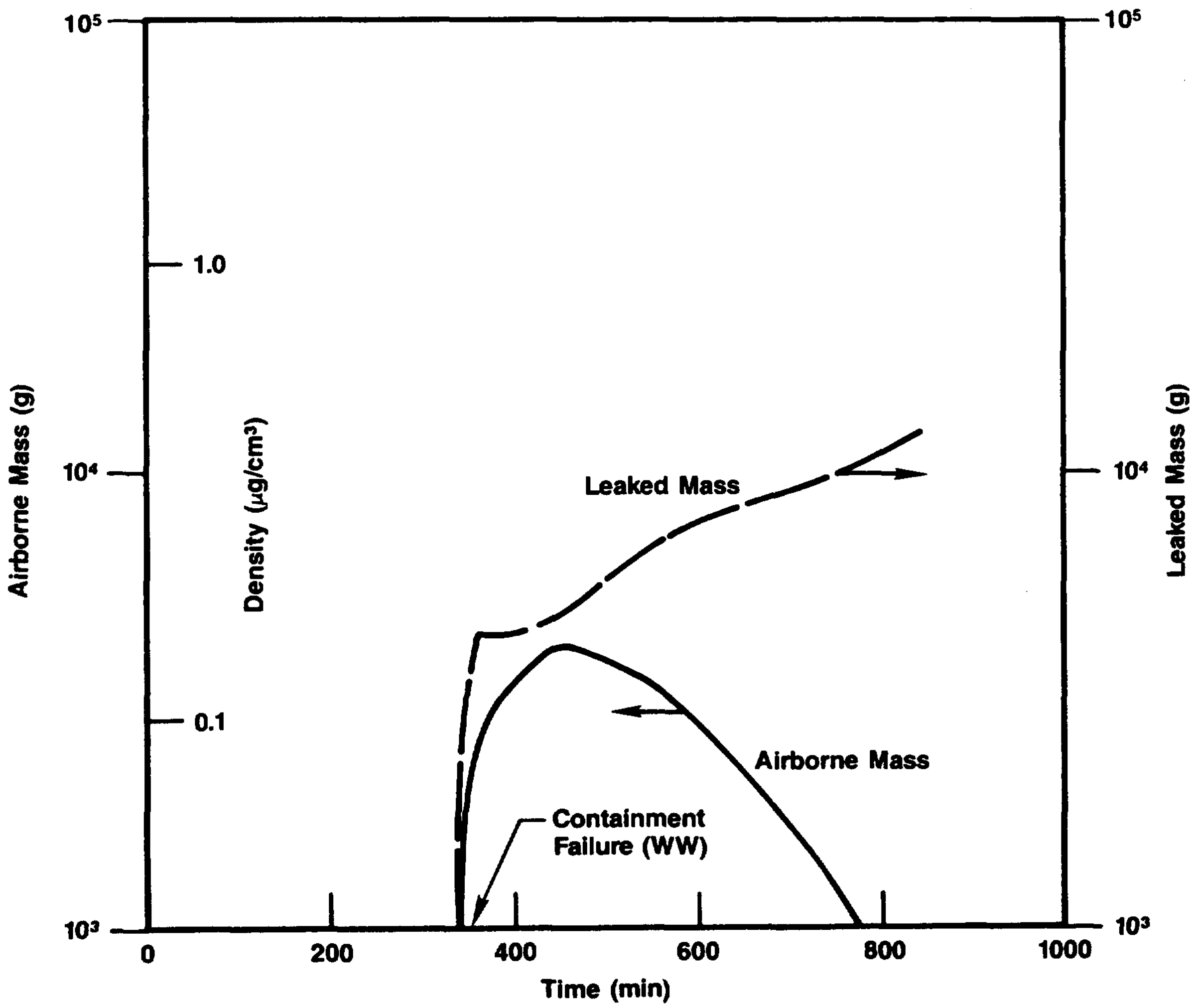

Figure 5 Airborne mass and leaked mass from refueling bay—Peach Bottom TBUX.

DISCUSSION

HULL: Two or three years ago, it seems that we were headed towards some redefinition of emergency planning zones in terms of what was believed to have been learned from the TMI-2 accident in which most of the activity which escaped from the fuel was shown to have been retained in the water and thus not available for airborne release. A study group from the American Physical Society found that the basis for this was that cesium and iodine had combined as soluble cesium iodide; but raised other contentions. Is the current NRC source term work tending toward a resolution of these contentions? Is it leading us toward a better definition whereby you can say with confidence that the emergency planning zones are large enough or should we be doing something else?

ROSS: Yes.

HULL: If so, when?

ROSS: I could probably give you a better answer around the middle of November than I can right now. As one of our previous speakers said, we do have an election. The system code package, the first slide I used, was indeed, in essence, what professor Wilson did his peer review on. Building on that, we published results in draft form last year for the five reactors. We had whole chapters on topics of applications and new source terms technology to various regulations, including emergency planning and uncertainties. We are busy redoing this draft and hope to reissue it this winter. We did show, at least to my satisfaction, that for the plants that we studied, beyond two or three miles a person was generally better off taking shelter then evacuating. The two or three mile distance depends quite a bit on the assumptions that are made, and we made a number of them. For example, do you start evacuation before or after the core starts giving significant release? What is the evacuation speed, etc? I think this winter there will exist a better technical basis for reconsidering protective action strategies. In many cases it is definitely important to move and move quickly, and not wait until the black cloud is circling over your house. You need to have adequate warning and perhaps move in advance of a release. Beyond that, I think a graded response such as sheltering would be favored. I think we will have the technical basis this winter. How to proceed is another problem.

WEBER, L.D.: I am taking a rather large step ahead, but only for the sake of discussion. When and if cleanup systems for vapors and gases released from potential accidents are developed, do you think that there will be space for installation of such equipment? You pointed out the dry well at Peach Bottom.

ROSS: I think Mr. Kovach mentioned Barseback, a system which I have seen. There is a lot of room for equipment outside. Some plants put hydrogen recombiners inside the containment already, but they are not too big. We have seen some designs where, when you have a spare penetration, you just come outside the system to the secondary auxiliary building and put in whatever system you want. I think the French put their system, at least temporarily, on the roof of the auxiliary building. So, as far as there being room or space, certainly there is. Do you need it in the U.S. reactors? That is yet to be decided and I hope it is decided technically. I think it will be a technical decision.

KUGLER: The time duration you showed between vessel failure and containment failure indicates that the wet well survives the steam explosion from the melt dropping into the wet well. Is that correct?

ROSS: Are you talking about Peach Bottom?

KUGLER: Yes.

ROSS: The geometry of Peach Bottom is such that I do not believe there is any credible way for molten materials to drop into the wet well pool. The wet well is outside and the core is in the middle. Because there is only about a foot from the bottom of the dry well to the downcomer pipes, it is possible for some of the molten materials, but not very much, to slowly run down the downcomer pipes. Eventually you might get something into the pool, but it would not be very much.

KUGLER: So, the center of the pedestal is isolated from the wet well? I recall that in the wet well analysis for WNP2 that there was a flow passage provided through the reactor pedestal so that heat sink could be utilized.

ROSS: The sixth plant, the LaSalle Plant, which I did not talk about, has only a few feet of concrete underneath, and then there would be entry into the water pool. When we do our calculation for LaSalle we are going to have to deal with that. Some people think there would not be an energetic steam explosion but Dr. Kelber, who is in the audience, has an opinion on this point. I do not know if there is time for him to express it.

KELBER: There is not enough time.