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Proceedings of the 23rd DOE/NRC Nuclear Air Cleaning Conference

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Held in Buffalo, New York
July 25–28, 1994

Edited by M. W. First

Sponsored by
U.S. Department of Energy
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International Society of Nuclear Air Treatment Technologies, Inc.
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ABSTRACT

This report contains the papers presented at the 23rd DOE/NRC Nuclear Air Cleaning Conference and the associated discussions. Major topics are: (1) nuclear air cleaning codes, (2) nuclear waste, (3) filters and filtration, (4) effluent stack monitoring, (5) gas processing, (6) adsorption, (7) air treatment systems, (8) source terms and accident analysis, and (9) fuel reprocessing.

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FOREWORD

We have learned a great deal at this and prior Conferences about the source term for power reactors being revised and refined, but we have not heard that a similar effort is under way to define a source term for decontamination, decommissioning, remediation, or for waste management in general, even though planning for these operations is well under way. The idea persists that all air cleaning and gas treatment requirements can be solved by some number of HEPA filters placed in series at the end of a pipe in spite of the knowledge that remediation, decontamination, and decommissioning requirements will be different from power reactor accident mitigation. Therefore, I was keenly interested in the panel session on gas processing as I believe that, here is where the future of nuclear air cleaning technology will reside for the next decade as we confront new problems and arrive at different solutions than we have been accustomed to accepting. We face a formidable task to make waste management operations safe, not only for the environment, but also for the people who are going to be doing the work. The West Valley operation is in the nature of a pilot study for the much larger task that must be tackled at other sites. From that standpoint, the presentations on the work at West Valley were very important.

Although HEPA filters continue to be the workhorse nuclear air cleaning device, we learned at this Conference about the vulnerability of HEPA filters to elevated temperature, humidity, dust loading, and aging, both in storage and in service. This brings up the question of how many years these filters should be kept in operation, even though they continue to pass the in-place filter test that is mandated as a periodic event. When we learn that some filters have been in service for thirty years, we can't help but wonder what their performance will be for an entire year following an in-place filter test, or how they will respond to emergencies. Do we really have confidence that this is the barrier we should be depending on? I have come to the conclusion that five years ought to be about as long as we should keep filters in service, even if they do not get loaded up. You might ask, where did the five year number come from; the answer is that, currently, we have no evidence that they will serve more than five years safely. Realistically, we do have to confront this issue with more than guesses; we need some specific goals that will make it mandatory to discard air cleaning equipment long before it becomes a questionable safety device. The many presentations and discussions at this Conference on the subjects of aging effects and residual safety factors promises that this topic will continue to receive serious research support that will help lead to resolution.

Another topic that engaged the interests of the conference was the many conflicts that exist among the many regulations, codes, standards, directives, *etc.*, that are invoked simultaneously at various sites. It will be obvious from the presentations and commentary that full compliance with all becomes impossible, and that additional complications arise from differences in the codes and standards adopted by the several nations insofar as they impact the free movement of equipment internationally. It is probable that everyone would be better off with internationally-agreed-upon standards, but it is unlikely to happen in a short time.

All of the papers in the reprocessing session came from nations other than the United States. This, of course, is the legacy of Jimmy Carter, the self-styled nuclear engineer, who also canceled the Clinch River LMFBR. I do not see any change in the public attitude regarding these matters; at least not in the immediate years ahead, unless all of us take it as a personal duty to propagandize for the safe operation of nuclear power in the United States, in Germany, and in other places where nuclear power is under severe attack. I think it is significant that in Japan and in France, nuclear power is doing very well. It simply means that we, in the United States, may have to wait until we

burn up all the natural gas and all of the oil in the ground before we again become serious about alternative energy sources. I hope we do so before we get into a terrible energy crunch.

For the future, it will be important for the Program Committee to encourage operating people to come forward with reports that can assist others who are likely to confront similar problems. Should the Air Cleaning Conferences become a forum only for research papers, people who conduct operations and those who do the engineering of the systems will lose interest in the Conferences. It is unfortunate that operating people do not write a lot of papers. Unlike research people, for whom a paper is the product, operating people have a different standard; they must keep the plant operating and make prompt repairs. The point is, those of us who are able to write with great facility should not only urge our operating people to come forward with their important information, particularly the "lessons learned" sort of thing, but also help them put it together so they can make presentations at future Air Cleaning Conferences. That way, we can get the benefit of the knowledge we need to learn about but that is currently locked away in the minds of individual operating people. Please keep this plea in mind, and when you find somebody who has learned something very interesting about operations, I hope you will help them prepare a paper for presentation at this forum. We should always be on the alert for reports of new air cleaning methods.

I would like to extend my sincere thanks to all of the members of the Program Committee who worked diligently over the past couple of years to put together the program of the 23rd Conference and who helped conduct it in an orderly manner. I am very grateful to each of them for their devotion to the Air Cleaning Conferences and for their willing assistance to the Chairman.

Richard Porco, chairman of ISNATT and a member of the Conference's Program Committee, deserves special mention for his diligence and skill in organizing the Conference exhibits. Fees from exhibitors help support the Conferences.

Thanks from all attendees, as well as the Chairman, are due Mary Kovach for her able assistance at the registration desk and to Julia First for her kind management of each morning's hospitality gathering of family members.

The Chairman extends his special thanks to Suzanne Reine for her able handling of all the business affairs of the Conference and for her careful and timely assembly of the camera-ready master copy of the Proceedings.

Melvin W. First
November 22, 1994

SESSION 1

OPENING OF THE CONFERENCE

Monday: July 25, 1994
Co-Chairmen: J. F. Leonard
K. Duvall

OPENING COMMENTS OF SESSION CO-CHAIRMAN LEONARD

A BRIEF HISTORY OF THE AIR CLEANING CONFERENCES
M. W. First

THE INTEGRATED MELTER OFF-GAS TREATMENT SYSTEMS AT THE WEST VALLEY DEMONSTRATION PROJECT
R. F. Vance

OPENING COMMENTS OF SESSION CO-CHAIRMAN LEONARD

As has been mentioned earlier, we have a great challenge ahead of us in Environmental Protection and Waste Management. It is an area that has been growing by leaps and bounds. A great deal of the budget of the US Department of Energy is now going in that direction. The DOE organization that I am in, Defense Programs, used to be one of the largest organizations but its emphasis is now going down and emphasis on Environmental Protection and Waste Management is going up. A few years ago, the Department of Energy, in order to take on this challenge, which is enormous, not only for the Department of Energy, but for the entire United States, began doing the research needed to figure out how to handle these materials and how to accomplish it in a safe manner. They have received DOE's foremost emphasis.

A Brief History of the Air Cleaning Conferences

Melvin W. First
Harvard Air Cleaning Laboratory

Introduction

I have been asked to prepare a history of the air cleaning conferences. Undertaking such a task is, of course, a wonderful opportunity for reminiscences and a chance to retell old war stories. I must admit that it has taken much longer than I anticipated because I found myself so completely engrossed rereading the old records that time seemed to stop, although the hours passed. But a history of the nuclear air cleaning conferences means more than a stroll down memory lane. The 23 recorded air cleaning conference proceedings reflect an important aspect of the history of major nuclear developments, both military and civilian, because engineered safety features designed to prevent dispersion of radioactive products to the environment have always been a necessity for progress in this field. For this reason, I hope the history of the nuclear air cleaning conferences will not only be enjoyable, but also have meaning for young people entering this field.

The air cleaning conferences were an outgrowth of the operations of the U. S. Atomic Energy Commission's (AEC) Stack Gas Working Group established in 1948 to review air cleaning operations at AEC installations. AEC's Division of Engineering sponsored and funded air cleaning research and development at Harvard University's School of Public Health, beginning about the same time. In addition to research and development, the Harvard contract called for consulting and educational services. The latter provided the opportunity for meetings devoted to information on air cleaning that could be applied to ongoing and anticipated nuclear operations.

The Early Meetings

The first meeting was held in Boston at the Harvard School of Public Health in 1951 under the direction of Professors Philip Drinker and Leslie Silverman. The state of knowledge at that time was such that the first meeting was devoted to sharing information on air cleaning in a seminar and laboratory format. No formal report was published. The only record is a photograph of the participants. One session was devoted to a discussion of classified topics; specifically, current operations requiring improved air cleaning equipment. Those without clearance were excluded by a security force that guarded all doors and windows of the school's lecture hall.

After air cleaning research programs began at various AEC laboratories and production sites, the agenda of the meetings expanded to include the presentation of formal papers as well as the exchange of operational information. A mixture of basic research, product and process development, and practical operational experience has characterized all subsequent conferences.

The 2nd Air Cleaning Seminar, exclusively for AEC personnel, was held at the Institute for Atomic Research, Iowa State College, Ames, Iowa, September 15-17, 1952. Its purpose, once again, was to acquaint personnel at national laboratories and AEC contractors who did not have a background in this field with the fundamental aspects of air and gas cleaning. In addition, it was an opportunity to discuss with individuals responsible for selecting and applying the various air cleaning devices new problems, recent applications, and the performance of equipment in the field. The first half day was devoted, again, to a review of the properties of aerosols, descriptions of basic air cleaning principles, and the major types of air cleaning equipment. Those attending were presented a copy of the newly

printed Handbook on Air Cleaning, the first of an extensive series of AEC-sponsored technical monographs covering many aspects of the developing technologies associated with environmental protection and occupational health and safety. The Handbook on Air Cleaning remained the principal reference source until the publication of the Nuclear Air Cleaning Handbook, authored by Burchsted and Fuller, 17 years later. An idea of the state of knowledge about air and gas cleaning technology in 1952 is conveyed by the information 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 more.

Among the topics covered during the second meeting, in addition to Harvard's air cleaning research studies, were reverse-jet filters for uranium refining operations at Mallinckrodt Chemical Co. and deep-bed sand filters at Hanford for treating dissolver offgases. It is interesting to learn from C. L. Lapple's account of the deep bed sand filter development at Hanford that it took but 3½ months from the start of experimental work to completion of the first full scale unit. Those were, indeed, heady times. Today, current projects of that magnitude will seldom take less than a year to clear the paper work to authorize and fund a study of project feasibility.

A part of the second meeting was concerned with removal of bacteria and bacteriophages from air by glass fiber filters; work conducted at Camp (later Fort) Detrick. Reports from Detrick of high efficiency air filtration for capture of biological agents continued for many conferences and ultimately contributed importantly to the development of biological safety cabinets. Other major topics of discussion were aerosol research at the University of Illinois and AEC-sponsored incinerator R&D programs at the U. S. Bureau of Mines, plus reports of incinerator operation (uniformly poor) at Los Alamos and Argonne National Laboratories. In fact, a review of later air cleaning proceedings shows a 40-year effort to design a safe, functional, and cost-effective unit for burning nuclear waste. Some never got built, some that were built never operated on nuclear waste, and the rest were soon abandoned. Nevertheless, interest in waste incineration persists as the search continues for a cradle-to-grave low-level rad-waste management system.

The reason I am spending so much time on the 2nd Seminar, when there are 21 more to go, is because I want to make the point that many of the problems that plague us today were recognized early on, in what we were then fond of calling "the nuclear age," and in spite of good intentions, we failed to solve them all before new ones caught up with us.

For the next few meetings, attention continued to center on aerosol sampling techniques and application of the chemical industry's wet gas cleaning methods to an atomic energy program that was mainly concerned with weapons-centered production activities. Two matters of concern were the release to populated areas of natural uranium compounds from ore processing and chemical purification steps, and the release of beryllium dust from ore refining and metal fabrication. In addition, there were needs for air cleaning research activities at the contract universities and national laboratories that involved the use of small quantities of radionuclides of an extremely diverse nature.

Starting with the third meeting, papers on the development and growth of the civilian nuclear power reactor program described methods for estimating fission product release rates and for collecting fission products, particularly radioiodine. For the latter, attention was directed toward the characterization and application of activated charcoal.

The third meeting took place at Los Alamos Scientific Laboratory, September 21-24, 1953, and was officially designated a "Conference" rather than a "Seminar." It included air cleaning status reports from nine AEC sites, four government agencies, (including the U. S. Weather Bureau and Bureau of Mines), six industrial companies, and five university research groups. Equipment manufacturers were invited to attend for the first time and Cambridge Filter Corp., Flanders Mill, Inc., and Mine Safety Appliances, Co., (all manufacturers of high efficiency (HEPA) filters), responded. Topics included the removal of radioiodine and particles from process gas streams at Hanford, the clean-up of stack gases from the Oak Ridge National Laboratory's (ORNL) gas cooled reactor with high efficiency filters, and the design of new ventilation and air filtration systems at Rocky Flats. The last topic was classified and not included in the unclassified proceedings of the Conference. As a sign of the euphoria then current regarding the nuclear industry, one contribution came from the aircraft nuclear propulsion program.

The 4th Conference, held at Argonne National Laboratory during November 1955, followed the same site reporting format of prior conferences. Prominent among the presentations were atmospheric diffusion studies of stack gas emissions that had been conducted at several AEC sites. Reports of blast effects on what were called AEC Type 1 Absolute Filters made their first appearance at this conference. Many more reports of filter resistance to shock followed and the subject remains a current interest, as may be seen from the Proceedings of the 23rd Conference. A method for changing contaminated AEC Type 1 filters without interrupting system airflow was described; it was a good try but it didn't make it. A functional system of this nature, from Harwell, was described at a later conference. Several of the presentations were classified, but all were later released. One described recent fires at four AEC installations and the effects on filter systems. A conclusion was that "heroic systems would be needed to protect large banks," and recommendations were made to make filter systems fire safe. Fire safety became a major topic at the 5th Conference and at all subsequent air cleaning conferences; it was on the agenda of the 23rd Conference.

Newly introduced topics of note at the 5th Conference were particulate airborne contamination at the startup of the experimental boiling water reactor (EBWR) at Argonne, and a Harvard Air Cleaning Laboratory progress report on filtration of freshly generated Nak (a sodium-potassium alloy) fume. Ultimately, Nak ruined a couple of nuclear submarines and burned down the Air Cleaning Lab. The Air Cleaning Lab made a recovery and later spent many years experimenting with molten elemental sodium coolant fires without incident. Air cleaning was being recognized as a significant cost factor in all aspects of nuclear operations, and cost accounting studies, then in progress, were reported. The 5th Conference was a totally open meeting.

The 6th Conference met at Idaho Falls during July 1959. It brought to the attention of the group a number of new topics of special importance, some of a decidedly disturbing nature, i.e., reports of radioactive particle fallout from tall discharge stacks and reports of serious defects in absolute filters delivered to AEC sites. Remedial measures for filter quality assurance included a mandatory retest of all filters at Hanford, WA, or at the Chemical Warfare Laboratory, before delivery to the user facilities, preparation of an Underwriters Laboratory standard for fire resistant filters, and a call for an industry standard for a fire-resistant filter. Euphoria continued, nonetheless, with reports of new air cleaning requirements for aircraft nuclear propulsion. Studies of adsorbents at several sites, principally activated charcoal, for retention of noble gases (in delay beds) and iodine (for permanent retention), were reported.

This was the first conference to which visitors from other countries were invited and a paper

entitled, Air Cleaning Practice at ACE, Aldermaston, was presented. This became a tradition that has continued unbroken through the years. At the 23rd Conference there was a paper on the performance of HEPA filters from AEA-Harwell. The 6th Conference was the first at which discussions following each presentation were tape recorded and reported in the proceedings.

The practice of publishing in the Proceedings the names of those attending each conference did not begin until the 6th Conference, so we do not know all those who were present at earlier conferences except for the listed speakers and those we are still able to identify in the photo of the reception held during the 1st Seminar. In the attendee list published in the 6th Conference Proceedings there are two names that were present then and were also present at the 23rd Conference: Humphrey Gilbert, formerly with the AEC Office of Operational Safety, and Wendell Anderson, formerly with the Naval Research Laboratory. Both have taken a major role in the planning and conduct of the air cleaning conferences and continue to do so to the present.

After nuclear equipment suppliers were invited, and they began to attend the air cleaning conferences in substantial numbers, Humphrey Gilbert made a habit of inviting them, along with some concerned government regulators, to his room in the evening for a relaxing drink and some sober talk. Many serious problems were worked out in this way but it was recognized that a mechanism was needed for announcing and publishing the results of these informal, but important, information exchanges, so an election was held to choose a reporter. Wendell Anderson was elected to take notes, present the information to the entire conference, and write up his notes for the proceedings. In retrospect, my recollection is that Gilbert and I voted for Andy and he won the election two to one. In any event, the Government-Industry meetings were so successful and important that they were later conducted in an open forum and became a regular program event at each conference, from the 10th through the 19th. What happened then, will be taken up later.

The 7th Conference took place at Brookhaven National Laboratory during October 1961. Visitors from the UKAEA (United Kingdom Atomic Energy Agency) and the Center for Nuclear Energy (CEN), Mol, Belgium, attended and described new techniques for gaseous fission product containment. Methods for retention of radioactive gaseous and particulate releases occupied a major share of the conference agenda. It was also reported that the filter retest stations in operation at Hanford, WA and Edgewood, MD were responsible for a very satisfactory decrease in new HEPA filter rejections. This topic was revisited during the 23rd Conference. To eliminate damage from faulty filter installation procedures, an installation manual was prepared and a portable in-place filter testing procedure, developed at the Naval Research Laboratory for submarine service, was described. Other topics highlighted at the conference were stack testing and atmospheric dispersion, environmental air monitoring, radioactivity control following nuclear accidents, and a variety of cleaning techniques that could be applied to incinerator offgases. The 7th Conference was the first to be summarized in the Nuclear Safety journal (March 1962). The content of all subsequent conferences has been reported there.

At the 8th Conference, held at Oak Ridge National Laboratory during October 1963, increasing attention was focused on power reactor fission product releases from in-pile burning of reactor fuels; what we now refer to as the "source term." This is, of course, essential information for the rational design of accident control air cleaning systems. A review of the air cleaning conference proceedings shows many changes in the source term over time as new data became available and analytical procedures were improved. The somewhat different concepts of accident "containment" vs. "confinement" were subjects of vigorous discussion. It was noted at this conference that the first

commercial chemical reprocessing plant was under construction in New York State (West Valley) and that future releases of radioactive noble gases from such sources could significantly increase atmospheric radioactivity world wide. At the conclusion of the 23rd Conference a site visit was conducted at West Valley to view the work being conducted there to decontaminate and decommission the facility and embed the residual high level waste in glass logs.

Four panel sessions were held on in-place filter testing and the development of fire resisting HEPA filter specifications. Panels proved to be an excellent way to explore emerging subjects and became a feature of all subsequent conferences. One of the unsettled questions brought out at this meeting, and that evoked many future research reports and panel discussions, was the chemical nature of the iodine released from fuel and its transformations between organic, inorganic, vapor, and particulate forms. In addition to the U. K., there were attendees from Canada, France, and the Netherlands.

The 8th Conference was the last one organized and attended by Leslie Silverman, "the originator and guiding spirit of the USAEC Air Cleaning Conferences." He died shortly before the 9th Conference, held in Boston during September, 1966.

Major topics on the agenda of the 9th Conference were source terms for water and gas cooled reactor accidents and for controlling sodium fires in liquid metal cooled fast breeder reactors (LMFBR); the chemical forms of iodine and their removal from air and steam; specifications for fire resistant HEPA filters; and studies of their resistance to heat, fire, and explosion. Definitive over-pressure tests were reported on all sizes of HEPA filters constructed in accordance with military specifications.

In-place filter testing continued to be a subject of interest and technical development and, for the first time, there were reports of in-place test procedures for activated carbon adsorber beds. Other new topics were air cleaning systems for vented containment and floating roof containment vessels for holding fission product emissions pending treatment. For the first time, there was an Open-end Session for reporting short papers, research in progress, and puzzling phenomena needing discussion. Nationals from the U. K., France, and Canada were present. For the first time, representatives attended from Denmark, Switzerland, Germany, and Mexico. The names of J. L. Kovach and Dade Moeller appear on the attendance list for the first time at this conference. Both made outstanding contributions to this and all following conferences.

The 10th Air Cleaning Conference turned out to be two related but separate meetings. With great confidence and an excess of naiveté, the planning committee for the 10th Conference decided it was time for an international air and gas cleaning meeting in recognition of the increasing international interest in the AEC air cleaning conferences, plus the fact that important air cleaning research was now underway in many countries. AEC Headquarters readily accepted the plan and offered assistance, an offer that could not be refused because, unbeknownst to the Conference Program Committee, the AEC had already agreed that it was time for the International Atomic Energy Agency (IAEA) to hold an international air cleaning conference. What better time than the upcoming 10th AEC Air Cleaning Conference and what better place than the United Nations Headquarters in New York City! Learning that the Conference Program Committee no longer had control of the program and would not be permitted to publish the proceedings, they took advantage of a day set aside for foreign nationals to visit Brookhaven National Laboratory and held a one-day 10th Conference at the nearby Waldorf-Astoria Hotel. It was well attended by all the U. S. delegates plus many international

visitors. This program, later published as the Proceedings of the 10th U. S. AEC Air Cleaning Conference, concentrated on a few topics of current importance: air cleaning system design, filter performance, and adsorbers for iodine. The International meeting at U. N. Headquarters was published by the IAEA and is entitled, "Environmental Aspects of Nuclear Power Stations", Vienna, 1971.

The Middle Period Conferences

The 11th AEC Air Cleaning Conference took place at Richland, Washington during August/September 1970. Major technical presentations included sodium aerosols from LMFBRs, water cooled reactor fission-product release estimations, removal of noble gases, iodine characterization and removal, filtration studies, instrument development and evaluation, effects of airborne radioactive materials, and standards development. Progress on engineered systems for the removal of the noble gases, krypton and xenon, from gaseous effluents was a major highlight. A computer model to simulate ignition of activated carbon adsorber banks by fission product decay heat was presented and, for the first time, the severe difficulties associated with extinguishing carbon fires were discussed. A number of presentations described the properties of inorganic adsorbents for iodine that would not be susceptible to ignition.

The 12th Conference was held during August 1972 at Oak Ridge National Laboratory. It was the last time the Air Cleaning Conference met at an AEC site. The large increase in the number of attendees from Europe, Asia, and South America made it expedient to meet at locations closer to an international airport.

In response to the announcement of plans for constructing a full scale LMFBR to be operated by TVA, plus knowledge that construction was underway on the Fast Flux Test Facility (FFTF) in Hanford, WA, an entire session was held on LMFBR accident analysis. In addition to numerous presentations on iodine, noble gases, and high efficiency filtration of fission product aerosols, entire sessions were devoted to underground uranium mine air cleaning, the function of air cleaning technology in helping to achieve "As Low As Reasonably Achievable" (ALARA) conditions, and air cleaning design for plutonium processing facilities (this was a panel session). There was a separate Government-Industry Conference on adsorbers and adsorbents, reported by Clifford Burchsted, in addition to a Government-Industry Committee on Filters, reported by Wendell Anderson.

A special ceremony was held at the beginning of this conference in recognition of the retirement of Humphrey Gilbert from the AEC. He was cited for having "sparked these conferences" as well as "a substantial portion of the research and quality control efforts that have transformed the air cleaning field in the past years."

The 13th Conference was held in San Francisco during August 1974. Dade Moeller presented the first of several analyses of air cleaning and airborne waste management system failures in U. S. nuclear facilities. This one covered 1966 to 1974. It was based on reports to the AEC of abnormal events. Over 75% of the abnormal events were directly attributable to errors in the design or operation of the systems. Intended as a remedial response by the AEC, the newly-published Regulatory Guide 1.52, entitled "The Design, Testing, And Maintenance Of Air Cleaning Systems In Water Cooled Nuclear Power Plants", received extensive discussion at several sessions during the conference. The first paper on nuclear power plant control room system design was presented at this conference. Many additional reports on this subject were given during later conferences.

Plutonium continued to be a major topic, along with filters, iodine, and fire protection.

Deterioration of filters and adsorbents during storage and in normal service was the subject of several papers and more was heard about their deterioration at the 23rd Conference. At the Government-Industry Meeting on Filters, Qualified Product Listings (QPLs), under revised standards for nuclear grade filters, were discussed. At the Government-Industry Meeting on Adsorbers, weathering of charcoal was shown to be a source of serious deterioration.

The 14th Conference took place in 1976 after the AEC was abolished and replaced by the U. S. Nuclear Regulatory Commission (NRC) and the U. S. Energy Research and Development Administration (ERDA); the latter agency was the sponsor of the 14th Conference. Under the stimulus of a continuing oil crisis, the design of the Clinch River Fast Breeder Reactor was accelerated. But at the same time, anti-nuclear groups were becoming a major deterrent to the continued expansion of civilian nuclear power. Among concepts first introduced at this conference was a plan for floating nuclear power plants that would be sited offshore to help reduce population exposure should a major accident occur. It would also be far from land-based voters. Although viewed as an innovated approach, this idea was never applied. However, a September 1994 item in the Wall Street Journal reported that "Russia's nuclear industry hopes to build four floating reactors to supply electricity to Siberia."

Gaseous waste treatment was always a principal focus at these conferences, and, at this one, volume reduction of solid waste products and their preparation for long and safe storage, including vitrification, constituted a major part of the proceedings. System protection from fire, explosion, and natural disasters received renewed attention, at least partially as a result of the Browns Ferry Nuclear Plant fire in 1975 and the publication that same year of the Reactor Safety Study, WASH-1400, that contained analyses related to generic types of failure mechanisms that could lead to failures of air cleaning systems. Of outstanding importance for nuclear engineered safety systems was the announcement that nuclear standards N509 (Nuclear Power Plant Air-Cleaning Units and Components) and N510 (Testing of Nuclear Air Treatment Systems) had been approved by the American Society of Mechanical Engineers' (ASME) Committee on Nuclear Codes and Standards and were awaiting approval by the American National Standards Institute (ANSI). An explanation of proposed modifications to R.G. 1.52, designed to respond to criticisms and to resolve possible conflicts with N509 and N510, was presented. Important changes included substitution of air cooling for water sprays to prevent decay heat ignition of carbon beds. It was announced that the custody of N509 and N510 would henceforth be the responsibility of the ASME Committee on Air and Gas Treatment Equipment (CONAGT), a group that has since that time played a prominent role at the air cleaning conferences.

The 15th Conference was held in Boston, MA during August 1978. It was renamed the DOE Nuclear Air Cleaning Conference. Not only was the new U. S. Department of Energy the successor agency of ERDA, but the word "nuclear" was added when the DOE designation in the call for papers brought in offerings on pulverized coal flyash, shale processing effluents, etc. The principal technical subjects considered at this conference were reactor safety R&D, regulatory activities to avoid accidents, and waste disposal, a topic that has grown in importance, year-by-year, at these conferences. Of special interest were explanations of the structure of the new DOE, at that time less than a year old, and planned activities within the newly organized management program for airborne radioactive waste.

A new report on failures of nuclear air cleaning systems listed over 1,200 that had occurred during 1975-1978. Inasmuch as a large fraction of the failures occurred in equipment installed to

sample and monitor air systems, the paper's author (Dade Moeller) cautioned that "inaccurate data on the performance of ... systems can lead to as many problems as inadequacies in the systems themselves." This was a wise and prophetic statement, as the Three Mile Island (TMI) event demonstrated less than one year later wherein inaccurate instrument readings played a major role in the disaster.

In a surprising announcement, NRC stated that their evaluation of recent test results from DOE's three filter test stations led them to conclude that it was no longer necessary for nuclear power plants to have their HEPA filters pretested prior to acceptance and installation; from this time on, in-place tests alone would provide confirmatory evidence of filter integrity. For the first time, minipleat filters of European design, having higher airflow capacity than the standard U. S. filter, were described, and representative performance data were presented.

The introduction of a new continuous automatic particle sampling, counting, and sizing instrument, the single particle intercavity laser spectrophotometer, developed at LASL (Los Alamos Scientific Laboratory, currently Los Alamos National Laboratory), made it possible to check more accurately the size of the monodisperse 0.3μm DOP aerosol used for bench-testing HEPA filters. Particle size was found not only to be off the mark but to vary from machine to machine. This discovery shook the foundation of HEPA filter technology and stimulated many re-studies of filter performance. Aerosol computer codes, designed to simulate the dynamic behavior of products released during core descriptive accidents, made a first appearance at this conference.

The 16th DOE Nuclear Air Cleaning Conference took place during October 1980. The full impact of the TMI-II accident figured prominently in the presentations at this meeting. Of special importance to the delegates were the sessions on air cleaning system performance at TMI-II. It was explained that noble gas leaks were the principal sources of radiation release to the environment but that HEPA filters prevented release of all particles, and that adsorbers reduced iodine release to a small value. Nevertheless, the TMI-II event initiated reports of a number of new U. S. studies of passive air cleaning devices, including deep sand filters that might be used to filter vented containment atmospheres. Many papers from Europe on vented containment were given at subsequent conferences.

A 1980 finding by the national toxicology program that DOP was a potential human carcinogen sent another shock wave through the nuclear air cleaning community and stimulated studies to find suitable substitutes for test stand and in-place filter testing. The search lasted 14 years. At the 23rd Conference there was a report from the Edgewood laboratory that they had, at last, discovered a perfect substitute, Emery 3004, a poly-alpha olefin synthetic lubricant. Simultaneously, the U. S. Environmental Protection Agency announced that they were considering the removal of DOP from the toxic products list.

Many of the presentations on reprocessing of spent fuels were now coming from Europe as this activity came to a halt in the U. S. during the Carter administration (1976-1980). In addition, the Clinch River LMFBR was cancelled, although the FFTF program at Hanford continued for a while. As a result, funding for most of the air cleaning research studies in the U. S. decreased or terminated, and a major share of the activities in this field shifted overseas. This was the first conference since the 10th to feature critical reviews. Three were given: on radioiodine control, aerosol filtration, and noble gas treatment systems. This was the first conference that was immediately preceded by a regularly scheduled CONAGT meeting, and CONAGT organized an entire session at the conference on the topic of nuclear air cleaning standards. ORNL provided an author, title, and permutated word

index to the proceedings of all the nuclear air cleaning conferences up through the 16th. It was included as a part of the Proceedings of the 16th Conference and was designed to make the vast archive of information readily available for consultation, reference, and review.

The 17th DOE Nuclear Air Cleaning Conference was held in Denver, CO during August 1982. A new DOE program in the field of HEPA filters was announced. It involved the development of a set of DOE standards for filter testing that would use the previously cited single particle intercavity laser spectrophotometer in place of a total light scattering photometer and would focus exclusively on the least filterable particle size, approximately 0.09 μm. NRC announced an initiative to prepare Revision 3 to R.G. 1.52 within 1½ to 2 years.

The conference agenda included sessions on the redefinition of nuclear power plant source terms based on studies of the TMI-II accident; the behavior and transport of specific radionuclides under accident conditions; performance standards, procedures, and instruments for evaluating the efficiency of air cleaning systems; air cleaning requirements for spent fuel reprocessing; special approaches for the control of ^{14}C, radioiodine, and the noble gases; development of computer models for evaluating component failures in air cleaning systems and the performance of specific systems under normal and accident conditions; contamination control and personnel protection; repair requirements and failure rates in air cleaning systems; and problems relating to quality assurance.

It was noted, once again, that the U. S. commitment to research on nuclear air and gas cleaning had been greatly reduced and that most research studies were now being conducted in Europe and Asia.

The 18th Conference, held in Baltimore, MD during August 1984, sported a new title, the DOE Nuclear Airborne Waste Management and Air Cleaning Conference. It signified a waning interest in the further development of U. S. government-supported research for engineered safety system air cleaning technology for commercial nuclear power plants and an increasing concern for nuclear waste management, an area where nuclear air cleaning technology was, and still is, desperately needed. Inasmuch as nuclear airborne waste management incorporates a very large component of air and gas cleaning technology, the combination is a natural one and it is logical to conclude that combining the two will enrich knowledge of both. The NRC, retaining a primary interest in commercial nuclear power, reported, at this and later conferences, continued refinements and improvements in the techniques for estimating accident source terms.

Performance and testing of adsorbers and filters installed in full-scale operating systems was the subject of several presentations at this conference, reflecting a determined effort by the Program Committee to respond more fully to the needs of operating personnel. Included were reports of a roundrobin test program for laboratory rating of nuclear carbon that demonstrated that the test protocol accepted by NRC was defective and required serious attention; this it received. The field of waste collection, processing, and storage was covered by numerous research papers, pilot plant operating reports, and theoretical studies. Of interest, was the first conference presentation by a representative from the People's Republic of China.

Recent Air Cleaning Conferences

The 19th Conference called for another title change when the NRC joined with the DOE to sponsor the first DOE/NRC Nuclear Air Cleaning Conference. It was held in Seattle, Washington during August 1986. Chernobyl was on the minds of all as they reflected on the consequences of a

failure to employ containment structures and engineered safety systems as primary mechanisms for controlling airborne radionuclide releases following major accidents in nuclear power plants. The outcome at Chernobyl was contrasted with the outcome of the TMI-II accident where air cleaning systems proved to be effective barriers to emissions, even under unusual circumstances. The heightened interest in vented containment resulting from Chernobyl emphasized the need for new types of ultra-high efficiency air and gas cleaning systems. Accident analysis and loss control were prominent topics on the agenda. The latest developments in adsorption for removal of iodine and other gases, and aerosol filtration for particle retention, were covered during several half day sessions. The many papers and panel sessions on standards and regulations reflected a vigorous effort to prepare and apply nuclear codes and standards, plus a drive to eliminate conflicts between like-minded documents. Nuclear gas processing systems intended for decontamination, demolition, and treatment of nuclear wastes were recognized to be seriously in need of the kinds of standards that were already being vigorously developed for air cleaning systems used in engineered safeguard systems of nuclear power plants.

Fire detection and suppression in activated carbon beds came to the fore once again as a result of a series of false triggerings of water sprinklers, required by American nuclear insurers. After a thorough exchange of views between regulators, insurers, and plant operators during a panel session, a satisfactory resolution was reached to provide positive protections against accidental carbon flooding by providing a pipeline discontinuity that must be closed manually to activate the system. New nuclear carbon laboratory test protocols were proposed; they were ultimately accepted by all stakeholders and incorporated into a revised ASTM Standard, D3803-89. A feature of this conference was a luncheon address by Dixy Lee Ray, an ardent and fiery spokesperson for nuclear energy.

The 20th DOE/NRC Nuclear Air Cleaning Conference was held in Boston, Massachusetts during August 1988 amid great excitement over the (unsuccessful) bid for the presidency by the state's governor. Papers on severe accident mitigation were a continuing legacy from the Chernobyl disaster. Reactor operations were a prominent part of the agenda. Papers on decommissioning made their first appearance at this conference. The effects of age, fire, and physical stress on HEPA filter performance were discussed by representatives from five different nations, evidence of a continuing universal concern. Waste treatment was discussed, but was not a major contributor to the agenda.

At this time, the Nuclear Regulatory Commission and the United States Department of Energy requested the industrial beneficiaries of these conferences to demonstrate that the conferences were useful, needed, and as productive for them as for the agencies involved, by becoming identified as co-sponsors and by helping to provide financial support. A number of industry leaders responded by organizing themselves as an *ad hoc* committee to explore the formation of a group that would help support future air cleaning conferences. This tentative organization developed into the International Society for Nuclear Air Treatment Technologies (ISNATT). It became a co-sponsor of the conferences and assumed responsibility for arranging the Government-Industry meetings. Today, ISNATT is solely responsible for organizing and conducting exhibits that are held jointly with the conference. ORNL prepared a cumulative index from the beginning through the 20th Conference (as they had for the 16th) and it was distributed with the Proceedings of the 20th as Volume 3 of 3.

The 21st Conference, held during August 1990, returned to San Diego. A feature of this meeting was an introduction to the development of advanced reactor designs. Management and disposal of spent fuel and high level chemical wastes received considerable attention in formal papers and discussion groups. Codes and standards continued to be an important topic with the appearance

of the first sections of an ASME/ANSI "Code on Nuclear Air and Gas Treatment, AG-1." There were continuing concerns expressed regarding compatibility of U. S. codes and standards with those being produced and adopted by the European Community. An entire session was devoted to the troublesome questions that arise when ASME N510 is invoked for testing systems that were not constructed in strict conformance with ASME N509. Incineration and vitrification of nuclear wastes were additional topics of interest. Vented containment studies from France, Germany, Sweden and Japan were presented. A panel session entitled, Nuclear Air Cleaning Programs Around the World, described current activities in 10 countries that ranged from reactor regulation to reprocessing and waste management.

The 22nd Conference was held in Denver during August 1992. In recognition of the greater interest and concern with the processing of gaseous effluents from nuclear waste treatment operations, a new title was chosen for the conferences, the DOE/NRC Nuclear Air Cleaning and Treatment Conference. The 23rd Conference retained the same designation. The opening session of the 22nd was devoted to a discussion of the current status of nuclear power in the U. S. and the importance of the advanced nuclear power reactors under construction in Asia. The impact of source term revisions on air and gas treatment requirements was the subject of a panel session. Another panel session reviewed new nuclear testing protocols. ORNL kindly provided an index for the 21st and 22nd Conferences that was included in volume 2 of the Proceedings of the 22nd Conference.

The most recent conference, the 23rd, was held in Buffalo, New York during July 1994. It was sponsored by ISNATT, in addition to DOE, NRC, and the Harvard Air Cleaning Laboratory. One hundred and ninety-two air cleaning specialists attended from the U. S. and 11 other countries. Proximity to West Valley, site of a major DOE program of decontamination, decommissioning, and waste encapsulation for permanent storage, made these topics a prominent part of the technical program. At the conclusion of the technical program, approximately 50 conference attendees visited the West Valley Demonstration Project and received a guided tour of a pilot plant used for operator training and a partially completed full-scale vitrification facility.

The development and application of nuclear codes and standards received a great deal of attention at this conference and, once again, re-emphasized the need to reconcile conflicts between codes and standards that are in good standing. Presentations about the ongoing West Valley Demonstration Project, a panel session on air cleaning for gas processing operations, and a description of the progress of cleanup efforts at the Fernald site brought to the fore the urgent need to develop, and have in place, nuclear gas processing standards that would apply uniquely to nuclear waste processing activities before major efforts are undertaken at numerous DOE sites. This need was communicated to representatives of CONAGT for action.

Deterioration of air cleaning systems after many years in service, and on standby, continues to be a major safety concern. Methods that might be used to downrate the "as-installed protection factors" provided by old systems were proposed, and continuation of work on this troublesome matter was promised by DOE representatives and their contractors. Changing materials and manufacturing methods for HEPA filters make predictions of future service life, based on analyses of old filters, somewhat precarious.

Refinements of the source term for nuclear power plant accidents were reported, and it was predicted that this activity will soon reach closure, at least for the present. During one of the panel sessions, efforts to define source terms applicable to the control of radioactive and chemical

contaminants during decontamination and dismantling operations were reported to be underway.

Commissioner Kenneth C. Rogers, U. S. Nuclear Regulatory Commission, gave a featured address entitled, "Clean Air and Clear Responsibility" that focused on a discussion of optimizing the principles of ALARA plus procedures for amicably reconciling conflicts between regulations of the NRC and the U. S. Environmental Protection Agency.

During the final session of the 23rd Conference, recommendations were received for selecting a site and a set of objectives for a 24th Conference that will be held during 1996.

Conclusion and Afterthoughts

I have recounted only a few of the highlights from the past. Each conference has recorded advancements in nuclear safety and revealed the arrival of new challenges. The conferences have been a force for advancement as well as a forum for refinement of current technology, and the conference proceedings serve as a history of achievements in this field.

What of the future? Surely the conferences will continue to be highly involved in the important task of maintaining and improving engineered safeguards for existing power reactors and with meeting the air cleaning and treatment needs of advanced reactors. In addition, the scope must broaden to include whole new areas that we always knew were there, but tended to postpone coming to grips with. This means, of course, the tasks of reactor decommissioning and disassembly, the cleaning up of weapons sites, and the safe storage of nuclear wastes of many types. All of these tasks call for air cleaning and gas treatment applications that differ significantly from those applied to power reactors. There is hope and expectation that we will learn more about these requirements and their solution during future nuclear air cleaning conferences.

An appendix to this "Brief History" summarizes the date and location of each air cleaning conference and gives a reference for each volume of the proceedings.

Acknowledgement: The liberal use of Dr. Dade Moeller's several publications in the Proceedings on the history of the nuclear air cleaning conferences as well as his perceptive comments on the original draft are acknowledged with thanks.

APPENDIX

Chronological Listing of Air Cleaning Conferences

Conference Number	Date	Site	Document Number	Number of Volumes
1	June 12-15, 1951	Boston, MA	No Report	
2	Sept. 15-17, 1952	Ames, IA	WASH-149	1
3	Sept. 21-23, 1953	Los Alamos, NM	WASH-170	1
4	Nov. 2-4, 1955	Argonne, IL	TID-7513(Pt. 1)	1
5	June 24-27, 1957	Boston, MA	TID-7551	1
6	July 7-9, 1959	Idaho Falls, ID	TID-7593	1
7	Oct. 10-12, 1961	Upton, NY	TID-7627	1
8	Oct. 22-25, 1963	Oak Ridge, TN	TID-7677	1
9	Sept. 13-16, 1966	Boston, MA	AEC-660904	2
10	Aug. 28, 1968	New York, NY	CONF-680821	1
11	Aug. 31-Sept. 3, 1970	Richland, WA	CONF-700816	2
12	Aug. 28-31, 1972	Oak Ridge, TN	CONF-720823	2
13	Aug. 12-15, 1974	San Francisco, CA	CONF-740807	2
14	Aug. 2-4, 1976	Sun Valley, ID	CONF-760822	2
15	Aug. 7-10, 1978	Boston, MA	CONF-780819	2
16	Oct. 20-23, 1980	San Diego, CA	CONF-801038	2
17	Aug. 2-5, 1982	Denver, CO	CONF-820833	2
18	Aug. 12-16, 1984	Baltimore, MD	CONF-840806	2
19	Aug. 18-21, 1986	Seattle, WA	NUREG/CP-0086	
			CONF-860820*	2
20	Aug. 22-25, 1988	Boston, MA	NUREG/CP-0098	
			CONF-880822*	3
21	Aug. 13-16, 1990	San Diego, CA	NUREG/CP-0116	
			CONF-900809*	2
22	Aug. 24-27, 1992	Denver, CO	NUREG/CP-0130	
			CONF-9020823*	2
23	July 25-28, 1994	Buffalo, NY	NUREG/CP-0141	
			CONF-940738*	1

All documents available from:
National Technical Information Service
Springfield, VA 22161

* Also available from:
Superintendent of Documents
U. S. Government Printing Office
Mail Stop SSOP
Washington, DC 20402-9328

THE INTEGRATED MELTER OFF-GAS TREATMENT SYSTEMS AT THE WEST VALLEY DEMONSTRATION PROJECT*

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Abstract

The West Valley Demonstration Project was established by Public Law 96-368, the "West Valley Demonstration Project Act," on October 1, 1980. Under this act, Congress directed the Department of Energy to carry out a high level radioactive waste management demonstration project at the Western New York Nuclear Service Center in Wast Valley, New York. The purpose of this project is to demonstrate solidification techniques which can be used for preparing high level radioactive waste for disposal. In addition to developing this technology, the West Valley Demonstration Project Act directs the Department of Energy to:

(1) develop containers suitable for permanent disposal of the high level waste;

(2) transport the solidified high level waste to a Federal repository;

(3) dispose of low level and transuranic waste produced under the project; and

(4) decontaminate and decommission the facilities and materials associated with project activities and the storage tanks originally used to store the liquid high level radioactive waste.

The process of vitrification will be used to solidify the high level radioactive liquid wastes into borosilicate glass. This report describes the functions, the controlling design criteria, and the resulting design of the melter off-gas treatment systems which are used in the vitrification process.

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* Work Performed Under Contract No. DE-AC07-81NE44139

I. Introduction

The primary objective of the West Valley Demonstration Project (WVDP), located at the former nuclear fuel reprocessing plant at West Valley, New York, is to solidify the high-level radioactive waste stored in underground tanks into a form suitable for transportation and disposal. Vitrification has been chosen as the method of solidification. The purpose of this report is to describe briefly the systems used to treat the vitrification system off-gases for the purpose of protecting the environment.

The gaseous effluent from the melter, mostly water vapor and oxides of nitrogen, are scrubbed in a submerged bed scrubber, processed through a high efficiency mist eliminator, and filtered before leaving the shielded vitrification cell. Before the off-gases are released to the environment, they are further filtered through High Efficiency Particulate Air (HEPA) filters, and the oxides of nitrogen (NO_x) are eliminated by selective catalytic destruction.

II. Functions and Design Criteria

The overall project approach was to incorporate the following principles:

(1) Make maximum use of existing technology, facilities and equipment.

(2) Minimize complexity.

(3) Produce a system that was safe, environmentally sound and cost effective.

Functions

The Vitrification Facility functional requirements that applied to the melter off-gas treatment systems are as follows:

(1) Remove radioactive particulate matter, toxic gases, and vapor from the process gases for release to the environment.

(2) Provide process solution re-use and recycle features to minimize derivative waste disposal demands.

(3) Provide process system redundancy and/or readily achievable remote removal and replacement capability for planned maintenance.

(4) Provide monitoring and control systems and maintain release levels below current state and federal regulatory limits.

(5) Use existing WVDP utility and support systems as far as practical.

(6) Provide design features to maintain internal and external radiation exposures to operating and maintenance personnel ALARA, and in no case exceed allowable design guidelines. Provide the capability to control contamination during routine and emergency operating conditions and during all hands-on and remote maintenance activities.

(7) Maximize the use of low maintenance process equipment in radioactive areas, and provide remotely operated decontamination capabilities for equipment to the extent necessary to support required maintenance.

(8) Provide alarm systems, and all other plant safety features essential to proper and safe plant operation.

(9) Evaluate remote equipment and components used in the vitrification for their service lives. Service lives may be enhanced, if required, by increased quality and reliability for the equipment, redundancy of equipment, and/or remote replaceability.

(10) Provide an alternate power supply and utilities as backup for essential equipment and systems.

Design Criteria

The primary design criteria applied to the melter off-gas treatment systems, excerpted from the design criteria document controlling the entire vitrification facility, are presented in the following paragraphs.

Operational Criteria The process off-gas system blowers will maintain a negative pressure on the major tankage and melter within the vitrification facility and throughout the upstream process train to assure any leaks are into the system. An installed back-up off-gas blower and an alternate power source shall be provided.

Maintenance Criteria Systems and components in contaminated areas shall be designed to be either remotely maintainable in place, or remotely removable and replaceable.

Spares shall be provided for critical components or components that have a high probability of failure during operation.

Connectors, bolts, flanges, wrenches, sockets shall be standardized to the maximum extent practical.

Equipment shall be movable, maintainable, and replaceable with minimum disturbance of adjacent equipment.

The vitrification facility systems and structures are to be

designed to facilitate post solidification decontamination.

Structural Criteria Structures and components that are required to confine radioactive material shall be able to withstand the effects of natural hazards without loss of capability to perform safety function(s) or prevent the release of radioactivity and shall be designed to the "Operational Safety Design Criteria Manual," (ID-12044).

New confinement structures shall be designed to an acceleration of 0.1 g at ground level (horizontal loads).

The equipment shall be designed for a service life of seven years.

Safety Criteria The principle of "As Low As Reasonably Achievable" (ALARA) shall apply to all aspects of radiation exposure. On-site personnel exposure levels less than one-fifth of the DOE Order 5480.11 dose equivalent limits should be used as a design objective.

The maximum radiation dose rate for a full-time access area shall be 2.5/t mRem/hr in which "t" is the maximum average time in hours per day that the area is expected to be occupied by any one individual. A full-time access area is one in which no physical or administrative control of entry exists.

Environmental Criteria Gaseous releases shall be subject to applicable Environmental Protection Agency (EPA), New York State Department of Environmental Conservation (NYSDEC), and DOE regulations, Orders, and directives. These regulations shall include, but not be limited to, 40 CFR 61, 6 NYCRR Series 200, and DOE Order 5480.1A.

Where possible, radioactive liquid waste will be recycled within the vitrification facility.

Quality Assurance The Quality Assurance program will be based on ANSI/ASME NQA-1, and all supplements.

Decommissioning The vitrification facility design shall incorporate features that facilitate future decommissioning of the facility.

III. Process Descriptions

The integrated melter off-gas treatment systems are depicted in Figure 1. The ALARA portion is located inside the vitrification cell and is identified as the In-Cell Off-Gas System. The atmospheric protection portion is located in the trench and in the 01-14 building and is identified as the Out-of-Cell Off-Gas system.

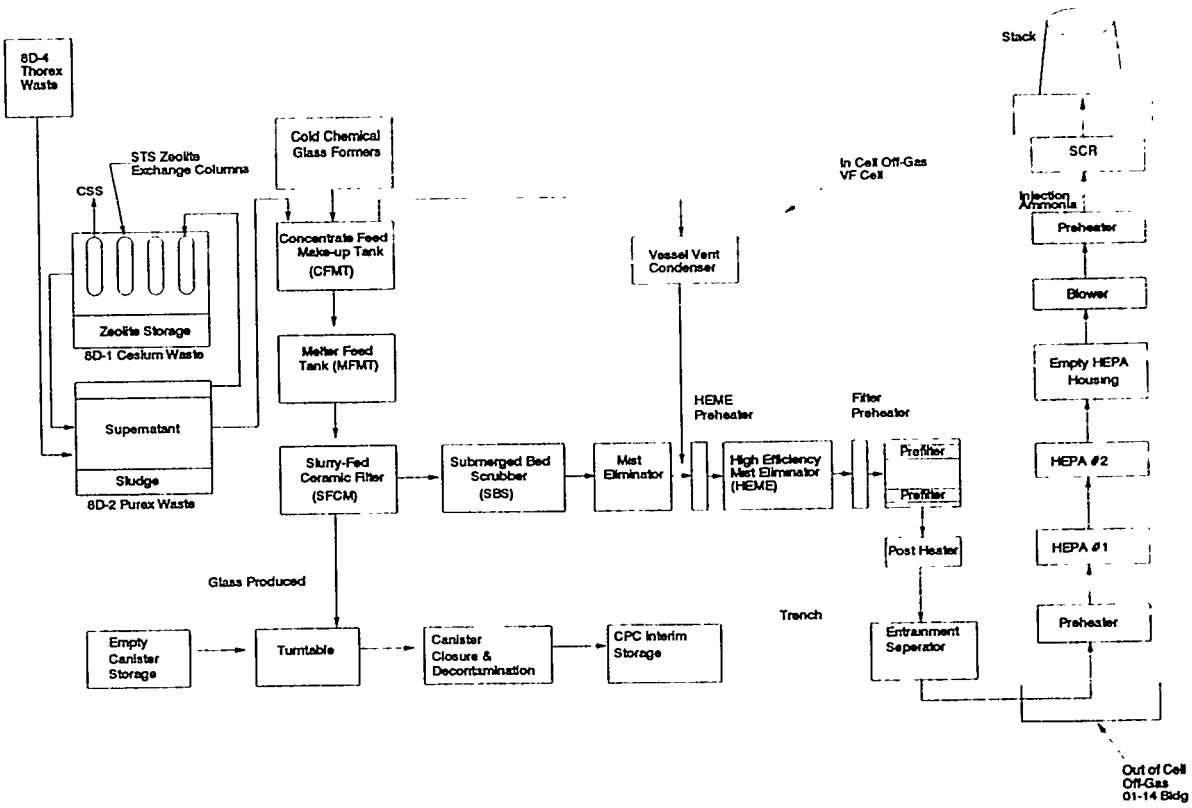


Figure 1: Integrated melter off-gas treatment systems.

ALARA Process

The ALARA melter off-gas treatment system, and associated vessel ventilation system, are depicted in Figures 2 and 3, and include all equipment required to treat, and transfer process gases and vapors from the melter and other in-cell equipment to the atmospheric protection off-gas treatment system.

Vessel Ventilation Ventilation gases and vapors from the concentrator, the feed hold tank, the canister turntable, canister decontamination station, and the vitrification cell waste header, are continuously collected into the vessel ventilation header. In the event that vacuum is lost in the melter, melter gases and vapors are also directed into the vessel vent header to re-establish the vacuum. The gases and vapors from the vessel vent header are directed into the shell side of the condenser. Closed loop cooling water is directed into the tubes, counterflow with the gases and vapors in the shell. The condensate from the condenser flows by gravity through a calibrated weir, and a liquid seal, to a header which directs the liquid to the tank farm for subsequent processing. Gases and uncondensed vapors are directed into the in-cell melter off-gas treatment system.

Quenching/Scrubbing The submerged bed scrubber is designed for the first stage scrubbing of melter off-gases, cooling and condensation of melter vapor emissions, and interim storage of condensed fluids. The scrubber is a passive device that uses water to remove particulate and to quench the off-gases. It functions by bubbling the off-gases through water in a bed packed with ceramic spheres. The rising bubbles of off-gas cause the liquid to circulate up through the packing. This simultaneously causes downward flow in the annular space outside the packed bed as liquid from the annular space replaces the liquid which is rising through the bed. The packing breaks larger bubbles into smaller ones to increase the gas-to-water contacting surface, thereby increasing the particulate removal and heat transfer efficiencies. The liquid circulation helps to prevent a buildup of captured material in the bed by constantly washing the material away. As the off-gases cool, water vapor condenses and increases the liquid water inventory. The excess water spills into the receiver, thereby maintaining a constant liquid depth in the scrubber. Heat absorbed by the water from the off-gases is removed by the cooling coils as the water flows downward in the annular space.

The mist eliminator consists of a mesh pad in a housing mounted directly on the submerged bed scrubber off-gas exit nozzle flange, plus a jumper that provides demineralized spray water to flush the pad, and pressure differential measurement across the pad during operation.

Off-gases enter from below the pad and exit from above. The mist eliminator pad collects entrained droplets by impaction against the pad fibers where they adhere and coalesce. The coalesced liquid flows by gravity back into the submerged bed scrubber.

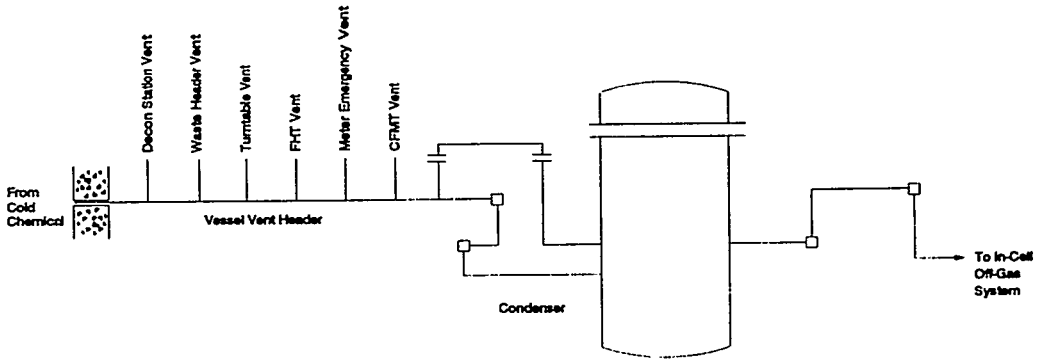


Figure 2: Vessel ventilation system.

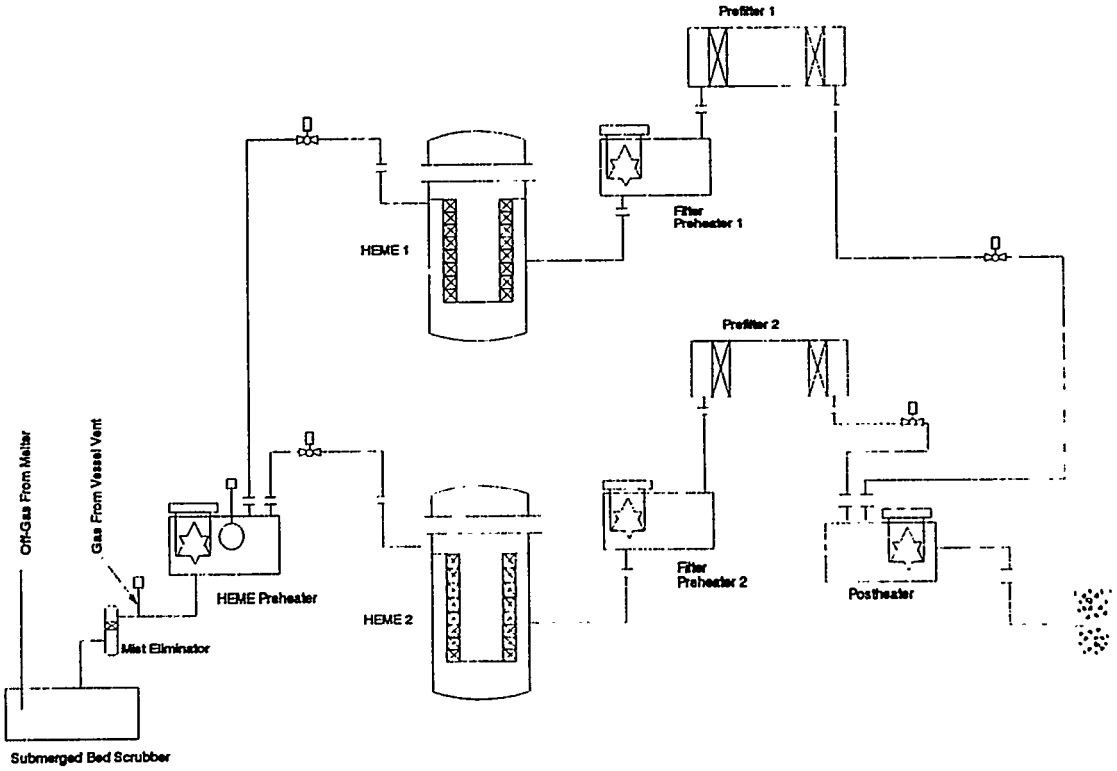


Figure 3: ALARA off-gas treatment system.

High Efficiency Mist Elimination Should the operations organization decide to employ the high efficiency mist eliminator as a dry filter, the 480 V electric preheater would be used to elevate the temperature of the off-gases above their dew point. The reference use for the high efficiency mist eliminator is mist elimination, and this preheating option is not expected to be employed.

Impact wrench operated isolation valves in the jumpers to the high efficiency mist eliminators are used to select which of the two installed parallel off-gas trains (high efficiency mist eliminator, filter preheater and prefilter) are used.

The high efficiency mist eliminator receives off-gases from the preheater. It collects and coalesces entrained liquid droplets, and is 99.8 weight percent efficient for droplets 3 microns in diameter and larger. Simultaneously it removes submicron particulate from the gases.

Off-gases enter the center of the cylindrical pad from the top and pass through the pad to the outside. Because the pad is so large, the velocity of the gas through the pad is slow. Due to Brownian movement, droplets and particles contact the fibers of the pad where they collect and coalesce. Collected particulate is carried to the high efficiency mist eliminator drain lines either by the gravity flow of the coalesced liquids, or by demineralized water spray.

The high efficiency mist eliminator vessel has a drain line which directs coalesced liquids, and spray water, to the submerged bed scrubber.

Prefiltration The filter preheaters operate by the same principles described for the high efficiency mist eliminator preheater.

The filter preheaters are used to elevate the temperature of the off-gases above the dew point to assure that no condensation occurs on the prefilter elements downstream.

The prefilter assemblies capture dry particulate to retain radioactive contamination inside the vitrification cell. This prevents significant contamination from reaching off-gas treatment equipment located downstream, outside the vitrification cell, thereby allowing hands-on maintenance there.

Heating The off-gases from the parallel off-gas trains are directed to a common "postheater."

NOTE

The original purpose of the postheater was to elevate the temperature of the off-gases above the dew point sufficiently to assure that no condensation occurred in the duct from the vitrification cell to the 01-14

Building. Insulation, an entrainment separator, and reheaters installed to protect the HEPA filters rendered the "postheater" redundant.

Atmospheric Protection Process

The Atmospheric Protection Off-Gas Treatment System is depicted in Figure 4, and includes all equipment required to treat and transfer process gases from the vitrification cell to the base of the plant stack.

HEPA Filtration The off-gases from the vitrification cell are directed through an insulated duct to the atmospheric protection off-gas treatment equipment. The insulation is intended to protect the HEPA filter elements by mitigating condensate formation between the vitrification cell and the final HEPA filters. Immediately upstream from the HEPA filters the off-gases are processed through an entrainment separator, and one of two redundant electric reheaters connected in parallel, which restore the off-gas temperature to above the dew point.

The off-gases then pass through the HEPA filters. In each of two parallel filter trains are two HEPA filter elements arranged in series. The gases pass through one filter train while the other remains available as an installed back-up. The purpose of the HEPA filters is to provide final atmospheric protection against dispersion of radioactive particulate. The integrity of the filter elements, and the seals between the elements and the housing, is verified by in-place DOP testing.

Following HEPA filtration, the off-gases pass through another, empty, filter housing, One housing is located immediately downstream from each HEPA filter. These previously existing housings were retained for possible future use.

Motivation Following filtration, the off-gases pass through one of three redundant, positive displacement, off-gas blowers installed in parallel. One blower operates while the others provide reliable, full capacity, backup service. The blower provides the motive force to maintain all of the vitrification equipment upstream under a slight vacuum for the purpose of contamination control. It also provides the motive force to discharge the treated off-gas into the base of the previously existing plant stack.

NO_x Destruction From the blower, the off-gas passes through a NO_x abatement system to destroy the noxious oxides of nitrogen (NO_x). The oxides of nitrogen are reacted with ammonia at an elevated temperature, and in the presence of a catalyst, to produce harmless water vapor and nitrogen.

The NO_x destruction equipment includes redundant off-gas preheaters, an ammonia supply system, and redundant catalytic reactors. The preheaters elevate the off-gas temperature to

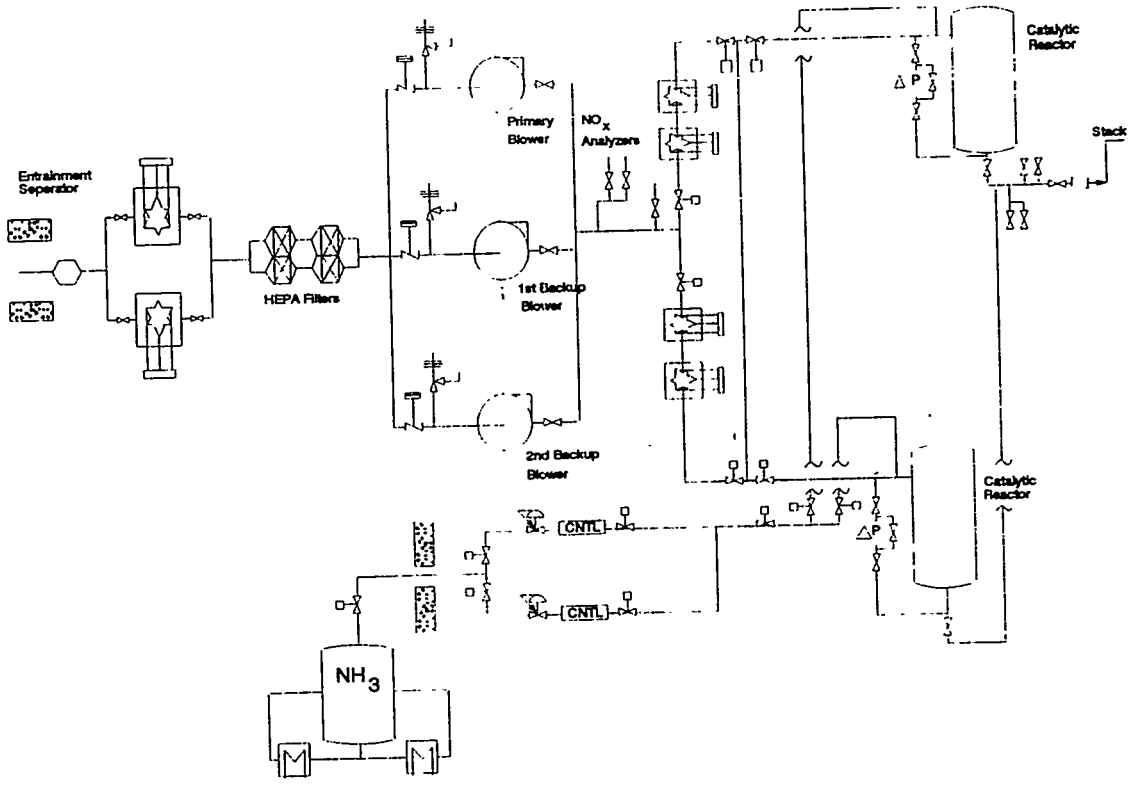


Figure 4: Atmospheric protection off-gas treatment system.

promote the desired reaction, the ammonia supply provides the necessary reactant, and the catalytic reactor accelerates the desired reaction.

The NO_x gases are destroyed by several competing chemical reactions involving NO_2, NO, N_2O, NH_3 and O_2. Test results indicate that, at WVDP operating conditions, the following reactions dominate.

$$2\ NO + O_2 \rightarrow 2\ NO_2$$

$$8\ NO_2 + 6\ NH_3 \rightarrow 7\ N_2O + 9\ H_2O$$

$$2\ N_2O \rightarrow 2\ N_2 + O_2$$

Following NO_x destruction, the treated off-gas is directed to the previously existing plant stack.

IV. Equipment Descriptions

ALARA Equipment

The vitrification cell environment which the melter off-gas treatment equipment must endure is tabulated below.

Temperature	16 to 35 °C	(60 to 95 °F)
Pressure	0 to -1 kPa	(0 to -4 InWC)
Relative Humidity	30 to 80 %	
Nitric Acid Fumes	100 ppm	
Radiation (SBS)	50 Gy/h	(5 x 10^3 Rad/hr)
(other)	1 Gy/h	(100 Rad/hr)

For corrosion resistance against nitric acid fumes and solutions, the fabrication materials used for surfaces of equipment exposed to the vitrification cell environment, and to process fluids, is generally Type 304L stainless steel. The structual supports are made from carbon steel, and are painted for corrosion resistance.

Vessel Ventilation The vessel ventilation header is located near the top of the vitrification cell walls. It rests on supports at a slope of 10 mm/m (1/8 in/ft) so liquids will drain to the condenser. The header is made from seamless 150 mm (6 in) pipe. Stresses in the header due to thermal expansion and contraction are accommodated by six, permanently installed, expansion joints. They are physically protected by external covers, and from accumulations of particulate by internal sleeves.

The vessel ventilation condenser is a vertical, 3.2 GJ/h (3.0 x 10^6 Btu/hr) shell and U-tube heat exchanger. The shell is 760 mm (30 in) in diameter, is 3400 mm (11 ft) high.

Quenching/Scrubbing Quenching and initial scrubbing is accomplished by the Submerged Bed Scrubber (SBS). It is comprised of a scrubber vessel, and a receiver vessel which completely envelopes the scrubber vessel.

The scrubber consists of two concentric right cylindrical vessels.

The inner vessel contains the bed, is 1220 mm (4 ft) tall, 910 mm (3 ft) in diameter, and is constantly flooded with the water contained by the outer vessel. An Inconel 690 bed support/gas distributor plate forms the bottom. It houses a 0.5 m^3 (18 ft^3) bed of 10 mm (3/8 in) diameter ceramic sphere packing. Off-gases are introduced through a 250 mm (10 in) diameter Inconel 690 downcomer entering from the top and discharging at the bottom of the bed.

The outer vessel is 1750 mm (69 in) tall and 1830 mm (6 ft) in diameter. It is open at the top, and is kept filled with water.

The scrubber contains cooling coils in the annular space outside the bed. Instrumentation to measure temperature, vapor space pressure, liquid specific gravity and liquid depth are provided.

The maximum operating volume of the receiver is 5.5 m^3 (1450 gal). The SBS receiver vessel is a right cylindrical vessel, 2440 mm (8 ft) in diameter and 3350 mm (11 ft) tall. Steam jets (not shown) are provided to evacuate the contents. Instrumentation is provided to measure temperature, vapor space pressure, liquid specific gravity, and liquid depth.

The mist eliminator housing is made from 460 mm (18 in) pipe, and the inlet and outlet ducting is from 150 mm (6 in) pipe. The pad is a 150 mm (6 in) thick, knitted mesh with a 360 mm (14 in) diameter exposed face. The pad is designed to minimize fouling, and consists of four pads in series with each successive pad having a greater packing density.

High Efficiency Mist Elimination The high efficiency mist eliminator (HEME) preheater 50 kW, electrical resistance heating element is operated with 480 V, electrical energy.

The HEME vessels are cylindrical, 1070 mm (42 in) diameter, 4060 mm (160 in) tall. The HEME elements are 760 mm (30 in) in diameter, 3050 mm (10 ft) tall, with a wound glass fiber element.

Prefiltration Each filter preheater consists of a housing and an element, with temperature instrumentation.

Each prefilter assembly consists of a housing, a perforated flow straightening baffle, and two HEPA filter elements in series. Provision is made to measure the differential pressure across the filter elements.

Atmospheric Protection Equipment

Equipment located inside the 01-Cell of the 01-14 Building was designed for remote manual switchover. These items include the off-gas reheater, HEPA filters, filter housings, reactor preheaters, and catalytic reactors.

HEPA Filtration The duct from the vitrification cell to the 01-14 Building is an insulated, 250 mm (10 in) stainless steel pipe. The trench through which the duct runs is heated to help maintain the temperature of the off-gas above its dew point. An entrainment separator is located at the low point between the vitrification cell and the 01-14 Building to collect any condensation that might occur in the duct.

Dual, redundant, electrical resistance reheaters are provided. Each reheater consists of a 1570 mm (62 in) tall, 250 mm (10 in) diameter housing with a 60 kW, 480 V, heating bundle. Thermocouples located in the outlets of the housings serve as process control sensors.

Parallel HEPA filter trains, consisting of two HEPA filter elements in series, are contained in a common housing. The housing accommodates filter elements 610 mm by 610 mm by 290 mm deep (2 ft X 2 ft X 11-1/2 in). Each filter element position is provided with a bag-out port.

Motivation The off-gas blower is a positive displacement rotary blower slightly oversized for 42 m^3/min (1500 ACFM). It is driven by an electric motor typically at a speed to motivate only 37 m^3/min (1300 ACFM). The blower suction is equipped with a vacuum relief valve.

NO_x Destruction The NO_x destruction system is designed to limit NO_x emissions during any 1-hour period to no more than 1.9 kg (4.2 lb_m).

The ammonia supply system was designed for anhydrous ammonia to minimize the equipment complexity and aqueous waste streams associated with use of aqueous ammonia. The ammonia storage tank is a carbon steel cylinder, 5180 mm (17 ft) tall, 1070 mm (42 in) in diameter, with elliptical dished heads. It is designed for a maximum inventory of 3.8 m^3 (1000 gal). The tank is equipped with two, redundant, 18 kW, 480 V, electric vaporizers. The vaporized ammonia is drawn from the top of the tank, and is routed to the mass flow control train. The ammonia is delivered to the off-gases downstream from the preheaters at the reactor inlets.

Dual, redundant, preheaters are provided. Each consists of two electric heater elements in series. Each heater element is rated at 480 V and 100 kW. Each preheater assembly includes a 3100 mm (122 in) tall, 300 mm (12 in) diameter housing. A thermocouple located in the housing outlet serves as the process control sensor.

Two identical reactors are installed in parallel. The reactor vessels are right cylinders made from Type 321 stainless steel, with the inlet at the top and the outlet at the bottom. The overall height of each reactor is 3480 mm (11 ft - 5 in). The cylindrical sections have inside diameters of 1000 mm (39-1/2 in).

The 1120 mm (44 in) deep catalyst bed is zeolite based. The primary catalyst bed consists of 890 mm (35 in) of 6 mm (1/4 in) Raschig rings. The polishing bed is located beneath the primary bed and consists of 230 mm (9 in) of 1.6 mm (1/16 in) catalyst extrudate.

The duct leading toward the stack is an insulated 200 mm (8 in) pipe. The insulation is specified to maintain the off-gas above its dew point.

V. Systems Control

ALARA Control

The motive forces for establishing a vacuum are provided by the blowers located downstream in the out-of-cell melter off-gas treatment system. The control valve located in the jumper that directs vessel ventilation gases and vapors to the melter off-gas treatment system is automatically modulated to maintain the vessel ventilation header at a predesignated vacuum. This is done to maintain a slight negative pressure on the primary process systems, relative to the vitrification cell atmosphere, for purposes of contamination control.

The amount of water that is discharged from the system, in the form of water vapor directed to the stack, is established by controlling the operating temperatures of the condenser and submerged bed scrubber. Temperature increases result in increased water vapor expulsion rates. The temperature in the submerged bed scrubber must be kept low enough, however, to prevent loss of solution from the scrubber. Temperature control is provided by modulation of the amount of closed loop cooling water sent to the cooling coils in the condenser and the scrubber vessel.

The temperature of the off-gases approaching the prefilter is maintained above the dew point by modulation of the power supplied to the filter preheater elements, based upon off-gas temperature readings immediately downstream from the preheater.

Atmospheric Protection Control

NO_x analyzers at the inlet of the NO_x abatement equipment, and at the outlet, are used to control the addition of the ammonia reactant and to monitor the NO_x destruction efficiency.

VI. Equipment Maintenance

ALARA Equipment

To allow for equipment maintenance without having to suspend melter feed operation, a redundant train of off-gas treatment equipment was installed. Normally, one of the installed redundant off-gas trains is operated while the other train is valved out-of-service. While out-of-service, maintenance operations can be performed.

Atmospheric Protection Equipment

The system is designed for hands-on maintenance. Installed redundant equipment was required for all equipment which would require removal from service for periodic maintenance, and whose removal from service would require suspension of melter feed operation.

The design for maintenance was based upon the philosophy that opening of a boundary, when pressurized NO_x is isolated by valving alone, would be an undue risk to personnel. Therefore, double isolation valves with bleed lines between the isolation valves leading to the blower suction, were provided to positively isolate process gases from equipment requiring maintenance.

The design for maintenance is also based upon the philosophy that routine access to a room or space in which pressurized NO_x exists in equipment would also be an unacceptable risk to personnel. Therefore, items of equipment on the discharge side of the blowers were located inside the 01-Cell of the 01-14 Building, and an off-gas by-pass line was provided for use when any of those items of equipment requires maintenance.

To allow for purging of NO_x, purge ports are provided for all major components exposed to off-gases.

SESSION 2

NUCLEAR AIR CLEANING CODES AND STANDARDS

Monday: July 25, 1994
Co-Chairmen: R. R. Weidler
J. J. Hayes

OPENING COMMENTS OF SESSION CO-CHAIRMAN HAYES

CHALLENGES OF EQUIPMENT QUALIFICATION USING TODAY'S STANDARDS WITH EMPHASIS ON A CLASS 1E MOTOR PROGRAM
K. Deaton

EVALUATION OF SELF-CONTAINED HEPA FILTER
T. E. Arndt

ASME N510 TEST RESULTS FOR SAVANNAH RIVER SITE AACS FILTER COMPARTMENTS
J. D. Paul, T. M. Punch

AN EVALUATION OF EFFORTS BY NUCLEAR POWER PLANTS TO USE ASTM D3803-89
W. P. Freeman

CLOSING COMMENTS OF SESSION CO-CHAIRMAN WEIDLER

Conference Number: 9407 38

Do Not prepare analytics for these papers

Conference 940738

23. DOE/NRC nuclear air cleaning and treatment conference, Buffalo, New York, 25-28 Jul 1994

Entered: 02/15/94 | Papers : 12 | Activities: 0 | Energy Meetings: Y
Changed: 06/15/95 | Volumes: 1 | Sponsors : 3

Conference Paper 1 of 12

Author : Paul, J. D.
Report Number: WSRC-MS--94-0352
Secondary : CONF-940738--1
Trac No. SC: : M94014870
Entered : 07/25/94
Changed :
Description:
Note :

ASME N510 test results for Savannah River Site AACS filter compartments

Conference Paper 2 of 12

Author : Davis, W. E.
Report Number: WHC-SA--2526
Secondary : CONF-940738--2-Pt.1
Trac No. SC: : M94015377
Entered : 07/28/94
Changed : 08/18/94
Description: Pt.1
Note :

Potential radionuclide emissions from stacks on the Hanford Site, Part 1: Dose assessment

Conference Paper 3 of 12

Author : Glissmeyer, J. A.
Report Number: PNL-SA--24626
Secondary : CONF-940738--3
Trac No. SC: : M94015734
Entered : 08/08/94
Changed :
Description:
Note :

Progress of revision to ANSI N13.1-1969 Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities

Conference Paper 4 of 12

Author : Arndt, T. E.
Report Number: WHC-SA--2313
Secondary : CONF-940738--4
Trac No. SC: : M94015709
Entered : 08/09/94
Changed :
Description:
Note :

Evaluation of self-contained HEPA filter

Conference Paper 5 of 12

Author : Barnett, J. M.
Report Number: WHC-SA--2517
Secondary : CONF-940738--5-Pt.2
Trac No. SC: : M94016465
Entered : 08/18/94
Changed :
Description: Pt.2
Note :

Potential radionuclide emissions from stacks on the Hanford Site, Part 2: Dose assessment methodology using portable ...

Conference Paper 6 of 12

Author : Arndt, T. E.
Report Number: WHC-SA--2516
Secondary : CONF-940738--6
Trac No. SC: : M94015708
Entered : 08/23/94
Changed :
Description:
Note :

An example of a component replacement when applying ASME N509 and ASME N510 to older entilation systems

Conference Paper 7 of 12

Author : Bergman, W.
Report Number: UCRL-JC--115889
Secondary : CONF-940738--7
Trac No. SC: : M94017411
Entered : 09/01/94
Changed :
Description:
Note :

Degradation of HEPA filters exposed to DMSO

Conference Paper 8 of 12

Author : Gilbert, H.
Report Number: UCRL-JC--115891
Secondary : CONF-940738--8
Trac No. SC: : M94017410
Entered : 09/06/94
Changed : 06/20/95
Description:
Note : B;Rev.1

Preliminary studies to determine the shelf life of HEPA filters

Conference Paper 9 of 12

Author : Bergman, W.
Report Number: UCRL-JC--115890
Secondary : CONF-940738--9
Trac No. SC: : M95003648
Entered : 12/20/94
Changed :
Description:
Note :

Performance of HEPA filters at LLNL following the 1980 and 1989 earthquakes

Conference Paper 10 of 12

Author : Rodgers, J. C.
Report Number: CONF-940738--10
Secondary :
Trac No. SC: : M95007767
Entered : 03/27/95
Changed :
Description:
Note :

Single point aerosol sampling: Evaluation of mixing and probe performance in a nuclear stack

Conference Paper 11 of 12

Author : Bergman, W.
Report Number: UCRL-JC--115893
Secondary : CONF-940738--11
Trac No. SC: : M95009465
Entered : 05/03/95
Changed :
Description:
Note :

Preliminary field evaluation of high efficiency steel filters

Conference Paper 12 of 12

Author : Bergman, W.
Report Number: UCRL-JC--119798
Secondary : CONF-940738--12
Trac No. SC: : M95011438
Entered : 06/19/95
Changed :
Description:
Note :

Criteria for calculating the efficiency of deep-pleated HEPA filters with aluminum separators during and after ....

OPENING COMMENTS OF SESSION CO-CHAIRMAN HAYES

This afternoon's session will be dealing with new nuclear air cleaning codes and standards. I think Dr. First's introduction this morning with respect to a synopsis of the past air cleaning conferences is rather appropriate. This afternoon we will be dealing with both the old and the new. We will be dealing with the design of new equipment to new standards. We will be dealing with the application of the existing standards to old equipment. And we will also be seeing how this implementation has taken place for various other systems.

CHALLENGES OF EQUIPMENT QUALIFICATION USING TODAY'S STANDARDS WITH EMPHASIS ON A CLASS 1E MOTOR PROGRAM

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ABSTRACT

This paper describes qualification of new equipment for safety related service in the nuclear power industry in accordance with current industry standards. This topic is presented from the perspective of an original equipment manufacturer (OEM). Equipment qualification is first discussed in a general way then an example is provided of an electric motor qualification. A review of alternative qualification methods including commercial dedication is included. Potential difficulties with alternative / expedited qualification methods are also discussed.

1. Introduction

Fewer companies today are qualifying equipment for the nuclear power industry. The primary reason for this is the fact that no new nuclear power plants are being built in the U.S. Growth in the overseas nuclear power industry has been the primary driving force behind much of the equipment qualification that is being done today.

Given the lack of manufacturers offering pre-qualified equipment, the original equipment manufacturer (OEM) can and often must conduct qualification programs on equipment he intends to use on a particular contract.

This paper describes some of the experiences and challenges in qualifying equipment from the OEM perspective. The Ellis & Watts experience in qualifying our 1E motor design will be used as a typical example of equipment qualification.

In our efforts to qualify equipment we have encountered some alternative qualification methods that will be discussed, including commercial dedication and its role in motor qualification.

2. Qualifying Electrical Equipment

This section can be used in connection with IEEE 323 but is not intended to replace the guidance found in IEEE 323. All references to IEEE 323 are from the 1983 edition.

There are two fundamental reasons to qualify equipment:

1. To provide an adequate level of safety in a nuclear power plant.

2. To meet regulatory requirements in order to keep an operating license.

Successful equipment qualification efforts will satisfy three significant uncertainties of operating equipment:

1. Will the equipment be resistant to common mode failures due to aging degradation?

2. Will non metallic materials survive anticipated environmental stresses?

3. Will equipment and its mountings withstand the seismic forces generated during anticipated earthquakes?

There is a broad range of effort that could be expended in answering these three questions. On one end of the scale is an effort that is insufficient to demonstrate qualification. At the opposite end is an effort that is limited only by time and money. In other words, unlimited time and money could be poured into an equipment qualification program. The goal is an effort that provides reasonable documented evidence that the answers to the three questions above will be yes.

2.1 Generic or Application Specific

One of the first things in qualifying equipment is to establish qualification goals. The program may be conducted as a generic or application specific qualification.

The generic qualification is probably best for the OEM since it opens up the possibility of using the qualified item on a variety of programs in the future. In many cases the generic qualification program is prompted by the need for a qualified item on an existing equipment contract. This type of program requires the determination of realistic test parameters that will probably exceed the needs of the current program but not be so extreme as to reduce the chances of a successful qualification.

Qualification for an application specific program limits use of the component or equipment to programs having the same or reduced environmental stresses.

2.2 Mild or Harsh Environment

A determination must be made regarding mild or harsh environment classification. A mild environment qualification can usually be accomplished without determination of a qualified life (per Section 4 of IEEE 323).

A harsh environment program usually requires testing to verify performance in extreme accident conditions. Simulated aging is necessary to arrive at the end of life condition prior to accident condition testing.

Figure A-1 (see Appendix A) describes the steps to determine if an equipment item is subject to harsh or mild conditions. Even with the direction provided by Figure A-1 there is still some subjectivity involved in making this determination. Only about half of the specifications we encounter make the distinction. Some customer specifications use the absorbed radiation dose as a criteria for distinguishing between mild and harsh. Others use a combination of radiation dose and temperature.

Even if the qualification program is based on a customer specification, there is often insufficient detail to determine what qualification steps are required. The foundation upon which any successful qualification program is built is a complete specification (Paragraph 6.1 of IEEE 323). As an OEM we see many examples of incomplete qualification specifications that must be clarified before the program can proceed.

2.3 Safety or Non-Safety Related Function

It is also necessary to determine if the equipment's function is safety related or non-safety related. Non-safety related items can often be excluded from the qualification process if it can be shown that a failure of that component would have no adverse effect on the safety function of the over all equipment (Paragraph 6.1.4 of IEEE 323). The qualification guidance in IEEE 323 is primarily directed towards equipment performing a safety function.

2.4 Qualification Plan

A qualification plan must also be developed in accordance with IEEE 323. The plan must include a determination of the qualification method, a listing of the environmental service conditions, a description of any required aging program, a prescription of the test sequence, and a definition of the accident test profiles.

An aging program would consist of such things as thermal aging, mechanical/cyclic aging, radiation exposure and mechanical vibration. All of these stressors are designed to simulate the conditions that would be encountered during the expected life of the test specimen prior to accident condition testing (seismic, HELB, LOCA, etc.).

Section 7 of IEEE 323 provides generic time/temperature/pressure profiles for simulated accident conditions. However, all values are to be provided by the developer of the test program.

Figure A-2 (see Appendix A) summarizes the requirements of IEEE 323 in the preparation of a qualification plan.

2.5 Documentation

The final requirement of a typical qualification program is the generation and compiling of program results (per Section 8 of IEEE 323). The documentation must be in a form that would permit confirmation of the results by an independent third party who is knowledgeable in the field of equipment qualification.

3. Ellis & Watts Motor Qualification Program

The qualification of an Ellis & Watts designed and fabricated 1E motor design will be presented as an example of the implementation of the program guidelines described above. IEEE 334-1974 was used as specific guidance for motor qualification in addition to IEEE 323.

This program was conducted in order to establish 1E (safety related) qualification for the Ellis and Watts motor design. The qualification process described herein is in accordance with IEEE Standards 323-1983, 334-1974 and IEEE Recommended Practice 344-1987. Testing performed for this program is for motors serving outside of containment areas.

The motor qualification program was conducted in two phases. The first phase consisted of the thermal aging of motorettes in order to establish a qualified life for the insulation system (as described in IEEE 117). Phase two consisted of prototype motor testing (type tests) (per Section 6 of IEEE 334-1974).

The purpose of the type test program was to demonstrate that the Ellis and Watts motor is capable of surviving a representative design basis event (DBE) after a simulated 40 year life and remaining functional for a specific time period after the DBE. The design basis events considered for this program were seismic and high energy line break (HELB) outside containment. A complete qualification test plan was prepared for the program.

3.1 Applicable Documents

A review of the available guidance for motor qualification revealed that the following documents would be useful and necessary in defining and conducting the motor qualification program:

IEEE 101-1987 Guide For The Statistical Analysis Of Thermal Life Test Data.

IEEE 117-1974 Standard Test Procedure For Evaluation Of Systems Insulating Materials For Random - Wound AC Electric Machinery.

IEEE 323-1983 Standard For Qualifying Class 1E Equipment For Nuclear Power Generating Stations.

IEEE 334-1974	Standard For Type Tests Of Continuous Duty Class 1E Motors For Nuclear Power Generating Stations.
IEEE 344-1987	Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.
UL 1446	Standard For Safety, Systems Of Insulating Materials.
ANSI / ASME NQA1-1989	Quality Assurance Program Requirements For Nuclear Facilities.

3.2 Qualification Test Summary

3.2.1 PHASE 1 Phase 1 of the qualification program established a qualified life for the Ellis and Watts insulation system of 44 years at a 130°C design temperature. Motorette insulation system samples were used and the motorettes were subjected to the battery of simulated aging tests described in IEEE 117. These tests consisted of heat aging, vibration, and moisture (in that sequence). IEEE 117 provides complete detailed instructions for fabrication of motorette insulation system samples.

In addition to IEEE 117 tests, the Ellis and Watts program added the cold shock test described in UL Standard 1446 to the environmental stress sequence. The cold shock was performed just before placing the motorettes into the humidity chamber. In general UL 1446 parallels the requirements of IEEE 117; however, IEEE 117 does not include the cold shock step.

Ten motorettes were taken through repeated cycles of heat, vibration, cold shock and humidity at each test temperature until the motorettes failed. Failure was determined by the voltage test (per Section 2.3 of IEEE 117).

After the motorettes failed, the number of cycles and the total number of hours of heat aging were recorded as test results. This data is provided as an average for each group of motorettes at each temperature. With this information a regression line for the insulation system was obtained using the analytical methods described in IEEE 101-1987. The curve developed from this data allowed the projection of a qualified life at a given motor temperature.

The insulation system was tested against a previously qualified, UL recognized Class H (180°C) control group. The life of the Ellis and Watts system exceeded the minimum life requirements of the Class H control group. A 44 year motor life is established by extrapolating the insulation system regression line to a Class B (130°C) maximum operating temperature. Ellis and

Watts motors are designed for a Class B maximum temperature rise in a 50°C ambient.

3.2.2 PHASE 2 After the 40 year minimum qualified life of the insulation system was established, the next phase of the program was simulated aging and design basis event testing on a prototype motor. The prototype motor (serial number 13703) used in the qualification program was a 7.5 horsepower, 1750 RPM, totally enclosed air over (TEAO) design in a 215T frame.

Phase 2 results are summarized below. The testing was conducted in the sequence listed:

1. THERMAL AGING: The prototype motor was subjected to thermal aging per Section 9 of IEEE 334-1974 to an equivalent of 44 (40 + 10% margin) years at 130°C operating temperature (NEMA Class B temperature rise). Based on the average life characteristic of the insulation system (established in Phase 1) the required number of thermal aging hours needed was determined to be 1698 at a selected aging temperature of 240°C. The prototype motor was thermally aged to an equivalent of 44 years (40 + 10% margin) at an assumed winding temperature of 130°C. The accelerated aging temperature and exposure time for this simulation was determined by use of the Arrenihus curve developed during the insulation system motorette tests conducted in phase 1 of the qualification. IEEE 334-1974 calls this curve the "implied average life characteristic" for the insulation system.

   A line parallel to the insulation system life line was drawn by starting at a point representing the desired life time (y axis) and the maximum operating temperature of the motor (x axis). This second line represents the aging characteristic of the motor. The point on this new line corresponding to an acceptable accelerated time will establish the accelerated aging temperature. This method of establishing the accelerated aging time and temperature is described in Section 9 of IEEE 334-1974.

   Testing showed motor operating temperatures to be below a Class B rise. Therefore, 130°C was used as the expected maximum operating temperature. At this temperature and 44 year life, 1698 hours at 240°C was selected as the accelerated thermal aging condition.

   For this portion of the test only the prototype motor housing and stator were placed in the oven since these were the only parts of the motor subject to the effects of thermal aging. Periodic monitoring of the oven temperature was documented over the 1698 hour time period.

2. MECHANICAL AGING: Mechanical aging was by means of the one hour mechanical vibration test described in paragraph 9.5 of IEEE 334 and paragraph 2.2.3 of IEEE 117. This test consisted of mechanical vibration of the stator for one hour at 1.5 times the acceleration of gravity at a frequency of 60 Hz. with 8 mils double amplitude displacement. This was the same vibration exposure seen by the motorettes during each cycle of phase 1 testing.

   After these first two steps in the test program, the prototype motor was reassembled and the complete motor, including the lubricant, bearing and seal system was subjected to all subsequent testing.

3. ROUTINE PERFORMANCE TEST: A routine performance test was conducted per IEEE 112 Form A1. This test provided baseline functional test data and was conducted with the motor unloaded both prior to and after thermal aging. There was no significant difference in the two sets of data which indicated that the motor was functioning properly after thermal and vibration aging.

4. RADIATION EXPOSURE: The thermally aged motor was exposed to a gamma radiation source to obtain a total accumulated dose of 3×10^7 rads minimum (equivalent air dose). The dose rate was between 250K and 1 megarad per hour. The tolerance of dosimetry instrumentation was taken into account to assure that a minimum dose of 3×10^7 rads gamma was achieved.

   The test lab that performed the radiation exposure provided a certification that the specified minimum dose was received and that the radiation source was Cobalt 60. The motor was rotated during this test to permit more equal radiation exposure to the various motor parts.

5. CYCLIC OPERATION: Additional mechanical aging was simulated by cyclic operation with the motor under load. The prototype motor was started and stopped under full load 520 times which would be equivalent to starting the motor approximately once a month over a 40 year life. In addition to these starts, it was estimated that the prototype motor was stopped and started another 20 to 30 times in the course of the other testing.

Each of the 520 cycles consisted of the following steps:

1) Start and operate motor for 1 minute.

2) Stop motor and keep off for 3 seconds after shaft rotation stops.

3) Stop motor for 10 minutes every seventh repetition of steps 1 and 2.

6. SEISMIC TESTS: The prototype motor was seismic tested in accordance with a generic response spectra created by Ellis & Watts to envelope the peak seismic conditions encountered in typical specifications. The motor under test was operating at full load current during the seismic event simulation.

Seismic tests consisted of 5 operating basis earthquakes (OBE's) and 1 safe shutdown earthquake (SSE) as described in Paragraph 6.1.4 of IEEE 344-1975. A detailed seismic test procedure was prepared by the independent test lab that performed the test. The test procedure contained the following general approach and was submitted to Ellis and Watts for approval.

A resonant frequency search was conducted in each principal axis. Ellis and Watts provided an identical duplicate test specimen for this portion of the seismic testing. The thermally aged motor was not subjected to a resonant frequency search. The input amplitude for the resonance search was the lowest possible that would enable determination of resonant frequencies.

The full level seismic test was performed on a tri-axial random motion vibration table. The test consisted of simultaneous horizontal and vertical inputs of a random motion over a frequency range of 1 to 33 Hz. The amplitude of these input motions were adjusted until the actual test response spectra (TRS) exceeded the Ellis and Watts generic response spectra. A damping value of 2% was used for OBE simulations and 3% was used in SSE simulations.

Acceptance criteria for the full level OBE and SSE simulations is as follows:

1) No observed structural failure is permitted. A thorough visual inspection is conducted after each simulation to locate any structural failures or loose hardware.

2) The motor must continue to operate properly at full load current during all simulations.

Any failure to meet these acceptance criterion will be reported to Ellis and Watts for resolution.

7. ACCIDENT SIMULATION: The high energy line break simulation (HELB) was performed at Wyle Laboratories. The motor operated under load for a period of six hours in the HELB chamber at temperatures which exceeded the required time/ temperature profile. The HELB simulation results were documented in Wyle Test Report Number 43030-1, Revision A .

The high energy line break (HELB) simulation was conducted with the motor operating under load. The test consisted of subjecting the motor to saturated steam to achieve the elevated temperature and high humidity conditions. The simulation ramped up to 212°F in the minimum time possible and held 212°F for 50 seconds minimum. Then the temperature gradually dropped following the generic time / temperature curve compiled by Ellis & Watts for this test. The overall duration of the HELB simulation was one hour minimum.

Ambient pressure was intended to be the prevailing ambient atmospheric pressure at the test laboratory. However, some pressure increase was experienced during the ramp up period. The ambient pressure surrounding the motor was recorded during the test and reported in the test report.

8. POST ACCIDENT ENDURANCE TEST: A post accident endurance test was conducted to simulate operation of the motor for 100 days, under load in an elevated temperature environment.

The post accident endurance test was also intended to address the additional guidelines on aging contained in IEEE 323-1983. The post accident endurance test was performed by Ellis and Watts. This test continued from the point where the HELB test stopped by simulating continuous operation of

the motor at maximum design conditions (130°C winding temperature) for 100 days.

In order to reduce the time required to conduct this test a regression analysis was performed using data obtained from the insulation system life test. The regression analysis showed that the 100 days (2400 hours) can be reduced to approximately 500 hours if the winding temperature is increased to 160°C.

A test enclosure was assembled in the Ellis and Watts test lab large enough to house the motor. The motor operated under load within the test enclosure and the ambient temperature was varied as required to maintain a 160°C winding temperature for approximately 500 hours.

A shorter test time could have been used with a corresponding increase in winding temperature in accordance with the regression line slope established in the insulation system life testing. The actual time and temperature used in the test were documented and supporting analysis was provided in the final test report written by Ellis & Watts.

9. FINAL ROUTINE PERFORMANCE TEST: After completion of the post accident endurance test, a second routine performance test was performed on the motor. This routine performance test was conducted in the same manner as all previous routine performance tests for the program so that a comparison of results would be meaningful. The comparison showed that the differences between initial and final routine performance test results is negligible. A discussion of this comparison is provided in the qualification test report prepared by Ellis & Watts.

10. VISUAL INSPECTION: After testing was completed, the motor end bells were removed to permit a visual inspection of the interior of the motor housing. The object of this inspection was to locate obvious signs of damage caused by the stresses of the test program.

A written description of the visual inspection is included in the test report prepared by Ellis & Watts.

4. Other Considerations

The motor qualification test program also addressed synergistic effects, lubricants, bearings and seal systems as required by IEEE 334.

5. Quality Assurance Provisions

The testing program covered the qualification of a safety related 1E electrical component. For such a program the requirements of 10CFR50 Appendix B and ANSI N45.2 applied. The Ellis and Watts QA program was in compliance with these documents and also met the basic requirements of ASME NQA-1. Ellis and Watts QA manual QC4000N documented the implementation of these requirements at our facility. The reporting requirements of 10CFR PART 21 also applied to the program.

All steps of the above described testing were documented in the Ellis & Watts motor qualification report (ENG 601).

6. COMMERCIAL DEDICATION

The term dedication refers to actions taken to use a commercial grade item in a safety related application within the nuclear power industry.

Commercial dedication is only applicable to parts which are classified as safety related. That is, their failure would prevent the equipment form performing its safety related function.

According to ANSI/ASME NQA-1-1986, a "commercial grade item" must meet all three of the following criterion:

1) It must not be subject to design or specification requirements that are unique to nuclear facilities.

2) It must be used in applications other than nuclear facilities.

3) It must be available from the supplier on the basis of specifications set forth in the manufacturer's literature such as a catalog.

A commercial grade item is said to be dedicated at the point when it is accepted for a safety related application. Deficiency reporting (10 CFR 21) then becomes the responsibility of the party performing the acceptance.

Dedication includes determining if the item performs a safety function and meets the definition of commercial grade. Another requirement is the identification of characteristics which are critical in performance of a safety function and therefore must be verified. Dedication also includes the selection and application of an acceptance method.

The Electric Power Institute (EPRI) Document NP 5652 describes four acceptance methods:

1) Special tests and inspections

2) Commercial grade survey of the supplier

3) Source verification

4) Acceptable supplier / item performance record

Figure A-4 (see Appendix A) is an excerpt from EPRI NP5652 which graphically describes the entire commercial dedication process.

Commercial dedication was first described in 10 CFR 21 in recognition of the fact that not all items used in a nuclear power facility could be obtained from a manufacturer or supplier to nuclear industry design and QA standards.

The evolution of commercial dedication in the nuclear power industry was motivated by the need to justify procurement of commercial parts because items qualified to nuclear industry standards were not available. Dedication was not developed to allow the use of commercial items in lieu of items that were available pre-qualified to nuclear power industry standards.

The use of commercial grade items in lieu of available items qualified to nuclear industry standards does not appear to be prohibited, but there have been difficulties for utilities when this type of procurement is practiced. This is especially true when third party qualifiers are used.

7. Evaluation of Qualification Programs - "Utility Pitfalls"

Procurement of nuclear qualified components can become a very costly and disappointing experience for a utility. Even though the specifying engineer includes all the correct IEEE standards, various interpretations may be taken by equipment and "third party" qualifiers. The NRC's position on qualification is that it is the owner's responsibility to insure compliance. As a result "short cut" qualification methods have been used that may not meet all the requirements of the standards. The "short cut" qualification methods seem to be driven by economic and/or schedule restraints. In many cases, alternative qualification methods do not include all the steps required in the current standards. Utilities must scrutinize these methods to assure compliance.

In relation to motor qualification, three alternative qualification methods that have been observed in the industry are commercial dedication, motor repair-other insulation systems, and expedited test methods. Each are briefly discussed below.

7.1 Commercial Dedication of Class 1E Motors

Commercial dedication of equipment is an acceptable method of equipment qualification. An electric motor, however, is a very questionable candidate for this type of qualification.

The Electric Power Institute (EPRI) Document NP 5652 provides the industry definition for commercial grade dedication and includes the requirement that a commercial grade item not be subject to design or specification requirements that are unique to nuclear facilities. Since there are nuclear industry standards unique for electric motors, commercial dedication is not applicable to Class 1E motors. IEEE 334-1974 is written specifically to address Class 1E motors used in nuclear power plants. IEEE 323 and 344 provide the environmental and seismic requirements respectively. However, utilities have been offered commercially dedicated Class 1E motors on the open market. Without performing all the qualification tests in the sequence described in Section 3.0 of this paper, a commercial dedicator cannot satisfy the requirements to assure all critical attributes are in compliance. As noted previously, the NRC holds the utility responsible for compliance.

7.2 Motor Repair - "Other" Insulation Systems

The qualified design life of a Class 1E motor is normally 40 years. However, utilities have experienced motor failures during normal usage and require motor repair. Ellis & Watts has encountered motor service shops providing this service to the utilities. The motor service shops are not the original supplier of the motors. When the motors are returned to the utility after repair/rewinding, the insulation system has usually been changed from that provided by the motor qualifier. The new insulation system has not endured the stringent qualification of the original insulation and in most cases is not the same material as that originally qualified. Motor manufacturers hold the insulation system used as proprietary because the insulation system is the key to qualification. Little justification is provided to demonstrate qualification of the substitute insulation system, yet the motor is returned to the utility as a "qualified" replacement. It is up to the utility to review and scrutinize to assure compliance.

In addition, one motor service shop claimed that seismic qualification was unnecessary because motors are inherently rugged.

7.3 Expedited Test Methods

Another phenomenon relating to motor qualification is third party qualifiers providing "expedited qualification". In one example, a single motor stator was aged in lieu of motorettes. Use of the motor stator is not, in itself, a problem. However, if motors are used for insulation system qualification instead of motorettes, a quantity of five (5) is required by IEEE 117.

No determination of the insulation system temperature class was performed. The third party qualifier assumed that the commercial motor manufacturer's advertised insulation class was correct. In addition, the statistical analysis of estimated qualified life for the insulation system was not performed. The single sample stator was merely aged and then subjected to accident condition testing. This motor was then offered as a Class 1E qualified motor for harsh environments.

8. CONCLUSION

The true purpose of equipment qualification is to verify design adequacy for assurance of public health and safety. Current IEEE Standards provide the guidance to accomplish this task in a manner that satisfies regulatory documents.

When purchasing equipment for safety related applications, utilities must be aware of potential problems associated with incomplete or abbreviated test methods. Expedient methods of qualification to improve delivery do not always satisfy all requirements of the applicable standards.

The NRC's position is very clear "the licensee has the ultimate responsibility to confirm that a given equipment or component is qualified for a safety related application".

Appendix A. Figures

FIGURE A-1

DETERMINATION OF MILD OR HARSH ENVIRONMENT

If your answers to all of the questions below are "Yes," then the equipment should be treated as mild environment equipment, otherwise, it is covered under the harsh environment qualification program.

1. The environment in the room where the equipment is located is unaffected during and after a DBA, i.e. no significant changes in temperature, radiation, etc.

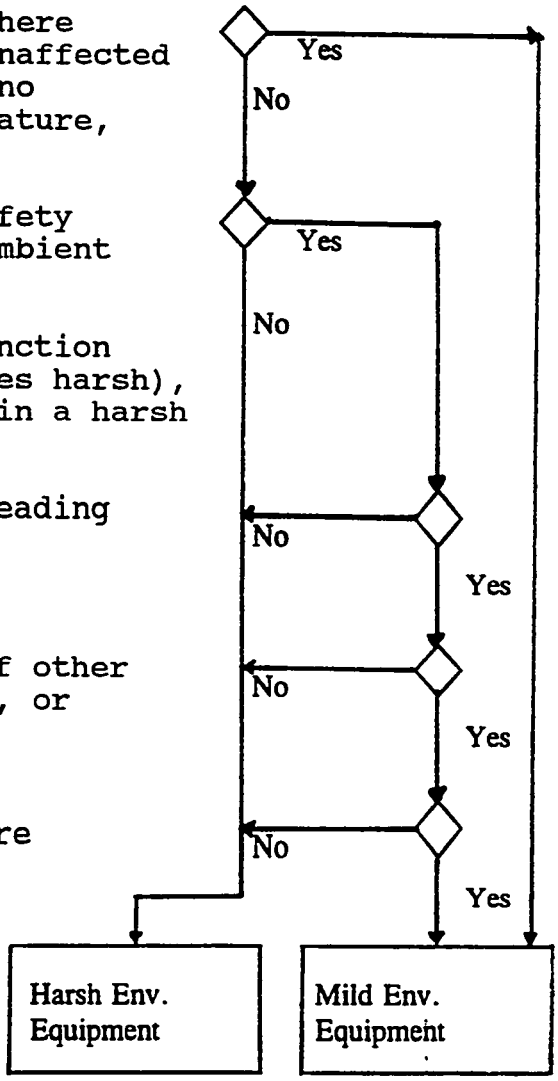
2. The equipment performs its safety related function before the ambient environment becomes harsh?

3. After it has performed its function (before the environment becomes harsh), the failure of the equipment in a harsh environment will **NOT**:

   a) result in providing misleading information, or

   b) affect the functioning of other safety related equipment, or

   c) cause a breach of pressure boundary integrity.



Source credit: EPRI/MOS environmental qualification training course. Used with permission.

FIGURE A-2

METHODS FOR QUALIFYING EQUIPMENT

1. TYPE TESTING

- ACCOUNTS FOR SIGNIFICANT AGING MECHANISMS
- SUBJECTS THE EQUIPMENT TO SPECIFIED SERVICE CONDITIONS
- DEMONSTRATES SUBSEQUENT ABILITY TO PERFORM SAFETY FUNCTION

2. OPERATING EXPERIENCE

- MUST BE COMPARED TO EQUIPMENT HAVING THE SAME GENERIC DESIGN
- VALIDITY DEPENDS ON DOCUMENTATION OF PAST SERVICE CONDITIONS, EQUIPMENT PERFORMANCE, MAINTENANCE AND SIMILARITY

3. ANALYSIS

- REQUIRES LOGICAL ASSESSMENT OR MATHEMATICAL MODEL OF THE EQUIPMENT
- REQUIRES SUPPORT OF TEST DATA, OPERATING EXPERIENCE OR PHYSICAL LAWS OF NATURE
- MUST BE DOCUMENTED IN A WAY THAT PERMITS VERIFICATION BY A COMPETENT THIRD PARTY.

4. COMBINED METHODS

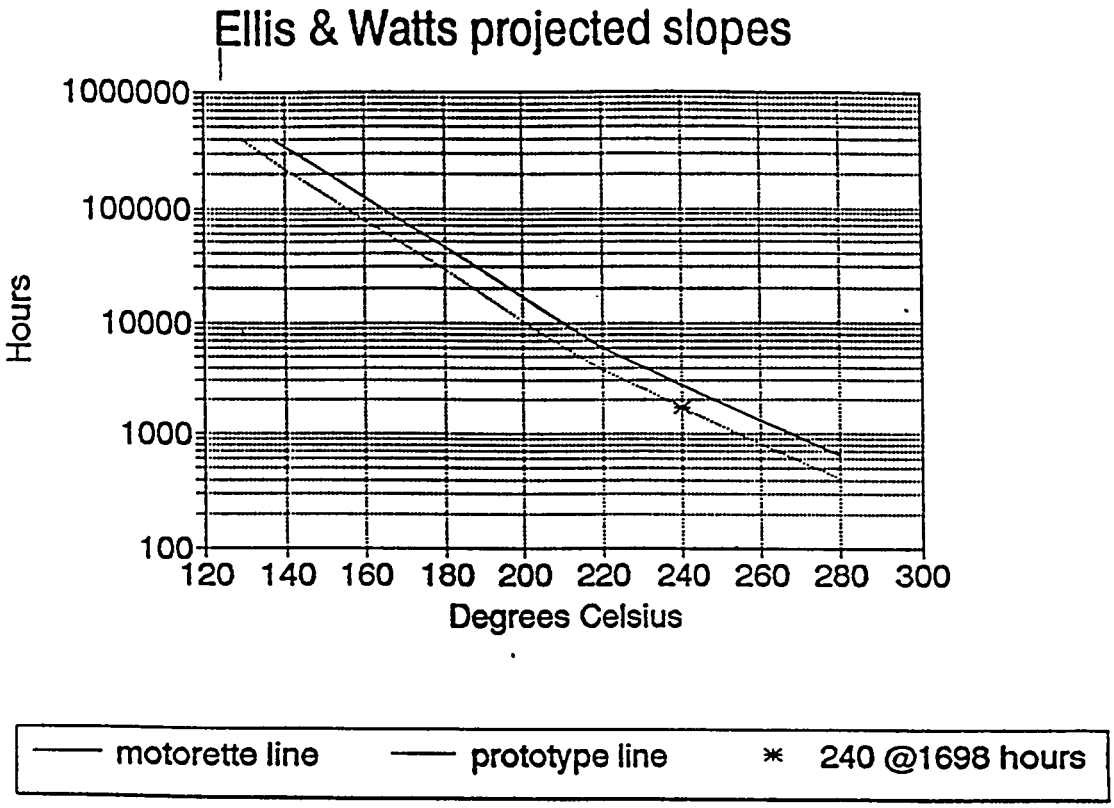
- REPRESENTS A COMBINATION OF ANY OF THE ABOVE QUALIFICATION METHODS

FIGURE A-3

AVERAGE LIFE REGRESSION LINE FOR

ELLIS & WATTS

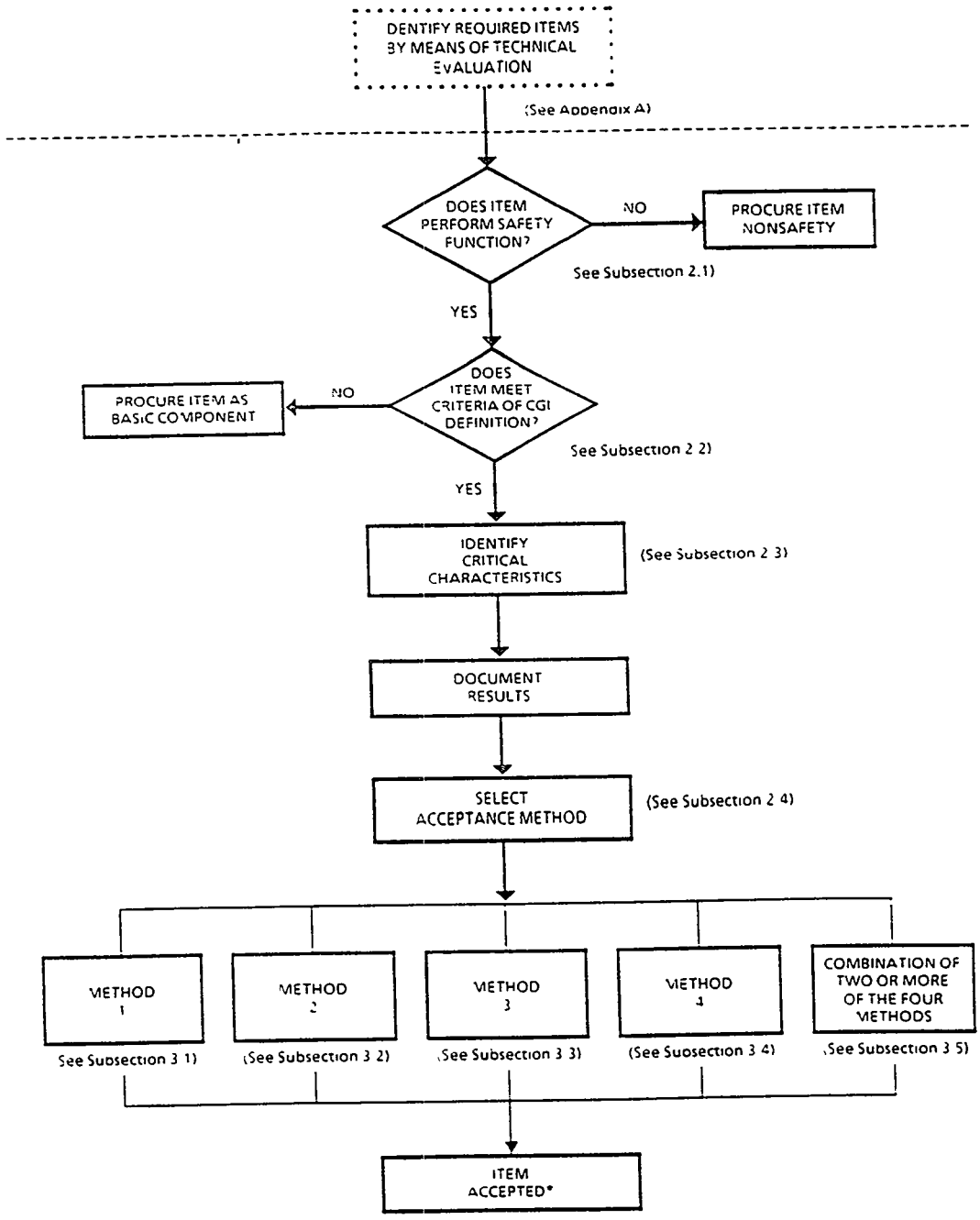
MOTOR INSULATION SYSTEM



This data as of Aug. 31, 1993.

FIGURE A-4

COMMERCIAL DEDICATION SEQUENCE



*Deficiency reporting responsibility accepted.

Source Credit: EPRI NP-5652 Project Q101-7, Final Report, June 1988 *Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Application (NCIG-07)*

DISCUSSION

ADAMS: A concern I have about motor qualification refers to an item on my desk right now. Often, when motors are qualified, they do voltage degradation testing while motors are coupled to a fan or something else. Does your qualification program account for voltage degradation with changes in efficiency and power factor at those degraded voltages? A lot of testing is based on 460 V but you take it up to 560 V and sometimes down to 460 V. A concern I have is that the power factor and the efficiency of a motor does change as voltage goes down, and current response changes drastically. You can actually lose a motor not within specification or covered by a qualification program.

DEATON: I can tell you that our qualification program was based on standard IEEE 334, 1974 edition. In the past, it did not go into much detail on your concerns. However, just in the past month there has been a 1994 revision to IEEE 334, and your concerns about voltage variation and degraded voltage testing are addressed there. Currently, we have another motor test in progress. We will be incorporating test sequences as a part of the program.

BELLAMY: I am from the Nuclear Regulatory Commission, so you have to understand that I am coming at this question from the regulatory side of the house. One of your first slides presented two fundamental reasons to qualify equipment. One was to provide an adequate level of safety, and the second one was to meet regulatory requirements. As a supplier (I think the term you use is, original equipment manufacturer), do you really see a difference between those two goals? Is there a difference between providing an adequate level of safety versus satisfying the regulatory requirements? As a regulator, I hope they are close together. But from a manufacturer's viewpoint, do you see them as significantly different?

DEATON: Not really. I would agree with you that the adequate level of safety is in conjunction with maintaining your license; to satisfy one satisfies the other, also.

EVALUATION OF SELF-CONTAINED HEPA FILTER

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Abstract

This paper presents the results of an evaluation of a self-contained high-efficiency particulate air filter (SCHEPA) used in nuclear applications.

A SCHEPA consists of filter medium encapsulated in a casing that is part of the system boundary. The SCHEPA filter serves as a combination of filter housing and filter. The filter medium is attached directly to the casing using adhesive as a bonding agent. A cylindrical connection in the middle of the end caps connects the filter assembly to adjoining ductwork.

The SCHEPA must perform the functions of a filter housing, filter frame, and filter. It was recognized that the codes and standards do not address the SCHEPA specifically. Therefore, the investigation evaluated the SCHEPA against current codes and standards related to the functional requirements of an air-cleaning system. The specific standards used are required by DOE Order 6430.1A[1] and include ASME N509[3], ASME N510[4], ERDA 76-21[5], MIL-F-51068F[6], NFPA 90A,[7] and NFPA 91[8].

The evaluation does not address whether the SCHEPA as a standard (off-the-shelf) filter could be upgraded to meet the current code requirements for an air-cleaning unit. The evaluation also did not consider how the SCHEPA was used in a system (e.g., whether it was under positive or negative pressure or whether it served as an air inlet filter to prevent contamination releases under system pressurization).

The results of the evaluation show that, the SCHEPA filter does not meet design, fabrication, testing, and documentation requirements of ASME N509[3] and ASME N510[4]. The paper will identify these deficiencies.

Specific exhaust system requirements and application should be considered when an evaluation of the SCHEPA filter is being performed in existing systems. When new designs are being contemplated, other types of HEPA filter housings can be used in lieu of the SCHEPA filter.

I. Introduction

Scope of Evaluation

The evaluation examined the self-contained high-efficiency particulate air (SCHEPA) filter for code compliance with DOE Order 6430.1A[1] relative to nuclear-grade HEPA filters. The specific standards in DOE Order 6430.1A[1] used to evaluate the SCHEPA included ASME N509[3], ASME N510[4], ERDA 76-21[5], MIL-F-51068F[6], NFPA 90A[7], and NFPA 91[8].

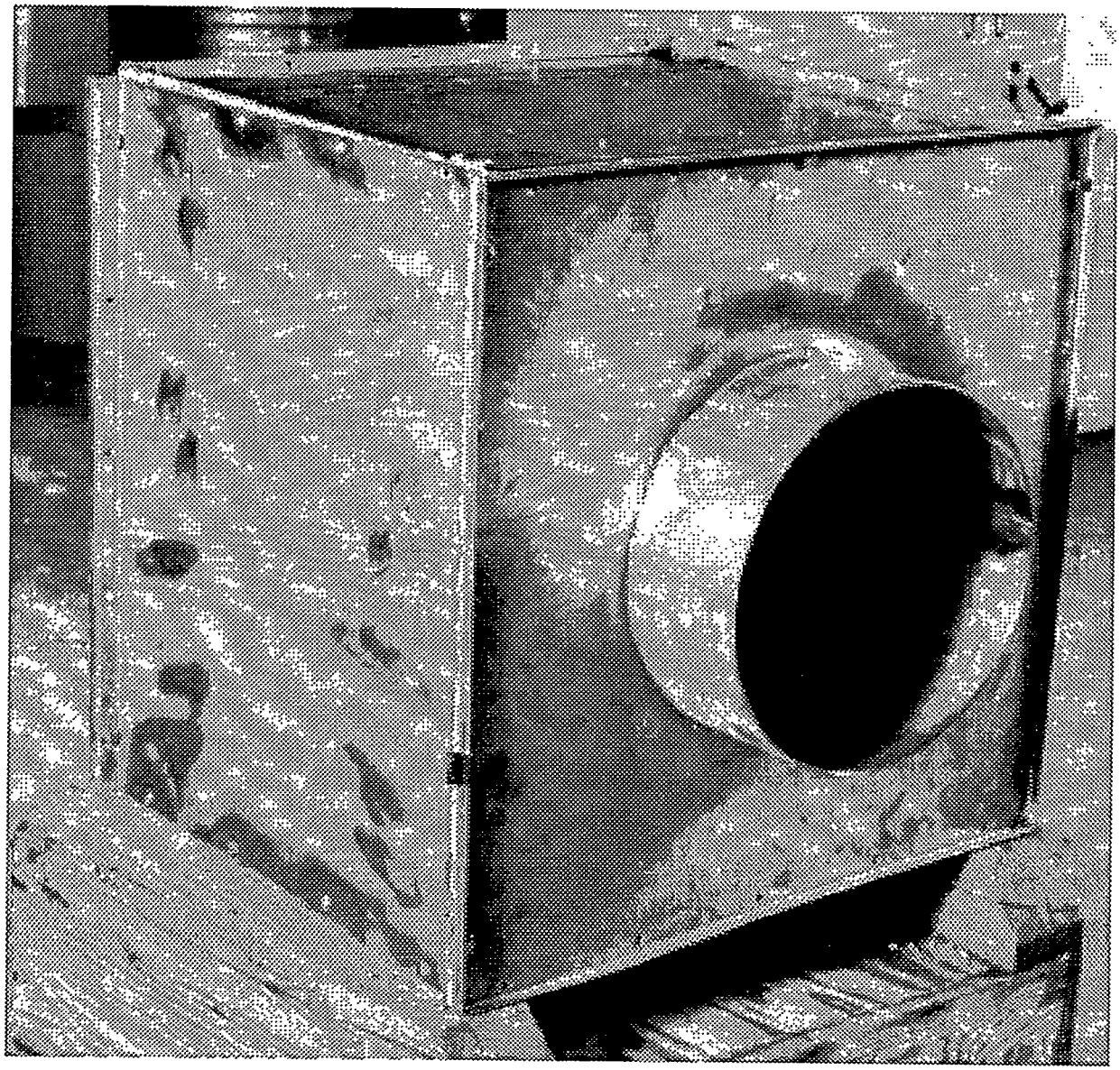


Figure 1. Self-Contained High-Efficiency Particulate Air Filter.

The evaluation took a general approach (i.e, that the SCHEPA filter in a nuclear safety system shall meet all applicable code requirements pertaining to nuclear air-cleaning components). The evaluation did not consider whether the SCHEPA, as a standard (off-the-shelf) filter, could be upgraded to meet the current code requirements for an air-cleaning unit nor did the evaluation consider how the SCHEPA was used in a system (e.g., whether under positive or negative pressure).

II. Description of SCHEPA Filter

The SCHEPA filter is a high-efficiency particulate air (HEPA) filter consisting of filter medium enclosed in a casing that is part of the system boundary. The casing consists of two end caps and four pieces of siding, which form a square frame around the filter media.

The filter media is centered in the casing and is attached directly to the casing with an adhesive used as a bonding agent. End caps cover the medium at the front and back of the frame. A cylindrical connection in the middle of the cap connects the filter assembly to adjacent ductwork. The distance between the filter medium and the caps varies with the type of filter media.

Because SCHEPA filters could be used in safety-related ventilation systems, a better understanding of these filters and their qualification requirements is needed to ensure safe operation of nuclear ventilation systems.

III. Definitions

The SCHEPA filters are not specifically addressed in the ASME Code and DOE Orders. To determine the applicable requirements of the code, the evaluation reviewed definitions of various components provided in the codes and assessed the SCHEPA on the bases of its configuration, design parameters, fabrication, application, and performance requirements.

The following definitions are quoted from ASME N509[3], Section 3.

- **Duct**--An enclosed passage through which air is transferred from point to point; typically will not include air-cleaning components such as HEPA filters or absorber air-cleaning units.

- **Housing**--The portion of an air-cleaning unit that encloses air-cleaning components and provides connections to adjacent ductwork.

- **Components, air-cleaning**--Equipment that is contained in nuclear air treatment systems. Typically components may include dampers, demisters, or moisture separators, heaters, prefilters, HEPA filters, charcoal absorbers, and fans.

- **Air-Cleaning unit**--An assembly of components comprising a self-contained subdivision of a complete air-cleaning system. It includes all the components necessary to achieve a unit air-cleaning function such as removing particulate matter (filter) or iodine vapor. A unit includes a housing plus internal

air-cleaning components and may include one or more auxiliary air-treatment components such as prefilters, post-filters, heaters, coils, and moisture separators.

IV. Basis for Code Compliance

The evaluation applied DOE Order 6430.1A[1] as the basis for code compliance. DOE Order 6430.1A[1] states in part that "These criteria apply to any building acquisition, new facility, facility addition and alteration, and leased facility that is required to comply with DOE 4300.1B[2]. This includes on-site constructed buildings, pre-engineered buildings, plant-fabricated modular buildings, and temporary facilities." DOE 6430.1A[1], Section 1550-2.5.5, titled "Air-Cleaning Devices," requires that all HEPA filtration systems comply with ASME N509[3] and be tested in accordance with ASME N510[4].

In addition to the foregoing requirements, RLIP 5480.4C[9], a Hanford Site requirement, titled *Environmental Protection, Safety, and Health Protection Standard*, Section 10, "Nuclear Safety Standards," paragraph b, "Nuclear Facility Safety," also invokes ASME N509[3] and ASME N510[4] as mandatory standards.

A review of ASME N509[3], ASME N510[4], and MIL-F-51068[6], showed that they do not address the SCHEPA nor were they written with the SCHEPA in mind.

On the basis of the definitions given in Section 3.0 of ASME N509[3], the SCHEPA filter was classified for the evaluation as an air-cleaning unit and, therefore, had to meet the applicable requirements identified for a filter housing.

V. Summary of Evaluation

A SCHEPA filter classified as an air-cleaning unit consists of two components, a casing, and a HEPA filter. Unlike the other type of HEPA filters, designed to be attached to the mounting frame within a housing in an exhaust system, the casing of a SCHEPA filter serves as a combination of filter, filter frame, and a filter housing that is a pressure boundary that must satisfy code requirements for a filter housing given in ASME N509[3] and N510[4]. The filter medium and frame must meet the HEPA filter specification given in MIL-F-51086[6].

The evaluation revealed that the SCHEPA filters do not have the documentation required for compliance with design, testing, and fabrication standards in ASME N509[3], ASME N510[4], and MIL-F-51068[6]. Without this documentation, filter adequacy cannot be verified against requirements of DOE Order 6430.1A[1]. The evaluation also revealed that the SCHEPA filter does not meet the airflow distribution requirements of ASME N509[3].

VI. Summary of Noncompliance

Findings of noncompliance with codes and standards are summarized in the following sections. These sections identify the specific paragraph(s) or section(s) with which the filter does not comply and explain why these criteria must be met.

Noncompliance Pertaining to ASME N509, *Nuclear Power Plant Air-Cleaning Units and Components*

Structural Load Requirements Section 4, paragraph 4.5, states, that "The engineered safety feature (ESF) systems and all of their components shall be shown, either by testing or by a mathematical technique, to remain functional under the structural loading specified in Section 4.2(k), 'Structural Loadings,' and described in ASME N509[3], Section 5, paragraph 5.10.3, 'Structural Requirements'." The SCHEPA filter does not have documentation demonstrating that it has met these criteria.

Leak-Test Pressure Section 4, paragraph 4.6.4, states that the leak-test pressures and components (i.e., ducts and housings) must be tested in the shop and field. Furthermore, it states that the test pressure must be documented by the engineer. The SCHEPA filter does not have this documentation.

Negative Pressure Section 4, paragraph 4.6.5.4.(b), states that "Air-cleaning units and components located on the inlet side of fan(s) which can be isolated by closure of an upstream damper, or potentially plugged components shall be designed to withstand a negative internal pressure equal to or more negative than the peak pressure of the fan(s)."

It is a requirement that filters be isolated with dampers for filter removal. If an upstream damper were suddenly closed or the entrance to the filter plugged during normal operation, with the fan peak pressure at a level that surpasses the structural integrity of the casing, the casing could fail and damage the filter medium. The casing of the SCHEPA filter must have test data available to show that it can withstand the structural requirements. The SCHEPA filter does not have this documentation.

HEPA Filters Section 5, paragraph 5.1, states that "HEPA filters shall meet the construction, material, test, and qualification requirements of military specification MIL-F-51068[6], except that listing of manufacturer's HEPA filter qualified products list (QPL) is not required." The SCHEPA is not identified in the MIL-F-51068(6).

Construction Section 5, paragraph 5.1.1, states that "Filters for use in containment or in ESF systems shall be metal case type (Type II frames as defined by MIL-F-51068[6]) and shall be compatible with chemical composition of the air stream. Filter systems exposed to temperatures greater than 200 °F shall have steel sides." A wooden SCHEPA filter in an ESF system does not comply with this requirement.

Documentation Section 5, paragraph 5.1.3, requires that a certificate of conformance be provided to the owner certifying that the filter assembly has been designed in accordance with MIL-F-51068(6). The SCHEPA does not have the required documentation.

General Requirements Section 5, paragraph 5.6.1, states that the layout of the filter housing shall have uniform airflow within ± 20% of the average through each bank of components. Because a SCHEPA is not designed in accordance with ASME N509(3), it does not meet this requirement.

Housing Doors Section 5, paragraph 5.6.2, (a), requires that a filter housing be equipped with doors. Doors are necessary for inspecting and removing the filter. The SCHEPA filter has no doors. The SCHEPA filter is inspected and replaced by removing the entire SCHEPA filter assembly from the system.

Drains Section 5, paragraph 5.6.2, (c), requires that the housing be equipped with a drain to remove water that may accumulate in the housing. Water may collect in the filter housing if a steam heater develops a leak. Rain or snow may infiltrate the housing through a loosened flanged connection, a crack in the filter housing, an improperly sealed access door, etc.

If the heater fails and the gas stream has high humidity or is saturated, condensation will occur in the housing. Systems that do not normally have high humidity in the air stream and do not have a heater may also experience condensation on the interior of the housing. This phenomenon will occur when the system is in a standby mode and the dew point temperature of the housing surface drops below the dew point temperature of the air inside the duct. The SCHEPA filter does not have a drain.

Housing Connections and Flexible Connections Section 5, paragraph 5.6.2, (e), "Housing Connections" and (f), "Flexible Connections," require that the filter housing connections be flanged and that the flexible connections be qualified for stress and pressure.

The connection between the self-contained filter housings and ducts are made of a light-weight fabric secured to the ductwork by hose-type clamps. This type of connection may not ensure structural stability and must be tested to meet the structural load requirements of ASME N509, Section 4.5.

A standard filter housing is mounted to the duct system by rigid metal flanged connections. The SCHEPA filter is not equipped with flanged connections.

Component Mounting Frame Section 5, paragraph 5.6.3, requires that the HEPA filter be secured within the housing to a mounting frame to ensure rigidity of the filter. One of the major differences between a SCHEPA and a standard 24-in. by 24-in. by 11.5-in. HEPA filter is the mounting arrangement of the HEPA filter.

Standard HEPA filters are bolted directly to a mounting frame or compressed against the mounting frame by a clamping mechanism; the gasket is compressed with a clamping force of at least 20 lb/in^2 of gasket surface or a minimum of 1,400 lb on the HEPA filter frame. The frame must meet the structural requirements of ERDA 76-12, paragraph 4.3, to prevent flexure between the frame and the HEPA filter.

The mounting frame is seal welded to the filter housing to prevent contaminants from bypassing the filter between the housing and mounting frame. The SCHEPA filter medium is attached directly to the casing by an adhesive used as a bonding agent. The SCHEPA filter does not have a mounting frame and does not meet these requirements.

Shop Leak Testing Section 5, paragraph 5.6.5.4, states that "Housings or housing sections shall be leak tested in the shop prior to shipment, in accordance with ASME N510[4], Section 6. Leakage shall be no greater than

acceptance criteria provided by the owner. Results of housing leak tests shall be transmitted to the owner for his records." The SCHEPA filter does not have documentation showing that this test has been performed.

Shop Airflow Distribution Test Section 5, paragraph 5.6.5.5, requires that the manufacturer perform an airflow distribution test in the shop in accordance with ASME N510[4], Section 8, before shipment. This test ensures that the manufacturer's design provides uniform air distribution. Again, the SCHEPA filter does not have documentation showing that this test has been performed.

Quality Assurance Required Documentation Section 8, paragraph 8.2 requires that as a minimum the following shall be documented and submitted to the owner:

- Maximum operating pressure
- Test pressure
- Structural capability pressure
- Factory housing leak-test report
- Factory airflow distribution report.

None of the above items have been documented for the SCHEPA filter.

Noncompliance Pertaining to ASME N510, *Testing of Nuclear Air-Treatment Systems*

Housing and Ducts Section 5, paragraph 5.5.1.1, (v), states that the system may not contain "sealant or caulking of any type on/in housings or component frames. Caulking on/in ducts may be permissible depending on projects specifications." The SCHEPA filter uses sealant on the end caps that connect the nipples to the filter housing.

Duct and Housing Leak and Structural Capability Tests Section 6 requires that both duct and housing be pressure tested either by the constant-pressure test (positive pressure) of paragraph 6.5.2 or by the pressure-decay method (negative pressure) of paragraph 6.5.3. These tests help ensure that the housing or duct will not release contaminants past the pressure boundary as a result of pressurization.

Standard filter housings are subjected to these tests with the test data documented. The SCHEPA filter casing functions as a housing in an air-cleaning system, but does not have the required documentation of test results.

Airflow Distribution, Air-Aerosol Mixing Uniformity, and In-Place Testing Section 8, paragraph 8.5.2, "Airflow Distribution Test," Section 9, "Air-Aerosol Mixing Uniformity Test," and Section 10, "HEPA Bank In-Place Test," state that the airflow distribution test is a prerequisite for the air-aerosol mixing test and the in-place HEPA filter test.

These tests ensure that a uniform distribution of the test aerosol is received on the entire face area of the HEPA filter bank. Without a uniform distribution of the test aerosol, there can be no assurance that, if a penetration exists in the filter medium, the penetration would be detected during the in-place testing.

The SCHEPA filter, as described in Section 1.2, does not have a smooth transition from ductwork to the filter to allow airflow passing through the filter medium uniformly. For example, the 1,000 ft^3/min SCHEPA filter has a 12-in.-diameter duct inlet. The SCHEPA has no transition from the 12-in-diameter circular cross section to the 24-in. by 24-in. by 11.5-in. filter.

The change in the size of the cross section at the duct inlet of the SCHEPA is abrupt and results in the lack of the nonuniformity in the airflow distribution at the face of the filter medium. Hence, during the filter in-place test, the entire filter face cannot be properly challenged with aerosol.

Further, the layout of the SCHEPA filter does not meet the criteria for even airflow distribution of ASME N509[3], and uniformity in airflow distribution at the face of the filter medium cannot be ensured. The air distribution must be tested to verify the design.

The SCHEPA filter is tested in the factory with the end caps removed to allow an even air/aerosol mixture. The manufacturer does not test the filter with the end caps installed. Therefore, no test data exist to illustrate the aerosol challenge of the filter face after final assembly as the filter is actually installed in the system.

The filter is handled many times before installation and is subject to damage. If the filter is damaged before installation, the adhesive that holds the filter medium in place may be separated from the housing. If in-place aerosol testing does not detect the damaged area, a release of radioactive contaminants could occur. When the entire filter face cannot be properly challenged with aerosol, the result is a false sense of filter integrity.

Noncompliance Pertaining to Military Specification MIL-F-51068

Conformance to Military Specification MIL-F-51068[6], *Filters, Particulate (High-Efficiency Fire Resistance)* is required by ASME N509[3], which states that the HEPA filter must meet the construction and testing of MIL-F-51068[6]. The SCHEPA filter does not comply with the following sections of MIL-F-51068[6].

Flatness and Squareness Section 3, paragraph 3.3.2, requires verification that the filter is flat and square when measured diagonally across the corners of both faces. There are no inspection records for the SCHEPA filter.

Resistance to Rough Handling Section 3, paragraph 3.4.3, requires that the filter pass the rough-handling test. This SCHEPA does not have documentation to show that it has passed such a test.

Noncompliance Pertaining to the National Fire Protection Association

This requirement applies only to wooden SCHEPA filters. The DOE Order 6430.1A[1], *General Design Criteria*, Section 1550-2.5.6, states that "Ductwork shall also be designed to comply with NFPA 90A[7]." Exhaust ductwork shall comply with NFPA 91[8].

Noncompliance Pertaining to NFPA 90A, *Installation of Air Conditioning and Ventilation Systems* Section 2.3.1.1 states that ducts shall be constructed from iron, steel, aluminum, copper, concrete, masonry, or clay tile. Some SCHEPA filters are constructed of wood.

Noncompliance Pertaining to NFPA 91, *Blower and Exhaust Systems for Stock and Vapor Removal or Conveying* Section 2.7 also states that the exhaust system shall be constructed entirely of sheet metal or other noncombustible material. The wooden SCHEPA filter cannot meet requirements of either of these sections.

In a fire, the wooden casing would cause the fire to escape to the surrounding area and spread. Therefore, the wooden SCHEPA filter should not be used where the possibility of fire exists.

Noncompliance Pertaining to ERDA 76-21, *Nuclear Air Cleaning Handbook*

The DOE Order 6430.1A[1], *General Design Criteria*, states that "In-place testing design requirements shall meet all the recommendations of UL 586[10], ASME N510[4], and ERDA 76-21[5]." The *Nuclear Air Cleaning Handbook*[5] recommends not using the SCHEPA where high levels of radioactivity are suspected.

Other Code-Related Issues

Visual Inspection ASME N510[4] requires visual inspection of the system, including HEPA filter and ductwork in conjunction with each test series. Direct visual inspection of the SCHEPA is not possible because of the obstruction of the caps and connectors. Because visual inspection must rely on special instruments/tools it becomes impractical; therefore, inspection may not occur.

Airflow Distribution Test for Single Filters As discussed in Section 7 above, the design requires that the layout of the housing shall allow uniform airflow within plus and minus 20% of the average through each bank of components.

Although the ASME N510[4] noted that airflow distribution tests are not required for a filter bank containing a single filter, it does not exempt the design requirement given in ASME N509[3].

Because the SCHEPA is not designed and constructed in accordance with ASME N509[3], the requirement for uniformity in airflow distribution at the filter cannot be assured.

VII. Conclusion

Commercially available SCHEPA filter(s) do not conform to the technical requirements nor have the documentation required to verify conformance to the mandatory codes and standards. The evaluation classified a SCHEPA filter as an "air-cleaning unit," in accordance with the definition given in ASME N509[3]. The SCHEPA filters address only the filter medium and casing material aspects and do not stipulate applicable ASME N509[3] air-cleaning unit requirements, such as airflow distribution, structural capability, pressure boundary integrity, seismic capability, and the rough-handling capability test of MIL-F-51068[6].

VIII. Recommendations

The SCHEPA filter should not be used for nuclear applications. When new designs are contemplated, other types of HEPA filter housings can be used in lieu of the SCHEPA filter.

IX. References

1. DOE Order 6430.1A, *General Design Criteria*, U.S. Department of Energy, Washington, D.C.

2. DOE Order 4300.1B, Real Property Management, U.S. Department of Energy, Washington, D.C.

3. ASME N509-1989, *Nuclear Power Plant Air Cleaning Units and Components* American Society of Mechanical Engineers, New York, New York.

4. ASME N510-1989, *Testing of Nuclear Air-Treatment Systems*, American Society of Mechanical Engineers, New York, New York.

5. ERDA 76-21, *Nuclear Air-Cleaning Handbook*, Energy Research and Development Administration, Washington, D.C.

6. MIL-F-51068F, *Military Specifications Filters, Particulate (High-Efficiency Fire Resistance)*, Aberdeen Proving Grounds, Maryland.

7. NFPA 90A, *Air Conditioning and Ventilation Systems*, National Fire Protection Association, Batterymarch Park, Quincy, Massachusetts.

8. NFPA 91, *Blower and Exhaust Systems*, National Fire Protection Association, Batterymarch Park, Quincy, Massachusetts.

9. RLIP 5480.4C, *Environmental Protection Safety and Health Practices Standards*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

10. UL 586, *High-Efficiency Particulate, Air Filter Units*, Underwriters Laboratories, Northbrook, Illinois.

DISCUSSION

TODD: To my knowledge there's no such thing as an off-the-shelf, self-contained HEPA (SCHEPA) filter. I can't speak for my competitors, but in our case the filter would only be built to the customer's specifications. And that brings to mind a question with regard to the reference to MIL-F-51068 for this filter. There are a number of DOD and DOE sites that specifically refer to the self-contained HEPA filter as a part of their individual site specifications. They refer to 51068 as the mother specification and then give size details and a description of the frame; but self-contained filters are not a part of it. This has been a matter of discussion for years. The filter has the components of a 51068 filter, and it is tested like a 51068 filter, but when you add face plates and nipple adapters, the filter is off specification. You are right, the matter does need to be resolved.

ARNDT: Off-the-shelf refers to a specific HEPA filter, or a SCHEPA filter. It has also been called an encapsulated filter or nipple-connected filter. There was a task team that evaluated this, and there was a specific self-contained filter that was addressed. It is in a catalog, and lists the data we used. We went to the vendor, but they did not have any documentation showing that the total assembly met the requirements of the rough handling test for MIL-F-51068.

KOVACH, L.: You were basing your evaluation on the DOE order requiring compliance with ASME N-509 and N-510. Don't you think that common sense HVAC engineering criteria were violated, and that if you were designing an air cleaning system without adding to the DOE order or N-509 requirements, just common sense engineering practice, it would be different from the SCHEPA design? Have you prepared an engineering evaluation, also?

ARNDT: The task we had was to evaluate the filter per the DOE order, and take all the emotion out of it. In other words, we could not put into the report that these filters were used for many years and there is no record of failure. What we had to do was to look at them in light of the DOE order and the documentation we are required to prepare prior to our designs. These filters are used on various applications. Some are used in ESF types of systems and we had to have the required documentation for our QA and for our safety records.

KOVACH, L.: Isn't it possible that the way they are built and the way that you test them in-place, you would not even find out if they leak?

ARNDT: When preliminary tests were performed on the SCHEPA filter, the vacuum pump could not keep up with the leakage. In addition, a rough handling test revealed several loosened bolts which connect the end cap to the casing.

WRIGHT: Considering that the standard is in need of updating, would it be worthwhile to seek an exemption to those specific parts of the order which don't make sense for this application? I can foresee within the environmental remediation effort where having a self-contained filter that you can just plug into an air enclosure would make a whole lot more sense than a permanent installation.

ARNDT: In a report we put out, we recommended that every facility look at their applications and, when technically justified, request a waiver from DOE. Some of these filters are used in systems handling highly radioactive contaminants, so a safety analysis must be performed prior to requesting a waiver from DOE. That is definitely an option. There was an earlier question about the difference between the safety basis and the regulatory basis. Within the DOE community this is a case where they are not necessarily consistent.

LAWTON: First, you can specify a SCHEPA filter to meet most of your specifications but it didn't sound like the one you tested did. Second, DOE order 6431A allows for good engineering practice. Third, if a contamination control specialist uses a SCHEPA filter properly, it can save a lot of work and problems elsewhere. And there are work-arounds. Is the SCHEPA filter really an engineered safety feature, or is it just a prefilter before

the effluent stream gets to the building plenum filters, the real engineering safety feature?

ARNDT: It depends entirely on the system function. Using it as a prefilter and not taking credit for it, is one application. But if you are taking credit for it, you have to perform aerosol testing, and a prerequisite is uniform air flow distribution. As I mentioned in the paper, a single filter is not required to meet air flow distribution requirements prior to in-place leak testing but you have to remember we address the enclosure as a filter housing according to N-509. N-509 does not exempt a single filter from uniform air flow. When there is technical justification, it is up to the project engineer, or whoever is responsible for project specifications, to request an exemption if they want to use a SCHEPA filter.

DICK: I question the appropriateness of comparing and basing the adequacy of the self-contained HEPA filter to a standard (ASME-N509) that was not established or considered for SCHEPA filter applications. Many filtration installations made prior to ASME-N509 cannot meet all the design requirements in ASME-N509 whether they are SCHEPA systems or not. However, installations are not inadequate simply because the existing filtration system does not have all the options that a N509 system is required to have.

ARNDT: As stated in the paper, it is recognized that the ASME's codes were not written with the SCHEPA filter in mind. However, if the SCHEPA filter is used in a nuclear application, then it is fair to use the requirements of nuclear filtration to analyze the SCHEPA filter. One of the main goals for nuclear filtration is protection to personnel, equipment, and the environment. Clearly, the SCHEPA filters do not have documentation to support their continued use. However, it is the responsibility of the individual user to make that decision.

BERGMAN: We are concerned that the box (SCHEPA) filters are being used for nuclear applications in several DOE facilities and yet do not meet all of the applicable requirements in ASME N509 and MIL-F-51068. We investigated the problem and discovered that documentation verifying the qualification tests either was not explicitly requested in purchase orders or was not enforced. In other cases, the supplied documentation was for the open-face HEPA filter made with the same materials as the box filter. Discussions with HEPA filter manufacturers confirmed that the box HEPA filters had never been qualified on the heated air, overpressure and rough handling tests. Further investigation also revealed that none of the filter qualification laboratories (Underwriters' Laboratories, the U. S. Army Product Assurance Directorate at Edgewood, MD, and the DOE Rocky Flats Filter Test Station) could even test the box HEPA filters. In addition to lacking the necessary adapters, the test equipment would require an expensive upgrade and relocation to provide additional space.

After the DOE/NRC Nuclear Air Cleaning Conference, we quickly initiated a test program at EG&G Rocky Flats Plant to conduct qualification tests on six box HEPA filters typical of those used at LLNL. Since the test facility at Rocky Flats did not have sufficient space to accommodate the box HEPA filters, we compromised on the design by sealing the test duct to the box frame rather than using inlet and exhaust adapters. The test results for the six, 1,000 cfm box HEPA filters from American Air Filter for the

heated air, overpressure, and rough handling tests are given below. We have also tabulated the penetration values allowed by ASME N509 and MIL-F-51068.

Heated Air Test

Serial No.	Percent Penetration at 1,000 cfm	
	Before	After
455566	0.009	0.022
41459567	0.012	0.042
Standard	0.030	3.000

Overpressure Test

Serial No.	Percent Penetration at 200 cfm	
	Before	After
459549	0.012	0.012
41459608	0.012	0.016
459549*	0.010	0.011
Standard	0.030	0.030

* Flanges were cut off the HEPA filter to reach 10 inch pressure drop.

Rough Handling Test

Serial No.	Percent Penetration at 200 cfm	
	Before	After
4159600	0.012	0.014
459560	0.018	0.020
Standard	0.030	0.030

In addition to the problem with adapting the box HEPA filters to the test equipment, we encountered two other problems. The blower on the overpressure test apparatus was not able to generate the required 10 inches of pressure across the filter medium pack because of the added resistance from the reduced entry and exit ports. The filters 459549 and 41459608 had pressure drops of 7.1 and 8.0 inches respectively. We cut off the entry and exit ports on the 459549 filter and repeated the test at the prescribed 10 inch pressure drop. The second problem was the periodic shut down of the rough handling machine due to the high weight of the box filters. The equipment will need extensive modifications for routine tests on the box HEPA filters.

Based on the tabulated test results, the box HEPA filters from American Air Filter passed the required qualification tests. The filters had a plywood frame, deep-pleated medium, aluminum separators, and neoprene sealant. Although box filters from other manufacturers and box filters from American Air Filter made using different materials

cannot be automatically qualified based on these tests, they also are expected to pass the tests. This follows because the only difference between a box filter and a standard open face filter is the restricted inlet and exit ports. Since the box HEPA filters from American Air Filter passed the qualification tests, any box HEPA filter made from the same materials and the same manufacturer as an open faced HEPA filter that passed the qualification test is also expected to pass.

ARNDT: A few questions come to mind regarding the testing of the SCHEPA filter for rough-handling, over-pressurization, and the heated air. Did the testing include a hole in the filter media of a known area and location to verify that the testing would detect a flaw in the filter medium during an in-place aerosol test? Due to the configuration and the uneven airflow distribution of the SCHEPA I would suggest, if possible, the filter be tested to failure, *i.e.*, that there be a series of holes of differing diameters in various locations to determine where the filter fails and passes. Also, are there any plans to pressure test the filter housing?

BERGMAN: The qualification tests on heated air, overpressure, and rough handling did not include a hole in the filter media. The qualification tests do not require that, and we see no purpose for artificially introducing such holes. The suggestion to introduce the holes was apparently made to investigate the air-aerosol mixing uniformity in the box HEPA filter. This is neither required by ASME N510, nor does it add to the accuracy of in-place leak tests. Since all box filters can only be used in single filter installations, there are no compelling reasons for doing the uniform air flow and aerosol mixing tests. ASME N510 exempts all single filter installations from such tests. In practice, the aerosols would be measured at some distance upstream and downstream of the entry and exit ducts, not inside the box. At these points, the aerosols would be well mixed. Regarding the final question, we plan to measure the leak rate through the box HEPA filter.

ASME N510 TEST RESULTS FOR SAVANNAH RIVER SITE AACS FILTER COMPARTMENTS

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Abstract

The K-Reactor at the Savannah River Site recently implemented design improvements for the Airborne Activity Confinement System (AACS) by procuring, installing, and testing new Air Cleaning Units, or filter compartments, to ASME AG-1, N509, and N510 requirements. Specifically, these new units provide documentable seismic resistance to a Design Basis Accident earthquake, provide 2" adsorber beds with 0.25 second residence time, and meet all AG-1, N509, and N510 requirements for testability and maintainability.

This paper presents the results of the Site acceptance testing and discusses an issue associated with sample manifold qualification testing.

I. Introduction

The Savannah River Site (SRS) is located on the Savannah River in South Carolina. The Site is owned by the Department Of Energy, with the Westinghouse Savannah River Company (WSRC), a Subsidiary of Westinghouse Electric Corp., as the primary contractor. The mission of the Savannah River Site was to produce special nuclear materials. Currently, 4 of the 5 production reactors at SRS have been shutdown and the fifth reactor, K-Reactor, is being maintained in a cold standby condition. The reactor never operated with these new filter compartments installed.

The production reactors at the Savannah River Site (SRS) were constructed in the 1950's. These reactors are non-boiling and vented to atmosphere (non-pressurized). The Airborne Activity Confinement System (AACS) for the reactors is designed to collect airborne radioactive particles and halogen vapors that might be released to the atmosphere following a reactor accident. The system is designed for confinement, not containment. Confinement is attained by sweeping outside air through the process areas and exhausting the air through parallel filtration-adsorption units (filter compartments) prior to exhausting to the atmosphere. Five compartments are available with three constantly on-line during reactor operation. Contamination within the reactor building is controlled by flowing air from areas of least expected contamination to areas with a higher potential for airborne contamination. The process areas of the reactor building are sufficiently sealed that they can be maintained at a negative pressure with respect to adjoining areas, including the atmosphere.

Originally, the process area exhaust air was exhausted directly to the atmosphere through a 200' stack. In the 1960's, design changes were implemented to exhaust air from the process areas through the 5 parallel filter compartments prior to exhausting to the atmosphere. Each unit contained a bank of moisture separators, a bank of HEPA filters, and an activated charcoal absorber bed. Typical building total flow rate during reactor operation with the filter compartments installed was 110,000 cfm.

In 1990, a new project was initiated to replace the existing filter compartments with new seismically qualified compartments designed, installed, and tested in accordance with ASME AG-1, N509, and N510 requirements. The new design also incorporated other changes to improve maintainability, operability, and to provide for longer periods of operation between filter replacement (see Figure 1).

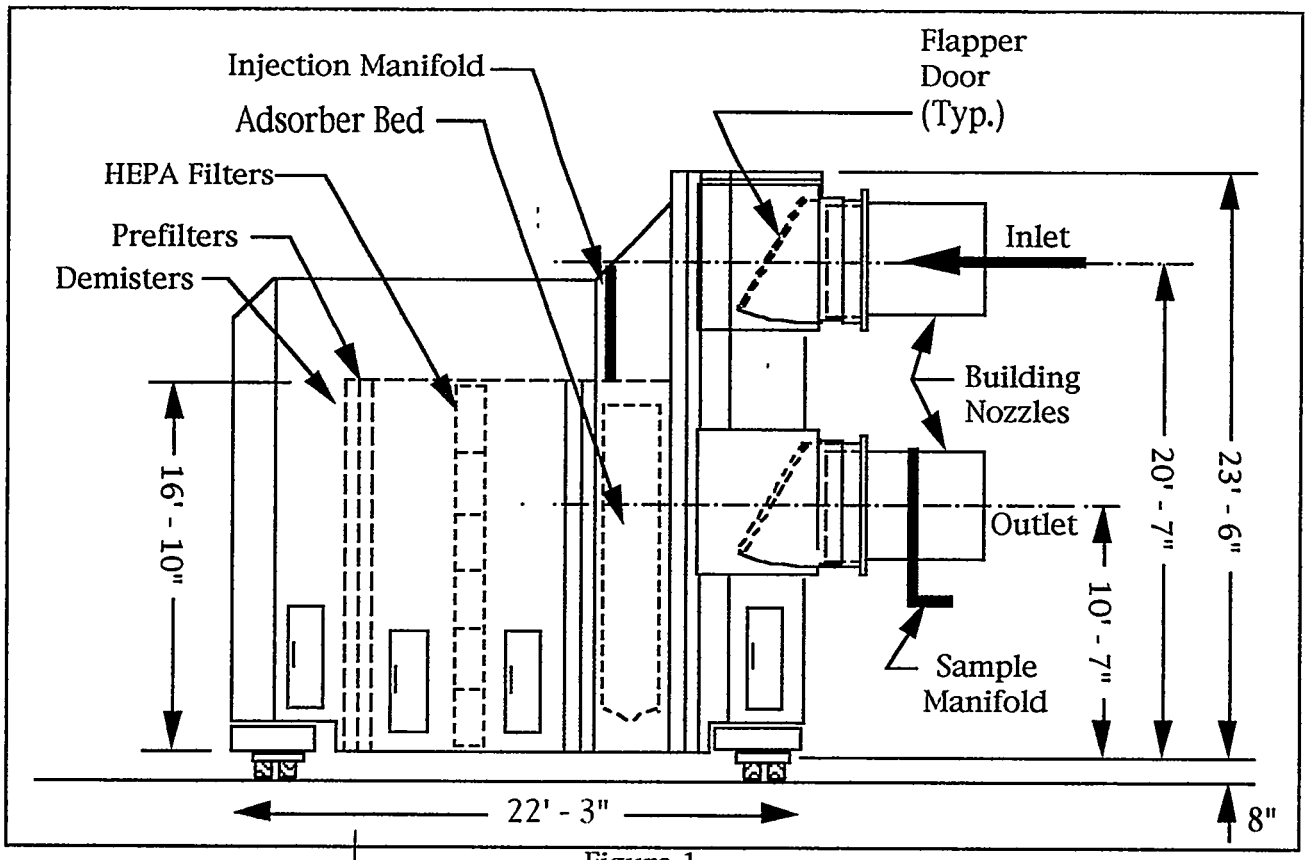


Figure 1
Elevation View - New Filter Compartment

The compartments reside on the roof of the building at an elevation 55' above the ground. Because the original AACS design did not include the filter compartments, a uniquely designed compartment was required to retrofit the existing system. Process area exhaust air enters the top of the compartment and makes a 180 degree turn prior to entering the moisture separators. The compartments are locked in place and sealed to the ventilation system using inflatable bladders on the building nozzles.

Neither the new or the old compartments were shielded. Therefore, both were designed for "remote removability" such that if a compartment became highly contaminated, it can be remotely disconnected from the ventilation system, sealed to prevent spread of contamination, and removed by a crane for burial.

The new compartments were constructed of stainless steel to resist corrosion in the acidic process air. Weight constraints were imposed to prevent degrading the building structure's ability to withstand a design basis earthquake (the existing compartments were made of aluminum). The design flow rate through the new compartments ranges from 16,000 to 32,000 cfm each.

II. ASME N510 Acceptance Testing

N510 qualification testing to prove AG-1 and N509 requirements were met was conducted at both an independent laboratory and at the K-Reactor. The test results reported herein were obtained during the Site acceptance testing of Filter Compartment N° 2 at the K-Reactor, unless otherwise noted.

Structural Capability Test

The filter housing was subjected to both a negative and positive structural capability test. The negative pressure test was performed at -9.25 inches of water gauge (w.g.) and the positive pressure test was performed at 69.5" w.g. Inspection of the filter compartment during pressurization and following test completion showed no evidence of permanent distortion or breach. Therefore the filter compartment met the acceptance as required by ASME N510.

Housing Leak Test

The filter housing was subjected to leak test to demonstrate its leakage integrity under maximum design pressure. The housing was pressurized to 55.9" w.g. Acceptance criteria was 1.0" w.g. decay over a ten minute period. Actual test result, after correction for temperature and barometric pressure changes, showed 0.99" w.g. decay during the ten minute test period.

HEPA And Carbon Adsorber Frame Leak Test

WSRC elected to perform the optional HEPA filter and carbon adsorber frame leak tests. Acceptance criteria per N509/510 is 0.1% of rated flow.

The HEPA frame, after sealing with plastic sheeting, was subjected to +3.0" ± 0.1" w.g. by pressurizing the housing upstream of the HEPA bank with air. Leakage was measured at the air flow required to maintain pressure. Actual average flow rate measured over a ten minute period was 4.7 cfm or 0.029% of the minimum design air flow.

The adsorber bed, after sealing with plastic sheeting, was subjected to +1.1 ± 0.1" w.g. by pressurizing the housing upstream of the adsorber bank with air. Leakage was measured as the air flow required to maintain pressure. Actual average flow rate measured over a ten minute period was 8.5 cfm or 0.053% of the minimum design air flow.

This test methodology produced a conservative result by assuming the measured leak rate was completely due to HEPA frame leakage and ignoring the other possible sources of leakage (personnel door seals, light fixtures, inadequate sealing of the plastic sheeting, etc.). Actual frame leakage is most probably less than the measured value. Both tests demonstrated that the sealing of the frames to the housing is adequate and meets all requirements.

Air Capacity Test

An Air Capacity Test was performed to demonstrate that at the maximum housing delta-P of 6" w.g., the process areas remained at a negative pressure with respect to atmosphere and that a minimum flow of 20,000 cfm can be maintained. Actual testing demonstrated that both criteria were met.

Air Flow Distribution Test

An Air Flow Distribution Test for both the HEPA filters and the Adsorber Bed air channels was performed. Both tests were conducted at the maximum design air flow of 32,000 cfm ± 10%.

HEPA Bank Once the air flow was established, the air velocity was measured at the downstream side of each HEPA filter (filter bank designed as 3 filters wide X 7 filters high). Actual measured velocities, shown in Table 1, demonstrated that the air flow distribution met the N510 criteria of ±20% of the average of the individual flows.

Table 1
Actual HEPA Air Flow (fpm)

	A	B	C
7	686	578	572
6	596	595	620
5	557	569	570
4	589	597	601
3	537	575	489
2	611	631	599
1	570	563	501

(Viewed From Upstream)

Average Air Flow = 581 fpm
Allowable Range: Min. = 465 fpm (-20%)
Max. = 698 fpm (+20%)
Measured Range: Min. = 489 fpm (-16%)
Max. = 686 fpm (+18%)

Table 2
Actual Adsorber Air Channel Air Flow (fpm)

	A	B	C	D	E	F	G	H
14	1870	1944	1842	1902	1929	2011	1895	1781
13	1924	1852	1939	1835	1871	1881	1730	1703
12	1771	1626	1541	1856	1823	1908	1566	1385
11	1514	1667	1831	1963	1859	1709	1646	1796
10	1601	1434	1850	1727	1834	1529	1575	1681
9	1961	1858	1833	1640	1612	1563	1523	1896
8	1872	1691	1785	1604	1671	1572	1535	1845
7	1798	1748	1674	1617	1607	1635	1552	1673
6	1597	1854	1793	1821	1680	1599	1645	1722
5	1898	1762	1660	1626	1684	1487	1580	1688
4	1756	1759	1664	1540	1624	1464	1408	1813
3	1785	1845	1701	1710	1657	1410	1429	1783
2	1872	1813	1909	1804	1786	1519	1445	1841
1	2007	1837	2008	1786	1775	1574	1417	1899

(Viewed From Upstream)

Average Air Flow = 1724 fpm
Allowable Range: Min. = 1379 fpm (-20%)
Max. = 2069 fpm (+20%)
Measured Range: Min. = 1385 fpm (-19.7%)
Max. = 2011 fpm (+16.6%)

Adsorber Bank Once the air flow was established, the air velocity was measured at fourteen approximately equal areas in the plane of the face of the eight inlet air channels. Actual measured velocities, shown in Table 2, demonstrated that the air flow distribution met the N510 criteria of ±20% of the average of the individual flows.

Air-Aerosol Mixing Uniformity Test

DOP was introduced into the housing at the maximum design air flow rate (± 10%) using the permanently installed injection manifold. The resultant DOP concentration measured one foot upstream of the HEPA filter bank (see Table 3) showed that the air-aerosol distribution at each of the 21 HEPA filters was within ±20% of the calculated average concentration.

It should be noted that even though the housing design requires the air to make a 180° turn prior to entering the moisture separators, both the air flow distribution and air-aerosol uniformity were achieved without the use of turning vanes or other mixing devices.

Table 3
Air-Aerosol Mixing - DOP Concentration

	A	B	C
7	86	85	86
6	84	85	83
5	78	70	75
4	78	66	76
3	75	62	75
2	72	62	70
1	68	68	66

(Viewed From Upstream)

Detector Scale Setting = 10%
Initial DOP Concentration = 63
Average DOP Concentration = 74.8
Allowable Range: Min. = 59.8 (-20%)
Max. = 89.8 (+20%)
Measured Range: Min. = 62 (-17%)
Max. = 86 (+15%)

HEPA Filter Bank In-Place Leak Test

Aerosol generators and detectors were connected to the housing for the performance of the HEPA filter bank in-place leak test. System dampers were properly positioned and the maximum design air flow rate (± 10%) was established. Upstream and downstream HEPA filter background readings were taken and recorded. Upstream concentration was measured using the single point sample port. This port was proven effective in measuring upstream concentrations during the air-aerosol mixing uniformity test. Downstream concentrations were measured using the permanently installed sample manifold. The aerosol generators were then pressurized, and DOP was injected into the inlet plenum manifold. Four sets of upstream and downstream HEPA filter DOP concentrations were taken and recorded. The final penetration calculated from the data demonstrated that the actual bypass leakage was virtually 0%.

Adsorber Bank In-Place Leak Test

Halide generators and one detector were connected to the filter housing for the performance of the adsorber in-place leak test. System dampers were properly positioned and the maximum design air flow (± 10%) was established. Upstream and downstream adsorber background readings were taken to verify stable background conditions for initiation of injection. The halide generators were turned on and halide gas (R-11) was injected into the inlet plenum manifold. Five sets of upstream and downstream readings were taken and recorded. The downstream concentration was measured using the permanently installed sample manifold. The resultant penetration bypass leakage extrapolated to time 0 from the data obtained was less than 0.0016%. However, WSRC questioned the acceptability of the sample manifold to detect bypass of the adsorber bed, even though the manifold satisfactorily passed the manifold qualification test as described in Mandatory Appendix D of N509, Performance Test For Qualification Of Sampling Manifolds. Refer to the following section on Manifold Testing for the basis of WSRC's concerns.

III. Manifold Qualification Testing

Manifold Testing

Testing to demonstrate that the proposed sampling manifold (see Figure 2 & 3) is adequate for sampling during both HEPA filter and adsorber bank bypass testing was conducted at an independent laboratory. Artificial leak sites were introduced in the form

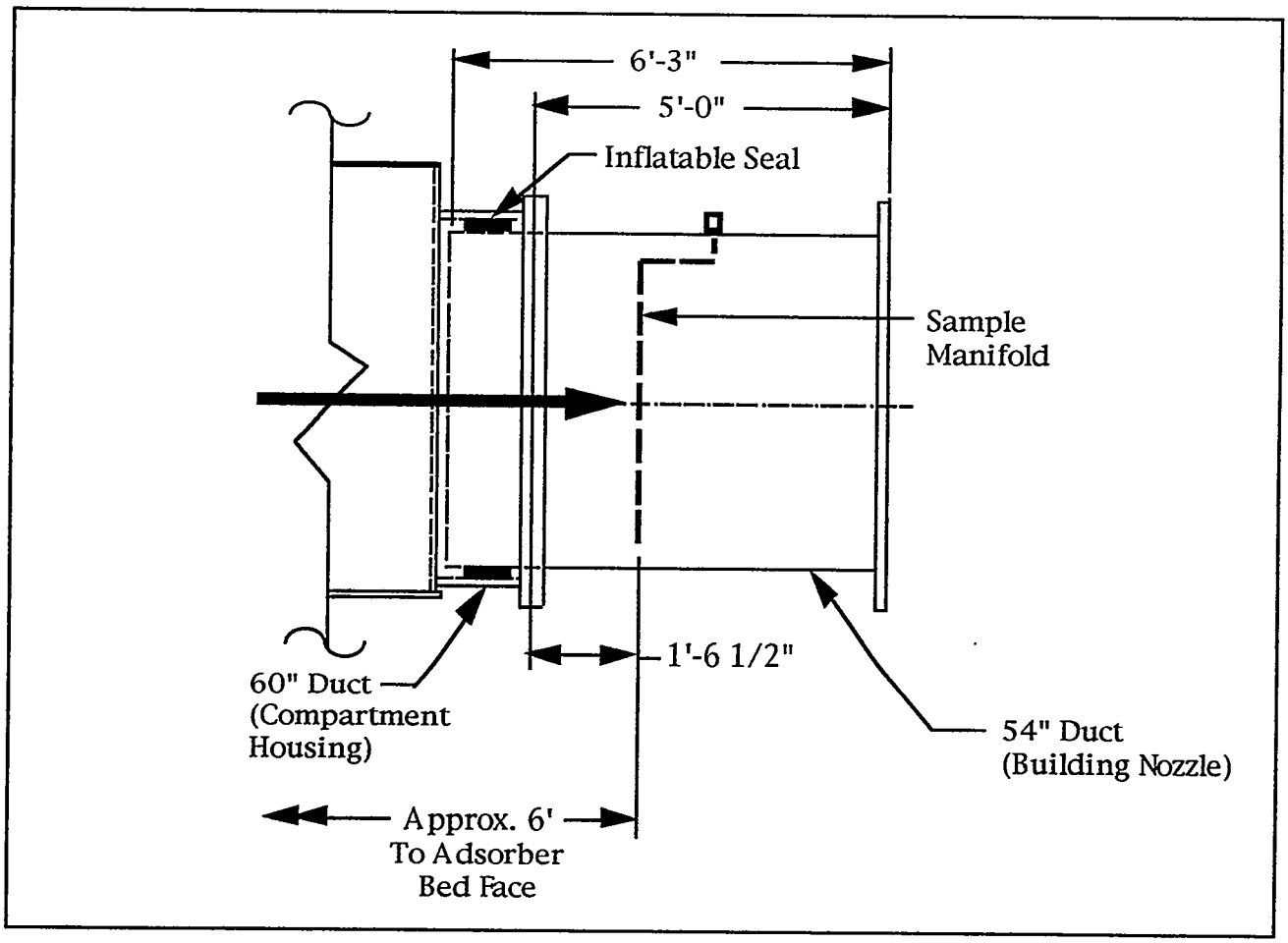


Figure 2
Sample Manifold Location

of holes in the HEPA frames, tears in the filter medium, and tears in the sealing face gaskets. No attempt was made to introduce artificial leak sites in the adsorber bed or frame.

WSRC questioned the completeness of the testing to show that the manifold is capable of detecting adsorber bypass. WSRC's interpretation of N509 Appendix D, paragraph D4.2(b)(1), which states "the artificial leak paths shall be located, one at a time, to simulate leaks in the filter/adsorbent face, frame-to-wall welds (including floor and ceiling), and gasket-to-frame seals (where applicable), and at structural welds on Type III adsorbers," is that both the HEPA bank and adsorber bank should have artificial leak sites introduced and the sample manifold tested for adequacy. The vendor responsible for the testing interpreted this paragraph as an either/or statement and that testing only the HEPA bank satisfied Appendix D requirements.

WSRC issued a formal inquiry, Inquiry 93-03, to the CONAGT committee to provide the correct code interpretation. The inquiry response is pending.

For any downstream sample manifold to work properly, it is essential that the air stream be well mixed prior to entering the sampler and that adequate bulk air mixing between the adsorber bed and sampler is achieved. WSRC hypothesized that adequate mixing downstream of the adsorber bed may not be achieved and that the Appendix D methodology is inadequate to demonstrate manifold acceptability. WSRC believes that the mixing experienced by the DOP may not be representative of the mixing that occurs downstream of the adsorber bed.

To prove the hypothesis, prior to receiving the response to the CONAGT inquiry, WSRC developed and initiated scoping tests to simulate point source leaks downstream of the adsorber bank to determine if preferential sampling was occurring at the sample manifold. DOP was chosen as the challenge gas.

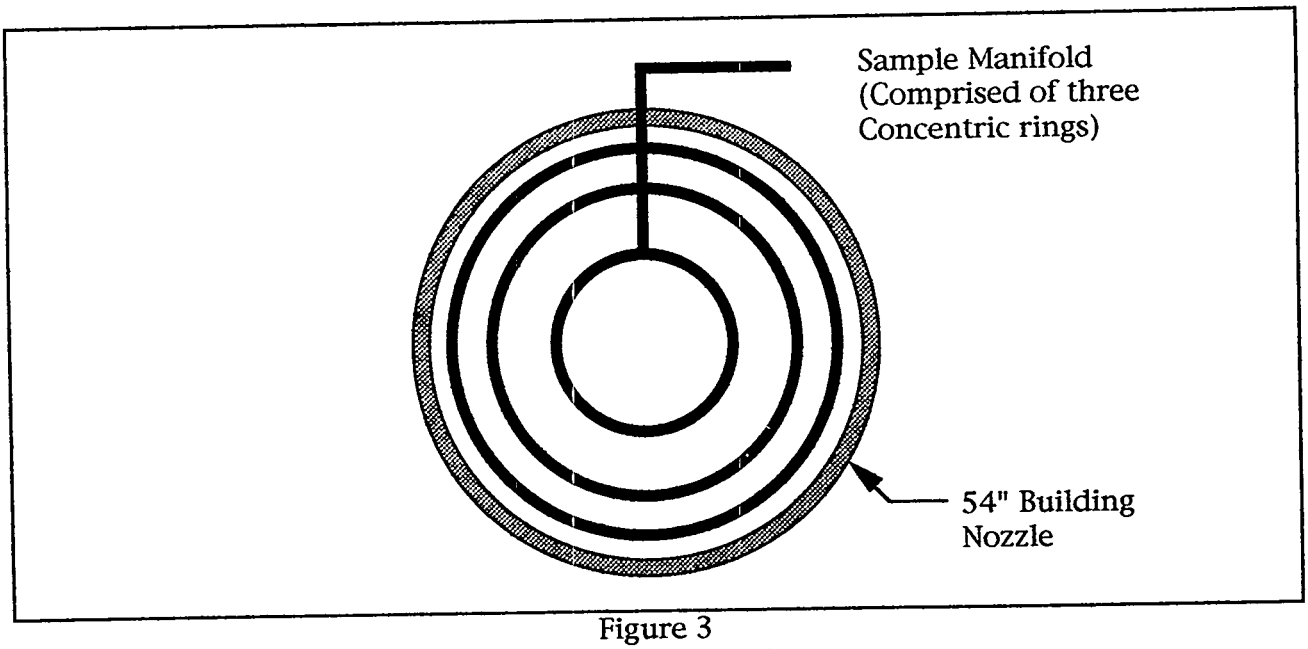


Figure 3
Sample Manifold

DOP was injected at 12 discrete points located on the downstream face of the adsorber bed (see Figure 4 & 5) by routing tubing from the aerosol generator through the housing wall and attaching to the adsorber bank face. Tubing routing was carefully chosen to prevent obstructing air flow. Downstream concentrations were measured at the sample manifold and at a single point downstream of the main exhaust fan. Testing showed the fan adequately mixed the air stream and the single point sample location produced a representative sample of the air stream.

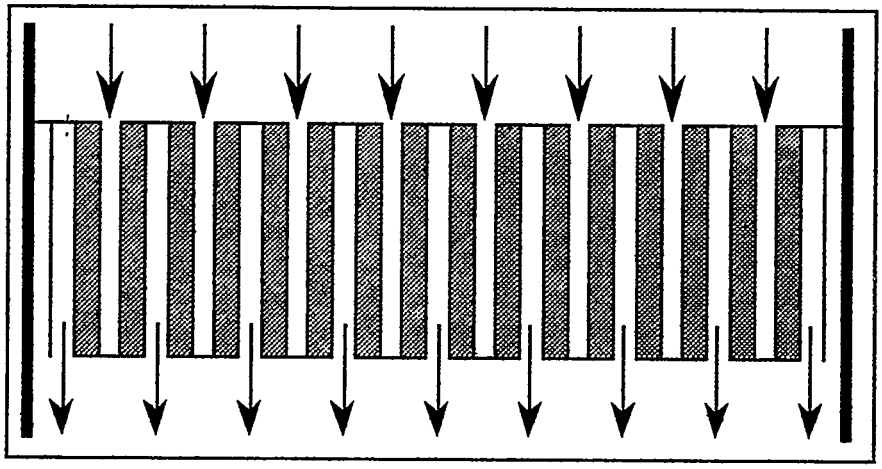
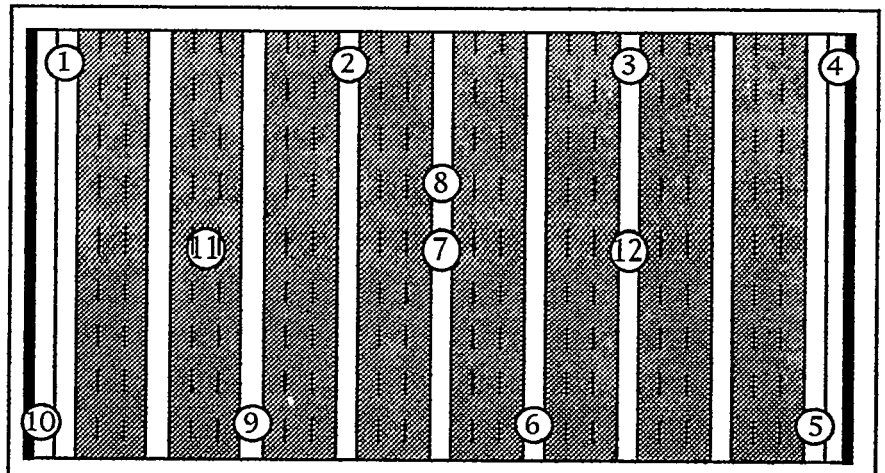


Figure 4
Adsorber Bed - Plan View



Viewed From Downstream

(1)(2)(3) - Located @ top of air channel
(4) - Located @ top of stagnant air gap between housing wall and adsorber bed
(5)(6)(9) - Located @ bottom of air channel
(7)(12) - Located @ middle of air channel
(8) - Located in-line with center of sample manifold
(11) - Located in middle of bed @ a seal weld
(10) - Located @ bottom of stagnant air gap between housing wall and adsorber bed

Figure 5
Adsorber Bed - Elevation View
Artificial Leak Site Locations

Since two different calibrated DOP detectors were used at the two locations, correlation tests between the two detectors were conducted at the sample manifold prior to the test and at the single point location following the test. While this test is not required per N509, it was felt that this data was essential for proving the hypothesis. Good agreement between the two detectors was demonstrated, as shown in Table 4.

Comparison between the concentrations measured at the sample manifold and the single point sample locations are shown in Table 5. These results suggest that testing of

Table 4
DOP Detector Comparison

Test Point N°	Post-Test Comparison (Sample Manifold)			Post-Test Comparison (Single Point Location)		
	Det. #1 Conc.	Det. #2 Conc.	% Diff.	Det. #1 Conc.	Det. #2 Conc.	% Diff.
1	75	75	0%	49	51	4%
2	54	52	4%	57	61	6%
3	115	105	9%	54	59	8%
4	100	95	5%	63	63	0%
5	125	110	12%	60	66	9%
6	65	64	2%	53	53	0%
7	40	40	0%	69	67	3%
8	54	54	0%	63	63	0%
9	92	94	2%	65	66	2%
10	110	100	9%	66	64	3%
11	88	85	3%	73	70	4%
12	72	75	4%	73	72	1%

the HEPA bank only may not sufficiently demonstrate that the sample manifold is adequate for detecting adsorber bypass.

WSRC initiated a test program to retest the sample manifold at the Site to determine if a different configuration is required. Other solutions such as installation of a temporary mixing baffle upstream of the sampler to promote bulk air mixing were also considered. This baffle would be installed prior to performing bypass testing and removed during reactor operation. The baffle would introduce a more tortuous air flow path to the manifold. However, project funding was canceled following the Department Of Energy decision to place the K-Reactor in a stand-by condition. No actual testing of a new manifold or mixing baffle was conducted.

Table 5
Sample Manifold Concentration vs.
Single Point Sample Concentration

Test Point N°	Manifold Sample Concentration	Single Point Sample Concentration	% Difference
1	50	47	6%
2	52	55	6%
3	51	80	57%
4	73	75	3%
5	51	52	2%
6	45	35	22%
7	54	25	54%
8	53	30	43%
9	54	49	9%
10	53	53	0%
11	57	56	2%
12	64	46	28%

IV. Summary

The new N° 2 filter compartment installed at the K-Reactor at the Savannah River Site successfully met all N509 acceptance test requirements, with the possible exception of the adsorber bed in-place leak test. These test results remain suspect pending further testing to demonstrate the acceptability of the sample manifold to detect adsorber bed bypass. Scoping test results suggest that successfully passing Appendix D manifold qualification testing in itself may not ensure that an adequate manifold design is provided. However, the additional testing necessary to prove this hypothesis was canceled following the Department Of Energy's decision to place the K-Reactor in cold stand-by. No further testing is foreseen in the near term.

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AN EVALUATION OF EFFORTS BY NUCLEAR POWER PLANTS TO USE ASTM D3803-89

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Abstract

A number of nuclear power plants are now using ASTM D3803-89, "Standard Test Method for Nuclear-Grade Activated Carbon" for routine surveillance testing of adsorbents. In order to judge the impact of this change, we have gathered radioiodine removal test results from our data base on a system-by-system basis (i. e. control room, technical support center, and spent fuel pool) and compared test results obtained for the same kind of systems using the new and older test methods. Included in this comparison are systems with and without humidity control.

Results are discussed from the standpoint of what to expect if a change to testing using ASTM D3803-89 is contemplated, especially regarding test results in light of existing acceptance criteria. Additionally, the results are discussed from the standpoint of the sensitivity of the ASTM test method to detect when the performance of the carbon in air cleaning systems has been compromised (compared to the older methods). Finally, we offer some suggestions for how other plants might upgrade their carbon testing to incorporate testing to ASTM D3803-89.

Introduction

It has now been seven years since the publication of NRC Information Notice No. 87-32, "Deficiencies in the Testing of Nuclear Grade-Activated Charcoal."[1] In a previous paper, we discussed the efforts of nuclear power facilities in the U. S. to interpret and apply the new standards for testing adsorbents used in their air cleaning systems and at the same time to address issues raised by the Information Notice.[2] In this paper, we have gathered radioiodine test results obtained from routine testing of particular air cleaning systems (control room, technical support center, and spent fuel pool) using the new and older test methods. Although we find more plants successfully using ASTM D3803-89[3] (or its equivalent), there are still many plants that do not. This situation will not change unless the NRC takes a stronger leadership role by either revising Regulatory Guide 1.52 or applying the available new standards in a consistent fashion.

Methyl Iodide Removal Efficiency Results

Table 1 lists results for testing carbon from Spent Fuel Pool air cleaning systems, Table 2 lists results from testing carbon from Technical Support Center air cleaning systems and Table 3 lists results from testing carbon from Control Room air cleaning systems. Included in the 30°C/95% R. H or 70% R. H. results are results from carbon samples tested at 25-30°C and equilibrated with 95 or 70% R. H. air. We consider this to be testing to ASTM D3803-89. Test results shown under ASTM D3803-79 are results from carbon samples tested at 25-30°C without equilibration at 95 or 70% R. H. For other results listed, no distinction was made between equilibrated or non-equilibrated samples. Results shown in parentheses are for 4" test beds, while single digit numbers in parentheses after test results are bed depths in inches.

TABLE 1
PERCENT METHYL IODIDE REMOVAL EFFICIENCY
FOR SPENT FUEL POOL CARBON SAMPLES
(INCLUDES FUEL HANDLING)

D3803-89 30°C / 95% R. H.	D3803-89 30°C / 70% R. H.	D3803-79 30°C / 70% R. H.	51.7°C / 95° R. H.
78.14	99.92	99.94	99.95
78.31	99.78	99.95	
96.25	99.31	99.99	66°C / 95% R. H.
96.24	94.89	99.97	82.32
97.71	99.99	98.77	99.66
87.21	99.24	98.81	99.81
93.30	99.66	99.80	91.36
98.00	99.83	99.79	99.88
92.02	99.93	99.98	99.96
99.45		99.99	99.94
84.20		99.81	
93.70		99.99	
98.65		99.96	80°C / 70% R. H.
98.51		99.58	99.40 (99.98)
93.12		99.90	99.45 (99.99)
99.55		99.96	99.89
96.74		99.59	99.88
92.69		98.37	99.97
95.71		99.84	99.92
96.75		99.95	99.43
88.35		99.96	
96.75		99.95	
99.37		99.98	80°C / 95% R. H.
96.49		99.91	95.98
88.89		99.98	88.10
99.09		99.90	99.93
93.80		99.94	
88.89		99.95	
88.82		99.94	130° / 95% R. H.
86.42		99.96	99.92
91.21		99.94	
99.94			
99.70			
99.54			
97.40			
97.95			
99.83			
98.01			
98.93			
99.16			
99.81			
94.05			
99.31			
96.92			
96.25			
98.64			

TABLE 2
PERCENT METHYL IODIDE REMOVAL EFFICIENCY
FOR TSC CARBON SAMPLES

D3803-89 30°C / 95% R. H.	D3803-89 30°C / 70% R. H.	D3803-79 30°C / 70% R. H.	51.7°C / 95% R. H.
91.47	99.58	99.98 (4)	99.92
93.80	99.89	97.07 (99.89)	99.76
84.70	76.59		96.96
78.73	99.30		87.10
90.54	99.62		99.93
99.37	99.69		90.49
99.51	99.96		
88.58	96.91		66°C / 95% R. H.
90.37	99.88 (99.98)		97.53
98.56	98.48 (99.97)		99.68
99.65	98.61 (99.93)		
94.25	99.96 (99.99)		
92.55	98.61 (99.98)		80°C / 70% R. H.
80.01	99.58 (99.99)		99.96
99.85	99.99+		99.87
93.85	97.29 (99.70)		99.96
99.94			99.98
99.85			99.81
93.87			99.94
98.90 (4)			
99.99 (4)			

TABLE 3
PERCENT METHYL IODIDE REMOVAL EFFICIENCY
FOR CONTROL ROOM SAMPLES

D3803-89 30°C / 95% R. H.	D3803-89 30°C/ 70% R. H.		D3803-79 30°C / 95% R. H.	80°C / 70% R. H.
98.73	99.04	97.88 (99.97)	94.60	99.89 (99.98)
91.60	99.86	99.80	96.54	99.66
97.64	99.50	98.38*		99.96
99.83	99.42	99.88*		99.94
99.86	92.35	99.93 (99.99)	D3803-79	99.72
96.85	90.89	99.55	30°C / 70% R. H.	99.86
89.59	99.85	99.95 (4)	99.99 (4)	99.98
99.00	99.30	99.09 (99.99+)	99.99 (4)	99.79 (99.99)
99.24	99.19	99.14*	99.96	99.93
92.81	99.90	84.45 (99.90)	99.95	99.99
94.95	99.86	99.69	99.96 (4)	99.52
92.75	99.19	85.95 (99.86)	99.97 (4)	99.99 (99.99+)
96.64	90.89	99.93	99.98 (4)	99.99
94.99	78.90		99.99 (4)	99.97 (99.99)
90.01	97.37		99.95	
99.31	99.92		99.95	80°C / 95% R. H.
91.36	99.09		99.92	99.77
83.73	84.45		99.99 (4)	99.67
99.31	99.69		99.89	92.03
97.36	85.95		99.96 (4)	84.85
97.35	99.93		99.95	99.98
99.06	99.78		99.99 (4)	99.91
49.69	99.93		98.30	
47.63	99.95		99.87	
90.56	99.92		99.98	130°C / 95% R. H.
95.29	99.45			99.97
96.69	99.68			99.93
93.58	86.53		51.7°C / 95% R. H.	99.79
97.84	99.96		99.90	99.96
99.86	82.13		99.47	99.91
99.83	99.90		99.80	
99.55	98.39		99.90	
83.98	99.93		99.76	
98.78	99.99		96.96	
99.54	99.97		99.67	
86.64	99.99		98.54	
99.30	99.30		98.36	
99.99+(6)	99.99		98.44	
99.56	99.91		96.01	
98.84	98.54		79.74	
98.63	99.03		99.47	
97.36	99.17			
97.35	99.69 (99.94)		66°C / 95% R. H.	
97.64	99.81		99.92	
	99.90		99.89	
	99.87		99.87	
	80.30 (99.72)		99.96	
	83.33 (99.74)			

Discussion of Results and Recommendations

Spent Fuel

It is encouraging to see that most of the test results shown here are obtained from using some version of the D3803 method. This is not too surprising given that the acceptance criterion for this system is usually an efficiency of >90%. The D3803-89 30°C/95% R. H. results of <90% are generally a first time result and represents a need for a carbon change. Thus, a >90% efficiency is a reasonable acceptance criterion for the D3803, 30°C/95% R.H. test. The D3803-89, 30°C/70% R. H. results indicate that an efficiency of >99% can be expected. Thus, we would recommend that systems currently tested at temperatures >30°C and 95% R. H. be tested at 30°C and 95% R.H. according to ASTM D3803-89. All the other systems currently being tested at some temperature and 70% R.H. should be tested at 30°C and 70% R.H. according to ASTM D3803-89. The 30°C/70% R.H. test according to ASTM D3803-79 (without equilibration) is not sensitive enough to the carbon's condition[4] and should be replaced with the 89 version.

TSC

Even though these systems are generally the newest (and best) in the plant, most, if not all, are non-ESF systems. Thus, these systems were designed to be tested at 25°C/70 or 95% R. H. with an efficiency of >90% and the change to D3803-89 at 70 or 95% R. H. has been rather painless. The systems tested using D3803-79 should changed to the 89 version as should the systems tested at the other temperatures >30°C. It was nice to see no 130°C tests in this data (from end of 92 to present), but one wonders why the other temperatures are used.

Control Room

The situation for the control room systems is more complicated. The required efficiency is typically >99% for 2 inch systems and >99.825% for 4 inch systems. This is a difficult requirements for 2 inch systems tested at 30°C and 95% R. H. to D3803-89. A >95% efficiency would be more reasonable, but would require the lowering of the assigned decontamination efficiencies. The results for samples tested at 30°C and 70% R. H. to D3803-89 indicate that an efficiency of >99% for two inch systems and >99.8% for 4 inch systems can be reasonably expected. The lower efficiency test results shown generally are first time results using the newer test methods after historically testing the carbon from these systems at higher temperatures. The high temperature testing of carbon samples from Control Room systems should be replaced with the 30°C/70 or 95% R. H. D3803-89 test. The same holds true for systems tested using D3803-79. It is discouraging to see that some Control Room systems are still tested at 130°C.

Conclusions

The industry must make a decision about testing carbon from air cleaning systems for methyl iodide removal. Do we want to test the carbon with a method that is sensitive to the condition of the carbon or with a test method that allows the carbon to meet the assigned decontamination efficiency for some assigned accident scenario? Systems that can use ASTM D3803-89 at 30°C and 70% R. H. (and hence have humidity control) can do both. But most ESF systems will require a derating of their assigned decontamination efficiency when tested to D3803-89 at 30°C and 95% R. H. Can the NRC provide a uniform, non-litigious means to accomplish this?

References

1. NRC Information Notice No. 87-32: "Deficiencies in the Testing of Nuclear Grade Activated Charcoal"; United States Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, DC 20555, July 10, 1987.

2. Freeman, W. P., "Testing of Adsorbents Used in Nuclear Power Plant Air Cleaning Systems Using the 'New' Standards." 22nd DOE/NRC Nuclear Air Cleaning Conference, CONF-9020823, Vol. 2, pp 661-671.

3. ASTM D3803-79, 86, 89; "Standard Test Methods for Radioiodine Testing of Nuclear-Grade Gas-Phase Adsorbents", ASTM, 1916 Race Street, Philadelphia, PA 19103-1187.

4. Scarpellino, C.D. and Sill, C.W.; "Final Technical Evaluation Report for the NRC/INEL Activated Carbon Testing Program," EGG-CS-7653 April 1987, Idaho Falls, Idaho 83415.

DISCUSSION

HAYES: In your conclusion you recommended that the NRC have some definitive guidelines with respect to decontamination efficiencies. Are you recommending that the NRC revise Table 2 of Regulatory Guide 1.52 and, based upon the data you have presented, show an adsorber efficiency credit of 85% or 90%, for the safety analysis with a laboratory testing acceptance criterion of 95%?

FREEMAN: Yes, for carbon samples from systems that have 2-inch beds tested at 30° C and 95% RH according to ASTM standard D3803-89.

KOVACH, L.: In last week's ASME meetings and several discussions that industry groups had with the NRC, the recommendation came out that if the industry knows better than the NRC, they should ask for a technical specification change on an individual plant basis. Not just as an industry representative, but also as a taxpayer, I am upset by this attitude that if there is something stupid in a technical specification (that was put in some time ago and blessed by NRC) and when I want to do something different, I have to go through all these hoops to change it to something that makes engineering common sense on the basis of what we have learned. I think we need and expect a little more leadership from NRC than has been demonstrated in relation to this particular problem. I know it is a political question and I know it is a legal question, but at the same time I realize that NRC is so busy licensing all these new plants which are being built that they don't have time to work on a problem that they didn't even discover. The industry, at its own expense and endeavor, pointed it out to them as being a problem. It is very frustrating to have this issue under discussion for years without resolution from the NRC. It is clearly defined by current knowledge that a problem exists. Presentations were made to the NRC without even a response. The rest of the world uses common sense test conditions. The U. S., with the most extensive regulatory organization, did not adopt the proper test conditions even after industry pointed at the non-conservative assumptions of the regulated test conditions.

FREEMAN: You are preaching to the choir here. From my standpoint, both as a taxpayer and a Joe Public, what discourages me is that, here we are trying to protect the safety of people that work in power plants, (and people outside power plants if there is an accident), and we are using test methods that are contrary to that endeavor. We can meet a 1% penetration requirement by currently required laboratory tests and yet the system may contain a carbon that would only perform in the low 40% removal efficiency range in a not-so-serious accident. That's really what disturbs me the most.

CLOSING COMMENTS OF SESSION CO-CHAIRMAN WEIDLER

I am going to give a brief summary, and then I'll follow up in a Southern way what Dr. Kovach said. We heard some diverse papers, and I am going to sum them up individually, but the common theme was that we need to continue development and refinement of codes and standards. It is a significant need.

All the presentations were excellent. Ken Deaton's paper on Challenges of Equipment Qualification, and his discussion of the commercial dedication of 1E motors versus the qualification issue point out that the licensee needs to exercise caution in the commercial dedication of items such as motors, because they are extremely complex. From my perspective, it begs a question: are the guidance documents that are currently available to the industry on commercial specifications adequate? I think it is a question we will have to explore down the road a bit.

In Tim Arndt's paper on Evaluation of the Self-Contained HEPA Filter, he talked about the specific design not being recognized by codes and standards, which I consider to be somewhat of a good point, because I don't think codes and standards should restrict specific designs. He pointed out the many areas that are in non-compliance with existing codes and standards for this filter, and the recommendation not to use this filter for nuclear applications. I would like to say that, from my perspective with codes and standards organizations, that when you run into a problem where you are trying to use new technology that doesn't "fit" into the existing standards, such as N-509 and N-510, please, by all means, contact that particular organization and have discussions with them.

The Paul and Punch paper on the new filter system at Savannah River noted that it passed all the acceptance tests in N-510 with the exception of the adsorber in-place leak test. This has not yet been resolved, due to a decision to put the K-reactor on standby. I checked my records on the inquiry to N-510, and they show that CONAGT still has not completed its response. We need to get together and resolve whether our (CONAGT's) records have gone astray or if we have given a wrong impression to Westinghouse Savannah River. I will certainly do that.

Freeman's paper on the Evaluation of Efforts by Nuclear Power Plants to Use ASTM D-3803, shows that a number of plants do use D-3803 for routine surveillance testing. He also pointed out that the adsorbent is being evaluated by different criteria. I conclude by saying that CONAGT took this matter up with the Nuclear Regulatory Commission in April of 1993. To this point, we have not had a response to the issue that we brought before them. We will continue to work for a resolution that is good for all, and we hope that by the next air cleaning conference we can close this issue out.

OPENING COMMENTS OF SESSION CO-CHAIRMAN JACOX

The title of this session is Application of Nuclear Air Cleaning and Treatment Codes and Standards. We are not going to limit it, but it will be oriented toward DOE facilities, because we have been talking primarily about NRC facilities for 22 conferences. We are fortunate that we have panel members from DOE headquarters, users, and consultants, so we can get a variety of viewpoints. We will start with relatively brief papers, so you will have an idea where we are coming from, and then leave at least an hour for open discussion. I think it is a subject that should engender a lot of questions and discussion. And if we do not get enough discussion from the floor, I, personally, have a couple of questions I want to ask the other panel members.



SESSION 3

PANEL SESSION: APPLICATION OF NUCLEAR AIR CLEANING AND TREATMENT CODES AND STANDARDS

Monday: July 25, 1994
Co-Chairmen: J. Jacox
C. Graves

Panel Members: G. Sherwood
B. Mokler
T. Arndt

OPENING COMMENTS OF SESSION CO-CHAIRMAN JACOX

HEPA FILTER TESTING - DEPARTMENT OF ENERGY OFFICE OF NUCLEAR ENERGY FACILITIES
G. L. Sherwood

AN EXAMPLE OF A COMPONENT REPLACEMENT WHEN APPLYING ASME N509 AND ASME N510 TO OLDER VENTILATION SYSTEMS
T. E. Arndt

A SINGLE STANDARD FOR IN-PLACE TESTING OF DOE HEPA FILTERS - NOT
B. V. Mokler

INTEGRATING INDUSTRY NUCLEAR CODES AND STANDARDS INTO UNITED STATES DEPARTMENT OF ENERGY FACILITIES
J. Jacox

PANEL DISCUSSION

HEPA FILTER TESTING - DEPARTMENT OF ENERGY OFFICE OF NUCLEAR ENERGY FACILITIES

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Abstract

This paper provides the background of, and some results from, a review of HEPA filter testing during 1993 at selected Department of Energy (DOE) facilities. Recommendations for improvements in standards resulting from the review are also presented.

I. Introduction

Testing of HEPA filters at selected Office of Nuclear Energy facilities was reviewed during 1993, by a team of well qualified reviewers. Members of the team were the following:

George L. Sherwood, Office of Nuclear Energy, DOE
Ronald C. Scripsick, Los Alamos National Laboratory
Brian V. Mokler, Environmental Health Services, Inc.
John Brockmann, Sandia National Laboratories
James E. Martin, University of Michigan

The review started following preparation and distribution of an information request (Figure 1) and analysis of responses to it (Figures 2 and 3). Details of the review will be provided in Brian Mokler's paper to be presented later in this conference.

The following preliminary observations resulted from the review:

1. Testing of HEPA filters at all NE facilities except the Portsmouth Gaseous Diffusion Plant is adequate.

2. Testing of HEPA filters at these sites can and should be improved.

3. Additional support for HEPA filter testing should be provided.

4. HEPA filter testing standards should be improved.

Each of these observations is explained in turn in the following paragraphs.

INFORMATION REQUEST (10/22/92)

1. LISTING OF HEPA FILTERS IN USE

2. TESTING PERFORMED

3. TESTING REQUIREMENTS

4. ASSISTANCE?

FIGURE 1

RESPONSES
(12/92 TO 3/93)

- o GDPs
  - PADUCAH
  - PORTSMOUTH
- o REACTORS
  - HFIR
  - BMRR
  - HFBR
  - EBR II (AND SUPPORT FACILITIES)
- o OTHER
  - MOUND (BLDGS 38 AND 50)
  - SAVANNAH RIVER (BLDG 235-F)
  - LANL (TA-55/PF-4)
  - ORNL (REDC AND IEF)

FIGURE 2

NE HEPA FILTER USAGE*

SERVICE	#
URANIUM/UF_6	~ 200
PLUTONIUM	~ 300
REACTOR EXHAUST	~ 200
OTHER	~ 300
TOTAL	~ 1000

*2' x 2' x 1' (1000 CFM)

FIGURE 3

II. Discussion

A thorough review of HEPA filter testing at NE facilities was completed during 1993. Testing was being performed, at all facilities but one, in a very professional manner. Testing was being performed as required, by experienced, well trained personnel. There were no serious gaps or flaws in the testing. However, sampling conditions for testing were often less than optimum. In addition, the depth of knowledge of HEPA filter testing at some facilities was less than desirable. Sampling is the area of HEPA filter testing where improvement and additional support should be provided.

This review has pointed out some areas where standards for HEPA filter testing can and should be improved, as follows:

1. Although HEPA filters used in most instances at NE facilities had been pretested at DOE filter test facilities prior to installation, this pretesting is not required in any of the standards or in any DOE Orders.

2. The HEPA filter standards may be too specific in some areas. In particular, flexibility and/or additional provisions in standards are needed for such variations as

   a. Testing multiple HEPA filters simultaneously,

   b. Allowing for ranges in rated air flow,

   c. Permitting alternative measuring devices (i.e., laser light scattering systems in addition to photometers), and

   d. Dealing with situations where the sampling conditions are not ideal (e.g., sampling ports not at least 10 duct diameters downstream).

III. Recommendations

The following recommendations are offered, with the hope that they will be responded to positively.

1. Provide Quality Assurance support not only for the DOE HEPA Filter Test Facilities, but also for the various organizations performing testing at DOE facilities.

2. Consider modifying existing HEPA filter standards to address the following; testing multiple filters

simultaneously, variations in air flow, use of alternative testing methods, and allowances for non-ideal sampling.

3. Make the practice of pretesting HEPA filters for use in DOE facilities an absolute requirement.

References:

1. LA-12763-SR (Status Report),
High Efficiency Filter Systems
General Observations, 1992-1993,
Los Alamos National Laboratory, May 1994

2. Safety and Health Issue,
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U.S. Department of Energy, NE-80, 10/22/92.

AN EXAMPLE OF A COMPONENT REPLACEMENT WHEN APPLYING ASME N509 AND ASME N510 TO OLDER VENTILATION SYSTEMS

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Abstract

This paper presents an example of a component replacement (electric heater) when installed in an older ventilation system that was constructed before the issuance of ASME N509[1] and N510[2]. Many of the existing ventilation systems at the Hanford Site were designed, fabricated, and installed before the issuance of ASME N509[1] and N510[2]. Requiring the application of these codes to existing ventilation systems presents challenges to the engineer when design changes are needed.

Although it may seem that the application of ASME N509[1] or N510[2] may be a hindrance at times, this does not need to occur. Proper preparation at the start of project or design modifications can minimize frustration to the engineer when it is judged that portions of ASME N509[1] and N510[2] do not apply in a particular application.

I. Introduction

Westinghouse Hanford Company (WHC) must comply with the U.S. Department of Energy (DOE) Order 6430.1A, *General Design Criteria*[3]. The order states in part "These criteria apply to any building acquisition, new facility, facility addition and alteration, and leased facility that is required to comply with DOE Order 4300.1B[4]. This includes on-site constructed buildings, pre-engineered buildings, plant-fabricated modular buildings, and temporary facilities." DOE 6430.1A, Section 1550-2.5.5, titled "Air-Cleaning Devices," requires that all high-efficiency particulate air (HEPA) filtration systems comply with ASME N509[1] and be tested in accordance with ASME N510[2].

In addition to DOE Order 6430.1A[3] requiring ASME N509[1] and ASME N510[2], RLIP 5480.4C[5], titled *Environmental Protection, Safety, and Health Protection Standards for RL*, Section 10, "Nuclear Safety Standards," paragraph b, "Nuclear Facility Safety," also invokes ASME N509[1] and ASME N510[2] as mandatory standards.

II. Existing Ventilation Systems

Many of the existing ventilation systems at the Hanford Site were designed, fabricated, and installed before the issuance of ASME N509[1] and N510[2]. Requiring the application of ASME N509[1] and N510[2] to existing ventilation systems presents challenges to the engineer when design changes are needed.

The following example will attempt to illustrate complications that are encountered when applying ASME N509[1] and N510[2] to older ventilation systems that were constructed before the issuance of these codes.

Replacement of a Steam Heater with an Electric Heater

ASME N509, paragraph 5.5, states "Heaters shall be electric and capable of meeting the requirements of paragraph 4.5 of this standard."[1]

Many of the older ventilation systems are installed using steam heat. When a modification to the exhaust train is required (i.e, a new HEPA filter housing is being replaced), problems are encountered when the use of a steam heater is requested. ASME N509 states "Heaters shall be electric and capable of meeting the requirements of paragraph 4.5 of this standard."[1] In some instances, requiring electric heat also requires the addition of electrical power because of the unavailability of power where the heater is located. The addition of the power will also require an electrical upgrade to meet the current electric code requirements.

In addition, within the tank farm complex certain tanks contain hydrogen. For tanks that contain certain levels of hydrogen it is required that ventilation systems and their components must be intrinsically safe.

A conflict arises when applying paragraph 5.5 of ASME N509.[1] Because an electric heater is required, the system cannot be intrinsically safe because of the amount of electric energy required to heat the gas stream. On the other hand, a steam heater will satisfy the requirement.

Where the intent was to use the existing steam heater and keep costs reasonable and within budget, now it becomes more costly and complex because an electric heater must be installed instead of reusing the steam heater.

III. Summary

ASME N509[1] and N510[2] are mandatory codes required by DOE Order 6430.1A[3]. Applying these codes to older ventilation systems that were constructed before their issuance is often frustrating. Often simple modifications can improve system performance, but the design will not progress unless full compliance to all the code requirements are adhered to.

Although it may seem that the application of ASME N509[1] or N510[2] may be a hindrance at times, this does not need to occur. Proper preparation at the start of project or design modifications can minimize frustration to the engineer when it is judged that portions of ASME N509[1] and N510[2] do not apply in a particular application.

If the engineer can provide a technical justification for deviation from ASME N509[1] or N510[2] and the justification will demonstrate that another method would be technically similar or produce the results intended by these ASME codes, then a waiver can be requested from the DOE for deviation.

DOE Order 6430.1A allows for waivers from the *General Design Criteria*, provided the deviation follows specific guidelines.[3] The following is quoted in part from Section 0101-2, titled "Criteria Deviations."

> For all projects subject to DOE 6430.1 series, these criteria are not intended to impose unnecessary design restrictions or requirements or to discourage design innovation. Professional architectural and engineering judgment shall be used in the

interpretation and application of these criteria to specific projects.

The contractor and/or DOE organizations responsible for facility projects shall review these criteria early in the planning phase and at later phases during the project construction process to determine if any of these criteria are not applicable or are not appropriate.

The contractor shall document the criteria being used for each project in the project's SAR (per Section 0110-5.2, "Safety Analysis") such that compliance with these criteria can be verified during design, construction, and facility operation. Site-specific criteria shall be included in this documentation.

The principal objective at the Hanford Site is to provide safety to personnel and the environment. The ASME N509[1] and N510[2] are necessary and aid in preventing unsafe HEPA filtration systems from being designed and installed.

If, in the opinion of the engineer, a particular paragraph does not apply, then the responsibility rests upon the engineer to convince the appropriate personnel that, based upon technical justification, a waiver from ASME N509[1] or N510[2] is warranted.

Frequently a simple modification will enhance the system performance and become safer than no modification at all.

IV. References

1. ASME N509, *Nuclear Power Plant Air Cleaning Units and Components*, ASME N509-1989, American Society of Mechanical Engineers, New York, New York.

2. ASME N510, *Testing of Nuclear Air-Cleaning Systems*, ASME N510-1989, American Society of Mechanical Engineers, New York, New York.

3. DOE Order 6430.1A, *General Design Criteria*, U.S. Department of Energy, Washington, D.C.

4. DOE Order 4300.1B, *Real Property Management*, U.S. Department of Energy, Washington, D.C.

5. RLIP 5480.4C, *Environmental Protection, Safety, and Health Protection Standards for RL*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

A Single Standard for In-Place Testing of DOE HEPA Filters - Not[1]

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In order to understand why I think it is inappropriate to apply a single in-place testing standard to all HEPA filter installations at U. S. Department of Energy HEPA facilities, I need to explain my views on two basic issues. The first is what is meant by "standard." To me a test standard is a document that specifically includes all procedures that **must** be completed to conduct a test. Acceptable tests are only those performed in strict accordance with these procedures. No options, additions, or alternatives are allowed by a standard. In this idealization of a standard the directions are all phrased in terms of "shall." To the extent procedures are written permissively, allowing options and alternatives, the document becomes less of a standard and more a description of suggested good practice.

The second issue is the variety of DOE HEPA systems that require in-place testing. They encompass a very wide range of design and construction practices. Many of the systems were designed and built before the virtues of and need for in-place testing were recognized. It would probably be possible, with minor modifications to the existing hardware, to adapt some systems to allow use of a standardized in-place test. In other cases, extensive modifications would be required. Finally, there are some systems that may have to be replaced before accurate in-place testing data can be obtained. In the latter two cases, we must recognize that many of the facilities served by these HEPA filtration systems are nearing the end of their technical and economic usefulness. Careful consideration must be given to all possible methods of risk management. Is it reasonable, on both practical and economic grounds, to use limited resources to retrofit these systems so they could, for a short time, be tested in exactly the same way newer systems are tested? Can the same ends be achieved by other means such as administrative controls?

These two issues do not mean that we cannot or should not test the performance of HEPA filter systems. The problem is to do it so that the measurements are as representative as possible and their limitations are understood and acknowledged. Both the reporting of the results and the required performance specified for a system must recognize that a calculated penetration and the aerosol concentration measurements from which it is derived are not absolute values, free of all uncertainty. Uncertainties in the measurements arise from several factors, primarily system design, sampling problems, and test instrument performance. Technically, we probably can measure aerosol concentrations with more assurance than we can establish that the measurement is representative of the average concentration at the sampling plane. The difficulty with many of the sampling locations on older HEPA filter systems at DOE facilities is that there has been no documented effort to establish the quality of the sample available when the facility's routine sampling procedures are followed. Before efforts are made to correct these problems, it would be appropriate for DOE to provide guidance to facility operators on what DOE expects to be accomplished by in-place testing. For example, are in-

place tests to be used only to detect gross system defects or are they also expected to verify that system performance meets design requirements? This guidance should be primarily technical in nature, stressing the need for valid measurements and describing means of achieving this goal. The guidance should also provide an approach for facilities to use in revising performance specifications to allow for the uncertainty inherent in filter bank penetration determinations.

HEPA filter installations at DOE facilities must be tested regularly to verify that they provide the filtration performance required of them. The test results must be an accurate measurement of the performance. It appears very unlikely that a true standard could be written to properly describe the testing requirements and procedures for all DOE HEPA filter installations. It would be more appropriate for DOE to provide guidance to its contractors on required elements of testing programs and models of good engineering practice associated with these elements. This guidance should provide a clear statement of DOE's expectations for in-place testing while being flexible enough to meet the needs and constraints of the wide variety of systems found at DOE facilities.

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[2] Environmental Health Sciences, Inc., contractor to Los Alamos National Laboratory.

23rd DOE/NRC/ISNATT NUCLEAR AIR CLEANING AND TREATMENT CONFERENCE

INTEGRATING INDUSTRY NUCLEAR CODES AND STANDARDS INTO UNITED STATES DEPARTMENT OF ENERGY FACILITIES

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Abstract

Recently the United States Department of Energy (DOE) has mandated facilities under their jurisdiction use various industry Codes and Standards developed for civilian power reactors that operate under U.S. Nuclear Regulatory Commission License. While this is a major step forward in putting all our nuclear facilities under common technical standards there are always problems associated with implementing such advances. This paper will discuss some of the advantages and problems experienced to date. These include the universal challenge of educating new users of any technical documents, repeating errors made by the NRC licensed facilities over the years and some unique problems specific to DOE facilities.

Introduction

As early as the 1960s, there have been Industry Codes and Standards for Nuclear Power Plants. These were first written when the original federal government agency for nuclear facilities was the "Atomic Energy Commission" (AEC). At the time, the AEC had jurisdiction of all United States nuclear facilities. Over the years, this common regulatory control has been split and gone through a number of political incarnations. This split responsibility has created a diversity of technical and engineering approaches to many common situations. With the recent decline in funding of weapons research and production, along with the great increase in "environmental" concerns, the use of common technical and engineering approaches as well as common hardware has again become the order of the day. In the area of nuclear air/gas treatment, this new commonality is mandated by DOE Order 6430.1A, "General Design Criteria" which requires, through various references, that all HEPA filtration systems comply with ASME N-509 "Nuclear Power Plant Air Cleaning and Components" (1) and ASME N-510-1980 "Testing of Nuclear Air Treatment Systems" (2). A revision of this DOE Order is being worked on that will add a third ASME document on Nuclear Air/Gas Treatment, ASME AG-1 "Code On Nuclear Air and Gas Treatment Systems" (3). When, or if, this revision will be issued is not known to the author. While there are many other industry Codes, Standards and documents that are also now mandated for DOE facilities, this paper will discuss these three documents and the related air and gas treatment systems they are mandated for application to. Note that the 1980 editions of N509 and N510 have been superseded by 1989 editions and are no longer in print.

The first and main point to keep in mind when using N509 and N510 is that through their currently three editions they have been written for a very specific application. This is to design, fabricate and test filtration systems for pressurized light water power reactors Licensed by the NRC. Even more specifically they are for systems intended to mitigate the consequences of a design basis accident as specified in NRC Regulatory Guide 1.52 (4). And even further, the gaseous contaminant of interest is primarily radioiodine 131. These documents have been reasonably successful in meeting the needs of the nuclear power plants for design, fabrication, commissioning, operation and maintenance of the subject systems. However, everyone using these Standards must always remember their specific and limited scope and intent. This is not to say they are limited, as excellent technical guidance, to only the Regulatory Guide definition of systems but there is a vast difference in using a Standard for technical guidance and "to the letter" bureaucratic enforcement of a Standard.

As a point of interest, the United States Department of Defense is also converting to industry standards as much as possible and is encountering essentially the same problems as the DOE has encountered.

Discussion

There are obvious and significant advantages for DOE facilities using Industry Standards. The Codes and Standards have evolved over decades and are, in most cases, very realistic from both technical and cost standpoints. As Industry consensus Standards they have been developed by manufacturers, users, regulators and other technical experts. These standards have been subject to public review and the ultimate test, actual use. Cost savings result from commercial industry input from the start and from economies of scale from using "standard" components, design, testing and integration. Additionally, there is a very large body of literature on these filtration systems. This is primarily in the Proceedings of these International Conferences. All these features add up to the ability to use well-proven components and systems. Unfortunately, the comprehensiveness of this material, physical and informational, means new users have considerable material to learn before they can properly benefit from it.

This self education phase is particularly difficult today given the scope of the environmental problems the DOE technical community is mandated to resolve, the great budget pressures to reduce costs, the ever increasing media attention to every detail of cleanup efforts and the somewhat insular working environment of DOE facilities worked in until recently. Engineers are trying to solve serious technical problems, often created decades ago, with limited time and resources, under politically created schedules and being second guessed every step of the way by non-technical insiders and outsiders. This leaves little time for the study required to fully understand the technical background and nuances of the newly mandated Standards. Further complicating the situation is the limited scope of these Standards. The author, as a consultant, as a member of the writing Committee - The "Committee On Nuclear Air And Gas Treatment" (CONAGT), sponsored by the American Society of Mechanical Engineers - and through many discussions with colleagues from all areas of involvement, has seen problems from use of N509 and N510 without true understanding of their content and scope. One of the most basic problems is "over-specification." This is usually done by someone who has not read and understood the documents he specifies. An example is specifying too many related, but not totally consistent, industry and government codes, Standards, Guides and Military Specifications. The most common error is to specify N509, N510 and AG-1. These are very similar documents in most ways but are NOT identical. Worse, the specifier may either mix editions or not specify the edition at all. The meticulous vendor will attempt to explain the often subtle problems that arise from this type of mis-specification but many vendors simply ignore them. Unfortunately the buyer, who is not as familiar with the documents, may prefer to work with the vendor who does not ask questions or "cause him trouble." This approach simply puts off the questions from the early stage when they are only on paper to the final stage when they are now hardware. Even a basic component such as the HEPA filter can be mis-specified very easily by referring to more documents than is necessary. This is detailed in a paper from the last Conference entitled "Review of HEPA Filtration Test Standards and Their Application to Nuclear Applications" (5).

Another all too common error in using the standards is simply not understanding exactly what you are specifying when invoking them contractually. Usually the result of this lack of understanding is not getting the performance and/or physical quality required. Vendors usually interpret requirements in the lowest cost manner. This may or may not actually meet the Standard but regularly does not result in a component or system that meets the users requirement. Worse, in nearly every case, the user does not return the item and require the vendor to correct the deficiency but somehow lives with the substandard item because of schedules. It appears that while there is never time to properly understand, specify, oversee, shop inspect, and test items there is always some technical or "paper" way to make deficient items acceptable to meet a schedule. Certainly this is a universal challenge for all complex technical endeavors, but the mind set of using Codes and Standards to the "legal letter" in all nuclear facilities can not replace good engineering experience, judgement, and understanding. Of course, Codes and Standards are not perfect. We who write them are painfully aware of this fact and always welcome constructive feedback on how to improve them. Any written comments to the sponsoring organization are formally transmitted to the responsible committee and given full consideration for inclusion in revision, addenda or correction. However, no Code or Standard can substitute for competent engineering and Quality Control by a user. Whether a deficiency arises from mis-application of a standard, mis-interpretation by the user or vendor or simply from the vendor not fully meeting the Standard requirements, such deficiencies are and must be the responsibility of the buyer/user. A set of major filtration systems costing millions of dollars purchased by a DOE facility was seriously deficient in a number of areas which were blamed by the vendor on poorly written sections of ASME AG-1. The vendor did not meet some

minimum dimensions called out in the Standard. After very lengthy investigations, discussions and formal inquiries the decision of CONAGT was that the Code Section was adequate and accurate and the problem resulted from inadequate buyer understanding and oversight of this vendor. If the buyer's responsible organization had more experience with the Standard used, this expensive problem would probably been avoided. This example includes the overlapping problems of becoming educated in the details of new documents and not repeating mistakes already made. Closer cooperation and cross education between NRC and DOE facilities will help reduce many repeat problems and headaches.

Complicating these general problems and learning curves there are unique circumstances that exist at DOE facilities making the challenges even greater. These circumstances are both technical and regulatory - administrative in nature. No other facilities have such considerations as criticality in liquid or gas/aerosol streams. Power plants do not have the problem of mega-curie sources in unknown chemical brews that generate heat and hydrogen in ways not totally modeled. Plutonium dust is encountered only in DOE facilities. All these and related technical details greatly complicate the use of Codes and Standards not written with them as a part of the documents scope. Administrative misunderstanding of the scope, intent and use of the documents also creates considerable problems for the user's technical staff.

Since the current versions of these documents do not address criticality, gases other than air with radioiodine and minor impregnated carbon poisons, corrosive or toxic gases or high temperature process flow, they must be augmented for many DOE facility projects. CONAGT is working on broadening the scope of AG-1, but the new Sections and revisions of existing sections will take years before they can be published. The Committee needs specific direction from the DOE facilities as to what is needed, the qualified technical members from DOE to help write the new Code Sections. All of these points do not invalidate the technical content of the three documents, they do point out their mandate has not eliminated the need for good engineering in design of new systems and upgrading of existing ones.

At the Hanford site where there are some waste storage tanks that contain highly radioactive liquid - or semi-liquid sludge - complex chemical mixtures. Equipment definition for the ventilation of these tanks; during material transfer, "breathing", to remove heat or to reduce an explosive hazard requires the system to handle complex organic and inorganic gaseous chemicals and prevent their release to the atmosphere is nowhere in N509 or AG-1 - directly. There is excellent technical guidance on how to build the different types of components required to process these chemicals. The successful combination of new technology with existing Codes and Standards is very difficult technically and nearly impossible administratively. It must, however, be accomplished.

The major administrative problems at present are: 1. That AG-1 has not been mandated and some facility administrations interpret this to mean it can not be used at all. 2. The interpretation that unless a component is explicitly defined and detailed in N509 it can not be used at all. 3. That industrial experience, no matter how extensive, can not be used to justify use of existing technology for nuclear application. Of course there are theoretically means to get around these road blocks, but the expenditure of resources to do so can exceed the productive engineering design effort. A suggested solution to many of these quandaries is for the DOE facility technical AND administrative personal to learn the intent of these documents formally from the writing Committee, CONAGT.

Being able to use the latest edition of AG-1 will be of significant help to most DOE sites. It is very much broader in scope than N509 and covers many more types of components. For example, in N509 the only type of heater covered is electrical which is not acceptable for potentially explosive process flows. AG-1 includes steam heaters which solve this problem. There are many other component types included that offer far more flexibility in design. AG-1 is made up of many Sections that are revised and expanded individually so it provides an easier and faster means to respond to industry needs. Additionally formal inquiries to ASME are possible to obtain binding new information and answers to questions until a new revision to the code is issued.

Conclusions and Recommendations

The main conclusion is that the use of Industry Codes and Standards such as ASME N509, N510 and AG-1 will benefit the facilities that use them, technically and economically. This statement is tempered with the strong caution that it will only be true if the documents are used in a proper and enlightened manner, both technically and administratively. Improper or overly restrictive use of these or any Codes and Standards can result in unnecessary expenditures of resources and, worse, inadequate system performance.

Users of these documents must fully understand them and their intent to benefit from their use. This applies to both the technical and administrative user. Some formal courses and training is available for the use of these documents. Harvard has an annual course, ASME has (usually) semiannual seminars and commercial training exists. All should be taken advantage of as well as using the experience of the Power plant industry.

Care should be taken to understand when a DOE facility situation, or part thereof, is truly unique and demands unique engineering and when it has a history in other areas, nuclear or industrial. We must use valid experience from what ever source we can find. When we waste resources reinventing a wheel, it is not only expensive but seldom round.

Maximum long term benefits of bringing the nuclear industry under a consistent set of Industry Codes and Standards will be obtained by both the DOE and NRC if their personnel teach and learn from each other. The DOE facilities must support the modification and expansion of Industry Codes and Standards by providing direction for their needs and qualified engineers to help write them. The existing committees must be actively open to the new challenges of supporting these new facilities and seeking out new members. Historically, CONAGT has sought out new members and had a very open policy for including new technical areas. I hope this paper will motivate DOE facility engineers to join CONAGT, ISNATT (6) and other necessary committees and related organizations for the benefit of all of us in the nuclear industry. Many members of these organizations are here at this conference. Feel free to ask any of us about membership.

References

1. ASME N509-1989; "Nuclear Power Plant Air Cleaning Units and Components", American Society of Mechanical Engineers, 345 East 47th Street, New York, NY 10017

2. ASME N510-1989; "Testing of Nuclear Air Treatment Systems", American Society of Mechanical Engineers, 345 East 47th Street, New York, NY 10017

3. ASME AG-1 1991; "Code on Nuclear Air and Gas Treatment Systems", American Society of Mechanical Engineers, 345 East 47th Street, New York, NY 10017

4. USNRC Regulatory Guide 1.52; "Design, Testing and Maintenance For Atmospheric Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants", Revision 2, March 1978, USNRC, Washington, DC 20556

5. Jacox, J., "Review of Filtration Test Standards and Their Application to Nuclear Applications", 22nd NACC, 1992, CONF-9020823, Superintendent of Documents, USGPO, Mail Stop SSOP, Washington, DC 20402-9328

6. ISNATT "International Society of Nuclear Air Treatment Technologies", P. O. Box 29246, Columbus, OH 43229

PANEL DISCUSSION

JACOX: I have some additional comments before we start the general discussion. Over the years I have worked with a manufacturer, with a testing and consulting firm, and as a consultant to DOE facilities, and can see the problems from most points of view. In regard to how DOE is trying to apply the standards, N-509 and N-510, Tim Arndt said it seems like a hinderance at times. He is a bit more generous than I. I think it's a serious hinderance, at least the way that the Hanford site is trying to apply it.

Here is a little bit of background. When you look at the scope and the limitations of both N-509 and N-510, it becomes apparent they are written for an extremely specific and limited purpose, and that is to meet the requirements of Regulatory Guide 1.52, which applies to engineered safeguard systems to mitigate the effects of a design-basis accident in light water reactors under NRC regulations. That is all these documents are written for. N-510 is written to go with N-509, but it must be done on an edition-by-edition basis because there are three pairs of these documents having dates '75-'76, '80-'80, and '89-'89. You can only apply N-510 to its companion N-509 edition to do things to the letter. Even NRC licensed facilities have a problem with that.

When a DOE facility tries to use these documents to the letter, rather than the spirit, they are making them impediments to doing anything correctly. There is, as Tim Arndt said, a way to get around it, but often you spend a lot more time, money, and effort getting a waiver to this administrative error (which is what it is when you are forced to use a document or a standard incorrectly), than if you just were doing the basic engineering. This is going to come out, I am sure, in our discussion. There will be a panel tomorrow afternoon, chaired by J. L. Kovach, that is strictly on gas processing. I recommend both Kovach's paper and the panel as they will, to a large extent, be a continuation of what we will be talking about here.

As a point of interest, the Department of Defense started using the N-509 and N-510 standards some years ago for some of their applications. They have been using nuclear industry technology on an *ad hoc* basis for the last fifteen or twenty years for air filtration of the safeguard missile silos at Cheyenne Mountain NORAD Defense Center, and in a number of other areas. I have a friend here who has been involved in the filter systems at Johnston Island for war gas demilitarization, and I am sure he could give some examples of the problems that arise when using a limited nuclear standard for other applications, and trying to do it to the letter rather than to the spirit of the standards.

One of the big differences between a DOE facility and an NRC regulated facility is that in an NRC regulated facility for light water reactors you will not see megacurie doses of anything except noble gases. You are not going to see chemical processing lines in power reactors. In DOE facilities these are chemical engineering processes. For the NRC regulated facilities that are designed to mitigate accidents, you design for a short-term use. These systems don't have to run for a long time. People may differ over the definition of an adequate source term, but the chemistry of a nuclear accident is somewhat better defined than the extreme range of conditions you can get in processing at DOE facilities. Long term versus short term requirements have not been discussed very much lately, but they are going to be discussed in some detail during tomorrow

afternoon's session.

The misapplication of a standard or code, any code or standard, is a very bad thing. The idea of using existing technology, which is why DOE is recommending the use of what the NRC has been using, is great, and when properly applied, could save everybody a lot of time or money. When we say DOE, we are talking about our time and money as taxpayers. Therefore, I think we should all be very much in favor of using existing technology. I certainly am. But when you misapply technologies, you waste a lot of time and money. So I am hoping that the discussion today, the panel tomorrow afternoon, and a number of formal papers will bring to light all of the problems. Their solution is simply to apply the spirit of the guidelines, rather than the blind administrative letter of the various documents.

The last point I want to make before we start the panel discussion is that ASME's Committee on Nuclear Air and Gas Treatment, CONAGT, is responsible for N-509 and N-510, plus a new code, which is much broader, called AG-1. AG-1 has not yet been approved for use by DOE. By simply doing that, it would eliminate quite a few of the current problems. CONAGT discussed this whole area last week and came to the conclusion that if we are going to have AG-1 used by DOE, we will need new CONAGT members who have the appropriate expertise. If anyone here has expertise in these areas, please see me or stop at either the ISNATT booth or the CONAGT booth and offer your help. Now, I would like to invite everyone to bring up whatever questions or comments they wish, and we will do our best to, at least, discuss the points, even if we can not answer them directly.

WRIGHT: We have been hearing about all the problems that we are facing and how standards are inadequate. It seems to me that standards that are adequate for the NRC may not be as adequate for the DOE, and the standard that will be written may or may not be different: I can't tell at this time. But here is a question for the panel. Can you, within a reasonable time frame, write a standard that makes sense for facilities that are not reactors, so that we can put this on the table and deal with it? Some of the facilities may be more dangerous than reactors, but most will be relatively benign, having an element of risk three, four, or five orders of magnitude below what you expect in a reactor. Within the DOE context, you have something called the S/RID, which is a process of identifying all of the applicable requirements for a facility, at which point you can call on a standard to do a job. However, you do not have to call on the standard in its entirety, you can call on those sections that make sense. Once it's approved, they become part of the basis for operating the facility. You also have at your disposal, on the DOE side, hazard categorization, which is the closest thing that we have to the Part 30, Part 50, Part 70 facilities in the NRC. You can write your standard in such a way that, when you are a hazard category 3, or below, you can invoke a specific level of a standard. But it seems to me that there are some solutions we can take, especially with a commitment to fix some of the DOE orders that are causing us all these headaches. And so, coming back to the essential question, can you achieve consensus and write a standard that makes sense, does the job, and does not put an onerous burden on the people that have to live under these standards?

SHERWOOD: I guess the answer is, I think so. I will be working on one soon. I have

already talked to the DOE standards writers, and you are more than welcome to join with us in that endeavor. We could use your help, to help us get consensus. I think we can write a standard fairly rapidly. The approval process at DOE, sometimes, is a little less rapid, and that might be the problem. But if we can get something in writing and get people responding to it, we are better off than we are today. And that's where I will be working after I get back to the office in Germantown.

JACOX: I would like to answer that from the standpoint of CONAGT. As I said, we discussed it, and, depending on what your definition of a relatively short time is, the answer is definitely, yes. And I agree that one of the most critical things in the use of any code or standard is that the people using it should invoke only the applicable parts, rather than invoking the whole standard. That is a serious problem in NRC-licensed facilities, when there is only one section that is really needed, but somebody invokes the whole standard. At some point you end up asking why we do all those things to ourselves. If the DOE is writing standards, it is very important that there will be some discussion. And I am not sure who the individuals will be. There should be good communication so that both CONAGT and DOE are not trying to do the same thing, or worse, trying to do the same thing in different directions. Ray Weidler (chairman of CONAGT), perhaps you could talk to the appropriate people, if you are not already doing it, to see that we get everybody moving in the same direction and most efficiently.

WEIDLER: As Jack Jacox has said, CONAGT is looking very seriously at expanding both AG-1, N-509, and N-510 to include DOE and DOD items. The definition of "a reasonably good time frame" is difficult because of the consensus process, and the fact that the Board of Nuclear Codes and Standards governs us. They, in turn, are governed by ANSI. So a code or standard requires approval of the CONAGT main committee, approval of the Board of Nuclear Codes and Standards, and then approval by ANSI. This can take up to five years. I will say that CONAGT is interested in expediting the process by getting these issues on the table and getting the codes and standards upgraded to address all DOD and DOE needs. We will be making a strong effort in the next year to get some writing effort under way.

KOVACH, L.: My full time affiliation is NUCON, International, but I will add DOE high level waste Technical Advisory Panel, as well, because I will be speaking with a forked tongue here. ANSI N-509 and N-510 were written because some of the architect-engineers were exercising all of their ingenuity. Originally, we had single reactors that had four or five different sizes of carbon adsorbers at the same site. So, it is not only that a standard is written to restrict people or to try to correct past mistakes, sometimes, it is written because of the horror of looking at existing systems. I have not seen a better installation average at DOE sites than at NRC regulated sites. There are some total abominations out there, installations that no decent architect-engineer in his sane mind could stand up and say that this is a good design, or this is an installation appropriate for the particular challenge. At the same time, we are looking at horrendous expenditures. We learned of one tank that was remediated at a cost of about $1.6 billion dollars, but there are more than 240 more out there. I never believe the budget that Westinghouse or DOE present. I always use a great multiplier on both the time and the money. But we cannot afford to have a separate design at West Valley, a separate design at Savannah River, a separate design at Hanford, a separate design at Rocky Flats, and all the other

places. We have to get to the point that anything that can be put on a common denominator will, in fact, be made uniform at all of the DOE sites, and that future construction, operation, and testing of air cleaning systems will be as uniform as possible, although some portion may have to be customized for certain applications. We are looking at air cleaning systems for processing that will be under continuous chemical and radioactive challenge for possibly twenty years. I hope that the melters will work that long. So far, we have spent more money on designing melters than on building them. It is a compromise, and within limits, I disagree with part of almost everything that is said, while I agree with some other parts. Regardless of how experienced the architect-engineers are, we are faced with some new processing problems that have been solved in very few places in the world. And we have to make sure that the common parts are standardized. For that I like to refer to the French approach because they certainly are more successful with their entire nuclear industry than we are. When you go into one of their reactors you know exactly where the air cleaning system is; it's the same size for all, and it's in the same place. The doors are the same, the components are the same. We will have certain processing equipment or area protection systems that will need to be identical, whereas some will need to be modified and good engineering judgement will be needed to back fit existing systems. It has to be done on a cost-benefit basis. Somebody should look at the uniformity of back fits, also, and not have a different back fit procedure at Hanford and a different back fit procedure at all the other sites. I started out in this business when it was still AEC operational safety for both agencies. At that time, several people, vendors, engineers, users, would sit down and come up with a common denominator approach. That was good at that time, but I don't think it will work ever again. Even though some past things were more successful than some of the things we are doing now, we can't repeat them. The consensus process for standards that we have now is very long. I am not sure we can come up with appropriate design criteria in time, particularly for Hanford, where they are now frantically trying to design a giant monolithic processing system. We cannot afford to leave another nuclear standard "memorial" to future generations. At the same time, if we all do something different, it will also create tremendous problems. So, my question to the panel is: what are the areas where we can achieve important uniformity, either by scissoring or mending current standards, and not let everybody go wild with their innovativeness at the particular sites? And which are the areas where we can afford to be a little more lenient, but make sure that an evaluation is made to justify deviation from the standards? But just shrugging off the problem by saying the standard was written for NRC, and therefore we have to drop the whole thing and start strictly on our own, is again, something that I don't think we can afford. There are a lot of good points in the current standards and I think we have to come to a compromise here. Some part has to be a rigid standard, but for some parts, there can be leeway when justified by engineering judgement.

SHERWOOD: One area that could be standardized a bit, but where there still could be some room for local modification, would be sampling itself. A lot of the sites with old facilities really cannot test according to all the requirements in the existing standards. There is not enough space and they can't get to the locations. The filters may have been installed too close to one another, and short of rebuilding the whole thing, you have to be able to modify testing to cover that case. Another one would be air flow. Some of the facilities have had drastic changes in air flow, and yet the same HEPA filters are still in place, performing for something like one third the load they used to get, in terms of

air flow or potential contamination. One area I would like to see more standardized within DOE is the requirement that the HEPA filters being used be pretested prior to installation. This is one place where DOE does more than others and it is well worth putting it into an absolute requirement, because in the long run you probably save more than the cost of testing.

MOKLER: An area that I think will require more flexibility, especially for older installations, is multiple-stage systems. Too often, it appears the goal is to test the individual banks, not to get accurate measurements of their performance. I believe there is a need for more flexibility in thinking about how we test multiple-stage systems and in describing how that test will differ from a standard test performed on a well-distributed, single bank system.

ARNDT: I also think that problems can be avoided when we start designing the systems. For example, during a conceptual design, or prior to when the FDC is being formulated, portions of the FDC may be able to stipulate certain sections of the codes that are, or are not, applicable, because then the proposal goes to DOE for approval. If DOE buys into it at that time, you are home free. But problems come in the FDC stage, when they put a blanket statement, in that all systems shall comply to ASME N-509 or N-510. The design agency takes it and may, or may not, in their judgement, feel that it is necessary. Then, at the 30% or 50% design review, someone else will start looking at quality assurance, safety, design engineering, and see a system that does not meet the letter of the law. This is, I think, when problems occur, whereas they could have been avoided in the initial stages. Now, because of cost and project scheduling, problems are encountered. But for a lot of existing systems, it is hard to standardize, because you have to retrofit. I feel that a lot of problems can be avoided if the appropriate people are brought in at the initial stages, at the design concept.

GRAVES: There is one tremendous advantage to using a standard even when you don't like to, and that is to use a standard as a starting point. While it is evident to me that some DOE facilities are not happy having to live with N-509 and N-510, they have the opportunity to avoid some of the bad things that they may have done in the past; certainly with air filtration. So, while N-509 and N-510 just do not fit, and it is somebody else's standard, it still is a tremendous advantage even when you feel that you have to take exception, chapter and verse. One of the problems in the light water nuclear power industry is that many mistakes get repeated until somebody latches onto the standard and finds a good design basis.

JACOX: In answer to some of Dr. Kovach's comments, I hope I didn't give the impression that I was saying that DOE should not use the standards. My point is that they shouldn't use them blindly, that is where the problem occurs. Even when we are talking about very similar sorts of applications, a requirement to meet all of the "to-the-letter type QA rigorousness" is the real problem. At least, I would like to think, having worked on them for nearly twenty years, that there is a lot of good technical content in the standards. However, given different applications, different source terms, and different situations, they should be used for technical guidance. As I mentioned earlier, in DOE there are chemical processing applications to which N-509, and even AG-1, the expanded code, simply do not apply, and, as guidance, there are certain large gaps. So, we do need some

kind of expansion. One of the points that Ray Weidler made was that it might take five years to get something approved. Having been involved with standard writing, I am painfully aware of that. Something other than a code or a standard that could be used by DOE, whether written by CONAGT, or by CONAGT in conjunction with DOE, or by DOE itself without CONAGT, would be of immense help in the application of good technical judgement based on experience and past mistakes. A bit of uniformity could result in a big saving.

WILLIAMS: I am with Parsons. We do decontamination and dismantlement. A lot of agencies, as well as DOE, call it decontamination and decommissioning, which essentially means the same thing. I had a question for George Sherwood. When you say you would like to have the HEPA filter pretest done, are you referring to the military standard, DOE's NEF 3-35, or the work done at Oak Ridge or Rocky Flats?

SHERWOOD: I am referring to DOE's HEPA Filter Test Stations that pretest HEPA filters before they are installed at DOE installations, particularly those at Oak Ridge and Rocky Flats, where I have visited and seen how they do the testing. I am satisfied they are doing a good job.

WILLIAMS: I agree with you totally. What we have run into, in the work we are doing at Fernald, is that we are an environmental remediation management contractor. We get paid faster and make more money if we get the job done quicker. When we have to send HEPA filters to Oak Ridge (Rocky Flats is not in our jurisdiction), it slows down the process. I agree with you that testing needs to be done at some centralized location, but in the case where we have to send things from our vendors, who are also doing tests to be sure that their products comply with the standards, then to Oak Ridge, and then on to Fernald, that slows down our lead time a lot.

SHERWOOD: Some of the DOE facilities that I have been to, for that very reason, have some pretested filters stored on site for immediate use.

WILLIAMS: That is another problem; we only buy what we need, and usually we require the subcontractor to buy them. We are no longer in the processing mode, we store nothing. Everything is subcontracted out. We, as a design firm, are the architecture-engineering firm. FERMCO is the managing contractor. We write performance specifications (sometimes hybrid designs/performance specifications and drawings) that the subcontractor utilizes to produce detailed work plans, drawings, or specifications. Nevertheless, we just don't have time to send HEPA filters from the manufacturer to Oak Ridge. We need to get them to the site as quickly as possible so we can get construction started.

SHERWOOD: You might want to look at something on the standardization issue that I did not bring up previously: I think we ought to take a real hard look at which HEPA filters need to be tested in-place and which ones do not. There is some variation among the DOE sites today on that very matter. I think everybody agrees that the filters on the main exhaust plenums need to be tested. But not everybody is testing prefilters, not everybody is testing intake filters. There are certain filters that may not require testing. Or filters may be in certain types of service, where it is not clear they are needed;

perhaps they were installed for an older system twenty years ago and they are still in place and still being used, but they are no longer appropriate. Having them in place and testing them does not make a lot of sense. We need to get over that kind of a hurdle. In your particular case, you might want to talk to the folks at Mound, who keep a fairly large stockpile.

WILLIAMS: That is something we will look into. As a matter of fact, I have some contacts up in Mound.

SHERWOOD: They are storing both tested and untested filters for use. They have a large stockpile of both. You might want to get in touch with them to get around some of your time constraints.

WILLIAMS: There is one more thing. At Fernald we are in a very public-sensitive area. If we deviate from the standards, we have no defense from a legal standpoint. That presents a major problem. In our specifications, we are in the process of pointing out the sentence, or the section number and the sentence number, that we need to observe to be in compliance with applicable standards, instead of trying to be in compliance with the whole document. That is where we are now. We prefer not to deviate from standards, because of problems we have had in the past. If you are familiar with Fernald's history, I think you would agree.

NEWTON: I am an aerosol scientist at DOE's Inhalation Toxicology Research Institute. I have a suggestion for people considering new standards. I don't think you have made enough use of the fact that we have radon progeny everywhere. I was struck recently, after I made it to Sandia National Laboratories, that we had one picocurie per liter of radon present in the upstream stack. Downstream of the HEPA filter it was $1.4x10^{-4}$ the upstream concentration. So, here we have a naturally occurring alpha emitting radionuclide that we can monitor on either side of a HEPA filter and find out when it starts to become degraded, or whether we have a catastrophic failure. That is my suggestion. People should use the presence of radon progeny to monitor the degradation of HEPA filters.

MOKLER: The measurement of radon progeny is an interesting concept as an alternative means for in-place testing of HEPA filters. However, I am not aware of any data comparing this approach to the conventional testing methods. A preliminary study to obtain such data is necessary before the utility, possible advantages, and appropriate situations for application of this approach can be evaluated.

GHOSH: You talked about standardizing the design, but most of the plants have already been designed, they have been installed, and they have been operating all this time. What you are really talking about are standardized testing criteria. Some of the filtering units that you are talking about were built prior to N-509. If you had a section in the N-510 to suggest testing criteria for units built prior to N-509, i.e., non-509 systems, that would be a big help.

JACOX: The first page of all three editions states that N-510 is to be rigorously applied only to systems built to specifications of the same edition of N-509, but it may be used

as technical guidance for all other systems. We have not been successful in getting people to read that. At some DOE sites, even when we point it out, they say, "but the administration says it has to be to the letter". It is the intent of the committee that wrote the standard, that it be used for technical guidance for any system, but rigorously applied only to a system that meets all of the requirements of N-509, same edition. What you suggest is an excellent idea. We have done our best, albeit without much success, to do just that.

GHOSH: If you would take it out and put it in a subsection, and title it for non-509 systems, it would stand out more.

JACOX: It is listed under "Limitations".

ARNDT: I'd like to add to that if I may. There are a lot of word engineers, so that when we use certain words, like "suggest" or "shall", they mean things. When we put a word in like "suggest", you can be assured that there will be people that say that you don't have to do it. And now you can have systems that will not be tested, because of using word like suggest. We do a lot of design reviews and you would be surprised at the type of designs that are sent in by agencies. The only time they may have seen N-509 was when it was included in their design packets. So, there are two ends of the spectrum, the "shall", which people say you have to meet according to the letter of the law, and "suggest", where people say, "well, since you used the term suggest, I don't have to do it." I think we need to come somewhere in the middle, and I really don't know at this point how that could be accomplished other than at the start of the project.

GHOSH: You're right, I agree with what you just mentioned. What I suggest is to include some flexibility: as a minimum, to say that "suggest" means you should do it, whereas "shall" means you have to do it. Flexibility is what we are talking about.

ARNDT: I understand. The point I am trying to make is that we have to be careful with the wording because there are people that will look at "shall" and say you have to follow everything written down whereas "suggest" means you don't have to do it.

MOKLER: To rephrase an important point that Tim Arndt made, we must specify the very general things that must be done. The way you accomplish these ends is where flexibility can come in. We have to remember that just as there is a very strict interpretation of things that you shall do, sometimes the permissiveness of a guidance document is used to avoid doing something.

DU BOIS: Our facility was designed before N-509. We are going through all our tech specs right now. It does not take long to go through the N-509 standard line-item-by-line-item to see if the system meets it and to justify the decision technically. I did it on one of our systems. In eight hours, I was able to go through the entire testing standard to see exactly what the system will meet and to document my technical justification. On the regulatory side, you are going to have further review, but people are willing to work with you. That is my first comment. The second one concerns HEPA filter testing at a qualified vendor facility. I am not against that, but you still need to do an in-place leak test because you have people who put these things in who do not know the fragility of

these filters. We had a case where a HEPA filter system we had just put in, failed. When I discussed it with the particular individuals who were installing it, someone dropped a filter and cracked it. You have to make sure that even though you do pretesting, you do an in-place leak test before you start the system.

MOKLER: For the last thirty years, the in-place test and the filter test facility test have been looked upon as adjuncts to each other. No one that I have been aware of has ever felt that one could stop doing in-place testing if filters were prequalified. You must do both. An important question, however, is the appropriate frequency for in-place testing.

JACOX: I have a question for someone from DOE. At least one DOE facility that I am personally aware of has taken the point of view that N-509 and N-510 must be used to the letter for all systems. Arndt's paper covered this in some detail with the exact phrasing. Although he did not discuss it, I understand that unless something is explicitly discussed and covered in N-509, it cannot be used, even when the document says that other designs that meet the basic criteria can be used; that there is an administrative decision which states that you can not do it. What are the chances that this DOE order can be revised to remedy this situation? I consider current practice to be a gross misinterpretation of the intent of the document that causes severe engineering and cost problems.

ANON: I can't answer that question, but I can get the question to people who can. Ask for an interpretation in writing from EA64, it used to be the old NE group. They just had a change in leadership, and that actually should make things easier. But it is Neil Goldenberg's old organization, safety policy. Put the question to them in writing, state the concern, and they are obligated to respond.

JACOX: To clarify that, did you say you would do that or we should do that?

ANON: You should probably do it, but if you need somebody to facilitate it, I shall.

JACOX: I am sure we will take you up on that.

ANON: I do not have any first hand information on the situation at Hanford. I have heard that they won't let you use anything that is not called out in the standards, such as new methods of testing filters. But I don't know that for a fact. Before I write a letter, I would like to take some people there and take a hard look at their overall testing program, and then make some recommendations.

KOVACH, L.: I looked recently at one of the Hanford designs, and I had a chance to look at the budget also. The system was slightly over ten million dollars. The same system built to N-509 for a power reactor would cost less than one million. In my mind, nine million was for customizing N-509 at Hanford, and that was the reason why the cost went up to ten million dollars. On a cost-benefit basis it is very difficult to justify going away from standardized designs, at least as far as the budget figures that are in the official documents are indicative. I can state with great confidence that they are an order of magnitude higher than anything I have seen for any other "very rigid application" air cleaning systems. In my mind it means that they are not going to be standard systems;

they will be N-509 systems except for something like this or something like that. I am not picking on Hanford for any particular reason, but I have recently looked at more of them than at Savannah River or other sites. My expectation would be that if they standardize, costs should be significantly lower, rather than higher. This puzzles me greatly, and I do not know if anybody else wants to comment.

JACOX: I have spent some weeks at Hanford on initial engineering design discussions for filter systems that will be N-509 systems with some additions because they will include some special process parameters. However, from 50% to 80% of the system will be a basic N-509 designed system. When I learned of the budget figures, I just couldn't believe them, they were incredibly high. And what could easily be done in eighteen months, they were hoping, we were told, to be able to finish by the end of the century. So there is something seriously wrong with the way this project is approached. It is what I would call mindless bureaucracy and politics. There are too many people involved that have to be stroked and satisfied. That is how an eighteen month, million dollar project becomes a six year, forty million dollar project. You end up with the same hardware doing the same thing in the same way. That scares me as a taxpayer, it upsets me as an engineer. I guess it does not upset me too much, because I still take the consulting money. There is something basically wrong when every attempt to standardize and use existing technology, to the extent that it is applicable, ends up with grossly longer schedules and at least an order of magnitude more money.

ARNDT: I would like to say something on that also. Westinghouse has tried to standardize designs on a lot of their projects, and I think you are familiar with one of them. Some of the problems they encountered involve design criteria when they are trying to determine the constituents in tanks and gas streams. It becomes difficult when characterization is unavailable and they have a very difficult time getting access to the tanks. With regard to air cleaning systems, they are trying to standardize how many stages to put on, one, two, or three; should they use charcoal adsorbers; do they need charcoal adsorbers; should they put deentrainers in, etc. I think they are trying to get the right answer, but are not there yet. I know it is the goal, but at times it is very frustrating because the criteria are just not available.

JACOX: I agree that what you say is factual. My feeling is that the concern about characterization is overemphasized. If you assume the worst and then design and build the system for the worst, you will save a tremendous amount of money over the millions and millions of dollars that are being spent on fine detail characterization. At one meeting, we were given limits, and told to use engineering judgement. At another meeting, we were given some limits on some particulate constituents, and a limit for a gaseous chemical constituent that was at least an order of magnitude higher than is physically possible for such a concentration to exist at the assumed temperature and pressure. One of the operational people said, "if we lose that much, my tank's empty in three weeks." So characterization is being approached in ridiculous ways. There may need to be detailed characterization for chemical processing, but for air and gas treatment you can say we may have this much and simply put in a filter. Isn't this better than spending millions of dollars to find out whether or not something is there? You do not have to know exactly what is in a stream to remove it, you merely need to know the range of concentration. When you look at the range of likely concentrations, it

represents a basic reality check. We get so caught up in the bureaucracy that we don't get an opportunity to use good engineering judgement. These were a couple of one week sessions where there were on-site people, people brought in from other DOE facilities, and independent consultants such as myself. I don't know what the meetings cost overall, but just on a couple of very basic points I think that they were more than paid for by the advice given. On one hand, I seem to be saying let's put a component, or a couple components, in to take care of things "just in case," signifying that I don't care about great detail in characterization. On the other hand, I seem to be saying, let's be more careful with the characterization. However, I think they are consistent positions because if we say we are not sure whether a particular species is present, but it is going to cost three million dollars to find out, whereas one or two standard components can handle the species at a cost of a hundred thousand dollars, installing the equipment is a real cost-effective solution.

SESSION 4

NUCLEAR WASTE

Monday: July 25, 1994
Co-Chairmen: R. Porco
W. Bergman

OPENING COMMENTS OF SESSION CO-CHAIRMAN PORCO

GENERATION AND RELEASE OF RADIOACTIVE GASES IN LLW DISPOSAL FACILITIES
M. S. Yim, S. A. Simonson

NUMERICAL ANALYSIS OF A NATURAL CONVECTION COOLING SYSTEM FOR RADIOACTIVE CANISTERS STORAGE
R. J. Tsal, S. Anwar, M. G. Mercado

DISPOSAL OF SLIGHTLY CONTAMINATED RADIOACTIVE WASTES FROM NUCLEAR POWER PLANTS
J. L. Minns

CLOSING COMMENTS OF SESSION CO-CHAIRMAN BERGMAN

OPENING COMMENTS OF SESSION CO-CHAIRMAN PORCO

This session is on nuclear waste. I think one of the most pertinent comments about nuclear waste made by Admiral Guimond this morning was that most of the DOE budget is for nuclear waste and nuclear cleanup, and that the largest environmental protection program in the world is the DOE projects. I think that sets the tone for the significance of this nuclear waste session.

GENERATION AND RELEASE OF RADIOACTIVE GASES IN LLW DISPOSAL FACILITIES

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Abstract

The atmospheric release of radioactive gases from a generic engineered LLW disposal facility and its radiological impacts were examined. To quantify the generation of radioactive gases, detailed characterization of source inventory for carbon-14, tritium, iodine-129, krypton-85, and radon-222, was performed in terms of their activity concentrations; their distribution within different waste classes, waste forms and containers; and their subsequent availability for release in volatile or gaseous form. The generation of gases was investigated for the processes of microbial activity, radiolysis, and corrosion of waste containers and metallic components in wastes. The release of radionuclides within these gases to the atmosphere was analyzed under the influence of atmospheric pressure changes.

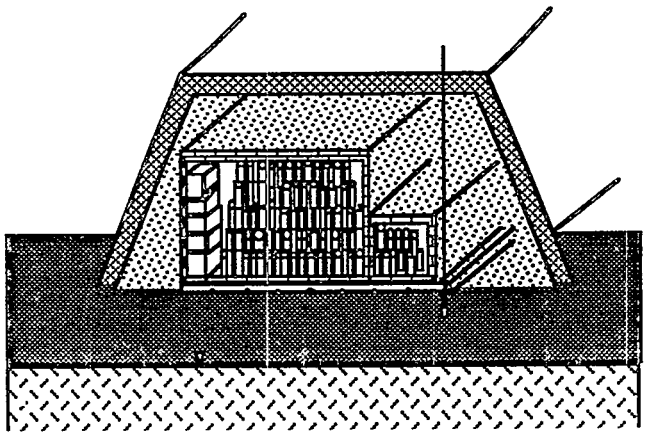
1. Introduction

Within LLW disposal facilities, contaminated gases may be introduced into the containment structure as a result of the failure of disposal canisters, anaerobic corrosion of steel containers or the decomposition of organic wastes[1,2].The presence of contaminated gases in the engineered barrier structure may result in radionuclides entering the atmosphere by diffusion or other mechanisms through the containment structure and soil cover. Because some amount of gaseous and/or organic waste is expected to exist in all disposal facilities, the gas-phase release of radionuclides needs to be characterized.

Due to the public's unfavorable view of the shallow land burial of LLW, a major share of the LLW disposal facilities being planned by the States and state compacts are to be above- or near grade bunkered facilities. According to the current design of certain of these facilities, there is a flow drain in the engineered concrete structure and a standpipe, for monitoring, which is connected to the flow drain. This feature potentially provides a direct pathway for gases to be transported out to the atmosphere from within the facility (Figure 1). If a significant amount of radioactive material can be incorporated into the gas-phase, it may be available for releases at rates far greater than would be normally experienced through the liquid (groundwater) pathway. For the overall performance assessment of the newly proposed LLW disposal facilities, the potential radiological dose from the airborne

release of radioactive materials through the drain system and monitoring well needs to be assessed. In this case, a major factor that is known to affect the transport and release of generated gases within a LLW disposal facility is the atmospheric pressure variation.

Figure 1. Schematic of an Earth Mounded Concrete Bunkered LLW Disposal Facility



2. Source Inventory Characterization for Gas-Phase Releases

Quantifying the generation of radioactive gases within a LLW disposal facility and the radiological impacts from the release of such gases requires understanding of the gaseous source term, i.e., the total amount and rate of gaseous radionuclide generation. This source term is influenced by the radionuclides inventory within various LLW streams and the waste forms and containers used to dispose of the inventory. In this chapter, summarized are data on these characteristics of specific radionuclides of concern along with the discussion of their subsequent availability for release to the atmosphere in volatile or gaseous form. Radionuclides of concern are ^{14}C, ^{3}H, ^{222}Rn, ^{85}Kr, and ^{129}I.

2.1 Radionuclide Concentrations in LLW

Radionuclide concentrations in various LLW streams had been estimated by several studies[3,4,5]. However, the information given in these estimates was not very useful for the purpose of this study for several reasons: No information was given on the distribution of activity according to waste class (A, B, & C) which precludes the estimation of activity distributions in different waste containers; the waste stream classifications were different from the ones used in LLW shipping manifests, and; no estimates were given for the concentration of ^{85}Kr and ^{226}Ra (precursor of ^{222}Rn).

For these reasons, it was necessary to develop new estimates of activity concentrations for the radionuclides of concern in a class-specific manner for various types of LLW generated from various sources. These estimates were made based on the information given in the U.S. Nuclear Regulatory Commission report,

Table 1 Activity Distributions (%) of Radionuclides in LLW

Radionuclides	Carbon-14			Tritium			Iodine -129			Krypton-85			Radium-226		
Class	A	B	C	A	B	C	A	B	C	A	B	C	A	B	C
Dry Solids	25.6	0.26	19.4	3.24	89.3	0.03	5.0	0	6.14	99.2	0	0	19.3	.001	67.2
Solidified Liquids	5.76	0	1.2	1.45	2.51	0	0.07	0	0	0.02	0	0	.001	0	0
Noncarcass Biological Wastes	0.12	0	0	0.02	0	0	0	0	0	0	0	0	0	0	0
Dewatered Resins	7.54	0.96	7.37	0.12	0.02	.002	41.0	5.46	12.8	0	0	0	0	0	0
Solidified Resins	0.30	2.30	0	0.03	0.02	0	0.29	0.19	0	0	0	0	0	0	0
Sorbed Aqueous Liquids	7.74	0	0	0.68	0	0	2.25	0	0	0.01	0	0	0	0	0
Sorbed Nonaqueous Liquids	.003	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Aqueous Liquids in Vials	2.10	0	0	0.06	0	0	0.01	0	0	0	0	0	0	0	0
Animal Carcasses	1.52	0	0	0.19	0	0	0	0	0	0	0	0	0	0	0
Evaporator Bottoms	0.57	0	0	0.21	0	0	3.83	0	0	0	0	0	0	0	0
Compacted Dry Active Wastes	7.69	0	0	0.26	0	0	13.9	0	5.41	0.02	0	0	0.93	0	0
Noncompacted Dry Active Wastes	0.50	0	0.16	0.07	0	.005	0.86	0	0	0.37	0	0	11.8	0	0
Cartridge-type Filter Media	1.37	0.21	2.83	0.03	0	.006	0.23	0	0.95	0	0	0	0	0	0
Noncartridge-type Filter Media	0.74	0.59	0	0.03	.006	0	0	0.05	0	0	0	0	0	0	0
Activated Reactor Hardware	0	0	2.92	0	0	1.68	0	0	0	0	0	0	0	0	0
Solidified Chelates	0.14	0	0	.001	0	0	0.06	0	0	0	0	0	0	0	0
Solidified Oils	0.16	0	0	.003	0	0	1.43	0	0	0	0	0	0	0	0
Gas	0	0	0	0	0	0	0	0	0	0.41	0	0	.003	0	0
Nonaqueous Liquids in Vials	0	0	0	0	0	0	0	0	0	0	0	0	.001	0	0
Subtotal	**61.9**	**4.3**	**33.8**	**6.4**	**91.9**	**1.7**	**69.0**	**5.7**	**25.3**	**100**	**0**	**0**	**32.8**	**.001**	**67.2**

NUREG-1418[6]. The details of this estimation and the results are given in the reference by Yim[7]. The estimates were made for 19 different types of waste (see Table 1) from 5 different LLW generators (utilities, industry, colleges, hospitals, and government).

2.2 Activity Distributions in Various Types of LLW

Based on the estimated concentrations, the distributions of the total activities of ^{14}C, ^{3}H, ^{129}I, ^{85}Kr, and ^{226}Ra in typical U.S. LLW streams were estimated as given in Table 1. For ^{14}C, 62 % of the activity was estimated to be in Class A, 34 % was estimated to be in Class C, and the remaining 4 % was estimated to be in Class B. Among various types of wastes, 45 % of the total ^{14}C activity was estimated to remain in "dry solids", followed by "dewatered resins" (16 %), "sorbed aqueous liquids" (7.7 %), "compacted dry active wastes" (7.7 %), "solidified liquids" (7.0 %), and "cartridge-type filter media" (4.4 %). The remainder (12 %) was in various types of other wastes.

For ^{3}H, most of the activity (about 90 %) was estimated to be in Class B as "dry solids". In the case of ^{129}I, 69 % of the activity was in Class A, followed by Class C (25 %) and Class B (6 %). Among various types of wastes, 60 % of the total ^{129}I activity was estimated to remain in "dewatered resins", followed by "compacted dry active wastes" (19 %) and "dry solids" (11 %). In the case of ^{85}Kr, most of all activity (99 %)

Table 2 Activity Distributions (%) of Radionuclides in Different Waste Containers and Waste Forms (a): Envirostone

	Carbon Steel Drums						Carbon Steel Liners						High Integrity Containers					
	Ce-ment	Sor-bent	None	Bitu-men	E stone (a)	Blank/ Other	Ce-ment	Sor-bent	None	Bitu-men	E stone (a)	Blank / Other	Ce-ment	Sor-bent	None	Bitu-men	E stone (a)	Blank / Other
C-14	5.1	9.7	27.4	0.2	0.2	7.1	1.5	2.4	6.9	0.04	0.06	1.9	22.5	0.04	4.2	0	2.2	8.8
H-3	1.2	0.9	2.8	0.02	0.02	0.2	0.3	0.2	1.6	.004	.005	0.04	2.6	0.03	90.0	0	.005	0.01
I-129	0.4	3.7	47.7	2.9	0.04	0.4	0.1	1.0	12.2	0.7	0.01	0.1	0.5	6.7	23.3	0	0	0.2
Kr-85	0	19.1	60.9	0	0	0	0	4.8	15.2	0	0	0	0	0	0	0	0	0
Ra226	1.5	1.1	23.3	0	0	0	1.1	0.3	5.5	0	0.02	0	67.2	0	0	0	0	0

was estimated to remain in Class A "dry solids". For ^{226}Ra, "dry solids" contain 87% of the activity, followed by "noncompacted dry active wastes" (12 %). About 67 % of the total ^{226}Ra activity was estimated to be in Class C wastes, with the remainder being in Class A.

2.3 Inventory for Gaseous Releases

In modeling the performance of engineered barriers in relation to the release of radionuclides, it is necessary to know what types of waste containers and forms are used and how much of each radionuclide is disposed in these containers or forms.

Detailed characterization of these types of information have been performed[7] and the results are shown in Table 2. According to the results presented in Table 2, most of the activity disposed, except for ^{226}Ra and ^{14}C, was in the "none required" category (however, for ^{3}H these "none required" waste forms were mostly disposed in special packages thus are not easily subject to release).

In the case of ^{14}C, about 29 % of the activity was reported to be buried in a "cement" waste form, with the majority (39 %) being estimated to be buried as the "none required" waste form. About 12 % of the activity was estimated to be stored in "sorbents". For about 18 % of the total ^{14}C activity disposed, no proper information of waste form was given.

Based on current container design characteristics and anticipated lifetimes, existing waste containers do not provide any significant barrier for releases in the gas-phase[7]. Even though high integrity containers (HICs) are designed to maintain structural stability for at least 300 years, the design requirement for the presence of a passive gas vent provides a direct release path for the gaseous radionuclides. This vent is designed with a charcoal filter in it and provides relief against pressure buildup. It is therefore permeable to air or water vapor flow[9].

The organic fraction of LLW will have the potential to degrade by microbially mediated processes to produce gas comprising a mixture of CH_4 and CO_2. About 31% of the total ^{14}C activity was estimated to exist in organic compounds and most of this is in Class A waste[7]. Among this 31%, roughly 39% resides in a cement waste

form (including Envirostone) in which the pH will remain high*. Excluding this fraction, about 20% of the total ^{14}C activity has the potential for gas-phase release through microbial attack. The estimate of 20% of the ^{14}C inventory to be biodegradable is a significant number in terms of its implications in overall LLW performance assessments. EPA has previously estimated[10] this biodegradable fraction to be about two-thirds of the total inventory, suggesting the possibility of a significant reduction in the ^{14}C inventory in the waste disposal facility through gaseous releases. However, EPA's estimate was based on a crude characterization of ^{14}C inventory in various waste streams.

Tritiated organic compounds have the potential to degrade by microbially mediated processes under anaerobic conditions. In this case, the generated gas, methane, will contain tritium. About 50% of the total ^{3}H activity was estimated to exist in organic compounds, mostly in Class B waste[7].

In the case of ^{129}I, about 30% of the activity which is in HICs may be subject to radiolysis, even though the radiation level may not be high enough to produce any significant amount of iodinated gas.

For ^{85}Kr, most of the activity was expected to remain in sealed sources. Considering the half-life of ^{85}Kr and lifetime of sealed containers, gas-phase release may not, as previously mentioned, be a concern for this radionuclide.

All of the ^{222}Rn activity, which reaches secular equilibrium with ^{226}Ra, is available for release as a gas. The actual amount of release will depend on how much time delay takes place for ^{222}Rn within the waste forms or facility before it reaches the atmosphere.

3. Release of Carbon-14

Carbon-14 is a radionuclide of major concern in LLW disposal due to its long half-life ($T_{1/2}$= 5730 years) and high mobility in the biosphere. For the groundwater pathway, ^{14}C is usually one of the most important contributors to the estimated risks. The characteristics of release of ^{14}C activity associated with these gases within a low-level radioactive waste disposal facility were analyzed in this chapter. The effects of radiolysis on the generation of ^{14}C contaminated gases were also described.

Gases that contain ^{14}C can be generated from a LLW disposal facility from either

* The optimum pH value range for methanogenesis is 6.7 to 7.4. Below 6.0 and above 8.0, very little methanogenesis occurs. The use of cement as a matrix would significantly alter the pH conditions in the waste drums. Cement is expected to serve as a pH buffer, the buffering being controlled by reactions between the waste material constituents. If sodium, potassium, calcium and silicon are removed from the contact with water, the pH will gradually decline from about 13. It will stay above 10.5 if, on average, there is more calcium than silicon in the cement's composition. In general, microbial activity virtually ceases above pH 10.5[8].

an aerobic or anaerobic process of degradation of organic components of LLW, depending on the level of oxygen in the surrounding environment. The organic fraction of LLW has the potential to degrade by microbially mediated processes that produce gas composed of a mixture of CH_4 and CO_2, depending on the nature of the waste and the type of waste forms. These microbially-mediated degradation pathways can be generalized into the following two reactions:

1) the aerobic degradation of organic carbon species,

$$(C_6H_{12}O_6)_n + 6nO_2 \xrightarrow{H_2O} 6nCO_2 + 6nH_2O \quad (1)$$

2) the anaerobic degradation of organic carbon species by fermentation

$$(C_6H_{12}O_6)_n \xrightarrow{H_2O} 3nCO_2 + 3nCH_4 \quad (2)$$

The disposal facility selected for the study was a concrete bunkered facility with a floor drain and a monitoring well* as shown in Figure 1. The fluctuations in barometric pressure, which is expected to affect the transport of gas within the facility, were investigated as a major driving mechanism for the release of gases through the monitoring well. To estimate the atmospheric pressure variation, two years of data collected in Boston (March 21, 1991 through March 29, 1993) were processed to produce the average pattern.

3.1 Modeling the Release of ^{14}C Contaminated Gases

The generation of ^{14}C contaminated gases within waste matrices by microbial activity needs to be analyzed along with the attending modeling of oxygen transport and consumption. The consumption of oxygen was calculated using the generation rate of CO_2 according to Equation (1). The transport of oxygen was treated the same as for other gases as described below.

Once ^{14}C containing gases are generated, they can leak out from waste containers and migrate through the disposal facility. Gases can be released from an undisturbed disposal facility through two main migration modes: (1) diffusion as impurities in air and water vapor, and (2) advection of air and water vapor from the disposal facility due to atmospheric pressure variations. For these investigations of gas production and transport, a computer code, called GETAR (Gas Evolution, Transport, and Reaction), was developed[11].

The mathematical model of the computer code, GETAR, includes the time-

* A monitoring well which is connected to the drainage system of the facility provides a conduit for barometric pumping.

dependent effect of gas generation, diffusion, advection, chemical reactions, and radioactive decay for each gas of concern. The resulting space- and time-dependent partial differential equations were solved through the model geometry with appropriate boundary conditions. To solve the model equation numerically, the method of lines has been applied in this study.

The facility modeled in this study was a concrete bunkered facility. It was assumed that the LLW disposal facility site is comprised of 100 cells, each cell with a dimension of 10 (ft) x25 (ft) x125 (ft). Due to the open space volume generated from the emplacement of containers, the actual volume of wastes that can be stored in a facility was assumed to be 54% of the total facility volume. The analysis of gas release was based on the modeling of one cell. The engineered cell facility was one-dimensionally modeled and the analysis tracked the migration of gases from the waste matrix to the atmosphere through a subsurface drain of a generic engineered structure. For the simulation of flow through the drain system, the model accounts for a pea gravel layer at the bottom of the vault structure, the primary drainage system and the cell drain sump monitor standpipe.

3.2 Estimates of ^{14}C Activity Releases

The generation and release of microbially-mediated gases from a LLW disposal facility were simulated using GETAR for a post-closure period. Analyses were performed before and after the assumed occurrence of a break in the waste containers.

For the purpose of modeling it was assumed that the waste containers were carbon steel*, and two scenarios were used: (1) All the waste containers fail from time zero. In this case, oxygen in the facility will immediately start to migrate into the waste matrix (Hereafter called the aerobic case); (2) Waste containers do not fail until after 23 years of service (this being the average estimated life of carbon steel containers). The gas generation for the first 23 years was assumed to occur under the as-sealed condition of the containers without any release of generated gases to the outside of a drum. This would be the most conservative case (the anaerobic case).

For the two scenarios assumed, the ^{14}C activity incorporated into generated gases was predicted by using the GETAR. The ^{14}C concentration within aerobic containers was estimated to approach about 2 pCi/L. In the anaerobic case, the concentration was predicted to approach 1000 pCi/L. These predicted values are found to be in good agreement to the range of values observed in existing or previous LLW facilities: The concentration observed at the Beatty facility was between 1.46 and 25.9 pCi/L[12]; at Maxey Flats, between 0.05 and 80 pCi/L[13]; and at West Valley, between 1,000 and 14,000 pCi/L[14].

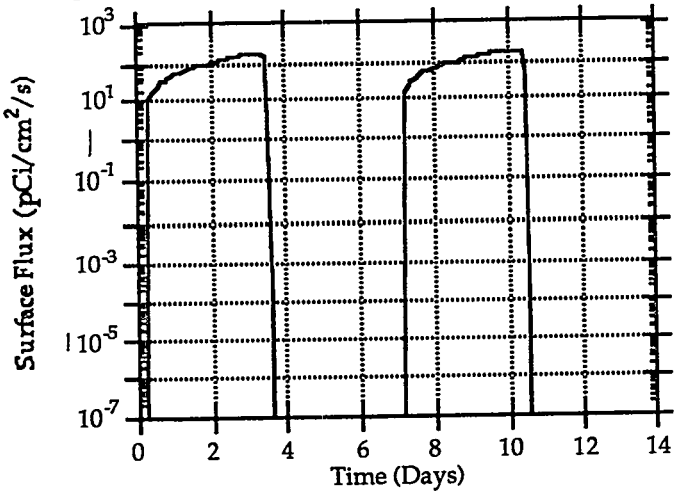
* According to the reference by Yim[7], only 5% of the biodegradable C-14 activity remains in high integrity containers.

To investigate the effects of fluctuations in atmospheric pressure, analyses of the potential release were performed. In describing the flow of gas by advection, the effects of atmospheric pressure variation on flow velocity (so-called atmospheric pressure pumping) was determined *a priori* and utilized in the model: The flow velocity in the open drainage system was calculated based on the ideal gas law considering the total volume of air affected under barometric pressure variation.

Figure 2 shows estimates of changes in the flux of ^{14}C activity at the surface of the soil cover (outside the standpipe) for the anaerobic generation of CO_2. As soon as the pressure change starts pumping out the gases within the facility, the flux increases significantly which was estimated to approach 200 pCi/cm^2/sec. The predicted cumulative release of ^{14}C during the simulated two weeks was about 2.8 mCi from one cell. Adding the contributions from other cells within a facility (100 cells assumed), the yearly estimated release of ^{14}C was about 7.3 Ci.

In the case of aerobic gas generation, the surface flux was estimated to be less than 2 pCi/cm^2/sec. The cumulative release of ^{14}C was about 2.6 μCi in two weeks. For the whole facility, the yearly ^{14}C release would be about 7 mCi.

Figure 2 Predicted Maximum C-14 (CO_2) Surface Flux Changes under Average Barometric Pressure Variations



The results of ^{14}C activity release were calculated based on many parameters in the model. These parameters are subject to different degrees of uncertainties and it is difficult to judge the confidence in the results of activity release without considering the effects of uncertainties. According to an uncertainty analysis for the release of ^{14}C[7], the release of activity varies with a median of 42 mCi. The 95th percentile of the ^{14}C release was estimated to be 141 mCi per cell.

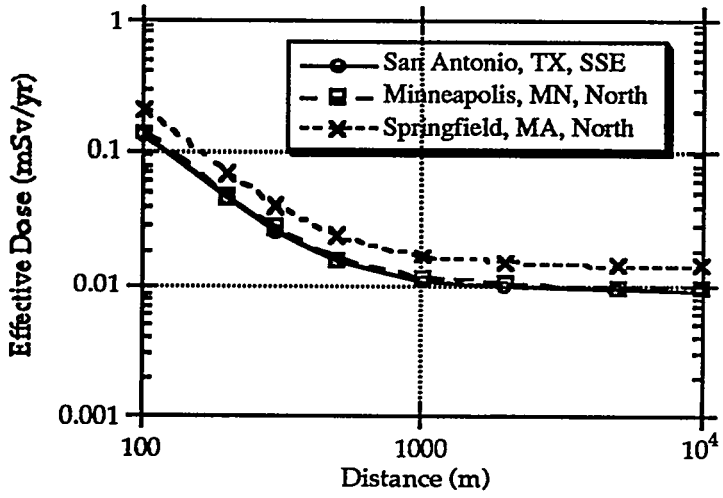
3.3 Radiological Impacts from the Release of ^{14}C

For the radiological assessments of airborne releases of ^{14}C, the CAP88-PC computer code[15] was used in this study. For the assessment of dose, major exposure pathways considered were: (1) inhalation of airborne ^{14}C; and (2) ingestion of ^{14}C which become incorporated into soil and is subsequently taken up by animals and plants. To reduce the uncertainties associated with using hypothetical site data, actual geographical locations were chosen and site specific data were used. Three sites were chosen to represent characteristics in different geographical locations in the U.S. These sites were Springfield, MA for the Northeast, Minneapolis, MN for the Midwest, and San Antonio, TX for the South. The site specific data were used for annual average meteorological conditions (atmospheric stability classes, and wind speeds and frequencies) and agricultural productivity (cattle and crop densities, etc.). For the sources of agricultural food consumed by the exposed individuals, nominal values expected in a rural setting[15] were used.

Figure 3 represents the effective dose estimates for the maximally exposed individual (MEI) due to the release of 7.3 Ci of ^{14}C per year. The 7.3 Ci of yearly ^{14}C release was about the 75th percentile of the estimated release.

According to the results of dose estimates, the ingestion pathway was of primary importance. In fact, inhalation is known as a minor exposure pathway for ^{14}C because dose rate factors of ^{14}C for inhalation are about 1% of those for ingestion[16]. Based on a separate sensitivity analysis, the most important route of exposure was found to be the ingestion of cereals, followed by the ingestion of milk and vegetables.

Figure 3 Dose for the Maximally Exposed Individual from the Release of C-14 (7.3 Ci/yr; Rural setting)



3.4 The Effects of Radiolysis on the Release of ^{14}C

Radiolysis, a decomposition mechanism of compounds by the deposition of energy from ionizing radiation, can lead to the release of ^{14}C. Using the dose rate, 161.5 R/hr, estimated for the class B/C wastes[7], and the composition of organic wastes estimated based on the characterization of LLW, the amount of ^{14}C entering the gas-phase was estimated. The results were about 0.16 mCi as CO_2 and about 0.04 mCi as CH_4 per year within one cell. For the assumed 100 cells within a facility, the total ^{14}C becoming incorporated into the gas-phase by radiolysis would be about 20 mCi per year. These estimates are in the same order as the estimated release of ^{14}C by microbial decomposition of organic wastes under aerobic conditions. According to the dose rate estimates for the release of ^{14}C generated from aerobic microbial activity, the resulting dose rate from the release of 20 mCi of ^{14}C is expected to be negligible.

The G-values used for the radiolysis of organic compounds and the composition of generated gases used are given in the reference by Molecke[17]. The assumed composition of organic wastes in volume was 19% cellulose, 81% plastics and about 0.1 % organic liquids.

4. Release of Tritium and Other Radionuclides

Other than ^{14}C, the radionuclides of concern for gaseous releases from LLW facilities include tritium, ^{222}Rn, ^{129}I, and ^{85}Kr.

The major process of tritiated gas generation is known to be the anaerobic corrosion of metal, and the tritiated water vapor generation and tritiated methane generation from anaerobic decomposition of organic waste. By assuming 100 cells present in one facility and considering all these processes, total ^{3}H activity entering the gas-phase was conservatively estimated to be about 80 Ci per year. The effective dose rate to the MEI from this release was estimated to be below 10^{-2} mSv (1 mrem) per year.

Due to the presence of ^{226}Ra within LLW streams, ^{222}Rn gas will be generated within LLW disposal facilities from the decay of ^{226}Ra. The total yearly ^{222}Rn release from the assumed facility was estimated to be in the order of Ci. The annual effective dose to the MEI from this release was estimated to be less than 10^{-2} mSv .

The release of ^{129}I as a gas from LLW depends on the methods used to capture and store the ^{129}I in the wastes and the mechanism for the release is known to be radiolysis[18,19]. The activity of ^{129}I to be released by radiolysis from a facility is estimated to be less than 2 μCi per year[7]. The resulting radiological dose are expected to be negligible (less than 10^{-4} mSv/year).

For ^{85}Kr, most all of the activity is expected to remain in sealed containers. Considering the half-life of ^{85}Kr and the anticipated lifetimes of the sealed

containers, gas-phase release is expected to be negligible. However, further study on the characterization of sealed containers used to store ^{85}Kr would be desirable.

Acknowledgment

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DISCUSSION

WREN: Was the microbial rate assumed constant in your C-14 model? If there is dependence, what kind of dependence did you include in your model? You briefly mentioned that it depends very strongly on the hydrolysis of organics, and that it may be very sensitive to the temperature as well.

YIM: Our carbon-14 model is based on a diaphasic description of aerobic and anaerobic digestion. The key parameter is the rate of hydrolysis, which is dependant on the type of organic wastes and temperature. Also there is a lot of data existing currently from civil engineers and environmental engineers who are trying to figure out what is going on with municipal waste dump sites. Some of their efforts are directed to producing biogases for fuel. Therefore, they want to maximize the gas generation and they maintain high temperature conditions, above 30° C. Our condition at a typical low level nuclear waste facility is 10° C. We tried to extrapolate the number down using the Arrhenius equation, which should be adequate for this purpose. But that needs to be verified through some of the experiments. Currently there are no experimental data to back up the results. That is something that has to be done. The effect of pH could be described in the growth kinetics model of microbial population. However, pH values do not change much in a dry engineered facility. Also in a high pH environment, such as in cement waste forms, microbial activity ceases.

KIKUCHI: C-14 will be mostly adsorbed chemically by concrete, by forming $Ca^{14}CO_3$. Did you consider this effect in your calculation? If so, how?

YIM: Current calculation of C-14 release did not consider the formation of calcium carbonate to be conservative. The computer model is capable of considering chemical reactions in the release pathway, however.

HINTENLANG: Site-specific barometric variations with time resolutions on the scale of hours should be used for dose predictions. In some parts of the country, periodic variations of 12 hr (for example) are superimposed on longer term meteorological patterns. Atmospheric stability classes do not include these small amplitude variations. Wouldn't these affect dose calculations dramatically?

YIM: In this study, two years of measured barometric pressure data (in Boston) were processed to represent the effect of atmospheric pressure pumping on the release. These data had hourly resolution. The release calculations were done for two cases: yearly average and severe storm case. The effects of small amplitude variations are included in the yearly average calculation and were not too important for cumulative yearly releases. Another thing that needs to be taken into account in release calculation is the rate of radioactive gas generation. In other words, even if there is a rapid pressure pumping going on in a facility, the eventual release of radionuclide will be dependant on the inventory of gaseous radionuclides from various generation mechanisms.

NUMERICAL ANALYSIS OF A NATURAL CONVECTION COOLING SYSTEM FOR RADIOACTIVE CANISTERS STORAGE

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ABSTRACT

This paper describes the use of numerical analysis for studying natural convection cooling systems for long term storage of heat producing radioactive materials, including special nuclear materials and nuclear waste. The paper explains the major design philosophy, and shares the experience of numerical modeling.

The strategy of storing radioactive material is to immobilize nuclear high-level waste by a vitrification process, converting it into borosilicate glass, and cast the glass into stainless steel canisters. These canisters are seal welded, decontaminated, inspected, and temporarily stored in an underground vault until they can be sent to a geologic repository for permanent storage. These canisters generate heat by nuclear decay of radioactive isotopes. The function of the storage facility ventilation system is to ensure that the glass centerline temperature does not exceed the glass transition temperature during storage and the vault concrete temperatures remain within the specified limits. A natural convection cooling system was proposed to meet these functions.

The effectiveness of a natural convection cooling system is dependent on two major factors that affect air movement through the vault for cooling the canisters: (1) thermal buoyancy forces inside the vault which create a stack effect, and (2) external wind forces, that may assist or oppose airflow through the vault.

Several numerical computer models were developed to analyze the thermal and hydraulic regimes in the storage vault. The **Site Model** is used to simulate the airflow around the building and to analyze different air inlet/outlet devices. The **Airflow Model** simulates the natural convection, thermal regime, and hydraulic resistance in the vault. The **Vault Model,** based on computational fluid dynamics techniques, is the main mechanism for analyzing internal vault temperature stratification; and, finally, the **Hot Area Model** is used for modeling concrete temperatures within the vault.

The end result of design and numerical modeling is a viable natural convection cooling system that performs its intended functions and is a safe practical alternative for the storage of radioactive canisters.

I. INTRODUCTION 4-2

The strategy of nuclear high-level waste is to immobilize by vitrification, converting it into borosilicate glass, then cast the glass into stainless steel canisters. The canisters are seal welded, decontaminated, inspected, and temporarily stored in an underground vault at an on-site canister storage facility until they can be sent to a geologic repository for permanent storage. The canister storage facility provides space for safe handling and temporary storage of canisters. The storage facility in this analyses is a two-level structure consisting of an above grade operating area, and a below-grade canister storage vault (Figure 1). The storage vault contains steel tubes to hold the canisters firmly in place. The loading and unloading of the canisters is done through cover plugs in the operating area floor. The canisters provide primary confinement; and the storage tubes (along with the associated cover plugs) form a secondary confinement system.

A natural ventilation system has many advantages as compared to a forced air ventilation system, such as high reliability, reduced operating and maintenance costs and no requirements for normal and emergency power due to the absence of active components. Radioactive decay heat, generated by the canisters in storage, is removed by the natural convective air cooling ventilation system. The air for cooling is drawn from the atmosphere through a number of air inlet towers, down through an inlet plenum, directed through underground vaults across the bank of vertical tubes, and exits through an exhaust stack. As an example, a plan view of the storage vault configuration is presented in Figure 2. The air is drawn through the vault by natural convection due to the buoyancy forces created by the density difference between the hot air inside the vault and cold air outside. HEPA filtration for the storage vault is unnecessary because potential contamination from a leaked canister will be confined to individual storage tubes and cannot enter the vault area.

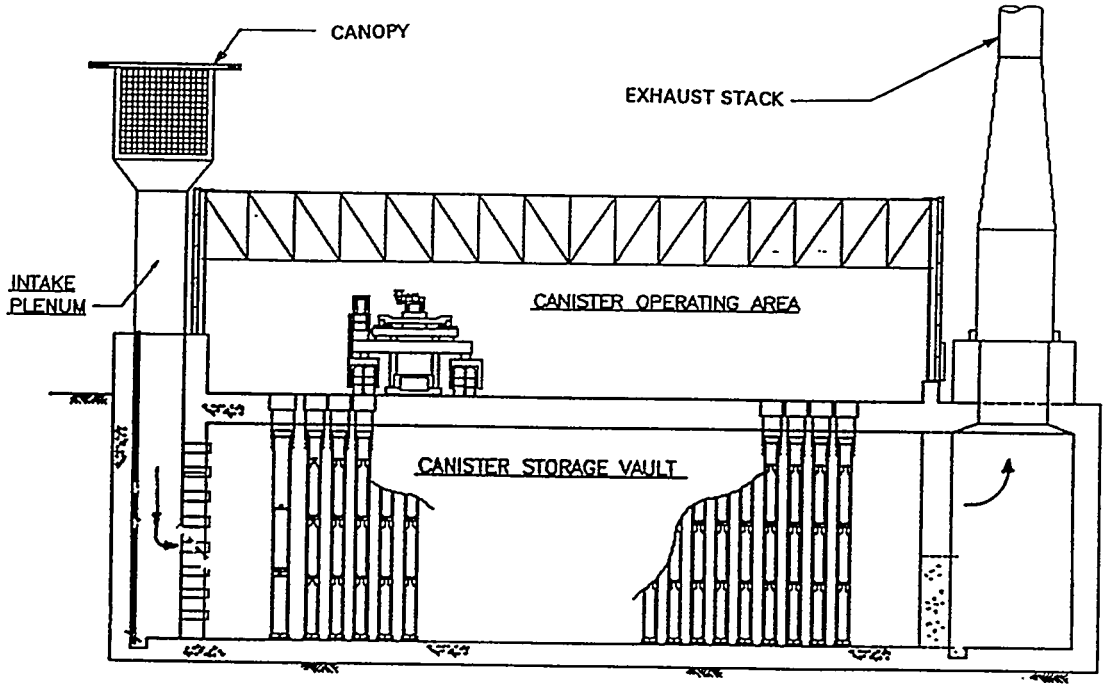


Figure 1. Section view of canister storage building.

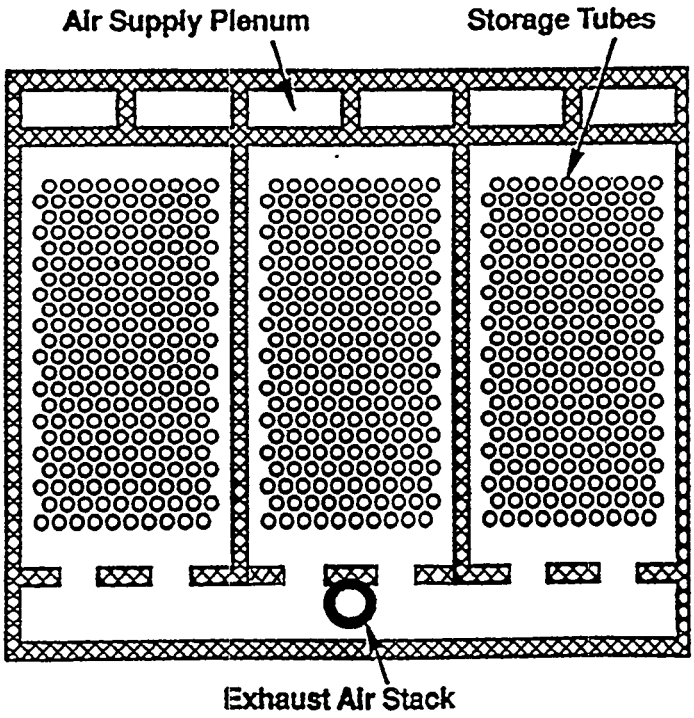


Figure 2. Plan view of canister vault.

Air movement through the vault cools the canisters and the surrounding concrete structure. The amount of airflow depends on the vault and ambient air temperature difference, stack height and diameter, and airflow resistances. Airflow resistances depend on air inlet configuration, internal vault air path geometry, stack size and height; and external wind forces, that may assist or oppose the airflow.

Several numerical models were developed to analyze the thermal and hydraulic regimes in the storage vault including the wind effect and the impact of volcanic ashfall on the overall efficiency of the passive cooling system.

II. DESIGN PHILOSOPHY

II.1 Objective

The primary objective of this study is to analyze the concept of natural convection cooling to assure that the glass centerline temperature does not exceed the glass transition temperature. Also, the vault concrete temperature must remain within specified temperature limits under varying vault loading conditions during normal, abnormal (high wind and ashfall), and accident conditions.

II.2 Description

The natural convection system consists of three major parts: inlet towers and plenums, vault loaded with canisters, and exhaust stack. Numerical modeling shows that wind pressure is one of the major factors affecting the performance of the cooling system. Therefore a number of design studies were conducted with high concern to utilize wind pressure: inlet towers contain wind braking vertical baffles and aerodynamic canopy, outlet

stacks are equipped with wind deflectors. Analysis indicates that wind may assist or oppose the airflow. Wind effect is simulated using Computation Fluid Dynamics (CFD) techniques. The simulation results show the necessity of using aerodynamic canopies on the top of inlet towers in order to eliminate the negative pressure at high wind in any direction (Figure 1). The inlet tower and outlet stack performance were verified through a wind tunnel test conducted by Colorado State University[1] showing the effectiveness of the proposed construction, which includes wind braking internal baffles, aerodynamic canopy, and wind deflector. The wind tunnel test has proven that for any wind direction and speed, total pressure of wind and stack effect will stay positive.

III. ANALYSIS

III.1 General

Theoretically, if negative and positive pressures that control air movement are equal, a choking effect may occur and stop the airflow. The result of numerical analysis shows that, without specific modifications to the air inlet towers and the exhaust stack, wind may partially stop airflow, which results in a corresponding increase in vault temperature. If this choking effect continues for an extended period the canister centerline temperature may exceed its limitation.

Numerical modeling proved that canisters storage tubes should be arranged in staggered configuration because it provides greater thermal efficiency than an in-line configuration (Figure 2). Also, due to the low air velocity through the vault, the increase in pressure loss due to staggered configuration is not significant when compared to a in-line storage tube arrangement.

Another concern addressed in the numerical study is the possibility that the volcanic ash may adhere to the storage tube surfaces which could create a thermal resistance layer around the tube surface, reducing heat transfer, thus causing the canister centerline temperature to rise above the limit. In this analysis the design of the air inlet towers limits ash from being drawn into the vault and also prevents entry screens from being blocked.

The performance of the passive ventilation system has been analyzed by simulating the thermal and hydraulic regimes in the vault area through the use of the following numerical models:

- **Site Model** for simulating airflow around relevant buildings on the site and air inlet/outlet devices at different wind speeds and directions.
- **Airflow Model** for simulating the natural convection cooling system by simultaneously considering the stack, wind, and vault thermal and hydraulic effects.
- **Vault Model** for simulating internal vault conditions, such as the air temperatures stratification and air velocity distribution.

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[1]M.Poreh, J.E.Cermak, H.G.C. Woo "Canister Storage Building Wind-tunnel Test", Colorado State University, Fort Collins, Colorado, 1992

Hot Area Model for simulating the maximum temperature in the vault environment. This information is used to design the vault concrete structure.

III.2 Site Model

The objectives of site modeling are: (1) to simulate airflow around the canister storage building in order to analyze static pressure at air entry openings, and (2) to study the effectiveness of different air inlet and outlet configurations.

Input parameters to the site model include climatological data on outside air temperature, wind speed and direction. It was suggested to analyze the extreme climatological conditions. For example, this would be an outside air temperature of 115°F (46°C) and wind speed of 31 mph (14 m/s) when the affect of these parameters is computed separately. When their combined affect is considered, the outside air temperature and wind speed assumed are 101°F (38°C) and 15 mph (6.7 m/s) respectively.

Site simulation is performed by several two and three-dimensional Computational Fluid Dynamics models using the PHOENICS[2] computer code. The results of the numerical analysis show that wind is one of the major factors effecting passive cooling system, particularly when storage vault is only partially loaded and stack effect is minimal. It was found that the wind is capable of reducing flow, reversing flow, or even completely stopping the airflow within the vault. Simulation was also performed for two kind of air inlets: tower inlets and traditional side inlets at ground level. The analysis indicates that tower inlets with internal baffles and outer aerodynamic canopy have significant pressure advantages over side inlets.

Evaluation of the numerical analysis was performed at Colorado State University by using large scale wind tunnel modeling. This evaluation includes the effect of wind pressure on the storage building at different wind speeds and directions, study of the effectiveness of the inlet structure, and selection of an efficient wind deflector. The wind tunnel test confirmed the performance of the natural convection system and helped to validate the site model, and to confirm the wind effects.

III.3 Airflow Model

The objective of the Airflow Model is to simulate temperature, airflow, pressure, and buoyancy effects in the vault in order to identify canister center line and structure temperatures at varying canister loading scenarios, air conditions, and hydraulic resistances. The Airflow Model considers the hydraulic resistance along the entire airflow path within the vault, the wind effect, and the stack effect. The airflow hydraulic resistances are determined as functions of air entry geometry, internal vault, and stack size and height using the T-DUCT[3] computer code. The amount of airflow ultimately depends on the vault and ambient air temperature difference, the stack height and diameter, and the airflow hydraulic resistances. Additional pressure created by wind can contribute to or hinder the natural convection effect. For a conservative approach the positive effect of wind is not considered, however for the same reasons, any negative effect of wind is carefully analyzed and deducted from stack

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[2]PHOENICS computer code developed by CHAM, Huntsville, Alabama

[3]T-DUCT computer code developed by NETSAL & Associates, Fountain Valley, California

pressure.

The Airflow Model assumes that airflow in the vault is moving in a horizontal direction from inlet to outlet with a limited upward temperature gradient. Later this approach is corrected in the Hot Area Model by accounting for inside air circulation and "hot spots". For a loaded vault the air is gradually heated as it passes through rows of canisters. As the first guess, air temperature at the outlet stack is assumed as the highest in the vault. Later this assumption is analyzed by using the Hot Area Model. The Airflow Model calculates maximum centerline canister temperatures for each sleeve in a row with a staggered or in-line arrangement and defines exhaust air temperature and airflow. Mostly, physical properties of nuclear materials (density, heat transfer coefficient) vary with temperature. The Airflow Model solves numerically the differential equation that describes the heat transfer process for each elemental layer in a canister. Then it does pressure balancing between stack effect and hydraulic resistance by iterations. Air densities, stack effect, and flow resistance calculations are the parts of the iteration process. The iteration process is considered converged when stack effect and total airflow resistances are matched.

As an example, the Airflow Model has been used to simulate a vault loaded with 2000 canisters. The heat loss is expected to be 1,500 kW. For this heat load, with assumptions of maximum air inlet temperature of 115°F (46°C) and zero wind resistance, the simulation predicts the vault air temperature distribution presented in Figure 3. The maximum temperature in the vault is estimated to be 187°F (87°C). The outlet air temperature is 171°F (77°C). For the above values of air inlet temperature and wind resistance, a series of simulations has been performed at various heat loads to determine the

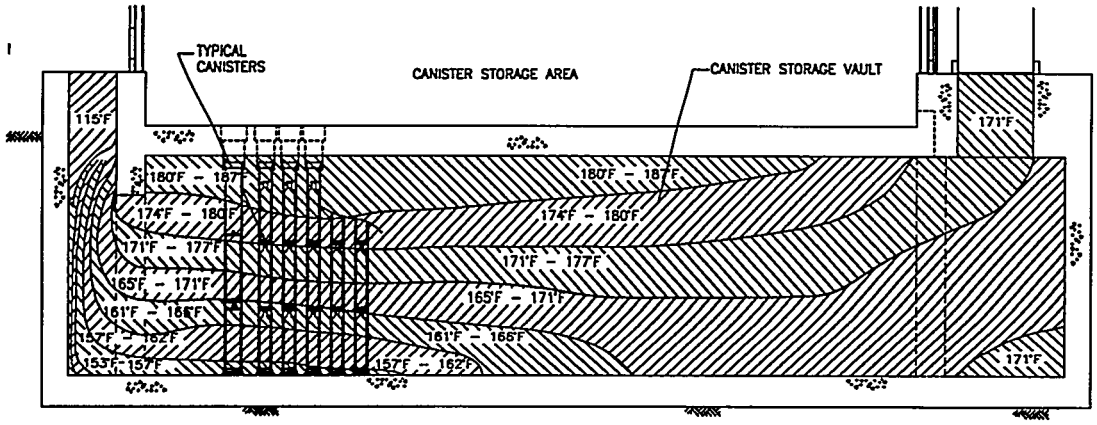


Figure 3. Temperature distribution in the vault.

total airflow through the vault and the outlet air temperatures as functions of the number of stored canisters. The results of these simulations are presented in Figures 4 and 5.

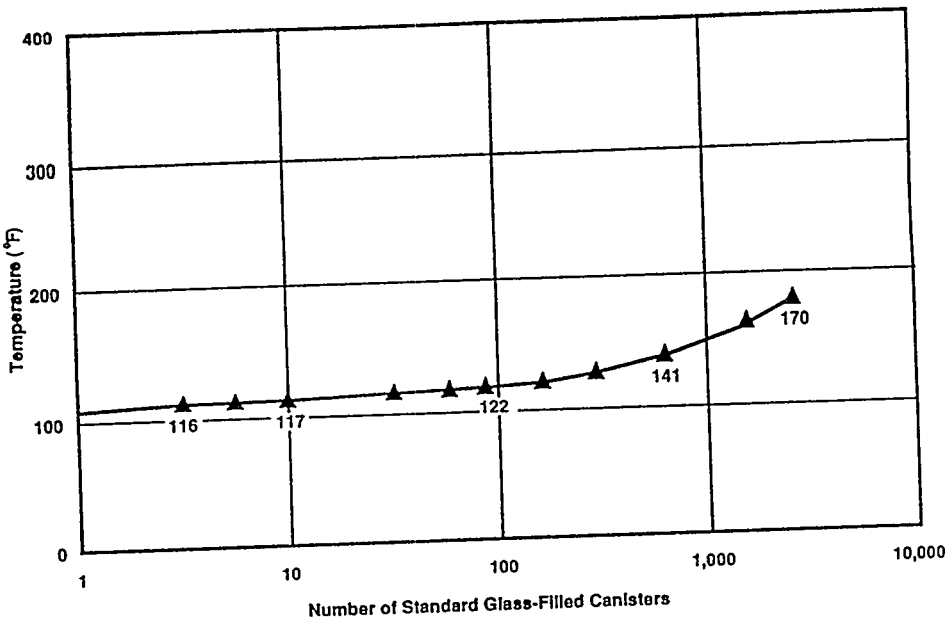


Figure 4. Outlet air temperature as a function of the number of stored canisters.

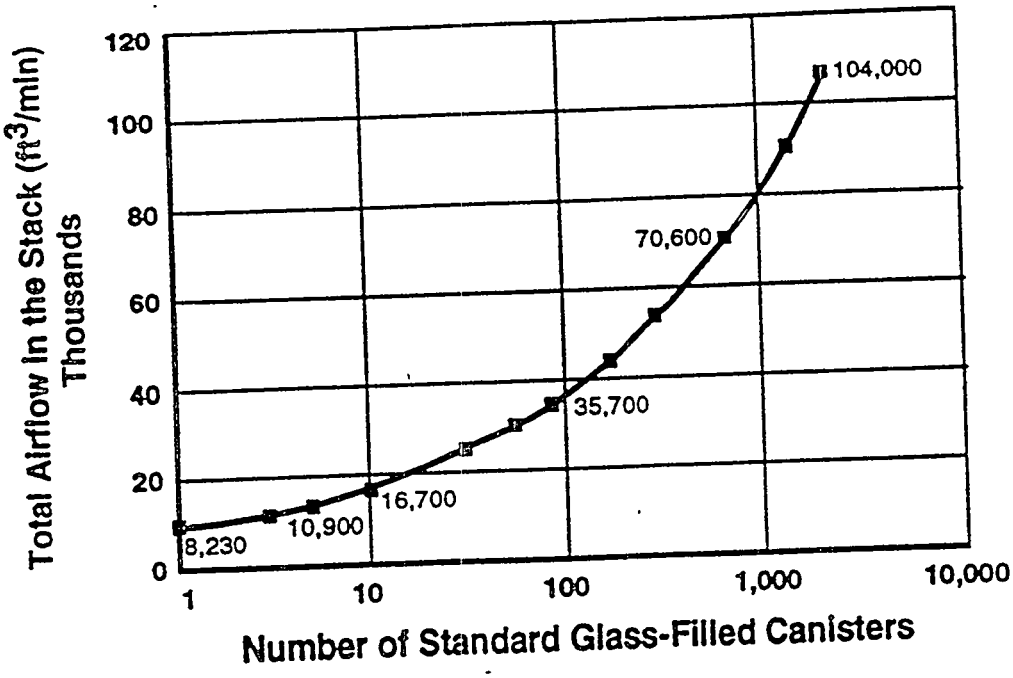


Figure 5. Total airflow in the outlet stack as a function of the number of stored canisters.

III.4 Vault Model

The objective of the **Vault Model,** based on CFD technique, is to simulate actual flow and temperature distribution inside the vault using the airflow calculated previously by the Airflow Model. The Vault Model locates the hottest temperature areas in the vault, called "hot spots." These temperatures are used: (1) to adjust temperature for the surrounding air in the Airflow Model when calculating canister center line temperatures, and (2) to calculate maximum concrete ceiling temperature.

There are three major airflow regions in the vault: (1) the beginning of the stream where entering air is moving down creating first entry vortex, (2) the middle of the stream divided into a lower part moving toward the outlet and an upper part moving toward the inlet, and (3) a third part which is a vortex located below the vault outlet opening.

Air buoyancy and slower air movement in the third flow region create higher temperatures at the top of the vault. This is called the "hot area". To calculate the actual air temperature in "hot area" is important for finding the surface temperature of the concrete. The numerical simulation, using the PHOENICS computer code, shows internal air temperatures at each cell including temperatures of inlet and outlet air. Then, the maximum difference between outlet and "hot area" air temperatures is added to the outlet temperature calculated by the Airflow Model. The result of this combined air temperature is considered as the "hot area" air temperature in the vault.

For the example above, it shows the maximum temperature difference between outlet air and air in the "hot area" is 9°F (equivalent to 5°C). This temperature difference was added to the highest air temperature in Airflow Model for identifying "hot area" temperature in the vault. It should be noted that by adjusting the geometry of the inlet and outlet plenums a minimum temperature difference between the "hot area" and the outlet air temperature can be attained.

For a partially loaded vault the outlet air temperature, the stack effect pressure as well as the airflow are lower. This may create hot areas at a lower heat load. A separate study indicates that lower heat load reduces air temperatures all over the vault. This means that the only hot area critical conditions that should be analyzed are at the highest canister loading.

III. 5 Hot Area Model

The objective of the Hot Area Model is to determine the maximum concrete structure temperature. The "Hot area" is a part of vault's inner space where, due to the buoyancy effect, the air temperature is highest. Calculation of temperatures and heat fluxes at the hot area is based on the results obtained by previous airflows and temperatures calculated in the Vault Model. The heat from hot area is transferred: (1) to the natural convection cooling system, and (2) to the operating area above the storage vault (see Figure 1).

The Hot Area Model is used for simulating temperatures in the upper vault area and to determine the concrete temperature profile. These temperatures are functions of many parameters such as outside air temperature, wind resistance, canister load, insulation thickness, etc. It is evident that concrete temperature rises with the increase in outside air temperature, however, stack effect moderates the temperature rise.

III.6 Model Interactions

A flowchart of model interaction is presented in Figure 6. Initially, the Site Model calculates static pressures differences at air inlet openings for different wind directions and velocities. In the event that wind may create a pressure resistance at inlet openings, this pressure is carried over to the following Airflow Model. If the wind creates a negative effect on the inlet, wind pressure is deducted from the total pressure balance which includes inlet wind pressure, outlet wind pressure, stack effect, and hydraulic resistance. If the wind creates a positive pressure force at inlet openings, conservative approach is taken and no wind pressure is assumed. The Airflow Model calculates the canister surface and centerline temperatures. Factors effecting these calculations are the stack effect and hydraulic resistances at the inlet towers and outlet stack calculated by the Site Model. Airflow is calculated for an equilibrium between path hydraulic resistance and the stack effect pressure.

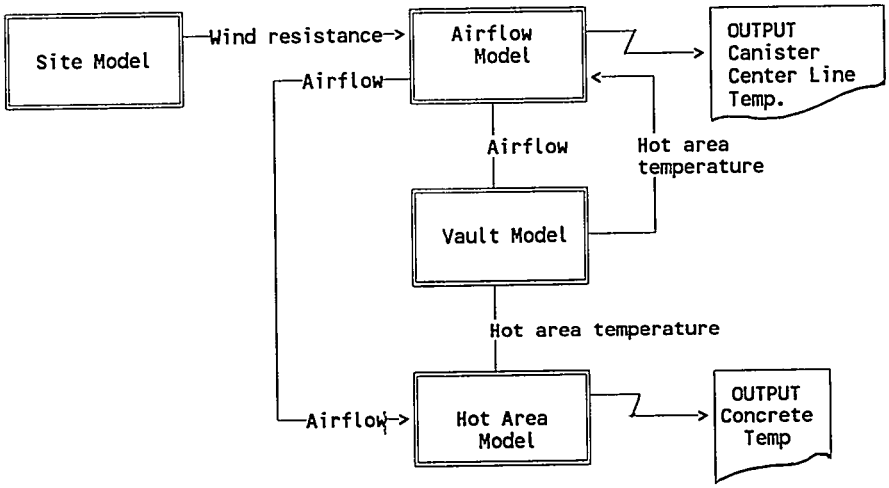


Figure 6. Computation flowchart.

The Vault Model simulates distribution of air temperatures in the vault using the amount of flow calculated by the Airflow Model. The CFD technique used in this model allows the user to identify the "hot areas" in the vault where air temperatures are the highest. It iterates into the Airflow Model and becomes a base for calculating canister temperatures in the most critical "hot areas." After several iterations between the Airflow Model and the Vault Model, the maximum "hot area" temperature is carried over to the Hot Area Model where it is used as a base for calculating concrete temperature.

IV. CONCLUSION AND RECOMMENDATIONS

The results of this engineering analysis, computer modeling, and wind tunnel testing show that a natural convection cooling system performs its intended function, and is a safe alternative design for the storage of radioactive materials.

The following are the recommendations based on the results of this study:

- The natural convection system for canister storage must have an exhaust stack with a wind deflector to reduce hydraulic resistance.
- Air inlet should be through one or more inlet towers designed and tested to prevent or minimize the negative wind pressure effect for all wind directions. Installation of the internal wind braking baffles, and aerodynamic canopy (Figure 1), are recommended.

DISCUSSION

MURTHY: Is your answer, yes; your study has now been accepted by the authorities, the DOE, or is it a new study and nobody knows about it?

TSAL: Mostly yes, because, first of all, we do have a utility in Colorado. There are several utilities in the world working on natural convection, for example in France. I know there are French people here and maybe they will help me find out where those utilities are. Also there is a utility in the United Kingdom. Unfortunately, in the United States we have to penetrate the wall of incomprehension and this is one of the reasons for my presentation. After working all my life in the HVAC field, dealing with natural convection, I truly believe that this is possible, legitimate, and that we have to go ahead with natural convection. It is much cheaper and more effective than the forced convection. But we are still going through the walls of incomprehension, it has not been easy to convince people. Fluor Daniel is in the stage of preconceptual design of natural convection (NC) for both nuclear waste and SNM storages for DOE.

DISPOSAL OF SLIGHTLY CONTAMINATED RADIOACTIVE WASTES FROM NUCLEAR POWER PLANTS

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ABSTRACT

With regard to the disposal of solid wastes, nuclear power plants basically have two options, disposal in a Part 61 licensed low-level waste site, or receive approval pursuant to 20.2002 for disposal in a manner not otherwise authorized by the NRC. Since 1981, the staff has reviewed and approved 30 requests for disposal of slightly contaminated radioactive materials pursuant to Section 20.2002 (formerly 20.302) for nuclear power plants located in non-Agreement States. NRC Agreement States have been delegated the authority for reviewing and approving such disposals (whether onsite or offsite) for nuclear power plants within their borders. This paper describes the characteristics of the waste disposed of, the review process, and the staff's guidelines.

INTRODUCTION

NRC regulations, at 10 CFR Part 20.2001, authorize four general alternatives for nuclear power plant licensees to dispose of slightly contaminated radioactive wastes from nuclear power plants: (1) by transfer to an authorized recipient as provided in 20.2006, or in Parts 30, 40, 60, 61, 70, or 72 of the NRC regulations or, (2) by decay in storage, (3) by release in effluent within the limits in 20.1301, or (4) as authorized under 20.2002, 20.2003, 20.2004, or 20.2005. In February 1983, the NRC staff published Information Notice No. 83-05 entitled "Obtaining Approval for Disposing of Very-Low-Level Radioactive Waste - 10 CFR Section 20.302." This Information Notice reminded nuclear power plant licensees that they could apply on a case-by-case basis for permission to use alternative methods for disposing of slightly contaminated radioactive materials (i.e., methods other than disposal at commercial wastes sites).

CHARACTERISTICS OF DISPOSED WASTES

During the 13-year period between 1981 and 1994, the NRC staff has reviewed and approved 30 requests for disposal of slightly contaminated radioactive materials from nuclear power plants (see Figure 1). Experience has shown that the review process can take from 2 weeks to 1 year. Table 1-5 contains a list of applications processed by the staff, as well as the general physical characteristics of the wastes. The types of waste disposed of are the following slightly contaminated materials: soil, sand, sediment from onsite settling ponds, sewage sludge, wood, spent resins used for cleaning the secondary side of pressurized water reactors (PWRs), roofing materials, and scrap metal from feedwater heaters used in the secondary side of PWRs (see Figure 2).

The principal nuclides in the wastes disposed are Co-58, Co-60, Cs-134, and Cs-137, with total activity concentrations in the range of about 1 to 50 pCi/g.

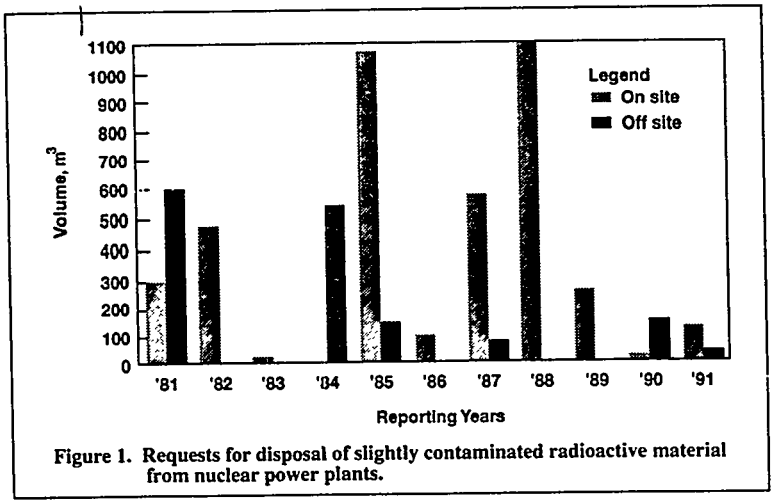


Figure 1. Requests for disposal of slightly contaminated radioactive material from nuclear power plants.

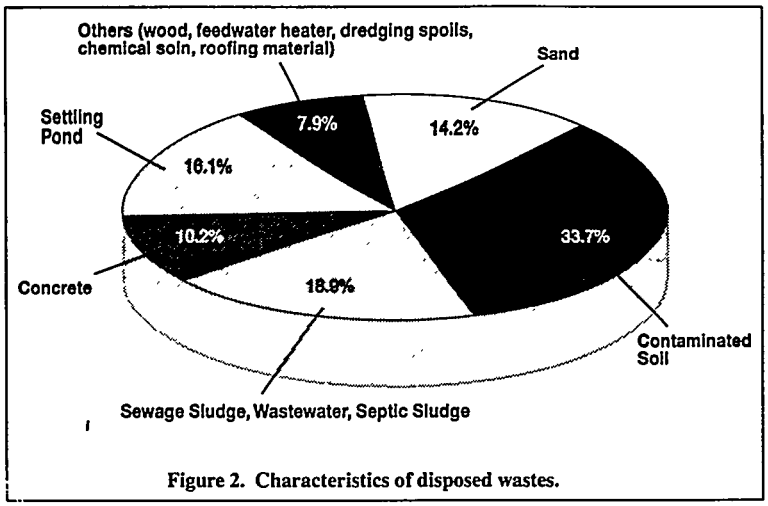


Figure 2. Characteristics of disposed wastes.

Figure 3 shows the total activity (mCi) versus volume of disposed waste.

Disposal methods have included municipal landfills, on-site burial, and processing at a hazardous waste disposal site. Although most of the requests have been for a one-time disposal, the NRC staff has approved requests for disposal of limited quantities contaminated materials on a repetitive basis, e.g., annual disposals of slightly contaminated wood, and disposals of contaminated sewage sludges every 1-2 years. For these repetitive disposals, the licensee must reapply to the NRC when a particular disposal would exceed the boundary conditions imposed by the staff's analysis (see Figure 4).

REGULATORY REVIEW PROCESS

Applications from reactor licensees in non-Agreement States should be submitted to the Office of Nuclear Reactor Regulation (NRR). Under 10 CFR 20.2002, licensees may request disposal of specific material on a case-by-case basis or licensees may request permission for routine disposal of specific types of wastes on a repetitive basis using approved procedures and systems. For disposal of radioactive materials in non-Agreement States, the application is reviewed solely by NRC. For disposal of radioactive material in an agreement State on or away from the reactor site, approval is needed from the Agreement State.

In their submittals licensees should describe the waste, the principal pathways of exposure at the disposal site, and the estimated dose to the maximally exposed individual from these pathways. For each planned request, the information regarding the waste for each planned request should briefly describe: (1) the item to be disposed, including the approximate volume or mass; (2) the principal nuclides expected to be in the waste; (3) the estimated radionuclide concentrations in the waste; (4) the estimated total activity of nuclides in the waste; and (5) the basis for the estimated concentrations and total activities (i.e., the number of samples measured, the representative nature of the samples, and the appropriateness of the instruments used to measure the activity in samples). Information regarding the disposal site should include: (1) the method of disposal (e.g., diluted with other sludge, burial in deep trenches, or spread over land and cover with "clean" soil); (2) the location of the disposal site (e.g., a legible map of the disposal site with compass direction and scale); (3) local land use (e.g., nearby residences and wells); and (4) physical or administrative barriers to prevent present or future use of the site for other than its intended purpose.[1]

Licensee should also briefly discuss the potential pathways of exposure and estimated doses to individuals from the principal pathways of exposure. Doses should be estimated for both a maximally exposed member of the public and a non-occupationally exposed worker. If a particular pathway is not of concern (e.g., inhalation of resuspended radionuclides) then this should be stated and the basis for the statement should also be stated (e.g., the nuclides are in an immobile form or the material is isolated from surface winds by several feet of earth cover.) Among pathways that are typically of concern include: 1) external exposure from standing or living above the disposal site (2)

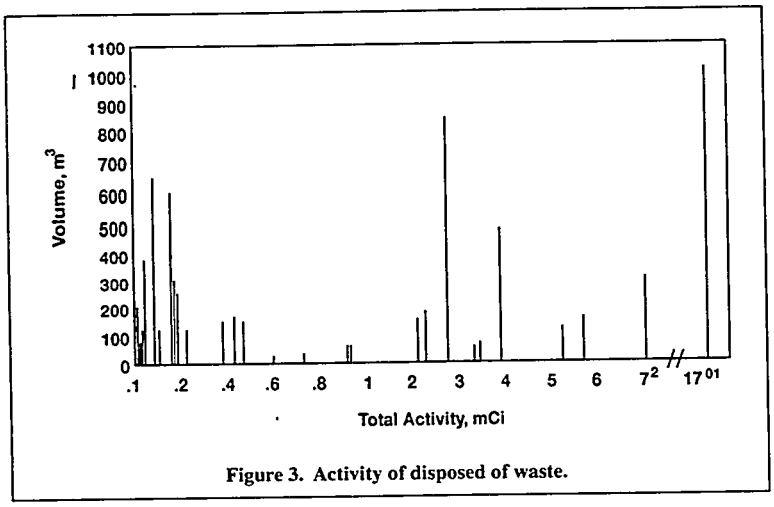


Figure 3. Activity of disposed of waste.

inhalation of resuspended radionuclides if the radioactive material is not covered promptly or effectively, (3) external and internal exposure to an inadvertent intruder, (4) external and internal exposure of an individual from assumed recycling of the material disposed of at the time the disposal site is released from regulatory control, (5) internal exposure from the ingestion of ground water, and (6) internal exposure from ingestion of food grown on the disposal site (see Figure 5).

DOSE GUIDELINES

In performing its safety evaluations of licensee submittals, the NRR staff has developed radiation dose guidelines for its staff. NRR is developing additional review guidelines such as standard review plan and staff report are under development to ensure that potential radiation doses that may result from the proposed method of disposal are calculated in a consistent manner. These guidelines are intended to apply to solid wastes (from reactor facilities) slightly contaminated with radionuclides with halflives less than 35 years. NRR reviews the licensee's waste stream description, radiological properties, and proposed disposal method, as well as radiological impacts, calculational methods, and assumptions, to ensure that the public health and safety is adequately protected. The staff will seek additional information from the licensee to further justify acceptance of the licensee's assessment methodology.

NRR developed its guidelines according to the following principles: first, the annual dose to a member of the public from exposure to the material disposed of should be a small fraction of annual exposure to natural background radiation. Second, the annual dose to a member of the public from exposure to the material disposed of should be no greater than the annual dose a maximally exposed individual would receive from exposure to radioactive effluents from normal operations at light water reactors. Third, concentrated sources of radioactive materials that might pose a health hazard before or after the time of release of the disposal site from all regulatory controls should not be permitted to be disposed of under 10 CFR 20.2002.

The NRR guidelines follow:

1. The radioactive material should be disposed of in such a manner that it is unlikely that the material would be recycled.

2. Doses to the whole body and to any body organ of a maximally exposed individual (a member of the general public or a non-occupationally exposed worker) from the probable pathways of exposure to the material disposed of should be less than 1 mrem/yr.

3. Doses to the whole body and to any body organ of an inadvertent intruder from the probable pathways of exposure should be less than 5 mrem/yr.

4. For onsite disposal, the dose to the whole body and to any body organ of an individual from assumed recycling of the material disposed of at the time the disposal site is released from regulatory control from all likely pathways of exposure should be less than 1 mrem/yr.

5. For disposal in a sanitary landfill, the dose to the whole body and to any body organ of an individual

from assumed recycling of the material disposed of at the time of disposal from all likely pathways of exposure should be less than 5 mrem/yr.

In conclusion, NRR believes that the use of 10 CFR 20.2002 for case-specific situations involving slightly contaminated radioactive wastes has been appropriate and any potential radiological impact on public health and safety or the environment has been minimized.

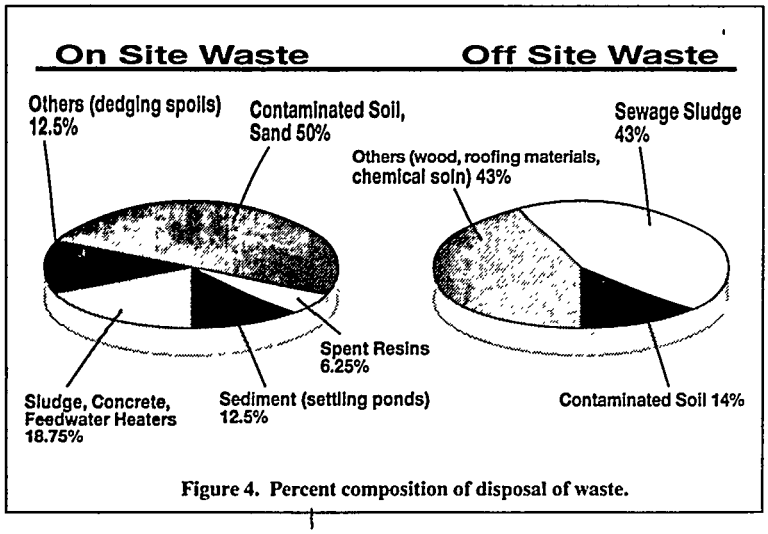


Figure 4. Percent composition of disposal of waste.

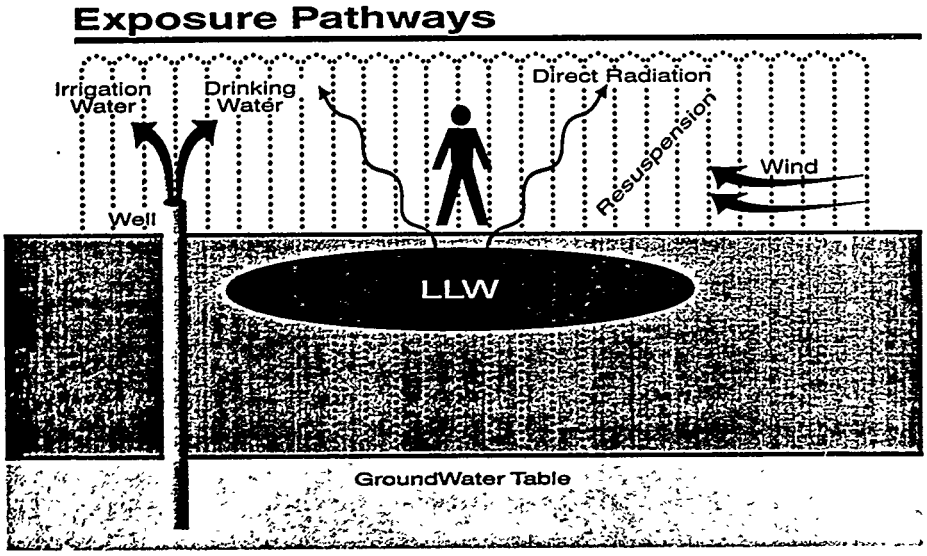


Figure 5. Potential pathways of exposure.

Table 1-5 Approved Requests for Disposal of Slightly Contaminated Radioactive Wastes from Nuclear Power Plants PUrsuant to 10 CFR 20.302.

Application Location	Waste Chars. (m^3)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
San Onofre	sand 300	onsite	Cs-137	0.2 mCi	APPROVAL: 1 mrem/yr whole body
Oyster Creek	contaminated. soil 480	onsite	Co-60 Cs-137 Mn-54 Cs-134	5 mCi	APPROVAL: 3 mrem/yr whole body
DC Cook	contaminated concrete, steam generator replacement 653	onsite	Co-60 Cs-134 Cs-137	0.1 mCi	APPROVAL: Insignificant impact because it involves pathways less significant than those considered in the FES.
Vermont Yankee	septic waste 262	onsite	Co-60 Mn-54 Cs-137 Cs-134 Zn-65	0.2 mCi per acre	APPROVAL: 0.2 mrem/yr maximally. exposed individual/organ:, 3.91 mrem/yr inadvertent intruder licensee evaluation..
Yankee Rowe	sewage 200 once every 1 to 2 years for 30 years.	offsite	Co-60 Mn-54 Cs-134 Cs-137	0.2 mCi	APPROVAL: 0.12 mrem/yr maximally exposed individual/whole body (child), ground irradiation,inhalation,stored vegetables,leafy vegetables,milk ingestion.
Big Rock Point	dredging spoils 15 yr	onsite	Co-60 Mn-54 Cs-137 Cs-134 Sr-90	0.9 mCi	APPROVAL: 0.03 mrem/yr whole body dose maximally exposed individual (groundshine,inhalation,groundwater)& 0.857 mrem/10 yr.
Palisades	soil 170	onsite	Co-60 Cs-137	0.03 mCi	APPROVAL: <1 mrem/yr whole body dose maximally exposed individual (groundshine, inhalation, groundwater ingestion)

*Disposal of Waste from Nuclear Power Plants.

Application Location	Waste Chars. (m^3)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
Maine Yankee	Hazardous chemical solution 40	offsite	Co-60 Zn-65 Cs-137	0.1 mCi	APPROVAL; 0.1 mrem/yr (lung) inhalation resuspended activity/ingestion of crops grown onsite; 0.0064 mrem/yr (whole body) ingestion drinking water; 0.34 mrem/yr (whole body) direct radiation buried activity; 0.12 (whole body) direct radiation waste transportation; 0.005 (whole body) direct radiation waste handling.
Sequoyah	trash 750	offsite		200 mCi	REJECTED: high-level of specific activity of less than 2 nCi/gm, and total activity proposed/ year of 200 mCi, are each orders of magnitude higher than similar parameters of any 20.302 proposal approved for offsite, also non-homogeneity of the trash
Fermi-2	cont soil 850	onsite	Cr-51 Mn-54 Co-58 Co-60	0.3 mCi	APPROVAL: 0.044 mrem/yr whole body public water and fishing ingestion and shoreline sediments, 0.0674 mrem/yr direct exposure to contaminated soil, resuspension of soil into air: 0.3 mrem/yr direct exposure.
Kewaunee	waste sludge 454	onsite	Co-60 Cs-137	0.2 mCi	APPROVAL; 0.034 mrem/yr (whole body) groundshine; 0.008 mrem/yr (whole body) inhalation; 0.007 mrem/yr (whole body) groundwater ingestion.

*Disposal of Waste from Nuclear Power Plants.

Application Location	Waste Chars. (m^3)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
Brunswick	dredging sediments, sand	onsite	Mn-54 Co-60 Cs-137		NON- APPROVAL: Meet NRC acceptance criteria, but the licensee is regulated by State. State approval is needed
Point Beach 1,2	sewage sludge 113	onsite	Co-60 Cs-137	0.003 mCi	APPROVAL: Based on licensee pathway analysis and the licensed materials <3% of the primarily nuclides, already acceptable in the FES,site-speciffic application to be insignificant radiological impact.
Surry 1,2	soil 300	onsite	Co-60 Cs-134 Cs-137 Mn-54	72	APPROVAL: Groundshine, inhalation dose breathing resuspended airborne radioactivity, and ingestion of radioactivity from contaminated water to maxiimally exposed member of the public <1 mrem/yr; inadvertent intruder <5 mrem/yr.
H.B. Robinson	sediment 170	Fossil plant ash pond in licensees controll area	Co-68	75	APPROVAL: <5 mrem (teenager, total body of person working 400 hr/yr above contaminated surface of soil cover zone.
H.B. Robinson	Soil (1.5)	Onsite along the bottom of a drainage ditch	Co-58 Co-60 Co-134 (10%) Cs-137 (23%) Mn-54 all nuclides	0.014	APPROVAL: <5mrem (direct radiation whole body of person working 400 hr/yr above contaminated surface of soil cover zone.

*Disposal of Waste from Nuclear Power Plants.

Application Location	Waste Chars. (m^3)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
Humboldt Bay 3	Sludge (36.8)	Offsite RCRA chemical waste disposal landfill (Martinez , Calif.)	Co-60, Cs-134 Cs-137 Th-234	267.8 3.1 155 19.3	APPROVAL: 1.5 mrem (direct radiation to worker standing on uncovered dried sludge)
Oconee Units 1,2,3	Sewage sludge 113	Offsite sanitary landfill	Co-58, Co-60 (27%) Cs-134 Cs-137 (45%) all nuclides	0.07	APPROVAL: 0.6 mrem (whole body direct radiation standing on uncovered dried sludge) 0.2 mrem(whole body ingestion of vegetable grown on landfill w/sludge). 0.3 mrem (highest dose to any organ, ingestion of vegetable).
H.B. Robinson	Setting pond sediment 60,000 cu meters	Onsite fossil ash pond	C0-60	1700 (over life of pond)	APPROVAL: 5 mrem (direct radiation) to teenager 67 hr/yr at ash pond
R.E.Ginna 1	Roofing materials <100 tons	Offsite municipal landfill	Co-60 Cs-134 Cs-137	0.30 0.23 0.92	APPROVAL: 4 mrem (1st year) 9 mrem (thereafter)
McGuire 1 and 2	Wastewater residue sludge <368	Onsite	Co-58 Co-60	0.05 0.05	APPROVAL: <1 mrem (whole body direct radiation standing 2000 hr/yr on soil cover); <0.1 mrem (lung dose, worker inhaling dust 2000 hr/yr); <0.1 mrem (highest dose to any organ, ingestion of vegetable).

*Disposal of Waste from Nuclear Power Plants.

Application Location	Waste Chars. (m^3)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
Oconee 1, 2,& 3	Feedwater heater a high activity tube bundls. 160 tons. b. very low activity heater shells 100 tons	Company controlled area (outside sec fence) Onsite	Co-60 (79%) Cs-137 (15%) Co-60 (80%)	6.5	APPROVAL: 0.01 mrem (whole body direct radiation standing on soil cover) <2mrem to maximally exposed individual
Oconee 1,2,3	Sand (45)	Onsite Company controlled area (outside security fence)	Cs-134 Cs-137 Co-60 Mn-54 all nuclides	1.2 3 0.1 0.005 <12.3	APPROVAL: <1mrem(groundshine to whole body, standing 2000 hr on soil cover); <0.1 mrem (10 days inhalation of dust from disposal process; <2 mrem (future ingestion of crops grown on burial site, whole body or any organ.)
Big Rock Point	Cont. soil leak in comdensate process monitor	Onsite retain soil in place			APPROVAL
Davis Besse	Secondary side resins 142 or 150 every 5 yrs.	Offsite company owned	Co-58 (34%) Co-60 (3%) Cs-134 (27%) Cs-134 (36%)	8.5/ every 5 yrs.	APPROVAL: 0.7 mrem (direct radiation standing on uncovered basin dredgings); adult eating vegetable. grown on disposal site <3 mrem (whole body); <4 mrem highest dose to any organ; <0.1 mrem (drinking ground water (licensee estimate)

*Disposal of Waste from Nuclear Power Plants.

Application Location	Waste Chars. (m^3)	Proposed Disposal	Nuclides Present	Total Act.	Pathways
Oconee 1,2,3 McGuire 1&2 Catawaba 1&2	Wood 12-21	Offsite sanitary landfill	Assume Cs-137 100%	0.4 to 0.7 per yr per station	APPROVAL: <1 mrem (resident on decommissioned. landfill direct radiation. whole body or any organ of adult eating vegetable grown on soil cover); <0.1 mrem (nuclear station workers, direct radiation. or inhalation).
Pilgrim	Contaminated soil (63	Onsite inplace	Co-60 Cs-137	0.19 mCi 0.4	APPROVAL: <1 mrem external exposure < 0.02 ground shine; <0.5 internal exposure inhalation.resuspended nuclide, (2000 hr/yr), direct radiation0 2000 hr/yr.
D.C. Cook	Contaminated sludge	Onsite pre-burial	Cs-137 Cs-134 Co-60 I-131	8.89 (1982) 5.02 (1991)	APPROVAL: <1 mrem external exposure from disposal site; <1 mrem internal form inhalation of resuspended radionuclide; (3) <0.7 mrem internal exposure ingesting ground water

1. For wastes containing mobile radionuclides (e.g., H-3) detailed information on geology and hydrology may be necessary.

*Disposal of Waste from Nuclear Power Plants.

CLOSING COMMENTS OF SESSION CO-CHAIRMAN BERGMAN

We have had three presentations, the first on the Generation and Release of Radioactive Gases in Low Level Waste Disposal Facilities, by Dr. Yim. He basically stated that the waste is not a static entity, it is dynamic. Just because you have it buried somewhere doesn't mean it is not going generate some surprises. Gases will be generated by a variety of mechanisms. He reviewed microbial, radiolysis, and corrosion. Although, from his analysis, a significant amount of gas would be released, it would be within regulatory limits. It may be within regulatory limits at the present time but looking at the historical picture, when people make measurements twenty or fifty years from now, I can guarantee that regulatory limits will have been decreased. So it is a very important topic and I think the information should be folded into the decision of what kind of solid waste form we should have.

The second paper was by Dr. Tsal on Numerical Analysis of Natural Convection Cooling Systems for Radioactive Canister Storage. He described a natural convective cooling system consisting of two parts: (1) thermal buoyancy inside a vault that creates a stack effect, raising the gas, and (2) external wind forces that may either assist or oppose the air flow through the vault. He used several numerical models to analyze temperature and airflow within the storage vault and showed that natural convective will not only work, but is a safe alternative to the design of current systems.

The third presentation, by Mr. Minns, on the Disposal of Slightly Contaminated Radioactive Wastes from Nuclear Power Plants, described methods for disposal, both on site and off site. He also described current regulations that cover disposal of these wastes. People fail to consider these wastes when they are designing the new facilities. I know DOE's EM section is preoccupied with the initial processes for treatment of waste, with less and less thought and attention given to the end. What they do not realize is that after all their hard work, it may not be acceptable (as was pointed out this morning), if they do not consider the full cycle. Waste disposal is one of the most critical components. Current practice is to do more and more burial on site or wherever people do not get excited about it. A lot of facilities, such as the DOE weapons complex, are leaving items in storage because they can't find a waste site. Also, we have critical facilities within the DOE complex that contain HEPA filters that are 20-30 years old; they are kept in place because no one can get rid of them. To borrow the human analogy of Dr. Tsal, just think what would happen if you kept on eating and eating and never disposed of your waste. It doesn't take much imagination to see the consequence. So consideration of waste is a very important part of nuclear activities.

SESSION 5

FILTERS AND FILTRATION

Monday: July 25, 1994
Co-Chairmen: H. Gilbert
R. G. Dorman

HEPA FILTER CONCERNS - AN OVERVIEW
J. F. Leonard

A NOVEL PERMANENTLY MAGNETISED HIGH GRADIENT MAGNETIC FILTER USING ASSISTED CAPTURE FOR FINE PARTICLES
J. H. P. Watson

THE DEVELOPMENT OF A HEPA FILTER WITH IMPROVED DUST HOLDING CHARACTERISTICS
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EFFECT OF HUMIDITY ON THE FILTER PRESSURE DROP
J. Vendel, P. Letourneau

PRELIMINARY FIELD EVALUATION OF HIGH EFFICIENCY STEEL FILTERS
W. Bergman, G. Larsen, R. Lopez, K. Wilson, K. Simon, L. Frye

CLOSING COMMENTS OF SESSION CO-CHAIRMAN DORMAN

HEPA FILTER CONCERNS - AN OVERVIEW

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Abstract

The U. S. Department of Energy (DOE) recently initiated a complete review of the DOE High Efficiency Particulate Air (HEPA) Filter Program to identify areas for improvement. Although this process is currently ongoing, various issues and problems have already been identified for action that not only impacts the DOE HEPA filter program, but potentially the national and international air cleaning community as well.

This paper briefly reviews a few of those concerns that may be of interest, and discusses actions initiated by the DOE to address the associated issues and problems.

Issues discussed include: guidance standards, in-place testing, specifications, Test Facilities, portable units, vacuum cleaners, substitute aerosols, filter efficiencies, aging/shelf life/service life, fire suppression, handbook, Quality Products List (QPL), QA testing, and evaluations.

I. Introduction

An overall review of HEPA filter materials, manufacturing techniques, uses, tests, research, and system designs, reveals that many improvements can be made in the HEPA filter arena.

In addition, problems with material quality, aging, filter failures, special applications, and lack of guidance, indicate that additional evaluations, research, and other actions are necessary to resolve a wide variety of issues and problems.

The DOE is currently evaluating its overall HEPA filter program and has identified numerous areas for improvement. A DOE HEPA Filter Working Group has been formed to look at the issues and advise management concerning resolutions and priorities.

Some of the concerns identified for resolution are included in this paper for information, including actions currently underway by the DOE.

II. HEPA Filter Issues

Aging: HEPA filters are fiberglass paper products that deteriorate (weaken) with age. The aging affects the ability of the HEPA filters to continue to meet normal and abnormal conditions.

No definitive guidance is available concerning Shelf Life or Service Life of HEPA Filters although we rely heavily on this safety component to protect workers, the public, and the environment.

In-place tests, delta pressure gauges, and visual inspections do not address the deteriorating structural strength issue.

Some HEPA filters in DOE facilities have been found to be reaching ages in excess of 30 years old. In addition, many filters are subjected to heat, pressure, moisture, and chemicals which can speed up the aging process. These older filters usually contain high quantities of hazardous materials. In some cases the filters contain non-radioactive materials or mixed hazardous materials.

Failure of the filters during normal and abnormal conditions can occur, particularly during accident conditions in which the facility is relying heavily on the ability of the filters.

A mini-study on HEPA filters using the DOE Occurrence Reporting and Processing System (ORPS) computer database, was performed to determine if there is evidence of filter problems. Results of the mini-study indicates that HEPA Filters are failing due to aging and other factors. Also, some separate, recent research into the issue revealed problems associated with the aging process. Enough information has been collected to suggest that additional research, evaluations, and testing is warranted.

The DOE is currently trying to determine if a time frame for replacing HEPA filters is reasonable and should be instituted. Additional research to help determine the scope of the problem is necessary.

Results of some recent research and limited testing by the DOE on aging is presented elsewhere in the conference papers.

Quality Products List (QPL) and Filter Testing: Testing of filters and filter materials for certification and inclusion on the QPL was previously conducted by the Department of Army, with some limited testing performed by the DOE. Based upon a letter from the Army dated March 16, 1994, the Army will maintain their

testing facility but will no longer require testing for the QPL and will therefore no longer maintain the list.

Whether to test (and how much), or not test, needs to be revisited. Although the NRC and the DOE previously co-funded a QA/Acceptance pre-installation filter testing program, the NRC does not now require the commercial user to perform quality assurance and acceptance testing prior to installation of the filters. However, the DOE continues to perform filter acceptance and quality tests before installation, and currently maintains 3 Filter Test Facilities (FTFs) for this purpose.

Results of the FTF QA/Acceptance tests (which are different than in-place tests) indicate that the testing is not only desirable but necessary. The overall failure rate is approximately 3-4% with some models/manufacturers reaching 85-100% failures. The most common mode of failure is excessive penetration. The failure rate of respirator cartridges is approximately 1%.

Some non-DOE organizations make use of the DOE FTFs and pay to have their filters tested prior to installation.

The DOE has also discovered failed filters where the manufacturer did not have their products tested and included on the QPL.

In addition, some research and testing by the DOE into the cause of some recently failed new HEPA filters indicate that not all manufacturers are maintaining their products to the necessary material requirements. The new HEPA filters were structurally weak and prone to failure. In many cases the new filters failed the in-place test performed immediately after installation. This created the problem that the new, but now contaminated, filters had to be disposed of as radioactive waste while subjecting workers to unnecessary exposures while performing the extra installation and removal of the new filters (some filter changes are high potential contamination/radiation jobs). The failure of the HEPA filters to meet the required quality assurance tests, despite the requirements in the purchase orders, illustrates the need for DOE to conduct its own QA tests.

The results of research and testing on these failed filters is presented elsewhere in these conference papers.

The DOE currently maintains QPL testing capability at the Rocky Flats Filter Test Facility at Golden, Colorado but is not presently performing routine QPL-type testing. The DOE is currently reviewing filter testing needs in light of recent changes to the Army QPL program, and the DOE research on filter failures.

Assumed Efficiencies of HEPA Filters During Accidents: There are no standards or definitive guidance concerning efficiencies of HEPA Filters during abnormal or accident conditions. The continued efficient operation of the filters under these conditions is important since many facilities rely heavily on the HEPA Filters to prevent exposure of radioactive materials to workers, the public, and the environment. However, facilities often do not consider or evaluate the effect of adverse conditions on the HEPA filters, and the resulting significantly reduced performance and efficiency.

During abnormal or accident conditions, HEPA filters could fail to perform their function of capturing hazardous materials, or, structurally fail releasing significant quantities of the hazardous materials they have accumulated (e.g. millions of curies of radioactive material or other hazardous substances).

The two items above on Aging and Quality Assurance Testing exemplify the real concerns of potential filter failures under adverse conditions.

The DOE is developing a standard to provide guidance concerning assumed efficiencies of HEPA Filters during abnormal and accident conditions. A draft of the standard has been completed and is starting through the DOE Standards Review Program. Information on the standard is presented elsewhere in these conference papers.

Nuclear Air Cleaning Handbook: The Nuclear Air Cleaning Handbook (ERDA 76-21), subtitled Design, Construction, and Testing of High Efficiency Air Cleaning Systems for Nuclear Applications, first issued in 1976 and used throughout the air cleaning community, needs updating to reflect latest technology improvements as well as changes in applicable codes and standards.

The Handbook, to promote development of safe, effective HEPA Filter systems, is currently undergoing revision by the DOE. Professor Mel First, at the Harvard University Air Cleaning Laboratory, is heading the revision effort.

Portable Air Cleaning Units and HEPA Vacuum Cleaners: Currently there are no standards that sufficiently cover special HEPA Filter applications such as the portable air cleaning units, radiological vacuum cleaners, and small in-line HEPA filters. Existing standards apply primarily to facilities.

Without specific direction and guidance, facilities are failing to properly test and use the filters. For example, some facilities run their portable HEPA units at twice the rated velocities of the HEPA Filters. This invalidates the certification of the filters and causes the filters to have an efficiency less than that required to qualify as a HEPA Filter. In addition it puts extra pressure on the filter structurally.

Failure of these type filters ejects hazardous materials directly into the workplace or environment, usually without any means to detect that this is happening since portable units do not usually have the type of backup instrumentation used with permanent facility applications.

During fiscal year 1995, the DOE intends to develop a guidance standard for these type HEPA Filter applications. In the meantime, until the standard can be developed, temporary minimum guidance on these special applications has been included in Article 464 of the DOE Radiological Control Manual.

Fire Suppression Systems: Many large HEPA filter systems have water fire suppression systems installed inside the filter plenum. Moisture can have a detrimental effect on the fiberglass paper media weakening the filter structurally and speeding the aging process.

Fire systems are routinely tested with the water impacting the filters. In addition, full use of the fire suppression system could cause significant damage, leakage, and/or filter blow out causing release of hazardous materials. On the other hand, a fire could also release hazardous materials.

The DOE is currently looking at this issue relative to new materials and technology.

Substitute Aerosols: Di-Octyl Phthalate (DOP) continues to be used as the challenge aerosol for testing HEPA Filters, although questions exist concerning its potential health hazard to humans.

Information on DOP, as a potential carcinogen, has been issued by the DOE Office of Health in a Health Hazard Alert dated May 1994. The Alert says that the DOE "is evaluating to what extent the current use of DOP poses a health hazard within the DOE complex." The Alert further requests employers to limit exposure to employees, but does not prohibit use of DOP.

However, the EPA, in the Federal Register, Vol 58, No. 191, Tuesday, October 5, 1993, deleted di-n-octyl phthalate (DnOP) from the list of toxic chemicals under section 313 of the Emergency Planning and Community Right-to-Know Act of 1986 (EPCRA). Based upon the rationale used to justify the action, there appears to be a loosening of requirements concerning the phthalates.

Substitute aerosols are an alternative and in some cases DOE contractors are already using substitutes such as Emery 3004.

The DOE is currently evaluating the issue.

HEPA Filter Specifications: The DOE has found that its facilities are not consistent in the use of HEPA Filter Specifications. For example, some facilities use out-of-date or otherwise incorrect specifications. This has resulted in problems at the facilities and at the filter manufacturers.

During visits to media and filter manufacturing plants over the last year, the manufacturers complained it is difficult to maintain a wide range of filter designs, materials, and certifications when the specifications vary among various organizations and facilities. For example, having such a wide range of filter materials tested and put on the Qualified Products List (QPL) is time consuming and costly.

The DOE is currently revising and revamping an internal standard on Specifications for HEPA Filters, and looking at newer materials that might provide greater structural strength and resistance to fire and heat. A draft of the standard has been completed and is about to be introduced into the DOE Standards program for review/issue.

In-Place Testing of HEPA Filters: The majority of DOE and non-DOE facilities (approximately 80% by some estimates) were built before the development of current applicable Standards for the design and in-place testing of HEPA Filters. Many of the facilities, with a wide assortment of designs, cannot meet the current standards and use a variety of self developed, non-consistent approaches.

An effort is underway within the DOE to develop a guidance document to provide for an improved, consistent, in-place testing program. Work on the in-place testing guidance document should be completed during FY95.

III. Conclusion

The above discussion on HEPA filter issues covers only a few of the many areas that can be addressed to collectively improve the design, use, and testing of HEPA filters.

Examples of other topical areas include gel seals, metal filters, new materials, delta pressure meters, laser spectrometers, training, and use of scrim materials.

The intent of this overview is to illustrate that when all the HEPA filter issues are taken as a whole, the conclusion is easily reached that numerous areas exist for improvement of the HEPA filter program as a whole.

DISCUSSION

PALMER: You stated that DOE is revamping an internal standard on specifications for a HEPA filter. One of the problems with the old standard is that it was never formally issued as a compliance document. What efforts will DOE make to more formally issue this specification, more like an order to follow when you specify HEPA filters, instead of just handing it over as a kind of guidance document? That is my first question. My second is, will it address both portable HEPA filtration devices and stationary devices? A lot of the vacuum cleaners you talked about use small round filters instead of the standard square filters mentioned in the specifications.

LEONARD: Vacuum cleaners are what we call a special application. We have an item in one of our contracts to work on it over the next fiscal year. We recognize that it is different, and we will probably pick up the specifications on it as we work on the standard. It won't be in the specification standard we are working on right now. For the specification standard we are working on now, we will have everybody in the Department of Energy (and I know there are some of you outside the Department of Energy who are interested) take a look at it. We are trying to find something that will cover most of the applications. I don't think we can cover them all, but we can cover most of them. It is amazing, but manufacturers are still getting requests to build filters to the old standard that we were using back in the early 70s for asbestos filters.

FIRST: I acknowledge the deterioration of filters in long storage. Is there any work going on by any manufacturers or government agencies to define the deterioration mechanism? In other words, instead of wringing our hands that the filters are failing, is anybody trying to find out how to keep them from failing as they age?

LEONARD: The manufacturers we have talked to have said they have been looking at some new materials and they have some new ideas. It is a tough question. We have been trying to figure out how to determine what the maximum storage time should be but I don't think there is a way to determine it. We can come up with a number, and just put it out there, but I don't think that is a way to go because the variables are so many: heat, moisture, pressure, chemicals. You can just go on and on. Our concern is, how far should we go with this. You can't simply let them sit there forever. Eventually, they will fail on you. The mechanism to determine the failure is an in-place test that is performed annually, or at some other interval. How do you know when the filter is going to fail? We are also very concerned about the structural strength of the filter. It can pass the in-place test and still be about ready to fall apart. That is one of the things we are concerned about with the aging process. So, if you have an accident, and you are depending on old filters to perform according to the specifications it had when new, it is not going to happen.

CROSBY: As a follow up to Palmer's question, a Battelle engineer at Oak Ridge named Val Bouchard (615-220-2654) has, in fact, a complete set of specifications for testing both the HEPA filter in vacuum cleaners and negative pressure filtration units. He has also written a procurement specification.

LEONARD: I am aware of that. We have been working with these people on some of the tests

and on other things we are doing. There are quite a few other issues but that is one of the facilities that has had problems with filters. They use a lot of HEPA filters, and they have a lot of old designs.

Let me leave you with this. I have come to the conclusion that HEPA filters are like small print. A man went to his attorney and said, "I am buying this house, but look at the small print, page after page of it. Should I read it all?" The attorney replies, "Well, technically you really should read all the small print, but I can give you the bottom line. The bottom line is this, you keep making your payments and there isn't anything in that small print that can hurt you. But if you stop making those payments, there isn't anything in that small print that can save you." And that seems like about where we are headed.

A Novel Permanently Magnetised High Gradient Magnetic filter using assisted capture for fine particles

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Abstract

This paper describes the structure and properties of a novel permanently magnetised magnetic filter for fine friable radioactive material. Previously a filter was described and tested. This filter was designed so that the holes in the filter are left open as capture proceeds which means the pressure drop builds up only slowly. This filter is not suitable for friable composite particles which can be broken by mechanical forces. The structure of magnetic part of the second filter has been changed so as to strongly capture particles composed of fine particles weakly bound together which tend to break when captured. This uses a principle of assisted-capture in which coarse particles aid the capture of the fine fragments. The technique has the unfortunate consequence that the pressure drop across the filter rises faster as capture capture proceeds than the filter described previously. These filters have the following characteristics: (1) No external magnet is required. (2) No external power is required. (3) Small is size and portable. (4) Easily interchangeable.(5) Can be cleaned without demagnetising.

I Introduction

This paper describes a permanently magnetised magnetic filter which uses the principle of assisted-capture into order to capture and retain extremely fine particles of PuO_2, for example, generated in the processing of materials in the nuclear industry.

As described in a previous paper[1], the basis of the use of such a filter in the nuclear industry relies on the fact that much of the radioactive material is paramagnetic and the permanently magnetised magnetic filters described in this paper have the same important features as those described previously [1], namely: (1) No external magnet is required. (2) No external power is required. (3) Small in size and portable. (4) Easily interchangeable. (5) Can be cleaned without demagnetising. Two uses have been proposed for such a filter:

(A) As a cleanable pre-filter in front of an absolute filter in order to reduce the load on the absolute filter and thereby increase its useful life. This greatly reduces the absolute filter diposal cost. This filter can also be used in de-commissioning to capture the magnetic fume produced by cutting steel with a plasma torch.

(B) To improve housekeeping in a glove-box by preventing the spread of fine PuO_2 through the box by constantly circulating the air or part of the air through the filter. By adopting this measure operator exposure could be reduced.

In the earlier paper [1] a different design for a permanently magnetised magnetic filter was presented. Experimental work with this filter was carried out at BNFL, Sellafield and in Mol, Belgium using Cr powder which is a good magnetic simulant for PuO_2. The extractive and retentive capability of this filter on Cr powder was found to be excellent down to a particle diameter of 0.5 μm which was the limit of the measurements[1]. Subsequently, work was carried out on plutonium oxide at BNFL, Sellafield which revealed inadequacies in the filter design when treating real plutonium dioxide.

The magnetic filter consisted of a perforated sheet of Chromindur wound on a central tube and consisted of approximately 76 turns. The particles pass out of holes in the central tube and then pass through 76 layers. The Chromindur was magnetised parallel to axis of the tube and consequently parallel to the plane of the sheet. Under these conditions when particles are captured by the sheet, it has been demonstated **the particles are repelled from the holes** in the Chromindur sheet and captured on the material between the holes[2]. This configuration is illustrated in Fig 1. This meant for a material such as chromium powder passing through the sheets under radial flow, as particles are captured, the holes remain open so that the pressure drop increases very slowly as material is captured by the

filter. Ironically, this advantage of the design also carried a severe weakness when dealing with a friable material which can generate very fine particles. This property of the configuration which retains particles in the filter in such a way that the holes are left open, often considered very desirable in a filter, is the main reason why the fines generated by the friable plutonium oxide pass easily through these open holes without being retained. The evidence for this conclusion comes from a number of sources. Scanning electron microscopy shows the particles of PuO_2 are composed of angular platelets arranged as rosettes around a nucleation centre. Ultrasonic agitation or other mechanical processes can be used to break up the agglomerates. Another observation was that the $Pu0_2$ agglomerates with other dust particles so there is a small amount of active particles together with non-active material. If this other dust is non-magnetic it could be a problem for the magnetic filter as the magnetic susceptibilities of the composite particles will be reduced. A number of these properties of PuO_2 were observed by Moss, Hyatt and Schulte[3].

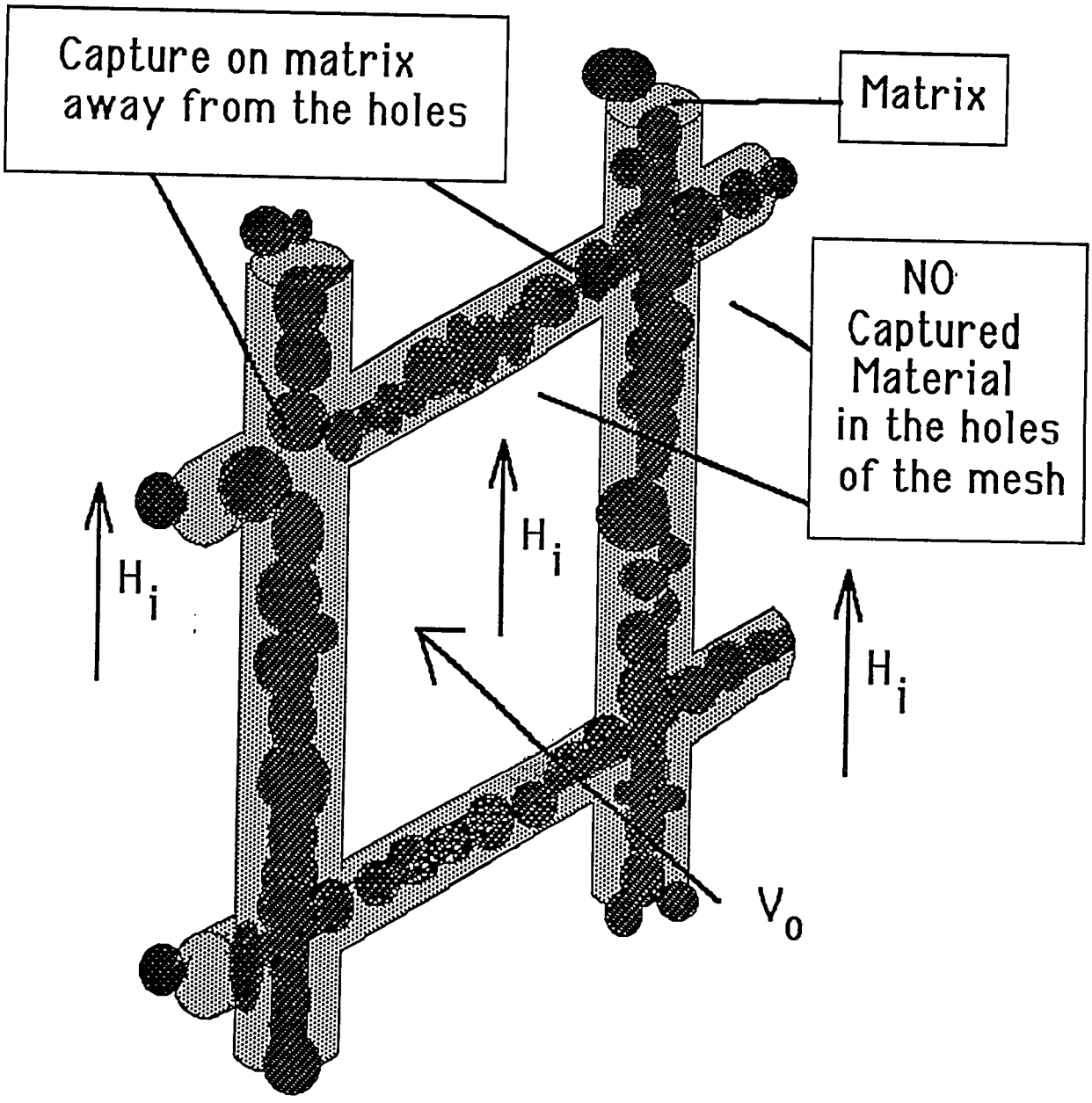


Fig 1 Shows a part of the permanently magnetised matrix. When the internal field H_i (demagnetising field) is in the plane of the mesh, no paramagnetic particles are captured in the holes. This means there is a small increase pressure drop as capture proceeds. The relatively coarse captured material is ineffectively placed in order to mechanically capture and entrain the fines which are lost.

The novel design presented here has a different configuration which should solve these problems for friable materials such as PuO_2 and this is done by making the **holes in the mesh attractive** to magnetic particles. This occurs when the demagnetising field H_i is perpendicular to the plane of the mesh and in this part of the magnetisation curve, the magnetisation of the mesh is in the opposite direction to H_i . This configuration is illusrated in Fig 2. The relatively coarse particles now are captured within the holes. The captured particles are situated in a very good position to mechanically filter the fine particles from the flow. This means the fines are retained by the coarse particles captured within the holes and the retention is aided by the observed property of PuO_2 to readily

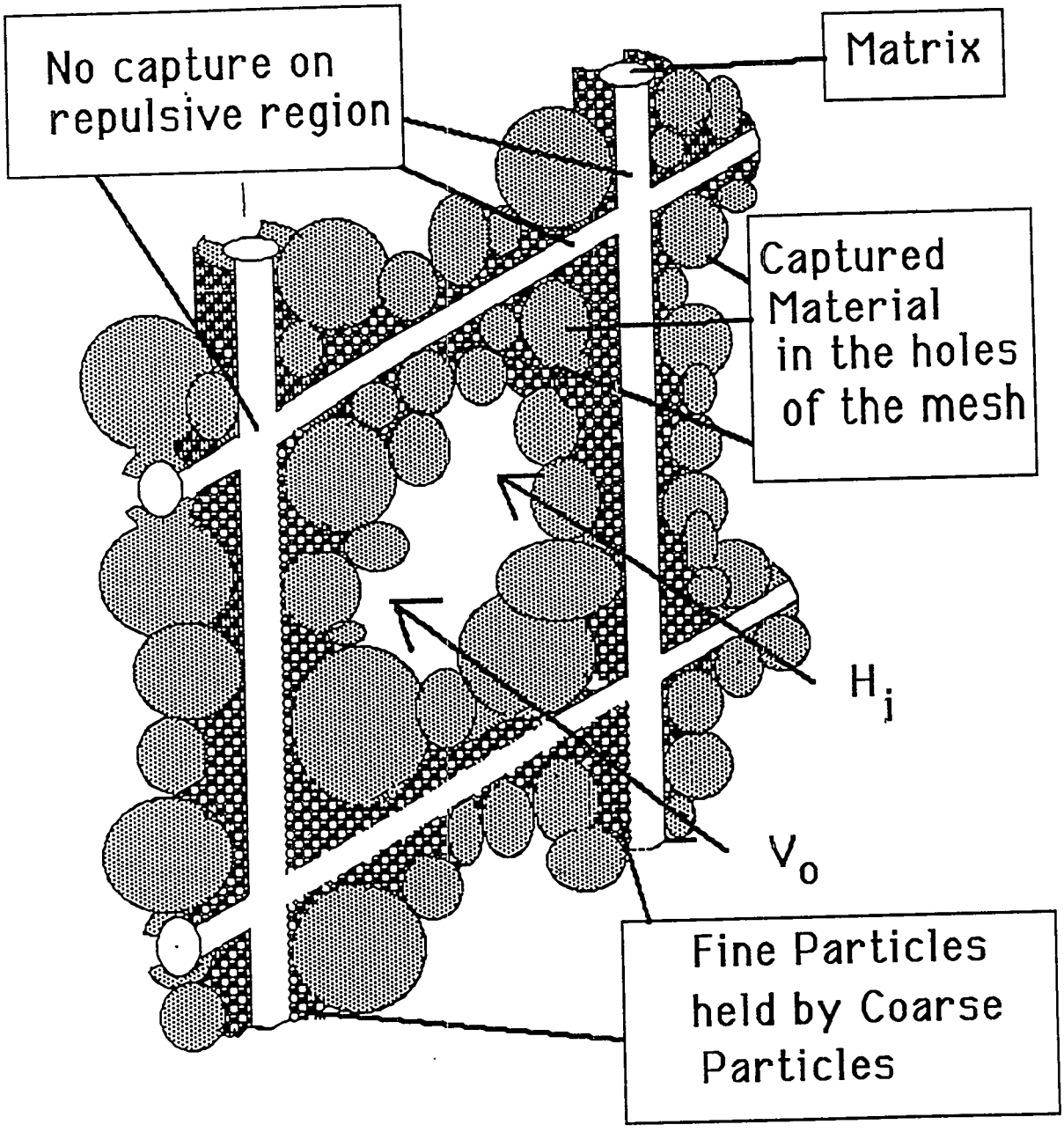


Fig 2 Shows part of the permanently magnetised matrix with the new configuration described in this paper. When the demanetising field H_i is perpendicular to the plane of the mesh and in the portion of the magnetisation loop where the magnetisation of the mesh is in the opposite direction to H_i, as it is here, the matrix surface within the holes becomes attractive to paramagnetic particles. These particles are favorably placed to capture and retain fines particles by mechanical filtration and magnetic forces, as shown schematically here.

agglomerate [3] and by the magnetic forces.

Further, as the mesh is not rolled, the requirement for mechanical flexibility is not required for the mesh. It is therefore possible to use materials with much better magnetic properties than Chromindur. The more concrete choice of materials for the filter are discussed in section IV below.
In the second section of the paper the theory of magnetic separation of monosize particles by permanently magnetised material is presented. In the third section the theory of **assisted capture** is discussed for the case where a wide range of particle size is present in the material to be filtered. The fourth section presents a new matrix configuration which will allow **assisted capture** to take place within the filter and which should considerably improve the performance of the filter.

II. Magnetic interaction of a particle and a magnetic wire

The theoretical model assumes a flow of paramagnetic particles carried by a fluid past a magnetised wire. We assume that the wire is ferromagnetic and hysteretic. The magnetic field in which the wire sits is the internal demagnetising field H_i of the whole assembly of hysteretic wires which have been magnetised to saturation after which the external applied field is reduced to zero. This leaves the residual demagnetising field as the magnetic field in which the wires and the paramagnetic particles sit. In this situation

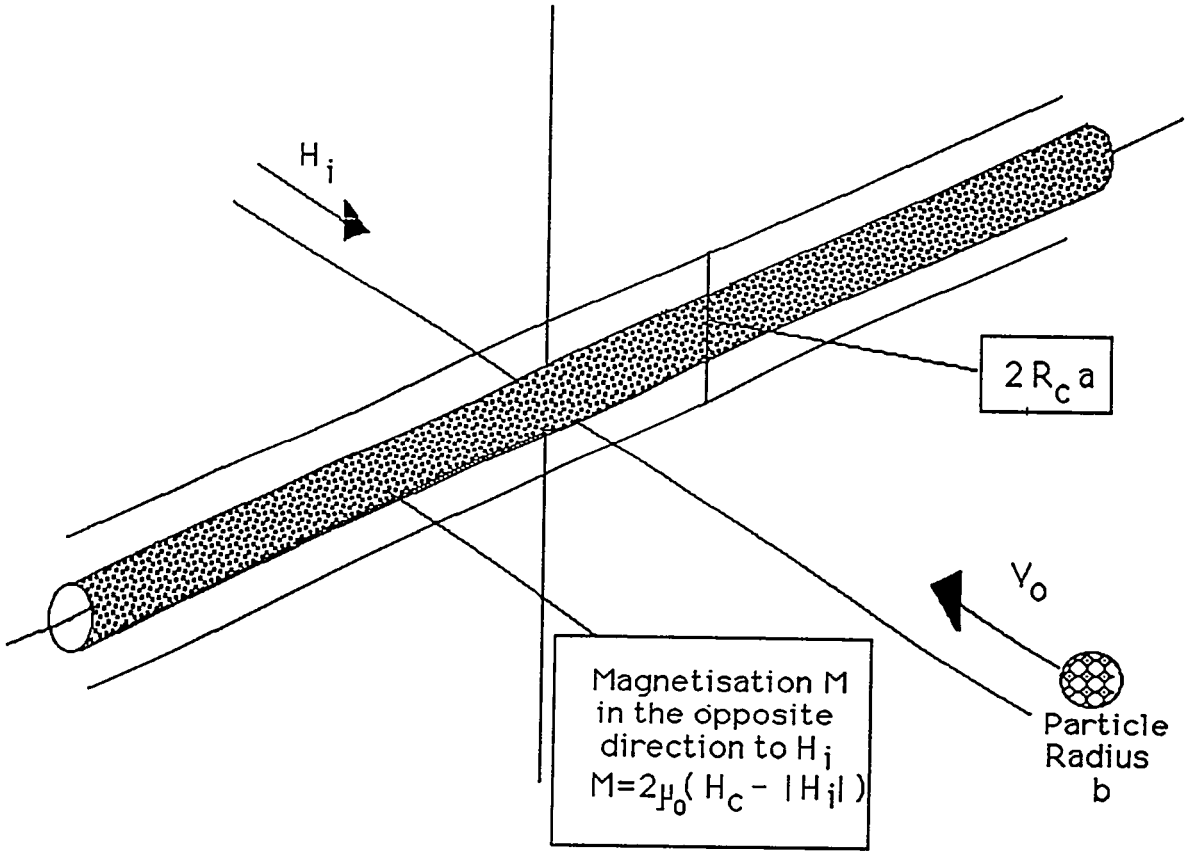


Fig 3 Shows the particle of radius b carried by a fluid of background velocity V_o moving parallel to the x-axis along which a magnetic field H_i is applied. It is necessary to determine the capture cross-section $2R_ca$, as shown. R_c is referred to as the capture radius.

the magnetisation M of the wire is in the opposite direction to the local magnetic field H_i and is given by $M = 2\mu_o(H_c-|H_i|)$ where H_c is the coercive force and $\mu_o = 4\pi\ (10^{-7})$H/m, the permeability of free space. It is assumed that the wire is perpendicular in direction to this local field. From this following the methods by Watson[4] and more recently by Lawson, Simons and Treat[5] and Simons and Treat[6], the equations of motion can be calculated and the capture cross-section $2R_ca$ for the particle can be calculated. Here R_c is termed the capture radius and a is the wire radius. The geometical significance of the capture cross-section is illustrated in Fig 3.

The equations of motion determining the particle trajectories show and, in consquence, the capture of particles by a fibre, in are background magnetic field H_i, are determined by three parameters;

i) V_m/V_o

where V_m is the 'magnetic velocity' containing all the important magnetic parameters in the system,

and V_o is the flow velocity. V_m is given by;

$$V_m = (2/9)\chi\, b^2\, M|H_i|/\eta a \qquad (1)$$

where χ is the difference in susceptibility between the fluid of viscocity η and the particle of radius b. M is the magnetisation of the ferromagnetic wires of radius a and H_i is the local magnetic field, which for the case of the permanently magnetised matrix is referred to as the internal field H_i. Then using the expression given above for the wire magnetisation M, equation (1) becomes

$$Vm = (4/9)\chi\, b^2\mu_o(H_c - |H_i|\,)|H_i|/\eta a \qquad (1a)$$

V_m is maximised with respect to H_i when $|H_i| = H_c/2$.

(ii) The short range parameter K, given by;

$$K = M/2\mu_o H_i = \{\, H_c/|H_i|\,\} - 1 \qquad (2)$$

If the wire is not magnetically saturated then K = 1. If V_m is maximised then K = 1, as the capture radius R_c depends on V_m/V_o and on K, the maximum in R_c does not necessarily occur at $|H_i| = H_c/2$.

iii) Stoke's number, N_{st};

$$N_{st} = (2/9)\rho_p b^2 V_o/\eta a = (Re/9)(\rho_p/\rho_L)(b^2/a^2) \qquad (3)$$

ρ_p and ρ_L are the particle density and the fluid density, respectively, Re is the wire Reynolds number where $Re = 2\,\rho_L V_o a/\eta$. The efficiency of capture by the fibre can be adjusted by changing these parameters.

In the case of a permanently magnetised filter the optimisation of these parameters is crucial. To understand why this is so some background is necessary. In HGMS, weakly magnetic particles are captured by the combination of high fields and high field gradients. This is possible through the use of large electromagnets, or superconducting magnets so large values of V_m can be obtained. However there are some disadvantages which must be considered. One of these concerns the short range term, K, and the fact that for a soft ferromagnet with low hysteresis, K can only have a values between 0 and 1. In the case of a permanently magnetised filter the magnetic field is significantly smaller than that experienced in HGMS, however, because of magnetic hysteresis, it is possible to create values of K much greater than unity. Consequently, the optimisation of the field and magnetisation is crucially important. Given this optimisation, magnetic particles passing the filter elements will be captured if they are inside the capture radius R_c of that element.

The magnitude of R_c can be determined from V_m, V_o and K. It is possible to build a model relating relating K to R_c for various V_m/V_o ratios. Fig 4 shows a graph of the capture radius as a function of the ratio of magnetic velocity to flow velocity. This graph has been obtained from data calculated by Watson[7]. These values were calculated for the case where dependence on the Stokes number is small and negligible.

It is interesting to look at the differences between conventional HGMS and HGMS with hysteretic materials. In conventional HGMS, the surface of the wire becomes divided into four regions, two of the regions are attractive to paramgnetic material and two repulsive to paramagnetic material. In hysteretic HGMS, the attractive regions become repulsive and vice versa. This has been discussed

previously by Boorman, Watson and Bahaj[2].

III. Theory of Assisted Capture

III.1 Additional factors occuring in the magnetic separation of polydisperse systems

The theory discussed here has been presented in detail eleswhere[8], however, because of the importance of the theory to the proposals made here, it appropriate to tto review the work in some detail.

Single-wire cells have been described by Watson & Rassi[9] and such cells have proved extremely useful in understanding the separation of many mineral systems. The particle trajectories and the magnetic build-up can be observed and recorded using a microscope, video camera and a video recorder. A video "frame grabber" in conjunction with a computing system has been used by Bahaj, Ellwood & Watson[10] to determine the magnetic properties of individual micron-sized particles from the particle trajectories near the wire. Watson and Li used a single-wire flow cell to investigate the

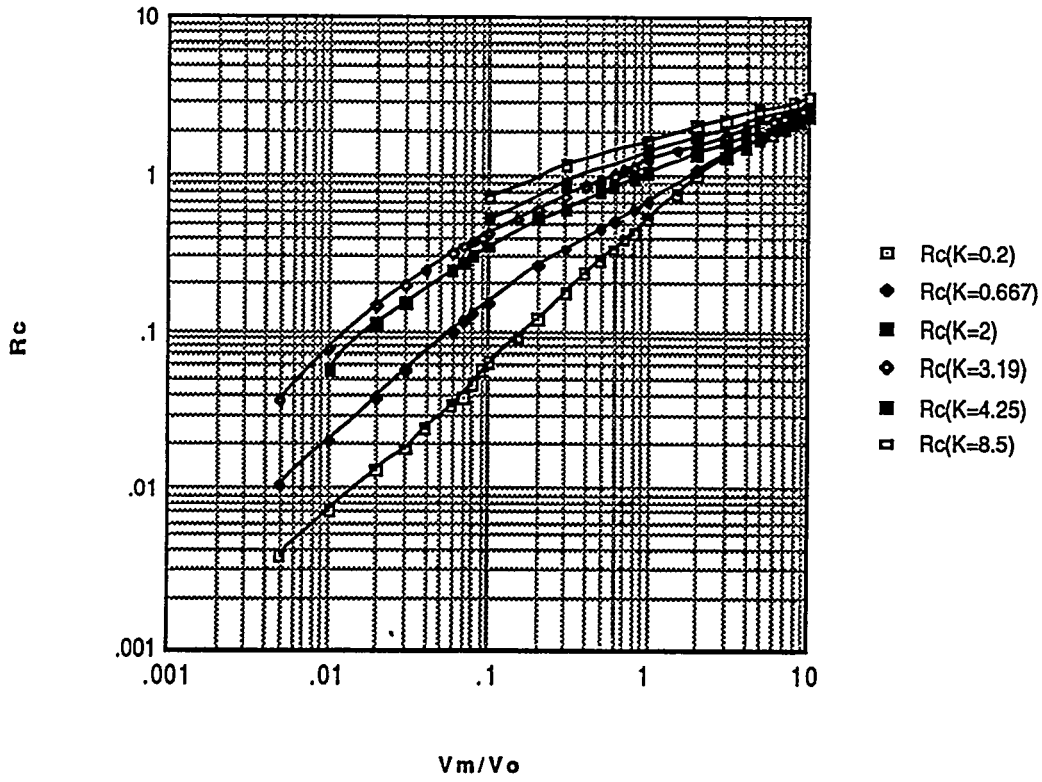


Fig 4 Shows R_c versus V_m/V_o for various values of the parameter K. For normal soft ferromagnetic material K can never be greater than 1, but for hysteretic material values of K>>1 are possible.

capture of polydisperse particle systems onto a ferromagnetic wire magnetised by a uniform applied magnetic field. The novel feature of the cell used by Watson & Li[11] is that when the wire is loaded, the channel, containing the wire and the captured magnetic material, is frozen and the ice core containing the magnetic deposit can be removed for further detailed examination. With this technique Watson and Li[11] effectively demonstrated the occurence of mechanical entrainment of non-magnetic material within the captured magnetics on the upstream side of the wire.

When particles collide with the surface of the magnetic deposit on the wire, they are acted on by a number of forces, such as, the viscous force due to the slurry flow which acts to drag the particles from the surface of the deposit, frictional and surface chemical forces which can hold the particles on

the surface. In the operation of a separator normally the slurry would be optimally dispersed to minimise the adhesion between the different mineral species to improve selectivity but this is about the only general rule and in filtration problems, there would be considerable advantage to have the particles coagulate. Here the frictional term is of most interest. It was clearly observed by Watson and Li[11] means of the microscope system that when a slurry containing a mixture of paramagnetic and non-magnetic particles was passed through the flow cell with the field turned off few particles would deposit on the clean wire surface as the particles were easily swept away by the hydrodynamic force due to the slurry flow. It was observed however the magnetic deposit is rough on a scale of the deposited particle radius, as shown in Fig 5. If particles of smaller radius are present, they are able to penetrate into the deposit and be trapped there. If the particles are of radius less than half of that of the primary magnetic deposit, the the hydrodynamic drag on the particle will be greatly reduced and the probability of retention greatly increased. A model developed in section III.2 which includes some of the complicating factors of the filtration of a polydisperse particle distribution, that is a wide particle size distribution, in which mechanical trapping is included. This is attempted with a bimodal distribution in which all the particles have the same susceptibility.

In the simple force balance model of magnetic capture[12,13] the fluid drag on a magnetic particle in relation to the magnetic forces is such that $V_m/V_o < 0.62$, the particle can not be retained. The quantity f is ratio of the volume of material captured to the volume of the wire and the maximum value taken by f is f_{max} which depends on V_m/V_o. The factor f can increase until the azimuthal magnetic force is less than the azimuthal fluid drag force and this condition determines f_{max} so that when $V_m/V_o = 0.62$ then $f_{max}=0$ [12,13].

$$f_{max} = (1/\pi)(V_m/V_o)^{2/3}\int_{3\pi/4}^{\pi} | - \cos\theta |^{2/3} d\theta - 1/4 \qquad (4)$$

Effects of mechanical entrainment can be examined with this model. The particle size for the bimodal distribution is chosen so that for the small particles $V_m/V_o < 0.62$ which means they are normally swept off the wire as $f_{max} = 0$. However, the large particles have $V_m/V_o >> 0.62$ and consequently any fine particles which are captured must be entrained mechanically. This therefore one of the simplest system which enable the consequencies of mechanical entrainment to be considered. This model can easily be extended be extended to the case where the magnetic susceptibilities of the two sizes are different as the appropriate parameter characterising the particles in V_m/V_o is χb^2. To extend the model to the case where $V_m/V_o > 0.62$ for all the particles is more complicated and is not considered here.

This model gives very different results from the simple application of monodisperse theory to each particle size; the reasons for this are two-fold. First, mechanical entraiment was not included and second, that the capture radius R_{co}, for a bare wire, is modified when capture proceeds, that is when f>0, by multplying R_c by a function G(f) where $G(f) = 1/(1 + 4f)$[13], $R_c = R_{co}G(f)$ and contributions to f occur from all particle sizes which have been captured. The method of adding contibutions from each particle size only gives the correct answer when f=0 and G(0)=1.

The results are also extremely interesting for filtration problems when $V_m/V_o >> 1$ for the large particles while maintaining $V_m/V_o < 0.62$ for the small particles and when it is desired to filter all the particles from the stream. The results are completely at variance with the conclusions arrived at using the method of adding contibutions from each particle size. However, the model shows how a bimodal distribution can be optimally filtered which has important implications for the filtration of polydisperse systems.

III.2 The magnetic separation of a paramagnetic bimodal distribution of particles with equal values of magnetic susceptibility

Now we can consider a bimodal particle distribution where the particles have radii b and B where B>>b and V_o is chosen so that $V_m(B)/V_o >> 0.62$ and $V_m(b) < 0.62$. If the small particles impinge upon the bare wire, they will be swept off as $V_m/V_o < 0.62$, but the large particles will be able to form a 2-dimensional layer over the part of the surface of the wire surface which is attractive to paramagnetic,

particles. If the small particles of radius b<<B can pack between the large ones. These particles will not feel the full drag as determined in the Force-Balance model and therefore they can be retained with the holes and pockets in the surface of the deposited B particles which the studies with the single. This is shown schematically in Fig 5. If it is assumed that the particles are spherical, then in each layer the space between the large particles B which can accomodate the small particles b is a volume $(4\pi/3))(\beta_{eB}-1)B^3$, where β_{eB} is the packing factor for the particles B. The packing factor is obtained from the total volume of the deposit V_D as follows:

$$\beta_{eB}=V_D/n_p(4\ \pi/3))B^3 \qquad (5)$$

where n_p is the number of particles in the deposit. If we assume only the top half is accesible to the small particles b, the volume available in which the small particles can be retained for every large particle captured is $(2\ \pi/3))(\beta_{eB}-1)B^3$. In the calculations which follow it is assumed that the packing is not very dense so that each large particle that is captured provides a volume of $2B^3(2 - \pi/3)$ in which small particles can be retained. From this assumption $\beta_{eB}=6/\pi=1.91$. For these small particles the probability of retention is identical to their capture probability which is not zero when $f > f_{max}(b)$ because when the particles b are retained between the partlcles B then the hydrodynamic drag is greatly reduced so that f_{max} is determined by the large particles B as $f_{max}(B)$. If the number of small particles captured is greater than the volume available to them which provided by the incoming large particles, the excess are small particles return to the slurry. Stating this more precisely, if

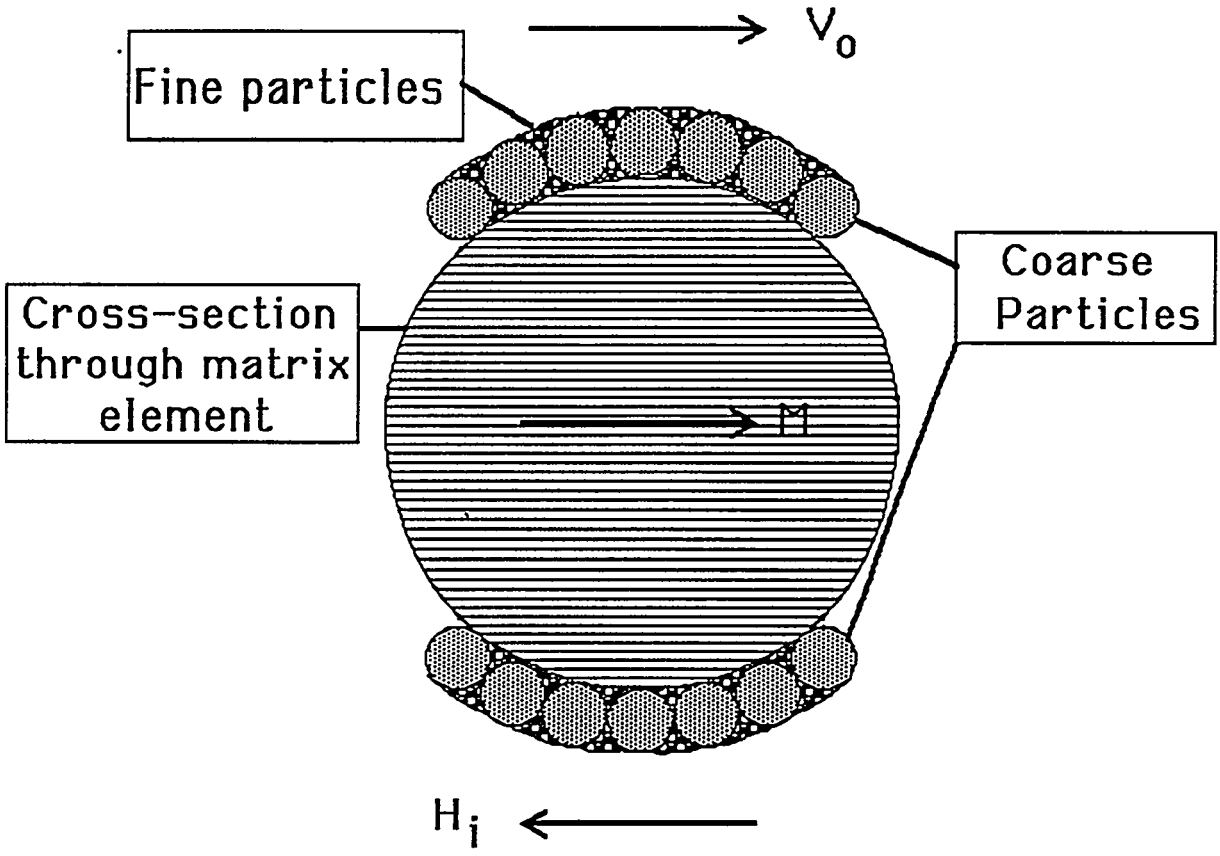


Fig. 5 Cross-section through the wire showing captured particles of radius B space shaded is occupied by particles of radius b wire .

$$dN_b(x)/dt < (\beta_{eB}/2\beta_{eb})(B/b)^3 dN_B(x)/dt \qquad (6)$$

where β_{eb} is the packing factor of the small particles into the volume provided by the large particles. $N_b(x)$ and $N_B(x)$ are the number of captured particles at position x/ m^3 of the matrix, for b and B respectively .
If $dN_b(x)/dt$, as determined by the separator equation with $R_c(b)$[13] given by $R_c(b)=(V_m/2V_o)(1/(4f+1))$ satisfies the inequality (6), as the space provided by the large particles B is occupied by the small particles therefore the value of f is completely determined by the large particles B. This simplifies the equations, as f=f(B) and f(b)=f(B) and therefore $R_c(b)=(V_m(b)/2V_o)(1/(4f(B)+1))$, for the small particles b, and for the large particles $R_c(B)=(V_m(B)/2V_o)(1/(4f(B)+1))$. Combining these expressions for $R_c(b)$ and for $R_c(B)$ with the separator equation gives:

$$(dN_B(x)/dt)/(d\ N_b(x)/dt)=(R_B(x)R_{c\ o}(B))/(R_b(x)R_{c\ o}(b)) \qquad (7)$$

This equation only applies when the particles b do not fill all the space provided by the captured B particles.
If the inequality expressed in (6) is not satisfied, then $dN_b(x)/dt$ is given by
$d\ N_b(x)/dt=(\beta_{eB}/2\beta_{eb})(B/b)^3 d\ N_B(x)/dt$ and then excess particles b, arriving at the wire, are not retained and remain in the slurry. This expression for $dN_b(x)/dt$ together with equation(8) determines the ratio $R_b(x)/(R_B(x)$ at which the space provided by B is just filled and is given by

$$R_b(x)/(R_B(x)=(R_{c\ o}(B))/(R_{c\ o}(b))/((\beta_{eB}/2\beta_{eb})(B/b)^3) \qquad (8)$$

If B=2b then $R_{c\ o}(B))/R_{c\ o}(b)=4$ and if $\beta_{eB}=\beta_{eb}$ then equation (8) gives $R_b(x)/(R_B(x)=1$.

The behaviour of the separator can now be investigated within this model by solving the separator equations with the appropriate boundary conditions. Here the boundary conditions used correspond to usual operating practice in magnetic separation, namely

$$\begin{aligned} &R_B(x,0)=N_B(x,0)=R_b(x,0)=N_b(x,0)=0 && L \geq x \geq 0 \\ &R_B(0,\tau)=R_B(0),\ R_b(0,\tau)=R_b(0) && \\ &R_B(x,\tau)=R_b(x,\tau)=0 && \text{for } \tau < \varepsilon_o x/V_o \end{aligned} \qquad (9a)$$

These conditions correspond to the separator containing no particles initially and the flux of particles at the entrance remains constant in time ,until, after some time t_f, the required volume of particles have been fed and then the flux of particles is set to zero. This boundary condition is as follows:

$$R_B(x,\tau)=N_B(x,\tau)=R_b(x,\tau)=N_b(x,\tau)=0 \quad \text{for } L \geq x \geq 0 \text{ and for } \tau > t_f \qquad (9b)$$

We wish now to consider the capture of a bimodal system by the separator under boundary conditions outlined in equations (9a) and (9b). To illustrate the solution of this model, the following conditions were applied. The matrix occupied 5% of a cylindrical space that was 0.03 m in length with a crossectional area of $1m^2$. The applied field was 3 Tesla with a velocity of $1ms^{-1}$. The large particles have radius of 10 μm and the fine particles a radius of 4 μm and γ, the ratio of the volumes of coarse to fine is γ=1. The magnetic velocity of the coarse particles $V_m(B)=1.804$ and for the fine particles $V_m(b)=0.289$. After the total volume of material applied to the separator, normalised to the volume of the matrix α = 0.08, the concentration of $N_B(x)$, the concentration of the coarse particles B, and $N_b(x)$, the concentration of the fine particles b, as a function of position within the matrix is shown in Fig 6.

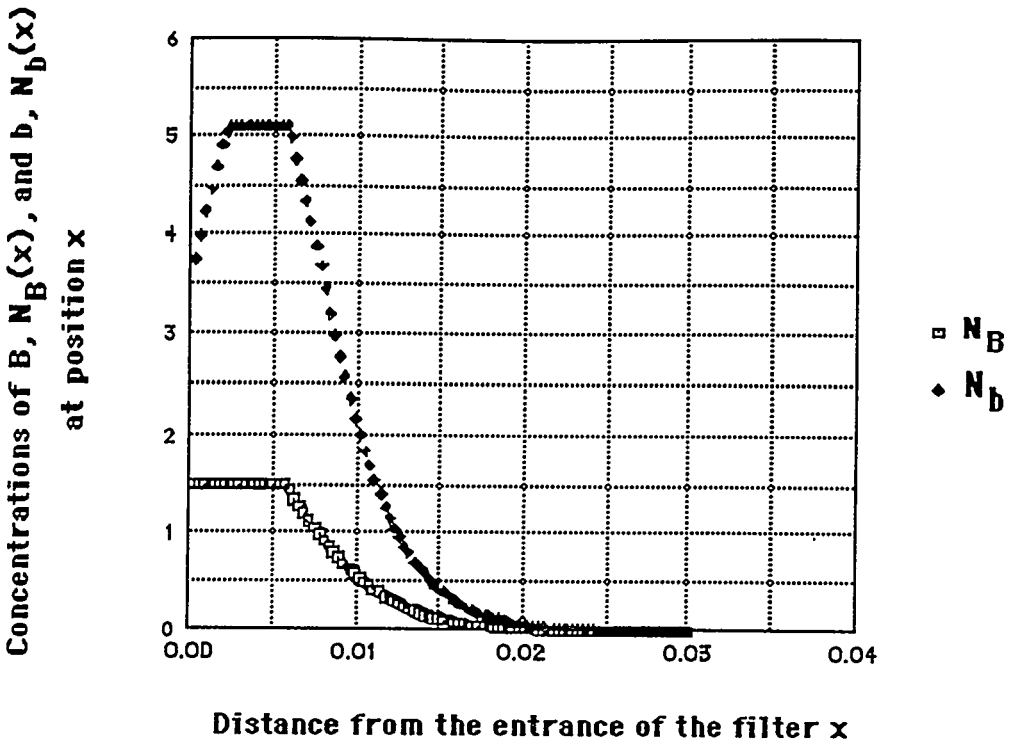


Fig 6 $N_B(x)$ and $N_b(x)$ the (number of particles).10^{-12} per unit volume of the coarse and the fine particles, respectively, in the separator at position X where X is the distance from the entrance of the separator. The magnetic velocities for the coarse and fine particles are 1.804 and 0.289 respectively. The volume of the particles fed to the filter divided by the volume of the matrix α = 0.08. γ, the ratio by volume of coarse to fine particles, is γ = 1.

The fraction of the fine particles transmitted through the separator is 0.781 if the coarse particles were not present then all the fines would be transmitted. All the coarse particles are retained within the separator. The flow rate used in the calculations was 1 m/s and the concentration was 0.04% by volume and the total volume of slurry passed was $0.3m^3$. The distribution of particles captured within the separator is substantially the same as for a more concentrated slurry of smaller volume provided that α, the total volume of the particles divided by the volume of the matrix and γ, the ratio by volume of coarse to fine particles, are held constant.

There are a number of features of interest in Fig 6, namely the concentration of the coarse particles in the separator $N_B(x)$ is initially constant with x and at the maximum value given by equation (8) of $N_B(x)=1.49(10^{12})/m^3$ with the fine particles b starting at x=0 with a concentration of $N_b(x)=3.76(10^{12})/m^3$, corresponding to a fraction of 0.672 of the total space provided within the coarse particle buildup which can be calculated from equation(4). The relative of buildup of the coarse and fine particles is determined by equation(7), the capture rate at x=0, is initially greater for the large particles B, but for greater values of x, the capture of the fine particles b increases with x until $(R_B(x)R_{co}(B))/(R_b(x)R_{co}(b))=1$ which, by equation (7) means that $dN_B(x)/dt)/(dN_b(x)/dt)=1$. This point is reached at x=0.002 after which the concentration of the large particles B is $N_B(x)=1.49(10^{12})/m^3$ and the fine particles b with a concentration of $N_b(x)=5.10^9(10^{12})/m^3$ which corresponds to a fraction 0.921 of the total space available to the fine particles b. Beyond x=0.006, the flux of coarse particles B has dropped so low that the separator is not filled to the maximum value possible and in consequence the number the fine particles begin to fill the total volume available to them but which decreases rapidly with x.

It is interesting to consider the transmission of the fine particles b as the total volume of the particles applied to the separator is increased, that is, as α is increased.

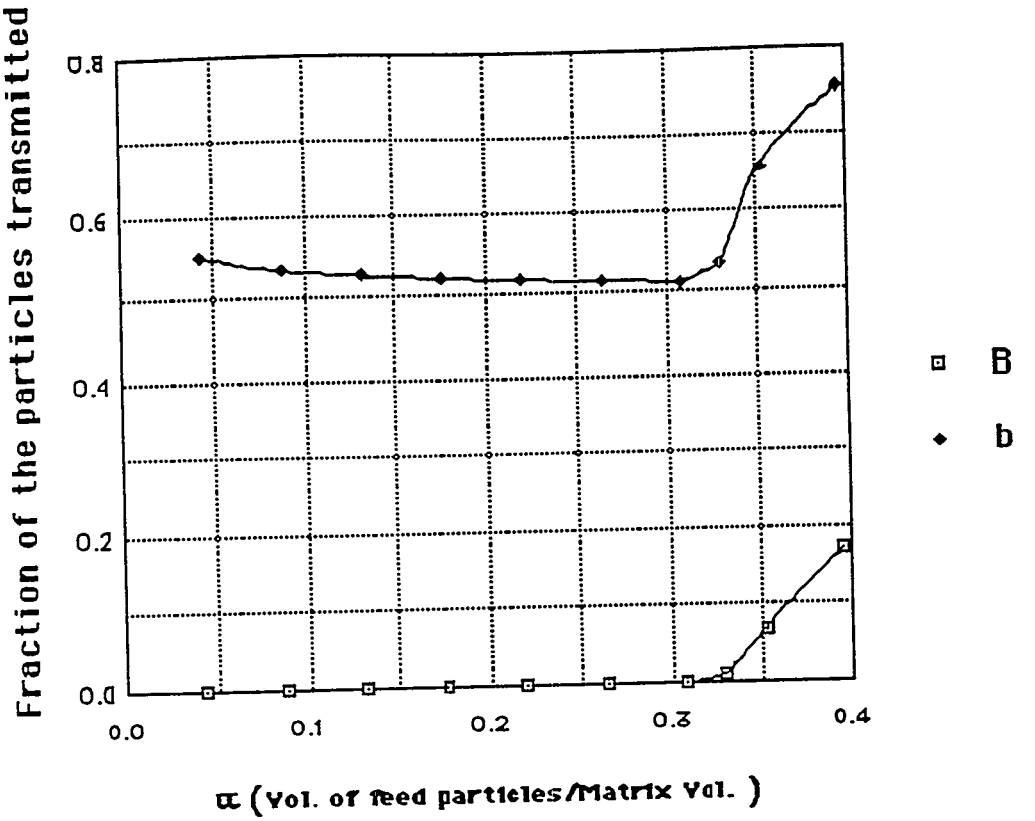


Fig 7 The fraction of the particles transmitted versus a ratio of the volume of the particles fed to the separator to the volume of the matrix for the coarse particles B and the fine particles b. The ratio of the volumes of coarse particles to fine particles γ is γ =10. The length of the separator is 0.1 m. All the fine particles b would be transmitted in the absence of the coarse particles.

As shown in Fig 7, there is a small decrease in the fraction of fine particles b transmitted as the value of α, the ratio of the volume of the particles fed to the separator to the volume of the matrix, is increased until breakthrough of the coarse particles B occurs when $\alpha = \alpha_{max} = 0.3$ which is the maximum efficiency for the filtration of the fine particles. If it is assumed that no coarse particles escape before complete breathrough then $a_{max} = f_{max}FL = N_BL$ where F is the fraction of the space in the filter occupied by the matrix and L is the length.

In the conventional treatment of magnetic separation for monosize particles, the number of particles leaving the separator decreases exponentially with the separator length L, it is interesting to consider the effect of length L on the transmission of fine particles in this assisted capture model. This can be done by examining the transmission of fine particles using the same ratio γ of coarse particles to fine particles of γ= 10, for a number of constant values of α, versus the separator length L. As shown in Fig 8 the fraction particles transmitted is almost independent of the length L, except when the coarse particles are beginning to breakthrough at high values of α. This is the type of behaviour expected for the coarse particles in the absence of an interaction with the fine particles, but in this case the transmission of the fine particles would be complete.

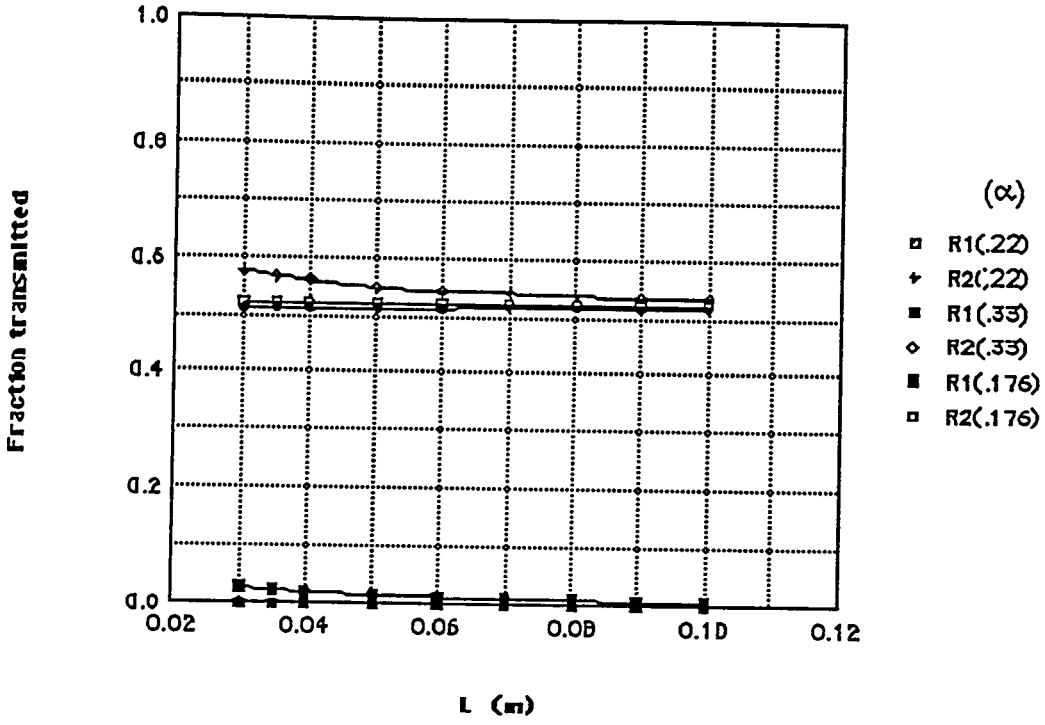


Fig 8 The fraction of particles transmitted through separators of various length L when the ratio of the volume of the particles in the feed to the volume of the matrix α is used as a parameter. The curves for the coarse particles are near zero but the values for the fine particles are near 0.5. The ratio of the volumes of coarse to fine particles γ is γ =10.

We can now consider reducing the number of fine particles transmitted, that is, we are attempting to filter out the fine particles by adjusting γ the volume ratio of coarse particles to fine particles. From Fig 8 it is known that the fraction of particles transmitted is almost independent of the length of the separator L if $\alpha << \alpha_{max}$. In the results of the calculations shown in Fig 9, the amount of the small particles b in each feed batch is held constant and this is fed mixed with various amounts of the coarse particles B so that γ reaches $\gamma = 10$ (so that every point on the curve represents a separate experiment. This sequence is repeated for for various applied magnetic fields. Fig 9 reveals that provided the coarse particles B do not breakthrough, the number of fine particles extracted depends almost entirely on γ, the ratio of the coarse particles to the fine particles. There is a weak dependence of number of small particles b retained in the separator on the applied magnetic field and, contrary to expectation from monodisperse theory, the number retained decreases slightly as the field increases.

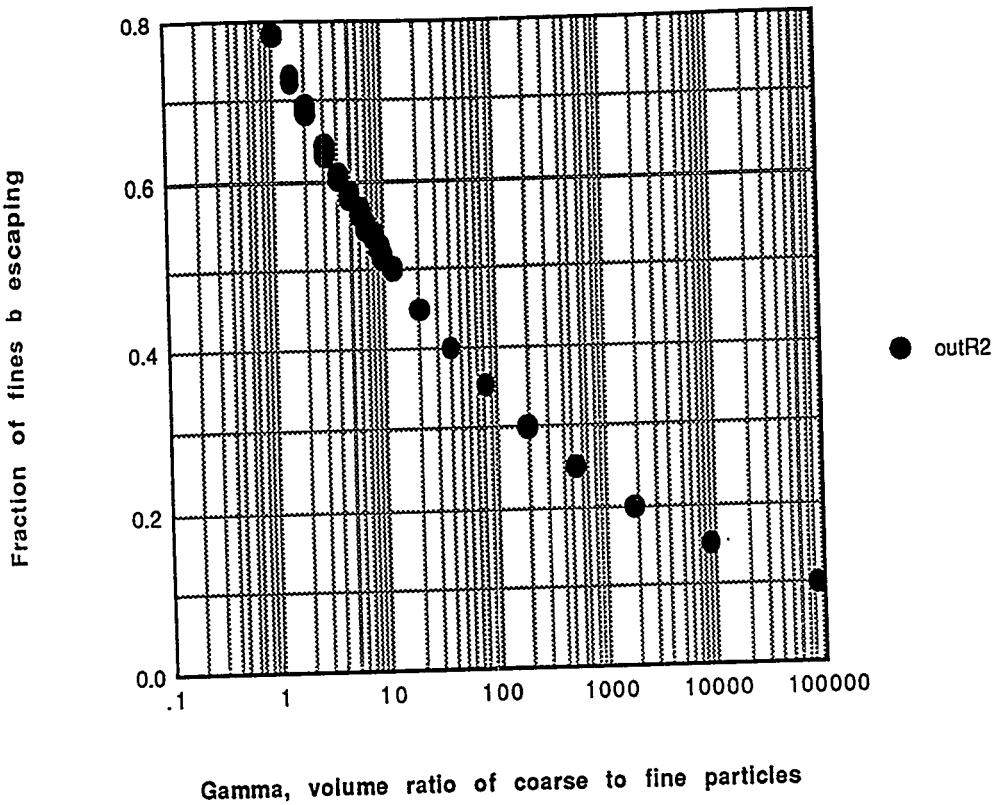


Fig 9 The fraction of small particles, radius b, transmitted through the separator versus γ, the volume ratio of coarse particles radius B to fine particles radius b. This conditional upon the the coarse particles not having filled the filter.

Summary of the effects produced on the filtration of fine particles

The result of assisted capture on the filtration and retention of the fines b can be summarised by saying the minimum transmission of b is achieved, for a particular filter, when the quantity of large particles B is just sufficent to achieve breakthrough. The fraction of b transmitted decreases as γ, the ratio of coarse particles B to fine particles b, increases but subject to the condition that breakthrough of B does not occur. High magnetic fields and long filters can therefore be used to decrease the transmission of the fine particles b by ensuring that breakthrough of B only occurs at smaller and smaller ratios of b to B, that is at large and larger values of γ.

IV. New design for the Permanently magnetised separator

Introduction

The development of this model requires an understanding of the filtration applications envisaged for the system. A filter is proposed consisting of a stack of expanded ferromagnetic metal mesh. The previous filter was made from a roll of Chromindur, as described in an earlier paper(1). The mesh was an hysteretic ferromagnetic, but it had the important property of is also flexibility with a low but usable coercive force of 300 Oersted. Chromindur is a product of Telcon Metals Ltd, Crawley, England. The new filter geometry as shown in Fig 10a does not require the property of flexibility so other materials, particularly those developed for the recording industry with their high values of coercivity, can be used.

Magnetic Materials for use in the filter

Basically their are two choices for materials, namely: Thin film materials and particulate materials of various types[14]. Most of the films available have have the residual magnetisation in the plane of the film which is not useful for the filter structure described here. The reason for this is the shape anisotropy energy of a thin sheet magentised perpendicular to the sheet. To stabilise the magnetisation perpendicular to the sheet, additional anisotropy must be introduced. This can be done in Co films by introducing Cr to a level of near 20 at. %[15]. At these compositions values of the coercivity of H_c = 65 - 105 kAm^{-1} can be achieved.

A better choice for a matrix material are metal particles, a material which again originates in the magnetic recording industry and in that industry has allowed high coercive forces to be achieved. For example, with pure iron particles, which are available commercially, values of coercivity are typically H_c = 91.5 kAm^{-1}. Pure iron particles are very reactive and must be passivated by exposing them to a mixture of N_2 and O_2. The final product is an acicular particle surrounded by a coating of Fe_2O_3 and Fe_3O_4.The stability of iron particles has been studied by Lee, Hu & Madudid[17]

Performance of the filter with a material composed of passivated iron particles

The matrix was magnetised perpendicular to the surface of the sheets in the stack; the sheets being perpendicular to the flow of the suspension and also parallel to the filter axis. The applied field was then reduced to zero, and, due to the hysteretic nature of the material, the filter remained magnetised with an internal field strength H_i of 21.8 kAm^{-1}. This determined from the dimensional ratio L/D = 1/3 having a demagnetising factor of N_z = 0.52. The magnetisation of the unit cell can be approximated by assuming the it is due to the strands perpendicular to H_i but in the opposite direction to it. Under these conditions the magnetisation of the strands is largely determined by the demagnetisation factor of the strand, approximately 2, and the coercivity of the material assumed to be 91.5 kAm^{-1}.

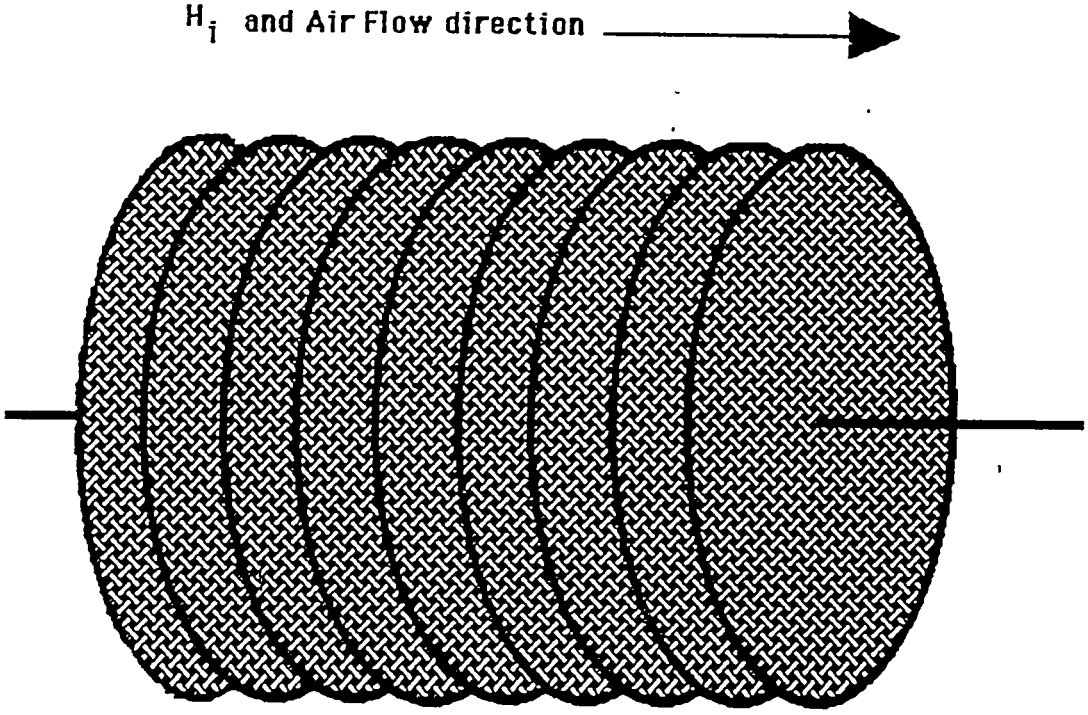


Fig 10a Structure of the prototype filter shows a few of the 124 sheets of circular cross-section in the form of a mesh stacked in a cylindrical tube made from a recording material composed of passivated iron particles. The sheets of material composing the filter are perendicular to the flow. The unit cell is the repeated unit generating the two-dimensional sheet. The separation of the sheets is 0.2 mm

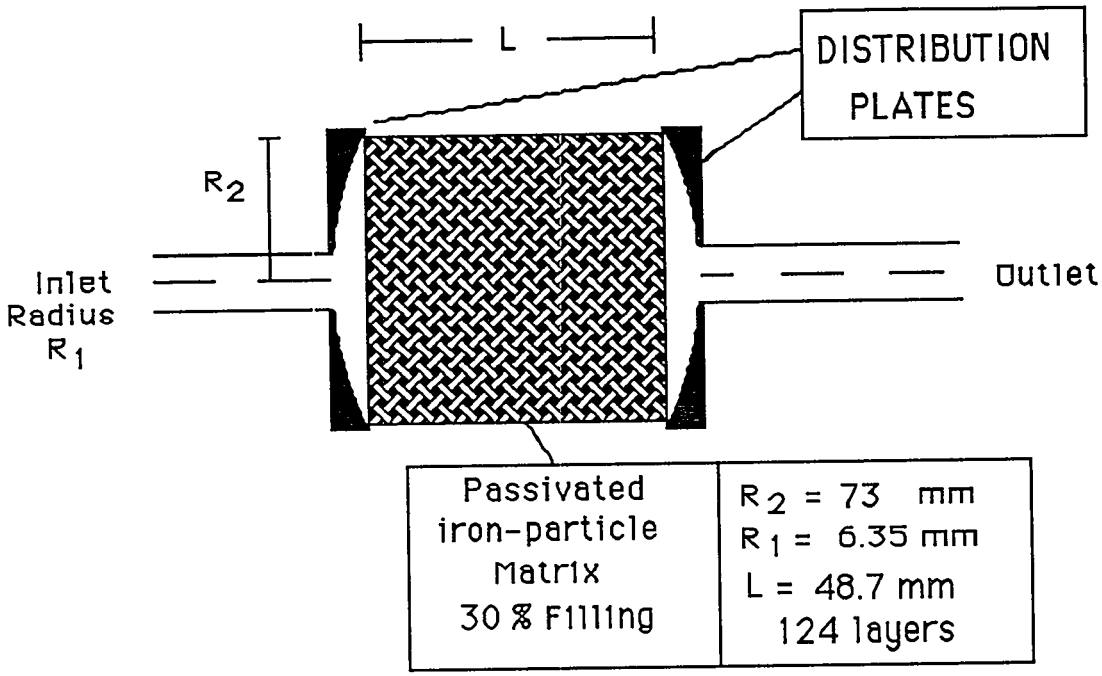


Fig 10b Shows a section parallel to and passing through the axis of the cylindrical filter. The gas flow is parallel to the cylindrical axis. The cross-section perpendicular to the flow is circular is proposed that L= 48.7 mm. The separation between the layers is 0.2 mm and the thickness of the sheet is 0.2 mm. The material is magnetised perpendicular to the plane of the sheets that is parallel to the axis of symmetry. The distribution plates are designed to equalise the Bernoulli pressures.

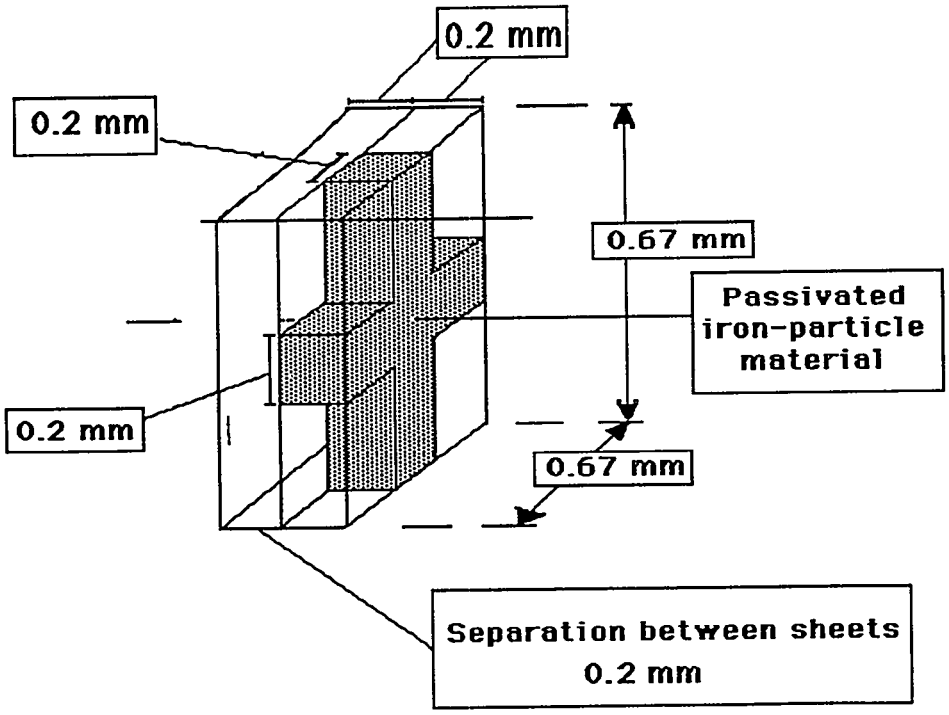


Fig 11 The unit cell of the structure is shown. The sheets of chromindur 0.2 mm thick are separated from each other by 0.2 mm and are generated by displacements of 6.7 mm parallel to the directions 6.7 mm shown. The stack is generated by displacements in the perpendicular direction.

Consequently, with the magnetisation of the wires M_W given by $M_W = 2(H_c + H_i)$, and H_i has the value $H_i = 21.8 kAm^{-1}$. The magnetisation M_W of those legs perpendicular to the field, on which most magnetic capture will occur due to the large field gradients, is therefore $M_W = 139.5\ kAm^{-1}$. The magnetic velocity V_m, given in equation (1a), can now be evaluated. For fibres of radius, a = 0.1mm, and particles of radius, b = 0.5 microns, with a susceptibility of 5×10^{-5} (S.I. units), the resulting magnetic velocity 0.038 mm/sec. In this case a value of $V_m/V_0 = 0.03$ would be suitable, i.e. a flow velocity of 1.25 mm/sec. The corresponding value of K, given in equation (2), is 3.19 and the value of R_c versus V_m/V_0 for K=3.19 is plotted in Fig 4. If we assume that the particles are captured on the legs of the matrix with the value of the capture radius shown in Fig 4, the effective cross-section per unit area, $\#_1$, for the first sheet is given by(using the dimensions shown in Fig 11 is $\#_1 = 0.204R_c$, therefore the fraction transmitted is $1 - \#_1 = 1 - 0.204R_c$. The number of particles captured in the next sheet was obtained for the if the second sheet by determining the extra cross-section provided by the second sheet when the second sheet is shifted parallel to the plane of the sheet from perfect registration with the first sheet. Assuming that all shifts have equal probability, the extra cross-section provided by the second sheet cabe averaged. The fraction transmitted by the second sheet is therefore $1 - \#_2 = 1.33\ R_c^2(1 - 0.3R_c)^2$. The next 123 sheets are all similar the total number of particles escaping is given by $(1- \#_1)(1 - \#_2)^{123}$. These expressions have been evaluated and are shown in Fig 12 for the case where the matrix is clean for a number of particle sizes.

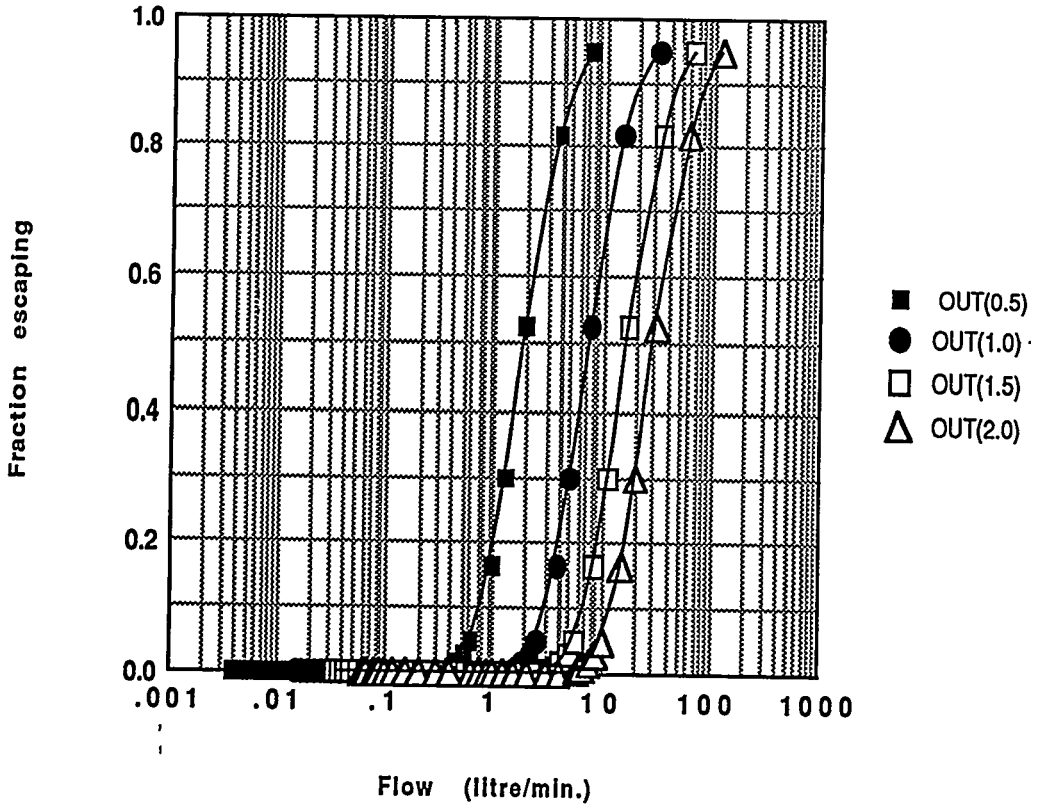


Fig 11 Shows the expected initial performance of the new filter orientation when it is clean.

Assuming a 30% filling factor, the total volume of the matrix is 2.4 x 10^{-4} m^3. As already mentioned, the mass loading envisaged in the flow to the filter is about 0.04gm/m^3. Nesset and Finch[16] introduced a quantity called the loading number which has been modified to incorporate interparticle frictional forces and this loading number has been used to calculate the total mass that can be theoretically captured by the filter. Using this approximation the total mass captured by the filter has been calculated to be 485 grams. If the mass loading in the gas to the filter is 0.04gm/m^3, and assuming a 30% packing factor for the material built up on the wires, this means the total processed volume will be 12,125 m^3, and, at a flow rate of 0.036 m^3/hr the overall lifetime of the filter will be 38 year. It should be realised that this is a small prototype filter. Practical filters will need to process somewhere in the region of 10 m^3/hr if they are to be useful.

The performance of the filter which is expected in the presence of assisted capture is shown in Fig 11.

VI. References

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DISCUSSION

WREN: Does the required magnetic field depend on the susceptability of the particles? If so, how will the required field be predetermined when the susceptability would change due to the change in the particle composition with time? Would the deposited particles affect the localized field? If so, what is the effect of that?

WATSON: For ordinary paramagnetic materials the changes in the magnetic field as build-up proceeds are small and can be neglected. If the materials were ferromagnetic they would affect local fields and would increase capture.

THE DEVELOPMENT OF A HEPA FILTER WITH IMPROVED DUST HOLDING CHARACTERISTICS

J Dyment[1] and C Hamblin[2]

(1) AWE Aldermaston, (2) UKAEA Harwell

Abstract

A limitation of the HEPA filters used in the extract of nuclear facilities is their relatively low capacity for captured dust. The costs associated with the disposal of a typical filter means that there are clear incentives to extend filter life. The work described in this report are the initial stages in the development of a filter which incorporates a medium which enhances its dust holding capacity.

Experimental equipment was installed to enable the dust loading characteristics of candidate media to be compared with those of the glass fibre based papers currently used in filter construction. These tests involved challenging representative samples of the media with an air stream containing a controlled concentration of thermally generated sodium chloride particles. The dust loading characteristics of the media were then compared in terms of the rate of increase in pressure differential. A number of "graded density" papers were subsequently identified which appeared to offer significant improvements in dust holding.

In the second phase of the programme deep-pleat filters (1,700 m^3h^{-1}) incorporating graded density papers were manufactured and tested. Improvements of up to 50 % were observed in their capacity for the sub-micron sodium chloride test dust. Smaller differences (15 %) were measured when a coarser, carbon black, challenge was used. This is attributed to the differences in the particles sizes of the two dusts.

1. Introduction

The high efficiency particulate air (HEPA) filters used in the United Kingdom nuclear industry are purchased against specifications[1] These documents detail the requirements for the materials of construction and the basic design, and prescribe general performance criteria. The increasing financial and technical implications of radioactive waste treatment, storage and disposal has heightened interest in the dust holding capacity of HEPA filters. In order to maintain compatibility with existing plant any increase in dust holding must be achieved without increasing the overall dimensions of current filter designs, or with detriment to operating parameters.

Previous workers have investigated and compared the dust holding capacity of various HEPA filter designs[2]. However, these studies did not specifically measure the performance of the filtration medium employed or the effects of different media in identical filters. It is possible therefore that a filter constructed with a medium possessing favourable dust loading characteristics could achieve longer life.

This report describes a method of rapidly testing the dust loading characteristics of small samples of candidate filtration media with a thermally generated sub-micron test aerosol. This procedure identified the potential advantages offered by a range of "graded density" papers developed by Whatman International Ltd. Results are also presented which show that improvements in dust loading are also evident for 1,700 m^3h^{-1} deep-pleat filters constructed with the most promising examples of these media.

2. Experimental Programme

The experimental programme was conducted in two phases. Initial experiments sought to establish the dust loading properties and the basic performance characteristics of a number of filtration media. The second phase involved the manufacture and testing of conventional deep-pleat HEPA filters which incorporated the most promising of the media identified.

2.1 Media Testing and Selection.

The strength properties and particle collection efficiency of the candidate materials were assessed for compliance with the current filter media specification[3]. A significant number of the media available (especially those manufactured from non-glass fibres) are at least an order of magnitude thicker than the glass fibre based papers currently used in HEPA filters. These materials were not included in the test programme due the practical difficulties of incorporating sufficient medium into existing filter size envelopes.

The dust loading characteristics of the samples of filter medium were investigated using the equipment shown in Figure 1. A test aerosol was generated by feeding a stick of sodium chloride into an oxy-propane flame to produce particles with a mass median diameter (mmd) of 0.20 μm and a geometric standard deviation of 1.4. The salt particles were then injected into the primary duct (250 mm ID) to become thoroughly mixed with the main system air flow. The flow rate within the primary duct was monitored by a standard orifice plate and controlled by adjusting speed of the fan. The provision of a rectangular HEPA filter, to protect the fan, allowed the air to be discharged back into the laboratory area.

A secondary (test) flow was extracted from the primary duct and passed through an 80 mm diameter disc of filter medium mounted 20 duct diameters downstream of the abstraction point. Flow within the test duct was continuously monitored and controlled to maintain a sample face velocity of 3 m min^{-1}. This value was selected to ensure loading times were not unnecessarily long; the results from experiments conducted at the usual operating face velocity (1.5 m min^{-1}) showed no significant differences for nominally identical samples.

Precise control of both of the air flow rates and the speed at which the salt stick was fed to the thermal generator was maintained throughout. In order to establish the reproducibility of the aerosol concentration, commissioning trials were conducted using samples of a "standard" glass fibre based paper commonly used in HEPA filter production within the UK. Variations of < 10 % were observed in the amount of salt collected over a 32 minute test. This permitted the relative performance of the candidate media to be assessed by simply comparing the changes in differential pressure with time.

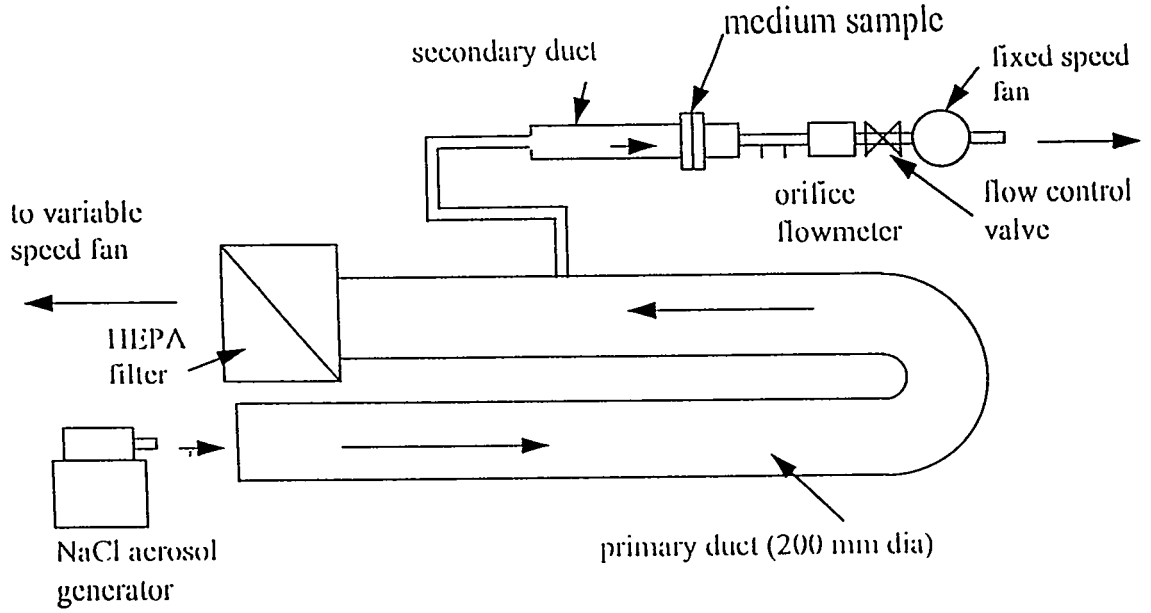


Figure 1 - Filter Media Dust Loading Rig (Schematic)

2.2 Filter Testing

The results from phase 1 of the programme were used to identify the most appropriate media to have incorporated into conventional deep-pleat filters, generally conforming to the requirements detailed in reference 1. The dust loading characteristics of these prototype filters were then established and compared with results obtained for deep-pleat filters containing a typical glass fibre paper.

The rig used (Figure 2) was originally constructed to meet the requirements of BS 3928[4]. It essentially comprises an open ended circular inlet duct, which receives the discharge from the aerosol dispenser, a standard filter housing and an outlet duct connected to the extract fan. Air flow rate was maintained at 1,700 m^3h^{-1} during dust loading, and the differential pressure generated across the filter was continuously monitored.

Filter loading trials were conducted using two test aerosols:

(a) thermally generated sodium chloride particles (as used for the media loading tests), fed at a rate of 0.33 g/min;

(b) carbon black (Acetylene 50% compressed) particles, fed at 2.4 g/min. Although the carbon black used was intrinsically sub-micron in size, significant agglomeration was evident during dispersion. The mmd of the resulting aerosol was measured, using an Andersen Cascade Impactor, to be 2.2 μm.

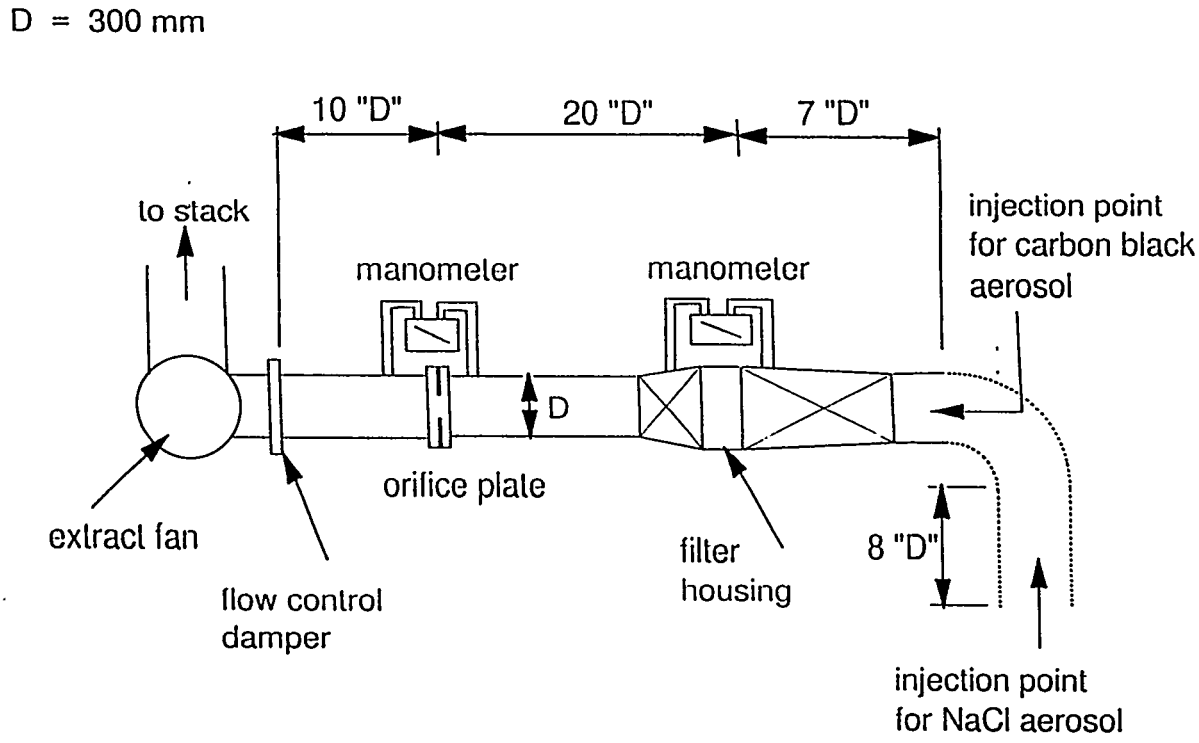


Figure 2 - General Arrangement of the Filter Dust Loading Rig

Dust loading was continued until the test filter pressure differential reached 1 kPa and 3 kPa for experiments involving the salt and carbon black aerosols respectively.

3. Results and Discussion

The dust loading characteristics of samples cut from a roll of a "standard" HEPA quality paper were routinely measured. This approach enabled the correct operation of the rig to be confirmed by reference to the results obtained for this material. Moreover, the relative dust holding capacity (RDC) of the test medium could be expressed as the ratio of its final pressure drop to the final pressure drop recorded for the standard paper, ie:

$$RDC = \frac{\Delta P \text{ Standard Paper}}{\Delta P \text{ Test Medium}}$$

Although this ratio did not fully quantify differences in the dust holding capacities of the papers, it allowed rapid identification of media for further investigation. Values > 1 indicated an improvement in dust holding. Approximately 40 different media were tested and ratios ranging from 0.6 to 2.0 were recorded.

Figure 3 shows examples of the loading curves obtained. The samples designated F-01 and F-18 are examples of "graded density" papers supplied by Whatman International Ltd. These papers are not commercially available, and were supplied as "development grades". Their structure differs from the uniform density normally associated with standard glass fibre papers. A continuous layer of very fine fibres is supported on a layer of coarser fibres which provide mechanical strength and act as a protective pre-filter. The results illustrated (Figure 3) are for dust loading from the pre-filter side. As may be expected, loading in the reverse direction was observed to nullify the advantage provided by the graded structure.

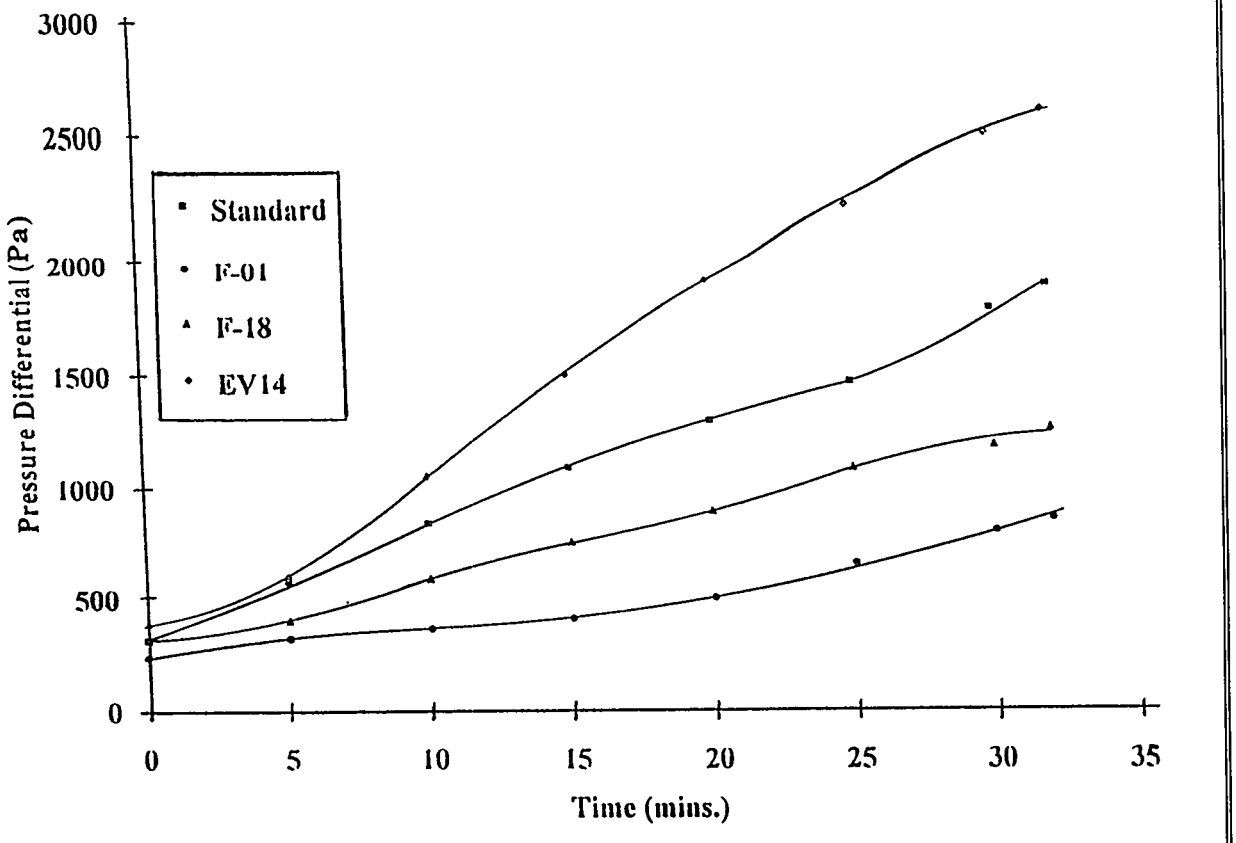


Figure 3 - Results of Media Loading at 3m min^{-1}

Although the potential advantage offered by the F-01 paper was established early in the programme, a number of its properties failed to meet the requirements stipulated in AESS 30/93400[3] (see Table 1). However, the manufacturing process allows the production of graded media to meet specific requirements in terms of filtration efficiency, strength and dust holding performance. Paper F-18 is the result of a development programme undertaken by the manufacturer to improve the strength properties of the F-01 medium. The increase in strength resulted in some reduction in dust holding, but it is felt that further development will achieve the optimum balance between strength and dust loading characteristics.

Table 1

Physical Property Data for Graded Density Papers

Property	Requirement (AESS 30/ 93400)	Paper Type	
		F-01	F-18
Tensile Strength[5] (kNm^{-1}):			
Machine Direction	> 0.8	0.9	1.2
Cross Direction	> 0.5	0.5	0.6
Creased	> 0.5	0.4	0.7
Air Burst Resistance (kPa)	>5.4	> 8.0	> 8.0
Grammage (gm^{-2})	> 85	72	95
Water Repellency (kPa)	> 2.5	0.5	3.1
Thickness (mm)	-	0.51	0.53
Particle penetration[6] (%)	< 0.005	0.013	0.003

Influences such as the deposition of dust between the pleats have been shown to adversely affect filter dust holding[2, 7]. This suggests that any advantage in the dust loading characteristics of the medium may not be fully realised in the performance of the filter. The experimental filters used in the second phase of the programme were fabricated to include the F-01 and F-18 papers; the dust loading performance of these papers was well characterised, and their use enabled such factors to be quantified.

Dust loading was assessed by comparison with filters manufactured to include the standard glass fibre paper used in the medium selection trials. In all cases the filters were fabricated to incorporate 23± 1 m^2 of paper. The initial BS 3928 penetration values are summarised in Table 2.

In general the penetration results recorded for the filters fabricated from the graded density papers fail the requirement (< 0.01%) stipulated in the relevant specification. However, it is considered that further medium developments can also improve this situation. No significant

increases in penetration were measured after filter dust loading, and there was no evidence of test dust in the downstream ducting etc.

The sodium chloride and carbon black filter loading results are illustrated in Figures 4 and 5 respectively. In each case the dust load per unit area of media has been calculated and the results are summarised in Table 3. The data presented are based on the nominal area of the media, and do not make allowance for the paper which is in contact with the corrugations of the spacers.

Table 2

Filter Penetration and Pressure Drop Results

Filter No	Medium Used	ΔP (Pa)	Pen (%)
7178	Standard	204	0.005
7181	Standard	204	0.003
7461	F-18	201	0.003
7462	F-18	186	0.049
7463	F-01	154	0.026
7464	F-01	154	0.044

Table 3

Filter Dust Load as a Function of Medium Area

Medium	Dust load / Unit area	
	Carbon Black ($g\ m^{-2}$ at 3 kPa)	Sodium Chloride ($g\ m^{-2}$ at 1 kPa)
Standard	43.1	5.3
F-18	47.3	6.6
F-01	48.5	8.2

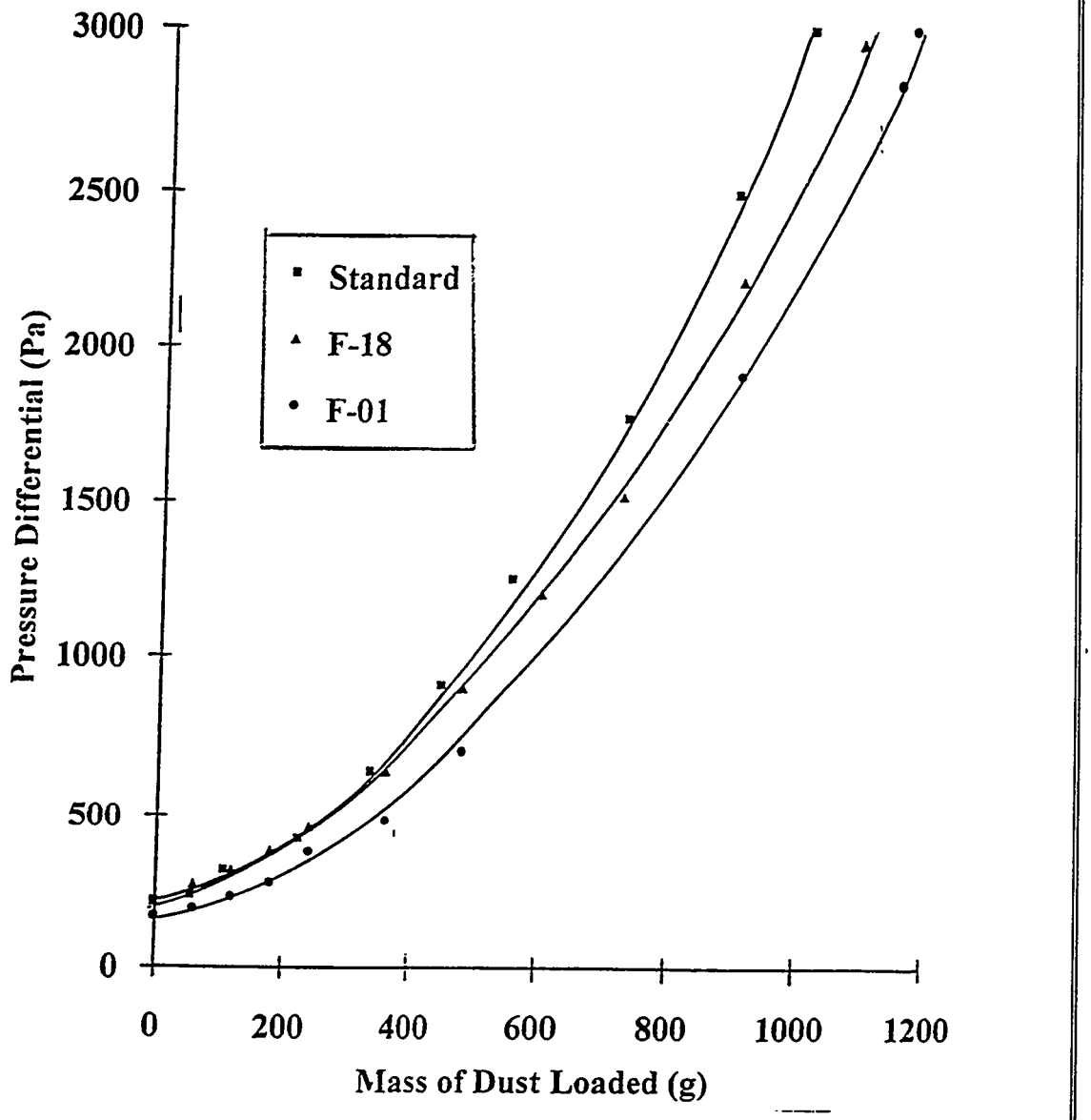


Figure 4 - Filter Dust Loading Results
(Carbon Black)

The proportionate differences in filter dust holding capacity (expressed in terms of dust capacity per unit area of medium) are less for carbon black than for the much finer sodium chloride particles. For carbon black the improvements were 25% and 12% at 1KPa for the F-01 and F-18 papers respectively. The improvements in filter loading, for sodium chloride were 55% and 25%. The corresponding values for the sheet media were 95% and 45% respectively.

Clearly, the potential advantages in dust holding offered by the papers have not been realised in the filters. There was noticeable deposition of carbon black within the filter channels; but virtually no deposition observed with the sodium chloride aerosol. Similar differences between

the medium and filter loading data were observed for the standard paper. Further work is necessary to fully quantify these effects.

The distinguishing characteristic of these particular graded density papers appears to be their influence on the changes in paper resistance during the initial particle deposition process. This is illustrated by the shape of the resistance curves in Figure 5. There appears to be a "once only" bonus effect, whereby the graded layer can absorb particles (in this instance up to 2.0 gm^{-2}; approximately 1 x $10^{-6}m^3$ per m^2) without significant increase in resistance to flow. The resistance then begins to rise and continue in a similar manner to the clean standard paper, ie the resistance rises as though for "cake deposition. In the case of the carbon black results this effect is not apparent.

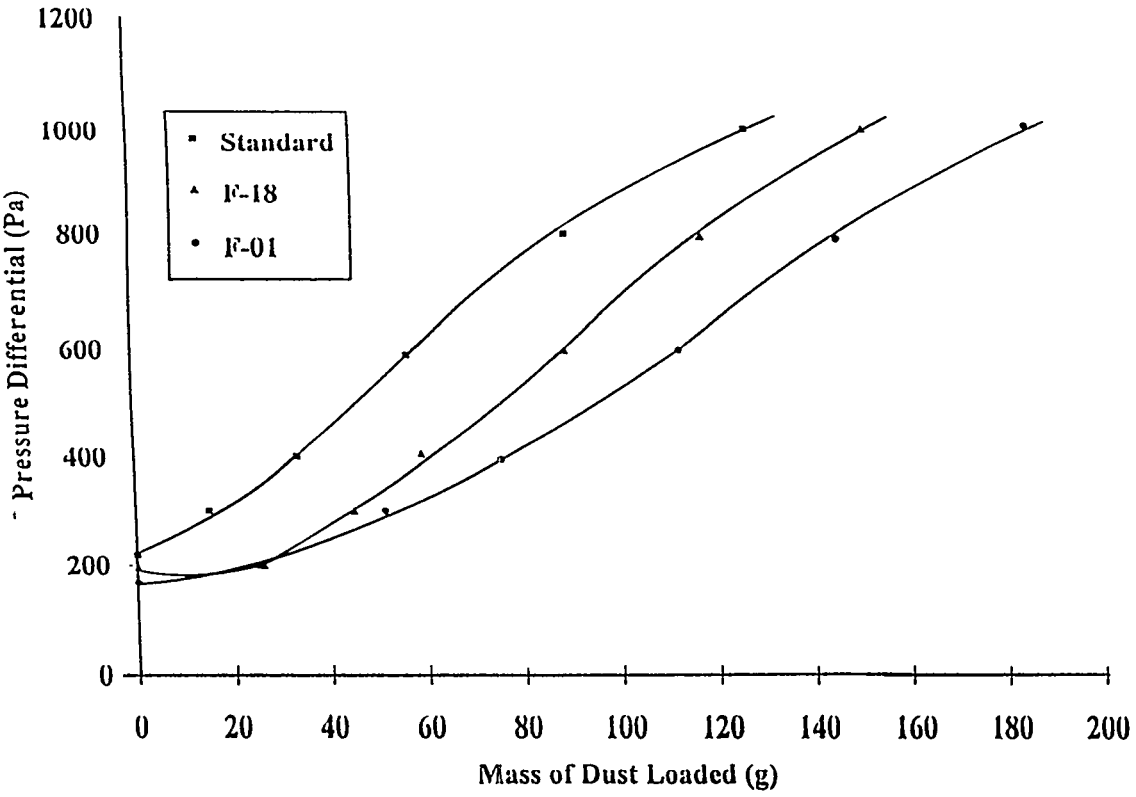


Figure 5 - Filter Dust loading Results
(Sodium Chloride)

This suggests that the structure of the graded paper can be expected to influence the increase in resistance due to dust deposition only while the deposit is forming within the depth of the

medium. Clearly this situation will be most marked in situations where the aerosol is within the most penetrating particle size range.

Such loading characteristics are likely to be most beneficial when HEPA (or ULPA) filters are operated in situations where the allowable increase in operating resistance is restricted, or when very small changes in resistance over a long period of operation are of particular advantage.

4. Conclusions

An initial programme to develop a HEPA filter with improved dust loading characteristics has been completed. This work has shown that:

- the relative dust holding of media samples can be rapidly assessed by comparing the increases in pressure differential under standard dust loading conditions;
- graded density glass fibre papers can offer significant improvements in dust holding. Recent manufacturing developments have provided examples of these materials which meet the requirements of the current filter medium specification;
- graded density papers can be incorporated in a standard sized deep-pleat filters. The resulting improvements in filter dust holding are comparable to the advantages demonstrated by the media.
- the increases in dust holding capacity achieved to date have the potential to significantly reduce waste arisings in applications where the challenge is predominantly sub-micron in size.

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6. Acknowledgements

This work was undertaken with financial support from the UK Department of Trade and Industry, under the management of the Research, Development and Strategy Department of the DRAWMOPS Directorate, at Harwell. The authors gratefully acknowledge the work of Whatman International Ltd in the development of the graded density papers used, and the kind co-operation of MC Air Filtration Ltd in the production of the deep-pleat filters for testing.

* * *

Effect of humidity on the filter pressure drop

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Abstract

The effects of humidity on the filter pressure drop have been reported in some previous studies in which it is difficult to draw definite conclusions.

These studies show contradictory effects of humidity on the pressure drop probably due to differences in the hygroscopicity of the test aerosols.

The objective of this paper is to present experimental results on the evolution of the filter pressure drop versus mass loading, for different test aerosols and relative humidities. Present results are compared to those found in various publications.

An experimental device has been designed to measure filter pressure drop as the function of the areal density for relative humidity varying in the range of 9 % to 85 %. Experiments have been conducted with hygroscopic (CsOH) and nonhygroscopic aerosols (TiO_2). Cesium hydroxyde (CsOH) of size of 2 μm AMMD has been generated by an ultrasonic generator and the 0.7 μm AMMD titanium oxyde has been dispersed by a "turn-table" generator.

As it is noted in the BISWAS'publication [3], present results show, in the case of nonhygroscopic aerosols, a linear relationship of pressure drop to mass loading. For hygroscopic aerosols two cases must be considered : for relative humidity below the deliquescent point of the aerosol, the relationship of pressure drop to mass loading remains linear ; above the deliquescent point, the results show a sudden increase in the pressure drop and the mass loading capacity of the filter is drastically reduced.

I. Introduction

In Nuclear Industry Particulate Air filters are always used as particulate containment device in gas cleaning and ventilation systems. In some postulated accident scenarios associated with water cooled nuclear reactor, a large quantity of steam could also be released along with a variety of aerosol. The interaction of water vapor with generated aerosols may significantly affect the ability of a filtration system to contain the radio-active aerosols.

Previous studies report [1] [2] contradictory effects of humidity on the pressure drop across filters. They have indicated trends but no definite conclusions could be drawn about the effects of relative humidity on the alteration of pressure drop.

The effects of the relative humidity on the filter pressure drop are bound to be dependant on the hygroscopicity of the particles and may also depend upon the particle size.

The objective of this paper is to present experimental results on the effects of humidity on the pressure drop-mass loading characteristics of a metallic filter according to the particle hygroscopicity. Experiments were conducted using different particle materials : non hygroscopic titanium oxyde and hygroscopic sodium chloride and cesium hydroxyde.

II. Experimental test rig

An experimental test rig, figure 1, was designed to measure the pressure drop across the metallic filter as a function of mass loading (mass/unit area of filter).

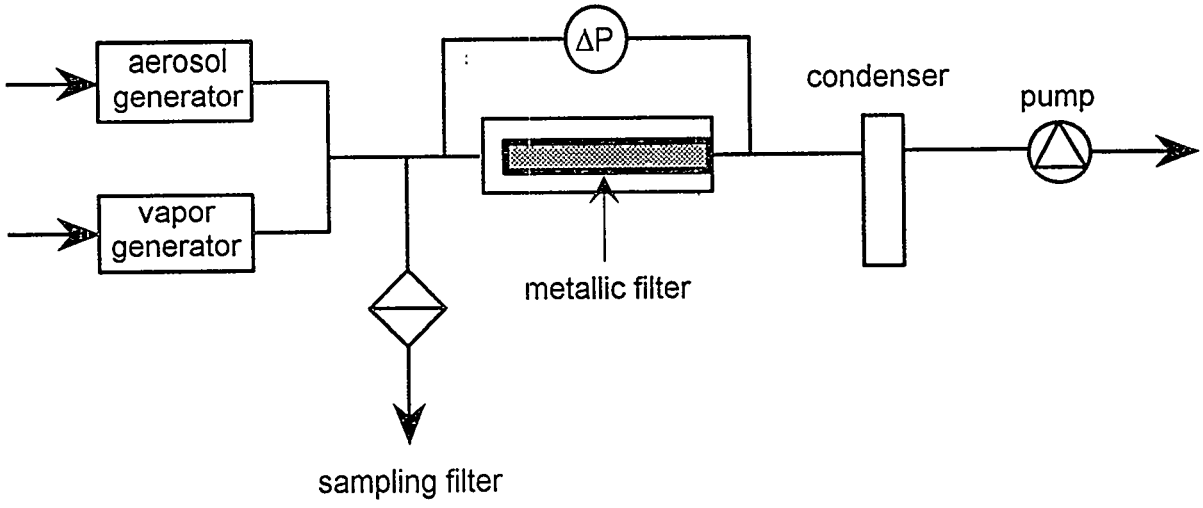


Figure 1 : Experimental test rig

The sodium chloride (NaCl) and cesium hydroxyde (CsOH) aerosols were generated by an ultrasonic atomisation of aqueous solutions of NaCl and CsOH. The droplets formed were drained and dryed in an additionnal dry air flow rate. Titanium oxyde powder were dispersed by a "turn table" generator.

In dry air conditions, an Andersen MKII impactor was used to determine the size distribution of the different test aerosols. The measured size distributions were lognormally distributed and the aerodynamic mass median diameters (AMMD) and the geometric standard deviations (σg) are listed in table 1.

Aerosol	AMMD(μm)	σg
NaCl	2.2	1.45
CsOH	2.4	1.50
TiO_2	0.7	1.55

Table 1 : Particles characteristics

The dry aerosol stream was inlet into the main test rig where the metallic filter was installed in a cylindrical cartridge. The metallic filter had a cylindrical shape of 6 cm in diameter and 30 cm long. The nominal filtration velocity used in these experiments was 5 cm/s.

A known flow rate of humidified air, delivered by a vapor generator, was inlet into the test rig in order to obtain different relative humidities.

Upstream the metallic filter, a sampling tube was installed and connected to a 47 mm filter holder. Glass fiber HEPA filters were used to determine the aerosol mass concentration during the mass loading experiments.

III. Results and discussion

For the different mass loading tests on the metallic filter the areal loading density (mass/unit area of filter) was computed from the aerosol mass concentration measurements performed on the 47 mm glass fiber filter. The pressure drop across the metallic filter was recorded all along the experiment.

Results for the titanium oxyde particles

The pressure drop across the filter was plotted as a function of areal loading density. The results are presented in figure 2 for tests performed under dry air conditions and at RH of 85 %; the pressure drop versus mass loading curves are linear and no significantly variation in the slopes of these curves can be observed. Similar observations have been made by BISWAS [3] for nonhygroscopic particles of 0.5 μm in diameter below a RH of 90 %.

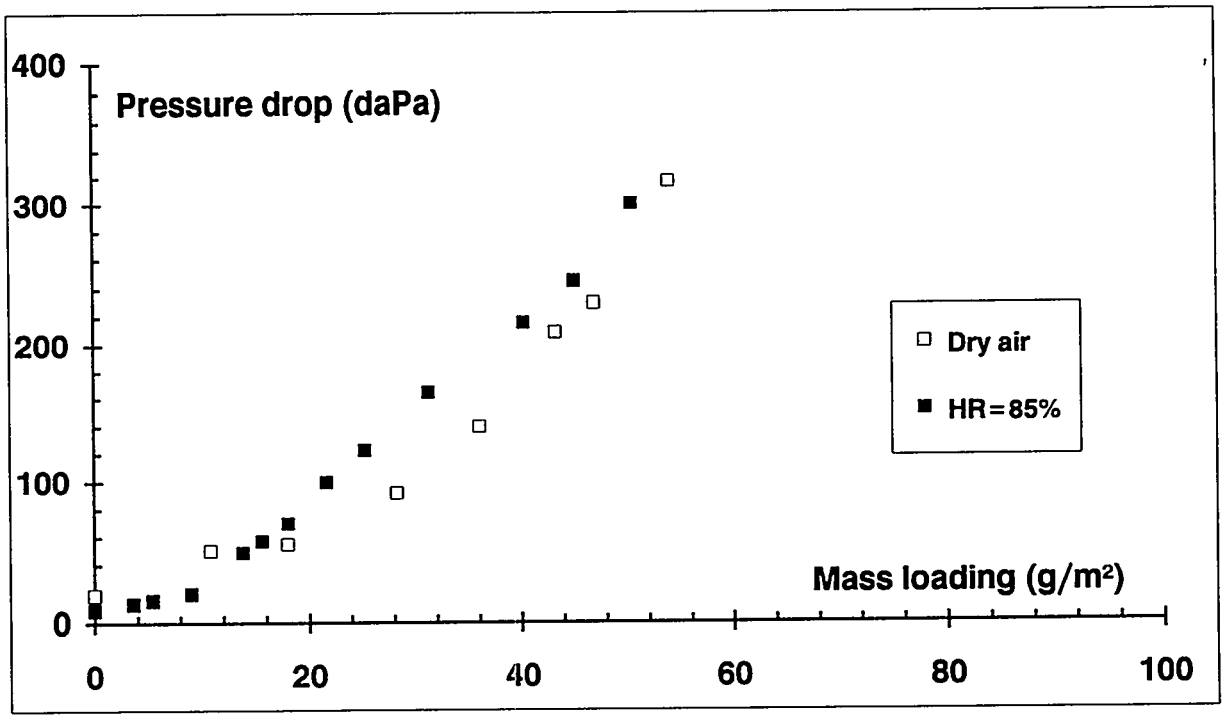


Figure 2 : Measured pressure drop versus areal density for titanium oxyde particles

Results for the NaCl particles

In figure 3 are presented the results of the mass loading experiments performed with a hygroscopic aerosol (NaCl) under dry air conditions and at RH of 23 % and 85 %.

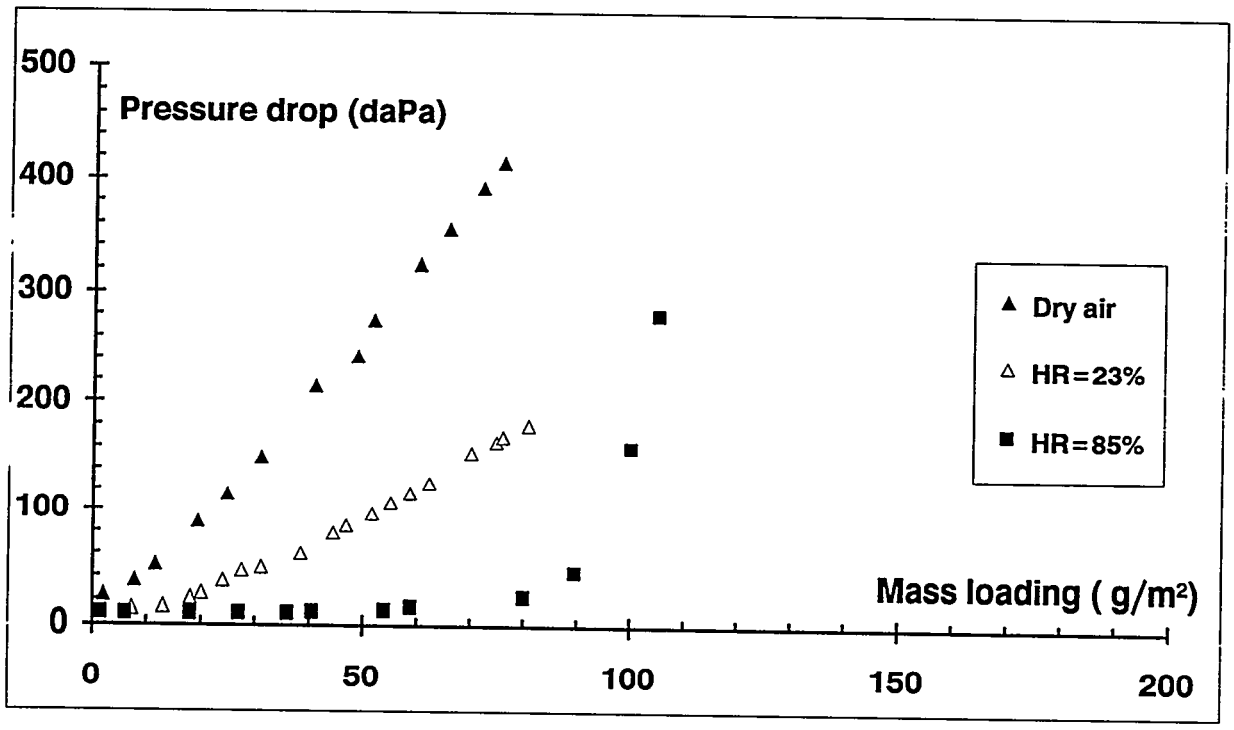


Figure 3 : Measured pressure drop versus areal density for sodium chloride particles

For dry air conditions and at RH of 23 %, a linear relationship of pressure drop to mass loading was obtained. At RH of 85 % this relationship was non-linear and it was observed a sudden increase in pressure drop.

These kind of results were also observed by BISWAS [3] and they show that the non-linear curves were only obtained for NaCl aerosol tests at humidities higher than the deliquescent point (RH = 75 %)

Results for the CsOH particles

For different humidities ranging from 9 % to 85 %, the results of the mass loading experiments performed with CsOH particles are presented in figure 4. On this figure the results obtained under dry air conditions are also noted.

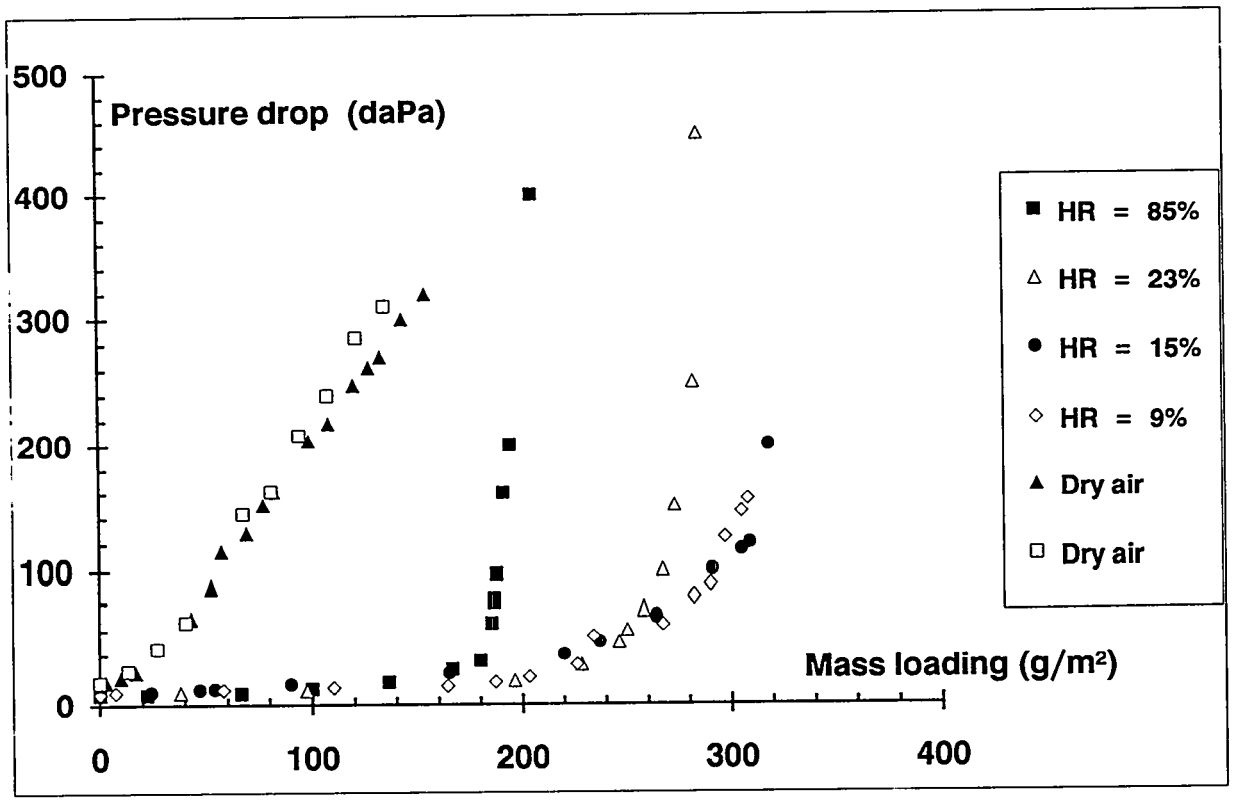


Figure 4 : Measured pressure drop versus aeral density for cesium hydroxyde particles

For dry air condition a linear relationship is observed. Because the deliquescent point for the CsOH-H_2O system [4] in the temperature range 20°C-100°C is about 2 % RH, non linear relationship of pressure drop to mass loading was obtained.

As the figure 4 shows, the sudden increase in pressure drop seems initiated at lower mass loading when the relative humidity is increased.

IV. Conclusion

The objective of the study was to present experimental results of the effect of humidity on the evolution of the filter pressure drop versus mass loading.

Linear relationships of pressure drop were obtained for non hygroscopic aerosols and for hygroscopic aerosols at humidity below the deliquescent point. Morever a decrease in pressure drop with mass loading in humidity conditions has been observed for the hygroscopic NaCl particles. For RH of 85 % ,this trend has not been noted for the non hygroscopic titanium oxyde particles.

Above the deliquescent point, non-linear relationship of pressure drop versus mass loading have been observed for hygroscopic NaCl and CsOH particles. In the case of CsOH particles, the sudden increase in pressure drop could make the filter completly clogged. The effect of humidity above the deliquescent point is to reduce the mass loading capacity of a filter. Hence to avoid filter clogging, caution must be exercised in filtration of hygroscopic aersols at humidities above their deliquescent point.

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DISCUSSION

WEBER: Did you have a chance to see what would happen after drying out a filter that had clogged because it was operated above the deliquescent point of the contaminant? Did the pressure drop change?

VENDEL: We did not make this kind of experiment. I don't know if the pressure drop changes when you dry the filters.

TSAL: In the last slide and the previous one, there is something that is difficult for me to understand. The 85% RH curve is a horizontal line close to zero pressure drop until it rises abruptly. To me this means that if I increase humidity I will bring the resistance down below what it would be in dry air until the sudden rise. Is it right? At 85% RH, your pressure drop is not increasing when you load the filter more and more.

VENDEL: Above the deliquescent point, the pressure drop does not increase with mass loading. One of the explanations is that a liquid film is formed on the fibers of the filter. In the case of cesium oxide, it is not necessary to have a relative humidity of 85%. This trait is also observed for humidity of about 9%. When you work above the deliquescent point, the pressure drop of the filters does not increase according to the mass loading, and a sudden increase in pressure drop can appear. In the case of cesium oxide, the filter is completely clogged and we can't work with these filters. It is not necessary to have 85%; it is strained when you work above the deliquescent point of the material that you use.

TSAL: It is very hard to understand because we are not used to this. We now understand that when you have a higher RH, resistance will be less until you plug the filter and then it becomes too wet. This is very important data because we do not take enough care about RH when calculating HVAC systems.

BERGMAN: Does the mass loading include the water content, or is it only particles? You plot pressure drop versus mass loading. Is that the weight of the filter when it's wet, or did you put it in an oven and then weigh it?

VENDEL: Yes, the mass loading presented includes water so the filters are weighed directly without any drying sequence.

Preliminary Field Evaluation of High Efficiency Steel Filters*

by

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ABSTRACT

We have conducted an evaluation of two high efficiency steel filters in the exhaust of an uranium oxide grit blaster at the Y-12 Plant in Oak Ridge Tennessee. The filters were installed in a specially designed filter housing with a reverse air-pulse cleaning system for automatically cleaning the filters in-place. Previous tests conducted on the same filters and housing at LLNL under controlled conditions using Arizona road dust showed good cleanability with reverse air pulses.

Two high efficiency steel filters, containing 64 pleated cartridge elements housed in the standard 2' x 2' x 1' HEPA frame, were evaluated in the filter test housing using a 1,000 cfm slip stream containing a high concentration of depleted uranium oxide dust. One filter had the pleated cartridges manufactured to our specifications by the Pall Corporation and the other by Memtec Corporation. Test results showed both filters had a rapid increase in pressure drop with time, and reverse air pulses could not decrease the pressure drop. We suspected moisture accumulation in the filters was the problem since there were heavy rains during the evaluations, and the pressure drop of the Memtec filter decreased dramatically after passing clean, dry air through the filter and after the filter sat idle for one week. Subsequent laboratory tests on a single filter cartridge confirmed that water accumulation in the filter was responsible for the increase in filter pressure drop and the inability to lower the pressure drop by reverse air pulses. No effort was made to identify the source of the water accumulation and correct the problem because the available funds were exhausted.

I. Introduction

This report describes the continuation of our development of a high efficiency particulate air (HEPA) filter that is made from stainless steel medium and is cleanable by reverse air pulses for nuclear air cleaning applications. Our initial study established the feasibility of developing a HEPA filter made from steel fiber medium (1). At that time, commercially available steel media had either a low efficiency or a

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high pressure drop. We evaluated a variety of steel filter media and concluded that media made from steel fibers yield higher efficiencies and lower pressure drops than comparable media made from powder metal. Another major conclusion of that study was that reducing the fiber diameter would simultaneously yield higher efficiencies and lower pressure drops. We then worked separately with Pall Corporation and Memtec of America Corporation to develop a steel fiber medium that had the lowest pressure drop while maintaining a minimum efficiency of 99.97% for 0.3 μm DOP particles. Both Pall and Memtec were able to make a filter medium that met the HEPA efficiency requirement but had a pressure drop of about three inches of water. Although we knew that smaller diameter fibers were needed to achieve HEPA performance for filters having the standard dimension of 2 ft. x 2 ft. x 1 ft., none were available.

Our next step was the development and evaluation of a prototype steel filter using the new media made from 2 μm diameter steel fibers [2]. We established the basic design parameters of the filter in a semi-empirical study. To obtain a sufficient filter area, we selected the pleated cartridge design shown in Figure 1. The cartridge parameters of pleat width and pleat depth were selected based on a compromise between maximum filter area and filter cleanability. We then issued contracts to Pall and Memtec to make a sufficient number of cartridges to fabricate a standard 1,000 cfm capacity filter from each company. Figure 2 shows the assembled filter that consisted of 64 cartridge elements. We fabricated two filter units, one using cartridges made by Pall, and the second using cartridges made by Memtec. Both of the filters were certified to have 99.99% efficiency for 0.3 μm DOP particles when tested at 1,000 cfm in the DOE filter test facility at Oak Ridge, Tennessee. The measured pressure drop for the Pall filter was 3.2 inches, while the Memtec filter had a pressure drop of 2.9 inches.

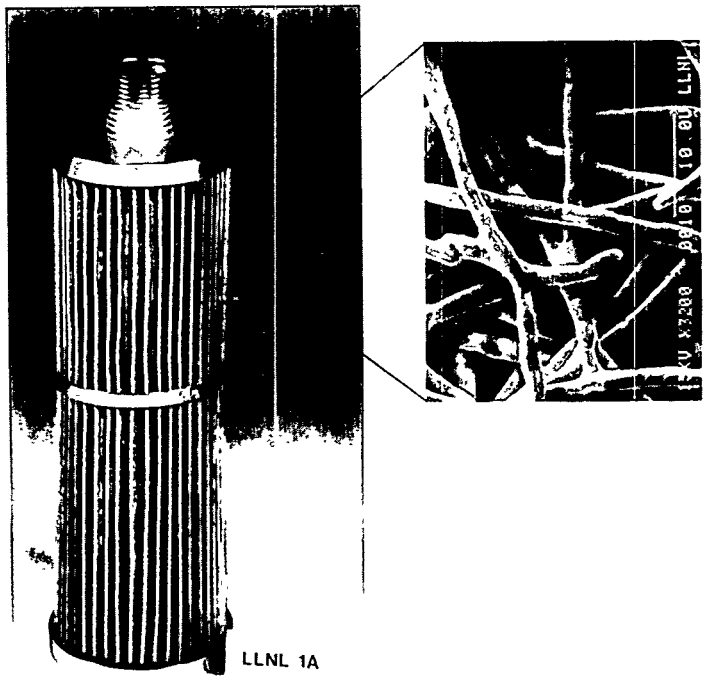


Figure 1. Single filter cartridge and electron micrograph of stainless steel fiber medium.

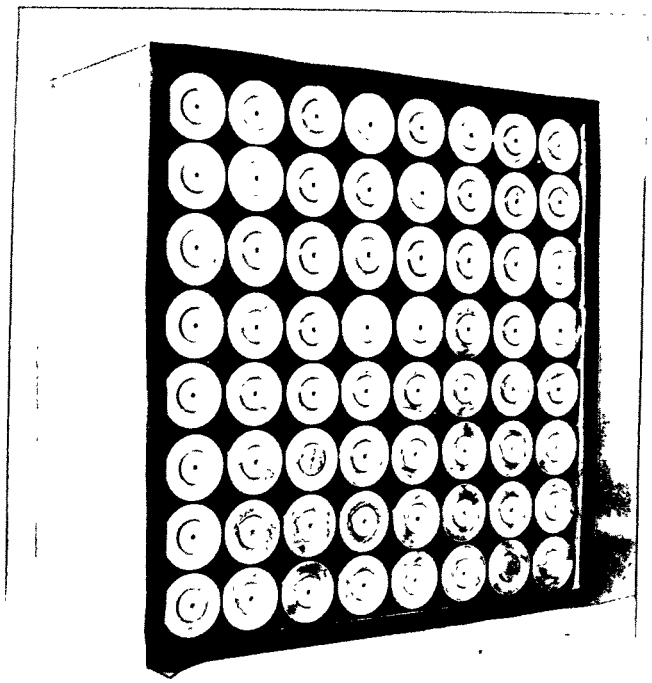


Figure 2. Photograph of the cleanable steel filter consisting of 64 cartridge elements

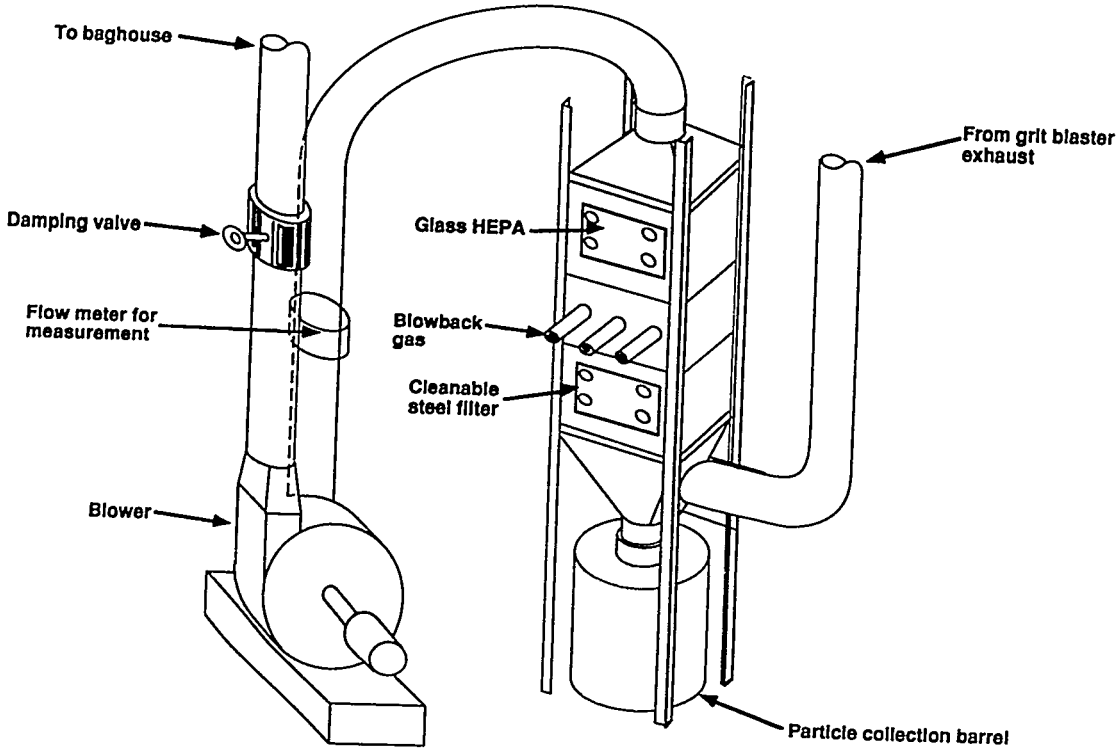


Figure 3. Schematic of the filter housing and blower assembly

The steel filters were evaluated in the filter housing assembly shown in the schematic in Figure 3. The steel filter was housed in the lower chamber. The middle chamber contained the tubing used for air pulse cleaning, and the upper chamber contained a standard glass HEPA filter. Figure 4 shows a photograph of the filter housing and blower assembly. The steel filters were subjected to multiple cycles of dust loading with Arizona road dust and cleaning with reverse air pulses as shown in Figure 5. The filter cleaning is done without interrupting the air flow because only four of the filter cartridges are cleaned at one time. As the dirty air is filtered, particle deposits form on the surface of the filter and cause the pressure drop to increase. After the filter reaches a preset pressure drop, a reverse air pulse blows back through a sequence of four cartridge elements at a time to dislodge the particle deposits, which are then collected in a waste drum. Cleaning tests conducted with Arizona road dust showed that, after the first few cleaning cycles, the filter maintained a constant pressure drop of 5 inches after each pulse cleaning.

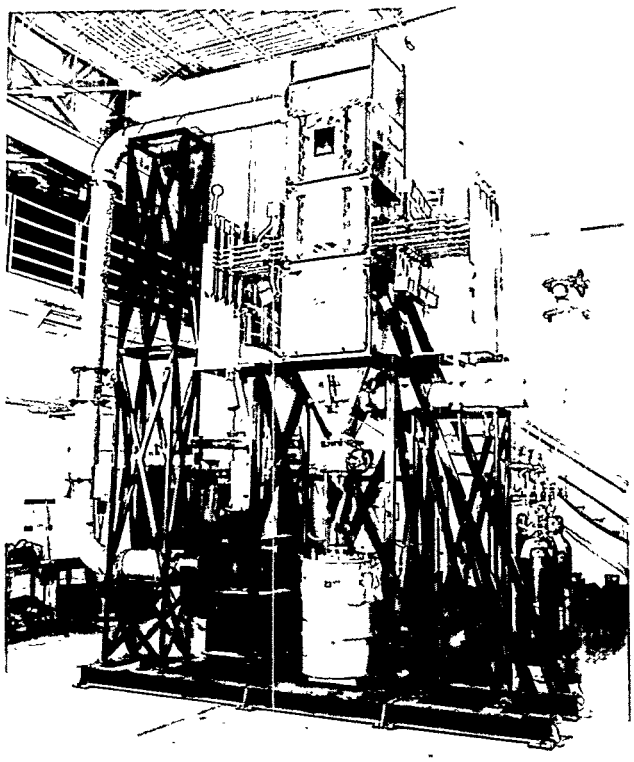


Figure 4. Photograph of the filter housing and blower assembly

II. Preparation for the Demonstration at the Y-12 Plant

After surveying potential sites at LLNL, Idaho National Engineering Laboratory, and the Y-12 Plant, we selected a grit blaster at the Y-12 Plant as the preferred demonstration site. The grit blaster uses abrasive grit to mechanically remove oxides and surface contamination from depleted uranium billets and other formed parts. The exhaust from the grit blaster first passes through a grit recovery system and then through a bag filter and HEPA filter system shown in Figure 6.

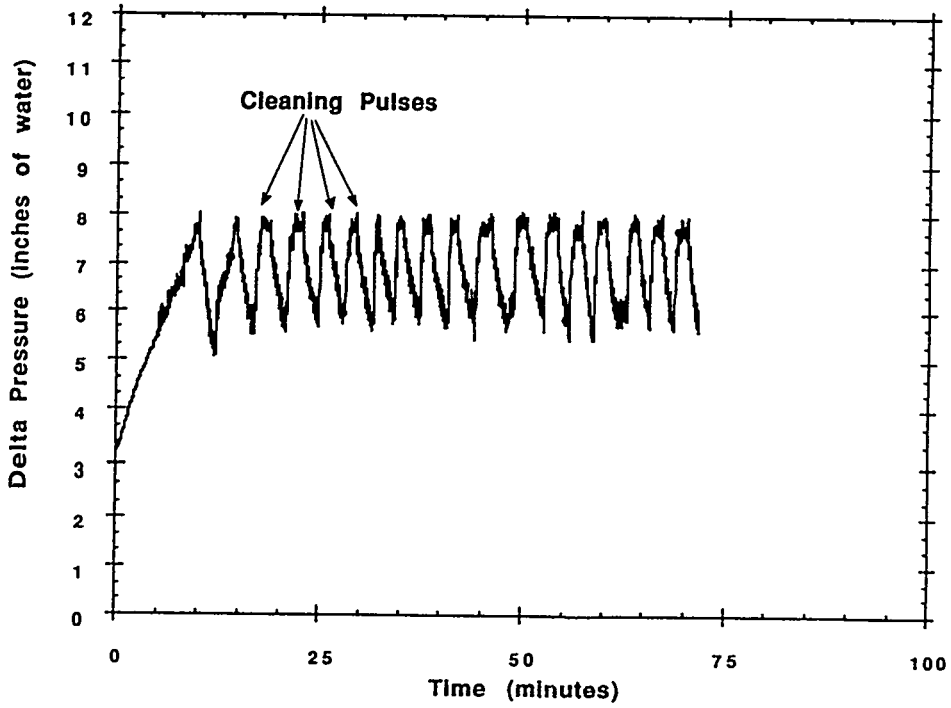


Figure 5. Multiple cycles of filter loading with Arizona road dust and filter cleaning with reverse air pulses.

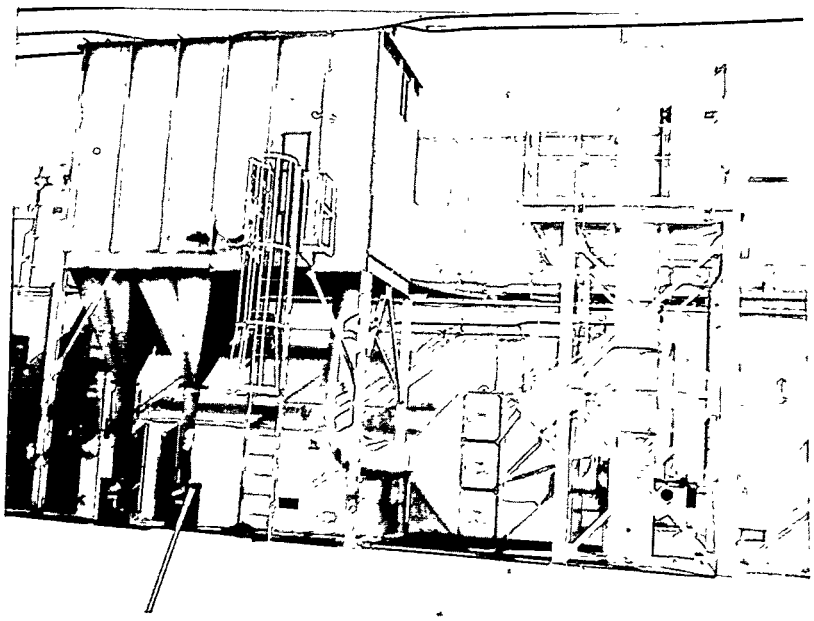


Figure 6. The exhaust system from the grit blaster consists of a bag and a HEPA filtration system.

The exhaust from the grit blaster had many attractive features for our field demonstration. The facility could generate a large quantity of dust on-demand and thereby would allow us to conduct many cycles of filter loading and cleaning in a short time. The depleted uranium oxide served to demonstrate the filter in a radioactive application; but since the radioactivity was very low, the added health and safety requirements from handling radioactive material was minimal. Another attractive feature of the demonstration site was that the grit blaster generates large particles in a dry process. Figure 7 shows a comparison of the uranium oxide dust measured at the Y-12 Plant and the Arizona road dust used in the LLNL filter loading and cleaning tests. The data were obtained with inertial impactors and are plotted as percent mass less than stated size as a function of particle size. The mass median diameter (50% points on the graphs) for the Arizona road dust and the uranium oxide dust is 7.5 μm and 200 μm respectively. The larger size of the uranium oxide dust should make it much easier to clean the steel filter than the smaller Arizona road dust.

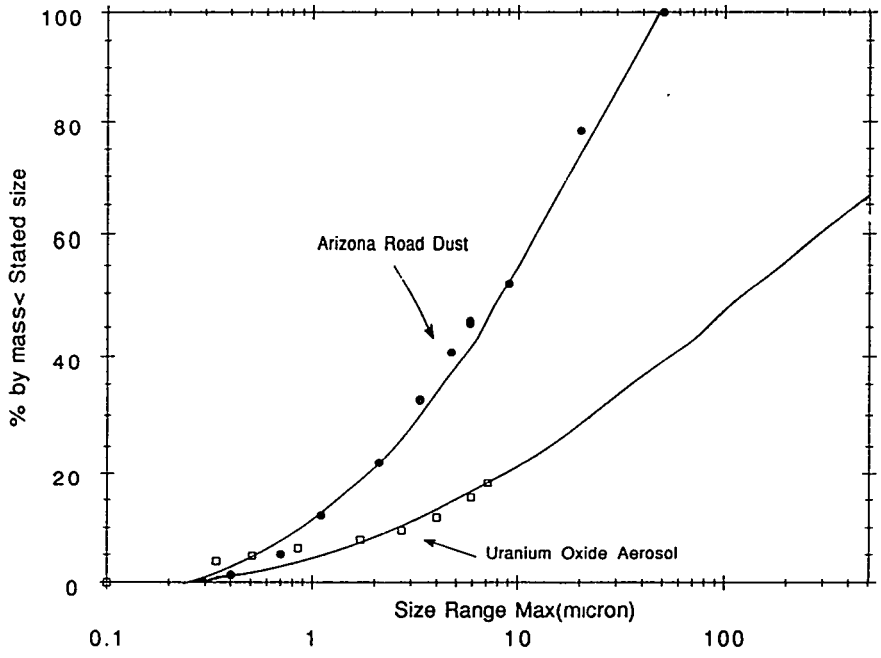


Figure 7. Size distribution measurements show the uranium oxide particles are much larger than the Arizona road dust.

The filter housing assembly was then installed outside the building next to the existing bag filter as seen in Figure 8. A separate 1,000 cfm slip stream was cut into the existing 3,000 cfm exhaust duct from the grit blaster and connected to the inlet of the filter housing assembly (the right vertical duct in Figures 3 and 4). The exhaust from the filter housing assembly was then reconnected to the existing duct leading to the bag house.

Most of the effort in the field demonstration was spent on health and safety reviews and preparations for the tests. In addition to the standard health and safety

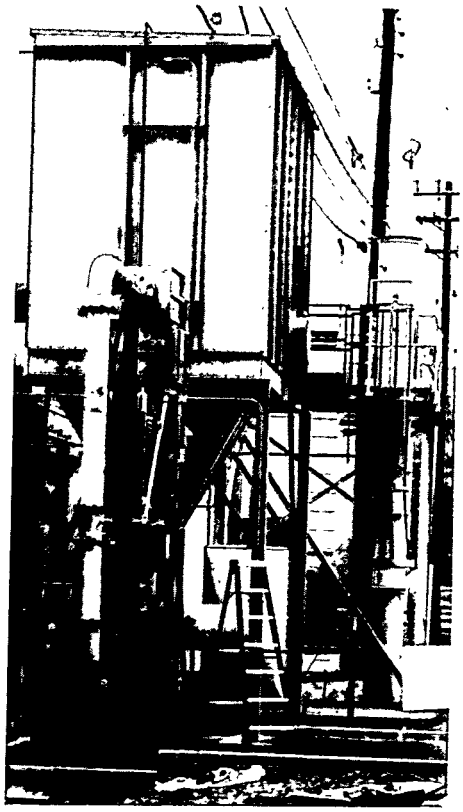


Figure 8 Filter housing assembly for evaluating the cleanable steel filter mounted next to the bag filtration system.

reviews, we conducted a seismic study on the filter housing assembly shown in Figure 4 to insure that the unit would not tip over. Because of the height, scaffolding had to be erected to allow for changing both the steel and the glass fiber filters. We also built an alignment table to support the 200 pound steel filter during filter installation. The alignment table was mounted on a scissors lift.

We also performed leak tests on the filter housing and blower assembly using the constant pressure method in ASME-N510-1989 (3). According to this method the inlet duct was sealed and a dry gas meter was installed between the blower and the exit duct from the glass HEPA. The initial test showed a leak of 0.68 cfm at 10 inches of vacuum, which met the most stringent requirement 0.1% maximum leak specified in Table B-3 in ASME-509 (4). The small leak was determined to come from the manifold plate gasket that supported the air pulse cleaning system. After sealing the leak with Glyptal 1202, the leak was reduced to 0.017 cfm. Leak tests based on ASME N510 were also performed on the two glass HEPA and two, steel, high efficiency filters after installation to insure that the filter leaks did not exceed 0.03%.

III. Y-12 Test Results Show Filter Plugging, Apparently Due To Moisture

The objective of our tests at the Y-12 Plant was to establish that the steel filter could be repeatedly cleaned in a field demonstration and thereby save the equivalent of 15 or more standard HEPA filters. A standard glass HEPA filter downstream of the

steel filter would verify that the steel filter trapped nearly all of the particles. The pressure drop across this filter remained constant at one inch of water during this evaluation, thereby insuring that virtually all of the particles were trapped on the first filter. The number of steel filter loadings and cleanings that correspond to the life of a single glass HEPA filter would be established in a baseline test of a glass HEPA filter. Details of the field demonstration were reported by Simon and Frye (5).

Figure 9 shows the pressure drop increase with increasing filtration time for one of two baseline glass HEPA filters tested. The HEPA filter was installed in the lower housing compartment on April 28, 1993 and tested for uranium oxide loading on the next day to provide a baseline test for the Memtec filter. Note that the initial pressure drop was 1.5 inches, which indicates a significant plugging even before exposure to the uranium oxide dust. We suspect high moisture was responsible for the high pressure drop based on observations from the subsequent test with the Memtec filter. Removing the filter from the housing at 4 inches of pressure and weighing it showed the filter had increased its weight by 1.7 pounds. We selected 4 inches pressure drop as the typical upper limit pressure drop for replacing HEPA filters. The baseline test for the Pall filter using a different HEPA filter showed a weight increase of 2.4 pounds. That HEPA filter had an initial pressure drop of 0.78 inches. Thus, an average weight gain of 2 pounds of uranium oxide would serve as the reference weight corresponding to one HEPA life.

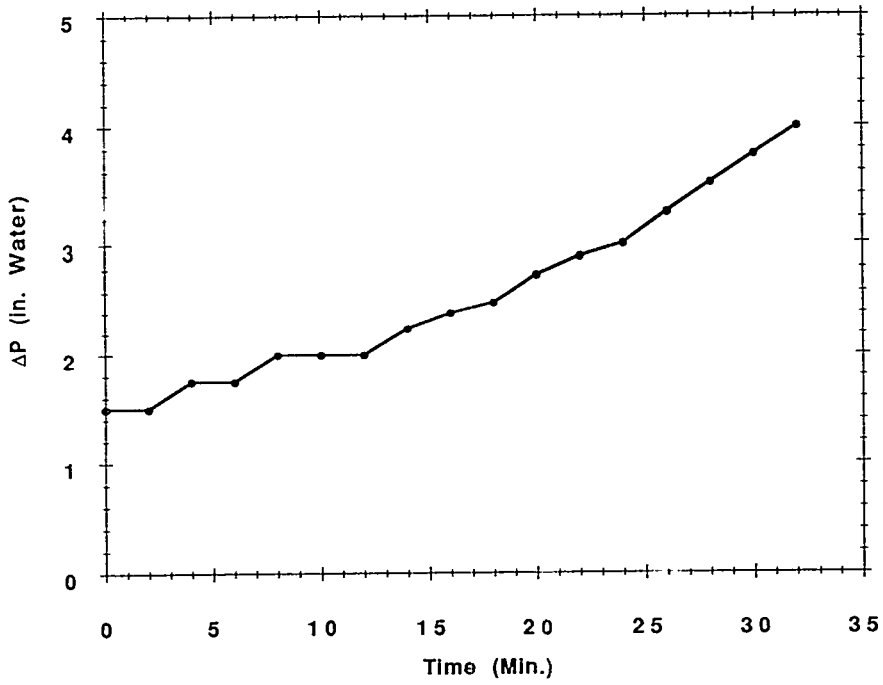


Figure 9. Pressure drop increase in a standard glass HEPA filter during loading with depleted uranium oxide dust.

The steel filter from Pall was installed and tested on April 15, 1993 for particle loading and cleaning. Figure 10 shows the pressure drop increase during the loading and the times where pulse cleaning was applied. Note that the pressure drop did not decrease after the pulse cleanings, indicating that the filter could not be cleaned. Visual examination of the filter showed that virtually no deposits had formed on the

filter surface. This is contrast to the heavy, irregular deposits observed in the LLNL tests with Arizona road dust (2).

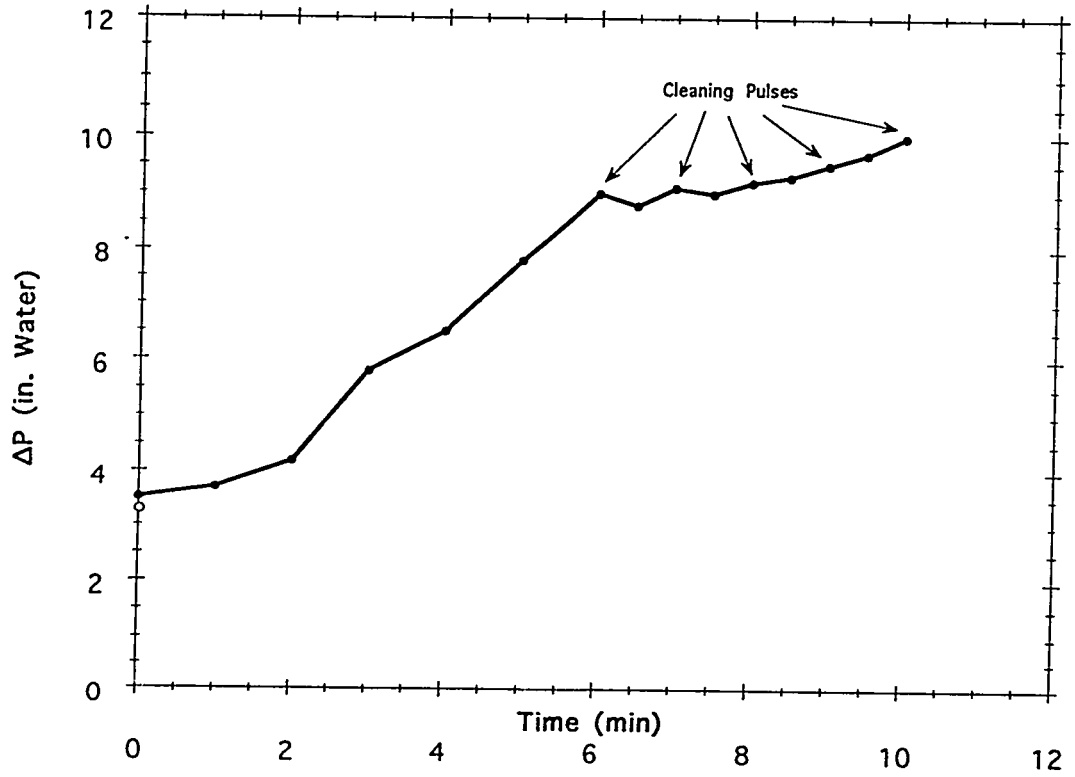


Figure 10. Pressure drop measurements on the Pall steel filter during loading with depleted uranium oxide dust and cleaning with reverse air pulses.

We suspected that moisture may have been responsible for our inability to clean the filter since there was a heavy rainstorm prior to and during the test. There was so much moisture in the system that water condensation had accumulated in one of the Magnahelic pressure gages and displaced the needle from its shaft. The Magnahelic pressure gage was connected to the Pitot tube flow meter that provided flow measurements for the test system. The defective gage was replaced prior to the test on the Pall filter. Unfortunately there were no humidity sensors in the exhaust duct to verify the high moisture content. To correct this deficiency, a humidity sensor was installed in the middle chamber of the filter housing assembly before proceeding with the next test.

The Memtec steel filter was then installed in the filter housing on May 4, 1993 for a new loading and cleaning test. Figure 11 shows the pressure drop across the filter during the field evaluation. Note that the initial pressure drop of the filter at the beginning of the test was 7 inches, which is much higher than the value obtained at LLNL or at the DOE Filter Certification Laboratory in K-25. Since the relative humidity measured after the filter indicated 90%, we suspected the filter was saturated with water, presumably from condensation. We then purged the filter by turning on the air blower without operating the grit blaster for 14 hours. This reduced the pressure drop to a constant 4.5 inches and dropped the relative humidity to 53%.

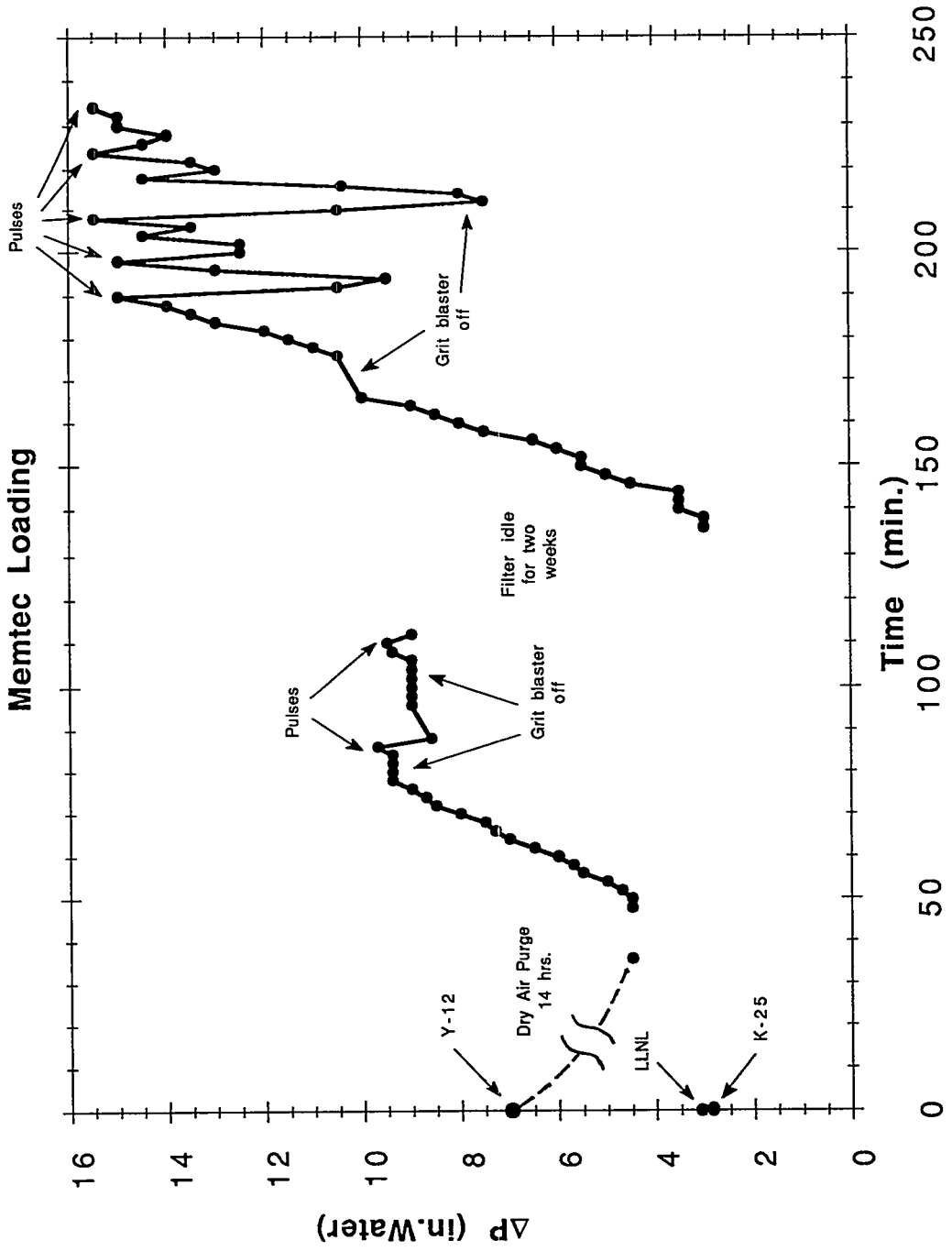


Figure 11 Pressure drop measurements on the Memtec steel filter during loading with depleted uranium oxide dust and cleaning with reverse air pulses.

The filter loading test was then started by operating the grit blaster on May 7, 1993 with the filter at 4.5 inches . Figure 11 shows the pressure drop increased with time in a similar fashion as the Pall filter in Figure 10. Attempts to clean the filter by reverse air pulses were also not successful; there was only a slight decrease in pressure drop after each pulse. The relative humidity after the filter varied between 48-55%.

We then stopped the loading test for nearly two weeks while we attempted to understand why reverse air pulses would not reduce the pressure drop. One hypothesis was that there was an insufficient dust cake to blow off by reverse air pulses. Visual inspection of the Pall filter in the previous test had confirmed that there was an insignificant deposit on the filter surface. We reasoned that a sufficient layer of particle deposits had not formed and therefore could not be blown off. Allowing the pressure drop to increase to 15 inches instead of 9 inches should increase the amount of particle deposits.

The loading test on the Memtec filter was then continued on May 20, 1993 to see if an increased particle deposit on the filter would make it easier to clean. Figure 11 shows that the filter pressure drop had decreased from 9 inches when the test was stopped to 3 inches when the test was continued. The filter pressure drop increased monotonically with increasing exposure to the uranium oxide stream until the reverse air pulse was activated at 15 inches of water, which reduced the pressure drop to 9.5 inches. The second cleaning had only a small effect. However after the third cleaning, the filter pressure drop decreased from 15 inches to 7.5 inches. The grit blaster was shut off at the same time and may have contributed to the decreased pressure drop. After the grit blaster was reactivated, the filter pressure drop rapidly increased to 14 inches and was not strongly affected by subsequent air pulses. The relative humidity in the filter duct was 44% at the start of the test and varied between 38% and 45% during the test. To verify that the pressure readings were correct, we performed a post-test calibration which showed the pressure gage was reading 0.5 inches lower. The data in this report were corrected for this error.

The following observations of the Memtec filter strongly suggested that water saturation on the filter was responsible for the pressure measurements: The pressure drop increased from 3 inches to 7 inches after installing the Memtec filter in the filter housing. It then dropped to 4.5 inches after passing clean air through the filter for 14 hours. The relative humidity measured after the filter dropped from 90% to 53% during the air purging. This data suggests that water is driven off the filter. The pressure drop also decreased from 9 inches to 3 inches after sitting idle for almost two weeks. Evaporation of water from the filter is the most likely explanation. The final observation that suggests that filter clogging is due to water and not particles is the inability of the reverse air pulses to reduce the pressure drop after filter clogging to either 9 inches or 15 inches.

Since high efficiency filters are not designed to operate under high moisture conditions, the source of the moisture should be identified and reduced. This task was the next logical step in our evaluation. Unfortunately, the field evaluation had to be terminated after the Memtec filter tests because the allocated funds were exhausted.

IV. Laboratory Tests Confirm Filter Plugging Is Due To Water

Since we could not complete the field evaluation, we conducted a series of laboratory tests on single cartridge filters to verify that water accumulation was responsible for filter plugging and the ineffective cleaning. Figure 12 shows the experimental apparatus used for the filter clogging and cleaning tests. The filter test housing consists of three chambers: a lower chamber that functions as a hopper to collect particle deposits, a middle chamber that houses the filter cartridge and an upper chamber that has the reverse air pulse system. Challenge air enters from the right into the lower part of the middle chamber, passes through the filter cartridge into the upper chamber and then leaves through the exit port on the left. Relative humidity measurements were made by replacing the sample probe in the middle chamber with a humidity probe. The same opening in the test chamber was also used to inject the iron oxide and Arizona road dust. Using the same port to measure relative humidity and inject aerosols reduces the relative humidity slightly during particle injection. However, since the injected air is only about 0.5 cfm, compared to the total flow of 15.6 cfm, the reduction in relative humidity is small.

We conducted a filter clogging and cleaning test with sub-micron iron oxide dust (Aldrich Chemical, Milwaukee, WI) to explore the possibility that sub-micron uranium oxide particles were responsible for the filter clogging. Although the average size of the uranium oxide particles was about 200 μm, there is a significant fraction of sub-micron particles that could potentially become deeply embedded in the steel fiber medium and be difficult to remove with reverse air pulses. The density of the iron oxide particles (5.2 g/cc) is also closer to the density of the uranium oxide (11 g/cc) than the Arizona road dust (2.5 g/cc) used in most of our laboratory tests. Figure 13 shows the filter clogging and cleaning test with iron oxide aerosols on a Pall cartridge operated at 15.6 cfm and about 30% relative humidity. This test demonstrated that deposits of high density and submicron aerosols could be readily cleaned from the filter with reverse air pulses.

Any explanation of filter plugging due to particle deposits must be eliminated because it is not possible for particle deposits to cause the large decreases in pressure drop seen with the Memtec filter in Figure 11. There is no conceivable mechanism based on particle deposits that could account for large decreases in filter pressure drop by passing clean air through the filer or by letting the filter sit idle for two weeks.

We also conducted filter clogging tests under high humidity using Arizona road dust. The high humidity was generated using several spray nebulizers in a separate chamber that was connected to the inlet of the filter test apparatus. Figure 14 shows the test results from a Memtec filter cartridge during four cycles of filter clogging and cleaning at 99% relative humidity. The filter pressure drop was reduced to 4-4.5 inches after each pulse cleaning at 7 inches. Figure 15 shows a Pall filter cartridge undergoing many cycles of clogging and cleaning at 86% relative humidity. The reverse air pulse was applied when the filter pressure drop reached 15 inches. Figures 14 and 15 demonstrate that reverse air pulses can successfully clean particle deposits from the filter even under high humidity conditions. However, Figure 15 shows that, at high humidities, the pulse cleaning became less efficient over time since the resulting pressure drop after each pulse cleaning gradually increased. Comparing the clogging and cleaning cycles under high humidity in Figure 15 with that under low humidity in Figure 5 suggests that high humidity causes a gradual decay in filter cleanability. We suspect the increasing

baseline pressure drop is due to the gradual accumulation of water in the filter and not particle deposits.



Figure 12. Filter test apparatus used for filter clogging and cleaning tests

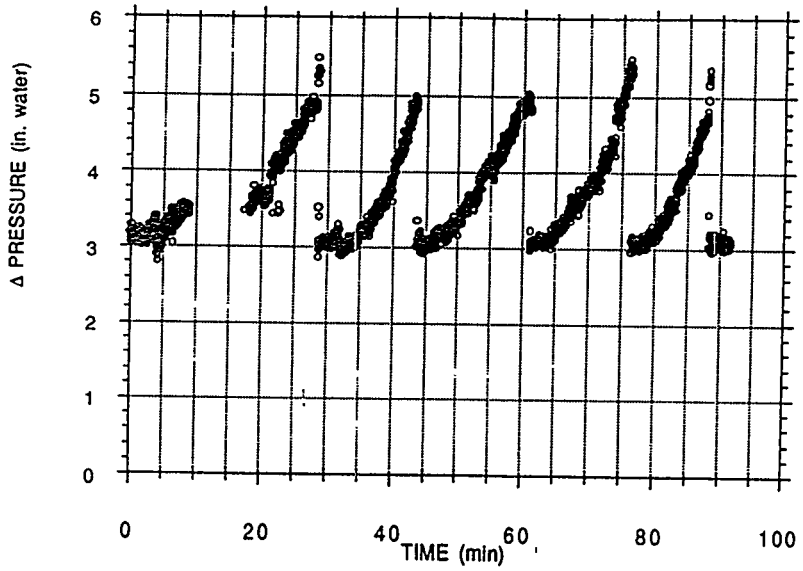


Figure 13. Filter clogging and cleaning test with submicron iron oxide dust on Pall filter cartridge at 15.6 cfm.

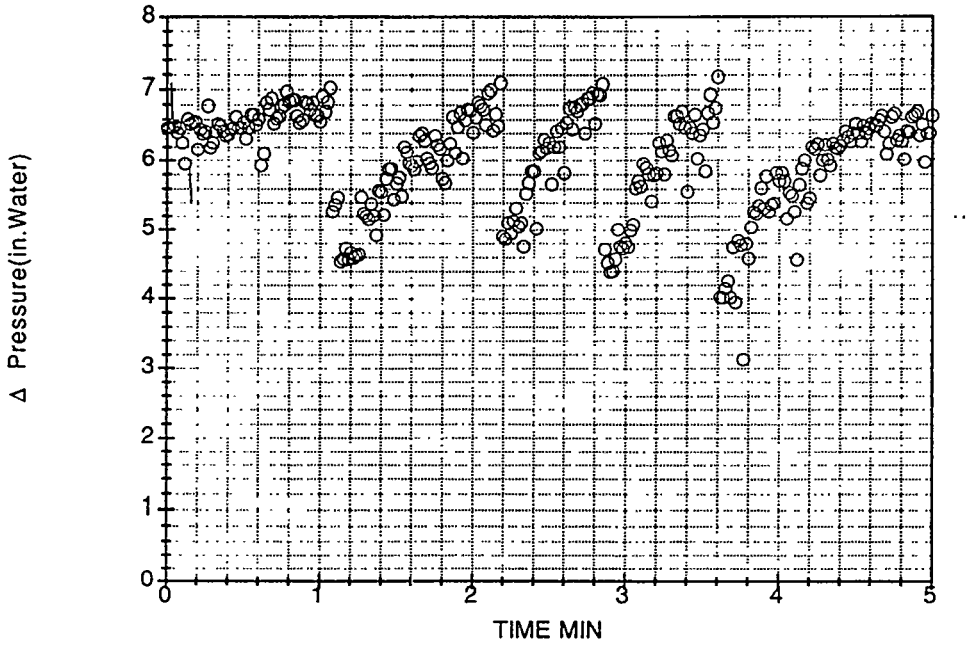


Figure 14. Filter clogging and cleaning cycles using Arizona road dust on Memtec filter cartridge at 15.6 cfm and 99% relative humidity

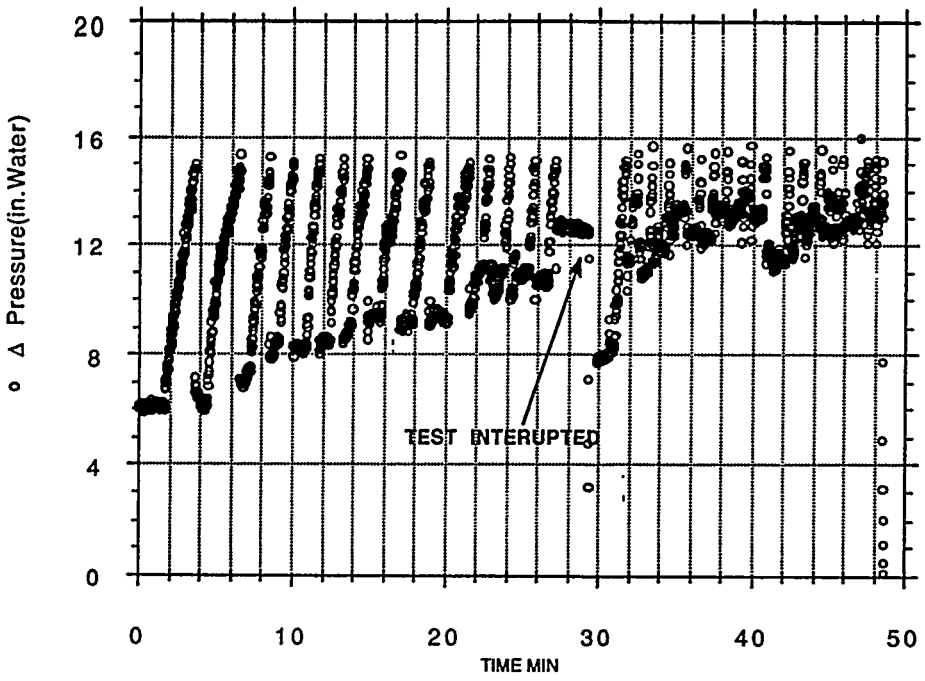


Figure 15 Filter clogging and cleaning cycles using Arizona road dust on Pall filter cartridge at 15.6 cfm and 86% relative humidity

Additional tests conducted under high humidity conditions with no aerosols confirm that the increased pressure drop and the ineffectiveness of the reverse air pulse cleaning system is due to water accumulation. Figure 16 shows the pressure drop increase as a function of time for a Pall filter cartridge when exposed to 15.6 cfm air at 100% relative humidity. The relative humidity after the filter dropped to 50%, which implies that half of the water vapor was removed by the filter. Reverse air pulses on the filter applied periodically during the water loading period had no effect on the pressure drop. The large breaks in the loading curve are due to flow adjustments required because of the increased pressure drop. The increasing pressure drop and the inability to lower it with air pulses in the laboratory tests duplicates the field observations for both the Pall filter in Figure 10 and the Memtec filter in Figure 11. The increasing pressure drop due to water accumulation is also the primary explanation for the increasing baseline pressure drop of the Pall filter being loaded with Arizona road dust at 86% relative humidity in Figure 15.

A filter clogging test using only high humidity was also conducted on a standard glass HEPA filter. Figure 17 shows the increase in pressure drop for a size 2 HEPA filter exposed to 40 cfm clean air at 100% relative humidity. Since the relative humidity downstream of the filter was 90%, only 10% of the water vapor was trapped in the filter. The large fluctuation in pressure drop in Figure 17 was due to an adjustment in the flow. Comparing the HEPA filter clogging with water in Figure 17 with the clogging seen with the grit blaster in Figure 9 suggests that water saturation may be responsible for the field clogging.

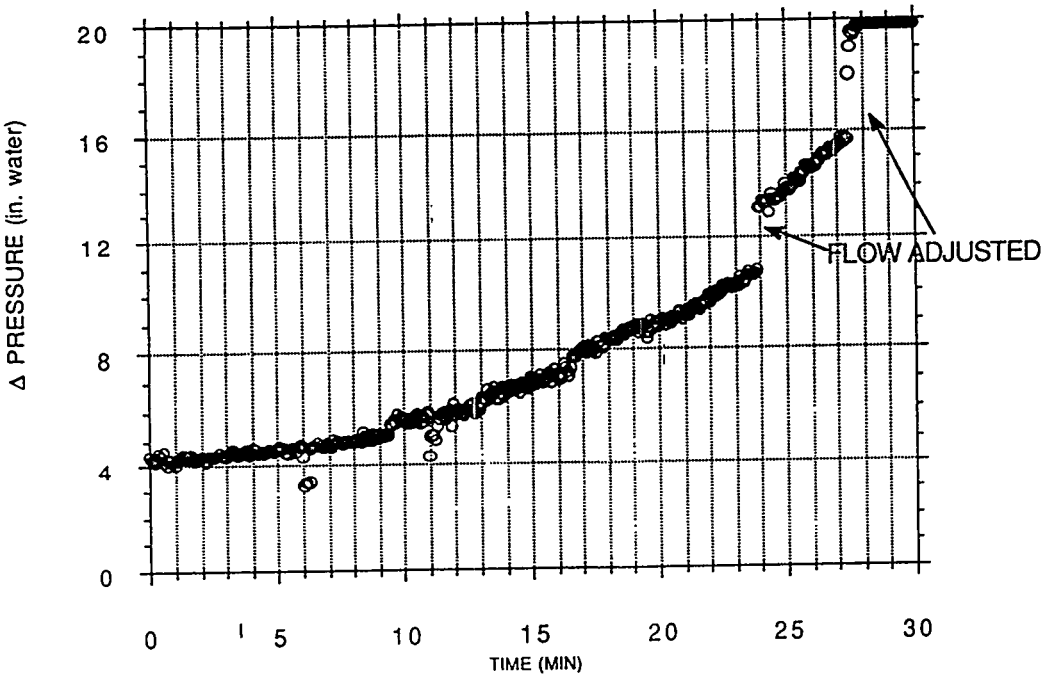


Figure 16 Filter clogging and cleaning tests with no aerosols on Pall filter cartridge at 15.6 cfm and 100% relative humidity.

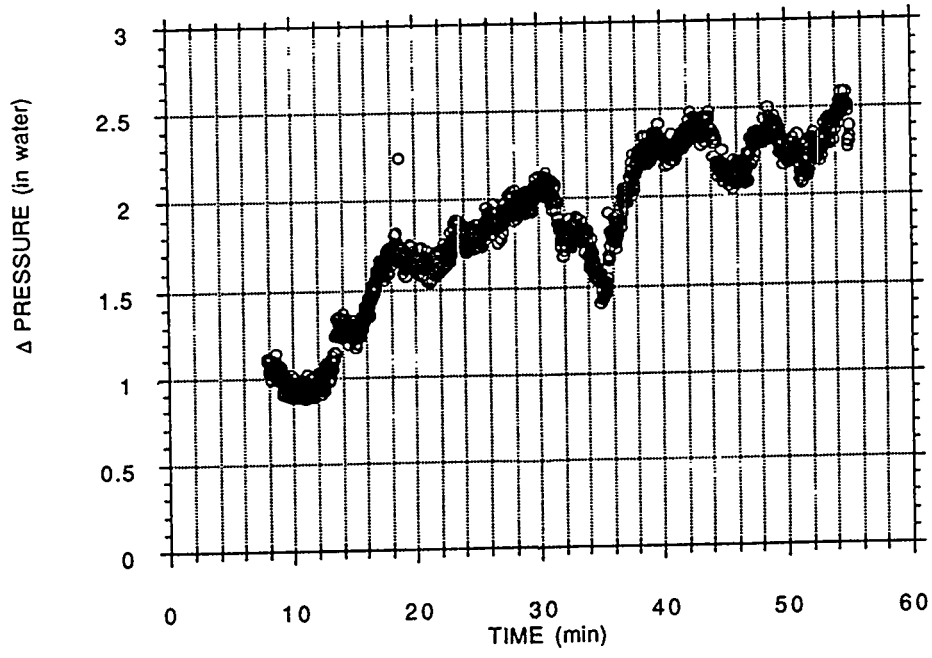


Figure 17 Filter clogging test with no aerosol on a standard glass HEPA filter at 40 cfm and 100% relative humidity

Our final laboratory test was conducted to illustrate purging a steel filter saturated with water. Figure 18 shows the pressure drop, percent relative humidity, and downstream temperature of a Memtec filter that had been fully saturated with water. Purging the filter drove off the condensed water and reduced the filter pressure drop. The evaporated water raised the relative humidity to 100% and lowered the temperature of the air downstream of the filter to about 11 C. The inlet air temperature and relative humidity were 20.7 C and 36% respectively. As the amount of condensed water in the filter decreased, the pressure drop decreased. With less water available for evaporation, the temperature increased and the relative humidity decreased. Figure 18 shows that the evaporation is complete after about 120 minutes.

The laboratory purge test on the water-saturated Memtec filter in Figure 18 was able to duplicate the field observations of the Memtec filter. Figure 11 shows the pressure drop of the Memtec filter at the beginning of the field evaluation decreased from 7 inches to 4.5 inches after 14 hours of purge. The relative humidity downstream of the filter decreased from 90% to 53% during this period. The Memtec filter also showed a drop in pressure drop from 9 inches to 3 inches after sitting idle for nearly two weeks. Evaporation of condensed water in the filter is the only reasonable explanation. The large decreases in filter pressure drop in Figure 11 due to the air purge and to the two-week idle period automatically eliminate the possibility that uranium oxide particles were responsible for the observed increases in filter pressure drop.

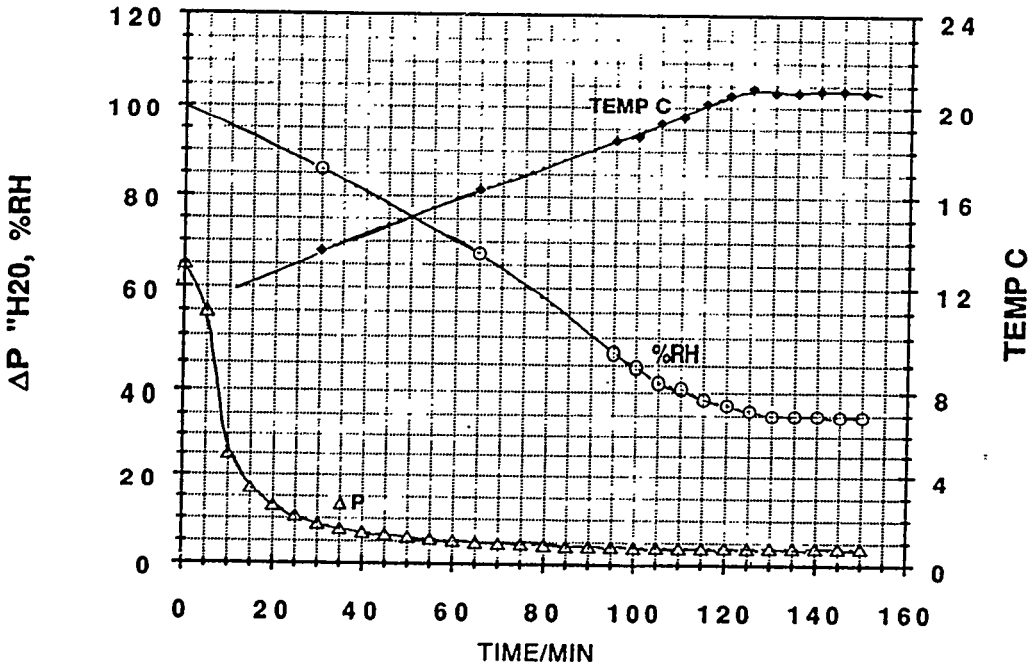


Figure 18. Purging a water-saturated Memtec filter with 15,6 cfm dry air

V. Conclusions

We have conducted a preliminary evaluation of two high efficiency steel filters in the exhaust of an uranium oxide grit blaster at the Y-12 Plant in Oak Ridge Tennessee. Test results showed both filters had a rapid increase in pressure drop with time, and reverse air pulses could not decrease the pressure drop. Subsequent laboratory tests on a single filter cartridge confirmed that water accumulation in the filter was responsible for the increase in filter pressure drop and the inability to lower the pressure drop by reverse air pulses. No effort was made to identify the source of the water accumulation and correct the problem because the available funds were exhausted.

V. Bibliography

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DISCUSSION

WREN: Have you correlated the pressure drop with the adsorbed amount of water on steel? The adsorption capacity of steel for water may not be linear but of a higher order dependance with water pressure. Thus the significant pressure drop may only be seen at high RH.

BERGMAN: We conducted a test at 100% relative humidity (see Figure 16) that showed water adsorption on the steel filter was responsible for a rapid increase in pressure drop. Unfortunately we did not have time to do a systematic study of water adsorption at other relative humidities.

HULL: Did I assume correctly that you did this in the summertime at Oak Ridge? I remember years ago when I was working down there that it certainly felt like the humidity was somewhere up around ninety percent during the summertime. Is that where the moisture came from?

BERGMAN: The tests were performed in April and May when there was a great deal of rain. There were heavy showers during the Pall filter evaluation.

CLOSING COMMENTS OF SESSION CO-CHAIRMAN DORMAN

We have had five papers, all dealing in some way with filters. Mr. Leonard gave his wide-ranging review of efficiency, testing, aging, quality control and so on. His paper wasn't such as to assure us that all is perfect in the filtration world. We then had two papers dealing with loading and pressure drop. Now this is a subject in which it is possible to have an infinitude of results, depending on particle size, hygroscopicity, density, shape, velocity, fiber diameter, and so on. All one can hope for in practice is to get something approaching qualitative answers and not quantitative. Nevertheless, it is highly important that we get results on this type of testing, because they help in forecasting what will happen with other types of filters; again, only qualitatively.

Figure 4 in both Mr. Dyment's and M. Vendel's paper showed rough agreement in the filter resistance vs. loading levels, up to about 300 g/m^2. Both authors gave 2 μm as the mass median diameter of the aerosol they used. I think the agreement is probably quite fortuitous, as the filters were of entirely different construction. Nevertheless, it is a point to note.

I also found it surprising in M. Vendel's Figure 4 that there was a bigger increase in pressure drop with the aerosol under dry and low humidity conditions than there was with very high humidity. I think this must be due to the hygroscopic nature of the aerosol. At high humidities, I am sure that a film is formed on the fibers, thus producing a smaller resistance rise until the liquid begins to fill the holes completely. I believe this was the thrust of one of the questioners.

The paper by Dr. Bergman suffered from the fact that he got no financial support. Nobody is rushing to give him any money at the moment, I see, but we live in hope.

The paper on magnetic filtration by Professor Watson is highly specialized. I think it is a bit out of the field in which most of us have worked. Perhaps I did not understand all the points. Perhaps some of you did. Magnetic filtration is different from other mechanisms in the fact that it is not obeying the ordinary rules of filtration, which are diffusion, interception, and inertia, and then the "stickability" of particles on the fibers. However, it is the third paper dealing with metal fibers and filters. These papers are indicative of the current interest in the performance of filters at elevated temperatures and under corrosive conditions. I foresee that for such conditions there will be a need for continuing work, and I think we shall be getting follow-up papers at the next conference.

Taken all in all, it has been quite an interesting session, with a wide variety of papers, from a general review to magnetic filters. I think Dr. Bergman's work has a future. Maybe the more we stress the finances, the more likely he is to get the money to carry on.

SESSION 6

PANEL SESSION: AEROSOL SAMPLING IN EFFLUENT AIR STREAMS

Tuesday: July 26, 1994
Chairman: A. R. McFarland

Panel Members: K. Duvall
J. A. Glissmeyer
J. M. Karhnak
G. J. Newton

OPENING COMMENTS OF SESSION CHAIRMAN MCFARLAND

SINGLE POINT AEROSOL SAMPLING: EVALUATION OF MIXING AND PROBE PERFORMANCE IN A NUCLEAR STACK
J. C. Rodgers, C. I. Fairchild, G. O. Wood, C. A. Ortiz, A. Muyshondt, A. R. McFarland

RADIONUCLIDE AIR EMISSIONS AT DEPARTMENT OF ENERGY FACILITIES
K. Duvall

EPA PERSPECTIVE ON RADIONUCLIDE AEROSOL SAMPLING
J. M. Karhnak

GUIDE TO SAMPLING AIRBORNE RADIOACTIVE MATERIALS IN NUCLEAR FACILITIES
J. A. Glissmeyer

PANEL DISCUSSION

CLOSING COMMENTS OF SESSION CHAIRMAN MCFARLAND

OPENING COMMENTS OF SESSION CHAIRMAN MCFARLAND

The session this morning is entitled, "Aerosol Sampling in Effluent Air Streams", but I think the title could also be, "What Has Been and What May Be in Aerosol Sampling." The current ANSI air sampling standard needs many changes. The current version contains problems that call for revisions that are currently under way. The last time the ANSI standard was up for revision, it was reaffirmed for another ten years. The latest revision has been a rather substantial exercise, as you will see from John Glissmeyer's presentation. I might mention that several of the committee members who participated in writing the ANSI standard are present as panelists and in the audience, and we would like to get your feedback on the proposed ANSI standard. I would like to start by describing the existing regulatory environment and then indicate where we are now and where the ANSI standard may eventually lead us.

SINGLE POINT AEROSOL SAMPLING: EVALUATION OF MIXING AND PROBE PERFORMANCE IN A NUCLEAR STACK*

John C. Rodgers[1], Charles I. Fairchild[2], Gerry O. Wood[2], Carlos A. Ortiz[3], Arnold Muyshondt[3] and Andrew R. McFarland[1,3].

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[3]Aerosol Technology Laboratory, Department of Mechanical Engineering, Texas A&M University, College Station, TX 77843.

Abstract

Alternative Reference Methodologies (ARMs) have been developed for sampling of radionuclides from stacks and ducts that differ from the methods required by the U.S. EPA. The EPA methods are prescriptive in selection of sampling locations and in design of sampling probes whereas the alternative methods are performance driven. Tests were conducted in a stack at Los Alamos National Laboratory to demonstrate the efficacy of the ARMs. Coefficients of variation of the velocity tracer gas, and aerosol particle profiles were determined at three sampling locations. Results showed numerical criteria placed upon the coefficients of variation by the ARMs were met at sampling stations located 9 and 14 stack diameters from flow entrance, but not at a location that is 1.5 diameters downstream from the inlet. Experiments were conducted to characterize the transmission of 10 μm aerodynamic equivalent diameter liquid aerosol particles through three types of sampling probes. The transmission ratio (ratio of aerosol concentration at the probe exit plane to the concentration in the free stream) was 107% for a 113 L/min (4-cfm) anisokinetic shrouded probe, but only 20% for an isokinetic probe that follows the EPA requirements. A specially designed isokinetic probe showed a transmission ratio of 63%. The shrouded probe performance would conform to the ARM criteria; however, the isokinetic probes would not.

I. INTRODUCTION

The U.S. Department of Energy (DOE) is required under the U.S. Environmental Protection Agency (EPA) National Emission Standards for Hazardous Pollutants (NESHAPs) to continuously monitor radionuclide emissions from stacks and ducts that could contribute more than 0.1 millirem per year to the most affected member of the public[1]. The NESHAPs require use of EPA Method 1[2] for determining the location of the sampling station in the duct, and use of American National Standards Institute N13.1-1969[3] for guidance in conducting the sampling. EPA Method 1 states that the sampling should be no closer than eight duct diameters from the nearest upstream flow disturbance (elbow, fan, etc.) and no closer than two duct diameters from the nearest downstream disturbance: This so-called '8- and 2-criterion,' is intended to provide users with assurance that the sampling site is suitable for collection of representative samples with the minimum number of sampling points (probes). Closer spacing between the sampling plane and the nearest disturbances is allowed if the '8- and 2-criterion' cannot be met, provided larger numbers of sampling points are used. EPA Method 1 also requires that the average swirl angle in the flow should not

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exceed 20°, which ostensibly limits problems that might be created by off-axis sampling by probes and minimizes errors in flow measurements in stacks and ducts..

ANSI N13.1-1969 serves several roles in implementation of the requirements of the nuclear NESHAPs. First, it is intended to provide guidance on the number of sampling points that should be used at a given site, with larger ducts requiring more sampling points than smaller ducts, and rectangularly-shaped ducts requiring more sampling points than circular ducts. As many as 20 sampling points are recommended for large rectangular ducts. However, the ANSI standard recognizes that fewer points may be used if careful evaluation of the sample extraction location shows that the concentration profile is relatively flat as a result of good mixing in the stack or duct. Second, the ANSI standard provides guidance on the design of probes; it recommends sharp-edged probes followed by 90° bends, with a constant internal diameter from the inlet through the elbow. Third, when multiple probes are required under the guidance of the ANSI standard, it provides designs for rakes of such probes.

It has been known for some time[4,5,6] that the methodology prescribed in the NESHAPs needed to be improved and updated. Use of the '8- and 2-criterion' is not a reliable predictor of stack mixing conditions. In particular, it does not provide assurance that fluid momentum and contaminant concentration are both well mixed at the sampling location. Hampl et al.[7] showed that 50 duct diameters may be needed for mixing of a tracer gas in a straight pipe whereas only two duct diameters were needed for mixing downstream of two elbows in series that are placed out-of-plane. Turner et al. [5] showed that representative aerosol samples could be obtained at a distance of 1.5 diameters from a downstream disturbance (elbow).

Use of ANSI-type probes can lead to significant internal wall losses of aerosol particles. Fan et al.[8] tested such a probe and found that approximately 75% of liquid 10 μm aerodynamic equivalent diameter (AED) aerosol particles were impacted on the internals walls and only 25% transmitted through an ANSI probe to a filter collector. As a consequence of these limitations, Los Alamos National Laboratory has prepared Alternative Reference Methodologies (ARMs) for representative sampling of stacks and ducts for emissions of radionuclides[6.] These have been submitted to the EPA Administrator for approval under the provisions of 40CRF61, Subpart H.

The core concept of the performance-based ARMs is that true representative sampling of stack effluents, whether contaminated by gaseous or particulate radioactive contaminants, requires that the contaminants have become well mixed with the effluent flow across the entire cross sectional area at the sampling location. Good mixing can be the result of natural turbulence in the flow, or as a result of the use of engineered mixing devices. A most important consequence of requiring demonstration that mixing at the desired sampling location meets certain performance criteria, is that sample extraction from a single point in that profile is amply justified.

We show here that the most accurate and effective method of achieving continuous representative sampling of radioactive aerosol effluents is through the use of a suitably designed shrouded probe extracting samples from a single properly prepared and located point in the flow. There are two components of the ARMs proposed for achieving representative samples from a single point. The first component is the use of numerical performance criteria for determining the suitability of a sampling location in lieu of the present prescriptive method. Extractive sampling will take place at suitably qualified locations where both fluid momentum (manifested by the shape of the velocity profile) and contaminant concentration (characterized by the shape of the concentration profile) are demonstrated by measurements to be well mixed. If only gaseous radionuclides could be sampled at the site, the criteria for suitability are that the coefficients of variation in the data for the velocity profile and the concentration profile of a tracer gas will each be ≤20% over the center 2/3 of the stack or duct area. The coefficient of variation, *COV* is defined as the ratio of the standard deviation of a data set to the mean value of the data set, i.e.:

$$COV = \frac{s}{\bar{x}} \qquad (1)$$

where the mean and standard deviation of the data are defined as:

$$\bar{x} = \frac{1}{n}\sum_{i=1}^{n} x_i \qquad (2)$$

and:

$$s = \sqrt{\frac{1}{n-1}\sum_{i=1}^{n} (x_i - \bar{x})^2} \qquad (3)$$

The parameter n is the number of data points; and, x_i is the value of the random variable (velocity or tracer concentration) at the ith location on a sampling grid.

To address the possibility of narrowly confined, high concentration flow envelopes being averaged out in the general performance criterion, an additional requirement is that over a grid set up in accordance with EPA Method 1[2], the concentration of tracer gas at any point will not be more than 30% greater than the mean concentration across the duct cross section. If aerosol radionuclide particles could be sampled at the site, the suitability criteria are the same as for gaseous radionuclides, but with the additional requirement that the *COV* of 10 μm AED aerosol particles will be ≤ 20% over the center 2/3 of the duct.

To preclude the possibility of significant emissions from a secondary flow being trapped in the boundary layer of a primary flow, no lateral flows may be introduced at a location downstream of the fan in a primary duct in a manner in which the secondary flow entrance would be flush with the wall of the primary flow duct without provision for downstream mixing elements which achieve complete mixing of the flows. This would not be a problem with junctions where the flows are of approximately the same magnitude.

The second component of the ARMs is the use of an anisokinetically operated shrouded probe for single point sampling of aerosols. This probe was developed by McFarland et al.[9]. Such a probe concept is a break with the provisions of ANSI N13.-1969 methods, which emphasize isokinetic sample withdrawal from multiple points in the profile to overcome limitations in mixing. A properly designed shrouded probe, operated at a single location in a well-mixed, stable profile, will provide more representative samples than a rake of numerous small probes due to dramatically reduced wall losses of larger size particles. For a shrouded probe to be acceptable for a given application, the design must have been tested in an aerosol wind tunnel with 10 μm AED aerosol particles over the range of anticipated operational free stream velocities and sampling flow rates. The transmission ratio must be between 0.80 and 1.30 for these conditions.

At Los Alamos National Laboratory, a Waste Assay Facility (WAF) has been constructed that will serve the role of providing non-intrusive examinations of containers of radioactive waste prior to their disposal. Building ventilation air from the WAF is passed through HEPA filters before being discharged to the environment through two stacks (one that is 250 mm, or 10-inches inner diameter, and the second that is 300 mm or 12-inches inner diameter); however, because of the potential for emissions of radionuclides, the stacks will be continuously monitored. The WAF stacks are new and preceded by HEPA filters, so it is unlikely that they have been contaminated. As a consequence, we selected this facility for studies on emissions monitoring. Tests were carried out in the 300 mm diameter stack. With reference to Figure 1, effluent air from the WAF passes through the bank of HEPA filters, into an induced draft fan and then into the 300 mm diameter stack. Air, discharged from the fan, enters the stack through a rectangularly-shaped lateral element on the south side of the stack. Thus, the flow pattern in the stack initially has a

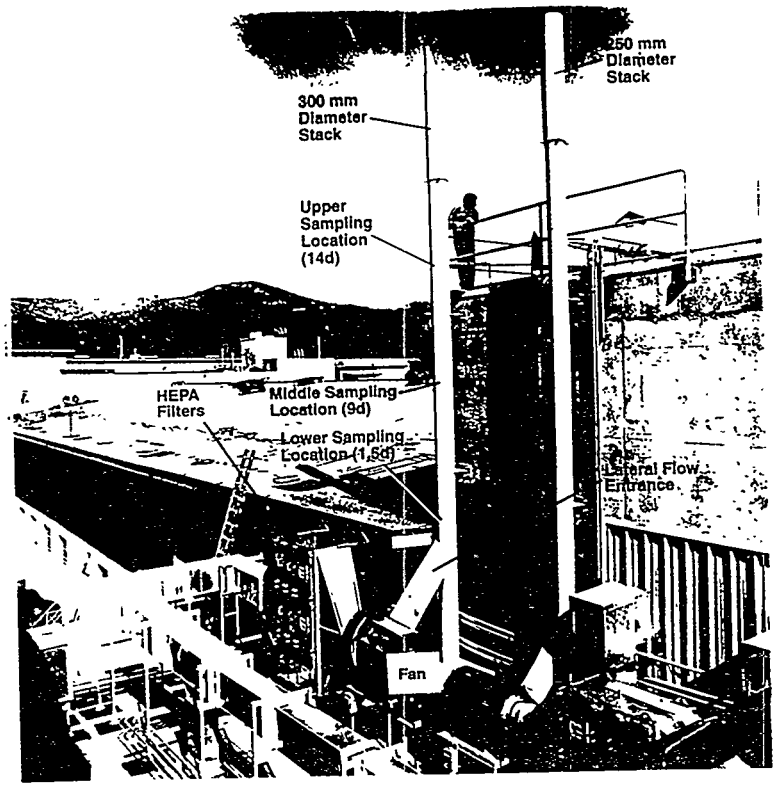


Figure 1. The air exhaust stacks of the Waste Assay Facility at Los Alamos National Laboratory.

pronounced north-south axis of disturbance. Based on pitot tube measurements that were taken by a facilities contractor before we started the study, the nominal mean velocity in the stack was assumed to be 21 m/s. Sampling stations, Figure 2, were placed in the stack at distances of 1.5 diameters, 9 diameters and 14 diameters from the flow entrance location. Totally, the height of the stack is over 20 diameters.

For single point representative sampling to be appropriate, the site must be qualified in terms of meeting numerical mixing criteria for both fluid momentum and contaminant concentration as manifested by the uniformity of the velocity and concentration profiles. In our study of the application of the proposed methodology and criteria in an unmodified operating stack, measurements were made at the three sampling locations of the velocity and concentration profiles. Two types of tests were conducted to characterize the concentration profiles; one set of tests dealt with a tracer gas and the second set dealt with aerosol particles. Sulfur hexafluoride was used as the gas tracer and oil droplets (oleic acid tagged with an analytical tracer) were used as the test aerosol.

Aerosol sampling experiments were conducted with both shrouded probes and isokinetic probes at a qualified location. We tested two different shrouded probes that had been designed to accommodate two different sampling flow rates, and made a comparison of their performances with those of corresponding isokinetic sampling probes. For these tests, 1 to 20 μm AED aerosol particles were used to challenge the probes.

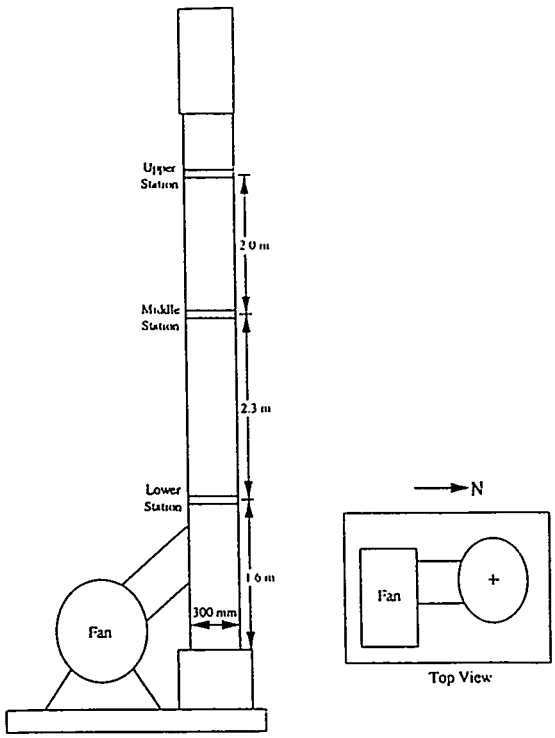


Figure 2. Location of sampling stations on the 300 mm diameter stack.

II. EXPERIMENTAL METHODS

Velocity Profiles

Velocity data were obtained at each of the three sampling locations with a two-channel hot film anemometer (TSI Model IFA 100/200, TSI, Inc., St. Paul, MN). The grid over which the velocity values were taken at each sampling location is shown in Figure 3. The hot film anemometer was initially calibrated at five different velocities in a free air jet against a pitot tube to establish the relationship between instrument output and air velocity. A daily single point calibration was used for assurance that the calibration had not shifted.

Velocity data from a hot film device are output in terms of a fixed set of reference conditions. These data were converted to actual velocity values in the stack through use of:

$$V = V_{ref}\left(\frac{P_{ref}}{P}\right)\left(\frac{T}{T_{ref}}\right) \qquad (4)$$

where: V = velocity; P = pressure; T = temperature; the subscript *ref* refers to the reference conditions for the hot film output; and, the unsubscripted parameters refer to the actual stack conditions.

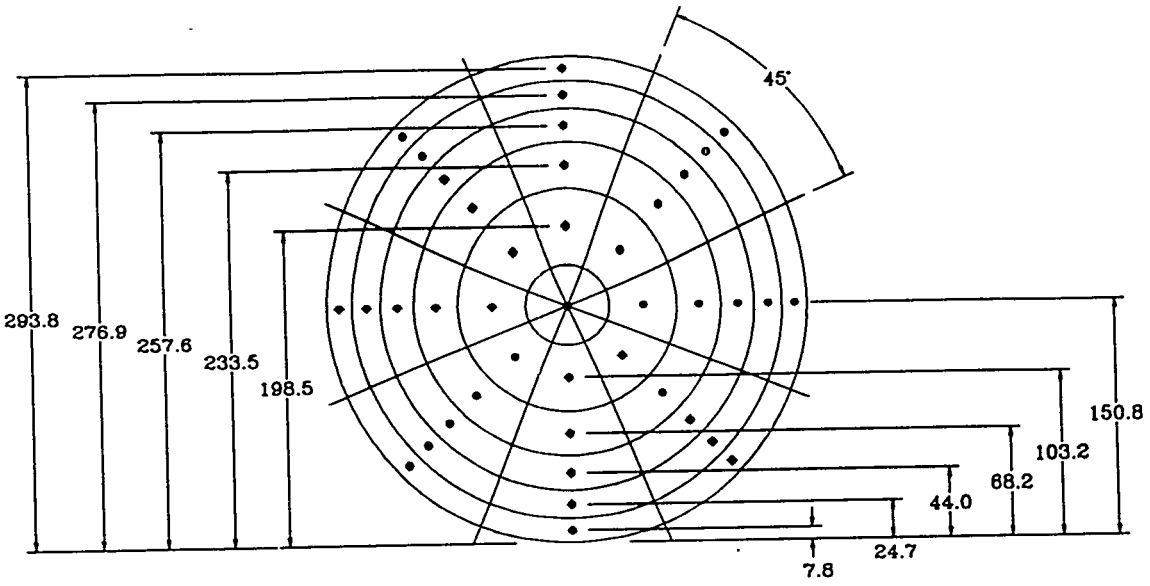


Figure 3. Grid over which velocity readings were taken. All dimensions are in mm.

Tracer Gas Profiles

Sulfur hexafluoride (SF_6) was introduced into the center of the lateral element at the stack entrance. A multipoint probe was used to sample SF_6 at various locations on two perpendicular diameters in the stack at each sampling location. The traverses were selected to be on north-south and east-west axes due to the orientation of the injected flow. Sampling positions in the stack were at distances of 13, 25, 46, 69, 104, 200, 236, and 279 mm (0.5, 1, 1.8, 2.7, 4.1, 7.9, 9.3, 10.2, 11, and 12 inches). The SF_6 concentration was determined with a photoacoustic infrared spectrometer (Multi-gas Monitor, Type 1302, Bruel & Kjaer, Naerum, Denmark).

Aerosol Concentration Profiles

Monodisperse particles were generated with a Berglund-Liu vibrating jet atomizer (TSI, Inc., St. Paul, MN) from the mixture of oleic acid and the analytical tracer, sodium fluorescein, dissolved in isopropyl alcohol. This aerosol was introduced into the center of the rectangular lateral flow element, which is located just upstream of the stack. Light scattering particle counters (MET-1, Grants Pass, OR) were used to measure the particle concentrations in the stack. Average particle size generated by the vibrating jet atomizer was 10.5 μm, which allowed a channel in an optical particle counter with a lower limit of 10 μm to provide size discrimination. The actual particle size was determined microscopically using the technique of Olan-Figureroa et al.[10]. Two particle counters were operated simultaneously at a given sampling location during a stack testing. One particle counter sampled through a probe from a position near the center of the stack profile. Data from this device was used as a reference for the experiments. The second particle counter sampled through a probe that was sequentially placed at each position on a traverse across the stack profile. The initial point was 25 mm (1 inch) from the stack wall and subsequent points were spaced 50 mm (2 inches) apart. Two traverses, at 90° to each other, were made at each of the sampling locations. The traverses were oriented so that one was along a north-south axis and the other along an east-west axis. Triplicate measurements were conducted at each location.

Tests of Sampling Probes

The Berglund-Liu vibrating jet atomizer was used to generate monodisperse aerosol, which was introduced into the lateral entrance section of the stack. Testing of the probes was performed only at the upper (14 diameter) location. The test protocol consisted of operating each probe alternately at the center of the stack for a period of 5 minutes and then replacing that probe with the next to be tested.

A set of tests was conducted to determine the effect of particle size on aerosol transmission through the probes. These tests were conducted with particle sizes from 1 to 20 μm AED at a velocity of 25 m/s. A second set of tests explored the effect of velocity upon the transmission of 10 μm AED aerosol particles. Here, the probes were tested at free stream velocities of 13 and 25 m/s. At least four replicate tests were conducted with each probe at each set of experimental conditions.

One of the isokinetic probes was constructed similar to the recommendations given in the ANSI standard -- it consists of a sharp edged inlet that is 6.5 mm (0.255 inch) in diameter followed by an expansion to 8.7 mm rather than having a constant internal diameter. Because the straight section of the probe and the subsequent elbow have a larger internal diameter than the inlet, it is to be expected the wall losses in this ANSI probe would be less than those in a probe that perfectly matches the ANSI recommendation. For the experiments reported herein, a filter was placed at the exit of the elbow. In the discussion that follows, this probe shall be referred to as the 'ANSI' probe.

A second isokinetic probe was fabricated following the design of Chandra[11]. It has a sharp-edged inlet that is 7.54 mm (0.297 inches) in diameter and it is followed by a gradual expansion of the flow stream to a diameter of 32 mm (1.25 inches). A filter sampler was placed at the exit of the expansion. In the discussion that follows, this probe shall be referred to as the 'isokinetic' probe.

Two shrouded probes were tested to determine aerosol transmission; one of the shrouded probes was designed to be operated at a nominal flow rate of 57 L/min (2 cfm) and the second was designed to be operated at 113 L/min (4 cfm). The shroud diameter of the 57 L/min unit was 50 mm (2-inches) and the diameter of the inner probe inlet was 15.5 mm (0.610-inches). The corresponding dimensions of the 113 L/min unit were a shroud diameter of 75 mm (3-inches) and an inlet diameter of the internal probe of 20.8 mm (0.818 mm).

Typically in the nuclear industry, the nominal flow rate for a stack sampling device is 57 L/min (2 cfm); however, it is commonplace to have two sampling systems operated at the same location with one used for alarming purposes and the second for collection of archival samples. In some applications a 113 L/min (4 cfm) probe is used to collect samples for both purposes. A flow splitter, placed outside of the duct, divides the flow stream so that a representative sample will be provided to each sampling device.

The parameter of principal interest in characterizing the probes is the transmission ratio, T, which is defined as the ratio of aerosol concentration at the exit plane of the sampling system to the aerosol concentration in the free stream. The parameter is determined for liquid aerosol particles and takes into account losses on the internal walls of a probe. Symbolically, it is expressed as:

$$T = \frac{C_e}{C_\infty} \tag{5}$$

where C_e = aerosol concentration in at the exit plane of the probe; and, C_∞ = aerosol concentration in the free stream. The parameter C_e is established from measurements of the aerosol mass that is transmitted through the probe and that which is collected on the filter, together with data on the volume of air sampled by the probe. Aerosol concentration in the free stream was determined from use of the Chandra-type probe. That probe was operated isokinetically, so the aerosol concentration at the 'isokinetic' probe inlet, $C_{i,iso}$ was the same as the free stream concentration, i.e.:

$$C_{i,iso} = C_{\infty} \tag{6}$$

A sample collected by a filter at the exit of this probe is deficient because of losses of aerosol particles to the internal walls of the probe, i.e.:

$$C_{e,iso} = C_{i,iso} - C_{wl,iso} \tag{7}$$

where: $C_{e,iso}$= aerosol concentration at the exit plane of the 'isokinetic' probe; and, $C_{wl,iso}$= aerosol concentration that is lost to the walls. In these experiments, the wall losses from the 'isokinetic' (Chandra-type) probe were recovered by washing the internal walls of the probe with isopropyl alcohol. Combining the concentration determined from the wall losses together with the concentration determined from aerosol transmitted through the probe allowed calculation of concentration at the probe inlet, which, from Equation 6, provided the value of the free stream aerosol concentration.

III. RESULTS

Velocity Profiles

With reference to Figures 4, plots are shown of the velocity profiles at the three sampling locations. Average velocity in the stack at operational conditions was about 25 m/s; however, data were also taken with the stack operated at about 1/2 that velocity to determine if the stack flow Reynolds number would significantly affect the mixing.

The profile at the 1.5 diameter station (Figure 4a) shows a reverse flow on the south side of the stack where the flow enters laterally, and a high speed region on the opposite side of the stack (north) where the velocity reaches a value of approximately 32 m/s. The *COV* of velocity at this station, calculated for the entire flow, is 28% while that over the region that includes 2/3 of the stack cross sectional area is 22%. These values, together with other *COV*s are shown in Table 1.

Data obtained at the second (9 diameters) level are shown in Figure 4b, where it may be noted the back flow has disappeared and the profile is much more uniform than at the lower level. However, there is still an excess velocity on the south side as compared with the north side. The coefficient of variation for the entire profile is 13% while that for the center 2/3 of the stack cross sectional area is 6%.

The velocity profile at the upper sampling station, which is 14 diameters downstream from the lateral entry, is shown in Figure 4c. Here, the profile is well developed, with a *COV* across the entire cross section of 12% and a *COV* of 4% for the center 2/3 of the stack. To determine if there was a flow Reynolds number influence on mixing, we measured the velocity profile at a flow rate of approximately 1/2 that of the normal operational value for the system. With reference to Figure 4d, the velocity profile for the middle (9 diameter) station at the reduced flow rate is still well developed and has a *COV* of 16% for the entire cross section.

Tracer Gas Profiles

Average velocity in the stack was 23 m/s when the SF_6 measurements were made. The SF_6 concentration profile at the lower level is shown in Figure 5a. The units of concentration are relative, with the measured concentration at each point normalized to the mean concentration. The range of relative concentration values shown in Figure 5a is 0.59 to 1.39. The *COV* is 26% for both the entire data set and for the center 2/3 of the duct area.

Mixing of tracer gas is much improved at the 9 diameter station as compared with that at the 1.5 diameter location. A plot of the concentration profile at the former location is shown in Figure 5b. The *COV* is 5.9% full data set, and 4.2% for the center 2/3 of the stack. At no location on the entire grid is the

Table 1. A comparison of the uniformity of velocity and concentration profiles recommended in the Alternative Reference Methodologies with the values experimentally observed in the WAF stack. All data are for the high velocity condition.

Criterion	ARM acceptability criteria	1.5 diameter location	9 diameter location	14 diameter location
Velocity *COV* over the center 2/3 of stack area	≤20%	27%	6%	4%
Velocity *COV* over the entire grid	No criterion	28%	13%	12%
Tracer gas *COV* over the center 2/3 of the stack area	≤20%	26%	4.2%	2.1%
Maximum of tracer gas relative to the mean	≤30%	39%	12%	5%
10 μm AED aerosol particle *COV*. Center 2/3 of stack area	≤20%	74%	5%	5%
Average swirl angle	≤20°	9°	6°	9°

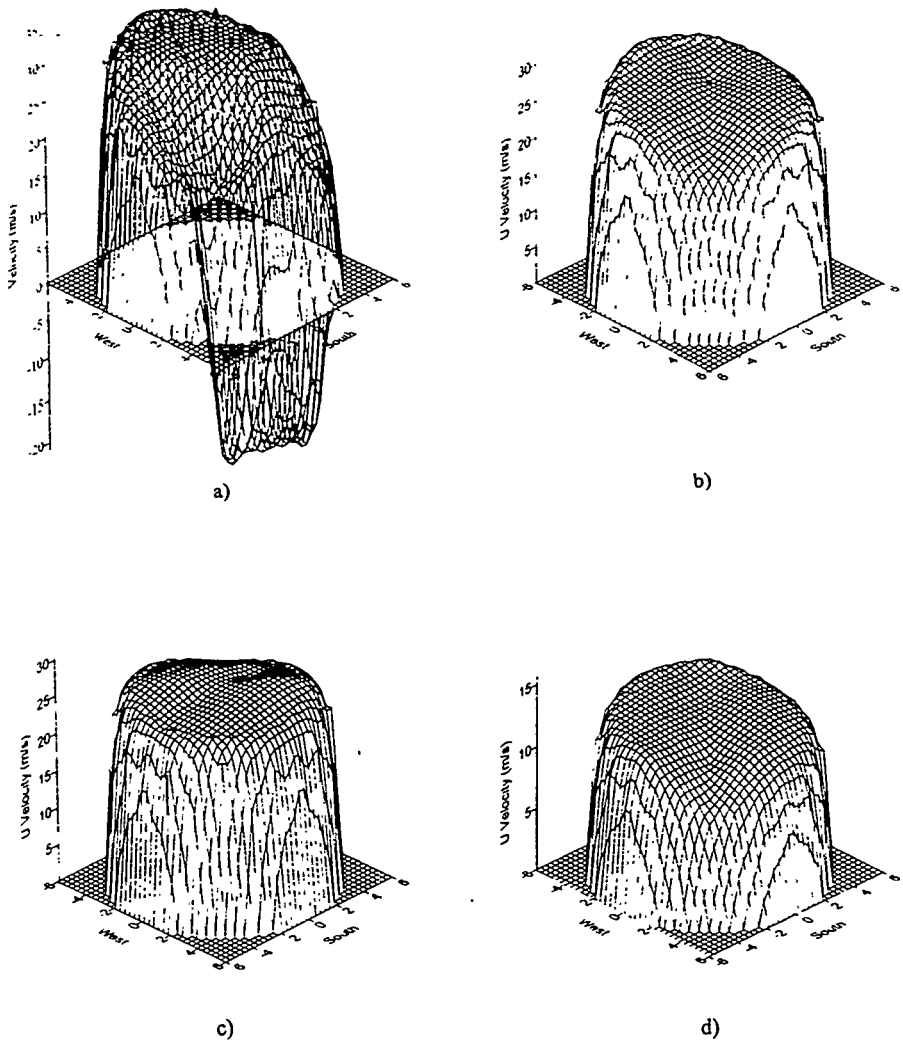


Figure 4. Velocity profiles at the three sampling locations: a) the 1.5 diameter location when the mean velocity was 26 m/s; b) the 9 diameter location when the mean velocity was 26 m/s; c) the 14 diameter location when the mean velocity was 26 m/s; and, d) the 14 diameter location at when the mean velocity was 11 m/s.

concentration more than 12% greater than the mean concentration (the range of measured concentration values was 0.92 - 1.12).

At the upper (14 diameter) location, Figure 5c, the mixing is slightly improved over that at the 9 diameter location. The coefficient of variation for the full set of data points at the upper level is calculated to be 2.8% and the *COV* for the center 2/3 of the stack area is 2.1%. The range of concentration values was 0.97 to 1.05.

Aerosol Concentration Profiles

The aerosol concentration profiles for the 1.5, 9 and 14 diameter locations are shown in Figure 6. The particle size for these data is 10.5 μm and the average velocity in the stack was 24 m/s. The concentration profile at the 1.5 diameter location, Figure 6a shows considerable skewness. Aerosol was introduced into the lateral on the south side of the stack and the data for a north-south traverse show the

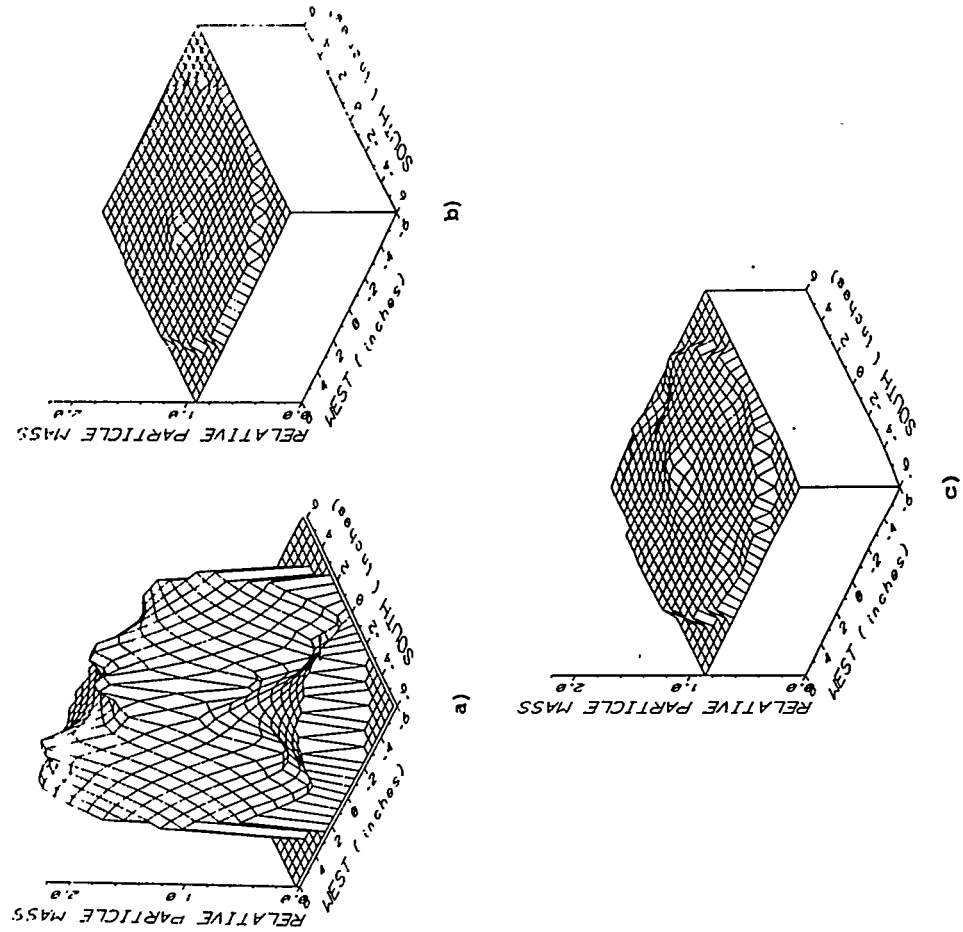


Figure 6. Aerosol concentration profiles for 10.5 µm diameter particles. The average stack velocity during these tests was 24 m/s. a) The 1.5 diameter location. b) The 9 diameter location. c) The 14 diameter location.

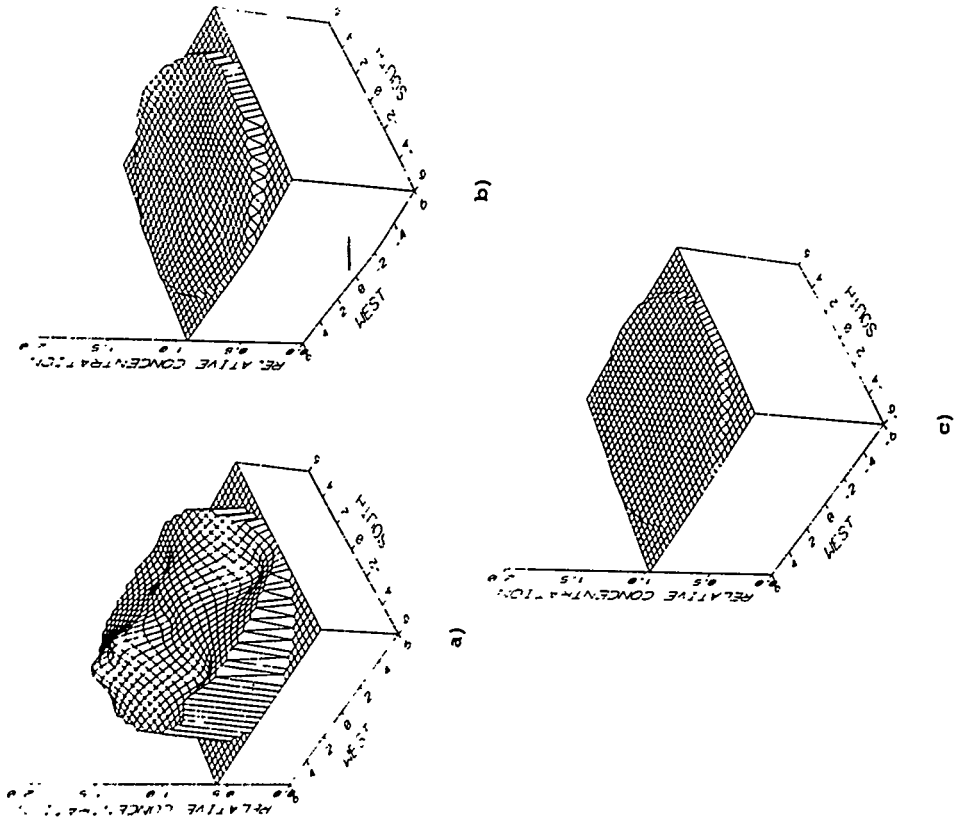


Figure 5. Tracer gas concentration profiles. Mean stack velocity was 23 m/s during these tests. a) The 1.5 diameter location. b) The 9 diameter location. c). The 14 diameter location.

south side has a concentration defect and the north side has considerable concentration enrichment. The peak concentration on the north side of the stack is about 200 relative mass units while a near-zero concentration was measured at a distance of 25 mm (1 inch) from the wall on the south side. The *COV* for the full data set at this location is approximately 80% while that for the center 2/3 of the stack is 74%. Also, the maximum concentration across the entire grid is 212% of the mean concentration.

The large-scale mixing produced by the lateral entrance has a significant effect on the particle concentration profile as may be observed from the data taken at the middle station (9 diameters), Figure 6b. Both of the traverses show relatively uniform concentration values. The *COV* of the entire data set is 4% and that for the center 2/3 of the duct area is 5%, where the latter value is an acceptable level under the ARMs (which stipulates the maximum *COV* should not exceed 20% over the center 2/3 of the stack). The ratio of the maximum concentration to the mean concentration across the entire sampling grid is 8%.

The traverses for aerosol concentration at the upper level produced the data shown in Figure 6c. A *COV* of 8% is associated with the full data set, while the *COV* for the data points that correspond to the center 2/3 of the stack area is 5%. The maximum concentration is 14% greater than the mean value over the entire sampling grid.

Probe Performance

The transmission ratios as functions of particle sizes for the four tested probes are shown in Figure 7. At a particle size of 1.0 μm AED, the transmission ratio of all probes is approximately unity. Larger particle sizes had an adverse effect upon the ANSI probe transmission performance. At 10 μm AED, the transmission ratio was 20% and at 20 μm AED, the transmission was only 4%. In contrast, both shrouded

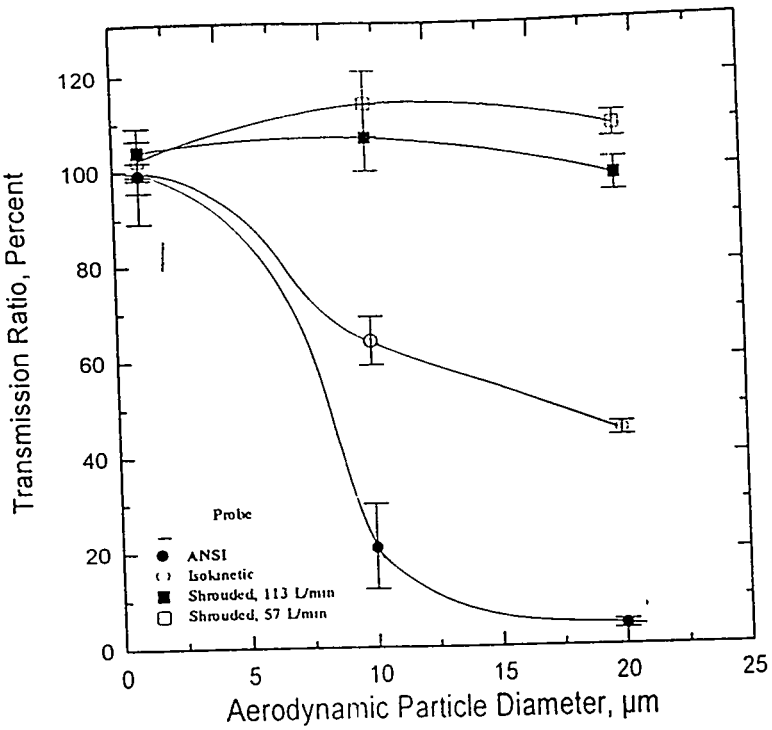


Figure 7. Effect of particle size on the transmission ratios of four different aerosol sampling probes. Mean velocity in the stack during these tests was 24 m/s.

probes showed relatively constant performance, with the 113 L/min (4 cfm) shrouded probe having transmission ratio values from 98% to 107% over the range of particle sizes of 1 to 20 μm AED. The 57 L/min (2 cfm) shrouded probe showed transmission values from 102 to 115% over the same range of particles sizes. Data for the 'isokinetic' probe showed transmission values intermediate to those of the ANSI and shrouded probes, with observed transmission ratio being 63% for a particle size of 10 μm AED.

The effect of stack velocity upon the transmission ratio of 10 μm AED aerosol particles through the various probes are shown in Figure 8. The transmission ratio of the 113 L/min shrouded probe changes from 107% to 92% (a relative change of 14%) as the velocity is decreased from 25 m/s to 13 m/s. In contrast, for the same change in velocity, the transmission ratio of the ANSI probe increases from 20% to 33% (a relative change of 65%). The transmission ratio of the 'isokinetic' was constant at about 63% as the velocity was changed.

Under the Alternate Reference Methodologies, a qualified probe will need to have a transmission ratio within the range of 80% to 130% for 10 μm AED aerosol particles and for the anticipated range of operational conditions. The data from stack tests show both of the shrouded probes (57 L/min and 113 L/min flow rate units) meet these criteria; however, neither the ANSI probe nor the Chandra-type 'isokinetic' probe would be suitable.

Estimate of Experimental Errors.

Replicate tests were conducted with each type of experiment. The normalized average standard deviation (standard deviation of the velocity measurements at each point divided by the mean at that point) of the velocity profile tests was 7% for the upper station, 9% for the mid station, and 16% for the lower station. For tests with tracer gas, the normalized average standard deviation averaged for all sampling

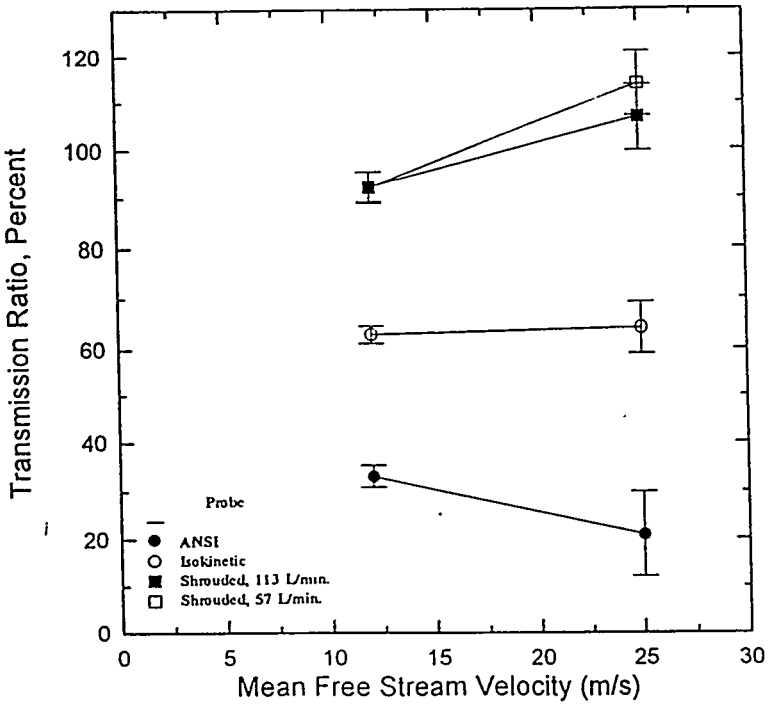


Figure 8. Effect of stack velocity on the transmission ratios of sampling probes. Particle size for these tests was 10 μm AED.

stations was 2%. Data for tests with aerosol particles showed the normalized average standard deviation was 18% at the lower lever and 10% at both the intermediate and upper levels. With respect to reproducibility of tests with probes, error bars that represent ± one standard deviation on the transmission ratio are shown in Figure 7.

IV. DISCUSSION

A comparison of the experimental results with the criteria presented as Alternative Reference Methodologies (ARMs) is given in Table 1. At the lower sampling station (1.5 diameters), the velocity profile over the center 2/3 of the stack area is 27% in contrast with the maximum value of 20% under the ARM . Also, the *COV*s of tracer gas and particle concentration are in excess of the proposed maximum values. Over the entire stack area, the maximum value of tracer gas was 39% more than the average value, which exceeds the proposed range of ≤30%. Average swirl angle was 9° at the lower location. Clearly, as indicated by the *COV*s of the velocity profile, tracer gas, and 10.5 μm diameter particle data, mixing at the lower sampling station is inadequate. However, the swirl data suggest that sampling stations located further downstream would not be rejected by the swirl angle criterion because it is anticipated that in a straight stack, the swirl angle would only decrease with downstream distance.

The 9 diameter location meets all of the numerical mixing criteria of the ARMs. The *COV*s over the center 2/3 of the stack area for velocity, tracer gas, and 10 μm aerosol particles are 6%, 4.2% and 5% respectively, which all compare favorably with maximum *COV*s of 20% stipulated in the ARMs. The maximum concentration of tracer gas was 12% greater than the mean value, as compared with the maximum of 30% allowed under the ARMs. Average swirl angle was 6°.

It should be anticipated that if mixing is suitable at a given location in a straight stack, then the mixing should also be suitable at any subsequent location, provided that indeed there are no obstructions or changes in the internal geometry. Data for velocity, tracer gas concentration and aerosol particle concentration demonstrate the nine diameter location is suitable for single point sampling and the data summarized in Table 1 show that the mixing is even better at the 14 diameter location. Although the nine diameter location would be suitable, the 14 diameter location is to be used for sampling the WAF stack because it can be serviced from the roof of the building.

The EPA '8- and 2-criterion'[2] may have provided acceptable guidance for selection of a sampling site in this stack. A sampling station placed at the 8-diameter location would probably have tested satisfactorily because the 9-diameter location is suitable. However, as demonstrated by the work of Hampl et al.[7], the guidance of the '8-and 2-criterion' would not be satisfactory as judged by the ARMs criteria for many configurations of stack flow. In the case of the WAF stacks, where there is a lateral entry followed by a straight section, the large scale eddy mixing transfers both sufficient fluid momentum and contaminant mass across the stack to render the profiles acceptable within a 9-diameter (and probably an 8-diameter) distance.

A test was conducted to determine if flow Reynolds number would produce a significant change in the mixing. The velocity profile at the 14 diameter location was characterized for a mean velocity of 11 m/s as well as for the velocity condition of 24 m/s. The results for the entire flow cross sectional grid showed a *COV* of 16% for the low flow condition as compared with a *COV* of 12% for the high flow rate. We do not consider this to be a major effect (see the section on experimental errors), and conclude that the Reynolds number, as impacted by mean velocity, does not appear to have a substantial effect on the mixing.

Tests of sampling probes showed the 113 L/min (4 cfm) shrouded probe to have the best performance. At the high velocity condition, the transmission ratio for this probe was between 98% and 107% for particles sizes in the range of 1 to 20 μm AED. In contrast, the ANSI probe and the 'isokinetic' probes only showed only 20% and 63% transmission of 10 μm AED aerosol particles at the high velocity

condition. The Alternate Reference Methodologies includes a performance criterion for probes; namely, that the transmission ratio of acceptable probe should within the range of 80% to 130% over the range of anticipated operating conditions. Based on the results of the stack tests, both of the shrouded probes would satisfy this criterion; but, neither the ANSI probe nor the 'isokinetic' probe would be acceptable.

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DISCUSSION

ADAMS: How do you think your methods would have changed had there been a stack or a duct that had internal stiffeners every four feet?

MCFARLAND: I assume that the internal stiffeners would be angle iron or something of that order. In general, they would not change things very much, because the stiffeners would not, for the most part, block flow. I would go away from a stiffener to get out of its wake. If a stiffener was only a couple of inches across, acceptable sampling could be conducted a couple feet away. The stiffeners would do a little mixing too; not much, but a little.

MISHIMA: I noticed that the 7 mm diameter sharp edged single tube had the knee close to the end where you expanded to 9 mm. How much internal deposition does that knee cause?

MCFARLAND: I think that is a very astute observation. The probe was not designed exactly in accordance with the ANSI standard. The ANSI standard requires a constant internal diameter. The probe that we tested happened to be one that had already been fabricated at a DOE facility. It was just through serendipity that I happened to encounter it. The unit was gold plated. The question is, how much deposition would occur at the elbow compared to that if it had a constant internal bore? The answer is that this is an improvement over the ANSI probe. We have done some studies with these types of expansions, and there is not a great deal of deposition. On the other hand, the expansion causes a reduction in velocity, which lowers deposition in the elbow. As a consequence I believe this probe is a better probe than the standard ANSI device.

RADIONUCLIDE AIR EMISSIONS AT DEPARTMENT OF ENERGY FACILITIES

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Abstract

Facilities operated by the U. S. Department of Energy (DOE) handle and process radioactive materials in conjunction with their research, nuclear materials production, remediation and waste disposal activities. Radionuclide emissions to the atmosphere from DOE facilities are regulated by the Environmental Protection Agency (EPA) under 40 CFR Part 61, Subpart H for emissions other than radon. Subpart H requires DOE to monitor emissions from stacks and calculate a potential offsite dose to an individual using EPA approved methods and procedures. DOE has applied to EPA for approval to use alternative methods for some of the EPA requirements for continuous monitoring. The use of alternative methods such as single-point sampling with a shrouded probe will have an impact at several major DOE facilities. These facilities are identified.

I. Introduction

The Department of Energy (DOE) is responsible for implementing the nations energy strategy. DOE also is responsible for the development and maintenance of the nations nuclear weapons arsenal. DOE facilities handle and process radioactive materials in conjunction with their research, nuclear materials production, remediation and waste disposal activities. During normal operations, some of these facilities have the potential to release radionuclides to the environment. DOE site reports indicate that about 90% of the release is to the atmosphere. Radionuclide emissions to the atmosphere from DOE facilities are regulated by the Environmental Protection Agency (EPA) under the authority of Section 112 of the Clean Air Act (CAA). The EPA regulations are codified in 40 CFR Part 61, National Emission Standards for Hazardous Air Pollutants (NESHAPs). Subpart H of the NESHAPs applies to radionuclide emissions from DOE facilities other than radon.

DOE is in compliance with the dose standard of Subpart H. However, meeting all of the EPA requirements for continuous emission monitoring at major stacks has been a challenge to DOE, particularly at the older facilities. DOE is attempting to use alternate methods to meet some of these EPA requirements. To utilize the alternate methods provision of 40 CFR Part 61, DOE must obtain EPA approval for its use. The use of alternative methods such as single-point sampling with a shrouded probe will have an impact at several major DOE facilities and should facilitate compliance with the EPA regulations for continuous monitoring. A brief discussion of the DOE facilities that are subject to Subpart H of 40 CFR Part 61, the

types of radionuclide emissions, the emission levels, and the corresponding potential doses to the public will be given.

II. DOE Facility Emissions

In Figure 1., DOE facilities that have radionuclide inventory, and therefore, subject to the radionuclide NESHAPs, are located on the map of the United States with respect to their cognizant EPA region and state. | The names of the facilities corresponding to the site abbreviations shown are listed in Appendix A. There are about forty-five of these DOE sites that are spread out through ten EPA regions, and twenty-two states. Each EPA region is responsible for interpreting and implementing the regulations as they determine appropriate for the sites under their purview. EPA is beginning to delegate the authority for regulatintg these sites under the radionuclide NESHAPs to the states. DOE headquarters has the role of forging consistency and uniformity in the implementation of radionuclide NESHAPs across the DOE complex.

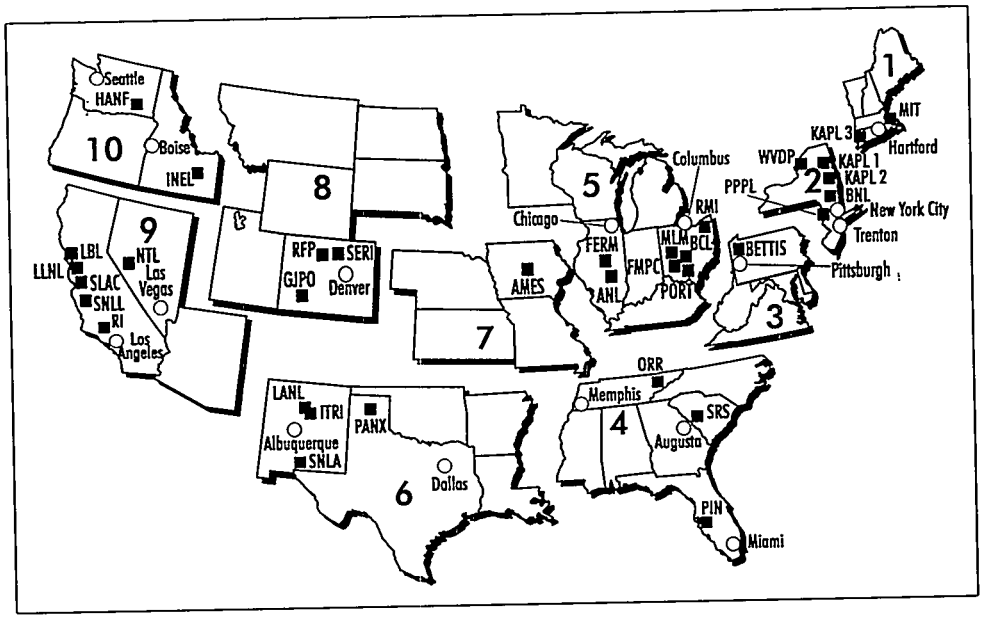


Figure 1. Department of Energy Site Locations and Environmental Protection Agency Regions

The types of radionuclide emissions from DOE facilities are illustrated in Figure 2. For calendar year 1992, most of the radioactivity released to the atmosphere was tritium. Tritium is a component of the emissions for over half of the DOE facilities reporting under NESHAPs. The category of other radionuclides, which mainly consists of air activation products produced by accelerators such as the Los Alamos Meson Physics Facility (LAMPF), was a major contributor to emissions. The nobel gas emissions are typically from nuclear reactor operations. The transuranic emissions, which were extremely low, are closely observed by DOE because this category represents a long term potential risk to the public at sufficient levels.

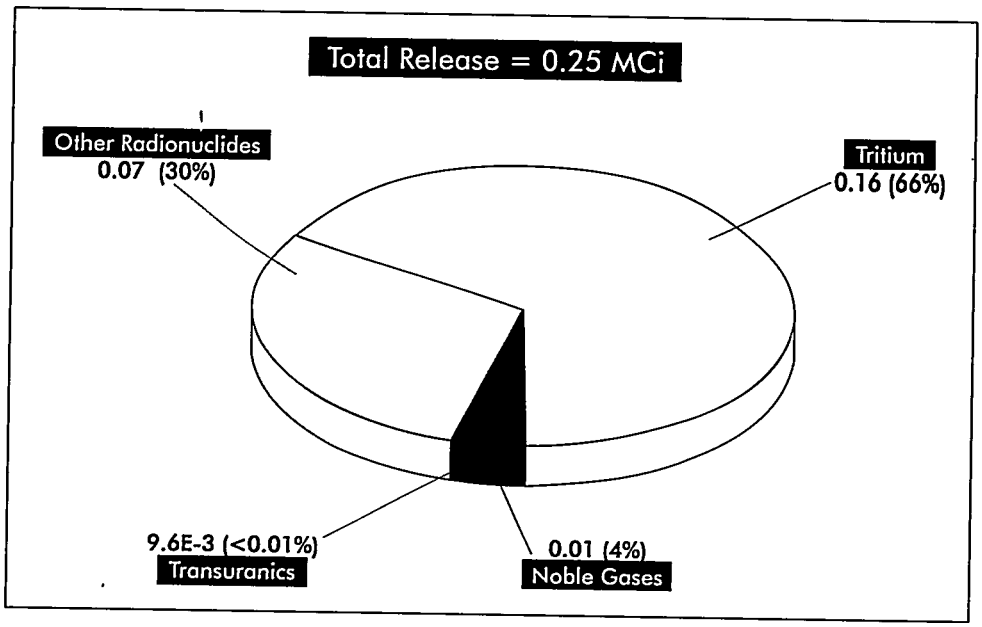


Figure 2. Composition of 1992 Airborne Releases from DOE Facilities (Megacuries)

Figure 3. indicates the level of emissions from DOE over a five year period. The levels of radioactivity released to the atmosphere have been reducing over this period and are expected to continue to decline as a result of a reduction in production activities DOE-wide.

III. DOE Facility Compliance

The radionuclide NESHAPs regulations in Subpart H of 40 CFR Part 61 have three basic tenets for which DOE must comply with. Each DOE site must (1) demonstrate compliance with the dose standard, (2) provide a report on June 30 of each year describing annual emissions, and (3) meet the prescriptive requirements for emission monitoring at its stacks. DOE sites have provided reports to EPA on an annual basis by the required date and are in compliance with the reporting requirement.

The EPA regulations identify specific methods and procedures for demonstrating compliance with the 10 mrem/year dose standard. The dose standard is defined at the receptor or individual who is most likely to receive the highest exposure from DOE operations. DOE emissions from its stacks are considered point sources of emission. The regulations require stack emissions to be monitored in accordance with EPA approved methods and procedures. The data from the stack emissions are used to assemble the source term that is input into computer air dispersion models, such as CAP88-PC. CAP88-PC models the dispersion of the radionuclides through the atmosphere and determines the fraction of the source term that reaches the location of the receptor where it is either inhaled or ingested. The radionuclide release fraction at the receptor is

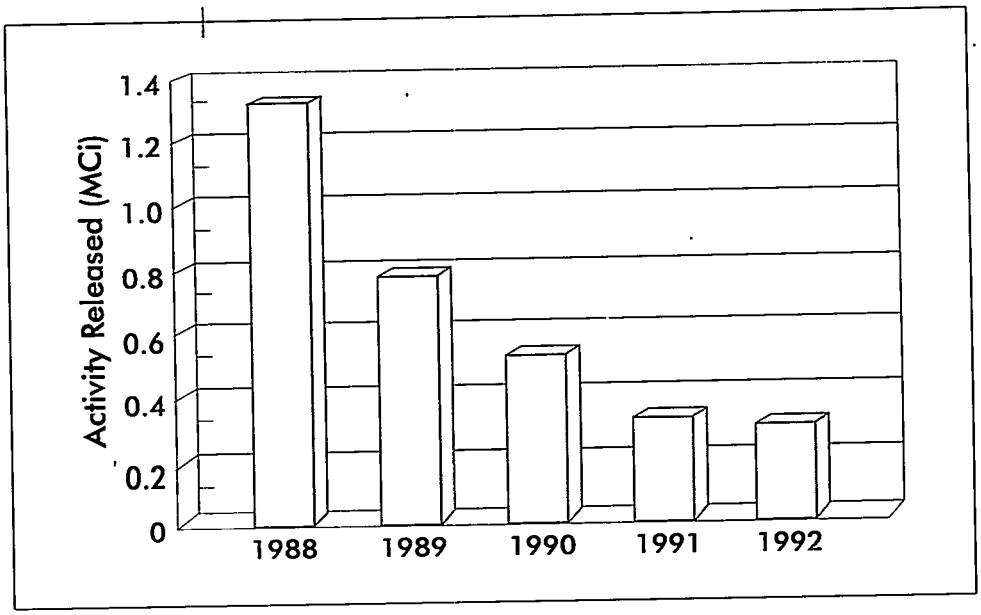


Figure 3. Total Airborne Radionuclide Releases from DOE Facilities

converted to dose and compared with the 10 mrem/year dose standard to ensure that the standard is not exceeded for that year. There are also contributions to the receptor dose from diffuse or non-point source emissions, however, EPA has not prescribed specific methods or procedures for assessing these emissions.

The overall status of DOE facility compliance with the 10 mrem/year dose standard for 1992 is illustrated in Figure 4. Los Alamos National Laboratory (LANL) is our highest emitter at 7.9 mrem/year with that dose resulting mainly from the operation of the LAMF accelerator. Next is the Oak Ridge Site which is at 1.4 mrem/year. There are a few other sites that are just above one percent of the standard such as Savannah River Site, Portsmouth Gaseous Diffusion Plant, and Brookhaven National Laboratory. However, in general, about eighty-five percent of the DOE sites are at least two orders of magnitude below the 10 mrem/year dose standard for their point source emissions. These results indicate that DOE is in compliance with the dose standard and continues to provide ample protection of the public from activities involving radionuclide release to the atmosphere.

Subpart H requires DOE to continuously monitor emissions from some of its stacks using EPA approved methods and procedures. These specific requirements were instituted when the rule was promulgated on December 15, 1989. The rule adopts the continuous monitoring guidelines established by the American National Standards Institute (ANSI) standard N13.1 which was finalized in 1969. Many of the monitoring systems in place at DOE facilities predate the considerations used to develop N13.1 and therefore these systems do not normally comply with the methods and procedures specified in the standard. It is these emission monitoring requirements for which DOE has had the most difficulty in meeting. DOE facilities are

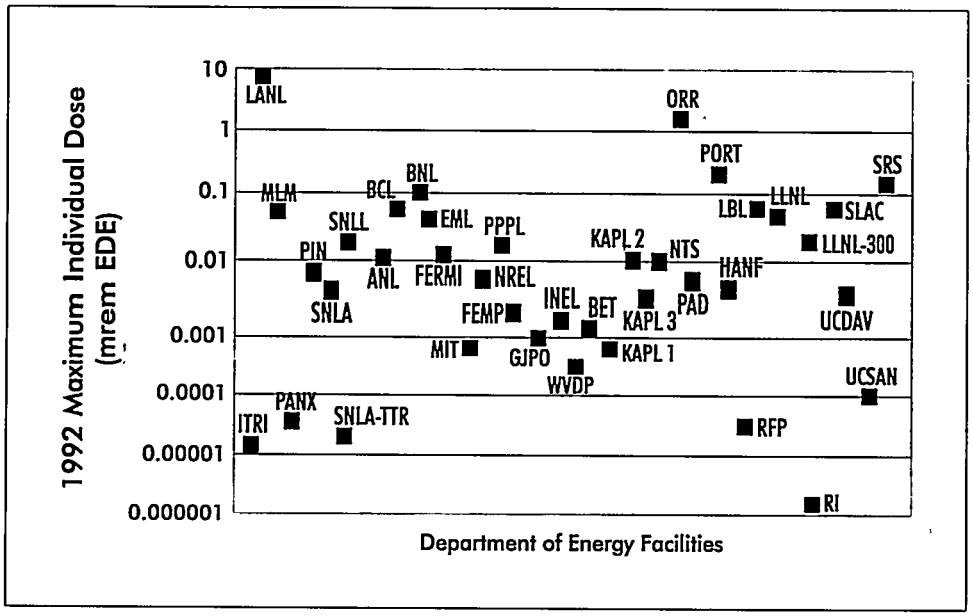


Figure 4. 1992 Annual Maximum Individual Dose Reported for Each DOE Facility

negotiating agreements with EPA on compliance plans and schedules for meeting the continuous monitoring requirements.

IV. Alternate Methods for Continuous Monitoring

The continuous emission monitoring requirements are imposed on those DOE stacks whose potential emissions exceed 1% of the dose standard, with no emission controls in place. These stacks are considered major point sources and compliance with emission monitoring requirements involves the implementation of EPA reference methods and procedures. The periodic confirmatory measurement (PCM) requirement applies to all other stacks. DOE interpretation of the rule implies that the Quality Assurance requirements of Method B114 apply to data collection at major point sources. The emission monitoring systems and plans must also be fully documented.

There are twelve DOE sites where continuous emission monitoring systems are required by 40 CFR Part 61, Subpart H. These sites are listed in Table 1 along with the cognizant EPA region. Some of these sites may require the use of alternate methods to demonstrate compliance with the continuous monitoring provisions. In many of these cases, use of single-point sampling with a shrouded probe is a viable alternative to EPA reference methods. Particularly, stacks that are not configured appropriately to meet the sample siting criteria are prime candidates for use of alternate methods. Furthermore, it has been demonstrated that single-point sampling with the shrouded probe provides more representative sampling and produces higher quality data. DOE sites may prefer to utilize the better technology in stack emissions measurement if that option is available. DOE has requested approval from EPA to utilize the

Table 1 DOE Facilities Subject to Continuous Monitoring

Facility	EPA Region
Brookhaven National Laboratory	2
Oak Ridge Site	4
Paducah Gaseous Diffusion Plant	4
Savannah River Site	4
Mound Site	5
Portsmouth Gaseous Diffusion Plant	5
Los Alamos National Laboratory	6
Rocky Flats Plant	8
Lawrence Berkeley Laboratory	9
Lawrence Livermore Laboratory	9
Hanford Site	10
Idaho National Engineering Laboratory	10

single-point sampling/shrouded probe approach DOE-wide. EPA is reviewing and evaluating the documentation submitted by DOE to determine if the approach is acceptable for alternate use on DOE major stacks.

Facilities that do not have at least one major stack are declared minor facilities and are required to meet only the PCM requirement. These facilities would not need EPA approval to use the single-point sampling/shrouded probe approach but could implement the newer technology based on the technologies own merits.

V. Conclusion

DOE has submitted a request to EPA for DOE-wide use of the single-point sampling/shrouded probe approach at major stacks that require continuous emission monitoring. EPA is evaluating this request and considering its approval. Twelve major DOE sites could benefit from the use of this approach in establishing compliance with the emission monitoring requirements. Other DOE sites could utilize the alternate approach because of the newer technologies' improved performance and higher data quality. In any case, EPA approval of the this alternate will enhance the widespread use of this measurement technology at DOE sites.

Appendix A

U.S. Department of Energy Sites by Operations Office and Location

DOE Operations Office	Site Abbreviation	Site Name	Site Location
U.S. DOE Production Sites and Research Facilities			
Albuquerque (AL)	ITRI	Inhalation Toxicology Research Institute	Albuquerque, NM
	KCP	Kansas City Plant, Allied-Signal Aerospace Co.	Kansas City, MO
	LANL	Los Alamos National Laboratory	Los Alamos, NM
	MLM	Mound Plant	Miamisburg, OH
	PANX	Pantex Plant	Amarillo, TX
	PIN	General Electric Neutron Devices Pinellas Plant	Largo, FL
	SNLA	Sandia National Laboratories, Albuquerque	Albuquerque, NM
	SNLA-TTR	Sandia National Laboratories, Albuquerque, Tonopah Test Range	Tonopah, NV
	SNLL	Sandia National Laboratories, Livermore	Livermore, CA
Chicago (CH)	AMES-1	Ames Laboratory, Iowa State University Site	Ames, IA
	AMES-2	Ames Laboratory, Alpha Containment Facility	Ames, IA
	ANL	Argonne National Laboratory	Argonne, IL
	BCL	Battelle Columbus Laboratory, West Jefferson Site	Columbus, OH
	BNL	Brookhaven National Laboratory	Upton, NY
	EML	Environmental Measurements Laboratory	New York, NY
	FERMI	Fermi National Accelerator Laboratory	Batavia, IL
	MIT	Massachusetts Institute of Technology Bates Linear Accelerator Center	Middleton, MA
	NREL	National Renewable Energy Laboratory	Golden, CO
	PPPL	Princeton Plasma Physics Laboratory	Princeton, NJ
Fernald (FN)	FEMP	Fernald Environmental Management Project	Fernald, OH
Idaho (ID)	GJPO	U.S. DOE Grand Junction Projects Office	Grand Junction, CO
	INEL	Idaho National Engineering Laboratory	Idaho Falls, ID
	WVDP	West Valley Demonstration Project	West Valley, NY
Naval Reactors (4) (NR)	BET	Bettis Atomic Power Laboratory, Bettis Site	West Mifflin, PA
	KAPL-1	Knolls Atomic Power Laboratory, Knolls Site	Schenectady, NY
	KAPL-2	Knolls Atomic Power Laboratory, Kesselring Site	West Milton, NY
	KAPL-3	Knolls Atomic Power Laboratory, Windsor Site	Windsor, CT
Nevada (NV)	NTS	Nevada Test Site	Mercury, NV
Oak Ridge (OR)	ORR	Oak Ridge Reservation	Oak Ridge, TN
	PAD	Paducah Gaseous Diffusion Plant	Paducah, KY
	PORT	Portsmouth Gaseous Diffusion Plant	Piketon, OH
	RMI	RMI Titanium Company Extrusion Plant	Ashtabula, OH
Richland (RL)	HANF	Hanford Site	Richland, WA
Rocky Flats (RFO)	RFP	Rocky Flats Plant	Golden, CO
San Francisco (SF)	LBL	Lawrence Berkeley Laboratory	Berkeley, CA
	LLNL	Lawrence Livermore National Laboratory	Livermore, CA
	LLNL Site 300	Lawrence Livermore Explosive Test Site	Livermore, CA
	RI	Rockwell International, Santa Susana Field Lab.	Canoga Park, CA
	SLAC	Stanford Linear Accelerator Center	Stanford, CA
	UCDAV	Laboratory for Energy-Related Health Research, University of California, Davis	Davis, CA
	UCSAN	Laboratory of Radiobiology and Environmental Health, University of California, San Francisco	San Francisco, CA
Savannah River (SR)	SRS	Savannah River Site	Aiken, SC

DISCUSSION

ANON: Since the standards are basically set for dose at the property line, are there any of the DOE facilities in which direct measurements at the property line can be used in lieu of stack sampling?

DUVALL: Yes there are and I can name a couple, Fernald and Pantex. This is because Fernald and Pantex do not have any major stacks, they have to measure. Most of their emissions are from diffuse, or non-point, sources. Since there are no stacks, there are no EPA required monitoring procedures. EPA has not specified how to assess emissions from diffuse sources. Ambient monitoring at the boundary is applicable to those sites but you have to request EPA approval. I think we are in the process of doing that for a couple of sites.

HULL: You mentioned that tritium was a major radionuclide in emissions. It ought to be relatively easy to monitor tritium in the field, particularly if it is in the form of HTO. A long sampling period would permit collection of a measurable sample that could be translated into a low integrated dose. Do you have any comment on that?

DUVALL: I do not know much about tritium monitoring. There is quite a bit of tritium emitted from DOE activities, but it represents a very low dose to the receptors. I have not heard of any problems from the sites in measuring tritium. Some sites have elaborate controls in place to reduce emissions, but tritium, in general, does not seem to be a problem except in modeling where they utilize a worst case situation, such as considering tritium as HTO, instead of HT. This overestimates emissions and needs to be corrected. I do not think it is a real problem.

EPA PERSPECTIVE ON RADIONUCLIDE AEROSOL SAMPLING

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Abstract

The Environmental Protection Agency (EPA) is concerned with radionuclide aerosol sampling primarily at Department of Energy (DOE) facilities in order to insure compliance with national air emission standards, known as NESHAPs. Sampling procedures are specified in "National Emission Standards for Emissions of Radionuclides other than Radon from Department of Energy Sites" (Subpart H). Subpart H also allows alternate procedures to be used if they meet certain requirements. This paper discusses some of the mission differences between EPA and DOE and how these differences are reflected in decisions that are made. It then describes how the EPA develops standards, considers alternate sampling procedures, and lists suggestions to speed up the review and acceptance process for alternate procedures. The paper concludes with a discussion of the process for delegation of Radionuclide NESHAPs responsibilities to the States, and responsibilities that would be retained by EPA.

The Difference in Perspective between EPA and DOE

Most of the EPA experience in radionuclide aerosol sampling involves facilities owned or operated by DOE. Those sites are subject to the requirements of 40 CFR 61, Subpart H, National Emission Standards for Emissions of Radionuclides other than Radon from Department of Energy Sites (Subpart H). The facilities typically are, or have been involved with weapons production or research. They are mostly large facilities, with some monitored stacks. The facilities may have some unmonitored stacks, and may have radionuclide emissions from fugitive or diffuse sources, such as contaminated soil. DOE reported radionuclide emissions from 38 facilities in 1992.

The DOE Approach

In order to characterize the EPA perspective on radionuclide aerosol sampling, it is necessary to start by characterizing the differences between the roles of EPA and DOE. The two organizations serve different roles in the Federal Government. The DOE has been production or research oriented, often working under severe time and security constraints. "Sufficient documentation" of a project was enough to show a small audience of supervisors and peers, educated in the science, that it was reasonable to proceed. Projects used sound logic, scientific laws and principles, and in some cases, educated assumptions. "Success" could be measured by verifiable experimental results or a functioning piece of hardware or software.

The EPA Mission

The EPA was formed on December 2, 1970. Less than a year before that, during the signing of the National Environmental Policy Act, President Nixon stated that he had "become further convinced that the 1970's absolutely must be the years when America pays its debt to the past by reclaiming the purity of its air, its waters, and our living environment. It is literally now or never." EPA's mission statement includes the commitment to ensuring that "National efforts to reduce environmental risks are based on the best available scientific information communicated clearly to the public." Note the need to communicate *scientific* information to the *public*, not to peers in the scientific community. Further, EPA is charged with protecting the public, as exemplified by the standard for the DOE facilities: "Emissions of radionuclides to the ambient air from Department of Energy facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 mrem/yr." (40 CFR 61.92).

"Members of the public" tend to be a diverse group whose interest in a problem varies, depending on the problem. The decisions made by EPA have an effect on both industry and members of the public. It is not uncommon for an EPA rule to be criticized by one group as being too stringent and by another as being too lenient.

EPA Standards Development

The development process includes research to determine the need for standards, selection of appropriate standards, and identification or development of the tests necessary to validate that the standards have been met. Since point source (stack or vent) emissions are the most significant at most DOE facilities, the emission monitoring and test procedures concentrate on point sources. The standard includes EPA Reference Method 1 (where to take a sample); EPA Reference Method 2 and 2a (velocity and volumetric flow rates); and inclusion by reference of ANSI N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," (how to get a good sample). Analysis methods, and quality control and quality assurance procedures are included as well. The procedures described above are used to determine emissions which are then used as input to computer models that calculate dose.

Once a standard is in place, EPA provides training for the technical staff and inspectors (who may be part of the technical staff) who will be dealing with the standard. For Subpart H, the training includes a review of the standard, the procedures for determining compliance and the computer codes used for calculating the dose. During the same session, the instructors present a class on sampling and analysis, which includes a detailed review of the test methods and the ANSI standard. The Subpart H training has been presented four times at four locations across the country. Current plans call for videotaping the course for future availability.

Roles of EPA Headquarters and Regions

EPA headquarters staff is responsible for developing regulations,

such as Subpart H, as described above. Once the regulations are in place, they are responsible for providing technical training, guidance on the uniform application of the regulation, and support to the regions on issues of national significance. The headquarters staff would also be responsible for any changes that subsequently might be made to the regulations.

Staff from the EPA Regions may participate in the regulation development process, depending on the skills available. Regional staff are the primary contact with the DOE facilities, and are responsible for inspections and determining facility compliance with the regulations. They work with the facilities to develop Federal Facility Compliance Agreements or other agreements as needed. The regions may call on support from the EPA laboratories, headquarters, or specialized expertise from EPA contractors for inspections. They also have authority to approve applications for construction and some alternate procedures for sampling as described below.

Alternate Procedures to Demonstrate Compliance

Because new technologies are constantly emerging, many EPA standards include a provision for approving alternate procedures for determining compliance. If a facility does not employ the procedures that are prescribed in the standard, the facility may request that EPA consider an alternate. Specific requirements are detailed in the standards, but generally are these: 1) the requirements in the original standard are impractical; 2) the alternate procedure will not significantly underestimate the emissions; 3) the alternate procedure is fully documented; and 4) the facility receives prior approval before using the alternate. An alternate procedure will typically include caveats on its use.

There are two possible review paths for an alternate procedure. If the request will apply to only one facility (or to the facilities in one region) the EPA region can approve the procedure. Frequently the EPA region discusses the application with headquarters before this decision is made. Generally, the approval will apply to only those facilities for which the original request was made. In some cases, there is a need or desire for the procedure to apply to a large number of facilities. In this case, approval would be a joint effort between EPA headquarters and the regions. Determining the level of review and the applicability of the alternate procedure calls for a cooperative effort between headquarters and regional staff.

Problems with alternate procedures

There has been some concern expressed about the length of time necessary for the review and approval of alternate procedures. EPA shares that concern, and has reviewed several requests to determine the cause of delays.

The review looked at the comparison of the requests with the requirements for approval. The first requirement is that the procedures in the standard for measuring the effluent flow rate, or monitoring or sampling the effluent stream are impractical. Although "impractical" is subject to some interpretation, EPA has been willing

to consider what would constitute sufficient evidence that a procedure was impractical.

The second requirement is that the alternate procedure does not underestimate the emissions. Proving that the new procedure does not underestimate emissions can be difficult. In order to show that the new procedure does not underestimate emissions, the procedure must be developed and tested. Simply comparing the new procedure with the standard procedure may not be possible, since site-specific conditions may affect the results. and the standard procedure is not used on the application, or the alternate would not be needed. Therefore, laboratory testing, calculations, and the use of scientific principles may be needed to provide enough information to convince EPA and *the public* (if necessary) that the procedure is indeed acceptable.

The third requirement is that the alternate procedure be fully (and properly) documented. This ties in somewhat with the second requirement, and is the one most frequently overlooked. One suggestion for a test of completeness is for someone familiar with the science but not with the application to be able to critically review the package and determine that the documentation is complete.

An example serves to illustrate the point. One request included a simple line drawing which were assumed to be a building and a stack. The drawing was not accurate from a mechanical drafting point of view, and had a notation at the bottom, "Flange Location for Stack Monitor". Only one dimension appeared on the drawing, and the only other notation was "Flange size and distance between to be determined." (The one dimension in the drawing was in conflict with the corresponding one in the text.) There was no identification of the facility on the drawing, no date, no location; in short, nothing that would lead the reader to any conclusion that the drawing was intended to be part of the submittal. The text was only slightly more informative. One statement from the request was "Ports are sufficiently downstream of the last major flow obstruction to provide a representative sample."

The fourth requirement, that the facility have approval for the procedure before it is used, would appear to be self-explanatory. However, it is worth noting that a positive approval is needed, not just a lack of a rejection.

Speeding up approval

What can be done to speed the approval process? The same things that lead to successes in other matters work here, too. Facilities that have had successes note that they work closely with, and have good working relationships with the personnel in their respective EPA regional offices. Good communication will alert EPA that a request is coming, and help to confirm that the level of detail supplied is sufficient for the review that is needed.

Internally, the facility should review the four criteria for the approval of an alternate procedure. The request and supporting doucments should be simple, clear and direct. The request should be complete enough to make the point without being verbose. Including

everything that has been ever written on the subject will only slow up the review. Data tables and figures should be clearly marked and referenced to the text. The bounds of applicability (ie, flowrates, temperatures, duct sizes, fluid viscosities, etc) should be clearly stated.

The facility should select an internal reviewer who has some knowledge of the science (to insure accuracy) but is not associated with the project (to insure an unbiased review.) The reviewer should put himself in the place of an EPA reviewer who could be subject to criticism by any member of the public for approving an alternate proposal that is not properly documented.

Incorporation of a Revised ANSI N13.1

The ANSI Standard, N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities" is presently being rewritten, and will likely contain significant changes from the version now in effect. Subpart H contains the current version of the ANSI standard. When the new version is approved, EPA will review it to determine its applicability to Subpart H and whether the current version should be replaced. Note that simply replacing the old version with the new version could lead to some problems for facilities. Sampling systems based on the old standard might not be in conformance with the new standard, and could lead to the need for retest or rework of the sampling systems. Thus EPA will have to determine if both versions should be accepted for a period of time. In the interim, a facility can propose to use the ANSI procedures as alternate procedures under Subpart H.

DOE "Shrouded Probe" Request

DOE has submitted a request for approval of a set of alternative procedures that have become known as the "Shrouded probe package". The package was submitted by DOE headquarters to EPA headquarters on behalf of all DOE facilities. DOE requested a generic approval that would be applicable in any DOE facility. As a result, EPA headquarters served as a focal point for the review, combining the responses from personnel in all EPA regions.

The individual procedures are 1) to use single-point sampling instead of multiple probe inlets; 2) use of a shrouded probe instead of an ANSI design probe; 3) use of test data to determine the proper placement of a probe, instead of the "8 and 2 rule"; and 4) use of the computer code DEPOSITION to determine particle losses in the sample transport lines, rather than the ANSI guidelines.

The four proposals provide performance-based criteria for approval. However, the first submission by DOE included a generic request, without information on the limitations of the procedures, or application guidelines. Although it appeared that the proposals would provide good data, there was not sufficient information to help the EPA reviewer to decide if this probe in this application would behave the same as the one in the request, even though the dimensions, flow rate, or other parameters might be different. A detailed request for

additional information was forwarded to DOE, and a response has just been received.

State Delegation of Authority

One of the priorities of the Administrator of EPA is to develop partnerships with state and local governments. To that end, EPA actively seeks to delegate to the states the authority to enforce the NESHAPs program. EPA is working with interested states to provide technical support, financial support in the form of grants, and technical training. To date, five states have received grants, and over 60 students from 12 states have participated in Subpart H training.

Delegation is important to DOE, since it changes the regulatory organization. To delegate its authority, EPA must make a finding that the State's procedures for implementing and enforcing the radionuclide NESHAPs are adequate. The EPA Regional Administrator must insure that several elements are addressed prior to delegating EPA's authority to the state. These elements include:

1) insuring that emission limits and test methods are consistent with federal regulations;
2) the NESHAPs reporting and monitoring requirements are implemented;
3) the state agency has adequate enforcement provisions;
4) procedures for public notification and disclosure of non-confidential source information are in place; and
5) the state has sufficient resources to implement the program.

After delegation of NESHAPs authority to a state, EPA retains concurrent authority to enforce the radionuclide NESHAPs in federal court where a State is unwilling or unable to pursue legal action in its own State court system. EPA will only exercise its concurrent authority when necessary to secure effective enforcement of radionuclide NESHAPs.

Responsibilities Not Delegated

Generally, there are two categories of responsibilities that will not be delegated to the States: any decision that requires a rulemaking to implement; or any decision where Federal oversight is needed to insure National consistency. For example, the EPA will retain the authority to approve an alternative method which effectively replaces a reference method, in order to insure uniformity and technical quality in the test methods used for enforcement. Or, approval of an equivalent means of emission limitation to any design, equipment, work practice, or operational standard must be adopted as a change to the individual subpart, and therefore can not be delegated to the States. These decisions would be proposed and subsequently promulgated in the Federal Register and will become part of 40 CFR Part 61. At that point, the methods are available for general use. These authorities will be retained by the Director, Office of Radiation and Indoor Air.

Conclusion

EPA has in place in its regulations standards for sampling radionuclides in air streams, primarily at DOE facilities. These standards are based on the state of the art that was available at the time the regulations were promulgated. As the state of the art advances and improves, EPA attempts to make use of the procedures in the regulations to allow the use of the improved procedures. In order to insure that the mission of protecting the public is properly carried out, EPA must insist on scientifically sound technical backup for the alternate procedures, and that the procedures provide at least the same level of protection as the procedures they replace. And finally, as states assume more responsibility for the regulations, EPA must insure that the states have adequate technical support to be able to discharge these duties properly.

References

(1) USEPA 1983. "Good Practices Manual for Delegation of NSPS and NESHAPs." U.S. Environmental Protection Agency, Office of Air Quality Planning and Standards, February 1983.

(2) USEPA 1989. National Emission Standards for Hazardous Air Pollutants: Radionuclides; Final Rule and Notice of Reconsideration. 40 CFR Part 61, December 15, 1989.

(3) Colli, et.al. "State Delegation of National Emission Standards for Hazardous Air Pollutants - Radionuclides". U.S. EPA Office of Radiation Programs.

(4) Clean Air Act 1990; 42 U.S.C.A. 7401 et.seq.

(5) USEPA 1991. Radionuclide NESHAP Grant Guidance. U.S. Environmental Protection Agency, Office of Radiation Programs.

(6) USEPA 1991. Requirements for Approval of State Requests for Delegation of Authority.

(7) Title 40, Code of Federal Regulations, Part 60, Appendix A, Test Methods

DISCUSSION

PALMER: I think Ohio has been one of the states designated to enforce NESHAPS. Will they be the people to whom we will send an alternative sampling request if we want to use alternative means for both near and long term? Or would it be a region office?

KARHNAK: There will be a time period during which the transition will take place. Ohio does not have delegation yet. When authority is eventually turned over to the state, it will include procedures to maintain national consistency. No state has taken delegation for NESHAPS yet and there is going to be a transition period during which we are going to have to work together. Once a state has complete delegation, it will be the one to get the application, but I do not want to commit EPA to something where there may be overriding circumstances. I think that it will go to the state, but EPA will have oversight to make sure there is not a national overriding concern that we should be involved with. There is not going to be a confrontation between EPA and the states, we are going to be working together, I anticipate, for five years.

OSBORNE: I have two questions. First, a lot of DOE facilities have spent millions of dollars to upgrade their monitoring equipment to meet the current requirements. After the ANSI standard is revised, will the facilities be grandfathered if their existing systems meet the old standard, or will they have to be upgraded again to meet the new ANSI standard? The way I see it, it makes more sense to grandfather them and have new sources meet the new standard. My second question is, some of the states, particularly Colorado, do not want to take over the NESHAPS program. This will cause problems for us with a Title 5 operating permit, e.g., who is going to issue the operating permit, the state or EPA?

KARHNAK: You gave me one very easy one and one very tough one. The first is rather easy but it is a very good question and I am glad you raised it. The question deals with whether or not the current ANSI standard will be grandfathered for facilities that have upgraded to meet it. We have discussed it ourselves, because it does not make sense to require everyone to make a change again. Let me talk just a moment about the procedure that is being proposed as an alternative to the ANSI standard specified in the rule. It would permit both to coexist. In order for the rule to be changed, and an updated ANSI N13.1 to be incorporated, we would have to go through a formal rule making process that includes publication in the Federal Register. We probably would not have to have an open hearing, because it is an administrative-type change rather than a change of dose level, which would be more significant. Until there is a rule change, both will coexist. After a rule change, we will have to consider that some facilities, that may have just been upgraded under the old rule, will not have to upgrade for some period of time. I am not sure that it makes sense for us to require changes, simply because we have incorporated the new standard in the rule, even though it contains improvements. My answer is that I expect both will be allowed for at least some period of time.

The second question deals with the states, what happens if a state does not take delegation, and how that will apply to Title 5. I do not have time this morning to answer that one, that is how big a question it is. There is a separate group working on the Title 5 issue and state delegation. I would like to talk to you about it, but it is too much to

get into now. There are some states that do not want the delegation, but there is a requirement that they take the delegation if they take any part of NESHAPS. Some of the states are balking at that because they do not want to have to take on the radiation NESHAPS, and I am not sure how that is all going to play out.

HULL: I assume that if a state is an NRC agreement state, they automatically have responsibility for NESHAPS, too? It sort of seems to go hand in hand.

KARHNAK: You would think that would be the case, wouldn't you. Let me try to explain, because it is an interesting subject. The 1990 Clean Air Act Amendment said that if EPA found the NRC program was sufficiently protective of public health, we could withdraw our rule and NRC could go forward. Legally, we could not make that finding and, as a result, Sub-part I went into effect. There are NRC agreement states, but they inspect according to the ALARA requirements, not to the 10 mrem limit and not to the COMPLY computer code. As a result, there is a difference. They are not far apart, but the two are different. To state it diplomatically, and politically correctly, we have been negotiating with NRC over some period of time to try to get them to agree as closely as possible.

HULL: Does that mean war?

KARHNAK: I did not say that. There is a difference of philosophy between the way the two organizations operate, and we have been trying to bridge the gaps to get together.

PROGRESS OF REVISION TO ANSI N13.1-1969

GUIDE TO SAMPLING AIRBORNE RADIOACTIVE MATERIALS IN NUCLEAR FACILITIES

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Abstract

The ANSI N13.1-1969 *Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities* is currently being revised. The revision is being drafted by a working group under the auspices of the Health Physics Society Standards Committee. The main differences between the original standard and the proposed revision are a narrowed scope, a greater emphasis on the design process, and the verification of meeting performance criteria. Compliance with the revised standard will present new challenges, especially in the area of performance validation. The progress made in the revision and key portions of the standard are discussed. The DOE has recently petitioned EPA for alternate approaches to complying with air-sampling regulations. Dealing with compliance issues until the revised standard is adopted will be a challenge for both designers and regulators.

I. Introduction

The objective of this paper is to briefly describe the content of the proposed revision, to point out significant differences from the old standard, and to describe new challenges that the proposed revision will present.

II. Change in Scope of the Standard

The original version of ANSI N13.1-1969, *Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities*, covered air sampling in the workplace, in the environment, and in stacks and ducts. The scope of the revision has been narrowed to air sampling in stacks and ducts; however, some supplementary material may be provided on sampling from calm air. This change was made to allow more coverage of the latest developments in this technology. Workplace air sampling will now be covered under Planco-57 (*Procedures and Instrumentation for Characterizing Airborne Radioactivity in the Workplace*, draft) and environmental air sampling will be covered in ANSI N13.9 (*Guide to Environmental Surveillance Around Nuclear Facilities*, draft). The proposed title for ANSI N13.1-199X is *Guide To Sampling Airborne Radioactive Materials in Stacks and Ducts.*

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III. Objectives and Approaches for Sampling Programs

The first major section of the revision covers the factors for selecting the objectives and design approaches for a sampling problem. This section emphasizes defining the objectives for a given sampling situation. Failure to understand the sampling objectives can lead to inappropriate or ineffective system design and implementation. The typical objective is to monitor the performance of air treatment systems that ensure that people in the surrounding environment are not exposed to levels of airborne materials deleterious to their health. Some examples of other objectives follow:

- To assess the need for a permanent sampling or monitoring program
- To call attention to deteriorating equipment, faulty processes, or other conditions leading to the loss of effective control of airborne materials in an operation, and to subsequently determine the effectiveness of corrective measures
- To help assess the possible consequences of non-routine incidents and to help in the selection of appropriate corrective action. This can include the integration of radioactive contamination released to the environment over various time periods.

The proposed revision describes a methodology for meeting the most stringent regulatory requirements; however, these requirements may not be universally applicable to all sampling programs, especially those with such limited objectives as process control. Therefore, when designing systems with objectives other than regulatory compliance, the designer must exercise judgment in the application of these requirements and explicitly document the sampling objectives and the reasons for any exceptions to the requirements of the standard.

After an agreement is reached on the purpose of a given sampling effort, the technical approach for meeting the objective must be formulated. Three factors determine the design approach to the sampling problem: 1) the potential emissions from the source, 2) the types of contaminants to be sampled, and 3) tne desired detection limits. For the first factor, the revision suggests a possible graded approach based on potential emissions that could be used to determine some basic sampling program capabilities as shown in Table 1. Guidelines are given for estimating potential emissions where actual measurements are not available.

The second factor determining the sampling approach is the type of contaminants being sampled. The differences in sampling for particles, gases, and vapors are discussed at a general level to draw attention to how they affect the choice of sampling methods. Particular problems associated with large particles, reactive gases, and off-normal conditions are discussed.

The third factor determining the sampling approach is the desired detection limits or action levels. Measurement uncertainties play a key role in determining practical action levels. The sources of uncertainty typical of air sampling and how they propagate are discussed. Guidance is then given to rank the sources of uncertainty and to control the most highly variable ones. Using this information, the designer can then set achievable action levels.

When going through this process, the designer also sets performance criteria for various measurement uncertainties specific for the particular sampling problem.

Table 1 Sample Graded Approach to Application of Sampling and Monitoring

Potential Effective Dose Equivalent Category	Required Monitoring and Sample Analysis Procedures	Potential Effective Dose Equivalent Range (mrem/y)
1	Continuous extractive sampling for a record of emissions, and real-time monitoring with alarm capability; consideration of separate accident monitoring system	>5
2	Continuous extractive sampling for record of emissions, with retrospective, off-line periodic analysis	>0.1 and ≤5
3	Periodic confirmatory extractive sampling and off-line analysis	>0.001 and ≤0.1
4	Annual administrative review of facility uses to confirm absence of radioactive materials in forms and quantities not conforming to prescribed specifications and limits	≤0.001

IV. Determining Representative Sampling Locations

Choosing the location from which to extract samples that are representative of the contaminant concentration in the airstream is the next topic covered in the proposed revision. This section includes characterizing the airstream, including composition, geometry, velocity mapping, and contaminant mapping. The original standard's requirement to locate the sampling plane at a certain distance from a flow disturbance or discharge point was never proven to be an effective means of ensuring sample extraction from a well-mixed system, and has been shown in instances to be ineffective. Also, the requirement for multi-point probes was a means to compensate for poor mixing by obtaining a mixed sample from several locations in the cross-section. The result was often poor sample penetration through complex probe designs and small extraction inlets. The proposed default criterion to ensure that samples are representative of the airstream is to demonstrate that the coefficient of variation of the contaminant is within +/- 15% over the cross section of the sampling plane. This demonstration is one of the major technical challenges of the revision, but the result will be simpler and more effective probe designs. Guidelines are given for measuring the contaminant distribution in the sampling plane.

V. Design of Effective Sampling Systems

The next major section covers, in general terms, the hardware aspects of designing effective sample systems. The requirements for bulk stream flow measurements are first discussed. Guidance is given concerning when to use continuous or periodic flow measurements. The methods appropriate for both of these modes are briefly described and literature references are given to detailed procedures.

Probe inlet designs are treated next. The relative merits of single inlet and multiple inlet probes are discussed. The proposed default requirement for demonstrated particle penetration through record or compliance sampling probes is given as 80 - 130% for particles of 10 micrometers aerodynamic equivalent diameter (AED). Slightly lesser requirements are proposed for systems used for alarm or trending purposes. Guidelines for probe materials, construction, and maintenance are provided. Performance demonstration methods are also given.

Other topics covered in this section are briefly described below.

- The minimization of sample line-loss for particles and reactive gases is discussed.
- Current data and references are given on selecting filters and gas collection devices.
- Guidelines are given on sample flow rate measurements and the pitfalls to avoid when using various instruments. Maintenance and calibration of flow instrumentation are discussed.
- The sampling system components that should have alarms and warning devices to indicate losses of performance and function are addressed.
- The optimization and upgrading of existing and new systems are discussed. Bringing existing systems up to new standards will be a major challenge, and each emission point's dose potential and the contaminant characteristics will be factors to consider in upgrade decisions. In analyzing the priority of an existing system's upgrade, the objective and approach issues described above must be re-evaluated. The use of the air-stream mixing and characterization methods and other performance testing methods are tools in determining whether upgrades will be necessary. The application of decision-aiding techniques in system upgrades is described.

VI. Quality Control and Quality Assurance

The final main section of the proposed revision addresses quality control and assurance issues. The first subject covered is the documentation of the system design. The design files will contain the rationale used in selecting objectives and approaches. Effluent stream characterization studies, detailed design drawings, vendor information, and operating procedures will also be maintained in the file. This section gives guidelines on the system components requiring regular inspections and calibrations. Inspection points will include probe position, corrosion, wear, leak

tightness, and the functioning of flow meters, pumps, controls, heat tracing, detectors, and other instruments. Operator training records will also be maintained. Sample handling will also be addressed.

VII. Appendices

The proposed revision was designed to present the general guidelines and requirements in the main body. To allow for more convenient standard updating, the appendices contain information on important methods for which there is a likelihood of continued technical developments. These include

- Air-stream flow measurements
- Techniques for characterizing sampling sites
- Current techniques for particle and iodine line-loss estimation
- Probe designs
- Methods for performance testing or sample validation.

VIII. Conclusion

The emphasis of the proposed revision is more on performance verification and less on look-alike design. The structure of the original standard made it easier to use as a source of a default design than to embrace the nuggets of performance-based design that it contained. The simplicity of the design concepts it contained were easy subjects for derivative guides and training materials that did not present information that the user needs to consider in the design phase and to verify adequate system performance. The requirements to demonstrate that the contaminants are well mixed in the sampling plane and that the system extracts and delivers a representative sample will be the major challenges of the proposed revision to system designers and users. The impact of the proposed revision on some existing systems may be significant, but careful consideration of the system objectives and the effluent dose impact will focus the upgrade efforts where they are most needed.

PANEL DISCUSSION

DAVIS: A general comment, we are in a Catch-22, so I would like to emphasize again the need for speed. We are spending multi-millions of dollars upgrading systems and, from what we are hearing, the upgrades we are making to our systems may become "junk". I would think that the EPA would be quite concerned about the fact that some of these probes we are putting in may have line losses, or probe deposition, upwards of 80%.

ANON: Your point is well taken. However, I would like to make the observation that the probe is only one portion of the sampling train and, in fact, there is a requirement in the ANSI standard, as it now exists, to look for deposition in the system. It has not gotten to the detail of using a deposition computer program to help plan how you put your lines in, but based on what you have learned, we certainly would not object to your putting in 1-in. sample lines instead of 0.5-in. lines, to make sure that what gets past the probe gets all the way to the collection point. There are things you can do without asking DOE to make changes or accept alternative procedures. On the other hand, we are moving ahead. I said that I do not think we have moved as fast as I would have liked to have moved, but, nonetheless, I think we are fairly close to closure.

NEWTON: As a prudent steward of the public money, I think that the least one should do is to say I am somewhat committed to putting in the ANSI probes. If I go ahead, in a few weeks' time you will probably okay a single point shrouded probe if you approve the supporting data. But if you run the deposition code, you can make your choice on which way to go. That is what I plan on doing for the people that I am advising.

GLADDEN: We do an awful lot of air sampling in our testing programs, etc. Would you advocate using the shrouded probe for sampling other things than radionuclides? Is there a commercial source for this new probe?

MCFARLAND: Let me address that, if I may. The answer is, I think the concept is generally applicable to situations that involve aerosols of larger sizes. When the particles are 0.3μm, or when it is a gas, it really does not matter what you sample with. For the nuclear industry, where 10μm particles could be present under accident conditions, such a system as has been proposed has some benefits. For sampling in other kinds of stacks that the EPA regulates, for example cement plants, a shrouded probe appears to offer benefits. I think it would also offer benefits in situations such as you have at Dugway, where you are interested in particles in the size range of 1-10μm. The probe appears to be able to tolerate variations in wind speed while sampling at a fixed flow rate, which would be appropriate for the type of sampling that you do. Graseby-Anderson of Atlanta has the license for manufacturing the probe.

ANON: I think it is important that the shrouded probe has brought a lot of visibility and a lot of attention to some of the other conditions that go along with sampling. A shrouded probe in a flow that is not well developed and not properly mixed is still not going to give you a representative sample. But I think the attention it has brought to these other points has helped to improve the science and process of sample taking.

ANON: In general terms, when you look at stack sampling, the real key is to be able to

sample representatively at a single point.

MISHIMA: I was the chairman for the Sub-part H, and we were sort of inclined to accept those things that were approved. But even then, we realized that there were very serious problems with ANSI 13.1. With respect to the gentleman who asked about being grandfathered in, that is a worthwhile effort, but I think we should bear in mind that ANSI 13.1 was not well based technically for such elements as the deposition code, as well as for others. When you need to upgrade, or when you have problems, a new code is the way to go. Because it is better based technically, it seems like a significant improvement over what we had available in the past to look at the nature of the pollutants, how to get the best sample, and how best to transport it. And as you pointed out, it extends further back into the system.

OSBORNE: A simple question for Mr. Glissmeyer. You showed a graded approach in your presentation for determining the monitoring required based on potential dose. Was that a controlled, or uncontrolled, potential dose?

GLISSMEYER: Uncontrolled.

MCFARLAND: I hope there will be a good correlation between the requirements of EPA sub-part H and the ANSI standard, so they won't be mutually contradictory.

KARHNAK: I think the goal of the committee was to make it easy to adhere to the rule. We wanted to make it simple, we wanted to have a little more guidance on when to make choices. When you see the final rule, it is going to be a lot thicker than it was, but there will be a lot more information. You won't be forced to accept rules of thumb without a technical basis. I think that the final result will be that we will have better, more reliable sampling procedures. We do not think it is necessary to convert all existing ANSI-style probes to the new design until you are faced with an upgrade, or a noncompliance situation. If you have a major source it is going to have to be negotiated, but if you have a minor source, it does not matter how you take your samples, as far as NESHAPS compliance is concerned. So, we hope that this is going to be an easy rule to follow.

DUVALL: I want to point out that the experience of getting the shrouded probe technology approved by EPA is an indication of what we need to have in the rule: a way to address new technology as it comes on line. There are a lot of EPA methods indicated in the rule that can be replaced as technology moves forward. A recent example is sampling uranium using the laser method; it gives much better sensitivity than the EPA method. Technology is going to move forward, and we need to be able to implement it across the DOE complex. One way we can do this is to adopt the new technologies, get them approved by EPA, and implement them across the DOE complex. I do not think that the development of technology was specifically addressed in the rule, but I think we now have an avenue to move forward on it.

GLISSMEYER: One quick comment. Now that you know that there will be deposition in a probe, when you order one, even when it is an ANSI-style probe, one of the hammers that you have over your suppliers is to ask for the deposition rate and ask for confirming

data. You are paying for that probe, so push them in the direction of less deposition in the probe. It probably can be done.

CLOSING COMMENTS OF SESSION CHAIRMAN MCFARLAND

I certainly appreciate the contributions of you the audience and the panelists. I would also like to mention that our colleague John Rogers who was scheduled to be the co-chairman for this session was unable to make it because of a medical problem. I would like to acknowledge his contributions over a very long period of time (since 1986) in attempting to improve the quality of air sampling in the nuclear industry.

SESSION 7

EFFLUENT STACK MONITORING

Tuesday: July 26, 1994
Co-Chairmen: J. J. Hayes
T. M. Punch

A REAL-TIME STACK RADIOACTIVITY MONITORING SYSTEM AND DOSE PROJECTION PROGRAM
A. P. Hull, P. A. Michael, H. J. Bernstein

POTENTIAL RADIONUCLIDE EMISSIONS FROM STACKS ON THE HANFORD SITE, PART 2: DOSE ASSESSMENT METHODOLOGY USING PORTABLE LOW-RESOLUTION GAMMA SPECTROSCOPY
J. M. Barnett

THE RETENTION OF IODINE IN STAINLESS STEEL SAMPLE LINES
G. J. Evans, C. Deir, J. M. Ball

23rd DOE/NRC NUCLEAR AIR CLEANING AND TREATMENT CONFERENCE

A REAL-TIME STACK RADIOACTIVITY MONITORING SYSTEM AND DOSE PROJECTION PROGRAM

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Abstract

At Brookhaven National Laboratory, a commercial Low- and High-Range Air Effluent Monitor has become operational at the 60 Mw (t) High Flux Beam Reactor. Its output data is combined with that from ground-level and elevated meteorological sensors to provide a real-time projection of the down-wind dose rates from noble gases and radioiodines released from the HFBR's 100 m stack.

The output of the monitor, and the meteorological sensors and the dose projections can be viewed at emergency response terminals located in the Reactor Control Room, its Technical Support Center and at the laboratory's separately located Meteorological Station and Monitoring and Assessment Center.

I. Introduction

A commercial low- and high-range monitoring system (Eberline SPING-3 and AXM-1) for the air effluent from the 60 MW(t) High Flux Beam Reactor (HFBR) at Brookhaven National Laboratory (BNL) has recently become operational. Commissioned in 1967, the HFBR is moderated and cooled with heavy water in order to optimize the thermal neutron flux at its beam ports, where most of its experimental facilities are situated. The 20,000 cfm ventilation of the HFBR's 4.9×10^4 m^3 confinement building and the reactor vessel off-gas are discharged from a 100 m stack (which was originally utilized for the 100,000 cfm flow through the now decommissioned Brookhaven Graphite Research Reactor). Up to the times of the installation of this system, the routine monitoring of the HFBR air effluent has been on a passive basis, except for an installed Kanne Chamber for the detection of HTO. Standard Operating Procedures were provided for the manual acquisition and analysis of non-routine samples and for dose projection in the event of abnormal concentrations resulting from fuel damage during an incident or accident.

At full power and at the end of the HFBR's 28 day fuel cycle, in which half of the 28 fuel elements in its core of 93% enriched uranium is discharged and the other half rotated, the HFBR's inventory of noble gases would be 7.1×10^6 Ci and of radioiodines 1.02×10^7 Ci. Since the HFBR operates at a relatively low pressure (~200 psi), it would not be subject to catastrophic vessel failure and its piping arrangement precludes the possibility of a pipe-break resulting in a rapid loss of coolant accident. However, in order to establish an upper limit on the concentrations of noble gases and

radioiodines that the monitoring equipment might see, a worst case accident scenario was postulated. The resulting peak effluent concentrations were of the order of several hundred $\mu Ci/cm^3$ of noble gases and several $\mu Ci/cm^3$ of radioiodine. These concentrations are well in excess of the range of a conventional low-range stack monitor, as well as of standard analytical laboratory capabilities. Thus, it was decided to acquire both a low- and high-range monitoring and sampling capability when the decision to install a real-time monitoring capability was made.

From its inception in 1949, the Laboratory has had a meteorological capability for obtaining measurements of surface and elevated wind directions and velocities and temperatures. Currently, these measurements are made at 10 m and 88 m, respectively (with a correction of the latter for the stack release elevation of 100 m). SOPs and map overlays were provided for the determination of stability classes and making dose projections.

At the time that the decision to acquire an active stack monitoring capability was made, it was also decided to develop a computer program which would merge the stack monitoring and meteorological data on a real-time basis. This decision was made to provide "State of the Art" compliance with DOE Orders requiring that its nuclear reactors have 24 hour monitoring capability for concentrations anticipated during routine operations and during accidents[1] and for consequence assessment and for protection actions during emergencies.[2]

II. Monitoring System

A. SPING-3A

As shown in Table 1, the SPING component of the monitoring system includes four channels for the measurement of the concentrations of radioactivity, one each for particulate beta activity and gaseous I-131 along with a low- and a mid-range channel for noble gasses. It also includes four other channels for the detection of "background" activity, some of which is subtracted from the active measurement channels. Each of the channel's detectors and its purposes are shown in Table 1. The ranges of the monitoring detectors are also shown. As also indicated in Table 1, three channels are devoted to the measurement of flow rates and one to the measurement of absolute pressure in the sample lines (used for the calculation of flow at STP).

B. AXM-1

As also shown in Table 1, the AXM component includes one channel which monitors the activity collected at a very low flow rate (~100 cm^3/min) on a particulate iodine collection cartridge, and two channels for the measurement noble gases in the high concentrations that would be anticipated during an accident which resulted in fuel damage or melting. One channel monitors "background," part of which is subtracted from the active measurement channels. Alarms for detector failure, loss of flow, "alarm" or "high alarm" conditions are provided for both the SPING and AXM components, along with "rate of increase" alarms.

The data from the SPING and AXM are accumulated by the individual channels, digital from radiation detectors or analog from measurement transducers (flow and pressure) and processed by these channels so that they can be displayed in any desired form. They are then sent to a control

Table 1

Radiological, Flow and Pressure Channels in the SPING-3A and AXM-1A

SPING Channel	Description/Detector		Range	Units
1	β Particulates		6×10^{-12} - 1×10^{-5}	
2	α Particulates (Rn)		Subtracted - Channel 1	cpm
3	Iodine-131	NaI-SCA	1.2×10^{-11} - 3×10^{-6}	$\mu Ci/cm^3$
4	Iodine 132-135	NaI-SCA		cpm
5	NG Low Range	β Scintillation	1.1×10^{-7} - 3.6×10^{-2}	$\mu Ci/cm^3$
6	γ Area	GM		mR/hr
7	NG Median Range	GM	9.2×10^{-4} - 1.5×10^{3}	$\mu Ci/cm^3$
8	γ Background		Subtracted Channel 5-7	cpm
10	Stack Flow Rate		-	cfm
11	SPING & AXM Flow Rate		-	cfm
14	Absolute Pressure		-	psig
15	SPING Flow Rate		-	cm^3/min
AXM Channel				
1	Grab Samples |	GM I-131	0.5 - 1×10^{6} 1×10^{-6} - 6×10^{1}	cpm $\mu Ci/cm^3$
2	NG Background		Subtracted - Channel 3&4	cpm
3	NG High Range	GM	6.9×10^{-1} - 1.2×10^{5}	$\mu Ci/cm^3$
4	NG Intermediate Range	GM	1.5×10^{-4} - 6.9×10^{1}	$\mu Ci/cm^3$

terminal (Eberline CT-1B). Control functions and data display are available at the control terminal. Various command and data readout capabilities and system status are available on four peripheral communication ports, one of which is utilized for a report generator interface (RGIF).

History files are available for twenty-four one-minute, ten-minute, one-hour and one-day periods. Through Eberline provided software, these data as well as channel parameters and system status, may be displayed and commands may be initiated from remote IBM-PC compatible computers.

III. Meteorological Measurements

Brookhaven's current meteorological capability includes the measurement of standard meteorological parameters, including the wind speed/direction and temperatures at 10 and 88 meters. These data are sampled once per second and are reported into a data logger. Averages (as well as some maxima, minima and variances) are reported once per minute as ASCII strings of 5-digit real numbers.

IV. Emergency Response Network and Terminals

Software has been developed for real-time emergency response displays of the stack monitor data and the meteorological data.[3] A Class A plume model is utilized to provide graphical and numerical projections of the selected environmental dose rates. This software runs in a complex environment of computers and network connections to PCs configured in emergency response terminals (ERTs). The source term and meteorological data flows are shown in Figure 1.

The plume model is a simple straight line, steady state, Gaussian Plume model of transport and dispersion of a type that has been discussed extensively in the literature.[4] Input to the model is the location of a source, its height above the surface, representative wind speed, wind direction, and Pasquill Gifford atmospheric stability classifications. The model calculates near surface relative concentrations as a function of distance and direction from a source. The output is a set of isopleths of (χ/Q) overlaid on a map of the BNL site or a listing of the maximum χ/Q, and other information on the spatial distribution of the ground level concentrations. The key to its utility of a Class A model is its link to real time wind directions and wind speed data and atmospheric stability and it is designed to provide those making decisions during emergencies with information on the region where action should be taken; and in conjunction with source information, the magnitude of potential threats to expose populations.

The effluent source term data is collected in Building 715 (the Stack Monitoring Facility) on the Eberline CT-1B, and distributed to Building 750 (the HFBR control room), Building 120 (the Reactor Technical Support Center), Building 703, (the backup Technical Support Center), Building 051, Room 19 (the Monitoring and Assessment Support Center), Building 515 (Communications & Computing Division) via a relay in Building 051, Room 7 (the Meteorological Forecast Room). The data from two meteorological towers and other instruments are collected in Building 051 on data loggers (Campbell Scientific Inc.) in and sent to two PCs in Building 051, a μVAX 3100 (BNLMET) in

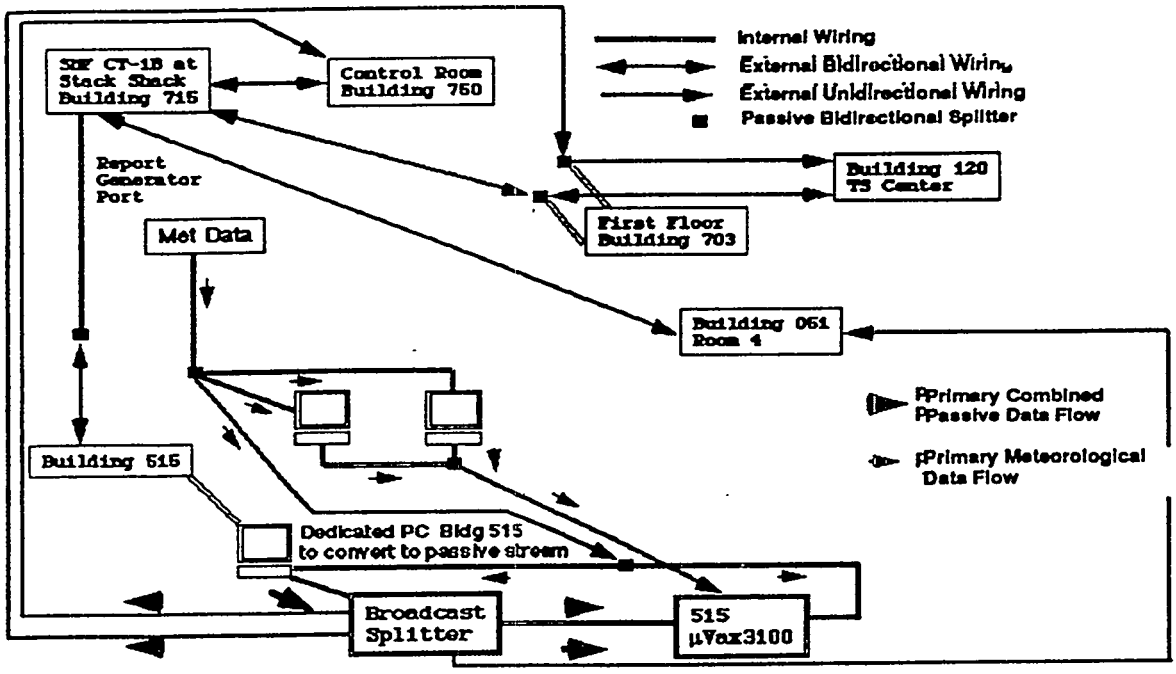


Figure 1. Effluent Source Term Data and Meteorological Data Flows for SMF

Building 515, and to a PC (METPC) in Building 515 where it is combined with the effluent source term data once per minute. The combined data is then redistributed to BNLMET and to ERT in Buildings 750, 120, 703 and 051.

The Emergency Response Terminals (ERTs) for the Stack Monitoring Facility are specially programmed Personal Computers, either Gateway 2000 clones of IBM PCs or IBM PS/2s. The machines run the IBM OS/2 2.0 operating system, which combines the capabilities of MS-DOS and Windows with the ability to run several real-time programs at once. As shown in Table 2, "Window List," many displays can be accessed. Some of the listed programs are not for end-users, but for diagnosis of problems. The principal user displays are listed in Table 3.

Within the SHOW15R.CMD, SHOW15.CMD and SHOW60.CMD displays, screens are available for the display of instantaneous present and past plume meteorological and dispersion data, and Source (monitor channel indication and release rates) Plume and Source Data Lists for the current and nineteen previous time intervals (two minute for the 15-minute running average, 15 minute for the 15-minute fixed windows, and hourly for the 60-minute fixed window screens).

Table 2. Window List

CMD.EXE
DAILYRUN.EXE
GETRAW.EXE
SHOW15R.CMD
SHOW15M.CMD
SHOW15P.CMD
SHOW15S.CMD
PCCRT.EXE
OS/2 2.0 Desktop - Icon View
OS/2 Full Screen - OS/2 Full Screen
LOGMET.EXE
LOGMET.EXE
ERTMENU.CMD
PLPLOT.EXE

Table 3. User Displays

SHOW 15R.CMD
SHOW15.CMD
SHOW 60.CMD

These three programs show 15 minute running average text displays updated **every 2 minutes** (...15R...), 15 minute fixed time-window displays (...15...) **updated every 15 minutes on the hour**, 15 minutes after the hour, 30 minutes after the hour and 45 minutes after the hour, and 60 minutes displays (...60...) **updated on the hour.**

PCCRT.EXE

This is the Eberline Instrument Corporation program which can control the instruments at the Stack Shack. It can be used to reset alarms and is an alternate means of access to source term data in case of failure of the system which combines meteorological and source term data.

PLPLOT.EXE

This is a graphic display program showing an approximate map of the BNL site overlaid with contour lines of an estimated release plume.

An example of a 15-minute running average Plume Screen for 16:01 EST on June 16, 1994 is shown in Figure 2. The wind direction and velocity are calculated for the 100 m release height of the HFBR stack on the basis of a power law extrapolation from the 88.1 m tower observations. The horizontal stability class is calculated from the 88.1 m $\sigma\theta$ (the standard deviation of the wind direction) and the vertical stability on the basis of the ΔT between 88.1 and 10.6 m. As shown, the maximum χ/Q, the distance and direction to it, and the plume half-widths at several distances are also calculated, as is the transit time to them. The 15-minute running average screen is updated every two minutes. The 15-minute fixed time window screen is updated on the hour and every 15 minutes thereafter, while the 60-minute screen is updated on the hour.

An example of the 15-minute running average Source Screen for 16:25 on the same date is shown in Figure 3. This screen shows a time post-shutdown, which is based on a user entry of the time of reactor shutdown. This time is used in the calculation of the release concentration (as Xe-133 effective) for the mixture of released noble gases and of radioiodines as I-131 effective. The 100 m release winds, stability classes and release concentrations and rates are also shown. The latter are calculated on the basis of the measured stack flow. The maximum χ/Q, as well as the direction and distance to it and to 10% of the maximum, is also shown, as are the concentrations of I-131 equivalent and Xe-133 equivalent and related dose rates at the location of the maximum. The SPING sampling flow rate and the SPING and AXM channel readings are also shown on the Source Screen to facilitate a quick review of them.

As indicated in Table 2 a Plume Display screen is also provided, an example of which is provided in Figure 4. The display is designed to be self-contained (i.e., the meteorological assumptions are listed, the definition of wind directions and the map scale and orientation are explicitly stated). The relevant meteorological parameters, the location of the ground maximum and isopleths at the 50%, 10%, 1%, 0.5% and 0.1% of maximum are shown, superimposed on some of the essential features of the Laboratory site and surrounding areas. This projection is automatically updated every fifteen minutes, and a more frequent update may be obtained manually on request, as may be its magnification factor to encompass a smaller or larger area than the standard 6 km × 6 km.

```
 PLUME SCREEN      TOWER DATA LOGGING PROGRAM  15 Min. Running Ave.
Computer Date: Jun 16,1994          Time: 15:48:34
Logger Julian Day 167.              Time: 16:01     EST
                          WIND SPEED/WIND DIRECTION
                                    ( mps /deg)
              100.0 M (Release Ht):     3.1/116.
               88.1 Meter:              3.1/117.
               10.6 Meter:              3.5/123.
    HORIZONTAL PG STABILITY (Winds)       D     ( .35)
    VERTICAL PG STABILITY                 A     ( .00)
    Maximum Chi/Q                      1.92E-05
    Dist/Direction TO max (km, deg)         .4/296.
    Dist to 10% of max (km)               1.9  @ max 1.6km 3.2km 8.0km 16 km
    Plume half-widths to 10% of max:                .06   .09   .00   .00   .00
    Transit time (hh:mm):                          0:01  0:08  0:17  0:43  1:26
Temperatures        (Deg C  Deg F)   Other Data Measurements
Shelter :             24.8   76.7    Precipitation (in):              .00
10 Meter (Field)      23.7   74.6    Relative Humidity (%):           65.
10 Meter (Tower)      23.8   74.8    Pressure (in of Hg):             30.08
87.3 Meter (Tower     22.1   71.8    Solar Radiation:                 .74
Delta T :             -1.64  -2.96   88.1 Meter Sigma Theta:          10.7
Minimum Shelter :     24.7   76.4    10.6 Meter Sigma Theta:          999.0
Maximum Shelter :     24.9   76.9    QA Count (% valid data)          100.00
 up/down arrows = scroll, space = +10, return = home, h = help, q = quit
```

Figure 2. Plume Screen, June 16, 1994

```
SOURCE SCREEN     TOWER DATA LOGGING PROGRAM  15 Min. Running Ave.
Today Is Thu, Jun 16,1994 jdate 167, Time: 16:12      |      .0 hrs postShutdown
  Logged Thu, Jun 16,1994 jdate 167, Time: 16:25 EST  |      .0 hrs postShutdown
--  100.0 METER RELEASE WINDS-----------------------------|
wind speed/direction (mps/deg)     3.5/134.               |
horizontal PG stability (Winds)    C ( .20)               |----PLUME MAXIMUM
vertical PG stability              A ( .00)               |  Concen Dose rate
ESTIMATED RELEASE RATES--------uCi/cc---Ci/sec---status---|  Ci/m3    Rem/hr
 Iodine (I-131 equivalent)      5.01E-11 2.57E-10 normal   | 3.21E-15  8.20E-10
 Noble Gas (Xe-133 equivalent) 2.72E-03 1.40E-02 normal    | 1.74E-07  8.16E-05
-------------------------------------------------------------
    STACK FLOW: 10885.cfm    5.14 m3/s | PLUME MAXIMUM CHI/Q:  1.25E-05 sec/m3
SPING+AXM FLOW:     1.cfm    normal    |  Dist/Dir TO max (km,deg)     .4/314.
-----------------------------------------|--------Dist to 10% of max   1.9 km

SPING flow  1.20E+00 cfm     normal     Gamma Bckgd 1.38E+00 cpm     normal
Beta part   6.66E-12 uCi/cc  normal     Abs Press.  2.81E+01 inHg A  normal
Alpha part  4.42E-01 cpm     normal     AXM GrabSmp 2.62E-02 mR/h    normal
Iodine      5.01E-11 uCi/cc  normal     AXM NG Bckg 1.06E+00 cpm     normal
I-Backgnd   8.18E+01 cpm     normal     NG hi rng   1.84E-01 uCi/cc  normal
NG low rng  5.31E-07 uCi/cc  normal     NG Intrmed  1.55E-04 uCi/cc  normal
Gamma Area  8.33E-02 mR/h    normal
NG med rng  2.72E-03 uCi/cc  normal     QA Count (% valid data)      100.00
 up/down arrows = scroll, space = +10, return = home, h = help, q = quit
```

Figure 3. Source Screen, June 16, 1994

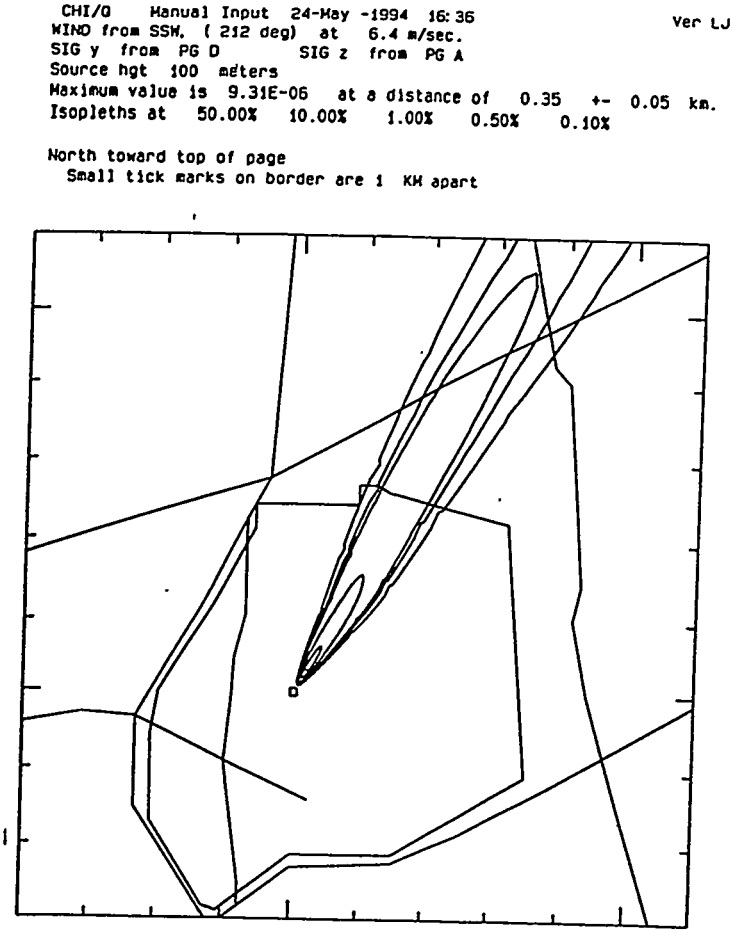


Figure 4. Plume Display

References

1. DOE Order 5480.30 "Nuclear Reactor Safety Design Criteria", 1-19-93.

2. DOE Order 5580.3A "Planning and Preparedness for Operational Emergencies", 4-30-91.

3. H. J. Bernstein, "SMF Software Manual," Bernstein & Sons, 5 Brewster Lane, Bellport NY 11713, April 7, 1993.

4. See, for example, F. Pasquill, Atmospheric Diffusion, 2nd Edition, John Wiley and Sons, 1974.

DISCUSSION

EVANS: I am curious about how you come up with the correction factors to get your ratio of iodine to noble gases. Is this based on some type of measurement or is it based on a release rate?

HULL: We do not do it by ratios, we do it by the assumption that there is a fixed mixture of noble gases (xenon and krypton) and a given mixture of iodine 131-135, which has been worked out for default accident scenario purposes. The very short life krypton and xenon are so short-lived that we do not put them in the calculation; they are not dose contributors at any distance away, at any time after shutdown. We start with the assumption of this mixture at shutdown and just correct it on the basis of decay of each of the individual constituents with time. We also throw in the dose factors, the effective energy of the mix for the gases and the iodides. A lot of the accident projection schemes work on the assumption of iodine-to-gas ratio. We do not do that, we operate on studying the shutdown mix.

EVANS: Maybe it is not a fair question to address to you, but you have assumptions of releases from the core for a given accident scenario, plus volatilities, etc. Has somebody else done these calculations to come up with the ratios that are going to become airborne in time? What sort of values go into this? You have assumptions there about what fraction of the iodine is going to volatilize.

HULL: There is a default assumption on the accident scenario as to the form of iodine. The 20,000 CFM effluent is continuously filtered by high efficiency filters for particulates. And the effluent also goes through an iodine adsorbent bed, so the assumption is that most of what gets out is in the form of metryl iodide.

HAYES: Is this particular monitoring system also utilized to report normal effluents?

HULL: We have this in mind, but it turns out, we have discovered, that we have to work around to get the instrumental backgrounds down to the point where we can see them. When we got the backgrounds down to the lowest predictable level, sensitivity was sufficient so that it turns out that we are putting out about 1×10^{-7} μci/cc of argon-41 on a continuous basis. Until we had the monitor, we could not see it. So, the answer is, yes, I think we will be utilizing it for routine effluents, too. But then you can not make assumptions about the mix, you have to do some grab sampling to establish the constituents.

HAYES: With respect to your normal effluents, would you be likely to change the program so that your mix would be based upon a weekly grab sample?

HULL: We haven't thought about it seriously, but I think we will probably continue to have it interpret the accident situation, because that is really what we developed it for. We do not want to have to change it the minute we have an accident, we want to perform in that mode. We will probably just do grab sampling to say we have so much in iodine-131 equivalents or so much in xenon-133 equivalents. We would reinterpret it, in terms of what we think is the principal constituent in the routine releases, and that will

be established by grab sampling.

GOOSSENS: How do you reckon with the presence of noble gases that might be emitted?

HULL: I am not sure I understand your question because major components of what we are monitoring are the noble gases that are routinely released, in very small quantities, so small that we can not measure them on an everyday basis. Our intention is to monitor the noble gas releases that might be expected after any damage to the reactor fuel. I should have said while I am with the Safety and Environmental Protection Division, our Reactor Division has sponsored this installation. The idea was that, sooner or later, somebody from the DOE was going to come around and say that our monitoring was insufficient in terms of industry practice and the precedents that were established by NUREG-737. So, we decided to stay ahead by doing it on our own. The Reactor Division paid for it and I should have expressed my appreciation to them. The meteorological component is very user friendly, it can be tailored for other elevations in the release and for other locations than the reactor stack that you saw in the last slide. At present, it assumes that the point of release is the top of the reactor stack at 100 m. In the future we will be trying to make it a little more flexible in terms of emergency response for other facilities and release heights.

POTENTIAL RADIONUCLIDE EMISSIONS FROM STACKS ON THE HANFORD SITE, PART 2: DOSE ASSESSMENT METHODOLOGY USING PORTABLE LOW-RESOLUTION GAMMA SPECTROSCOPY*

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Abstract

In September 1992, the Westinghouse Hanford Company began developing an *in situ* measurement method to assess gamma radiation emanating from high-efficiency particulate air filters using portable low-resolution gamma spectroscopy. The purpose of the new method was to assess radioactive exhaust stack air emissions from empirical data rather than from theoretical models and to determine the potential unabated dose to an offsite theoretical maximally exposed individual. In accordance with Title 40, Code of Federal Regulations, Part 61, Subpart H, "National Emission Standards for Hazardous Air Pollutants", stacks that have the potential to emit ≥ 1 μSv y^{-1} (0.1 mrem y^{-1}) to the maximally exposed individual are considered "major" and must meet the continuous monitoring requirements. After the method was tested and verified, the U.S. Environmental Protection Agency, Region 10, approved its use in June 1993. Of the 125 stacks operated by the Westinghouse Hanford Company, 22 were targeted for evaluation by this method, and 15 were assessed. (The method could not be applied at seven stacks because of excessive background radiation or because no gamma emitting particles appear in the emission stream.) The most significant result from this study was the redesignation of the T Plant main stack. The stack was assessed as being "minor", and it now only requires periodic confirmatory measurements and meets federally imposed sampling requirements.

I. Introduction

In September 1992, the U.S. Environmental Protection Agency (EPA), Region 10, concurred with a Westinghouse Hanford Company (WHC) proposal to develop a test method to measure radioactive particulate emissions [particularly gamma emitters from high-efficiency particulate air (HEPA) filters] in stack exhauster systems across the Hanford Site. Its intended purpose was to show the compliance status of potential unabated stack emissions and the estimated dose to the offsite maximally exposed individual (MEI) through the use of empirical data. A nondestructive assessment, in situ method was developed, tested, and implemented using portable low-resolution gamma spectroscopy instrumentation (Microspec-1**) to evaluate gamma emissions from operating HEPA filters. This method has become commonly known as the nondestructive assessment (NDA) method.

* Work conducted for the U.S. Department of Energy under contract number DE-AC06-87RL10930.

** Microspec-1 is a registered trademark of Bubble Technology Industries, Inc., Chalk River, Ontario, Canada.

Guidance for the developed method comes from Title 40, Code of Federal Regulations (CFR), Part 61, Subpart H[1] and Appendix B, Method 114[2].

On May 19, 1993, the Microspec-1 was taken to the B Plant and used to measure gamma emissions from the 296-B-13 stack HEPA filters to test the NDA method and to provide concrete results to the Region 10 EPA office. This location was chosen because it is readily accessible, and gamma counting may be performed without difficulty. Data acquired from the experiment was used to determine an offsite effective dose equivalent (EDE) to the MEI over a 1-y period assuming no filtration of the exhausted air. The results were presented, and a demonstration of the NDA method was given on June 10, 1993 to the EPA, Region 10. The method was accepted as an alternative to the other approved EPA methods for determining potential offsite doses, and implementation began across the Hanford Site where applicable.

The NDA method provides a technical basis for empirically measuring potential emissions, and it uses low-resolution portable gamma spectroscopy to collect *in situ* spectra from HEPA filters. By calibrating the instrument to the same geometry as that observed in the field, data acquired from field measurements may be decay corrected for filter lifetime to determine the potential annualized activity release to the atmosphere. Several assumptions made regarding the activity calculation make the reported value conservative: (1) filters otherwise considered to have zero activity are assigned the minimum detectable activity (MDA); (2) the total activity is placed on the filter at the time of installation; and (3) alpha and beta emissions are established based on known relationships to the gamma emitters. Dose conversions from the activity determination are made from EPA-approved dose conversion factors. Stacks with resultant dose estimates less than 1 μSv y^{-1} (0.1 mrem y^{-1}) to the offsite MEI are considered "minor" and do not require continuous monitoring for radionuclide emissions. High radiation near or emitted from HEPA filters limits the application of this method.

II. Objective and Scope

The objective of this test method is to identify gamma emitting radionuclides collected on HEPA filters, in a quantitative manner, over a discrete time period at the Hanford Site facilities managed by WHC using portable gamma spectrometry. Portable gamma spectrometry may also be used qualitatively to confirm the presence of gamma radiation. These data, in conjunction with process and plant history records, are used to determine an empirical potential unabated source term that has the potential for release to the atmosphere. As a result of determining the unabated source term, a dose to the MEI may be calculated, which will determine if the emissions from the stack are greater than the EPA threshold of 1 μSv y^{-1}. Alpha and beta emitting radionuclides may not be measured using the NDA method but are established using known facility ratios. Volatile nuclides and other nonradioactive materials are also beyond the scope of this method.

III. Test Method

The Microspec-1 is used, and data may be collected with or without a lead sleeve [i.e., a collimated shield over the NaI(Tl) detector]. Before each field use, the instrument is calibrated to the in-field HEPA filter geometry under controlled conditions. A background spectrum is collected at the test site, and

several HEPA filter spectra are collected. The setup, results, and offsite dose calculations are discussed for the method.

Sampling was initially conducted on 9 of the 84 WHC stacks registered with the Washington State Department of Health[3]. All 84 stack assessments were completed by December 17, 1993[4]. Beginning in 1994, unregistered WHC stacks were evaluated for potential emissions greater than the EPA threshold of 1 μSv y^{-1}; all reporting was completed by August 31, 1994.

Instrument Calibration

In the laboratory, the spectrometer is energy calibrated using the two peaks obtained from a ^{22}Na check source placed in a representative HEPA filter geometry. ^{22}Na is preferred for an energy calibration because it has both a low-energy (511 keV) and a high-energy (1274.5 keV) gamma emission. National Institute of Standards and Technology (NIST) traceable radioactive check sources are placed in the center of the calibration HEPA filter. Because of the distance from the detector to the source, the point source is approximated for calibration, even though the activity in a HEPA filter to be monitored is widely deposited. For all measurements of the same geometry, the detector head is placed up to the clean HEPA filter or housing at a location similar to where the actual measurement will be taken. After the energy calibration, a region of interest (ROI) is established for each isotope expected to be found during measurement for the given HEPA filter geometry using NIST traceable gamma check sources implanted in the clean HEPA filter. Typical isotopes include ^{54}Mn, ^{60}Co, and ^{137}Cs. The absolute efficiency is given by Equation 1.

$$\eta = \frac{\left[\frac{counts}{t(1-DT)}\right]_{source} - \left[\frac{counts}{t(1-DT)}\right]_{background}}{A_o e^{-\lambda \Delta t}} \times 100 \quad [\%] \qquad (1)$$

where:

η = absolute efficiency in percent
counts = gross counts in region of interest
t = count time of source, background (min.)
DT = instrument dead time i.e., fractional time instrument does not respond
A_o = initial check source activity (dpm)
λ = decay constant of check source (y^{-1})
Δt = time elapsed since check source manufacture date.

Data Collection

Before sampling begins, background measurements are collected. The background spectrum is collected within 10 to 20 m of outdoor systems; for indoor systems, the background spectrum is collected within the room or facility. The minimum detectable activity for a particular ROI is given by Equation 2[5].

$$MDA = \frac{4.66\sqrt{\left[\frac{counts}{t^2(1-DT)}\right]_{background}}}{2.22\eta} \quad [pCi] \qquad (2)$$

where:

MDA = minimum detectable activity for given ROI
4.66 = the value used to determine the instrument reading with a difference from background that is large enough to be statistically significant at the 99% confidence level
2.22 = the conversion from counts per minute to picocuries.

Each stack evaluation is governed by an individual WHC test plan. In the field, *in situ* measurements are made of the gamma radiation emanating from the HEPA filters and prefilters. Instrument data are logged, and the resultant spectra are recorded electronically. The instrument is placed near the HEPA filter in the same geometry as during calibration. This method eliminates the risk of radiation dose from HEPA filter changeout and the resultant disruption of the exhaust system. Count times vary between three and five minutes during which a gamma spectrum is collected. For the specific ROI(s), the collected counts (corresponding to a gamma ray) are converted to a particular activity. Stacks that cause excessive instrument dead time (≈> 70%) cannot be assessed using this method.

Measured activities are decay corrected and annualized. The HEPA filter activity is given by Equation 3. In systems that have them, prefilters are included in the analysis for the total activity calculation. Each resultant value is compared to the MDA. If the value is below the MDA, the MDA is assigned as a measure of conservativism. Additionally, activity estimates are made for alpha and beta emitters from ratios derived from the gamma emissions. Dose conversions from the activity determination are made from EPA-approved dose conversion factors. Summing the reported annual isotopic emissions and the derived average annual potential emissions from the HEPA filters results in the annual total potential emissions. Stacks with resultant dose estimates less than 1 μSv y^{-1} to the offsite MEI are considered minor stacks needing only periodic confirmatory measurements.

$$A_{i_{HEPA}} = \frac{\left[\frac{counts}{t(1-DT)}\right]_{i_{HEPA}} - \left[\frac{counts}{t(1-DT)}\right]_{background}}{2.22\eta e^{-\lambda t_{i_{HEPA}}}} \quad [pCi] \qquad (3)$$

where:

A_{iHEPA} = activity of i th HEPA filter (pCi)
t_{iHEPA} = time elapsed from HEPA filter installation to measurement date (y).

Potential Dose Calculation

The Hanford Site employs the CAP-88 dose model[6] to demonstrate compliance with 40 CFR 61.92. This model calculates a unit dose factor for each radionuclide discharged from the Hanford Site by discharge area to predict an EDE for the offsite MEI. The EDE is multiplied by the activity of each isotope, and the results are summed to produce the offsite potential dose to the MEI as shown in Equation 4[7, 8].

$$\dot{H}_{total} = \Sigma(EDE_{(i)} \times A_{(i)}) \quad [mrem\ y^{-1}] \qquad (4)$$

where:

$\dot{H}$ = potential unabated dose to the MEI
$EDE_{(i)}$ = effective dose equivalent for each radioisotope.
i = each applicable radioisotope

IV. Results

Calibration of the Microspec-1 2 x 2 NaI(Tl) well detector has consistently yielded for the ^{137}Cs ROI an intrinsic efficiency ranging from 11% to 16%. The calculated detector resolution using ^{137}Cs for a typical HEPA filter geometry is 8.5%.

Although the background radiation may vary with location, the Hanford Site typically has a natural background on the order of 10 to 15 $\mu R\ hr^{-1}$. Using the

in-field background data collected, the average MDA is about 100 nCi. MDA values range from about 50 nCi to 200 nCi.

Of the 84 registered WHC stacks, 14 were considered for NDA; however, only 9 were evaluated using the NDA method. Although one stack was readily accessible, the background and HEPA filter radiation emanating from the area rendered the instrument useless because of the excessive instrument dead time. Four of the available stacks could not be assessed with this method because the source term was solely alpha contamination. Figure 1 compares the results of the NDA assessment with the back calculation method (i.e., $3{,}000^{n}$ x emitted dose, where n is the number of HEPA filter stages) for the registered stacks. Without using the NDA method, all nine stacks assessed would have been subject to full compliance with the NESHAP requirements, which would have subjected them to upgrades.

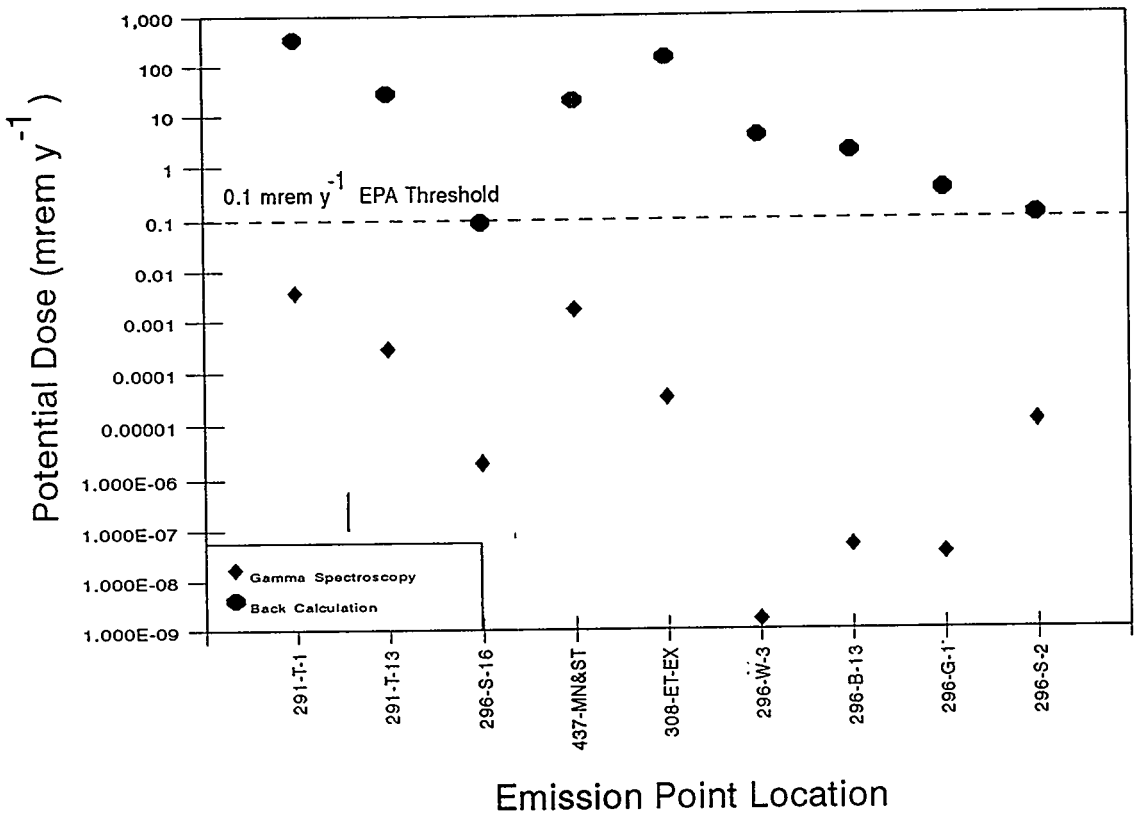


Figure 1 Comparison of the empirical results obtained from portable gamma spectrometry versus the theoretical back calculation model[8].

A significant result was obtained for the T Plant main stack. This is a 60 m (200 ft) stack and had been considered a "major" source of potential radioactive emissions. It had been designated in 1991 by the EPA, Region 10, as requiring upgrades to meet all of the NESHAP and Clean Air Act requirements. However, the empirical results using the NDA method showed conclusively that the potential emissions were not only below the 1 μSv y^{-1} EPA threshold but also five orders of magnitude below the theoretical estimated potential. As a result, a petition for redesignation was submitted to the EPA, Region 10[9], and the EPA, Region 10, accepted the proposal in a letter transmitted January 3, 1994. All actions for this stack are now considered complete, and it only requires periodic confirmatory measurements as specified in 40 CFR 61.93.

There were 41 WHC unregistered stacks assessed for potential emissions in 1994. Eight stacks were considered for NDA, and six of the unregistered stacks were assessed for potential unabated emissions using the NDA method. None of the NDA sampled stacks were above the EPA threshold. Two stacks could not be assessed because the background radiation interfered with the gamma radiation coming from the HEPA filter system.

V. Discussion and Conclusions

The data confirm the ability to measure *in situ* environmental, low-level concentrations of gamma radiation emanating from HEPA filters using portable low-resolution gamma spectrometry. Increasing the count time decreases the instrument dead time, resulting in better counting statistics. Using the collimated shield reduces the instrument dead time and also increases the data reliability. Using a point source within the HEPA filter to calibrate the instrument and establish the ROI is sufficient and enables specific and repeatable geometries; this is important considering the many stack geometries on the Hanford Site. Results have shown NDA to be as much as nine orders of magnitude below the theoretical back calculation model.

Conservative measures were used to calculate the potential unabated dose to the MEI. The first of these includes assigning the MDA to a filter that shows an activity less than the MDA. This assignment may represent greater than 25% of the actual measured activity; indeed in some cases it is 100% of the measured activity. The second conservative influence involves back-calculating the deposited activity to the date of the HEPA filter installation. This assumes all the activity accumulates on the HEPA filter the day it is installed instead of accumulating over time, as actually occurs. Third, the prefilter contribution to the potential dose is derived from empirical data. Fourth, although no alpha or beta activity is measured by this method, alpha and beta contributions are established based on known ratios of these emitters to the measured gamma emitter (e.g., ^{137}Cs). The alpha and beta components may account for as much as 70% of the potential unabated dose to the MEI.

This method clearly demonstrates the ability to use NDA for ascertaining the collected gamma-emitting isotopes on HEPA filters. Guidance for the method development comes from 40 CFR 61, Appendix B, Method 114, the intent of which was followed as much as possible. However, unlike Method 114, which requires taking a sample from the HEPA filter and sending it to a laboratory for analysis, the method developed uses nondestructive, *in situ* sampling. Advantages include reduced exposure and dose to personnel, time savings over laboratory analysis, and cost savings gained from reduced labor time and analytical laboratory

analyses. However, the variety of systems at the Hanford Site prevent applying this method universally, among other reasons, because of difficulties encountered with higher radiation areas, limited access into radiation zones, and restricted proximity to the HEPA filters.

VI. Acknowledgements

The author thanks Ms. Barbara Boucher, Victoreen, Inc., without whose help this project would not have been completed in a timely manner, and Mr. Jack Mizner who spent many hours learning the method and working on the T Plant main (291-T-1) stack redesignation petition. Additional thanks go to Mr. Larry Diediker, Manager of Effluent and Emission Monitoring (WHC), and the dedicated WHC air emissions cognizant engineers, facility operators, health physics technicians, and publication services editors.

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DISCUSSION

WREN: You are looking at other parameters besides cesium-137, is that correct?

BARNETT: That is correct.

WREN: Because the viewing volume of the gamma spectrometer is quite small, you are looking at a very small fraction of the HEPA filter. Do you see any changes in total activity, depending on where you are? Does the composition of the trapped particles change depending on where you are looking along the HEPA filter?

BARNETT: We have looked at that and have determined efficiency and activity based on a point source calibration. We found that moving sources around, i.e. implanting sources in one of nine locations on a grid on a clean HEPA filter, did not change efficiency significantly. This has resulted in a point source calibration approximation.

DUVALL: It seems as though you have some very large correction factors, particularly the geometry correction factor, because you are assaying the entire concentration of radionuclides on the HEPA filter. Therefore, you are measuring a fraction of that and you have to correct for four-pie geometries to assess the entire filter. In addition to corrections for alpha and beta contributions, which are a significant fraction of the activity, you are taking a correction factor for geometry that may also be large, and applying them to your overall result. How do those two factors impact your measurements in terms of the uncertainty of the overall results?

BARNETT: First, the instrument is calibrated to the particular geometry that we are looking at i.e., one HEPA filter, so geometry is taken into consideration at the point of making the efficiency determination. Second, we have used the geometry calibration to determine what the alpha and beta contribution is going to be. It can vary somewhat, based on laboratory results, and there are errors associated with it also. When you combine them to make a final dose estimate, the error, typically, is on the order of 30%. The alpha and beta contributions are assessed; they add about 5% to the total error. In some locations, because of the types of activity, errors can be on the order of 50%, but, typically they are 25-30%.

HULL: I do not understand why all this attention is given to what is on the HEPA filters. If you monitor the stack, and know what is going on at that monitoring point, why would you be concerned about what did not get out?

BARNETT: We are looking at the <u>potential</u> for emissions from the stack, as required by NESHAPS. What we did was to look at what is on the filter and use that to determine our potential release in a year. We take that value and develop what the maximally exposed individual would receive as a dose from a particular stack.

HULL: I am not familiar with NESHAPS; is "potential" defined by the regulation?

BARNETT: Yes, "potential" is the theoretical stack emission in a year under normal operating conditions without filtration. The potential is considered unabated. We are looking at

the filters to see what has been collected.

KARHNAK: Perhaps I can offer a little clarification. If a stack can potentially discharge more than 0.1 mrem/y, it requires continuous monitoring. If it is below that, it does not require continuous monitoring. That is the reason for measuring what is upstream in the filter.

HAYES: Have you determined what isotopes are collected on the HEPA filter on a routine basis, or is the historical data sufficient? Have you taken upstream samples to test the assumption utilized in your evaluation that the mix would not change?

BARNETT: Before we go into a facility, we do a characterization. We also have the facility's environmental monitoring plans and other documents that tell us what types of activity we can expect to see. If that changes from one year to the next, or the mission of the facility changes, we are able to go back and make changes based on the new isotopes and the new activities that we would expect to see. The portable gamma spectrometer has an isotope identification capability. Therefore, gamma emitting isotopes collected on HEPA filters may be tagged. Upstream samples can be taken to establish radioactive material present in a system. I haven't done this, but have relied on recent and historical publications to determine the radioactive materials in the exhaust stream.

THE RETENTION OF IODINE IN STAINLESS STEEL SAMPLE LINES

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Abstract

Following an accident in a multi-unit CANDU nuclear generating station, decontamination of air vented from containment, by the emergency filtered air discharge system (EFADS), would play a critical role in minimizing the release of iodine to the environment. The concentration of gas phase iodine in containment air would be measured using the post accident radiation monitoring system (PARMS), requiring that air samples be passed through a considerable length of tubing to a remote location where the desired measurements could safely be made. A significant loss of iodine, due to adsorption on the sample line surfaces, could greatly distort the measurement. In this study, the retention of $I_2(g)$ on stainless steel was evaluated in bench scale experiments in order to evaluate, and if possible minimise, the extent of any such line losses.

Experiments at the University of Toronto were performed using 6" lengths of 1/4" type 316 stainless steel tubing. Air, containing ^{131}I labelled $I_2(g)$, ranging in concentration from 10^{-10} to 10^{-6} mol/dm^3 and relative humidity (RH) from 20 to 90 %, was passed through tubing samples maintained at temperatures ranging from 25 to 90 °C. Adsorption at low gas phase iodine concentrations differed substantially from that at higher concentrations. However, in general, the rate of deposition was proportional to the gas phase concentration, giving support to the concept of a first order deposition velocity. The surface loading increased with increasing relative humidity, particularly at low RH values, while the deposition rate decreased with increasing temperature, in contradiction to the behaviour expected based on an Arrhenius type relationship. It is speculated that surface water on the steel may play an important role in the deposition process.

Measurements using XPS and SEM at Whiteshell Laboratories revealed that the chemisorbed iodine was located primarily in areas of corrosion. Furthermore, water used to wash the steel contained Fe, Mn and iodine in the form of iodide, suggesting that I_2 reacted to form metal iodides. The deposition of I_2 was also found to depend on the initial surface condition, such as the degree and type of oxidation present. Pretreatment with nitric acid reduced the retention as did electropolishing.

The results to date suggest that extensive loss of $I_2(g)$ would likely occur in the PARMS and that corrections for the extent of such loss will be unreliable, given the apparent complexity of the phenomena involved.

Introduction

Following an accident in a multi-unit CANDU nuclear generating station, decontamination of air vented from containment, by the emergency filtered air discharge system (EFADS), would play a critical role in minimizing the release of iodine to the environment. It is important that the concentration of gas phase iodine be measured, in order to to estimate the potential radiological impact of any iodine released to the surrounding population. This requires that air samples be passed through a considerable length of tubing to a remote location where the desired measurements can safely be made. The equipment required for this sampling and measurement is referred to as the Post Accident Radiation Monitoring System (PARMS). The vented air may contain gaseous iodine in various forms such as $I_2(g)$ and $CH_3I(g)$, particulates and noble gases. A significant loss of iodine during the passage through the tubing, due to adsorption on surfaces, could greatly distort the measurement.

Previous work has established that $I_2(g)$ can readily be adsorbed by stainless steel under some conditions[1]. Both physical and chemical adsorption appears to be involved, with the chemical sorption being related to corrosion of the steel surface[1,3] and the formations of metal iodides[2]. However, large differences exist in the deposition rates and iodine loadings that have been reported[1,4]. Furthermore, most of the previous work has involved higher temperatures and gas phase concentrations than those relevant to the PARMS.

In this study, the retention of $I_2(g)$ on stainless steel was evaluated in bench scale experiments. The overall objective was to evaluate, and if possible find methods to minimize, iodine deposition under conditions relevant to the PARMS. Specific objectives included evaluating the impact of operational conditions, such as air flow rate, temperature, humidity and gas phase concentration, and identifying chemical transformations occurring on the steel surface. Even though this study is still ongoing, some useful trends have already been identified suggesting tentative conclusions. However, further work is required in order to test properly and support these conclusions and to understand more fully the underlying phenomena.

Experimental

Studies at the University of Toronto

The rate and extent of iodine adsorption on stainless steel tubing was investigated for a range of gas phase concentrations, surface temperatures and relative humidities, in an apparatus at the University of Toronto (Figure 1).

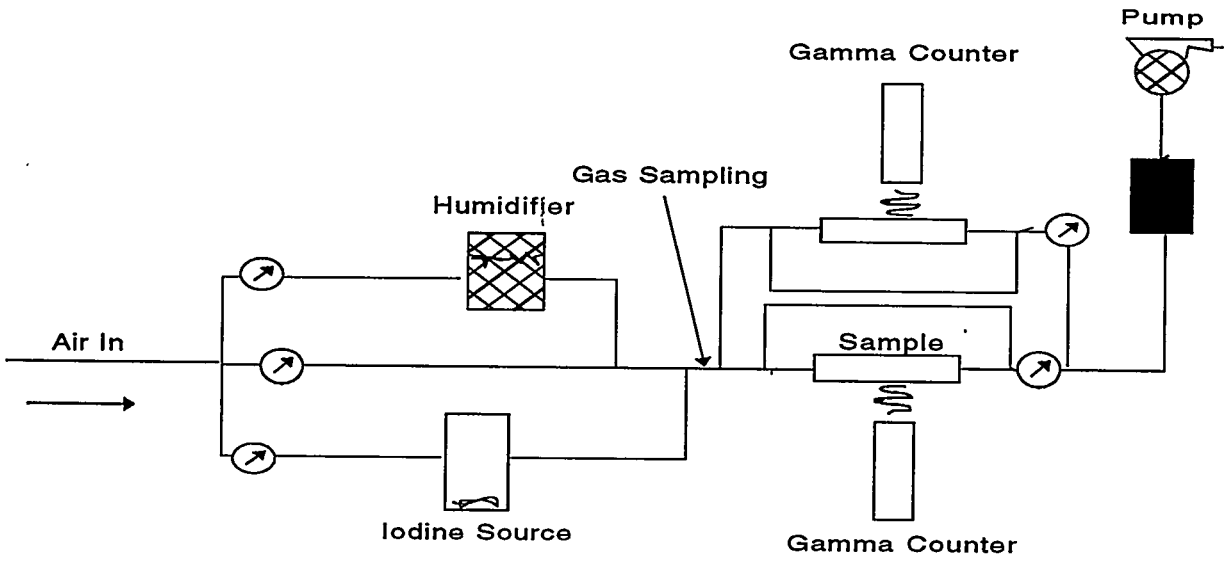


Figure 1: Apparatus used for the experiments performed at the University of Toronto

Laboratory air was drawn into the apparatus and divided into three streams. Most of the air passed through the first two streams, the second of which passed through a water bubbler before rejoining the first stream. Varying the relative flow of air in these two streams allowed the relative humidity of the air to be controlled. The relative humidity and temperature of the air was measured at the point where the two streams combined using a Cole Parmer™ Relative Humidity and Temperature probe.

The third stream was connected in parallel to a 350 mL Teflon vessel containing a solution of ^{131}I labelled I_2. The gas phase I_2 concentration could be controlled by varying the fraction of the third stream that passed through the vessel using a needle valve. The ^{131}I labelled I_2 solutions were prepared using two methods. For many of the experiments 5 mL 30% H_2O_2 and 1 mL H_2SO_4 was added to 1 mL of 0.5 mol/dm^3 CsI containing 7.4 to 37 MBq of ^{131}I. This produced a saturated I_2 solution with most of the I_2 formed precipitating. The premise was that as I_2 was lost from solution through volatilisation, it would be replenished through dissolution of the solid I_2. Unfortunately, the $I_2(g)$ concentration produced using this technique was found to drop substantially over the duration of the experiments. It is suspected that the rate of dissolution of the $I_2(s)$ varied due to differences in the size of the crystals.

In more recent experiments, an alternate method was used: An aqueous solution of saturated I_2 was added to a 2 mol/dm^3 NaI solution to produce a solution containing 1.6 mol/dm^3 I_3^- in equilibrium with I_2 and I^-. A small volume of the prepared stock solution, typically 15 to 20 mL, was then labelled with 37 to 74 MBq of ^{131}I. As the I_2 was removed it was instantaneously replenished through the rapid

equilibrium between I_2 and I_3^-. Gas phase I_2 concentrations produced using this technique were found to be more reliable.

The three streams were recombined and then split into two, each of which was passed through a section of ¼" ID type 316 stainless steel tubing, typically 15 cm in length. The tubing was trace heated and its temperature was set using a Cole Parmer DigiSense™ temperature controller. Duplication of the system allowed any variations between the sections of tubing to be observed. Also, this arrangement facilitated doing experiments at two different temperatures.

A 2x2 NaI scintillation detector was placed in close proximity (within 3 cm) of each of the tube sections. This allowed the accumulation of iodine on the tubing to be continuously monitored. The two detectors were coupled to Aptec™ multichannel analyzers (MCA) placed in an Intel™ based 80386-40 MHz personal computer. The ^{131}I activity on the tube surface was measured every 30 minutes. This counting interval was later decreased to 10 minutes. The counting efficiency was determined after every experiment by cutting the tube and counting the pieces in a well characterised LKB Compugamma™ system equipped with a 3x3 well type NaI detector.

The gas phase iodine concentration was determined using three methods. Usually a known volume of the air stream, upstream of the stainless steel tubing, was temporarily diverted through a short (5 cm) piece of Tygon tubing containing TEDA impregnated charcoal which was subsequently counted in the LKB Compugamma™ system. Air samples were also collected downstream of the stainless steel sections in order to ensure that the gas phase I_2 concentration did not vary significantly over the length of the sections. In some tests, tubing containing charcoal was present downstream of the steel sections throughout the test, in order to provide a mass balance. These results confirmed that only a small portion of the I_2 was retained by the stainless steel. Finally, in a few tests, one of the stainless steel sections was replaced by a tube containing charcoal to allow the gas phase iodine concentration to be monitored continuously.

Air flow rates in each of the three incoming streams was measured using variable area flowtubes as was the flow rate leaving both sections of stainless steel tubing. The flow though each piece of tubing was typically on the order of 1 - 3 L/min giving velocities on the order of 100 cm/s. This high flow was used to ensure rapid gas phase mass transfer within the system. The gas phase mass transfer coefficient was estimated based on a standard engineering correlation for flow through a tube[5] to be 1 - 2 cm/s. This value was confirmed in experiments involving tubing coated with an epoxy paint known to rapidly adsorb I_2.

A charcoal trap was used to decontaminate air leaving the apparatus, which was drawn using a GAST™ fluid pump placed at the end. Using a vacuum ensured that should any leakage occur it would be inward, thereby minimising the potential for release of radioiodine from the apparatus. All the tubing, fittings and valves used in the apparatus were made of Teflon™ and the entire apparatus was located in one of the fume hoods of a medium level radioisotope laboratory.

Studies at Whiteshell Laboratories

The adsorption, desorption experiments used 304-L stainless steel coupons with the dimensions of 1.25 cm in diameter by 0.32 cm in thickness. All the coupons were prewashed in 2-propanol and deionized water, and then air dried. Electro-polish samples were treated in an acid bath (H_2SO_4, H_3PO_4, H_2O) for 30 min. at 3 amps. and 4 volts, then washed in deionized water, acetone, methanol and dried at 700 °C for 30 min.

The experiments were carried out in a closed, circulating system. The system consisted of a 1 L Teflon vessel in which I_2 was generated by the addition of HCl and H_2O_2 to a solution of ^{131}I labelled CsI. The generated gas was circulated through Teflon lines and the sample chamber using a recirculating pump, capable of flows of between 1 to 5 L/min. The gas phase I_2 concentration was measured by trapping on TEDA Charcoal , which was then removed and counted.

After placing a coupon in the sample chamber, the chamber was valved into the circulating gas system. When the adsorption was complete, the coupon was removed and counted in an LKB 1282 Compugamma, then returned to the sample chamber. Ultra pure air was passed through the sample chamber and the desorption was followed using a NaI gamma detector. Following desorption, the coupon was removed and again counted in the LKB counter.

Results and Discussion

The Role of Surface Conditions and Chemistry

Scoping experiments were performed at Whiteshell Laboratories to investigate the impact of surface condition on the retention of iodine. The rate of adsorption and the amount of surface loading achieved was dependent upon the degree and type of oxidation present on the metal. Heating caused surface oxidation, and also extensive contamination of the surface with a thick layer containing carbon. Coupons that were heated in this manner with no subsequent polishing, absorbed relatively small quantities of iodine, likely due to protection by this layer (Table 1).

Table 1 Loading and removal of iodine on stainless steel coupons after various treatments

Treatment	$[I_2(g)]$ nmol/L	Loading (ng/cm^2)	After desorption (ng/cm^2)	After washing (ng/cm^2)
Heat (500°C) polished	3-7 >10	150-4000 10000	100-600 1500	12-40
Heat (500°C) unpolished	3-5	20-30	15-20	3 - 10
Acetone	2-4	200-1000	100-500	
Nitric Acid	2-5	35-100	30-50	10-15

In contrast, coupons heated with subsequent removal of the carbon layer by polishing, adsorbed large quantities of iodine. The loading observed was reasonably similar to that for a control sample that was only washed with acetone, or untreated stainless steel tubing (Table 2), suggesting that heating with subsequent polishing did not substantially affect the deposition of iodine. Treatment with nitric acid reduced the loading, possibly due to enrichment of the surface with chromium oxides. Similarly, electropolishing was found to greatly reduce the retention of iodine by the tubing.

Table 2: Loading and removal of iodine on stainless steel tubing after various treatments

Treatment	$[I_2(g)]$ nmol/L	Loading (ng/cm^2)	After desorption (ng/cm^2)	After washing (ng/cm^2)
untreated	0.3 3 20	45 9525 25400		
electropolished	2-8	11-33	10-30	0.5-1.5
nitric acid	2-8	18-50	14-36	2-31

Insight into the nature of the adsorbed iodine was provided by evaluating the removal of iodine due to desorption into air, washing with water and surface analysis using scanning electron microscopy (SEM) and X-ray photoelectron spectroscopy (XPS). A large portion of the iodine on the unpolished coupons, or those treated with acetone, could be removed by desorption into air. Analysis of the desorbed iodine using gas chromatography and species selective adsorbents[6] indicated that this iodine was predominantly in an inorganic form, likely I_2, rather than being present as an organic iodide. This suggests that for these coupons part of the iodine on the surface was physically (reversibly) bound I_2.

Water was found to be very effective in removing the iodine remaining on the surface following desorption into air. This suggested that the majority of the "chemically" bound iodine on the surface was in a water soluble form. Radiochemical solvent extraction of the wash solution [6] indicated that almost all (>95%) of the iodine removed was in the form of iodide. Furthermore, Fe and Mn were observed in the wash solution using Atomic Absorption Spectroscopy. These results suggest that the chemically bound iodine on the surface was predominantly in the form of metal iodides, possibly FeI_2 and MnI_2, consistent with observation of others. For example, Rosenberg et al[1] observed extensive removal of iodine, through washing with either water or an I_2 solution of identical specific activity. However, little iodine was lost through washing with FeI_2 solutions of identical specific activity, suggesting that the iodine on the surface was in the form of FeI_2. The lack of Cr in the wash is surprising given the large amount of Cr in stainless steel. However, Rosenberg et al found that although Fe readily adsorbed I_2, Cr did not, achieving no more than monolayer coverage. This finding is consistent with the lower retention of iodine observed following pretreatment with nitric acid in the present work.

Analysis of the coupons by SEM and XPS, indicated that iodine was retained in localised deposits, presumably oxide scale, whereas adjacent un-corroded regions contained no iodine. Fe and Cr were also detected in the corroded regions however no Mn was observed. The absence of Mn was surprising given that it was observed in the wash solutions.

The relationship between iodine deposition on steel and corrosion has long been established although the relevant mechanism remains uncertain. Recently Tsukaue et al have related corrosion to the adsorption of I_2 by water droplets on the surface, reaction with I^- to form I_3^-, oxidation of Fe to form more I^-, leading to further adsorption of I_2. Such a mechanism, involving corrosion due to I_2 or I_3^- reduction to I^- is consistent with the conclusion by Tyler that iodine is present as a metal iodide on the surface, based on binding energy measurements using XPS.

The Role of Operating Conditions

The impact on deposition of operating conditions such as the $I_2(g)$ concentration, relative humidity, and temperature of the steel was studied using the apparatus at the University of Toronto. These experiments were performed using 316 stainless steel with no prior treatment, as it was felt that this would be more representative of the tubing in the existing PARMS. To remove any loose contamination, air was blown through the tubing samples before use.

The lack of any pretreatment resulted in some variability in the behaviour observed under any given condition, likely due to variation in the initial condition of the surfaces. The use of the duplicated system, such that two sections of tubing could be investigated simultaneously, allowed this variability to easily be evaluated. In general, the duplicate tube samples agreed to within a factor of two. In the few cases there were larger discrepancies between the samples, and some runs had to be repeated or rejected.

Additional problems with reproducibility were caused by variations in the $I_2(g)$ concentration during experiments, which sometimes decreased by up to a factor of 10 over the course of a twenty hour experiment. Usually most of this decrease occurred during the initial hours of the test. This variability could be accounted for to some extent in calculations, using the numerous $I_2(g)$ measurements that were collected throughout an experiment. However, the variation in $I_2(g)$ caused uncertainty in the interpretation of some results. Consequently, the method for producing $^{131}I_2(g)$ was changed, as described in the experimental section, resulting in far more consistent $I_2(g)$ concentrations.

The variation between the duplicate samples was relatively small compared to that due to the parameters investigated. The errors caused by the variability in the gas phase was somewhat larger. Although these sources of error may have obscured some phenomena, many important trends were still clearly evident. For example, the results of two experiments performed with the same temperature (70°C) and similar gas phase concentrations but different relative humidities (6% and 60%) are shown in Figure 2. In both these tests, the gas phase iodine concentration decreased from $2x10^{-7}$ to $2x10^{-8}$ mol/dm^3 within the first four hours. The duplicate tubes differed by 10% at the low humidity and by up

to a factor of three at the higher humidity. Despite these variations, there was a clear difference between the loadings achieved at the two humidities which differed by a factor of 200.

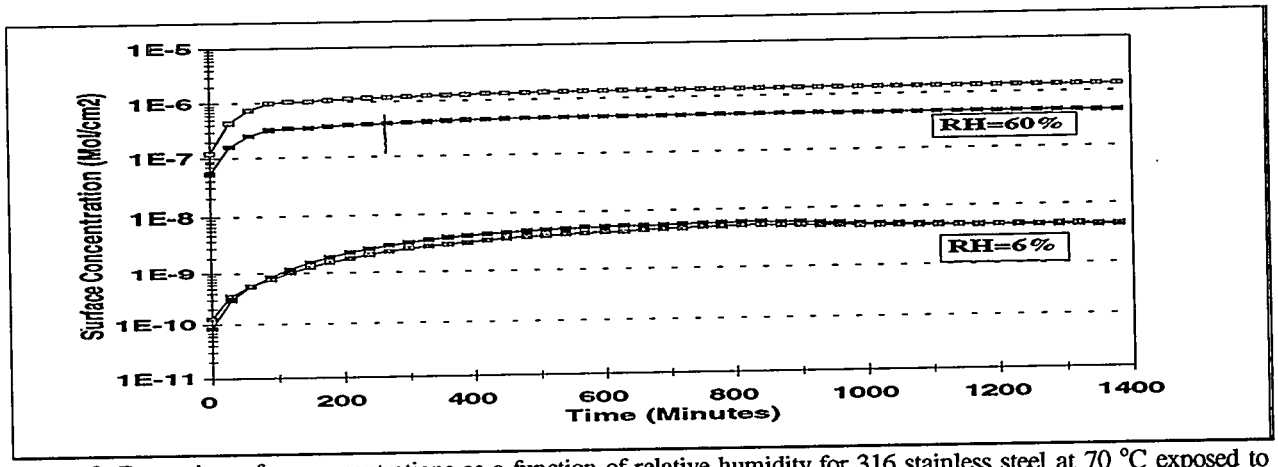


Figure 2: Dynamic surface concentrations as a function of relative humidity for 316 stainless steel at 70 °C exposed to $2x10^{-8}$ mol/dm^3 I_2(g)

In general, the surface loading increased with increasing relative humidity. The dependence on relative humidity was particularly strong at low RH values (Figure 3). It should be noted that this trend is in conflict with any suggestion that water and I_2 are competing for sites on the steel surface. However, the observed dependence on relative humidity could be considered consistent with I_2 deposition being related to corrosion of the steel. According to Rosenberg et. al.[1], a number of authors have reported that I_2 induced corrosion of steel is inhibited at low moisture contents. One of the authors referenced by Rosenberg et al proposed that at ambient temperatures, I_2 only corrodes iron at humidities above 30%.

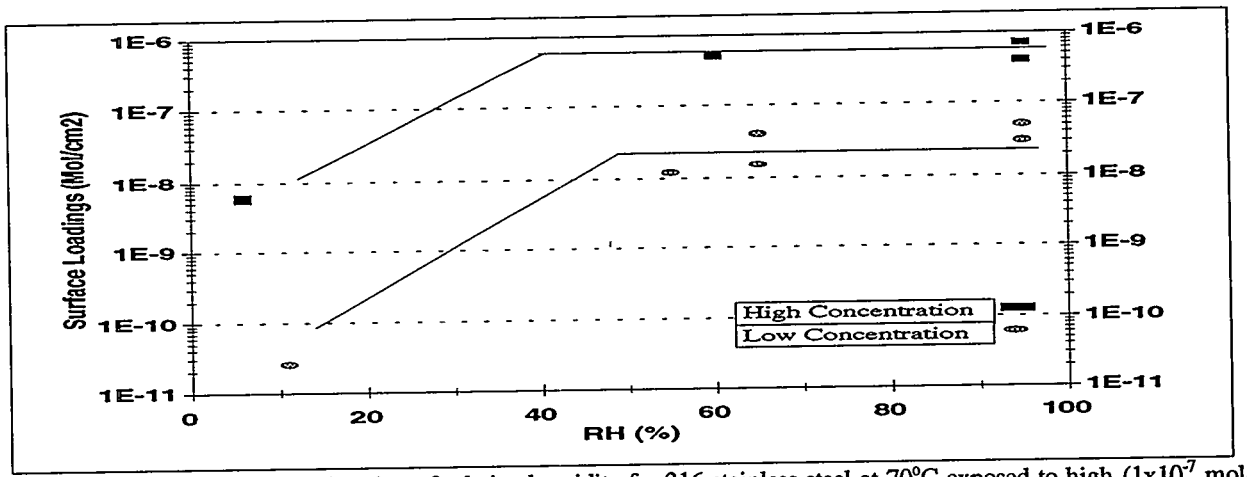


Figure 3: Surface loading as a function of relative humidity for 316 stainless steel at 70°C exposed to high ($1x10^{-7}$ mol/dm^3) and low ($1x10^{-9}$ mol/dm^3) concentrations of I_2(g)

Experiments were performed using I_2(g) concentrations ranging from 10^{-10} to 10^{-6} mol/dm^3. This range encompassed the low gas phase concentrations that might be expected following a reactor accident, the intermediate concentrations that have been used for in-plant testing of the PARMS and the relatively high concentrations that were used in previous studies[1, 4] The behaviour at low concentrations differed substantially from that at higher concentrations. In particular, at low I_2(g)

concentrations, the surface loading tended to increase throughout a test whereas at high concentrations what appeared to be asymptotes were observed (Figure 4).

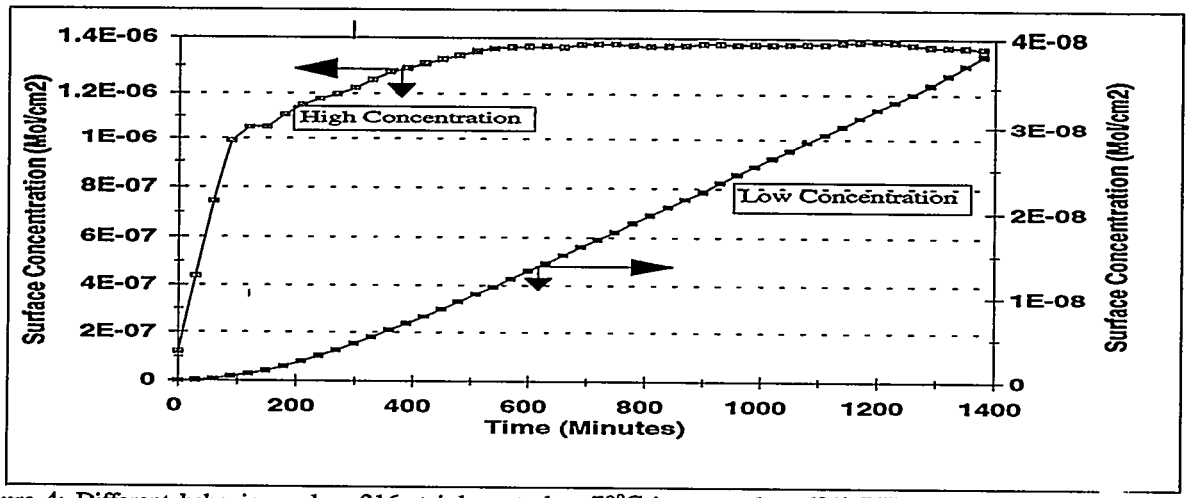


Figure 4: Different behaviour when 316 stainless steel at 70°C is exposed to 60% RH air containing a high ($2x10^{-7}$ mol/dm^3) or a low (10^{-9} mol/dm^3) concentration of I_2(g)

Gas phase concentration had a substantial impact on the initial rate of surface deposition (Figure 5). The rate depended on the gas phase concentration to the first power, giving support to the concept of the deposition velocity. In some experiments the deposition rate was initially quite slow but then suddenly increased by up to a factor 10. The reason for this change in deposition rate, which typically occurred during the early stages of a test, is unclear. Often this transition occurred once a surface loading on the order of 10^{-9} moles/cm^2 was achieved (Figure 4), suggesting some critical loading was required in order to initiate the corrosion mechanism. It should be noted that in their study on steel corrosion in triiodide solutions, Tsukaue et al[3] observed a delay between exposing a sample and the onset of corrosion which they accounted for using an incubation time for pit formation.

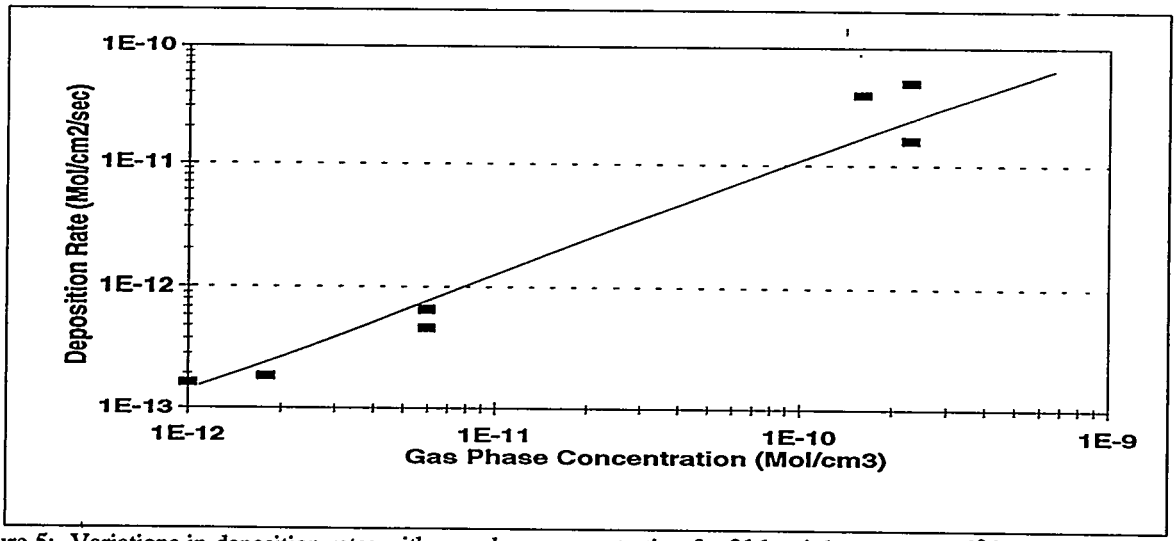


Figure 5: Variations in deposition rates with gas phase concentration for 316 stainless steel at 70°C exposed in air with 60% RH

The rate of iodine deposition decreased with increasing temperature (Figure 6), decreasing by a factor less than 5 over the temperature range for high gas phase concentrations, and for low gas phase concentrations by a factor of 100, finally dropping below detection at 90 °C. Deposition velocities, calculated from these deposition rates and the measured gas phase concentrations varied from 1 cm/s at 25°C to 0.1 cm/s at 70 °C for high gas phase concentrations. Clearly, other than at 25°C, the rate of deposition was slower than gas phase mass transfer (1- 2 cm/s) and hence was limited by reaction on the surface. This limitation may have been a combination of pore diffusion and chemical reaction.

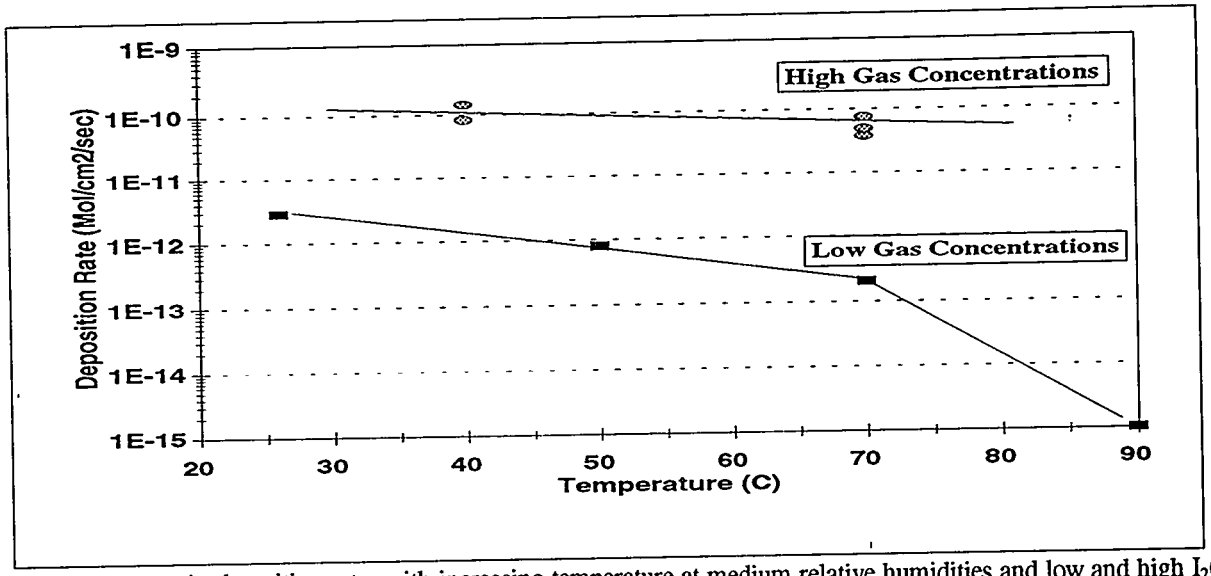


Figure 6: Decrease in deposition rates with increasing temperature at medium relative humidities and low and high I_2(g) concentrations of 10^{-9} mol/dm^3 and 10^{-7} mol/dm^3 respectively.

Rosenberg et al[1] observed a similar trend in experiments performed at temperatures above 100°C. However in the current work, the rate of deposition at 90°C was below detection and the implicit deposition velocity was far lower than that reported by Rosenberg et al at 115°C. One possible reason for this difference is that in the present work, the reported temperature was that of the stainless steel; the air was not heated and passed through the tubing so rapidly that it essentially remained at ambient temperature. In the study by Rosenberg et al, the air was heated rather than the steel. In theory, the loading of physically adsorbed I_2 on steel in contact with cooler air would be lower than that on steel in contact with air at the same temperature. Perhaps more importantly, the loading of surface water would be lower on steel in contact with cooler air.

Rosenberg et al[1] interpreted the temperature dependence in terms of an Arrhenius plot and thereby calculated an activation energy for the adsorption reaction. However, it should be noted that the temperature dependence observed here, and by Rosenberg et al is the opposite of that expected based on an Arrhenius type relationship for irreversible chemisorption; in theory, the reaction rate should increase not decrease with increasing temperature. Furthermore, the rates of gas phase mass transfer and diffusion within any pores on the surface would also be expected to rise with increasing temperature. Clearly, the steady state loading of I_2 on the surface at higher temperature would be expected to be lower given that adsorption is exothermic. However, this trend is usually attributed to more rapid desorption rather than slower adsorption. It is speculated based on the observed dependence on temperature and relative humidity, that localised water on the surface plays an

important role in the deposition process. Less surface water would be expected at lower relative humidity and higher temperature. Furthermore, if deposition was mostly occurring on localised areas of the surface, rather than in a homogeneous manner, the rate of deposition would be slower than that expected based on gas phase mass transfer. This would be the case even if physical adsorption as opposed to chemisorption was the main process.

Conclusions

This study is still at an intermediate stage however, some useful trends have been identified and tentative conclusions proposed. Experiments were performed using $I_2(g)$ concentrations ranging from 10^{-10} to 10^{-6} mol/dm^3 and the behaviour at low concentrations differed substantially from that at higher concentrations. In general, the rate of deposition was proportional to the gas phase concentration, giving support to the concept of a first order deposition velocity. The surface loading increased with increasing relative humidity, particularly at low RH values, while the deposition rate decreased with increasing temperature, in contradiction to the behaviour expected in theory. It is speculated that surface water may play an important role in the adsorption process.

The deposition of I_2 occurred through a combination of physical and chemical adsorption and the extent was dependent upon the initial surface condition. Pretreatment with nitric acid reduced the loading as did electropolishing. Examination of the steel surface following exposure to $I_2(g)$ revealed that the chemisorbed iodine was located primarily is areas of corrosion. Furthermore, water used to wash the steel contained Fe and Mn and iodine in the form of iodide, suggesting that I_2 reacted to form metal iodides.

Considerable further work is required in order to properly test and support these conclusions and to more fully understanding the underlying phenomena. However, it is already clear that extensive loss of $I_2(g)$ would likely occur in the PARMS. Furthermore, reliable correction for the extent of such loss is likely near impossible, given the large number of parameters that could have an impact. Future work will focus on modification of operating or surface conditions leading to reduced iodine deposition rates, in order to provide recommendations regarding methods for enhancing iodine recovery in the PARMS.

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Acknowledgements

The authors are grateful for the financial support provided by the CANDU Owners Group and the Natural Science and Engineering Research Council of Canada. The useful discussions with Clara Wren are also acknowledged.

DISCUSSION

WREN: I have a few comments. First, you said that the dissolution rate is higher than you expected from the mass transfer rate. That was based on a smooth surface area, but the stainless steel surface probably is not smooth, so the surface area is not quite the same as what you calculated. Thus, comparison of deposition rate vs. mass transfer rate is not very useful. I do not know how it affects the mass transfer calculation.

EVANS: I said the deposition rate was slower than the mass transfer rate. It is about 0.1-0.8 whereas gas phase mass transfer is 1-2 cm/s. The observed deposition velocity may exceed the mass transfer coefficient if the true surface area for deposition is greater than the geometric area. However, in this study the deposition rate was slower than that based on gas phase mass transfer indicating that gas phase mass transfer was not the limiting step or that the area on the tube surface involved in deposition was smaller than the geometric area.

WREN: The second comment is that your Arrhenius-type plot has a negative slope. If you are only looking at the adsorption rate, you should always have a positive Arrhenius plot (activation energy is always positive). The negative Arrhenius plot indicates that significant desorption occurs during your adsorption rate measurement, or that water adsorption-desorption has significant impact on iodine adsorption. Did you look at the the temperature effect at 0% RH on the Arrhenius type plot to see whether there is a different Arrhenius behavior?

EVANS: I think that is a good question. I do not think we have yet plotted the temperature dependence at very low relative humidity. The plot I showed you was for 60% RH. We observed that the deposition rate, not just the surface loading, decreased with increasing temperature. Based on an Arrhenius relationship for chemisorption or diffusion in the boundary layer, or pores for physisorption, the deposition rate would be expected to increase with increasing temperature. If, as suggested in the presentation, sorption involved water droplets on the tubing surface, then this trend could be due to

less water on the surface at higher temperatures. Looking at the temperature effect at low humidity would be a good way to investigate this hypothesis.

WREN: Did you perform the desorption of iodine following loading? That will tell what fraction of iodine is physically adsorbed vs. chemically adsorbed.

EVANS: Desorption was performed in some experiments. However, results have not yet been compiled so I can't comment at this time. Establishing what fraction of the iodine is physisorbed is certainly important.

WREN: That will tell you the fraction of adsorption due to physical and chemical adsorption?

EVANS: That is right.

GLISSMEYER: What size stainless steel tubing was used in the PARM system?

EVANS: The PARMS differs between stations but typically involved ¼" or ½" lines approximately 20 m in length.

GLISSMEYER: What kind of tubing are you going to use for the PARM system?

EVANS: The main objective of the work over the next year or so is to see whether we can recommend a way to get the existing system to work. Ontario Hydro has no desire to rip out the existing systems in three nuclear stations. It would be very nice to get the existing system to work at a level that we have confidence in. If it appears that this will not be possible, we will look at different types of tubing, and we have to look at how the tubing ages over time. We showed some possible treatments that would reduce deposition. However, we do not know whether the effects of such treatments are permanent. So I cannot answer your question.

GLISSMEYER: You might want to consider a larger diameter tubing, or separating the iodine sampling system from the particulate system and using teflon or polyethylene tubing for your iodine system.

EVANS: The apparatus we use is entirely constructed of teflon. I find teflon tubing is very good, but teflon fittings are a nightmare for adsorbing iodine. I really do not think teflon has the right properties. Glass would be very nice to use, except there's no way to get seismic qualification. These options should certainly be considered if we reach the point of recommending to Ontario Hydro that the existing lines be replaced. However, we would prefer to find a way to improve the performance of the existing lines. The experimental apparatus was constructed of teflon and, clearly, iodine is not retained significantly on teflon tubing. However, we have noticed that it is retained on teflon fittings and solid pieces of teflon.

SESSION 8

PANEL SESSION: GAS PROCESSING

Tuesday: July 26, 1994
Co-Chairmen: J. L. Kovach
J. Jacox

Panel Members: J. R. Kriskovich
T. M. Punch
R. F. Vance

OPENING COMMENTS OF SESSION CO-CHAIRMAN KOVACH

GAS PROCESSING IN THE NUCLEAR INDUSTRY
J. L. Kovach

APPLICATION OF NUCLEAR AIR CLEANING AND TREATMENT CODES
J. R. Kriskovich

GAS PROCESSING AT DOE NUCLEAR FACILITIES
J. Jacox

PANEL DISCUSSION

OPENING COMMENTS OF SESSION CO-CHAIRMAN KOVACH

The panel title is Gas Processing. Gas processing primarily takes place at DOE sites, rather than in power reactors, at least as far as safety related systems are concerned. One of the reasons we decided to have a panel session on this topic is to discuss some of the problems and differences in air and gas treatment systems as they are applied to nuclear processing facilities versus conventional HVAC-type facilities. Conventional facilities are fairly well standardized, although not in a manner to make them user friendly for many applications.

GAS PROCESSING IN THE NUCLEAR INDUSTRY

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NUCON International Inc.

The American Society of Mechanical Engineers (ASME) maintains the Committee on Nuclear Air and Gas Treatment (CONAGT) which developed the AG-1 Code (1). CONAGT also maintains the ASME N509 (2) and ASME N510 (3) Standards, relating to the construction and testing of nuclear air cleaning systems. The AG-1 Code sections and the above standards were developed for the safety related air cleaning and treatment systems of nuclear power reactors. In the past, the criteria for "safety related" were generally established on the basis of the US Nuclear Regulatory Commission (NRC) guidelines. However, there are major nuclear engineering projects which do not fall under the purview of the NRC and are based mainly at US Department of Energy facilities. Additionally, the nuclear fuel cycle's gas processing activity is broader than that for power reactors, and particularly outside the USA, it also includes fuel reprocessing also. At the present time, the currently written sections of the AG-1 Code cover mainly HVAC and low pressure air cleaning activities.

The USDOE only recently established a DOE standard for the "Evaluation Guidelines for Accident Analysis and Safety Structures, Systems, and Components." (4) The evaluation guidelines established by the USDOE are somewhat different from the USNRC criteria for safety related items. This divergence also causes problems in strictly applying the AG-1 Code sections for other than NRC (or similar agency) regulated power reactor sites.

The USDOE classification for Safety Class Structures, Systems, and Components (SCSSCs) and Safety Significant Structures, Systems, and Components (SSSSCs) has a different definition both in accident levels and worker safety protection than the conventional USNRC criteria, the former being much broader than those established by the USNRC.

Thus two major problems face the applicability or expendability of the ASME AG-1 Code to areas where the major design and construction activity is taking place. One is the clarification of the appropriate applicability of the safety classification, the other is the current narrow scope of the component sections of the AG-1 and the specificity of the N509 and N510 standards to power reactor NRC dictated systems. At the same time the USDOE has endorsed the use of the two standards and is considering the endorsement of the AG-1 Code for applicability to USDOE sites.

The solution to the safety classification problem is more administrative than an engineering issue, and probably the ASME Board of Nuclear Codes and Standards needs to act for resolution of the issue. However, the current disharmony between existing standards and the lack of component sections specific to areas other than power reactors, in the AG-1, is an extensive and time consuming engineering problem.

At the same time, there are numerous military application uses of air treatment and gas processing, particularly in chemical agent demobilization, where some sections of the existing nuclear codes and standards are used. However, the existing codes and standards do not exactly match the nuclear power air cleaning applications either. At some point it will be important both from an availability and cost standpoint to use common consensus standards, rather than customizing for each specific application. Thus the possibility of including other than nuclear safety related structures, systems and components also needs to be considered.

The typical non-power reactor nuclear application, as an example, include air cleaning systems for the Tank Waste Remediation Systems (TWRS) project, which relate to the treatment of both non-standard air filtration systems and components which are not defined in any of the code sections.

Typically, the power reactor safety grade use would occur only for limited periods, i.e. after an accident, while for waste processing application air and gas cleaning, the design life has to be years under high input load conditions. Continuously operating systems, mainly installed for worker protection in the nuclear power reactor industry are not considered safety grade.

Typical differences in air and gas composition in USDOE applications are:

1) The presence of ammonia, hydrogen, and nitrogen oxides.

2) The possibility of supersaturated corrosive environments.

3) The normal presence of alpha, beta and gamma activity.

4) The potential presence of Transuranic isotopes.

5) Intrusive activity in the stored waste, generating high particulate loadings.

6) Intrusive activity potentially generating high liquid droplet levels.

7) Elevated temperature operation (from melters).

8) The potentially very high levels of nitric oxides from the decomposing nitrate salts from the melters.

9) The presence of radioactive nuclides other than iodine. (Ru, Tc, etc.)

10) Extensive condensation in the process of air cleaning

Typical components for DOE processing applications that are currently not covered in AG-1 code sections:

Scrubbers
Metallic media filters
Irrigated filters
Incineratable filters
Adsorbents for volatile isotopes
Adsorbents for mercury
Nitrogen oxide reduction catalysts
Nonflammable adsorbents
Cooling and heating components in the operating temperature ranges.
In-place regenerable adsorbents
Other than slab (Type II) or in-place fillable (Type III) adsorbers.
Instrumentation for specific processing conditions (larger number of remote readout and control).
Thermal and chemical resistance of components (gaskets, seals, damper seals, etc.) beyond the power reactor accident conditions.

Typical systems currently not considered by N-509 and N-510 standards.

Remote handling ability for component changes.
Personnel protection for alpha contaminated components.
The continuous manner of system operation.
Maintainability of highly contaminated systems.
Thermal fatigue of system and components.
Corrosion resistance of systems and components.
Instrumentation for other than HVAC type operation.
In-place testability of highly contaminated systems.
Decontamination and final disposal of components and systems.

The above tabulations are examples and can not be considered all inclusive. There are probably numerous others that need to be considered both for the applicability of the existing codes and standards and for any possible future standardization work.

One of the major problems, even in the nuclear power related standard and code writing effort, was the simultaneous development of the standards and codes and the actual design and construction of the components. This resulted in the installation of many systems which do not meet the current standards and codes either for the actual design criteria or the testability requirements, because the testing standards were developed tied to specific design requirements. At the same time, it is often claimed, that a particular system can not be tested according to a specific test conditions, while the truth is that it can be tested, but it would fail the originally presumed acceptance criteria.

Many of the processes in the DOE field of air and gas cleaning application are already in the conceptual stage and if the standardization process is going to be applied to them, it is imperative that the standard writing activity starts immediately for the currently expected components (many of which are available currently in commercial grade).

The design and standardization of systems is a much more complicated problem, because the total number of systems in many cases, but not always, will be limited, therefore the effort of standard writing has to be commensurate with the need. This is an area, where more generic "guides" rather than cast-in-stone codes or standards are required.

Regardless of the methodology used, participation from the organizations involved in the conceptual and detail design of the systems and the manufacturers of the currently "commercial" grade systems or components is very important, otherwise the writing of standards and codes will not include all of those who are intimately familiar with the specific requirements of the new applications.

Conclusions and Recommendations.

It would be desirable that the ASME Board of Nuclear Codes and Standards review the problems of the application of the AG-1 code for non-power reactor conditions and advise CONAGT of direction. It is also desirable that the USDOE reviews the AG-1 Code and the N509 and N510 Standards for the problems of applicability to different uses as an example the TWRS.

It is also worthwhile to establish whether a specific nuclear power reactor air and gas processing code and standard is needed and affordable, or the potential preparation of a broader code and standard which could include military, biohazard, pharmaceutical, etc. applications should replace the existing highly specific code sections and standards.

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APPLICATION OF NUCLEAR AIR CLEANING AND TREATMENT CODES

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ABSTRACT

All modifications to existing ventilation systems, as well as any new ventilation systems used on the Hanford Site are required to meet both American Society of Mechanical Engineers (ASME) codes N509 and N510. Difficulties encountered when applying code N509 at the Hanford Site include the composition of the ventilation air stream and requirements related to ventilation equipment procurement. Also, the existing ventilation systems for the waste tanks at the Hanford Site cannot be tested in accordance with code N510 because of the current configuration of these systems.

I. INTRODUCTION

The Hanford Site is a 1,424-km^2 (550-mi^2) Department of Energy Environmental clean-up site. The site originally was constructed in the early 1940's and it continued operation throughout the cold war. The Hanford Site's mission was to produce nuclear materials for the war effort. In the process, millions of liters of liquid waste products were generated. The highly radioactive wastes were pumped into underground tanks. These tanks were to be used as temporary storage. However, to this day the waste remains. There are currently a total of 177 underground waste storage tanks in the Tank Farms on the Hanford Site. They consist of 149 Single Shell Tanks (SSTs) and 28 Double Shell Tanks (DSTs), ranging in size between 208,000 L (55,000 gal) to 4,390,000 L (1,160,000 gal) on the Hanford Site.

As the cold war ended, the Hanford Site's mission changed from producing nuclear material to environmental restoration. The center of this effort includes the waste stored in the 177 waste tanks. To ensure the safety of facility workers as well as the offsite population, the waste in the tanks is continuously monitored. This includes monitoring the liquid level, temperature, and pressure to ensure these parameters remain within prescribed limits. To help maintain the waste (as well as the tanks) within these limits, some of the tanks are actively or force ventilated.

To date, all 28 DSTs are actively ventilated as are 18 of the 149 SSTs. A typical ventilation system for these tanks consists of duct work coming from the top of the tank, which then enters a ventilation train. In this typical system, the ventilation train consists of a de-entrainer, a heater, a pre-filter, 2 stages of HEPA filtration, a centrifugal fan, an exhaust stack, and a stack sampler. Two of the main reasons for actively ventilating these tanks are to maintain confinement (i.e., negative pressure with respect to atmosphere) and to help cool the waste.

The remaining 131 SSTs that are not actively ventilated are passively ventilated. A typical passive ventilation system consists of a HEPA filter in a filter housing connected to the top of the tank by a riser made from commercial pipe. Rather than having forced ventilation through the tank, as in the case of the tanks that are actively ventilated, the tank breathes through the HEPA filter because of changes in atmospheric pressure.

All modifications to the systems mentioned above, as well as any new ventilation systems designed and fabricated for use at the Hanford Site, are required by the Department of Energy (DOE 1989 and DOE-RL 1989) to meet ASME code N509, *Nuclear Power Plant Air-Cleaning Units and Components*, and ASME code N510, *Testing of Nuclear Air Treatment Systems*. The difficulties in applying the ASME codes to the ventilation systems at the Hanford Site are the topic of the following discussions.

II. DIFFICULTIES APPLYING ASME N509

Some of the difficulties encountered when applying code N509 at the Hanford Site are related to the composition of the ventilation stream and requirements related to ventilation equipment procurement.

The exhaust stream from the waste tanks contains not only radioactive particles, but in some instances gases in excess of explosive limits (e.g, hydrogen) or toxic vapors. For those tanks that contain hydrogen gas, it is very difficult to meet code N509 in its entirety when making modifications to the existing system or designing a new system.

In Section 5.5 of ASME code N509 it is stated that, "Heaters shall be electric." Currently, for those waste tanks that are actively ventilated, steam heaters are used in most applications where hydrogen is present in the exhaust stream. If an electric heater were to be used in these applications, it would need to be intrinsically safe. As a result, the heater would be physically very large, limited in availability, and very expensive. Therefore, in this situation it is more appropriate to use a steam heater.

Although a steam heater has the potential to leak, it is not expected to have an adverse effect on a typical Tank Farm ventilation system. A likely result of a steam leak would be an increased differential pressure across the HEPA filters. The ventilation system automatically shuts down on high differential pressure across the filters well within the 25 cm (10 in.) w.g. rating for the filters.

Cleaning or decontaminating the ventilation equipment after a steam leak could be difficult because the steam leak would be located upstream of or before the HEPA filters. As a result, this area of the ventilation train would be contaminated by radiation. However, clean up of that area may be a more acceptable risk than an explosion that could be caused by an electric heater.

Procuring fans that meet code N509 for the ventilation systems is very difficult. Under paragraph 5.7.2, "Rating or Test," code N509 states the following, for engineered safety features (ESF) and non-ESF fans:

> *"ESF fans shall be tested in accordance with AMCA 210 and the applicable special sections of AMCA 211A...Non-ESF fans shall be either rated and listed in accordance with AMCA 211A or tested the same as ESF fans."*

One interpretation is that the requirements for non-ESF fans are the same as ESF fans. The reason is because any fan rated and listed in accordance with AMCA code 211A (1985) must bear the AMCA Seal because AMCA code 211A outlines the AMCA Certified Ratings Program and use of the AMCA Seal. Furthermore, for ESF fans, code N509 seems to imply that they must bear the AMCA Seal because the special sections in AMCA code 211A read as if the Certified Ratings Program is being used. Furthermore, please note that AMCA code 211A is an incorrect reference to an outdated publication. The current publication is AMCA code 211-87. In addition, WHC has asked ASME for a formal technical inquiry into this issue.

This, by itself, does not pose a difficult problem for procuring a new fan because there are numerous vendors who are AMCA approved. However, imposing this requirement in conjunction with additional code N509 requirements makes it difficult to locate qualified suppliers. One such additional requirement is NQA-1.

The Quality Assurance requirements of code N509 are stated in section 8, as follows:

> *"The design organization, manufacturers of components, and constructors (including subcontractors) shall each establish and comply with a comprehensive quality assurance program and plan which meets the requirements of ANSI/ASME NQA-1-1986."*

One of the problems is the interpretation of this section. Is this section implying vendors are acceptable, even though the vendor may not be NQA-1 certified, as long as they have a quality assurance program in place that is similar to NQA-1? Or, are they required to have an NQA-1 program in place and be certified by ASME?

If, in fact, the statement from Section 8 is interpreted to mean that a vendor needs to be NQA-1 certified, it is very difficult to locate a vendor that has obtained that certification.

III. DIFFICULTIES APPLYING N510

Another problem encountered with all the existing ventilation systems for the waste tanks at the Hanford Site is that the systems cannot be tested in accordance with code N510. Although DOE directives state the ventilation systems at the Hanford Site are to be tested to code N510, the existing systems cannot and are not required to meet this constraint. This is because the existing systems were not designed or fabricated to code N509. However, code N510 is used as guidance because it provides a basis and a method for performing the applicable tests.

Code N510 lists several tests or inspections that are required to be completed on the ventilation systems. The tests discussed herein will be the in-place leak test for HEPA filters, as well as the test for airflow distribution through HEPA filter banks.

Section 10.5 of code N510 explains the procedure for performing the in-place leak test. In this section it is a requirement to check the background (i.e., any stray particles in the air stream that may effect the test) and the concentration of the aerosol being used in the test upstream of the filters. On a few of the ventilation systems for the waste tanks at the Hanford Site it is impossible to verify the concentration upstream. Injection and sample ports are either located in the wrong locations or do not exist. As a result, the 100 percent upstream concentration sample reading is in question because it can not be taken at the appropriate location.

Several of the ventilation systems cannot be aerosol tested properly between filter stages. The housings that were designed and installed were not equipped with aerosol injection ports or sample ports. Consequently, rather than testing the second filter in the stage independently, an overall test of the system is completed to verify integrity.

The airflow distribution through the HEPA filters is also a troublesome area for some of the systems ventilating the waste tanks at the Hanford Site. These systems do not have gradual transition pieces into and out of the filter housings. Without these transition pieces it is difficult to determine if the aerosol is properly challenging the filter face and edge seals as required according to code N510, paragraph 8.6.2. As a result, the seal around the filter may in fact be breached or degraded. Since the aerosol is not able to challenge this particular area, detecting a leak is virtually impossible.

IV. CONCLUSION

There are several problems encountered in trying to comply with ASME codes N509 and N510 for the ventilation systems for the waste tanks at the Hanford Site. Problems range from interpretation of these two codes to design and field testing deficiencies. These deficiencies sometimes make it very difficult to apply code N509 and N510 to the ventilation systems for the waste tanks at the Hanford Site.

Actions are being taken to resolve these problems. New systems that will alleviate the problem with testing are being designed and fabricated. These systems, as well as modifications to existing systems are designed to meet both code N509 and N510 wherever practical. In some instances waivers to DOE and/or DOE Richland Field Office (DOE 1989 and DOE-RL 1989) are, or will be written where it is determined and justified to do so and if it does not jeopardize the safety of the facility worker, the off-site population, or

equipment. And finally, DOE requirements, as well as ASME codes are being evaluated at the Hanford Site to determine if clarification or revisions to these requirements are necessary.

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23rd DOE/NRC/ISNATT NUCLEAR AIR CLEANING CONFERENCE

GAS PROCESSING AT DOE NUCLEAR FACILITIES

Jack Jacox
JACOX ASSOCIATES

Abstract

The term "Gas Processing" has many possible meanings and understandings. In this paper, and panel, we will be using it to generally mean the treatment of gas by methods other than those common to HVAC and Nuclear Air Treatment. This is only a working guideline not a rigorous definition. Whether a rigorous definition is desirable, or even possible is a question for some other forum. Here we will be discussing the practical aspects of what "Gas Processing" includes and how existing Codes, Standards and industry experience can, and should, apply to DOE and NRC Licensed facilities.

A major impediment to use of the best engineering and technology in many nuclear facilities is the administrative mandate that only systems and equipment that meet specified "nuclear" documents are permissible. This paper will highlight some of the limitations created by this approach.

Introduction

Since the 1960s, handbooks, Standards, Codes and other formal documents have been written and then given the force of law by their use in facility Licenses or by other contractual or government mandates to specify many aspects of air-cleaning and treatment components and systems. The "American Association for Contamination Control" (The AACC is now defunct. Their Standards have been taken over by the Institute of Environmental Sciences, 940 East Northwest Highway, Mount Prospect, IL 60056.) published a tentative HEPA filter standard "CS-1T, HEPA Filter Units" in 1968 and a tentative radioiodine adsorber "CS-8T, high Efficiency Gas-Phase Adsorber Cells" in 1972. The best source for information of this technical area was, and still is, these International Nuclear Air Cleaning Conferences. They have been held since the early 1950s. The first major general work on components and systems available to the public was the handbook "Design, Construction, and Testing of High Efficiency Air Filtration Systems For Nuclear Applications" (1). With very few exceptions these documents have been limited to HEPA filter systems and radioiodine carbon filter system. Control Room systems include minor expansion in scope to remove chlorine but the scope has been narrow. Some closely related components such as prefilters, moisture separators, electrical heaters, ducts, housings and frames have been historically included. However until the advent of ASME AG-1 (1) in 1988 there has been little effort to broaden this scope. Of course AG-1 had been under development for nearly a decade before its initial release. Still the original scope of AG-1 as given to the American Society of Mechanical Engineers (ASME) Code "Committee on Nuclear Air and Gas Treatment" in January of 1976 was to develop Codes and Standards for Light-Water Cooled commercial power plants. Their first assignment was to revise ASME N509-76 "Nuclear Power Plant Air Cleaning Units and Components" (3) and ASME N510-75 "Testing of Nuclear Air Treatment Systems" (4) which had been written by ad hoc committees which disbanded upon issue of the original documents.

The American Nuclear Society has written Standards on a slightly broader scope of treatment systems but their mandate is to basically define Process Standards, not to write equipment Standards. The ASME and ANS Standards are intended to be complimentary since ASME writes equipment Standards but not Process. Other than Xe and Kr control for Light-Water Cooled commercial power plants this has been essentially the limit of formal Industry Codes and Standards in the air and gas treatment area. Such important technical challenges such as incineration exhaust, hot cell exhaust, tank venting, reprocessing and general chemical processing gas streams have not been addressed. With the ever increasing pressure to reduce hazardous emissions and clean-up existing nuclear sites of all types new Codes and Standards are clearly required as quickly as possible. Equally importantly, the proper use of existing Codes, Standards and related documents is critical. Unfortunately in many cases the existing documents have been, and are being, badly misused.

Discussion

In recent years Codes and Standards written for Light-Water Cooled power reactors have been applied to a variety of new applications. This has been the result of both mandated requirements from the government as well as "unofficial" and well intentioned attempts to use existing documents, rather than writing new duplicative ones. Unfortunately in most cases this excellent idea has resulted in more problems than positive results. This particular problem area is covered in other papers and a panel in this conference so will not be discussed here. The thrust of this paper is to point out the immense engineering, technical and experience base for many, if not all, the Gas Processing challenges now requiring solutions in the nuclear area.

Along with the good technical solutions formal Codes and Standards are also being administratively required. Helpful Codes and Standards are not possible until the engineering/technical solutions have been at least somewhat developed and standardized. Unfortunately various regulatory and other administrative organizations have been trying to mandate the use of Codes and Standards before the technical solutions have been developed. This forces the engineering staff in an extremely difficult, of not impossible, situation.

Few current "nuclear" Gas Processing applications are truly unique. With very few exceptions they have been solved in industrial applications long ago. Certainly such things as criticality concerns are unique, but few Gas Processing applications have a criticality aspect. Radiation fields are also unique but not too difficult to handle with shielding and some engineering fore-thought. Radioactive contamination in most Gas Processing situations is little different than contamination from chemical or biological sources. These can be extremely dangerous and even deadly but are regularly dealt with by industry. It is instructive to recall that one of the major breakthroughs in the separation of Uranium isotopes at Oak Ridge was not a "nuclear" problem but the large scale industrial handling of pure Fluorine. Few compounds or biologicals are as deadly or difficult to work with as Fluorine, whether or not it is radioactive.

The engineering for cleaning the exhaust from the storage tanks at Hanford has been largely worked out by the chemical industry for many years. Addition of more conservative design margins, seismic requirements, material traceability, use of "Qualified" welders, more comprehensive QC/QA and such requirements is not overly difficult (But certainly very expensive). Rather than mandating the use of existing "nuclear" Codes and Standards that were written for extremely different and usually narrow applications the better approach is to use well proven industry experience then add the "nuclear" safety margins as applicable. This can be done in a reasonable and cost effective manner if common sense and good judgement is used, rather than blind adherence to administrative mandates. It is in everyone's best self interest to accomplish a project in the best technical and most economic way possible. This fact can be lost by overly rigid application of and Code or Standard. In nearly ever case the project will be complete in far less time and for far less money with the "industry approach"

Every industry tends to believe it has a totally unique set of problems. Many have a few unique aspects but most are the same as other industries, they just have a different set of labels and regulations. Unfortunately regulations tend to reinforce this aura of uniqueness and foster a mind-set that restricts the use of "outside" technology. The nuclear industry has a particularly bad case of this malady. This is partly from the truly unique aspects that do exist, partly from the requirement for security classification in many areas and from the total regulation of the industry from it's creation. Breaking out of this limiting mind-set is necessary to solve the current and future gas processing problems we face.

Conclusions

The conclusions of this paper and thesis are as obvious as they are difficult to achieve.

1. To approach the solution to technical challenges from an open minded engineering point of view, rather starting with regulations, Codes or Standards.

2. Use the vast industrial experience that exists to help solve a problem. Usually it has already been solved in some industrial application.

3. Use the existing "nuclear" Codes and Standards knowledgeably as technical guides when the situation is not exactly the one the documents were written. Do not try to torture a technical problem to fit a Code or Standard!

4. Regulatory bodies and organizations that administer regulations must understand the above points and not stifle good engineering solutions to technical problems with overly rigid and inflexible enforcement of the letter of a document when it is technically inapplicable.

5. Use Codes and Standards from other industries, modified if and as necessary, for new nuclear applications.

6. New Codes and Standards must be written - or existing ones expanded - to cover these new and broader areas or general nuclear gas processing.

7. DOE facilities must support these new documents by supporting their personnel to join the committees that write the Codes and Standards.

8. DOE facilities must be more open in discussion of their challenges and share both problems and solutions with the rest of the nuclear industry and general industry to the extent national security allows. The formation of open industry groups as the nuclear power industry has so successfully promoted is a possible means to help accomplish these goal.

9. As with all industries do not be blinded by the idea you are totally unique. Use the work, mistakes and successes of others to solve your problems.

These goal are easy to state but notoriously difficult to implement since they require significant changes in mind-set and the culture of a large and highly structured industry. The fact that it is totally regulated makes the implementation much more difficult since some regulations and regulator's outlook must also change. However, to solve the gas processing problems change is inevitable.

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PANEL DISCUSSION

JACOX: When you are writing codes and standards, you are not allowed to put into them the kinds of explanations you have discussed. I am glad that you were able to find some very innovative uses for the documents. I will point out that they were written for ESF systems in light water nuclear reactors. You are absolutely right, some of the things you have to do are not covered, and that is where flexibility, rather than an administrative law approach, is necessary. When we write these standards, we rail against some of the bureaucratic rules that we have to follow. In some cases, we have put in non-mandatory appendices because they offer a great deal more freedom than is allowed in the body of a code and standard. However, we did not do it in the particular area you discussed. In retrospect, you are right; that section does need to be fleshed out. At the time it was prepared, we were looking strictly at accident mitigation and safe shut down systems. None of the other considerations were addressed.

VANCE: I would just like to say that I appreciate that we all have constraints.

KOVACH, L.: To comment further on the testing problems of two HEPA filters in series, I recall the early days of the nuclear power reactor industry and the first time people had a system with two HEPAs and a carbon bed inbetween. To test the system, they actually started out by taking out the front bank and testing the back bank, then putting back the front bank and taking out the back bank to test the front one. Finally, they put back the rear bank. It was extremely lucrative for testing companies. Then people realized that an injection manifold or a test manifold would be more helpful than continually taking out and putting back filters. To take out a bank of filters to test a second bank is eminently feasible on a system that is in a standby mode, waiting for a potential accident. But if you contaminate the filter banks, the worst thing you can do is to take them out and put them back inside the filter housing. The criteria, as written, and as applied, are different. It is only in the last issue of N-509 that the use of manifolds for testing multiple filters is discussed in order to prevent this type of a problem.

I would like to add that we have personnel here from CONAGT, from the committee chairman to a number of individual sub-committee chairmen, and if any of you are interested in contributing to better codes and standards, please volunteer to participate. I feel that it is much more useful to put a little time in up front to generate better codes and standards, written in an engineering manner, than it is to spend money and time later on either fighting them or trying to live with them. Currently, we are in the reactive mode of taking an existing standard or code and spending a lot of time and effort fighting it or trying to go around it, while it would have taken much less time to participate originally and make sure that the problems that are different in different areas are in fact covered logically in the standards.

At the same time, I would like to state that I do not believe that there ever will be a code or a standard that is applicable for all cases, there is no such thing. It is utopian to think that we can write a standard for all possible future applications. I do not think anyone should look on this code as a substitute for engineers. Unfortunately, I think some management or administrative levels, in fact, do look on these standards and codes as a substitute for engineers and inhouse technical knowledge. We cannot do

anything about it in the code section itself, but maybe we can make a statement that will filter back to management and the regulatory agencies that codes and standards are not substitutes for common sense and good engineering judgement.

MYERS: I do not have a problem with what you are presenting, but I am wondering whether, in addition to attending to needs for other applications and information, there is going to be an effort to relax some of these standards. I recall that Dr. Bellamy indicated last year that some plants could apply for removal of certain components. Is it going to be up to each licensee to justify their own design? Or will there be statements in the standard to say that when you have a carbon bed there is no need for a following HEPA filter? Can you comment on that?

KOVACH, L.: I am not going to speak for the NRC. They have accused me several times of doing so and I do not want to fall into that trap again. I know of some reactors where the application of Regulatory Guide 1.52 and the ASME/ANSI code sections were relaxed. I can list a few, TMI-2, one in Long Island, one in Portland. I am not sure if this is the type of relaxation you are looking for. As far as a HEPA filter following a carbon bank is concerned, I think it was written to be super-conservative. The carbon particle size has never been at a level that requires a HEPA filter to remove it. The idea behind a second HEPA filter was that if carbon picks up radioiodine and then breaks off, it should not pass through the system. To my knowledge, several utilities requested that their second HEPA filter bank not be treated as a second HEPA bank and therefore, there would not be a need to test it. My eminent colleague, Jack Hayes, who I think participated in one of these jurisdictional disputes, is standing at the microphone and I am going to let him speak officially for the NRC.

HAYES: In the guidance presented in the NRC's Standard Review Plan (SRP) only SRP 15.7.4 states that an ESF filter system should be provided. In all other cases, such systems are only required if acceptable data cannot be obtained. Operating plants can provide revised accident analyses to the NRC to justify the elimination of ESF filters.

With respect to the comment on the need for a HEPA filter downstream of the charcoal adsorber, the NRC has approved the use of a post-filter which is not a HEPA grade filter for the Advanced Boiling Water Reactor (ABWR). The NRC would consider requests for other applications, also. This is consistent with the guidance in ASME Standard N-509-1989.

JACOX: The 1989 edition of N-509 explicitly recognizes this problem. I believe the wording states there is an afterfilter because there is a potential for carbon dusting. A HEPA filter may not be the right type of filter, so the standard designates it as an afterfilter, which can be a medium efficiency filter that does not require testing.

ARNDT: The only thing I would ask is that if anyone has experience in procuring materials, such as housings, dampers, or fans, for systems, and has either written justification for not meeting NQA-1, N-509, etc., I would love to talk with them just to determine if I might apply it at Hanford, as well. I am here to get information that will help me save as much money as possible in my area.

KOVACH, L.: I would rather get people together to see how they can meet the standard, rather than get people together to see how they can get around it. However, when there is something wrong in the standard or its application, there are prescribed enquiry methods to get clarification. The outcome may be that the standard writing group will review the questioned section and be willing to change it. An enquiry can result in an interpretation of a code or a standard; it can also result in a modification of a standard or code. Spending time to correct standards is more important than finding out how to get around them. Somebody mentioned to me that they considered N-509 and N-510 to be a bible. My answer was to point out how different religions interpret the bible in different ways. I would not have you look on the standards as a bible, but as engineering advice. They contained a number of mistakes and that is why you see revisions. The people writing them realize that they need to be changed and improved. Users are justified in making recommendations when they encounter problems and demanding technical justification for their preparation and modification. Look on them as living documents that can be changed and help make sure that poor engineering is removed from the codes and standards.

VANCE: A closing remark about the value of codes and standards: the availability of these codes and standards helps to provide us with the ability to do our jobs because they provide legitimacy and acceptance by others. When we can show through the code process that what we have done makes good engineering judgement, it becomes easier to convince people that what we are doing is sound. They provide a tool to get our jobs done. Without basic standards, engineering would be much more difficult. We just want them to be the best possible codes and standards.

KOVACH, L.: Most of us have used a hammer and hit our fingers but that is not a good reason for throwing the hammer away. I think it is how we apply codes and standards, how we use them, and how we interpret their application that is important.

SESSION 9

ADSORPTION

Tuesday: July 26, 1994
Co-Chairmen: P. Sigli
J. L. Kovach

RESIDENCE TIME DETERMINATION FOR ADSORBENT BEDS OF DIFFERENT CONFIGURATIONS
J. E. Otermat, W. O. Wikoff, J. L. Kovach

HALIDE TEST AGENT REPLACEMENT STUDY
E. M. Banks, W. P. Freeman, B. J. Kovach, J. L. Kovach, R. R. Sommer

RESIDENCE TIME DETERMINATION FOR ADSORBENT BEDS OF DIFFERENT CONFIGURATIONS

J.E. Otermat
W.O. Wikoff
J.L. Kovach
NUCON International Inc.

Abstract:

The residence time calculations of ASME AG-1 Code, Section FC, currently specify a screen surface area method, that is technically incorrect. Test data has been obtained on Type II adsorber trays of different configurations to establish residence time in the adsorber trays. These data indicate that the air volume/carbon volume ratio or the average screen area are more appropriate for the calculation of the residence time calculation than the currently used, smallest screen area basis.

Introduction:

The original adsorber residence time calculations of ASME/ANSI N-509 Standard were based on total screen area. The subsequently issued AG-1, Section FC, Mandatory Appendix I-1000, "Residence Time" calculations added new wording, changing from total screen area to an arbitrary value of the "gross screen area of all screens on inlet side or outlet side, whichever is smaller". This new definition, introduces a bias that favors adsorbers which are strictly parallel bed designs. Adsorbent beds, which do not conform to the parallel bed design, such as wrap around designs are penalized by the wording used. The easiest way to visualize the difference in residence time calculation is, by analogy with concentric adsorber beds, where the internal bed diameter is always smaller than the outer diameter. The current, major quantity military application adsorber, as an example, uses such configuration. The correct method of calculating residence time, for adsorbers with different inlet and outlet screen areas, is to use the average bed cross section and not the "smallest" screen area based flow path. The same thing is true for the wrap-around adsorber bed designs, discussed in Section FC of AG-1. In adsorption type unit operations, it is important to use strictly geometric average cross section for calculation, because the rate controlling steps of the adsorption process are also velocity dependent.

Test data has been generated to evaluate several Type II adsorber designs to establish actual residence time and to compare the adsorption based behavior with geometric configurations.

Experimental Methods.

Standard Type II adsorber cells of ASME/ANSI 509 cross section requirements (identical to the AG-1 Section FC cross section dimensions), but of different lengths and internal construction were filled with the same Lot/Batch adsorbent and tested in a wind tunnel at 333 SCFM. The pressure drop and the penetration of R-11 vapor were measured. The tests were conducted at ambient temperature and pressure in a wind tunnel where relative humidity and temperature were adjusted to result in uniform flow. The R-11 challenge vapor was introduced at a 1000 vppm inlet concentration and the outlet concentration was continuously monitored. The test was terminated when the adsorber tray outlet concentration reached 250 vppm R-11. The inlet and outlet concentrations were monitored by open diode type ionization detectors. The carbon fillable volume of the adsorbent beds was calculated both based on the physical measurements of the adsorber trays and from the adsorbent density and weight used to fill the adsorbent beds. All flow and R-11 penetration measurements were performed at 333 SCFM.

The adsorbent Lot/Batch used was in a moist condition, 19,0 wt. % water content, to increase the transport rate of the challenge gas through the adsorbent trays.

After filling, the adsorber trays were exposed to the ambient atmosphere for 30 minutes, then placed in the wind tunnel, where the 333 SCFM clean air was passed through the trays for 30 minutes prior to the initiation of R-11 challenge gas injection.

The breakthrough curves are shown on Figure No 1.

Test Results. Table 1.

	26 inch tray		30 inch tray	
	wrap around	parallel	wrap around	parallel
Adsorbent wt. lbs.	72.4	65.6	81.5	70.1
Pressure drop wg."	1.22	0.98	1.03	0.85
Cell volume cu.in.	3102	2770	3496	3019
Carbon volume cu.in. based on ASTM AD	3119	2826	3511	3020
Time to 100 vppm outlet conc. min.	5.75	5.00	7.18	6.77
Time to 250 vppm outlet conc. min.	12.52	11.43	15.75	14.43

Evaluation Results. Table 2

	26 inch tray		30 inch tray	
	wrap around	parallel	wrap around	parallel
Residence time based on AG-1,sec.	0.20	0.22	0.23	0.25
Residence time based on air/carbon volume, sec.	0.32	0.29	0.37	0.31
Residence time based on avg. screen area, sec.	0.24	0.22	0.27	0.25
Carbon weight/AG-1 residence time	362.0	298.2	354.3	280.4
Carbon weight/Air/ Carb.Volume res.time	226.3	226.2	220.3	226.1

Evaluation Results. Table 2 (Continued)

	26 inch tray		30 inch tray	
	wrap around	parallel	wrap around	parallel
Carbon weight/Avg. Screen Area res. time	301.7	298.2	301.9	280.4
Ratio of time to 250 vppm/ carb wt.	0.173	0.174	0.193	0.205
Ratio of time to 250 vppm/AG-1 calc. residence time	62.6	52.0	68.5	57.7
Ratio of time to 250 vppm/Avg Screen area res. time	53.1	52.0	58.3	57.7
Ratio of time to 250 vppm/ carbon vol. residence time	39.1	39.4	42.6	46.5

The data indicate that the residence time as derived from AG-1 Section FC Mandatory Appendix gives a misleading nominal residence time value which is biased toward parallel beds design. Either of the other two methods, i.e. the air to carbon volume or the average screen area (less baffles, etc.) gives a more correct actual residence time for any adsorber design, than the formula stated in AG-1, Section FC, Appendix I-1000.

Conclusions:

It is critical that formulae included in Code sections be based on sound technical information, and not favor any particular design style, as long as the other relevant parameters of the Code are met. It is equally important that changes and revisions of the Code sections are evaluated according to existing technical justification and supporting data and not on an arbitrary basis.

DISCUSSION

WREN: Did you look at the leakage rate as a function of flow rate? What kind of dependence do you expect?

KOVACH, L.: Leak rates were looked at under identical flow conditions. Our interest was mainly to determine whether different test agents, under the same conditions, gave an identical leak. Therefore, we used the standard conditions that are prescribed for this, i.e., 40 fpm through the carbon bed.

WREN: It is my understanding that *in situ* tests do not necessarily use the same flow rate that is expected during an accident.

KOVACH, L.: I agree with that, because a leak is basically an orifice. Therefore, if you use a different flow rate, you will have a different pressure drop, and that would mean a different leak quantity.

WREN: Do you want to speculate about what kind of dependence you would expect?

KOVACH, L.: No, but I think it would not be difficult, based on the standard orifice equation, to predict what the pressure differential would be based on an equivalent leak rate.

WREN: In your methyl iodide removal efficiency, what fraction was not trapped due to leakage, and what fraction due to penetration?

KOVACH, L.: The methyl iodine test was performed in a laboratory test rig, so there was no leakage. Leaks occur only in the full-sized leak equivalency test.

PEARSON: In your paper, you described the method of detection for the leak test, and you mentioned a gas chromatograph equipped with an electron capture detector. You stated that the gas chromatographic column's purpose is to separate oxygen from the tracer gas. The real purpose of a chromatographic column is to separate the tracer gas from other interferences, such as Refrigerants-11, -112, -114, and -13B1, which are commonly found in varying amounts, hour to hour, minute to minute, in nuclear power plants. You also described an open diode detector. My question to you is, what method does the open diode detector use to separate the interfering substances commonly found during filter leak testing?

KOVACH, L.: We are using the installed carbon bed in the same manner that the chromatographic column is used.

PEARSON: When Refrigerant-11 is adsorbed, it is very likely that you will desorb Refrigerant-12 or -22, which can be mistaken for a leak, when, in fact, the material has nothing to do with a leak.

KOVACH, L.: I have not seen a case like that in any industrial application when using either instrument. There has been no problem determining whether there was a leak or not. I think tests that show that you get identical results when running tests, one after another, using different test agents are a good indication of that.

PEARSON: Equipment manufacturers have suggested that anyone using HCFC-123 in air conditioning equipment should install detection monitors. When running leak tests using HCFC-123, would you need detection equipment in the area to make sure that the recommended TLV is not exceeded? How would you go about assuring compliance?

KOVACH, L.: The Federal Register describes very clearly what is required and under what conditions. I can tell you that the answer is, no, you do not need to install any other detectors than the one you are using for the leak test with HCFC-123.

PEARSON: Even when you are filling your generator, or getting your test equipment ready?

KOVACH, L.: That is correct.

BELLAMY: A lot of chemical names appeared on your slide. What compound are you using today for in-place testing?

KOVACH, L.: R-11.

BELLAMY: How long do you expect R-11 to be available?

KOVACH, L.: The reason we are using R-11 currently is because you cannot use a substitute according to the code. Until there is a change in the code, you will have to use R-11.

BELLAMY: And the cost is basically immaterial, it just has to get passed along to somebody.

KOVACH, L.: At the present time, I do not think anybody has changed the price for testing based on the increased price of R-11.

BELLAMY: If I can take the liberty, I would like to ask an NCS representative what compound they are using today?

PEARSON: R-11

BELLAMY: So, everybody is still using R-11 compound simply because it is a regulatory requirement and that is what we will have to use until the code case is approved?

KOVACH, L.: That is correct. One of the questions that may not have come up directly is why we are still looking at other test agents. It is because EPA's regulations relating to the use of some of the alternative compounds is also changing. EPA is publishing, under the Stratospheric Protection Division's Questions and Answers on Ozone Depleting Refrigerants and Their Substitutes, almost all of the various CFC applications. Concerning one of the presumed inert materials, perfluorocarbons, the EPA statement is as follows: "CFCs may only be used where no other agent is technically feasible, due to the performance or safety requirements. The user must conduct an evaluation of the other alternatives and must determine that they either will not perform properly, or that they will pose a risk to human health. In addition, CFC use must meet conditions discussed in..." and they refer to other sections. EPA's requirement is to look at test agents based on perfluorocarbons and show that you tested everything and found nothing suitable before they will agree to using some other product. The published data on perfluorocarbon reactions with carbon, that decomposition of the perfluorocarbon occurs, made it not worthwhile to look at perfluorocarbons as potential alternatives. At some point we may look at them again, if we have to show that we tested everything else. We would rather start looking at those that may be suitable, rather than proving there are none. This is the reason we are not conducting work on perfluorocarbons at the present time.

HALIDE TEST AGENT REPLACEMENT STUDY

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Abstract

The intended phaseout of the chlorofluorocarbons (CFCs) from commercial use required the evaluation of substitute materials for the testing for leak paths through both individual adsorbers and installed adsorbent banks. The American Society of Mechanical Engineers (ASME) Committee on Nuclear Air and Gas Treatment (CONAGT) is in charge of maintaining the standards and codes specifying adsorbent leak test methods for the nuclear safety related air cleaning systems. The currently published standards and codes cite the use of R-11, R-12 and R-112 for leak path test agents. All of these compounds are CFCs. There are other agencies and organizations (USDOE, USDOD and USNRC) also specifying testing for leak paths or in some cases for special life tests using the above compounds. The CONAGT has recently developed criteria for the suitability evaluation of substitute test agents. On the basis of these criteria, several compounds were evaluated for their acceptability as adsorbent bed leak and life test agents.

The ASME CONAGT Test Agent Qualification Criteria.

The test agent qualification is based on the following parameters:

1) Similar retention times on activated carbons at the same concentration levels as one of the following: R-11, R-12, R-112 or R-112a.

2) Similar lower detection limit sensitivity and precision in the concentration range of use as R-11, R-12, R-112 and R-112a.

3) Gives the same in-place leak test results as R-11, R-12, R-112 or R-112a.

4) Chemical and radiological stability under the use conditions.

5) Causes no degradation of the carbon and its impregnant or of the other NATS components under the use conditions.

6) Is listed in the USEPA Toxic Substances Control Act (TSCA) inventory for commercial use.

Current Test Agent Use Information.

The utilization of CFCs has been pioneered by the USDOD at Edgewood Arsenal, to determine the presence of leak paths and bed packing uniformity for adsorbent beds intended for chemical agent protection. The development of radiological vapor protection adsorbers for the USAEC Savannah River site production reactors lead to the application of this methodology for nuclear safety related adsorbers. Initially these adsorbers operated at an approximately 75 FPM velocity through a one inch deep carbon bed. The evaluation of the various test agents is reported in regard to the suitability of the test agent under these conditions. (1)(2)(3)(4)(5)(6). On the basis of the original studies R-112 was found to be suitable for the leak testing of both the shop testing of individual adsorbers and the leak testing of the installed adsorber banks. R-112a was found to be an acceptable alternative agent to R-112. The other lower molecular weight and lower boiling point compounds were found to be unsuitable, mainly for the in-place tests, due to the fast penetration of the adsorbent bed at the high ambient temperature and humidity conditions of the site.

The identification of the presence of organic iodine compounds and the commensurate lower removal efficiency and capacity by the conventionally used adsorbents, led to the increase in adsorbent bed depth to a minimum of two inches and the operation at 40 FPM velocity through the adsorbent beds. Under these design and operating conditions, some of the smaller molecular weight and lower boiling point CFCs were suitable for the determination of the absence of the leak paths in both the individual adsorbers and the installed adsorber banks. The industrial practice became using R-12 for the leak testing of individual adsorbers, because when filled with new adsorbent and tested under controlled environmental conditions the adsorptive retention time was sufficiently long to enable the differentiation between a leak and the penetration of the adsorption wave front through the adsorbent bed. At the same time, it was relatively easy to purge off the R-12 used in the adsorber leak test and prevent potential interference with the subsequent in-place leak test. The R-12 was not suitable for the performance of the in-place leak test, because after the build-up of common organic compounds in the installed adsorbent beds and their equilibration with the environmental moisture content, the penetration of the R-12 was too fast. Therefore a slightly larger molecular weight and higher boiling point compound, R-11, was selected as the in-place leak test agent. There was no specific development work performed anywhere for the selection of the R-12 and R-11 test agents. The selection was based on the available data from the original Savannah River site studies. However, the standard writing organizations included the specification for these two compounds on a "grandfathering" manner, without further evaluation. (7)(8)

When the perceived concern about the survival of the CFCs in the upper atmosphere became legislated, the continued use of the CFCs for leak testing of adsorbent beds became a concern of the users, standard writing organizations, testing agencies and test equipment manufacturers. At the present time, the USEPA policy is that the use of CFCs in the nuclear industry does not violate the prohibition of the intentional CFC release rules. However, the phase out of the manufacture of CFCs and the yearly increasing excise tax costs for the still manufactured CFCs, is an important driver for the evaluation of suitable replacement agents.

Basically, the real technical criterion for a leak test agent is the detectability limit of the compound. The lower that limit, the less of the particular compound is used for leak testing. At the same time, it is important that the test agent is not strongly adsorbed on the particular adsorbent, otherwise the test becomes a destructive test. Because the current standards and codes specify a maximum leak rate of 0.05 % for the installed adsorbent banks, the required operating range for the test agent is in the 10,000 to 1 concentration range, thus the test instruments have to be able to detect the test agent in the same concentration range.

Currently, there are two typical detection systems in use. One is an electron capture detector utilizing radioactive ionization in the detector element and the separation of the oxygen peak from the test agent peak by the use of a chromatographic column. The test consist of extracting sub- milliliter quantities of sample from the air cleaning unit's air stream and injecting it into the column of the chromatograph. The other is an open diode detector, which is capable of detecting the presence of the test agent above the oxygen response on a continuous basis at sample flow rates of hundreds of ml per minute. The sensitivity of both detectors is in the ppb range for the conventional CFCs used as test agents and it is important that the same sensitivity range is maintained for the substitute test agents. There is also an economic reason for the utilization of the currently manufactured test equipment with the new test agents.

Alternative Test Agent Review.

In several countries, alternate test agents are being used even at the present time, because the US developed regulations and standards do not apply. The most commonly used in the non US nuclear safety related leak test application is non-radioactive methyl iodide or a mixture of radioactive and non-radioactive methyl iodide. While the detectability limit is satisfactory for
methyl iodide by both commonly used detector systems and the penetration (retention) of the methyl iodide is not too different from that of the CFCs used in the past, the utilization of methyl iodide is complicated by the chemical hazards associated with it.

It is currently forbidden to air ship methyl iodide and it is a known carcinogen. The use of combination leak and efficiency test, using radioactive methyl iodide, is also dependent on the collection and detection limits of the upstream and downstream sampling cartridges, and the potential hazard of using radioactive compounds to determine the efficiency of the installed, but prior to testing, unknown efficiency air cleaning unit is questionable. The use of radioactive methyl iodide is also a health physics problem in US facilities due to current regulations.

Several NATO related agencies have selected 1-bromobutane as a test agent, after also evaluating chloroethane, iodomethane, chloroform, tert-butylchloride, chlorobenzene as candidate test agents and hexane-1-ol as a potential "poisoning" agent, (9) (10). The detectability limit for the 1-bromobutane is similar to that of the test agents used in the past. In the process of this test agent evaluation, an extensive study was also conducted determining the adsorptive and ambient environment effects on a parametric basis, (11).

NUCON has performed extensive testing with both HCFC-123 (a chlorofluorohydrocarbon) and 1-bromobutane as candidate materials. The behavior of the HCFC-123 is near identical in properties and leak test performance to the R-11. The 2-bromobutane is between the R-112 and the R-11 in chemical properties and the same as the R-11 and R-112 in leak test performance.

NCS Inc. has reported on the use of perflourocompounds as potential test agents, (12). The perflourocompounds are easily detected by electron capture detectors, but less easily by open diode detectors.

The NCS proposed optimum material, Dodecafluorodimethylcyclobutane (PDCB) was claimed as similar in leak test performance to that of R-11. (12)

Future Regulation Considerations.

The USEPA, while currently not prohibiting the use of CFCs for leak testing is discouraging its use for any application by the phase-out of manufacture and the high excise tax applied. It is also expected, that barring additional technical evidence relating to the behavior of HCFC materials, their use will also be restricted in future years. The USEPA ruled for many current applications of replacement for R-11, the use of HCFC-123 as "acceptable". (13) It is important to keep in mind, at least for US applications, that the USEPA ruled that "It is illegal to replace an ozone depleting chemical with a substitute listed by the Significant New Alternatives Policy (SNAP) as "unacceptable".

Recently, the EPA has advised of restrictions on the use of perflourocompounds (PFC) for other than essential use, " PFCs may only be used where no other agent is technically feasible due to performance or safety requirements. The user must conduct an evaluation of the other alternatives and must determine that they either will not perform properly or that they will pose risk to human health". The restriction is based on the long term environmental survival of perflourocarbons in the environment, (14). Thus, while multi-halogenated compounds are ideal from the detectability standpoint, the use of chlorofluoro, hydrogenchlorofluoro or perflouro compounds does have some future use regulation uncertainty.

Potential Test Agent Properties.

TEST AGENT	BP C	MW	TSCA
Chloromethane****	-24.2	50.49	yes
R-12***	-29.8	120.92	yes
Chloroethane*	12.2	64.52	yes
R-11***	23.7	137.38	yes
HCFC-123	27.6	140.92	yes
Iodomethane*	42.5	141.95	yes
PDCB**	45.0	300.6	yes
t-Butyl chloride*	51.0	92.57	yes
Chloroform*	61.3	119.39	yes
Bromochloromethane****	68.0	129.39	yes
PMCH**	76.0	350.07	yes
n-Butyl chloride*	78.0	92.57	yes
R-112***	92.8	203.80	yes
1-Bromobutane*	101.6	137.03	yes
Chlorobenzene*	132.0	112.56	yes

* Additional evaluation in Ref 11.
** Additional evaluation in Ref 12.
*** Additional evaluation in Refs 1 through 6.
**** Currently under evaluation at NUCON.

Adsorption Considerations for Test Agents.

There are different considerations for the leak test agent, when it is used on new adsorbent beds which have low moisture content and no "poisoning" contamination and for "in-place" leak testing agents, which need to differentiate between a mechanical leak and the penetration of the test agent through the adsorbent bed in installed systems, which may have unknown quantities of adsorbed moisture and other contaminants.

According to the work of Smith(11), the adsorbent moisture content interference effect becomes strong when the boiling point of the adsorbate is below 85 C. Thus, if a compound is selected for the determination of the mechanical leak, under most "as is" condition,installed adsorber banks, it is advantageous to use compounds with a high boiling point. In this case, there is a longer residence time of the test agent on the adsorber bed, however, the quantity of test agent used does not need to be at a level that would significantly influence the radioiodine control properties of the adsorbent, even after repeated testing. If the pulse mode leak test technique is applied, then even less test agent is used.

For the in-place test agent, it is also preferable, if the test agent is not identical to compounds commonly present in the plant environment, where the adsorbent banks are installed. This parameter was not considered originally, when the common refrigerant compounds were proposed and standardized into leak test methods. It would also be advantageous if the same test agent could be used for both nuclear safety system and military chemical agent protection adsorber banks, containing different adsorbents and various adsorbent bed depths, operated under varying face velocities.

The quality control testing on the freshly filled or refilled adsorbers, tested under controlled conditions, does not have the above discussed, high boiling point test agent, requirement.

For quality control testing of individual adsorbent beds, it is advantageous to use a test agent, that is easily stripped from the adsorbent, so no potential interference would exist when the installed adsorber bank is tested. However, it is possible to use the same test agent for both applications, if the quantity used is small (for example, as in pulse mode leak testing).

Equivalency of In-place Leak Test Results.

A 6000 CFM full NATS installed at NUCON International Inc. has been used to evaluate the equivalency of the in-place leak test results using the different test agents on the full size systems. One of the sampling cartridges has been removed to simulate the leak. The total airflow, temperature and relative humidity were identical for the test series at 27 C, 55% RH.
The leak percentages determined using the NUCON F-1000 test agent generators and detectors were as follows:

	In-place Leak Test Results	
Test Agent	Continuous Test	Pulse Test
R-11	0.26%	0.27%
HCFC-123	0.25%	0.26%
1-bromobutane	0.26%	0.27%

The inlet concentration, for each of the test series, was either a continuous ~15 vppm, or three successive 6.0 ml pulses. The downstream detector was always connected to the duct downstream of the blower, to assure good mixing. The injection point was verified according to ASME/ANSI N-510 requirements.

The lower detectability limits for the evaluated compounds using the F-1000 instruments were as follows:

R-11	0.75 vppb
HCFC-123	1.00 vppb
1-bromobutane	1.00 vppb
Bromochloromethane	0.75 vppb

It is expected that other currently used test agent detector instruments have similar detection sensitivity levels.

The change in radioactive methyl iodide penetration of adsorbent beds (NUCON NUSORB KITEG II, Lot 55, Batch 54), when exposed to 0.5% by weight of each of the above test agents, which corresponds to approximately 500 in-place leak tests total test agent load, was insignificant. (Within the error limit of the tests.)

Unexposed	99.66% efficiency at 30 C, 95% RH,
R-11 exposed	99.60%
HCFC-123 exposed	99.55%
1-bromobutane exposed	99.30%

It is expected that longer than 2 hours of sweep time would displace more of the test agents and the co-loading effect would even be less if the NRC required air sweeps are performed on a monthly basis on the NATS. It is estimated that an adsorbent bed installed in a NATs may see approximately 50 tests in it's life.

So the above test was conducted at 10 times higher loading than the expected normal test agent load.

In connection with the evaluation of organic compound poisoning of adsorbent beds, a detailed study was made of the quantity of organic compounds that would result in a significant effect on the radioactive methyl iodide removal efficiency of the adsorbent.

While, for that particular case, not the most stringent test conditions were used, the data indicated that higher than 1 wt. % loadings of organic compounds are required to show differences in the adsorbent performance.(15)

A technical evaluation of the potential effect of the proposed test agents on other NATS components was based on the effects of potential dissolution of HEPA filter potting and sealant compounds and the potential detrimental effect on the commonly used gasket materials, neoprene and silicone rubber. In the quantities used there were no measurable effects.

The retention on ASME/ANSI N-509, 1980 grade activated carbon for the test agents was performed under identical process conditions of 40 FPM, 30 C and preequilibrated at the appropriate %RH, on a two inch deep bed. The results of a 5 minute, 20 vppm test agent penetration test through the adsorbent bed were:

Test Agent	Penetration time to 10% breakthrough on Preequilibrated Adsorbent Minutes	
	At 95 %RH	At 70 %RH
R-11	1.3	3.1
HCFC-123	1.9	4.3
1-bromobutane	120 +	120 +
R-112	120 +	120 +

So far, only screening tests were performed with chloromethane, as a potential substitute for R-12, in the quality control tests on freshly filled adsorbers. The preliminary data indicates, that the substitution will be probably acceptable, however additional data needs to be generated.

Conclusions from the Test Program.

On the basis of the evaluation of the properties and the performance of actual in-place leak tests and of the other relevant parameters of utilization of the evaluated test agents, it was found that either HCFC-123 or 1-bromobutane are acceptable substitutes for the R-11 in-place leak test agent.

Based on the very extensive testing at Porton Down (11), the 1-bromobutane is not affected by the expected in situ moisture content of the adsorbent or the potentially existing "poisoning" compound levels in either nuclear or military applications.

None of the test agents evaluated so far, were found to adversely effect either the adsorbent performance or the downstream NATS components at the loading levels expected in NATs in-place leak testing.

It is expected from the screening tests performed with electron capture detectors, that other manufacturer's test instruments would have similar detector sensitivity as that of the NUCON instruments.

Additional compounds are being tested to further optimize test agents under the CONAGT criteria. Of these bromochloromethane is currently in the evaluation stage for replacement of R-11 and chloromethane for the replacement of R-12.

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SESSION 10/11

PROCESSING AND FUEL REPROCESSING

Tuesday: July 26, 1994
Co-Chairmen: J. G. Wilhelm
M. Kikuchi

A STUDY ON THE EXPULSION OF IODINE FROM SPENT-FUEL SOLUTIONS
T. Sakurai, A. Takahashi, N. Ishikawa, Y. Komaki, M. Ohnuki, K. Kato

THE DEVELOPMENT AND DESIGN OF THE OFF-GAS TREATMENT SYSTEM FOR THE THERMAL OXIDE REPROCESSING PLANT (THORP) AT SELLAFIELD
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CLOSING COMMENTS OF SESSION CO-CHAIRMAN WILHELM

CLOSING COMMENTS OF SESSION CO-CHAIRMAN KIKUCHI

A STUDY ON THE EXPULSION OF IODINE FROM SPENT-FUEL SOLUTIONS

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Abstract

During dissolution of spent nuclear fuels, some radioiodine remains in spent-fuel solutions. Its expulsion to dissolver off-gas is important to minimize iodine escape to the environment. In our current work, the iodine remaining in spent-fuel solutions varied from 0 to 10% after dissolution of spent PWR-fuel specimens (approximately 3 g each). The amount remaining probably was dependent upon the dissolution time required. The cause is ascribable to the increased nitrous acid concentration that results from NOx generated during dissolution. The presence of nitrous acid was confirmed spectrophotometrically in an $NO-HNO_3$ system at 100°C. Experiments examining NOx concentration versus the quantity of iodine in a simulated spent-fuel solution indicate that iodine (I^-) in spent fuels is subjected to the following three reactions: (1) oxidation into I_2 by nitric acid, (2) oxidation into I_2 by nitrous acid arising from NOx, and (3) formation of colloidal iodine (AgI, PdI_2), the major iodine species in a spent-fuel solution. Reaction (2) competes with reaction (3) to control the quantity of iodine remaining in solution. The following two-step expulsion process to remove iodine from a spent-fuel solution was derived from these experiments:

Step One - Heat spent-fuel solutions without NOx sparging. When aged colloidal iodine is present, an excess amount of iodate should be added to the solution.

Step Two - Sparge the fuel solution with NOx while heating.

Effect of this new method was confirmed by use of a spent PWR-fuel solution.

I. Introduction

Radioiodine (^{129}I) freed during dissolution of spent nuclear fuels in reprocessing is prone to escape to the environment. Its strict control is required for environmental safety. High-efficiency silver-impregnated adsorbents recently have been developed that provide sufficiently high decontamination factors for radioiodine in off-gas.[1] The next consideration is the treatment of iodine remaining in the fuel solution. Untreated, it distributes itself into the Purex process where it causes an increase of radioiodine released to the environment. Thus, as much iodine as possible must be expelled from the fuel solution to the dissolver off-gas where it will be fixed on the high-efficiency adsorbent. Our current work is concerned with this expulsion of iodine from fuel solutions.

Earlier workers postulated iodate (IO_3^-) to be the main iodine species remaining in spent-fuel solutions. NOx sparging of the solutions was proposed to change the iodate into volatile iodine (I_2).[2] However, recent research provides a different understanding. Most remaining iodine may be colloids of insoluble iodides of silver

and palladium[3] or nonvolatile organic iodides.[4] Moreover, our recent research has found that, although the amount of fission-product iodine increases with increasing burnup of nuclear fuels, its amount is not necessarily proportional to the burnup.[3] Thus, there are ambiguities about the behavior and chemical form of iodine in the dissolution step. Our current research determined factors that influence the iodine species and their quantities in spent-fuel solutions. This allowed the derivation of a new method for effective expulsion of iodine from these solutions.

II. Experimental Method

Spent PWR-fuel specimens with burnups between 21 and 39 GWd/t (approximately 3 g each) were dissolved in 30 mℓ of 4 M HNO_3 at 100°C. The volatilized iodine was carried onto an iodine filter by bubbling helium (40 mℓ/min) through the solution. Details of the experimental apparatus and procedure were described elsewhere.[5] Dissolution progress was monitored by continuous measurement of ^{85}Kr radioactivity in the dissolver off-gas. After the insoluble residue had been removed by centrifugation, the expulsion of iodine from the solution was also performed using this experimental apparatus.

The simulated spent-fuel solution used had the same composition as that described previously,[6] that is, uranium of 50 g-U/ℓ and amounts of simulated fission-products corresponding to those in spent fuels with a burnup of 40 GWd/t. A corresponding amount of iodine (as potassium iodide) labeled with ^{131}I was added to 30 mℓ samples of the solution at 100°C with and without an NO or NO_2 supply. The iodine species in the solutions were analyzed by the carbon tetrachloride extraction method that determined the quantities of I_2, I^-, IO_3^-, organic iodide, and colloidal iodine present.[7] Except the nitrogen dioxide, purified by flowing oxygen through its liquid, commercial grade reagents were used without purification.

III. Results and Discussion

1. Reactions of iodine in the dissolution step

Study of the dissolution of the spent-fuel specimens suggests the amount of iodine remaining in the fuel solutions is associated with the rates of dissolution. Figure 1 shows two typical ^{85}Kr concentrations in dissolver off-gas during dissolution. Table 1 provides specific details. In the case marked UO_2 in the figure, the ^{85}Kr concentration reached a maximum about 18 min after the start of the dissolution and then decreased rapidly. The dissolution was complete within about 1 h. Dissolution of most ordinary UO_2 fuel specimens provided similar ^{85}Kr concentration curves. Only small amounts of iodine (0 to 3.3% of the total iodine quantity) remained in solution, despite the high burnup of the fuels (Table 1).

In the other case, the ^{85}Kr concentration provided a long-tailed curve as shown by the Gd_2O_3-UO_2 curve in Fig. 1. This showed the dissolution continued for more than 2 h. Most of the Gd_2O_3-UO_2 fuel specimens used showed similar curves. Considerable amounts of iodine (4.5 to 9.7%) remained in solution, even for the fuel specimen having the lowest burnup (Table 1). Gadolinium does not form its insoluble iodide. A smaller dissolution rate causes a smaller NOx generation

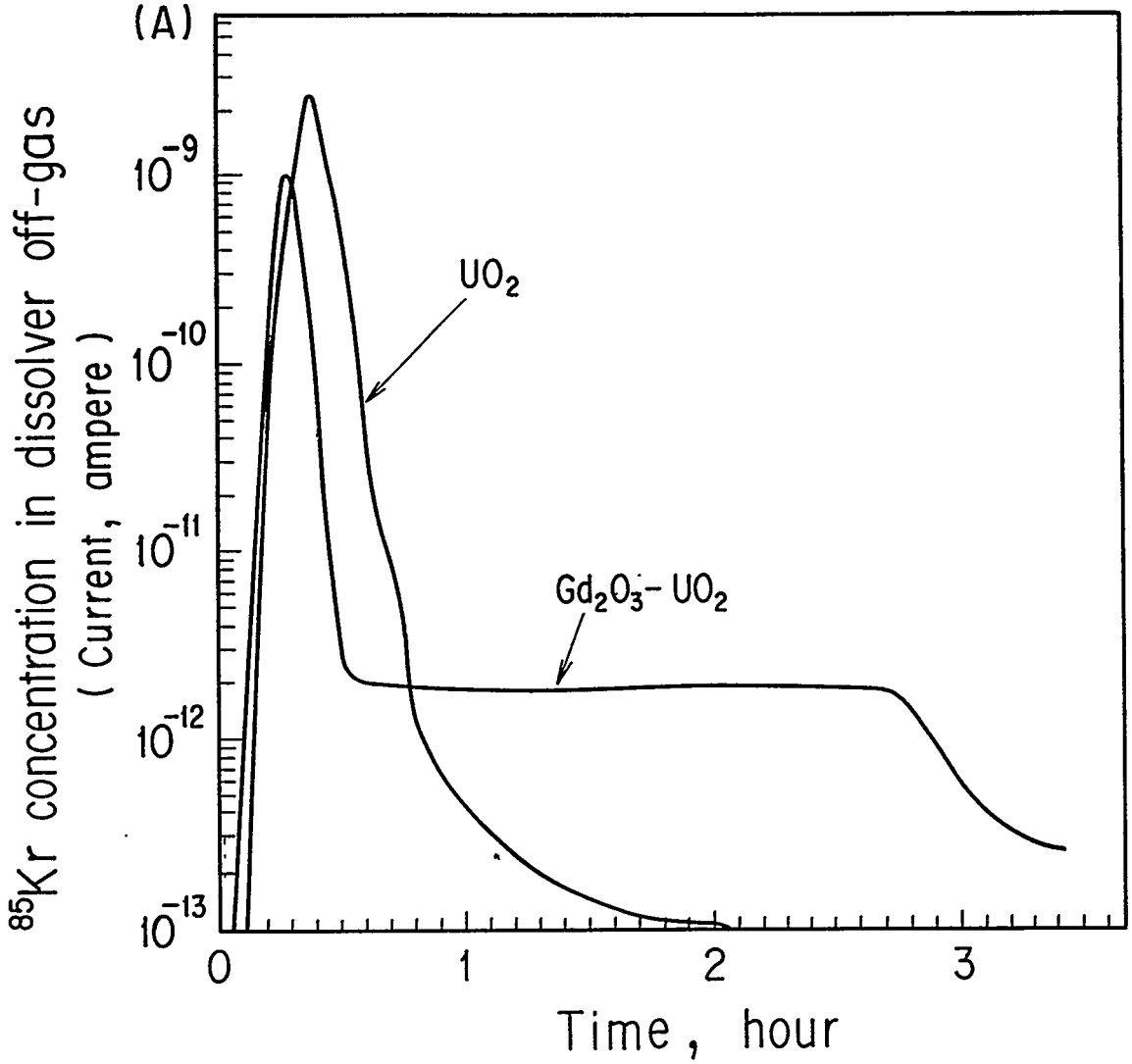


Figure 1 Two types of ^{85}Kr concentrations in dissolver off-gas during the dissolution of spent fuel specimens (approximately 3 g each).

Table 1 Details for the examples shown in Fig.1

Type of sample (No.)	Weight (g)	Burnup* (GWd/t)	Iodine quantity remaining in solution (%)**	Ratio of total iodine quantity to ORIGEN (%)***
UO_2 (89G09)	1.769	39.0	<0.1	85.2
$Gd_2O_3 - UO_2$ (89G06)	2.140	25.3	4.5	62.1

* The iodine quantities in the UO_2 and $Gd_2O_3 - UO_2$ samples, calculated from the ORIGEN-2 code, are 1400 and 860 Bq/g-U_0, respectively.

** Ratio of iodine quantity in the solution to total iodine quantity.

*** Ratio of the total iodine quantity to that calculated by the ORIGEN-2 code.

rate. Thus, the amount of iodine in solution was thought to be associated with the amount of NOx produced per unit time. Next, the relation between NOx concentration and the remaining quantity of iodine was examined using simulated spent-fuel solution.

2. Influence of NOx concentration on the quantity of colloidal iodine

Figure 2 shows the quantities of iodine species remaining 30 min, 1 h, and 2 h after the addition of 0.42 mg of iodine (I^-) to 30 mℓ of simulated spent-fuel solution at 100°C. NO_2 (24%) in N_2 (2.6×10^{-4} mol-NO_2/min) was bubbled through the solution. The main iodine species was colloidal iodine, which only slightly decreased with time. The amount of iodate (IO_3^-) was small, as predicted from a thermochemical calculation.[8]

Figure 3 shows the dependence of colloidal iodine quantity on NOx concentration in 100°C fuel solutions 30 min after the addition of iodine (0.42 mg-I^-). Both NO and NO_2 suppress the formation of colloidal iodine in proportion to their concentration up to about 10%. However, the effect from NO was greater than from NO_2 and the data for NO_2 concentration varied considerably.

The weights of the spent PWR-fuel specimens (Table 1) and their dissolution rates (Fig. 1) were used to calculate the amounts of NO generated during dissolution. The average concentrations in dissolver off-gas are shown in Fig. 3. These averages are in the region where the colloidal iodine quantity is very sensitive to a change in NO concentration, a small change in NO concentration results in a considerable change in the quantity of colloidal iodine. Calculation suggested that the quantity of colloidal iodine would be greater for Gd_2O_3-UO_2 than UO_2 shown in Fig. 1. This is consistent with the experimental results shown in Table 1. The quantity of iodine remaining in fuel solutions is less for spent-fuels with higher rates of dissolution than for fuel with lower rates, despite the amount of burnup.

3. Chemical reactions of iodine (I^-) during dissolution

Introducing (bubbling) NOx into nitric acid solution produces nitrous acid (HNO_2) in the solution according to the following reactions:

$$2\ NO(g) + H_2O(l) + NO_3^- = 3\ NO_2^- + 2\ H^+, \quad (1)$$

$$2\ NO_2(g) + H_2O(l) = NO_2^- + NO_3^- + 2\ H^+, \quad (2)$$

Combining (1) and (2):

$$NO(g) + 2\ H^+ + 2\ NO_3^- = 3\ NO_2(g) + H_2O(l). \quad (3)$$

Figure 4 shows the absorption spectrum of HNO_2 detected in 3.5 M HNO_3 bubbled with an NO-N_2 flow at 100°C. This implies the influence of NOx on colloidal iodine quantity is ascribable to the influence of the nitrous acid formed. Iodine in spent fuel, postulated to be in the form of CsI, most probably is subjected to the following three reactions during dissolution:

(a) Oxidation into I_2 by nitric acid:

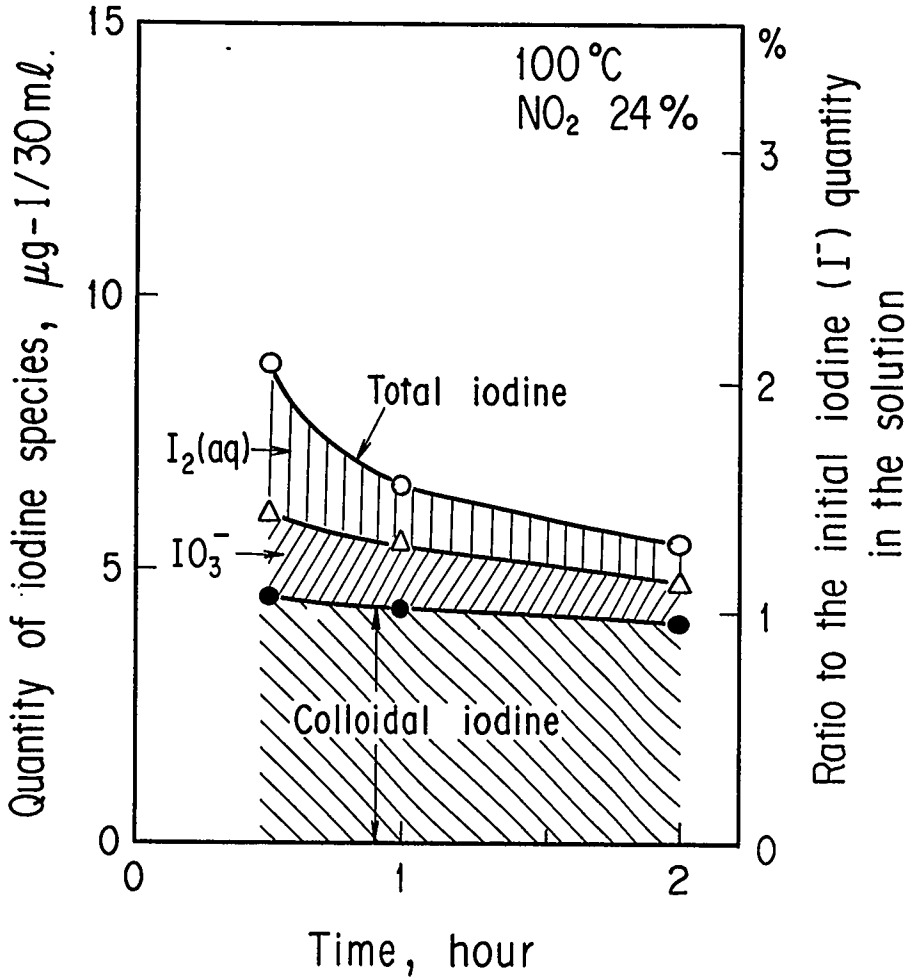


Figure 2 Iodine species produced by addition of iodine (I^-) into the simulated spent-fuel solution and their change in quantity with time. 30 mℓ of simulated spent-fuel solution was heated to 100°C, 24% NO_2 in N_2 was bubbled through the solution, and then (5 min later) iodine (0.42 mg-I^-) was added. 0.5 mℓ samples of the solution were analyzed 30 min, 1 h, and 2 h after the addition of the iodine.

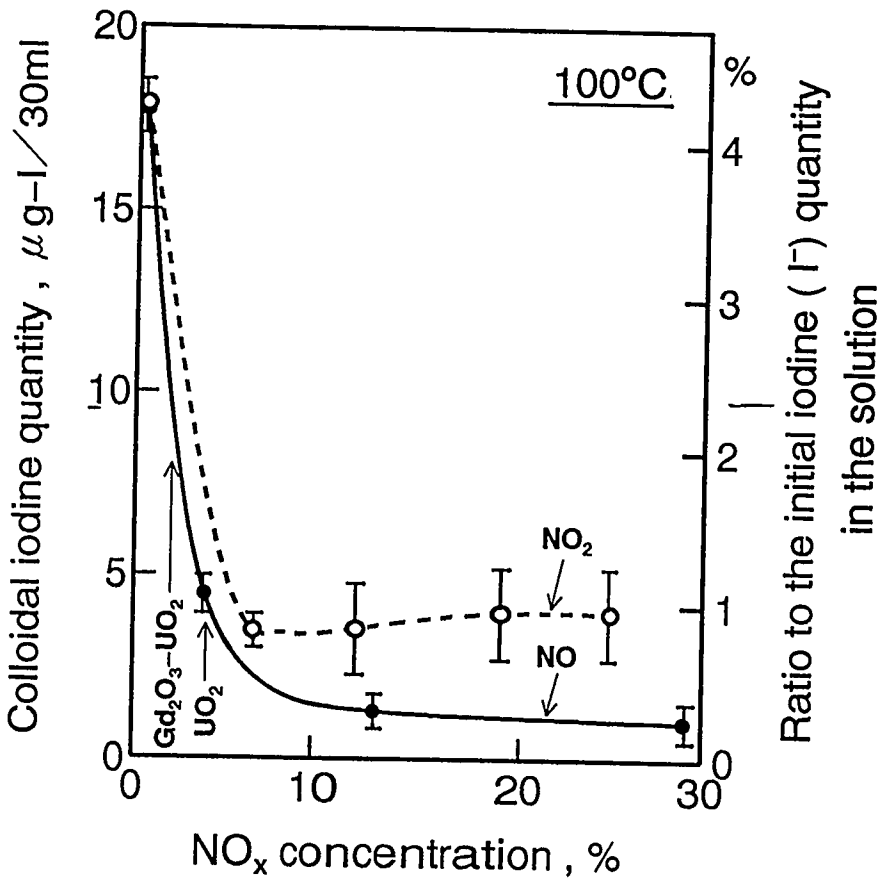


Figure 3. Dependence of colloidal iodine quantity on NOx concentration in simulated spent-fuel solutions at 100°C. Iodine (0.42 mg-I^-) was added to the solutions having NOx in various concentrations. The iodine species in the solutions were measured 30 min after the addition of the iodine. UO_2 and Gd_2O_3-UO_2 denote the calculated values of the average concentration of NO in the dissolution of the two spent-fuel specimens described in Fig. 1 and Table 1.

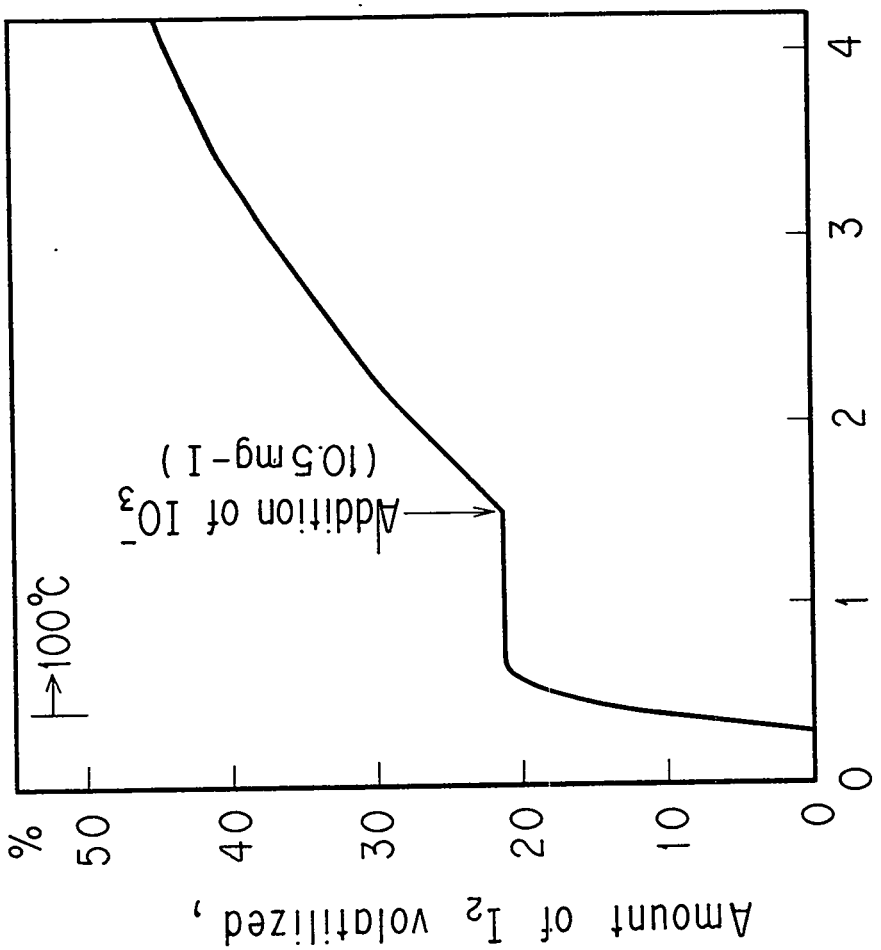


Figure 5. Effect of iodate (IO_3^-) on decomposition of silver iodide into I_2 in 3.4 M HNO_3 at 100°C. Heating of 30 mℓ of 3.4 M HNO_3 containing 19.3 mg of AgI volatilized a small amount of iodine (I_2). Addition of more iodate (10.5 mg-I) into the solution at 100°C caused a second volatilization of I_2.

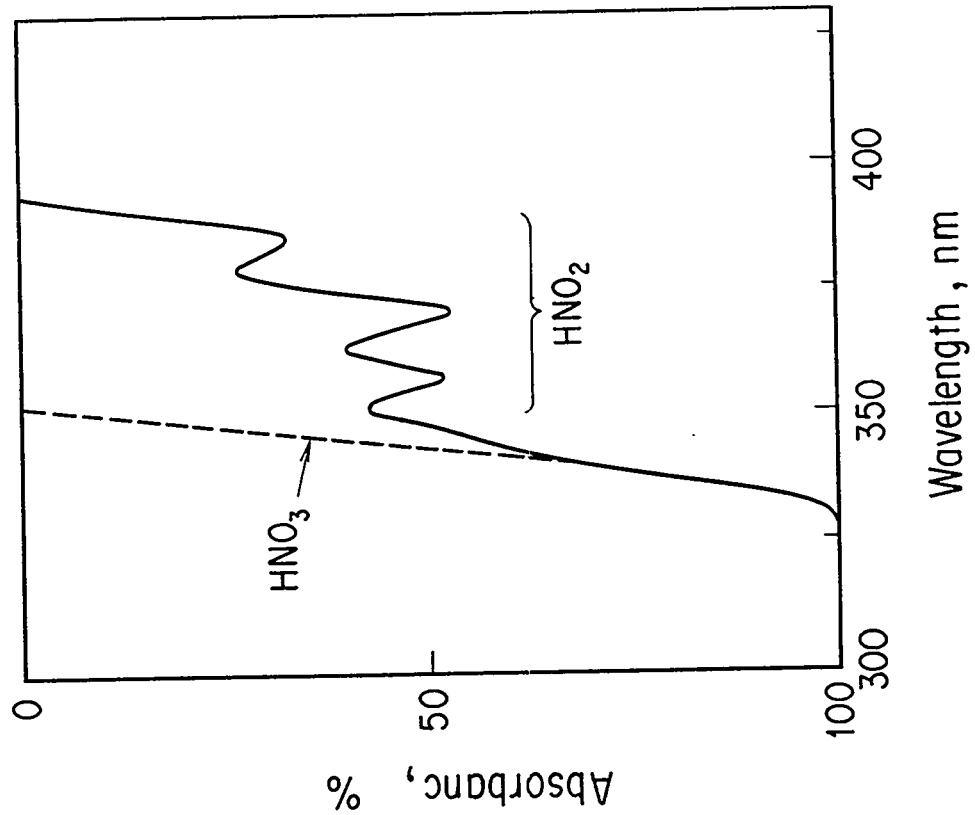


Figure 4. Absorption spectrum of HNO_2 produced when passing NO through 3.5 M HNO_3 at 100°C. While 14% NO in N_2 was bubbled into the solution, a 2 mℓ sample of the solution was cooled rapidly to 20°C for this measurement.

$$I^- + 2\,H^+ + NO_3^- = 1/2\,I_2(aq) + NO_2(g) + H_2O(l), \quad (4)$$

(b) Oxidation into I_2 by nitrous acid formed from NOx:

$$I^- + 4H^+ + NO_2^- + 2NO_3^- = 1/2I_2(aq) + 3NO_2(g) + 2H_2O(l), \quad (5)$$

(c) Formation of colloidal iodine:

$$3\,I^- + Ag^+ + Pd_2^+ \rightleftharpoons AgI(s) + PdI_2(s). \quad (6)$$

Of these, reaction (5) competes with reaction (6) in the presence of NOx and controls the quantity of colloidal iodine produced during dissolution. At 100°C, the equilibrium of reaction (3) lies so far to the right, that is, 80% of NOx is NO_2 and 20% is NO.[8] Nitrous acid needed in reaction (5) is likely to be provided by reaction (1). Therefore, the colloidal iodine quantity is suppressed more by NO than by NO_2, as already shown in Fig. 3.

4. Expulsion of iodine from spent-fuel solutions

Colloidal iodine is unstable immediately after its formation and is decomposed in a hot nitric acid solution by the following reaction:

$$AgI(s) + 2H^+ + NO_3^- = 1/2I_2(aq) + Ag^+ + NO_2(g) + H_2O(l). \quad (7)$$

Two factors retard this decomposition, the presence of NOx and the aging of the colloidal iodine. While suppressing the formation of colloidal iodine, NOx interferes with the decomposition of the already-formed colloid by reversing reaction (7).

The aging of the remaining colloidal iodine proceeds concurrently with its decomposition. To decompose the aged colloid, the addition of an excess amount of iodate (IO_3^-) is effective, as Figure 5 shows.[6] In this procedure, a precipitate of AgI was prepared from potassium iodide solution (which had been labeled with ^{131}I) and silver nitrate solution. The precipitate was dried and the solid AgI (19.3 mg) was heated in 3.4 M HNO_3. A small amount of I_2 was observed to volatilize at the beginning of heating; however, further volatilization needed the addition of iodate.

Effects of these two factors on the decomposition of colloidal iodine are shown in Figure 6 together with the experimental conditions. NOx retards the decomposition (curves 2 and 3). Effect of iodate is more prominent when it was added at the later part of the process (curve 4) than when it was added at the beginning of the process (curve 1). This is because the aging proceeds rapidly with time. Based on these results, the following two-step process for the expulsion of iodine remaining in spent-fuel solutions is proposed:

Step One - Heat the fuel solution without supplying NOx to decompose the colloidal iodine. When aging of colloidal iodine has proceeded after dissolution, an excess amount of iodate should be added.

Step Two - Supply (bubble) NOx through the solution while heating to reduce the iodate ($^{129}IO_3^-$) to volatile I_2.

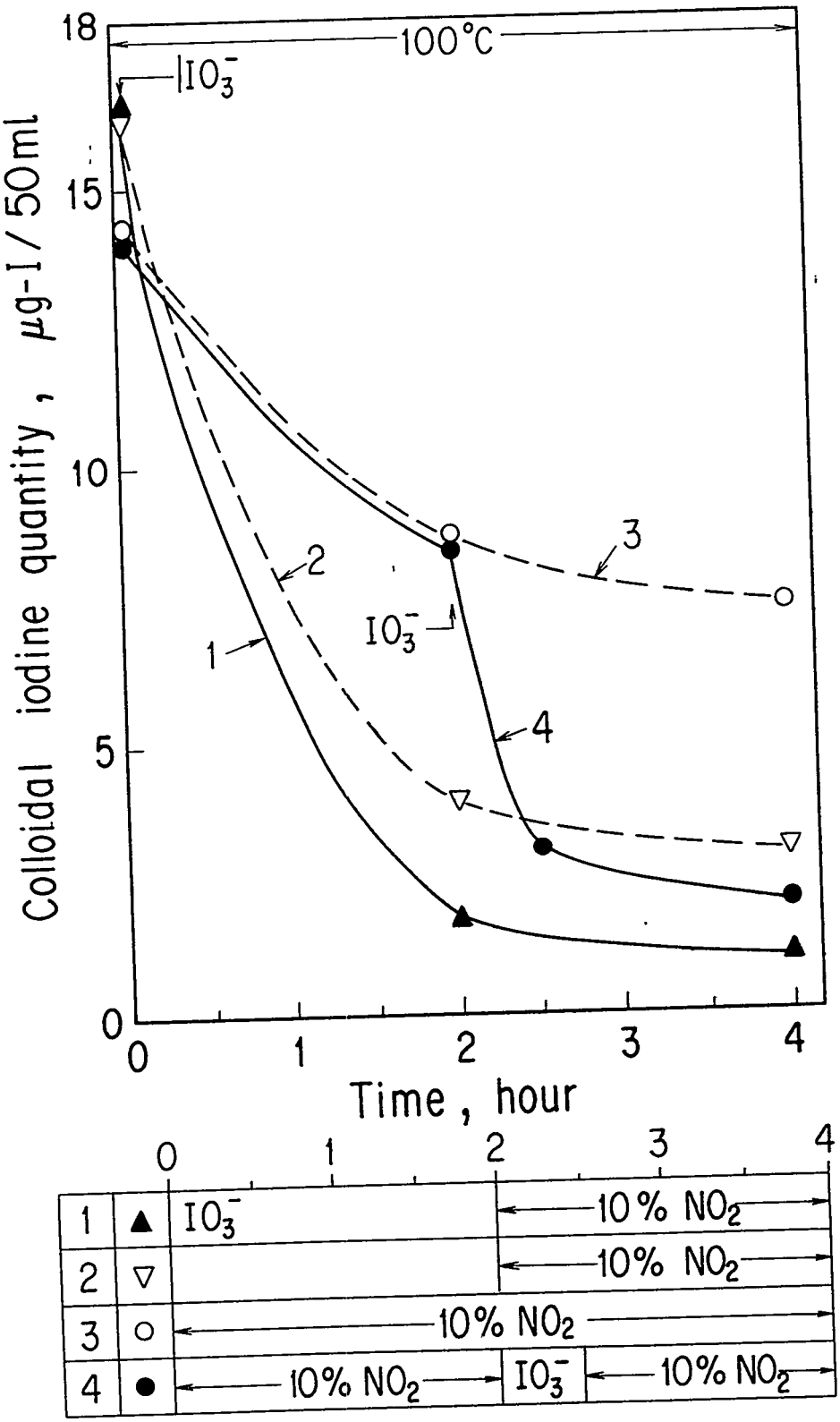


Figure 6. Effects of NOx and iodate on decomposition of colloidal iodine. The simulated spent-fuel solution (50 mfℓ each) containing colloidal iodine was heated at 100°C with and without a NO_2 supply. An amount of iodate (2 mg-I) was added at the beginning of heating for the run marked 1 and 2 h later for the run marked 4.

Table 2 Comparison of conventional NOx sparging method with the new process for expulsion of iodine from solution

Operation to expel iodine	Spent fuel specimens dissolved			Iodine quantity in the solution	
	Type (No)	Burnup (GWd/t)	Weight (g)	Before the expulsion process, Bq (%)*	After the expulsion process, Bq (%)**
Supply of NO (30% in N_2) at 100°C for 2 h. (The conventional method)	UO_2 (89G08)	30.8	1.302	23.6 (3.3)	7.9 (33.3)
	UO_2 (87H02)	39.7	2.250	11.7 (1.2)	3.2 (27.4)
	$G_2O_3 - UO_2$ (89G01)	21.9	1.659	34.8 (6.6)	15.9 (45.7)
	$G_2O_3 - UO_2$ (89G05)	21.2	1.382	63.9 (9.7)	28.2 (44.1)
Addition of IO_3^- (1 mg-I/30mℓ) and heating at 100°C for 2 h without NO_x supply (The new process)	UO_2 (89G10)	39.3	1.746	9.6 (1.2)	N.D*** (0)
	UO_2 (87C05)	~ 32.1	2.008	2.1 (0.3)	N.D (0)
	$G_2O_3 - UO_2$ (89G06)	25.3	2.140	24.0 (4.5)	N.D (0)

* Ratio of the initial iodine quantity in the solution to the total iodine quantity collected during dissolution and expulsion processes

** Ratio of the final iodine quantity to the initial iodine quantity in the solution

*** Not detected

The efficiency of this process is compared with that of the conventional method (NOx sparging) for spent PWR-fuel specimen solutions (30 mℓ each) in Table 2.[3] After two hours of NOx sparging (30% NO in N_2) at 100°C, 27 to 46% of the initial iodine quantity remained in solution. However, two hours (without NOx supply) at 100°C after addition of iodate (1 mg-I^-) removed all iodine from solution. For both ordinary UO_2 and Gd_2O_3-UO_2 specimens, the new process was much more effective. Here, Step Two was not necessary. However, there is a possibility that a small amount of iodate may remain in solution as shown in Fig. 2 and that a small portion of I_2 may be oxidized to iodate during Step One. So, NOx feeding, viz. Step Two, is desirable for more effective expulsion. These findings show that the main iodine species remaining in actual spent-fuel solutions is colloidal. Thermochemical calculations have determined that the formation of iodate during dissolution was suppressed because of the generated NOx.[8]

In our current work, the expulsion was performed on spent-fuel solution previously separated from insoluble residue. In actual reprocessing, the expulsion would be carried out in the presence of insoluble residue. Then, it is expected that part of iodine deposited on the insoluble residue will be expelled by the new expulsion process.[9]

Boukis and Henrich attributed the residual iodine to nonvolatile organic iodine that originated from organic impurities in nitric acid.[4] They noticed that recycled acid that had not been carefully purified increased the residual iodine quantity. In our case, however, colloidal iodine was observed only in simulated spent-fuel solutions containing Ag^+ and Pd_2^+. When iodide (I^-) was added into $UO_2(NO_3)_2$-HNO_3 and HNO_3 solutions, the amount of colloidal iodine was negligibly small. Thus, we understand the residual iodine cannot be nonvolatile organic iodine arising from organic impurities in nitric acid.

IV. Conclusion

1. In dissolution of spent nuclear fuels, iodine (I^-) from CsI in the fuels is subjected to the following three reactions:
(a) Oxidation into I_2 by nitric acid,
(b) Oxidation into I_2 by nitrous acid (HNO_2) arising from NOx generated in the dissolution, and
(c) Formation of the colloid of insoluble iodides such as AgI and PdI_2.
Depending upon the NOx concentration, NOx suppresses formation of colloidal iodine.

2. The main iodine species in spent-fuel solutions is colloidal iodine.

3. The following two-step process is more effective than the conventional process (NOx sparging) for expulsion of the residual iodine from the fuel solution:

Step One - Heat the fuel solution without NOx supply. When aged colloidal iodine is present after the dissolution step, an excess amount of iodate should be added into the solution.

Step Two - Supply NOx to the fuel solution while heating.

Acknowledgment

The authors wish to express their thanks to Drs. Mitsuru Maeda and Takeo Adachi of the Japan Atomic Energy Research Institute (JAERI) for their advice and support. We also thank Mr. Haruka Shinohara for his assistance in preparing this manuscript. This paper is one of several reporting dissolution experiments on high-burnup PWR spent fuels. The authors express their appreciation to the Science and Technology Agency of Japan for their support of this work.

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DISCUSSION

WREN: Have you looked at the effect of the oxidation state of UO_2 (e.g. UO_{2+x}) on the rate and behavior of iodine repulsion? Depending on the oxidation state of UO_2, I think the oxidation state of iodine and the release rate of iodine from fuel would change. Another question: what is the effect of radiation on AgI stability?

SAKURAI: The oxidation state is known to influence the dissolution rate of fuels. But I have not considered its effect on behavior of iodine. Your comment is very useful. Silver iodide was stable against Co-60 irradiation up to 10^3 C/kg.

KOVACH, L.: Have you considered that in this type of experiment, the actual sequence of dissolution is slightly different than in a continuous desorber in a fuel processing plant? Have you considered the differences in heat-up rates and contact temperatures in your experiment versus an actual reprocessing plant application? Do you consider that the experimental sequence described matches the thermal sequence in an actual reprocessing plant?

SAKURAI: I know the difference. Our experiment was on a very small scale. In the near future we have to do the same experiments on a larger scale, for certain actual data points. However, the behavior of iodine that I reported here is thermochemically reasonable; hence, it is expected to hold for actual reprocessing plants. The process we proposed can be incorporated with a reprocessing plant that adopts the continuous dissolution of spent nuclear-fuels. It is desirable to confirm its efficiency on a much larger scale.

KOVACH, L.: In these experiments you started out by placing the fuel into your container and then heating it up, which is slightly different from the actual desorber in a reprocessing plant.

SAKURAI: In actual reprocessing plants, the fuels are put into hot nitric acid solutions. The dissolution rate will be faster than in these experiments.

EVANS: Could you give us some more information on how you separated the iodide, iodate, and the colloidal forms of iodine using chloroform?

SAKURAI: In our analytical scheme, 'colloidal iodine is the final species remaining in the aqueous solution after ionic species, as iodide and iodate, were extracted in the CCl_4 phase.

Details of the analytical scheme are described in the literature cited in the text (Ref. 7).

THE DEVELOPMENT AND DESIGN OF THE OFF-GAS TREATMENT SYSTEM FOR THE THERMAL OXIDE REPROCESSING PLANT (THORP) AT SELLAFIELD

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Sellafield [1] or Risley[2],
UK

Abstract

British Nuclear Fuels plc (BNFL) completed construction of its Thermal Oxide Reprocessing Plant (THORP) at Sellafield in 1992, at a cost of £1,850M. After Government and Regulatory approval, active commissioning was initiated on 17 January 1994.

From the outset, the need to protect the workforce, the public and the environment in general from the plant's discharges was clearly recognised. The design intent was to limit radiation exposure of members of the general public to "As Low as Reasonably Practicable" (ALARP). Furthermore no member of the most highly exposed (critical) group should receive an annual dose exceeding 50 microsieverts from either the aerial or marine discharge routes.

This paper describes how the design intent has been met with respect to aerial discharges. It outlines the development programme which was undertaken to address the more demanding aspects of the performance specification. This ranged from small-scale experiments with irradiated fuel to inactive pilot plant trials and full-scale plant measurements. The resulting information was then used, with the aid of mathematical models, in the design of an off-gas treatment system which could achieve the overall goal.

The principal species requiring treatment in the THORP off-gas system are iodine-129, carbon-14, nitrogen oxides (NOx), fuel dust particles and aerosols containing plutonium or mixed fission products. The paper describes the combination of abatement equipment used in different parts of the plant, including counter-current absorption columns, electrostatic precipitators, dehumidifiers and High Efficiency Particulate Air (HEPA) filters. Because a number of separate off-gas streams are combined before discharge, special depression control systems were developed which have already proved successful during plant commissioning.

BNFL is confident that the detailed attention given to the development and design phases of the THORP off-gas system will ensure good performance when the plant moves into fully radioactive operation.

I - Introduction

Brief History

THORP is the third reprocessing plant to be constructed on BNFL's Sellafield site in Cumbria, UK. The plant is designed to reprocess irradiated fuel from Advanced Gas-Cooled Reactors (AGR) in the UK and from Light Water Reactors (LWR) in Europe and Japan: virtually all of its first 10 years operation are committed with contracts for 7000t(U).

Conceptual design of the plant started in 1974, supported by construction and operation of a number of major development facilities during the later 1970's and early 1980's. Design of the plant was delayed by the Government's decision to hold a public inquiry (the Windscale Inquiry) in 1977. Following the successful outcome of the inquiry, design recommenced in 1983, leading to the start of civil construction in 1984 and eventual handover of the completed plant in 1992. Plant commissioning then proceeded but the progression to active commissioning, planned for December 1992, was delayed by a protracted public consultation exercise to satisfy recent environmental legislation. Government and Regulatory approval to proceed was finally given, leading to the commencement of active commissioning on 17 January 1994.

Design Intent

All BNFL's operations on the Sellafield site are strictly controlled by Government Authorising Bodies. For the control of discharges to the environment, the relevant bodies are Her Majesty's Inspectorate of Pollution (HMIP) and the Ministry of Agriculture, Fisheries and Food (MAFF). Certificates of Authorisation issued by the Departments impose strict conditions on the discharge and disposal of radioactive waste. They also require BNFL to carry out detailed environmental radioactivity monitoring programmes (in addition to their own independent measurements) and to use "Best Practicable Means" to further limit the radioactivity in the waste discharged.

BNFL's strategy for the management of aerial and marine radioactive effluents arising from THORP operations were first formally stated at the Windscale Inquiry in 1977. In summary the design intent was that:

(a) radiation exposure of members of the general public, arising from discharges from THORP and associated facilities, should be "As Low As is Reasonably Practicable" (ALARP);

(b) in any case, no member of the most highly exposed (critical) group should receive an annual committed effective dose equivalent (CEDE) in excess of 50μSv from either aerial or marine discharge routes.

In the context of exposure to the general public, it is important to consider both:

(a) individual risk, as measured by doses to members of the critical group, and

(b) societal risk, as measured by the collective dose to large populations over a long time period.

In addition to dose uptake constraints, HMIP has more recently set an Aerial Discharge Authorisation for THORP which limits the quantity of specific radionuclides which can be released from the THORP stack on a daily, quarterly or annual basis. The annual discharge limits for the most significant radionuclides are given in Table 1.

Scope of Paper

This paper describes how the design intent on aerial discharges has been met. It examines the different sources of off-gas arisings from the THORP process and describes how the different types of stream have been segregated for treatment prior to extraction and discharge from a common 125m stack. The development work associated with some of the key process equipment is described and key design features of the equipment are covered, including depression control in such a complex system. Finally the predicted discharges from the plant are compared with the latest Discharge Authorisation to illustrate how well the plant design is expected to meet its objectives.

Table 1 - Comparison of Predicted THORP Discharges with Discharge Authorisation

Radionuclide	THORP Discharges (TBq/yr)	THORP Downstream Plant Discharges (TBq/yr)	THORP[(1)] Total Discharges (TBq/yr)	THORP Authorisation (TBq/yr)
H -3	21.6	0.0007	21.6	43
C-14	0.434	~0	0.434	0.87
Kr-85	369,000	6.61	369,007	470,000
Sr-90	0.004	0.0036	0.0076	0.0078
Ru-106	0.0355	0.0016	0.0371	0.05
I-129	0.0218	0.0031	0.0249	0.044
Cs-137	0.0055	0.005	0.0105	0.011
Pu(Alpha)	0.00027	~0	0.00027	0.0005
Total Alpha	0.00048	~0	0.00048	0.001
Total Beta (β5)	0.152	0.017	0.169	0.28

Critical Group dose [(1)] = 22μSv/yr
Target dose = 50μSv/yr

(1) Based on 1200t(U)/yr of reference fuel

II Gaseous Arisings

THORP is an integrated reprocessing plant with all the major process steps being carried out in a single building (See Fig 1). Only where use could be made of waste and effluent treatment facilities serving existing process plants on the Sellafield site are certain operations performed outside the THORP building. The THORP process breaks down principally into a shear/leach "Head End" section with clarification of the dissolver solution, followed by chemical separation of the uranium, plutonium and fission products by solvent extraction in pulsed column contactors. The uranium and plutonium products are purified by further solvent extraction prior to conversion to solid forms suitable for storage or return to customers.

Within the plant as a whole, a variety of different off-gases arise.

Dissolver Off-gas

Leaching of the sheared fuel in boiling nitric acid releases virtually all the volatile components of the irradiated fuel assemblies, principally I-129, C-14 and Kr-85. Also Nitrogen oxides (NOx) are generated by dissolution of the fuel. The dissolver off-gas (DOG) stream consists of the ventilation air from the shear machine and the off-gases from each of three dissolvers, amounting to a flow of about 400m^3/hour. The stream also contains very fine particles of fuel dust which are not captured in the dissolver and, in total, represents by far the biggest off-gas challenge in THORP.



Figure 1 THORP building

Central Off-gas

Apart from the dissolvers, most of the other vessels and process equipment within the Head End and Chemical Separation Plant are vented via a separate Central Off-gas (COG) system. This has to accept a large number of streams with widely differing flows and compositions.

The most environmentally significant component of the COG arisings is I-129. Although the vast majority is volatilised into the DOG system, any residual amount is assumed to be released from downstream vessels into the COG system. In addition, aerosols created in the chemical treatment and liquor transfer processes throughout the plant carry particles of non-volatile actinides and fission products into the ventilation stream. The nuclides that contribute significantly to the aerial activity levels are Ru-106, Pu, Sr-90 and Cs-137.

There are certain non-active arisings which have to be considered in the design of the COG system. The only one subject to regulatory discharge limits is NOx.

Cell Ventilation

THORP is designed on the principle of cascading depressions between areas to provide barriers against spread of contamination. Cells and caves which contain the most highly active processes in the plant are therefore maintained under a depression with reference to the adjoining access areas. Generally inleakage at cell depression is adequate to provide the air flow required in a cell with little or no heat emitting equipment. In cells with large heat sources, additional air is

provided to dissipate the heat by purpose-built engineered inlets comprising HEPA filters and control/fire dampers.

Air extracted from cells and caves exhibits the same distribution of radionuclides (footprint) as is present in the process equipment in the cell. Arisings from mechanical operations of fuel shearing in head end caves has been assessed on empirical data from similar operations on other plants. On cells where operations should be clean under normal operating conditions (washdown provisions are installed to clean-up spillages and leaks), the arisings have been based on potential re-suspension of liquors from sprays or pools on the bunded floors.

Building Ventilation

Building ventilation and heat systems provide a comfortable working environment in 'normal' and 'limited' access areas of the plant.

Normally accessible (unrestricted access) areas, despite having no definitive arisings (except under abnormal circumstances), are allocated a mean airborne contamination which, if inhaled over a 2000 hour year, would result in 1% of the whole body dose target being received by inhalation.

Limited (restricted) access areas normally have very low arisings but have the potential for activity being present for short periods or in abnormal circumstances. Such areas are allocated a mean airborne contamination of ten times that allocated to the normal access areas. During such circumstances personnel would wear respiratory protection. In each instance the nuclides present are in proportion to the 'footprint' of the active material being processed.

Total Ventilation System

A schematic diagram showing the inter-relationship of the various components of the THORP ventilation system is given in Figure 2. It illustrates that main ventilation streams are kept separate until they enter the 125m stack from which they are discharged.

III Flowsheet and Equipment Development

Dissolver Off-gas

The prime task of the DOG system is to remove nitrogen oxides generated by the dissolution of the UO2 fuel, together with the major volatile radioactive species released as the fuel is dissolved. The DOG challenge is illustrated in Table 2, together with the flowsheeted Decontamination Factors (DFs) for each item of equipment.

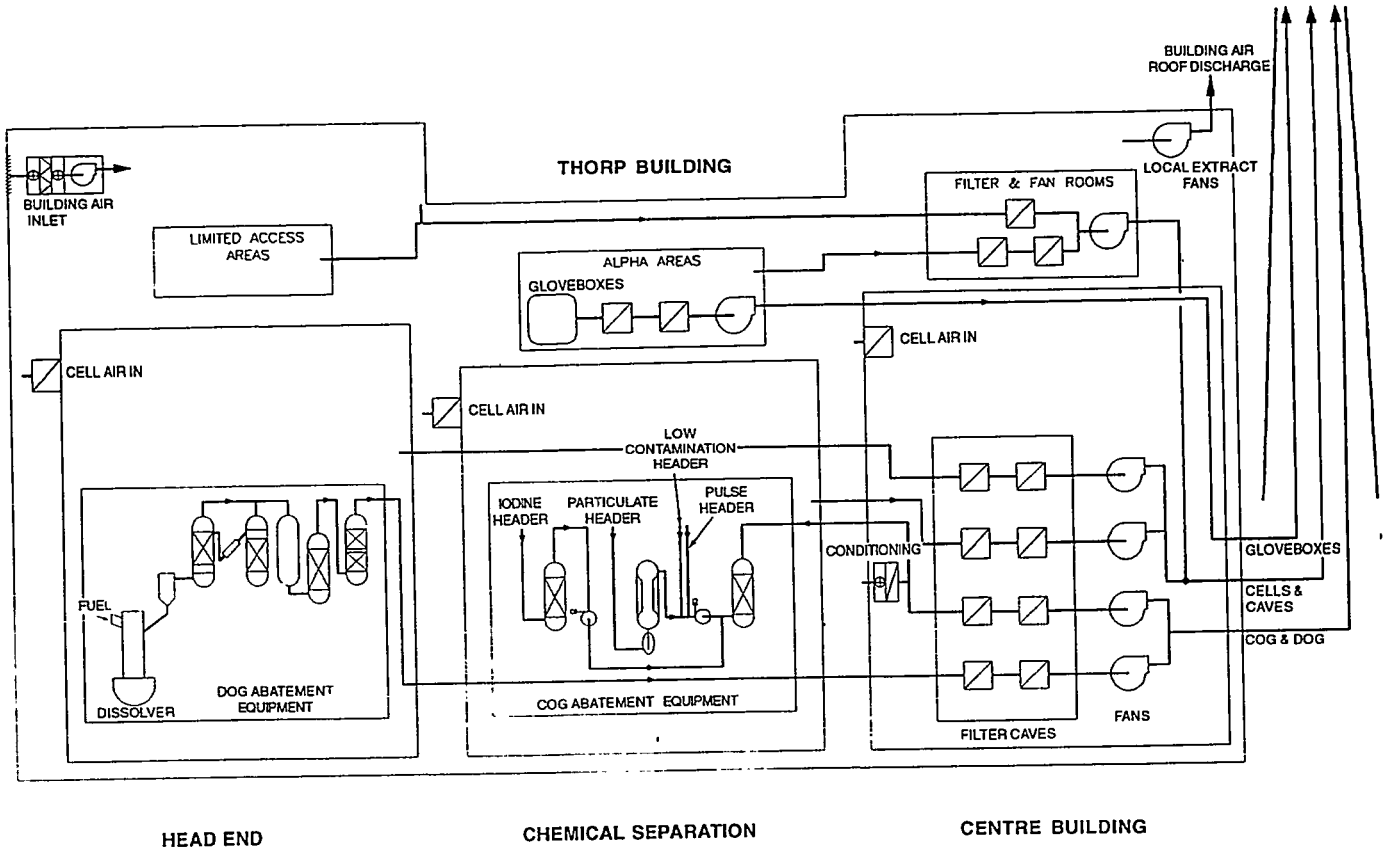


Figure 2 Schematic of THORP active ventilation systems

Table 2 DOG System Performance

Radionuclide or species	Arising[(1)] (TBq/yr)	Flowsheet Decontamination Factor (DF)				
		Condenser	Acid Scrubber	Caustic Scrubber	Weak Acid Scrubber	HEPA
H-3	97.2	1	3	1.5	1	1
C-14	28.9	1	1	70	1	1
Kr-85	3.69E5	1	1	1	1	1
Ru-106(gas)	37.5*		20	100	1	1
Ru-106 (solid)	37.5*		1	1	1	10E4
I-129	1.41	1	1.05	100	1	1
NOx	8.1E4 m^3	1.5	3	100	1	1
Fuel Dust	2.6E3 kg	25	70	1.2	1.4	10E4

* post condenser

(i) Based on 1200t(U)/yr of reference fuel

Recognizing the need for highly efficient I-129 removal in THORP, the first requirement is to volatilise as much as possible at the dissolver stage. Once converted to organic forms by contact with solvent (TBP/OK) further down the process, iodine capture is much more difficult. Experimental work has demonstrated that the NOx evolved during dissolution of the fuel is effective in maintaining the I-129 in its volatile molecular form, which renders it easier to remove from the off-gas system.

Treatment of the DOG stream is performed principally by acid and caustic scrubbing in packed columns. The full train of DOG equipment is shown in Fig 3

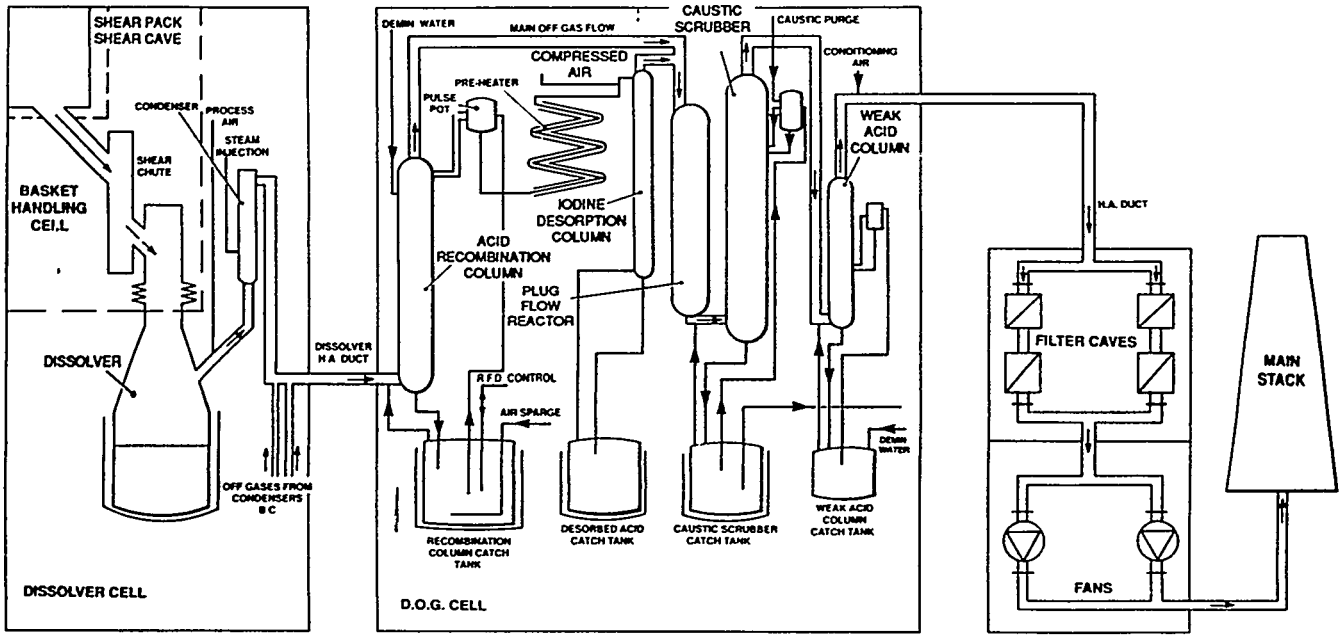


Figure 3 Dissolver off gas (DOG) extract system

The off-gases from each dissolver pass through their own dedicated reflux condenser and are then combined to be fed to the recombined acid column. Here, scrubbing with 6M nitric acid serves to remove the bulk of the oxides of nitrogen and two impingement plates at the base of the column assist removal of the entrained fuel dust.

A small amount of the iodine is also absorbed in the acidic scrubber liquor and this has to be removed before the liquor is fed back into the main process stream. This iodine desorption is achieved by heating a bleed of scrubber liquor and passing it through a small diameter packed column, counter current to a flow of air, which then joins the main off-gas flow carrying the desorbed iodine.

Following the recombined acid column is a large empty vessel, termed the "noxidiser", which provides a residence time of about two minutes in which the nitric oxide present is substantially oxidised to nitrogen dioxide by the oxygen in the air (which makes up the major proportion of the off-gas flow).

Increasing the proportion of nitrogen dioxide in this way improves the absorption of the nitrogen oxides in the caustic scrubber immediately downstream. The caustic scrubber will achieve a reduction in the nitrogen oxide concentration to below 1000ppm and a reduction in the I-129 and C-14 by factors of 100 and 70 respectively. Following the caustic scrubber the gas stream will be scrubbed with chilled recirculating water to remove entrained caustic soda droplets and dehumidify the gas. Finally the DOG stream undergoes two stages of HEPA filtration for final particulate removal before discharge to atmosphere.

DOG Supporting Development

In view of the dependence of the off-gas system upon aqueous scrubbing, the major facility in the development programme was a pilot scale rig - the dissolver off-gas pilot plant - to enable study of each of the scrubbing processes in the DOG system.

The rig, depicted in Fig 4, consisted of a small packed column, 100 mm diameter, with a packed height of 3m. Equipment was provided to generate a suitable mixture of oxides of nitrogen in air to which gaseous iodine could be added as appropriate. The scrub liquor could either be nitric acid or caustic soda solution and the liquor recirculated continuously through the column or treated in a single pass.

Sampling of the gases by means of heated sample lines enabled gas phase analysis by UV or IR spectrometers for nitrogen oxides and carbon dioxide respectively. Sampling for iodine was achieved by selective absorption on heated beds of silver nitrate impregnated absorbers.

As originally conceived the pilot plant was to have handled iodine traced with I-131 in order to facilitate analysis to the low levels required. This increased significantly the complexity of the rig and would have made experimental work much more unwieldy. However,

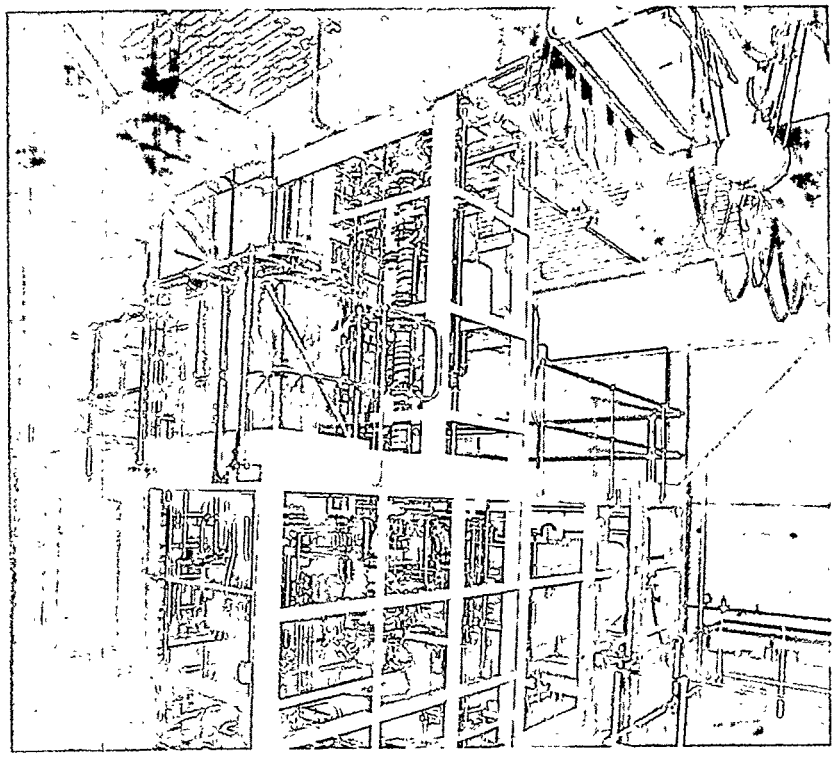


Figure 4 Dissolver off gas pilot plant

satisfactory development of ion selective electrode techniques sensitive to iodine concentrations as low as 1×10^{-7} M removed the need for trace active work and considerably simplified the experimental programme. The outstanding analytical difficulty was the determination of inactive iodine on the silver nitrate absorbers used for sampling the gas streams but this was readily overcome by the use of neutron activation analysis.

Experimental investigation of acid and alkaline scrubbing of nitrogen oxides was supported by work on computer simulation of the processes in a package developed by BNFL. This package, NOXSIM, was thoroughly validated against the pilot plant experimental results and thus could be used with confidence in the design of the THORP off-gas columns. NOXSIM is discussed further in Section IV.

Apart from the development work on the DOG pilot plant, other supplementary development was undertaken to address specific issues within DOG system. For example, experiments were performed to investigate iodine behaviour during dissolution of irradiated fuel and the subsequent treatment of dissolver off-gases. These investigations were part of the experimental work carried out on the THORP miniature pilot plant, a highly active 1:6500 scale pilot plant built primarily to study solvent extraction flowsheets. The dissolution of irradiated fuel served to show that the I-129 was present as molecular iodine and hence the simulation in the DOG pilot plant was valid.

Carbon-14 behaviour was less of a problem than I-129 since the bulk of it would be present as carbon dioxide and this would mix with the much larger quantities of CO_2 in air. There is abundant information on caustic scrubbing of CO_2 in air but a number of confirmatory experiments were necessary on the DOG pilot plant to ensure that the presence of 1 to 2 per cent NOx would not affect the CO_2 removal.

Not all the C-14 released from the dissolver is in the form of CO_2: a small amount is present as CO. Measurements performed during dissolution of irradiated fuel on the THORP miniature pilot plant confirmed literature information that the CO component was less than 1% and would not compromise the ability of the caustic scrubber to reduce C-14 levels by a factor of 70.

Measurement of CO_2 absorption has also proved important in the detection of potential maloperation. Experiments with the DOG pilot plant have shown that monitoring for increased CO_2 in the off-gases downstream of the scrubber provides adequate warning of approaching loss of alkalinity.

One issue raised well into the design phase of the THORP DOG system was a suggestion that troublesome quantities of volatile Ru-106 might penetrate into and even through the off-gas system. At first this question was explored by means of dissolution of irradiated fuel in a highly active cell and then in greater depth using trace active laboratory experiments. These two pieces of work were sufficient to indicate that only relatively small amounts of volatile Ru-106 would be generated and that the off-gas system as designed would ensure adequate removal. Subsequent studies with irradiated fuel dissolution in the THORP miniature pilot plant have indicated that the removal efficiency in the THORP dissolver off-gas system will be ten times greater than initially estimated and thus the impact of volatile Ru-106 has proved to be of little significance.

Central Off-Gas

As already described, the COG system serves most of the vessels and equipment downstream of the dissolvers. The full train of COG equipment is shown in Fig 5. The off-gas arisings from different parts of the plant or from different types of equipment are combined into a series of "headers", which feed into the COG system at an appropriate point according to the type of decontamination required. The arising of key components in the COG feed is shown in Table 3, together with the flowsheeted DFs for each item of equipment.

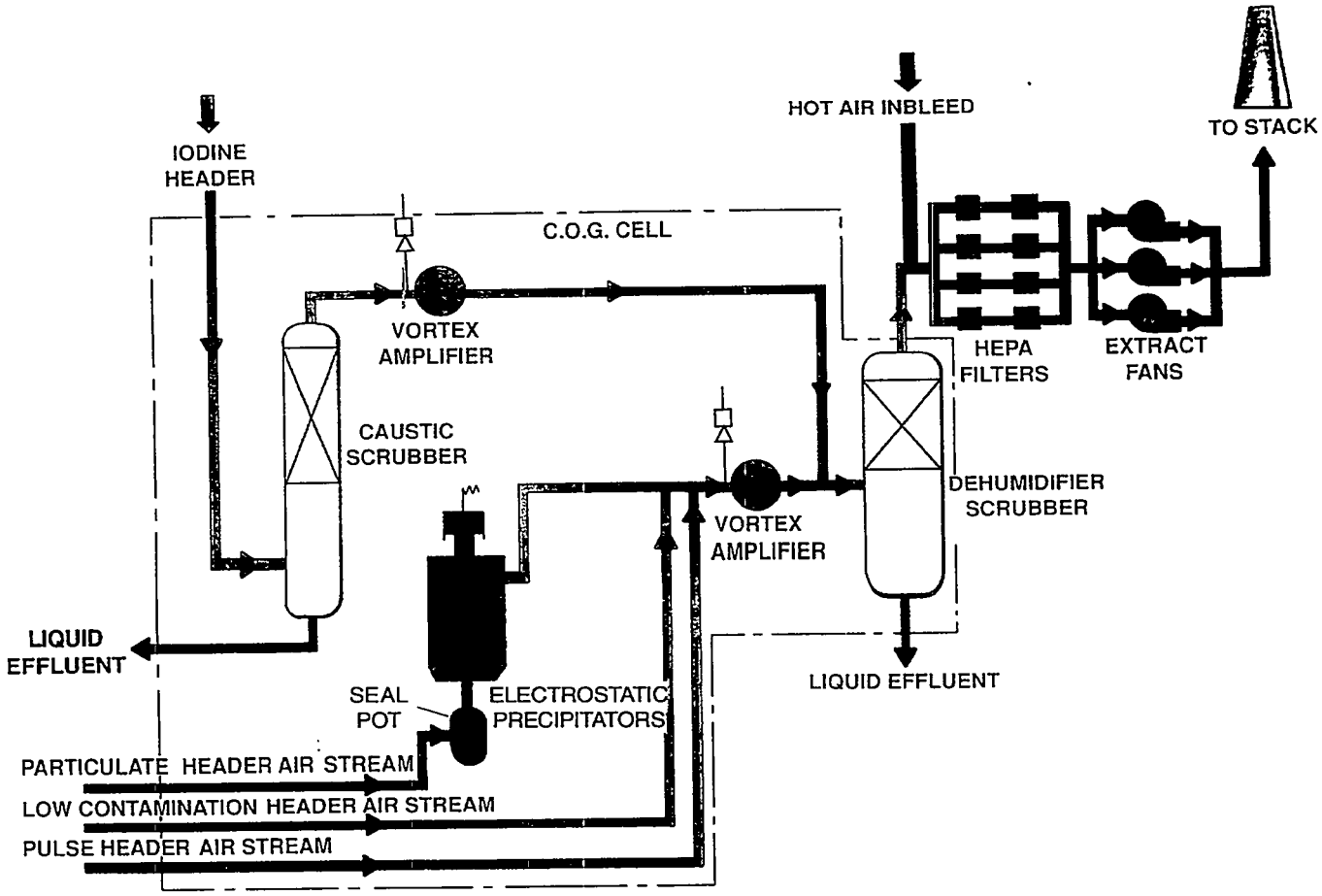


Figure 5 Central off gas (COG) extract system

First in the COG system is a 1.8m diameter caustic scrubber which serves the iodine header. This is primarily for iodine removal though it removes NOx and some particulate as well. A conservative Decontamination Factor (DF) of 2 has been assessed for iodine removal because the residual iodine from these downstream operations will be substantially organic. The scrubber performs a secondary function as an iodine trap in the event of accidental neutralisation of iodine-bearing streams in the Low Active Effluent Treatment Plant.

Table 3 COG System Performance

Radionuclide or species	Arising[(1)] (TBq/yr)	Flowsheet Decontamination Factor (DF)			
		Caustic Scrubber	ESP	Dehumidifier	HEPA
Sr-90	56.4	10	100	10	10E4
Ru-106(gas)	0.15	1	1	5	1
Ru-106(solid)	14.4	10	100	10	10E4
I-129	0.009	2	1	1	1
Cs-137	79.9	10	100	10	10E4
Pu(alpha)	6.19	10	100	10	10E4
NOx	84.9 m^3	43	1	1.6	1

(1) Based on 1200 t (U)/yr of reference fuel

The next equipment in the train is a pair of 680 m^3/hr Electrostatic Precipitators (ESPs) whose purpose is the removal of entrained droplets and aerosols from the particulate header. These devices have proved to be very reliable and effective in other applications in highly active plant at Sellafield and only generate a low volume of liquid waste for treatment. Furthermore, by reducing the load on the HEPA filters at the end of the train, they reduce filter change frequencies and waste disposal costs. However care has to be taken to ensure that any solvent vapour in the feed stream is well below its lower flammability limit because the high voltage across the electrodes represents a potential ignition source.

The dehumidifier scrubber downstream of the ESPs is primarily aimed at removing moisture from the gas stream prior to HEPA filtration. This would cause deterioration of the filter elements, leading to their eventual collapse and hence increased solid waste arisings. In order to prevent this the vent gas has to be cooled below its dew point, using a chiller unit to cool the dehumidifier scrubber liquor.

In addition, the dehumidifier acts as the primary disentrainment device for streams for which treatment in the ESPs is unnecessary. These include the low contamination vent header and the pulse header from the pulsed column drive legs in the highly active and plutonium purification cycles. With a large number of pulse systems operating independently the header could, theoretically, receive a very high flowrate if all the pulse systems discharged simultaneously. A Monte Carlo approach was used to predict the highest realistic flow likely to arise, thereby minimising the degree of over- design.

The final equipment in the train are banks of primary and secondary remote-change HEPA filters, which act as a final polishing step for particulates and as a back-up in the event of equipment maloperation upstream. Hot air is bled into the vent before the filters to ensure that the feed is well above its dew point.

The COG stream is finally discharged from a 125m stack, together with other THORP off-gases.

COG Supporting Development

In comparison with the DOG system, very little specific development work was necessary to support the COG system design. In relation to scrubber design, data generated from the DOG pilot plant were used, with the help of the NOXSIM computer model, to assist design of the COG unit.

One area which did require special attention was the characterization of the aerosols generated by pulsing systems, agitation systems and various liquor transfer devices within the plant. Information on quantity and droplet size distribution was generated by development trials on replica equipment. This work showed that treatment of many of these streams in the ESPs was unnecessary; scrubbing in the dehumidifier gave sufficient decontamination.

Some development was performed on the ESP units during the design phase. Experience on earlier plants and collaboration with the manufacturer enabled a simpler and more compact design to be used on THORP without losing any efficiency. The simplification of the central electrode design within the collection tubes was demonstrated on a test rig.

Glovebox Ventilation

The finishing processes on the purified plutonium stream are largely conducted in glovebox containments. As gloveboxes with their high alpha inventories have to be protected against both overpressure and excessive depression under fault conditions, they are separately vented from the COG system.

The glovebox extracts are fitted with vortex amplifier (VXA) control devices operating under ambient pressure control to provide a controlled normal operating depression with small gas purge flows. VXAs and other fluidic devices are used extensively in THORP because their maintenance-free operation is invaluable in a highly radioactive environment. If a glovebox containment develops a leak or is breached the VXA device automatically opens to provide enhanced containment flow through the breach. These devices have proved highly reliable on both plutonium glovebox and alpha laboratory analytical suites in existing Sellafield facilities.

The glovebox extracts are collected together and pulled through primary and secondary HEPA filters prior to discharge up the 125m stack in a dedicated flue.

Cell and Cave Ventilation

The plant is designed on the principle of segregation of ventilation streams of different activity levels to restrict the spread of contamination and minimise the amount of highly active waste materials.

Cell and cave air is extracted through fully shielded ducting to two active filter caves located in the centre building where the streams are filtered through 2-stages of HEPA filtration prior to discharge through the main stack. Several cells with minimal contamination potential have local manual-bag-change, circular filters, as do cells with predominantly alpha-bearing streams.

To prove the concept of the filter cave, prototype filter modules were constructed to minimise the pressure drop and improve the isolation damper maintenance problems. The grab assembly, which is essential for reliable filter change operations, underwent several revisions which were

extensively tested to ensure the action was positive and reliable over a long period in a hazardous environment.

Building Ventilation

The largest proportion of THORP building is devoted to normally accessible work-areas and interconnecting corridors. The air required for ventilation purposes is supplied filtered at a constant temperature of 19°C in winter. Overall air change rates are inappropriate for use on large plants and would have resulted in an expensive and uneconomic installation. Each ventilated space within the THORP building has had its requirement for fresh air assessed on the need for heat, fume or humidity removal, occupation level and function. This approach has resulted in a smaller, more economic ventilation plant able to meet the large variations in individual room requirements.

To maintain the principles of containment, air is transferred from zones of low contamination potential to zones of higher contamination potential, thus minimising the spread of activity. Air extracted from the various zones is collected in separate dedicated ducting systems. In the case of the building air from beta-gamma areas, this is discharged unfiltered from short ducts terminating above the building parapet height. Air from alpha areas where there is normal access is passed through single stage HEPA filters before discharge from the main stack.

Air from limited access areas is extracted through single-stage HEPA filters in manually changed (bagged out) housings and discharged to atmosphere from the main stack. Limited access alpha areas are filtered through two stages of HEPA filters before stack discharge.

IV Specific Design Issues

The Role of Mathematical Modelling

A number of computer-based mathematical models were used in the design of the THORP ventilation systems. Some were commercial computer programmes such as PIPENET and others bespoke programmes such as HYPO which was used for analysing the effects of pulsed air flows on depression control. One of the most useful bespoke programmes was a suite of computer codes called NOXSIM to model the complex behaviour of NOx used in various process stages in THORP.

The NOx chemistry and absorption mechanisms are complex involving up to 10 reacting species with 7 reversible reactions. The NOXSIM model was validated against experimental data obtained both from the DOG pilot plant and from a caustic scrubber on a nitric acid recovery plant at Sellafield. The validation confirmed accuracy to within 330 ppm at ambient temperatures and normal flow rates.

As indicated previously the design of the caustic scrubbers in both DOG and COG and the acid recombination column in DOG was largely based on NOXSIM performance predictions. The most dramatic benefit was a reduction in the packed height of the DOG columns by a factor of nearly two compared with earlier design predictions. The impact of this was a marked reduction in both the cell height and the overall THORP building height. Other benefits have been the recognition of the importance of controlling the nitrous acid concentration in the recombined acid column (to improve its performance) and increasing the NO_2 : NO ratio before caustic scrubbing, by means of the noxidiser, to achieve high scrubbing efficiency for nitrogen oxides.

COG System Depression Control

The pressure in the ullage spaces of the process vessels and the interconnecting transfer routes is controlled at around 10mbarg below atmospheric, which is below the pressure of the surrounding cells so that any air or gas leakage is into the primary containments. The control system has to maintain this depression in a system of over 7000 m^3, with air inputs varying from 900 m^3/hr under plant shut down conditions to 8000 m^3/hr under peak operating conditions, with air inputs from fluidic pumps, agitators and pulsed column drive legs.

The control system employs a transducer-controlled VXA (TC-VXA). This is a device based on the glovebox control VXAs with the ventilation duct depression measured by a transducer which controls a valve in a compressed air line routed to the VXA control ports. The attached line diagram (Figure 6) illustrates the control links. The control air flow is up to 2000 m^3/hr at plant shut down which is added to the plant arisings of 900 m^3/hr. As the plant flow increases the control flow reduces.

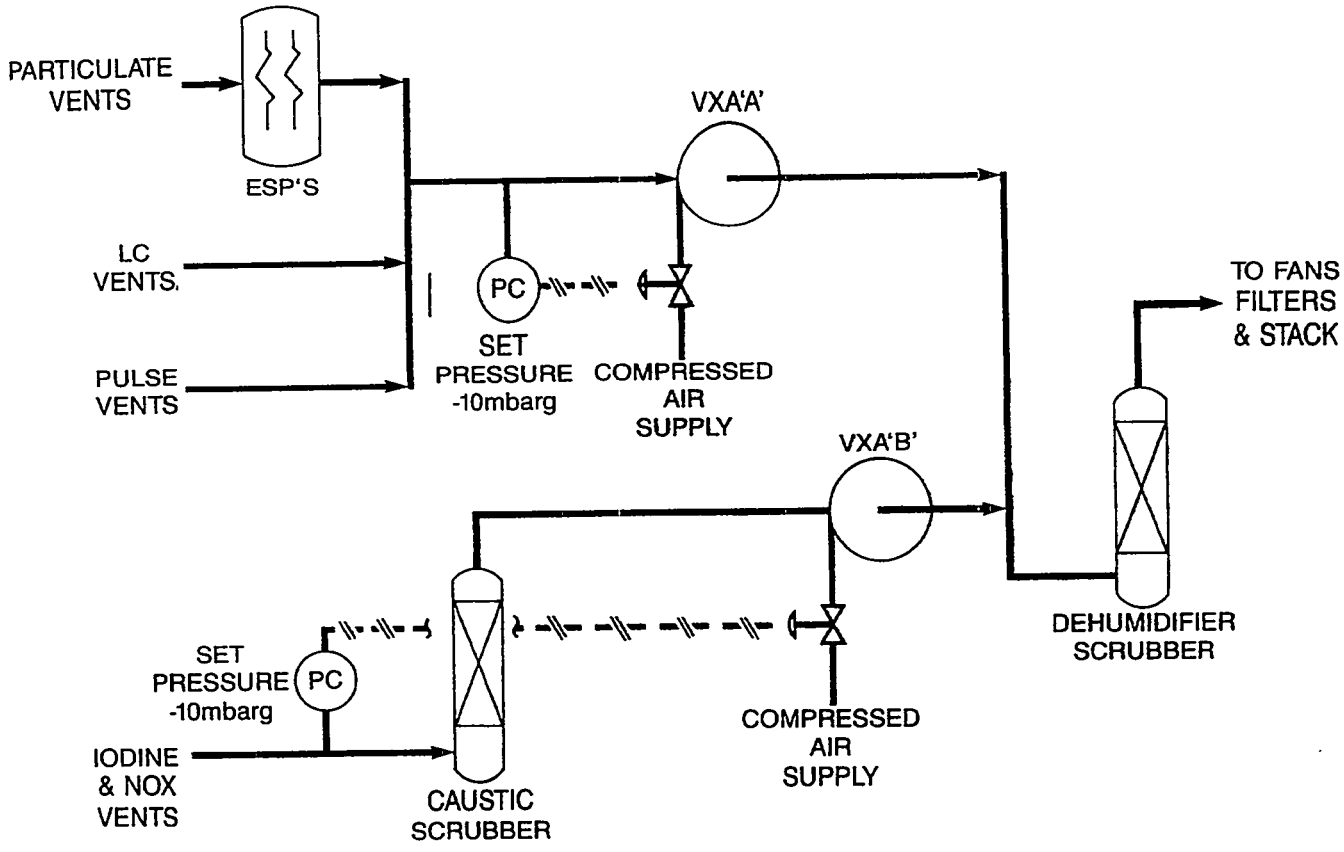


Figure 6 Vessel vent depression control system

The pressure drop through the caustic scrubber is significantly higher than through the other treatment devices and hence this stream is separately controlled with its own TC-VXA. The remaining streams with similar pressure drops are controlled by a single TC-VXA device.

The TC-VXA was developed and tested on an inactive laboratory test rig which demonstrated that close control could be maintained on ventilation system depression over the large flow ranges required with short response times. The transducer control circuit and control valve are duplicated and the air supply is reinforced to provide a high degree of reliability. The failure mode leads to high depression which is considered preferable to loss of depression.

HEPA filtration

Circular HEPA filters are used extensively on THORP to clean up contaminated airflows. For low contamination air streams, unshielded manually changed canister-type units are employed which allow filters to be changed off-line whilst maintaining containment by the use of fully sealed bagging-out techniques. Single or two-stage filtration may be required, depending on the DF requirement. A typical canister-type safe change unit is shown in Figure 7.

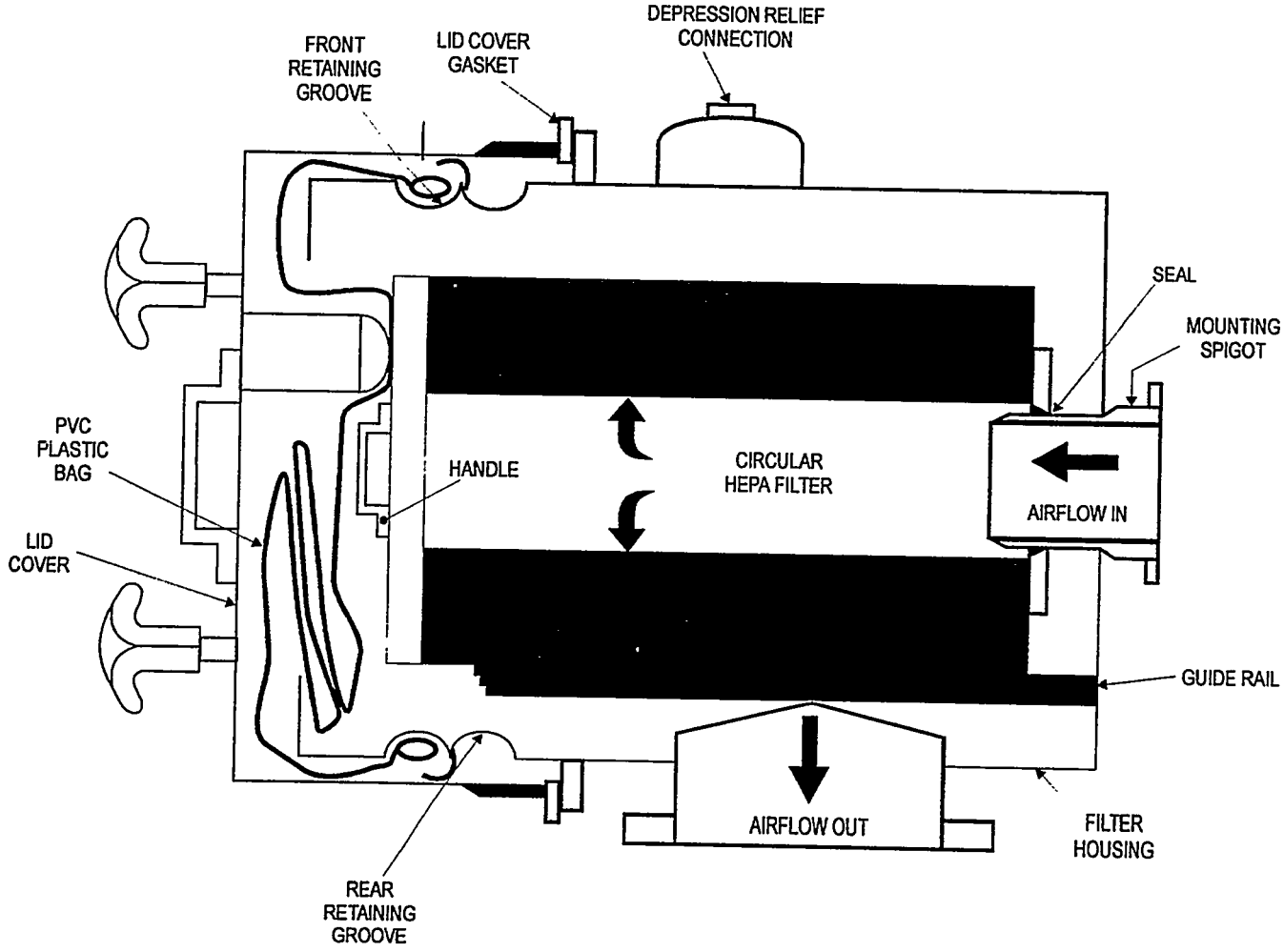


Figure 7 Typical HEPA filter

On remote shielded installations, multiple filter units are installed in purpose-built filter modules located in dedicated filter caves (shown in Fig 8), serviced by an overhead crane-mounted, power manipulator/hoist assembly.

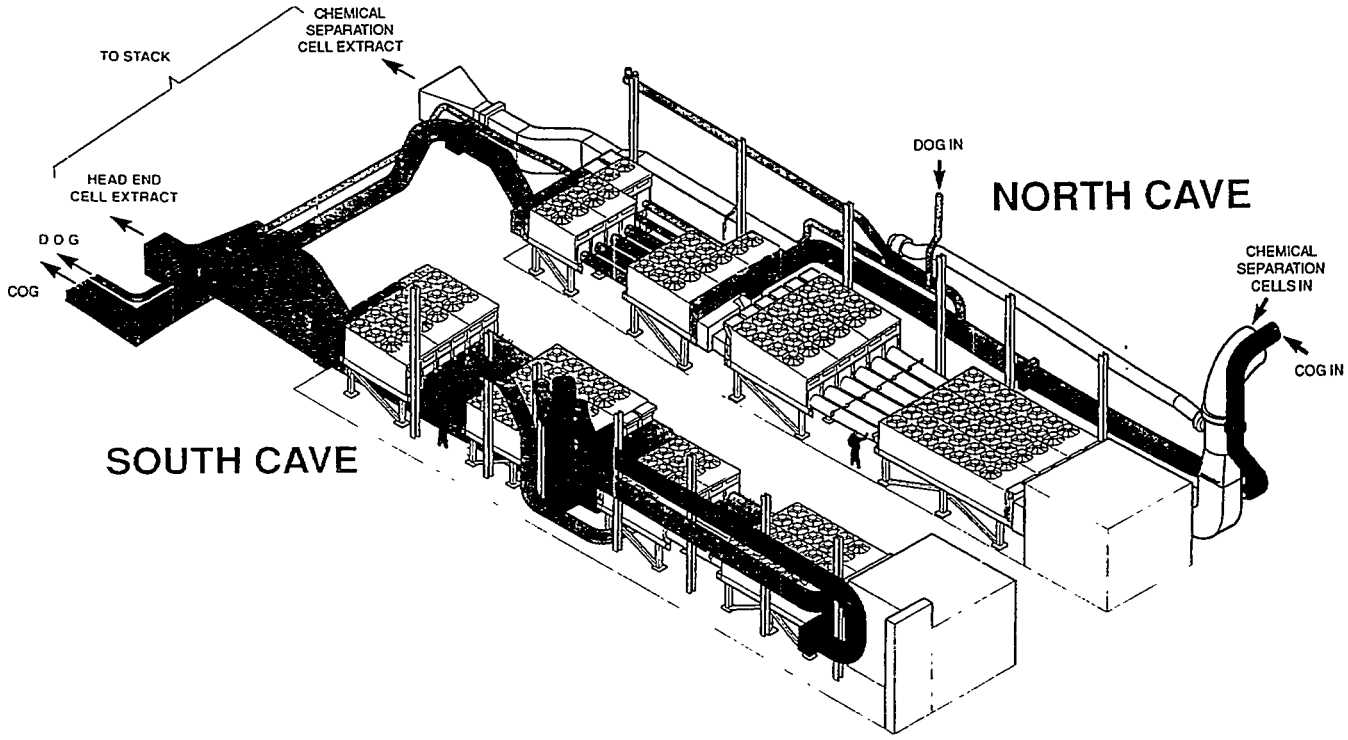


Figure 8 Filter cave isometric

The filters are changed after the removal of the filter access lid (using the hoist and auto-latching grab assembly), allowing the filter element to be removed from the module by use of the same grab. Refitting of the clean element and replacement of the lid is a reversal of this procedure. Dirty filters are posted out of the facility through a gamma-gate and into a flask, having first monitored the spent element for activity level before ultimate disposal. Clean filters are posted into the facility utilising the shielded and protected crane maintenance/decontamination facility. The central control room supervises all operations within the filter caves, the filter-change operation being remote-controlled from local stations situated adjacent to the caves themselves. Operations within the caves are normally overviewed from the window stations with remote CCTV enhancement for such operations as filter seat cleaning.

Stack Dispersion Modelling

The first detailed flowsheets identified a requirement for a stack capable of discharging 485,000 m^3/hour of effluent at a discharge height of between 80m and 120m based on a building height of 35m. A multi-flue connected to the main plant (centre building) was proposed, the main ducts being contained within a bridge section which would be used for flow measurement and sampling/monitoring.

Wind tunnel tests were carried out to assess the performance of both stacks (80m and 120m) on a 1:200 scale site model, with varying windspeeds and directions and a range of discharge flue

velocities. Due to the lack of sufficiently accurate gas concentration measurement equipment at this time, much of the work was done using high speed camera techniques with a neutral buoyancy smoke tracer in the flue discharge. The 120m high stack, employing a discharge velocity in excess of 30m/sec gave an effective height of 120m based on the visual appearance of the plume.

Further wind tunnel tests were carried out in 1988 to verify the performance of the stack based on final flowsheet values and using modern Flame Ionisation Detector systems which can detect levels an order of magnitude smaller than the earlier technology. In the interim period, the stack design had been simplified to one central flue of 3m dia housing the major ventilation streams but incorporating COG and DOG streams at 108m level. The glovebox discharge joined the main flue at 108m level but continued inside the flue in its own discharge pipe to prevent back-pressure.

The later tests were used as a basis for quantitative effective height assessment, and demonstrated an effective height of only 85m for a 120m stack when buildings are present and in the most limiting weather conditions. Further tests with the model were able to show that an additional flue extension of 5m to give a height of 125m would produce an effective discharge height of 92m, which was sufficient to keep the critical group dose impact well below the target value and ensure the plume clears all site buildings. Thus the final design has a flue discharge height of 125m, a windshield roof level of 115m and a flue velocity of 34m/s.

Reverse Flow Protection

The use of a common flue provided the potential for pressure reversals on any failed fan system where isolation dampers did not close. This could result in a back-flow of contaminated air down the affected system. To minimise this potential, duplicate (inlet/outlet) actuated isolation dampers were fitted on some fan systems, spring closure was included on actuators serving critical isolation dampers and diverse failure detection systems were incorporated.

To protect manned areas against pressurisation from adjacent containment areas of higher contamination potential, filters are included on engineered inlet connections to cells and caves. Provisions are also incorporated to check DFs of filters in situ in these installations.

V Commissioning/Testing Experience

At the time of writing, only inactive equipment commissioning experience was available.

The building ventilation systems were commissioned during 1991 and early 1992. The main air flow balances were within the design target and equipment performed to specification. The accessible area systems could not be finally commissioned and balanced until the non-access area systems had completed their initial stage of setting to work and inleakage measurement. This process was accelerated by using temporary fan rigs to exhaust cells and caves to enable constructional deficiencies to be made good. It proved to be a lengthy process to achieve the desired degree of sealing but the integrity of the cell containment eventually established was excellent - typically ≤ 0.5 air changes/hour at 2mbars depression in a cell. The variable-speed fan drives enabled the systems to be adjusted to keep within volumetric and depression control tolerances whilst compensating for filter dirtying characteristics.

The final balanced airflow requirements for the Head End caves and cells was 68% of the design figure and 73% on the Chemical Separation cells. This gave a substantial reduction on design discharges which increased the usable margin on filter/fan capacity.

The DOG system was originally intended to function under depression control from the two 100% fans linked to a pressure controller situated in the common duct downstream of the dissolvers. This, however, proved to be difficult to control due to inleakages in the shear chute and dissolver lid areas. Increasing the fan speed increased the airflow which caused flooding of the reflux condensers without increasing the depression at the shear. The inleakage problem was solved by improving the mechanical joints on the articulated shear chute and fitting dissolver lids with closer tolerances.

Within the COG system, VXAs on the pulsed and non-pulsed headers have proved to be reliable and effective. The system was commissioned with few problems and although minor control changes have been incorporated into the fan start/stop sequences, the overall system has performed as designed. The successful use of fluidic controls in this application has confirmed the advantages to be gained by having non-maintainable equipment in a highly active environment.

On filter testing, DFs across both types of filter module have exceeded expectation. On canister-type (unshielded) units, DFs in excess of 10^3 have been achieved, and 10^4 has been the norm. In remote change, fully-shielded installations, 10^3 - 10^4 per stage has been measured and on two stages, DFs have exceeded 10^6 generally.

VI Conclusions

1 BNFL's Thermal Oxide Reprocessing Plant provides a major off-gas treatment challenge.

2 This challenge has been met by subdivision of the arisings into discrete treatment trains, viz: dissolver off-gas, central off-gas, gloveboxes, cells and caves, and building ventilation.

3 Each stream is treated by a combination of process steps appropriate to its composition and the required degree of decontamination.

4 Development work on key parts of the off-gas treatment system, supported by theoretical modelling, has ensured that plant design has a firm foundation.

5 All active streams are discharged, after treatment, from a 125m stack whose effective height has been demonstrated to be 92m.

6 Modelling work has demonstrated that the maximum critical group dose from THORP aerial discharges will be 22μSv compared with a design target of 50μSv.

7 As illustrated in Table 1, all the discharge targets on individual radionuclides specified in HMIP's Discharge Authorisation will be met.

8 Although commissioning experience is limited, early indications are that confidence in the design of the THORP off-gas system is well founded.

DISCUSSION

EVANS: I am curious as to what is limiting the decontamination factor of iodine in your caustic scrubber. Can you give me an idea what the residence time was and what sort of bubble size you had in there?

HUDSON: When you say what is limiting it, did you expect a higher DF than a hundred?

EVANS: I have no idea what sort of flows you are dealing with and what the bubble size is. I am just curious.

HUDSON: The gas residence time in the THORP caustic scrubber is 2 minutes. Since iodine absorption is gas-phase controlled, the gas residence time is an important factor in determining the DF. The flowsheeted DF of 100 is thought to be conservative because measured DFs of several hundred were achieved during pilot plant trials. Provided organics are excluded, the flowsheet DF should be exceeded during plant operation. There will be a limit caused by a small proportion of organic iodine being present, which will not be removed very readily.

MYERS: I was surprised to learn that the government allows you to not treat noble gases. Is this typical, or is it that you just do not have enough to be concerned about?

HUDSON: There are various aspects of this question. The impact of noble gases on the critical group is very small, it is less than five percent of the total impact on the critical group. We are already less than a half of the discharge target so there was no concern there. What is perhaps more difficult is the societal risk caused by noble gas release. There is no proven full-scale method of removing noble gases. Lots of people have done development work on it and there are prototypes in operation but there is no full-scale operating plant. We did estimate what it would cost to build and operate such a plant based on German technology, and the cost came out to about a hundred million pounds. Based on that figure, the dose detriment saved did not justify spending the money. As a precaution, the plant is designed to enable the backfitting of a krypton removal process, should suitable technology be developed in time.

DEVELOPMENT OF SILVER IMPREGNATED ALUMINA FOR IODINE SEPARATION FROM OFF-GAS STREAMS

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Abstract

An inorganic iodine adsorbent, silver impregnated alumina (AgA), has been developed to separate iodine effectively from off-gas streams of nuclear facilities and to decrease the volume of waste (spent adsorbent). Iodine removal efficiency was improved at relatively high humidity by using alumina carrier with two different pore diameters. Waste volume reduction was achieved by impregnating relatively large amounts of silver into the alumina pores. The developed adsorbent was tested first with simulated off-gas streams under various experimental conditions and finally with actual off-gas streams of the Karlsruhe reprocessing plant. The decontamination factor (DF) was about 100 with the AgA bed depth of 2cm at 70% relative humidity, which was a DF one order higher than that when AgA with one pore size was used. Iodine adsorption capacity was checked by passing excess iodine into the AgA bed. Values were about 0.12 and 0.35 g-I/cm³-AgA bed for 10 and 24wt% silver impregnated AgA, respectively. The results obtained in this study demonstrated the applicability of the developed AgA to the off-gas treatment system of nuclear facilities.

I. Introduction

Nuclear facilities, such as nuclear power plants and reprocessing plants, separate radioactive iodine from off-gas streams and then release the off-gas to the environment.[1-3] Nuclear fission reactions of reactor fuel material produce several iodine isotopes. Most iodine isotopes have short half lives and/or low fission yields, except for ^{129}I and ^{131}I. ^{129}I is the more important because of its long half life. But ^{131}I should be radiochemically considered when dealing with short decay times and/or in the presence of spontaneous fission nuclides such as ^{244}Cm.[4] Account must also be taken of the chemical amount of a stable isotope, ^{127}I, which has a fission yield about

one third that of ^{129}I.[1]

A number of iodine separation methods from off-gas streams are known and have been summarized in the literature.[1,5-6] Scrubbing the off-gas streams with acid or alkaline solution can absorb iodine into the solution. These liquid scrubbing methods need large amounts of solution, so beginning about 15 years ago the authors started to develop an adsorption method with iodine adsorbent. Although several solid adsorbents were known at that time, such as activated carbon and silver zeolite,[5-6] much room remained for improvement. The authors finally developed a silver impregnated alumina adsorbent (AgA) after various experimental studies.[7-15] This paper reviews the development process for the adsorbent and the demonstration of its suitable characteristics for iodine separation in off-gas treatment systems of nuclear facilities.

The adsorbent development placed an emphasis on effective iodine separation at high relative humidity. Water vapor was found to clog the small pores of the adsorbent carrier, activated alumina, and to interfer with the reaction of iodine and impregnated silver. So mainly, the pore distribution was considered to avoid clogging of pores with impurities. In this case, effectiveness of the iodine separation was evaluated with a parameter, decontamination factor (DF), which is defined as the iodine concentration at the inlet of the adsorbent column divided by that at the outlet.

After developing the optimum adsorbent, laboratory tests were carried out to check the effects on DF of various conditions, namely iodine form, concentrations of iodine and impurities, and linear velocity of iodine gas. A plant test was finally carried out in the dissolver off-gas treatment line of the Karlsruhe reprocessing facility in Germany.

Iodine adsorption capacity was investigated by passing simulated off-gas with a relatively high iodine concentration through the adsorbent column under various conditions. Adsorption capacity should strongly relate to the waste (spent adsorbent) volume. Iodine release property was also studied from the adsorbent with a saturated amount of iodine, considering the interim storage of spent adsorbent.

II. Experimental

Activated alumina was used as a carrier (base material) of the adsorbent. Industrially obtained aluminium sulphate was the starting material for the activated alumina. Addition of ammonium hydroxide solution into aluminium sulphate solution produced aluminium hydroxide precipitate. The precipitate was filtered and washed by demineralized water to remove ammonium and sulphate ions. The dried precipitate was calcinated at 600-800℃ to get aluminium oxide (alumina) and, if necessary, crushed to smaller particles and calcinated again. Crushing and second calcination processes provided alumina with different pore diameters (distribution). Then, the alumina was put into the silver nitrate

solution, discharged from the solution and dried in air to get the final product of silver impregnated alumina (AgA) adsorbent. Repeating the impregnation of silver nitrate yielded AgA with different silver contents. The AgA tested here had pore diameters of 10 to 100nm, silver contents of 3 to 30wt%, particle diameter of about 2mm, specific surface area of about 10m^2/g and bulk density of about 1.5g/cm^3.

The test apparatus is schematically shown in Fig. 1. The iodine species were methyl iodide and elementary mainly labelled with ^{131}I. Iodine, air, steame-air and nitrogen oxides were mixed to provide various conditions of simulated off-gas. The gas temperature was kept constant at 150℃ before making contact with AgA in an adsorption column. The column was divided into several beds to get the decontamination factor (DF) and iodine distribution (adsorption profile) in a column. Iodine concentrations at the inlet and outlet of the column were measured with an NaI(Tl) scintillation counter or spectrophotometer after absorbing the iodine in an alkaline solution. ^{131}I in each bed was measured with an NaI counter after removing the bed from the column. The DF value was determined by equation (1),

$$DF = [I]o/[I]x \quad (1),$$

where [I]o and [I]x are iodine concentrations at the inlet and outlet of the column (bed) of x cm length, respectively. As the iodine concentration was constant during the test and the iodine amount introduced into the column was quite low compared with the adsorption capacity of the adsorbent, adsorbed iodine amount in all beds and in beds after x cm should be propotional to [I]o and [I]x, respectively. Then, DF value could also be obtained by measuring iodine amount in each bed.

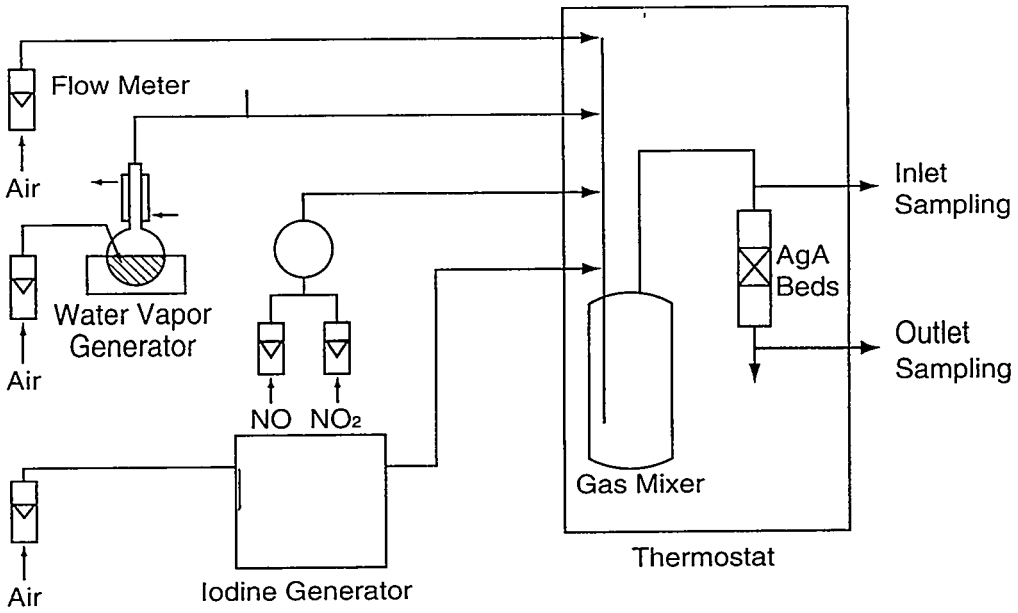


Fig.1 Outline of the Test Apparatus

The DF value was finally checked by putting AgA in a column of the dissolver off-gas sampling line of the Karlsruhe reprocessing plant. Average iodine concentration, relative humidity and NOx concentration of the actual off-gas were 0.1volppm, 5vol% and 4vol% at the column, respectively. Radioactivity of ^{129}I was measured in each bed with a pure Ge detector.

Adsorption capacity for AgA was determined from the adsorption profile after passing simulated off-gas with sufficient iodine quantity into the column. Stability of adsorbed iodine was checked by blowing air onto AgA with a saturated amount of iodine. The temperature of the air was held constant at 28℃ and 50℃ during the desorption test. Desorbed iodine was trapped in two stages, containing alkaline solution. Iodine in the solution was measured by an ordinary photospectrometric method using ferric thiocyanate. Desorption ratio, which is defined by the amount ratio of desorbed iodine to initial adsorbed iodine, was calculated to evaluate the stability of adsorbed iodine.

Tests conditions were set to cover the range actually encountered in iodine removal operations at nuclear power and reprocessing plants. Severer conditions, such as high impurity concentrations, were also included. A severe desorption test was carried out at higher temperature than usually considered for adsorbed (spent) iodine adsorbent.

III. Development of Silver Impregnated Alumina Adsorbent

The optimum pore diameter (distribution) was first determined for activated alumina carrier with constant silver content of 10wt%. From preliminary experiment results, three kinds of AgA were prepared with main pore diametes of ~15nm, ~60nm, and both ~15nm and ~60nm. These AgA's were tested to compare the DF values at different relative humidities. The results are shown in Fig. 2. AgA with ~15nm pores was efficient at low relative humidity, but showed a low DF value at higher relative humidity. By contrast, AgA with ~60nm pores was not so efficient at low relative humidity, and it did not show such an abrupt DF decrease at higher relative humidity. This could be explained by clogging of small (~15nm) pores by water molecules at high relative humidity. Iodine reacts with silver impregnated in small pores and is effectively adsorbed onto AgA with small pores. But at high humidity, small pores are occupied by water, which disturbs the reaction of iodine with silver. Large (~60nm) pores do not promote iodine reaction with impregnated silver at low humidity and they can avoid occupation by water, which mean there are relatively constant DF values for different humidities. Thus, the combination of small and large pores was predicted to be suitable for iodine separation over a wide humidity range. The results confirmed this prediction as shown in Fig. 2.

The DF value dependency on silver content was next examined at higher relative humidity. The results are shown in Fig. 3, which shows DF increased from 3 to 10 wt% and decreased from 24 to 30wt% of the AgA. This could be

explained by an effective surface area change for the reaction of iodine with silver. Small pores should be covered by water under this condition. AgA with lower silver content provides an insufficient reaction surface area for a small silver amount. AgA with higher silver content also has a small surface area because large pores are covered with a large amount of silver nitrate. Thus, the optimum silver content for AgA was found to be in the region between 10wt% and 24wt%.

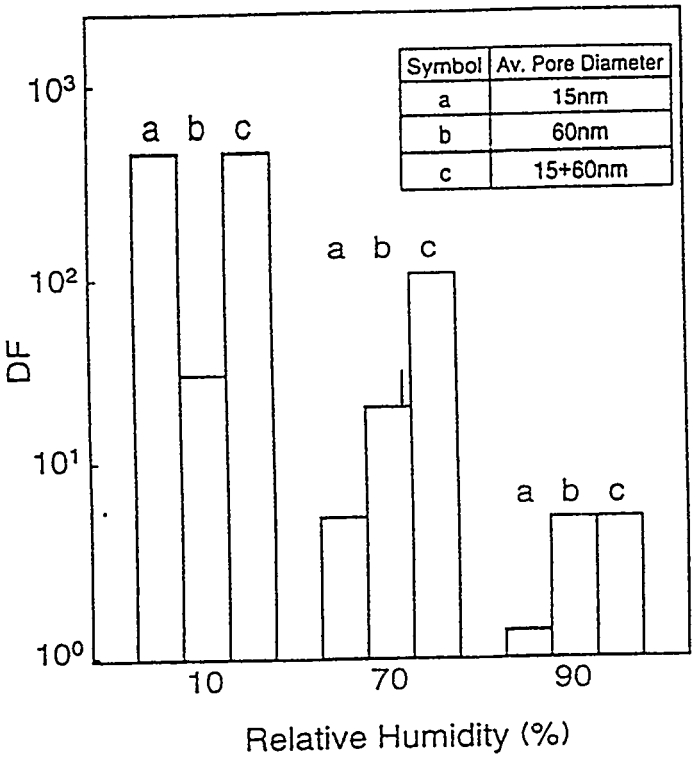


Fig.2 DF of AgA with Different Pore Diameters at Different Relative Humidities

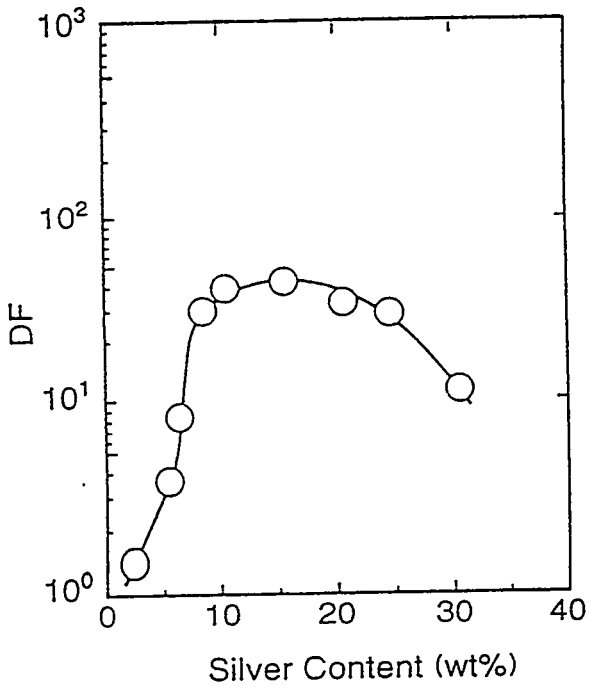


Fig.3 Dependency of AgA DF for CH_3I on Silver Content at 90% RH and 30℃.

IV. Decontamination Factors at Various Conditions

Typical iodine forms selected here were methyl iodide (CH_3I) and molecular iodine (I_2). CH_3I is known to be the most difficult species to separate from simulated off-gas among the major organic iodines, and I_2 is the main form in the dissolver off-gas of spent fuel reprocessing[1,13]. The DF values for both iodines with AgA are shown in Fig. 4 as a function of gas temperature. I_2 showed a higher DF because the reactivity of CH_3I with silver would be low. DF values of over 100 could be obtained for both iodines at 100℃ and 150℃ with AgA of 3cm bed depth. Temperatures below 100℃ lowered the DF values for both CH_3I and I_2, but NOx of 1.5vol% did not lower the DF values in this experiment.

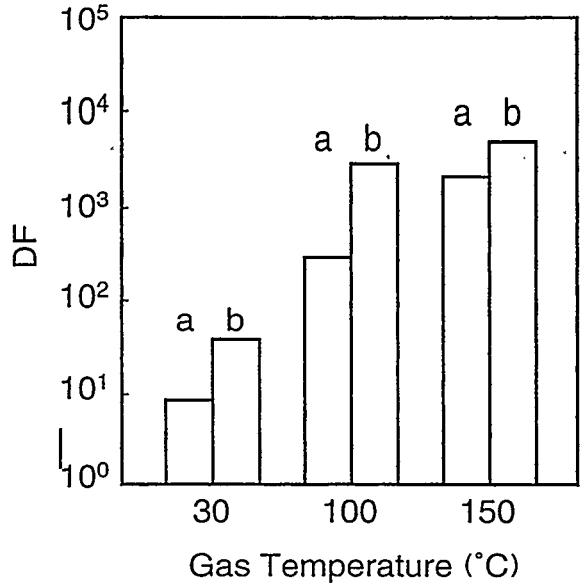


Fig.4 DF for CH_3I (a) and I_2 (b) as a Function of Gas Temperature

The DF dependencies for CH_3I and I_2 were also investigated for several conditions of simulated off-gas other than temperature. The results are shown in Figs. 5 to 9 which were obtained by changing parameters of iodine concentration (standard condition: ~20volppm), linear gas velocity (20cm/s), AgA bed depth (3cm), NOx concentration (0vol%), and H_2O concentration (0vol%), respectively. The parameters were widely changed for I_2 compared with CH_3I. When the iodine concentration at the column outlet was under the detection limit, a DF value was calculated by substituting the detection limit for [I]x in equation (1); these values are indicated with an upward pointing arrow in the figures. Although most DF values were the minimum ones, no apparent dependencies were observed for any parameters except AgA bed depth. CH_3I seemed to show relatively lower DF values than I_2, which was also observed in Fig. 4. DF values of over 10^3 were obtained for both iodine forms for any combination of conditions in Figs. 5 to 9 if the AgA bed depth was larger than 3cm. Thus it was concluded that DF for CH_3I and I_2 with AgA was independent of various parameters, except bed depth and temperature.

Finally the plant test was carried out to check the properties of AgA and to demonstrate its applicability to an actual reprocessing plant. The results are shown in Fig. 10 after loading dissolver off-gas of the Karlsruhe reprocessing plant into a column of AgA beds. DF values were measured for different bed depths and operation times. The DF values depended on bed depth and were independent of operation time up to 118 days. DF values of over 10^2, 10^3 and 10^4 were obtained for bed depths of 2.5, 5 and 10cm, respectively. DF values obtained here were almost the same as those obtained in laboratory tests. Little DF change during 118 days operation confirmed the stable separation characteristics of AgA. In all cases, DF of the 10cm AgA bed depth was more

than 10^3 in the presence of NOx and water vapor in off-gas of the actual reprocessing plant, which revealed that AgA could be applied in an actual off-gas treatment system.

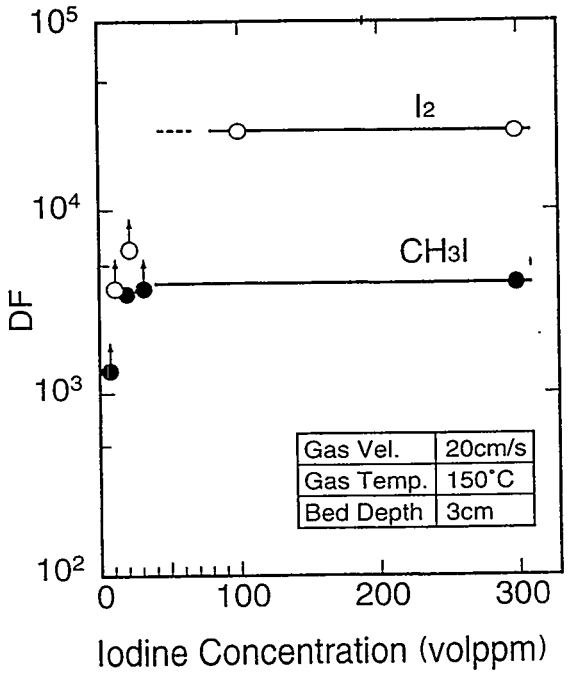


Fig.5 DF Dependency on Iodine Concentration

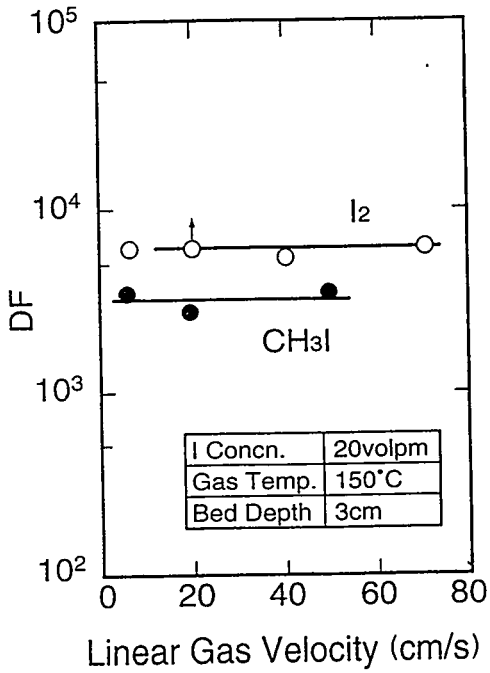


Fig.6 DF Dependency on Linear Gas Velocity

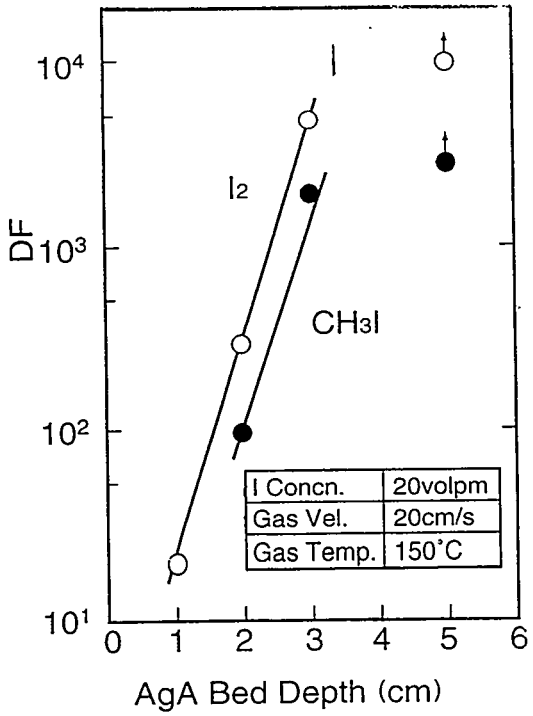


Fig.7 DF Dependency on Bed Depth

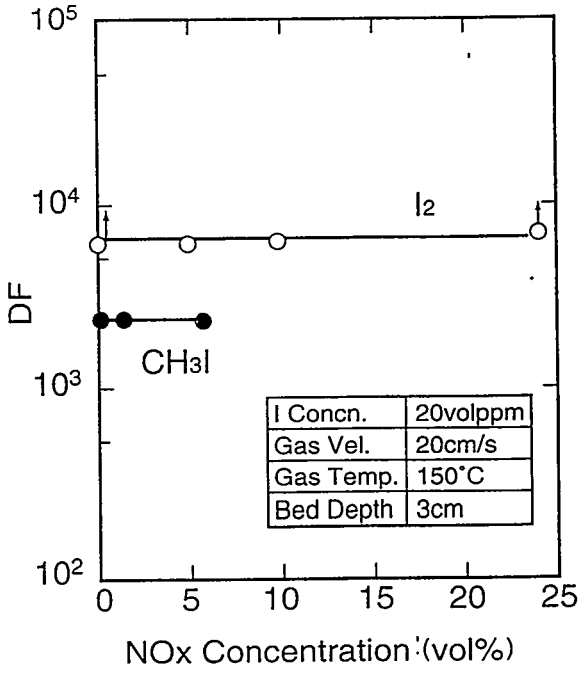


Fig.8 DF Dependency on NOx Concentration

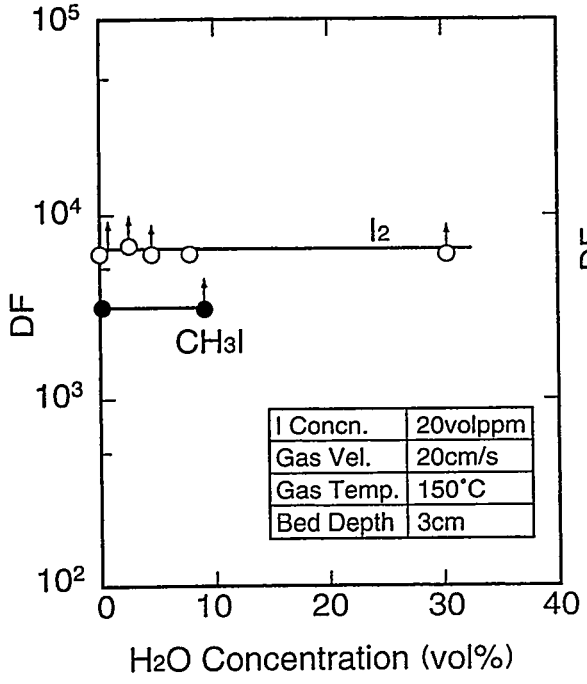


Fig.9 DF Dependency on H_2O Concentration

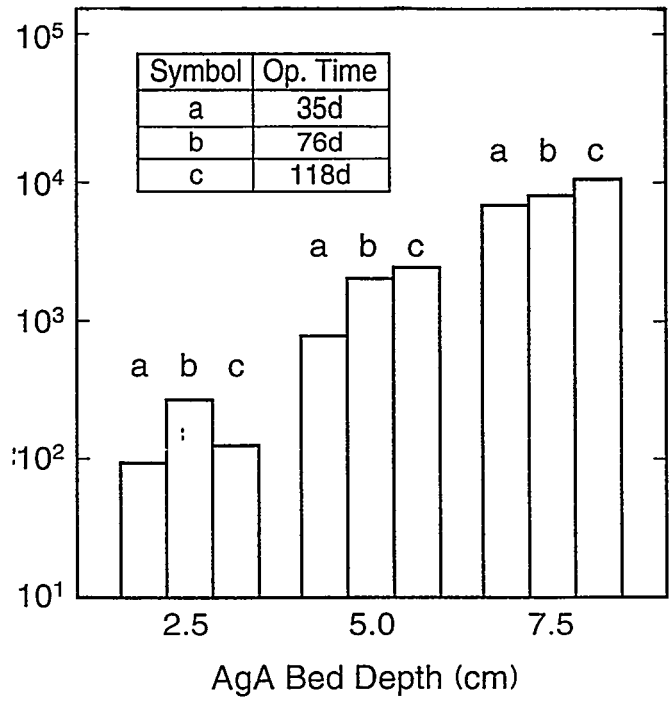


Fig.10 DF Dependency on Bed Depth and Operation Time Obtained in the Karlsruhe Plant Test

V. Iodine Adsorption Capacity of AgA

In order to estimate the volume of waste (spent AgA), iodine adsorption capacity of AgA was checked by adsorbing iodide onto AgA. The results are shown in Fig. 11 for AgA's with silver contents of 10wt% and 24wt%. Both CH_3I and I_2 adsorptions showed the same results. The vertical axis of Fig. 11 is the adsorbed weight of iodine per unit bulk volume ($1cm^3$) of AgA adsorbent. The shape of the AgA particles is a sphere with a diameter of 1-2mm and their bulk volume is important for estimation of waste volume. Iodine adsorption capacity of AgA depended on its silver content and was about 0.12 g-I/cm^3-AgA(10wt%) and 0.35 g-I/cm^3-AgA(24wt%). Iodine reacts with the same number of moles of silver in AgA and capacity data obtained indicated about 80% silver utilization.

Iodine adsorption capacities could also be calculated for other iodine adsorbents from the published data.[12,16] The capacities were in the region of 0.07 to 0.17 g-I/cm^3-adsorbent for previously developed adsorbents, silver zeolite, silver mordenite and silver silca gel. Thus AgA with 24wt% silver content had a higher iodine adsorption capacity than other adsorbents and it could reduce the waste volume to 1/5 - 1/2. Such a reduction would cut the volume of waste treatment, and size of storage and disposal facilities.

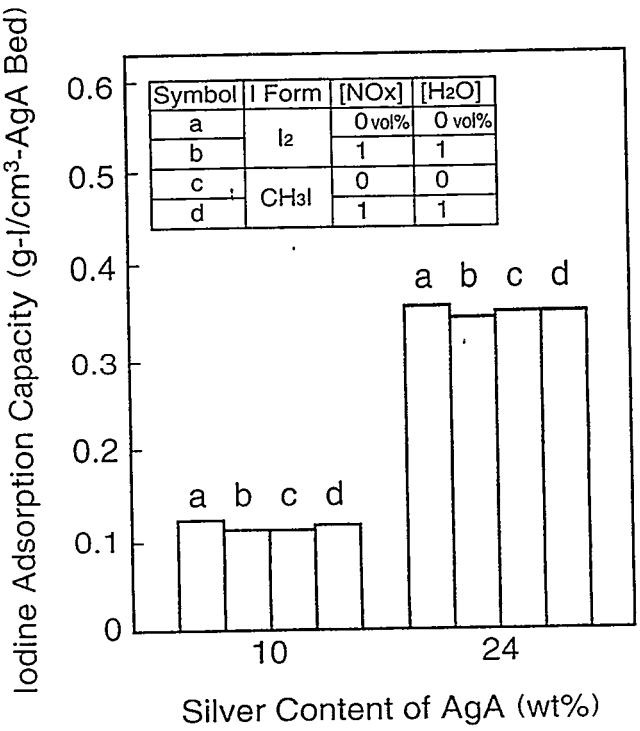


Fig. 11 Iodine Adsorption Capacity of AgA

VI. Desorption of Adsorbed Iodine from AgA

Iodine holding capability for AgA after adsorption was evaluated by desorption tests. Desorbed amount of iodine from AgA was measured by blowing air through the AgA (24wt% silver) with a saturated amount of iodine (I_2) in a column. The temperature was 28℃ and 50℃, and the relative humidity was 100%. The absolute humidity was then 2.4% and 8.6% for 28℃ and 50℃, respectively. The amount of desorbed iodine absorbed in the two series for a 0.05mol/ℓ sodium hydroxide solution was below the detection limit (5ppb) after 144 hours of air blowing. This meant that the desorption ratio was below 0.00011% and 0.00019% at the desorption air temperature of 28℃ and 50℃, respectively. The difference between the two series was caused by the water vapor content difference, that is, much water was absorbed in the alkaline solution at 50℃.

The annual iodine desorption ratio were calulated from these data to be below 0.007%/y and 0.012%/y at 28℃ and 50℃, respectively. This desorption test was carried out under a severe condition such as high humidity and large amount of blowing gas. The flow rate of the humid air was 60cm³/s. As the air volume for this test in a column was about 3cm³, the air was replaced by new air at a frequency of 20 times per second. No iodine desorption was detected even under such severe conditions, which demonstrated the stability of adsorbed iodine in AgA.

Ⅶ. Conclusions

An iodine separation adsorbent, silver impregnated activated alumina (AgA), was developed and its characteristics were investigated to get an effective iodine separation technology for off-gas treatment systems of nuclear facilities. The studies consisted of AgA development and AgA property investigations for iodine separation, iodine adsorption capacity, and iodine desorption. Various tests were conducted and the following conclusions were obtained, which confirmed the applicability of the iodine separation technology with AgA to the off-gas treatment systems.

1. Silver alumina adsorbent with pores of two different diameters had excellent iodine separation capability at both low and high humidities. Silver in smaller and larger pores reacted with iodine at relatively low and high humidities, respectively. Silver content of AgA was also optimized to be in the region of 10-24wt%.
2. Test conditions such as iodine concentration, linear gas velocity, nitrogen oxides concentration and humidity had no effect on iodine separation efficiency of AgA, whereas iodine form, gas temperature and AgA bed depth did. DF values for elemnental iodine were higher than those for methyl iodide probably because of the higher reactivity of I_2 with silver. DF values increased with temperature and bed depth because of the increase of reactivity and adsorption site number, respectively. DF values of over 10^3 were obtained in the actual plant test as well as in laboratory tests.
3. Iodine adsorption capacities were ~0.12 and ~0.35 g/cm³-AgA bed for AgA with 10 and 24 wt% silver contents. These values did not affected by iodine form and impurities. AgA with 24wt% silver content could reduce the waste volume to 1/5 -1/2 compared with previously developed adsorbents.
4. No iodine desorption was observed from AgA saturated with iodine after about 6 days blowing air (60cm³/s) with 100% relative humidity at 28 and 50℃, which demonstrated the stable iodine holding capability of AgA.

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DISCUSSION

WREN: I have a few questions. Did you look at the temperature effect on adsorption capacity? I know you looked at the DF, but what about the adsorption capacity at a high temperature?

KIKUCHI: Adsorption capacity was measured up to 150° C. Adsorption capacity is almost the same in the temperature range of our experiments (30°-150° C).

WREN: I understand that it does not have any temperature dependence. When you looked at the effect of NO_x, did you look at it at various relative humidities? What was the combined effect of relative humidity and NO_x? The adsorption behavior of NO_x on surfaces is likely to change depending on humidity. Thus the combined effect could be quite different from what one would expect from just simple addition of two effects. Could NO_x under dry conditions be more detrimental than NO_x under humid condition, depending on how NO_x is adsorbed?

KIKUCHI: Your question is about the combination of relative humidity and NO_x concentration. I have done the experiments, and the most severe condition is high relative humidity and high NO_x concentration. We have evaluated the combination of factors at about 90% RH and the highest NO_x concentrations. Addition of NO_2 under humid condition lowers more the adsorption capacity than simple addition of two effects, that is, NO_2 under humid condition is more detrimental than NO_2 under dry condition. This might be due to the fact that the formation of NO_3^- in the micropores by NO_2 dissolution to capillary water increases the apparent relative humidity.

WREN: Do you have any idea how NO_x was adsorbed? Was it completely saturated?

KIKUCHI: It is really hard to measure because of the impregnated material, silver nitrate. Unless you have a labelled compound of NO_x, you cannot measure the profile of the

adsorbed NO_x.

WREN: Was there any effect from radiation exposure?

KIKUCHI: No radiation effect was observed on iodine adsorption.

WILHELM: Experiments are not needed for reprocessing plants at 90% RH, because at 150° C it would give several bars pressure in the offgas.

KIKUCHI: We have two problems. One is the reprocessing plant (150° C, ~1% RH). In that case, we don't have to worry about relative humidity, only about NO_x. Eventually we plan to install this adsorbent in gas treatment systems in nuclear power plants, which have a running temperature of about 30°-40° C and 50-90% RH. Therefore we have done experiments at high relative humidity conditions.

TREATMENT OF TRITIATED EXHAUST GASES AT THE TRITIUM LABORATORY KARLSRUHE

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Abstract

The Tritium Laboratory Karlsruhe (TLK) accomplished commissioning; tritium involving activities will start this year. The laboratory is destined mainly to investigating processing of fusion reactor fuel and to developing analytic devices for determination of tritium and tritiated species in view of control and accountancy requirements.

The area for experimental work in the laboratory is about 800 m^2. The tritium infrastructure including systems for tritium storage, transfer within the laboratory and processing by cleanup and isotope separation methods has been installed on an additional 400 m^2 area. All tritium processing systems (=primary systems), either of the tritium infrastructure or of the experiments, are enclosed in secondary containments which consist of gloveboxes, each of them connected to the central depressurization system, a part integrated in the central detritiation system. The atmosphere of each glovebox is cleaned in a closed cycle by local detritiation units controlled by two tritium monitors.

Additionally, the TLK is equipped with a central detritiation system in which all gases discharged from the primary systems and the secondary systems are processed.

All detritiation units consist of a catalyst for oxidizing gaseous tritium or tritiated hydrocarbons to water, a heat exchanger for cooling the catalyst reactor exhaust gas to room temperature, and a molecular sieve bed for adsorbing the water. Experiments with tracer amounts of tritium have shown that decontamination factors >3000 can be achieved with the TLK detritiation units.

The central detritiation system was carefully tested and adjusted under normal and abnormal operation conditions. Test results and the behavior of the tritium barrier preventing tritiated exhaust gases from escaping into the atmosphere will be reported.

I. Introduction

The Tritium Laboratory Karlsruhe (TLK) provides the technical means for R&D work on the fuel cycle and components of the next fusion machine within the framework of the European Technology Program (NET/ITER). Work at TLK will concentrate on the development of advanced processes of plasma exhaust fuel cleanup.(1) In experiments performed on a technical scale up to 10 g of tritium will be handled.

The government license in accordance with Sec. 3 of the Radiation Protection Ordinance, being part of the German Atomic Energy Act, was granted in April 1993 by the State Ministry for the Environment of Baden-Württemberg.

II. Treatment of Tritiated Exhaust Gases

Description of the Multi-barrier Concept of Depressurization and of the Tritium Retention Concept

According to a multi-barrier concept (Figure 1), tritium is prevented from escaping into the ambient atmosphere. Most of the tritium carrying primary systems of the TLK infrastructure and of the experiments have been designed as metal-made standard UHV components. The specified leak tightness is $< 1 \times 10^{-9}$ mbarl/s for the single leak and $< 1 \times 10^{-8}$ mbarl/s for the entire primary system (integral leak).

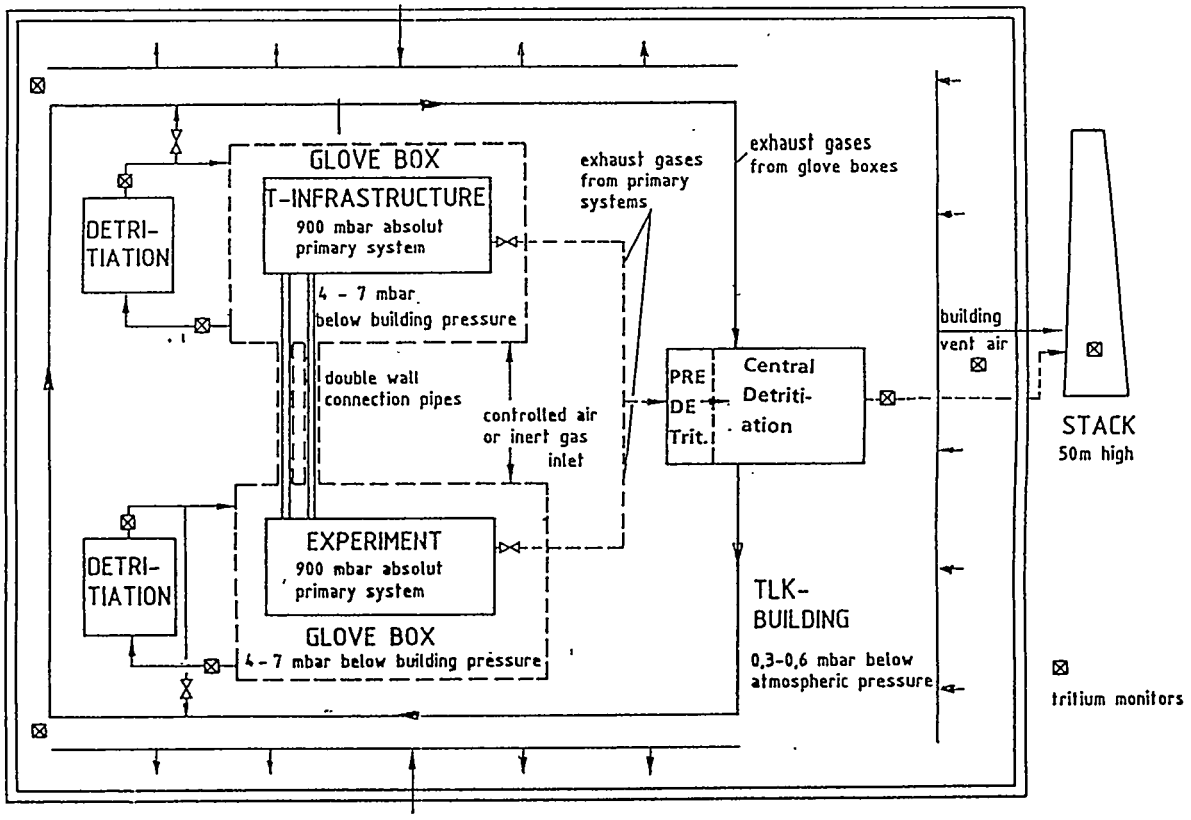


Figure 1 Containment and exhaust gas concept at TLK.

Each of these primary systems is enclosed in a secondary containment (most frequently a glovebox) filled with either air or an inert gas. Typical features of the gloveboxes are stricter requirements on tightness, namely < 0.1 vol.%/h leak rate compared to the DIN standard of 0.25 vol.%/h. The gloveboxes are kept at 4 - 7 mbar negative pressure rela-

tive to the laboratory atmosphere, which, in turn, is kept at a negative pressure of 0.3 to 0.6 mbar relative to atmospheric pressure.

The secondary containments are equipped with individual Tritium Retention Systems (TRS) the exhaust gases of which are passed for safety reasons through a Central Tritium Retention System (ZTS) - likewise responsible for depressurization in the gloveboxes - before they are discharged via the vent air stack. Both systems will be described in more detail below.

Depressurization

The secondary containments are the gloveboxes, which are connected to a central depressurization system (Figure 2). A closed circular pipeline installed in the tritium infrastructure rooms and in the experimental rooms generates through blowers a negative pressure of 11 - 18 mbar relative to the laboratory atmosphere. The gloveboxes release their leaking air via a pressure control valve into the closed circular pipeline (4 - 7 mbar operating range below laboratory atmosphere). Most of the boxes previously installed have a volume of 6 - 12 m^3. About 10 boxes have already been installed. The only exception is the 32 m^3 box of the Hydrogen Isotope Separation System. This means that the amount of air to be exhausted per box is small. However, should a glove be torn off, up to 50 m^3 /h of air would have to flow through the glovebox hole into the interior of the box in order to prevent tritium from diffusing out of the box again. In this case, the higher rate flow is fed into the depressurization system through a safety valve, which opens when the negative pressure in the glovebox has attained approx. - 1 mbar. The leaking air exhausted by the depressurization system from the gloveboxes and from other suppliers is discharged as vent air via the stack.

The safety-related components of the depressurization system - blowers, control valves - are made redundant.

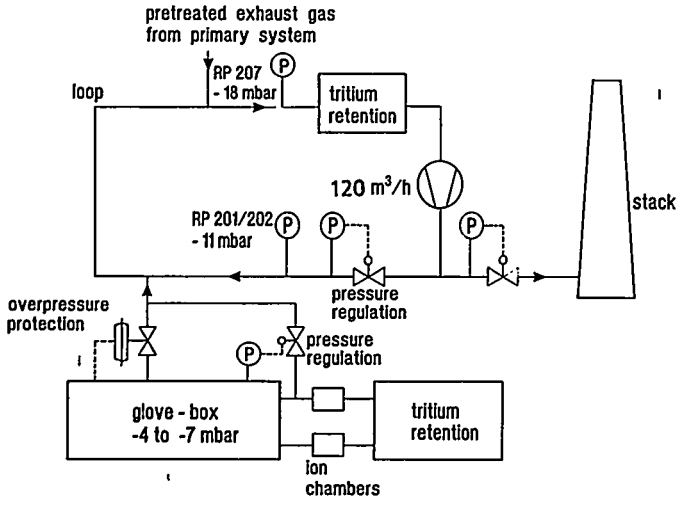


Figure 2 Flow sheet of the Central Tritium Retention System (ZTS).

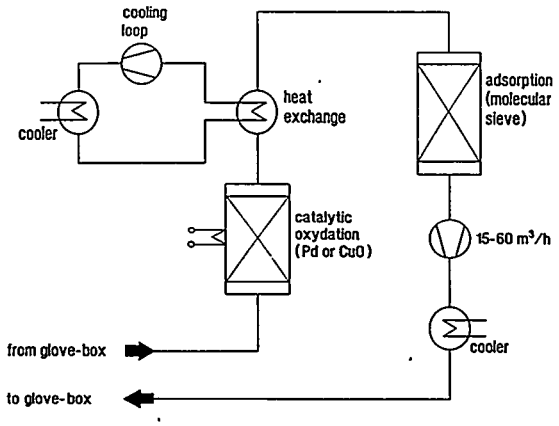


Figure 3 Flow sheet of glove box Tritium Retention System (TRS).

Tritium Retention System (TRS)

Each glovebox is equipped with a separate Tritium Retention System (Figure 3) based on conventional technology and consisting of a catalyst bed for hydrogen and hydrocarbon oxidation (0.5% Pd on alumina carrier or CuO) and a molecular sieve adsorber connected in series.

Depending on the size of the box, the dimensions selected for the Tritium Retention Systems are such that during normal operation the air is changed five times per hour. Should an elevated tritium level be measured in the box atmosphere, the TRSs are switched over to 10 times air change and the operating temperature of the catalysts is raised. The construction sizes (Table 1) are defined by the modular parallel connected expansion of the catalyst and molecular sieve units.

Table 1 TRS construction sizes.

Size	Gasflow	Catalyst	Molecular sieve	Heat exchanger	Refrigerator	Blower
I	15/30 m^3h^{-1}	1	1	1	1	1
II	30/60 m^3h^{-1}	2	2	2	1	1
III[1)]	60/120 m^3h^{-1}	4	4	4	2	2

1) Size III consists of two parallel connected units size II

Dependent on the nature of the processes and experiments, the primary systems do not only contain pure hydrogen species (H_2, HD; D_2, HT, DT, T_2) but also hydrogen compounds such as ammonia and methane. In order to be able to retain these substances too, which escape into the box during malfunctions, the catalysts of TRSs are operated at different temperatures (Table 2):

- Boxes filled with air and processes without hydrocarbons (L) are operated at 100 °C. If tritium escapes, the temperature is raised to 180 °C.
- Boxes filled with air and processes with hydrocarbons (L/CH) are operated at 300 °C. If tritium escapes, the temperature is raised to 450 °C.

Besides, gloveboxes with inert gas (IG) atmosphere are used for experimental reasons. The TRSs connected to this type of box need an oxygen donator for oxidation of any tritium which might escape. In these boxes the Pd catalyst has been replaced by CuO operated at 250 °C. The latter substance must be regenerated periodically with air.

Table 2 Types of catalyst and modes of operation of TRS.

Application	Internal indication	Catalyst material	Regeneration necessary	Operation temperature
TRS for air without hydrocarbons	L	Pd	no	100 °C - 180 °C
TRS for air with hydrocarbons	LCH	Pd	no	300 °C - 450 °C
TRS for inert gas without hydrocarbons	IG	Cu O	yes	200 °C
TRS for inert gas with hydrocarbons	IGCH	Cu O	yes	250 °C

The exhaust gas removed from the secondary containments for depressurization is extracted from the return flow from the TRS to the gloveboxes. Thus, two-stage tritium retention is ensured during normal operation (but not if gloves have been torn off) (Figure 2).

The specified activity of the glovebox atmosphere during normal operation is $< 4 \times 10^6$ Bq/m^3. It is monitored by one tritium monitor each (ionization chamber) in the feed and return flows to and from the TRS. In case of alarm due to rise in activity the gas flow rate is doubled, as stated before.

Central Tritium Retention System (ZTS)

In the Central Tritium Retention System (Figure 2) all exhaust gases of the secondary containment, the pretreated exhaust gases of the primary systems, and all other exhaust gases which might have been contaminated are processed in a single loop. This loop generates the negative pressure for all gloveboxes in the laboratory. A constant gas flow rate of 110 - 120 m^3/h circulates in the loop at a negative pressure (see III). The constant operating conditions are achieved by controlled release of the vent air into the stack.

In the main stage (RHH) of the ZTS (Figure 4) all the gas circulating in the loop is processed in the once-through mode. This is achieved by two redundant cleanup units of the same make and fulfilling the same function as the TRS, construction size III (L/CH) described before. An activity monitor placed at the outlet of these units monitors the clean gas and hence the partial gas stream released to the atmosphere.

When a limit is attained (5×10^6 Bq/m^3), the catalyst temperature is automatically increased from 300 °C to 450 °C. In case the result is unsatisfactory, the second standby main stage of the retention system is connected in parallel. At the same time, operations in the laboratory are restricted by administrative measures which means that all ongoing work in the gloveboxes is temporarily discontinued. Should other malfunctions occur (e.g. failure of the catalyst heating system) switching over takes place to the redundant system.

Different from the TRSs, the two main stages of the retention system are equipped with emergency cooling sets. These are switched on and maintain cooling of the process gas between the catalysts and the molecular sieves in cases where the laboratory coolant water system fails. In this way, the danger is eliminated that the non-cooled gas stream warms up the molecular sieves and releases already adsorbed tritiated water.

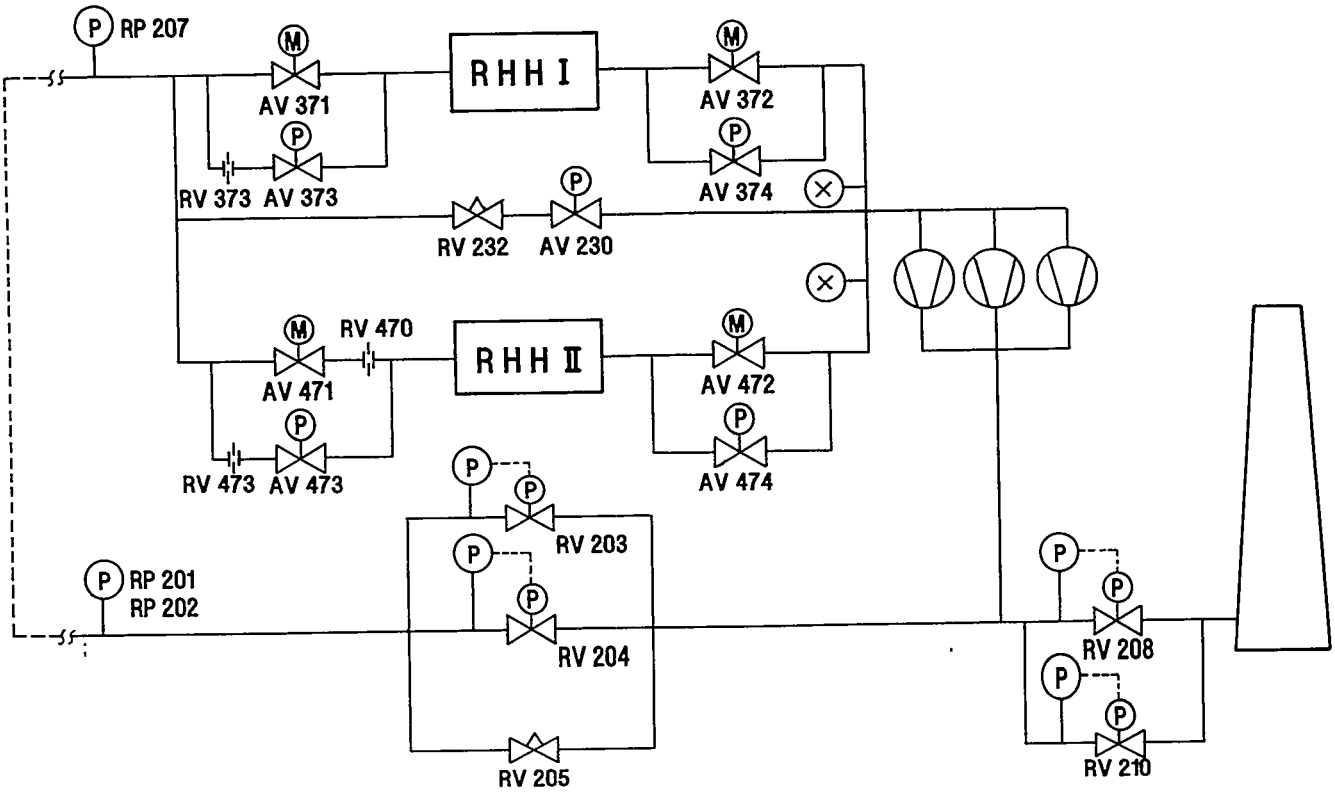


Figure 4 Improved concept of control valves of the closed circular pipeline.

At the preliminary stage (RHV) of the ZTS all exhaust gases originating in the primary systems of the tritium-carrying units at TLK are precleaned, i.e. mainly the exhaust gases from the vacuum pumps. All these exhaust gases are collected in buffer tanks (three units). Following an analysis and in case of necessity, these gases are treated, in the way already described, in a separate Tritium Retention System (TRS construction size I for inert gas). After a residual activity of about 10^{10} Bq/m^3 has been achieved, this gas can be discharged into the closed circular pipeline and into RHH.

These are the alternating modes of operation of the three buffer tanks:

- reception,
- inspection and treatment, if necessary,
- delivery and conditioning for reception of new batches.

As it cannot be excluded that considerable amounts of hydrogen from the primary systems are misrouted, the buffer tanks, before being switched to the gas reception mode, are flushed with nitrogen (700 mbar) in order to exclude the formation of an explosive gas mixture.

Like all systems of the TLK infrastructure, this system is program controlled and monitored by the Siemens Teleperm M process control system (PLS). The switches of the components made redundant for safety reasons (blowers and control valves) are hard wired and released by relays.

III. Difficulties Encountered during the Commissioning Phase

During commissioning of the ZTS weak points were detected. These, together with the solutions found, will be described below.

Negative Pressure in the Closed Circular Pipeline at Different Throughputs of Gas Delivery

Independent of the throughput of external gas deliveries (between 0 and 77 m^3/h according to specification), the negative pressure in the closed circular pipeline must be kept within a tolerable narrow band. The minimum negative pressure (p_e = -10 mbar) is necessary to guarantee the performance of the safety valves of the gloveboxes. The maximum negative pressure (p_e = -20 mbar) must not be underrated to ensure that the mechanical integrity of the glovebox is not put at risk upon failure of the regular depressurization system of a glovebox. On account of the preciseness of the measuring instruments (approx. ± 0.5 mbar) and to obtain a normal range of responses, the tolerable range at the end of the closed circular pipeline (in direction of flow) is defined as follows: RP207: p_e = -11.5 mbar at the minimum; p_e = -18.5 mbar at the maximum. Taking into account a maximum pressure drop of 2.5 mbar in the closed circular pipeline, about 100 m in length, nominal width DN 100, the tolerable range at the beginning of the closed circular pipeline (RP201, RP202) is defined as follows: p_e = -11.5 mbar at the minimum; p_e = -16 mbar at the maximum (Figure 4).

Keeping the pressure level in the range mentioned before is to be ensured by one of the redundant negative pressure-controlling valves installed (RV203 or RV204). These mechanical spring actuated and diaphragm controlled valves control the back pressure in direction of flow.

It was not possible during the first tests to maintain the values specified for the negative pressure because the admission pressure at the control valve and at the bypass valve (RV205) underwent excessive variations as a result of the different throughputs of external gas deliveries.

This is the reason why the exhaust gas pipe leading to the stack was retrofitted with admission pressure control valves, likewise two in number (RV208 and RV210). These mechanical spring actuated and diapraghm controlled valves serve to maintain between 8 and 12 mbar the admission pressure at the negative pressure control valves (RV203 and RV204) and at the bypass valve (RV205), independent of the throughput of the exhaust gases given off to the stack.

The rate of exhaust gas removal may vary between 7 and 84 m^3/h. Besides the external gas deliveries (0 to 77 m^3/h), the closed circular pipeline within the ZTS is permanently flushed at a rate of 7 m^3/h. This flushing gas serves to change the air in the sampling and valve cabinet of the system where gas samples are collected from the RHV tanks while the necessary protective measures are taken as in a hood.

The modified system has since produced satisfactory results. Some measured values have been entered in Figure 5 and in Table 3.

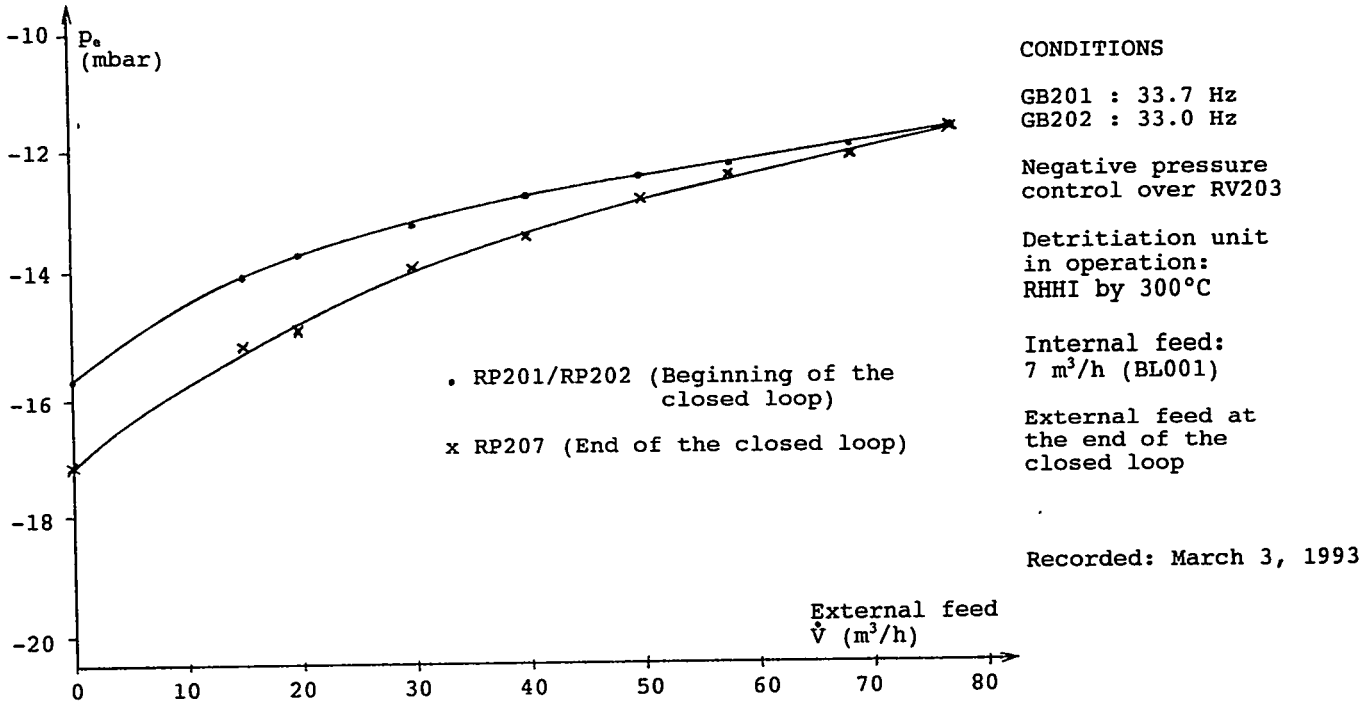


Figure 5 Variation of the negative pressure in the closed circular pipeline in function of the volume flowrate of external feed.

Table 3 Data related to figure 5.

$\dot{V}$	0 m^3/h	15 m^3/h	20 m^3/h	30 m^3/h	40 m^3/h	50 m^3/h	58 m^3/h	68.5m^3/h	77.5m^3/h
RF201 (m^3/h)	51/52	51/52	51/52	51/52	51/53	52/53	52/53	52/53	52/53
RF202 (m^3/h)	54/55	55	55	55/56	55/56	55/56	55/56	55/56	56
RF211 (m^3/h)	103/104	103/104	103/104	103/104	104/105	103/105	104/105	104/105	104/105
RP201 (mbar)	-15.8/ -15.6	-14.2/ -13.9	-13.8/ -13.7	-13.4/ -13.2	-12.9/ -12.7	-12.6/ -12.4	-12.3/ -12.2	-12.1/ -11.9	-11.7/ -11.6
RP202 (mbar)	-15.8/ -15.5	-14.2/ -14.0	-13.9/ -13.8	-13.4/ -13.2	-12.9/ -12.8	-12.6/ -12.4	-12.4/ -12.3	-12.1/ -12.0	-11.8/ -11.7
RP207 (mbar)	-17.3/ -17.1	-15.3/ -15.1	-15.0/ -14.8	-14.1/ -13.9	-13.6/ -13.4	-13.0/ -12.8	-12.6/ -12.4	-12.3/ -12.1	-11.8/ -11.7
RF213 (m^3/h)	20.2	20.2	20.2	20.2	20.2	20.2	20.3	20.3	20.3

Negative Pressure in the Closed Circular Pipeline as a Function of the Tritium Retention Systems in Operation

The tritium rentention function can be fulfilled either by RHH I or by RHH II or by both stages connected in parallel. During this operation the negative pressure in the closed circular pipeline must obviously remain unchanged, independent of the passage through one RHH or the combined RHH I/RHH II.

As had been expected, this condition was not fulfilled in the first tests.

The pressure drop at 300 °C of RHH I together with its connecting pipework and the related motor driven valves was taken as a reference value and assumed to apply to RHH II as well, by installation of a diaphragm (RV 470) downstream of the motor driven valve AV471.

The same pressure drop was generated for the two parallel connected RHHs by installation of diaphragms into the inlet nozzles of the pneumatic valves AV373 and AV473.

Operation of the Central Depressurization System without Tritium Retention Function

Regular operation of the ZTS is characterized by passage of flow through one RHH (RHH I or RHH II). In case of a defect of the RHH in operation, automatic switchover to the standby system is effected via a program controlled by the process control system (PLS). During the tests concerning takeover of the retention function by the standby system difficulties occasionally emerged in the performance of the refrigerators of the RHH just connected. This deficiency could cause in future operation excessive heating of components and, maybe, the release of HTO from partially laden molecular sieves. As a countermeasure, the process of takeover is now monitored more precisely by the process control system and, in case of failure of the refrigerator taking over, the RHH just connected would no longer be passed by gas. However, the central depressurization function of the closed circular pipeline is maintained by the flow of permanently circulated gases via a bypass valve (AV230). The pressure drop characteristic of the bypass section was compensated by means of the valve RV232 with that of RHH I. During this emergency mode of operation any tritium involving activities in the laboratory are interrupted.

IV. Tests on Hydrogen Retention at the Tritium Retention Systems with Palladium as the Catalyst

The Tritium Retention Systems with palladium as the catalyst used in the main cleaning stages (RHH) of ZTS were examined in a test series for their capability of hydrogen retention.

The retention capability tested related to hydrogen in the range of concentrations from 10,000 ppm (= 1%) down to 1 ppm, as a function of the catalyst temperature (100 °C, 180 °C, 300 °C, 450 °C) and the air flow.

Test Setup

A Tritium Retention System (TRS), construction size II, with palladium as the catalyst, was connected to a glovebox of approx. 5 m^3 volume. At the operating stage I of the blower (= 30 m^3/h) this corresponds to six air changes per hour; at the operating stage II (= 60 m^3/h) this is equivalent to twelve air changes per hour.

The test configuration was such that the TRS can be operated either within the loop via the glovebox or in the bypass mode outside the glovebox. Hydrogen can be introduced

into the glovebox at a defined flow rate. The atmosphere of the glovebox can be homogenized in the bypass mode by means of a separate blower mounted outside the TRS.

The hydrogen concentration of the processed gas was measured downstream of the Tritium Retention System and in the box.

Hydrogen Measuring Technology

Two leak detectors UL 100 Plus, supplied by Leybold AG, were used for continuous measurement of the hydrogen concentration downstream of the TRS and in the glovebox. These instruments (quadrupole mass spectrometers) have been designed above all for use with helium; however, they can be switched over to the mass 2 (H_2). The detectors were sniffing detectors which means that a gas stream of constant flow rate is analyzed for its H_2 fraction.

The instruments were calibrated using eleven calibrating gas mixtures of different concentrations, between 5 ppm and 2% hydrogen in air.

Experimental

At the beginning of a test TRS operated via the glovebox was brought to the respective operating temperature corresponding to a specified gas flow rate (30 or 60 m^3/h). When steady-state conditions had established, TRS was switched over to the bypass mode and the blower was connected to circulate the atmosphere in the glovebox. Then hydrogen was fed into the glovebox until the readout at the point of hydrogen measurement showed the desired concentration.

When steady-state conditions and the desired hydrogen concentration, respectively, had established in the glovebox, the test could be started. For this, the recirculating blower of the glovebox was disconnected and TRS operated via the glovebox. During the test the hydrogen concentrations at the gas outlet of TRS and in the glovebox were measured continuously.

When the hydrogen concentration in the glovebox approached the limit of detection of 1 ppm, the test was terminated. To do this, the glovebox was flushed with laboratory atmosphere while TRS was operated in the bypass mode and the recirculating blower was running. Flushing was done by opening one nozzle of the glovebox and by exhausting the gas at the opposite end of the glovebox using the mobile exhaust system of TLK.

Results Obtained

The dependence of the reduction of hydrogen concentration in the glovebox on the temperature of the catalyst and the gas flow rate has been plotted in Figure 6. The following results can be derived from that figure:

- The reduction of hydrogen concentration in the glovebox is nearly independent of the catalyst temperature.
- The reduction of hydrogen concentration in the glovebox is heavily dependent on the gas flow rate and hence on the number of air changes.

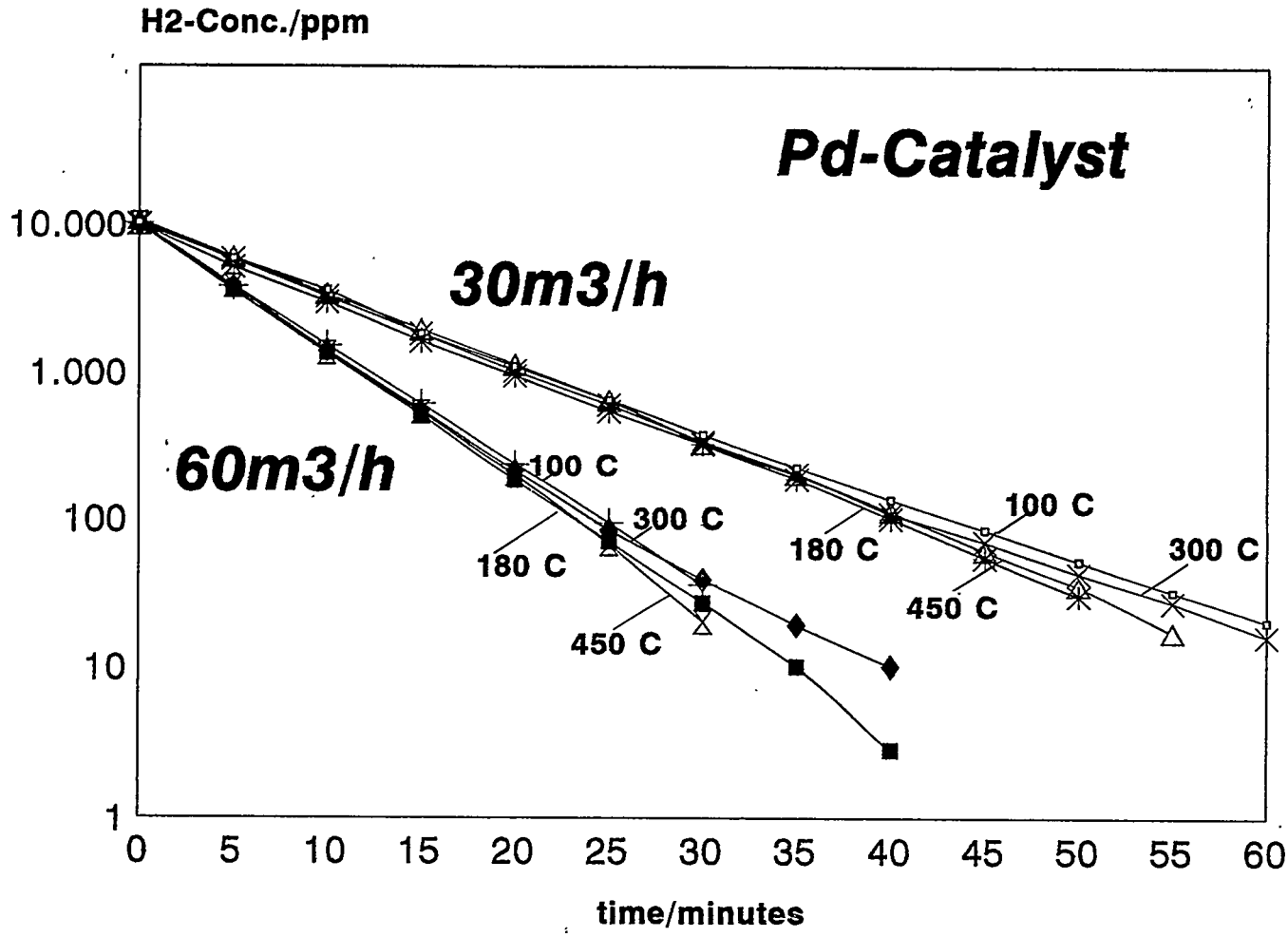


Figure 6 Dependance of decrease of concentration in the glovebox on catalyst temperature and gasflowrate.

Also Table 4 makes evident these results. It has been entered in the table after which time interval the hydrogen concentration in the glovebox has dropped to half its former value, i.e. from approximately 10,000 ppm to approximately 5000 ppm (half-life) and after which time interval a concentration of 100 ppm has been attained. starting from an initial concentration of 10,000 ppm, both as a function of the gas flow rate and the catalyst temperature.

In order to be able to determine the anticipated cleaning effect in once-through gas flow - the operating mode of RHH - experiments were performed at the Radiochemistry Institute of KfK regarding the determination of the decontamination factor (DF), prior to conceiving the Central Tritium Retention System. The desired low residual concentrations of hydrogen (tritium) could be measured only with tracer amounts of tritium used. A DF >3000 was measured which is deemed adequate considering the anticipated tritium concentrations in the TLK exhaust gases (10^{11} Bq/m^3 at the maximum).

Table 4 Reduction of hydrogen concentration in the glovebox.

Gas Flow Rate	Catalyst Temperature	Half-life	Duration of Concentration Drop from 10,000 to 100 ppm
30 m^3/h	100 °C	6 min	43 min
	180 °C	6 min	40 min
	300 °C	6 min	40 min
	450 °C	6 min	41 min
60 m^3/h	100 °C	4 min	25 min
	180 °C	4 min	24 min
	300 °C	4 min	24 min
	450 °C	4 min	23 min

Reference

(1) H.J. Ache, M. Glugla, E. Hutter, G. Jourdan, R.-D. Penzhorn, D. Röhrig, K. Schubert, H.Sebening and J.E. Vetter, "The Tritium Laboratory Karlsruhe: R&D Work and Laboratory Layout", Fusion Engineering and Design 12 (1990) 331-341, North Holland

DISCUSSION

GOOSSENS: I would like to ask about peak releases of tritium from the primary equipment into the glove boxes. How do you handle a peak release by accident of a relatively large quantity, including the explosion hazard?

HUTTER: I pointed out that we have multi-barriers, and if you have an escape of tritium out of the primary system into the glove box, you always have your glove box sealed. If you want to discuss a broken primary systems and a broken glove box, in the first case, the glove box is always tight, there is a glove box atmosphere treatment system that takes out the hydrogen escaping to the glove box within about two hours. I must say, for the moment, we have limited the amount of tritium per glove box to 25,000 ci, about 2.5 g of tritium. That gives in the smallest existing box of 3 m^3 volume a maximum tritium concentration (homogenized) of 0.3 v%. Taking into account the continuous air flow through the box by the glove box Tritium Retention System (TRS), we don't expect a tritium concentration higher than 1% even in the vicinity of the leak (explosion limit is 4 v%). Systems with a high content of hydrogen isotope mixtures (H, D, T) are equipped with an inerted box atmosphere. Remember there are two stages of tritium retention, the glove box clean-up system and the offgas out of the box that is given to the large loop under negative pressure. This gas flow is small and is treated with the TRS-like system. In the case of a glove box disintegration, there must be a large hole to discharge tritium into the laboratory atmosphere. That is, however, the reason we have restricted the tritium amount per box. A specialist calculated that this is far from the activity accidentally allowed in the environment.

MYERS: I am not sure I understand the process, but it looks like you are taking hydrogen in the tritium state and, by treating it with a catalyst, converting it into a liquid state, to water. Is this correct? How is the tritiated water treated, removed, or retained?

HUTTER: This is correct. The water is retained on molecular sieve beds, 30 kg each by weight. These are taken out of the cleanup system and regenerated in a special unit by heating the molecular sieve to 300° C in a nitrogen stream as carrier gas. The desorbed tritiated water is condensed in a heat exchange (25° C) followed by a freezer (-196° C) to dry the nitrogen stream. The water separated and collected in this way in a vessel is transported to the KfK Decontamination Department where it is prepared for storage.

Cracking of water (hot metal bed or electrolysis) to recover the tritium is actually not forseen in TLK because of the low tritium content expected in the waste water.

CLOSING COMMENTS OF SESSION CO-CHAIRMAN WILHELM

I would like to give a short summary of the first three papers. The first paper handles a very important issue, how much iodine will be transferred into the dissolver offgas when a desorption process is used to prevent iodine from remaining in the spent fuel solution. Iodine may remain in small amounts in the solutions in all of the vessels of the final process of separating uranium and plutonium and may be in the vessel offgas. To take it out of this large volume of offgas is very expensive, and the decontamination factors for the vessel offgas are much lower than for the dissolver offgas, the volume of which is much smaller. So, it is very important to come down with the rest of the iodine in the dissolver dissolution of the spent fuel. I found it very interesting to get figures for these operations. The question is whether they are figures that refer to a big reprocessing plant. Still, there is a question: what form is the iodine which does not come out from the dissolver solution? The colloidal form may be only a part. It may be an important part, or it could be that it is not, as we may expect with a recycled acid solution in the dissolver. Then we will have more problems.

I was happy to hear that the THORP is finished, and I hope very much that it goes into operation. It will be quite interesting to learn what the real decontamination factor will be with this offgas system design. There is a difference in the philosophy of the units used to retain iodine. At the moment, there are two processes discussed and used. One is a dry process. People expect the decontamination factor to be a little higher than in the wet process, the target being 10^3 compared to 10^2 for the wet process. The dry process was developed to get rid of all the problems of disposing of iodine in solution; after you trap it in an alkaline solution, you have to do something with it. We have seen a lot of work in the United States to transfer iodine into a dry form for storage. It is just hard work, and I do not think it will pay out because the English can release it as an alkaline solution to the sea where long-lived iodine-129 will be diluted. In some countries, this would not be allowed, so they have to use the dry process, where they will have the material ready for final disposal in one step.

The last paper covered work done on solid material to trap iodine in the form of silver iodide. A very important point was brought out in the paper that this process is very insensitive to changes of operating parameters in the reprocessing plant, and especially in the dissolving process. That means that even when there is a huge range of operating parameters, or parameters which will change by some small accident, one can still expect high decontamination factors. I think that is very much in favor of this process and of the material used.

CLOSING COMMENTS OF SESSION CO-CHAIRMAN KIKUCHI

I want to comment on the last paper. I think development of a tritium gas treatment system will become a very important technique for nuclear fusion research. Of course, nuclear fusion is far from commercial operation, but several places are doing research using tritium. The basic technology discussed in the paper uses a catalytic reaction which oxidizes tritium gas to water. The second paper on the program (that was not given today) uses the same technology. These technologies could become even more important, not only in fusion research but also for fuel reprocessing plants, when environmental standards become more severe. Therefore this research should be continued.

SESSION 12

AIR TREATMENT SYSTEMS AND ACCIDENT CONTROL

Tuesday: July 26, 1994
Co-Chairmen: R. D. Porco
W. R. A. Goossens
X. Chen

VALUE-IMPACT ASSESSMENT FOR RESOLUTION OF GENERIC SAFETY ISSUE 143 - AVAILABILITY OF HVAC AND CHILLED WATER SYSTEMS
P. M. Daling, J. E. Marler, T. V. Vo, H. K. Phan, J. R. Friley, V. T. Leung

LEAK TESTING OF BUBBLE-TIGHT DAMPERS USING TRACER GAS TECHNIQUES
P. L. Lagus, L. J. DuBois, K. M. Fleming, J. H. Brown

CONSTANT DEPRESSION FAN SYSTEM
A NOVEL GLOVEBOX VENTILATION SYSTEM
W. V. Milliner

CALCULATION CODE EVALUATING THE CONFINEMENT OF A NUCLEAR FACILITY IN CASE OF FIRES
J. C. Laborde, J. L. Peirano, G. Perrin, C. Prevost, S. Raboin, J. Vendel

PERFORMANCE OF HEPA FILTERS AT LLNL FOLLOWING THE 1980 AND 1989 EARTHQUAKES
W. Bergman, J. Elliott, K. Wilson

THE ACTUAL PRACTICE OF AIR CLEANING IN BELGIAN NUCLEAR FACILITIES
W. R. A. Goossens

AN INTRODUCTION TO THE DESIGN, COMMISSIONING AND OPERATION OF NUCLEAR AIR CLEANING SYSTEMS FOR QINSHAN NUCLEAR POWER PLANT
X. Chen, J. Qu, M. Shi, L. Yang

AP600 CONTAINMENT PURGE RADIOLOGICAL ANALYSIS
M. O'Connor, J. Schulz, C. Tan, M. Kasjaka, N. Alper

A LOW PRESSURE FILTER SYSTEM FOR NEW CONTAINMENT CONCEPTS
H. G. Dillmann, H. Pasler

CLOSING COMMENTS OF SESSION CO-CHAIRMAN PORCO

23rd DOE/NRC NUCLEAR AIR CLEANING AND TREATMENT CONFERENCE

VALUE-IMPACT ASSESSMENT FOR RESOLUTION OF GENERIC SAFETY ISSUE 143 - AVAILABILITY OF HVAC AND CHILLED WATER SYSTEMS

P. M. Daling, J. E. Marler, T. V. Vo, H. K. Phan, and J. R. Friley
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Richland, Washington

V. T. Leung
U.S. Nuclear Regulatory Commission
Washington D.C.

Abstract

The Pacific Northwest Laboratory (PNL), under contract to the U.S. Nuclear Regulatory Commission (NRC),(a)* has conducted an assessment of the values (benefits) and impacts (costs) associated with potential resolutions to Generic Issue 143, "Availability of Heating, Ventilation, and Air Conditioning (HVAC) and Chilled Water Systems." This assessment was conducted to identify vulnerabilities related to failures of HVAC, chilled water and room cooling systems and develop estimates of the core damage frequencies and public risks associated with failures of these systems. This information was used to develop proposed resolution strategies to this generic issue and perform a value/impact assessment to determine their cost-effectiveness. Probabilistic risk assessments (PRAs) for four representative plants form the basis for the core damage frequency and public risk calculations. Internally-initiated core damage sequences as well as external events were considered. Three proposed resolution strategies were developed for this safety issue and it was determined that all three were not cost-effective. Additional evaluations were performed to develop "generic" insights on potential design-related vulnerabilities and potential high-frequency accident sequences that involve failures of HVAC/room cooling functions.

I. Introduction and Background

Heating, Ventilation, and Air Conditioning (HVAC), chilled water, and room cooling systems perform important safety-related functions at nuclear power plants that are primarily directed toward maintaining appropriate environmental conditions in areas that contain safety-related equipment. The function evaluated in this paper involves removal of heat from rooms containing safety-related equipment, such as control rooms, emergency core cooling system (ECCS) pump rooms, electrical switchgear rooms, DC equipment rooms; and emergency diesel generator (EDG) enclosures. Heat removal from these rooms is necessary to ensure that room temperatures are maintained below the design specifications and environmental qualification parameters of the safety-related equipment in each room. Other functions performed by these systems, such as heating and air cleaning functions, are not in the scope of this analysis.

This paper evaluates the present designs and reliability of HVAC and room cooler systems in nuclear power plants. The specific objectives include: (1) identify areas that are particularly vulnerable to loss of HVAC or room cooling, (2) determine the vulnerabilities of safety-related systems and components to failure of HVAC and chilled water systems that are reflected in actual operating experience, (3) estimate the core damage frequency and public risks that are associated with failure of HVAC and chilled water systems, (4) develop alternative resolutions to this issue, and (5) perform value/impact assessments of each alternative. Complete documentation of this study is provided in NUREG/CR-6084.[1]

II. Analysis of Affected Core Damage Frequency and Public Risk Estimates

Existing probabilistic risk assessments (PRAs) for four representative plants, including one from each Nuclear Steam Supply System (NSSS) vendor, were used as the basis for quantifying the effects of HVAC and room cooler systems on core damage frequency and public risk estimates. Both internally- and externally-initiated events were

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(a)* Work performed for the U.S. Nuclear Regulatory Commission under a Related services Agreement with the U.S. Department of Energy under Contract DE AC06 76RLO 1830.

considered in the PRA calculations. Modifications to the PRAs were made, where necessary, to incorporate room cooler failures as potential contributors to losses of safety system functions. Losses of room cooling was also evaluated as potential accident sequence initiating events.

Transient thermal calculations were used to develop time-temperature profiles in various safety-related equipment rooms, including Engineered Safety Feature (ESF) pump rooms, AC and DC equipment rooms, and the control room. These calculations were used to identify rooms or areas that are particularly vulnerable to loss of HVAC/ room cooling functions and to provide indications of the amount of time available to recover from losses of room cooling. Limiting failure temperatures for mechanical and electrical equipment in these rooms were determined based on information obtained from utilities and vendors. The key conclusions from these evaluations are:

- Heatup of the control room leads to heatup of electrical components inside cabinets and panelboards containing temperature-sensitive components. This could lead to spurious signals, readings, alarms, and equipment actuations.

- DC equipment rooms contain numerous temperature-sensitive components, including battery chargers and inverters. Loss of this equipment due to room cooling failure could lead to a plant trip and simultaneous failure of DC control power to safety-related components needed to respond to the plant trip.

- Diesel generator ventilation systems are important to the operability of diesel generator control cabinets. Failure of the diesel generator room ventilation system could result in diesel generator control system failure in as little as 15 to 20 minutes.

- Heatup of ESF pump rooms following loss of room cooling could lead to pump motor failure. However, heatup rates are such that a substantial amount of time is available to restore room cooling or take other corrective action to cool ESF pump rooms. Relatively simple recovery measures, such as opening the ESF pump room door to allow air to circulate from an adjacent cooled room into the ESF pump room, can increase allowable recovery times substantially, as illustrated in Figure 1.

The key results from the PRA evaluation are shown in Table 1. This table presents the results of the core damage frequency and public risk calculations performed to quantify the contributions of HVAC/room cooler failures to internally- and externally-initiated accident sequences. The term "affected" refers to the fraction of the total plant core damage frequency (CDF) and public risk values that include failures of HVAC/room cooling systems. The values in Table 1 that are listed in the columns labeled "Internal" include accident sequences that are initiated by losses of HVAC/room cooling systems and accident sequences that are initiated by other plant transients (e.g., loss of offsite power) which include HVAC/room cooler failures as contributing events. The values in the columns labeled "External" are the affected CDF and public risks attributable to core damage sequences initiated by external events, including earthquakes, floods, fires, and tornadoes.

The total affected core damage frequencies for all four representative plants were calculated to be approximately 1E-05/RY. The affected CDF from external events (earthquakes, fires, and floods) contributes about 68 to 94% of this value, depending upon the plant. The affected public risk values for all four plants were calculated to be in the range from about 2 to 4 person-rem/RY. The affected public risk values were also dominated by external events.

The results presented in Table 1 are applicable to the specific plants evaluated. Due to major differences in HVAC and room cooling system configurations among plants, other plants may have higher or lower affected core damage frequencies and public risk values.

III. External Events Analysis

The CDF and public risks associated with external-event-induced failures of HVAC/room cooling functions were quantified in this assessment. The external events analyses supporting the NUREG-1150[(2)] PRAs and plant-specific Individual Plant Examinations (IPEs) formed the basis for this analysis.

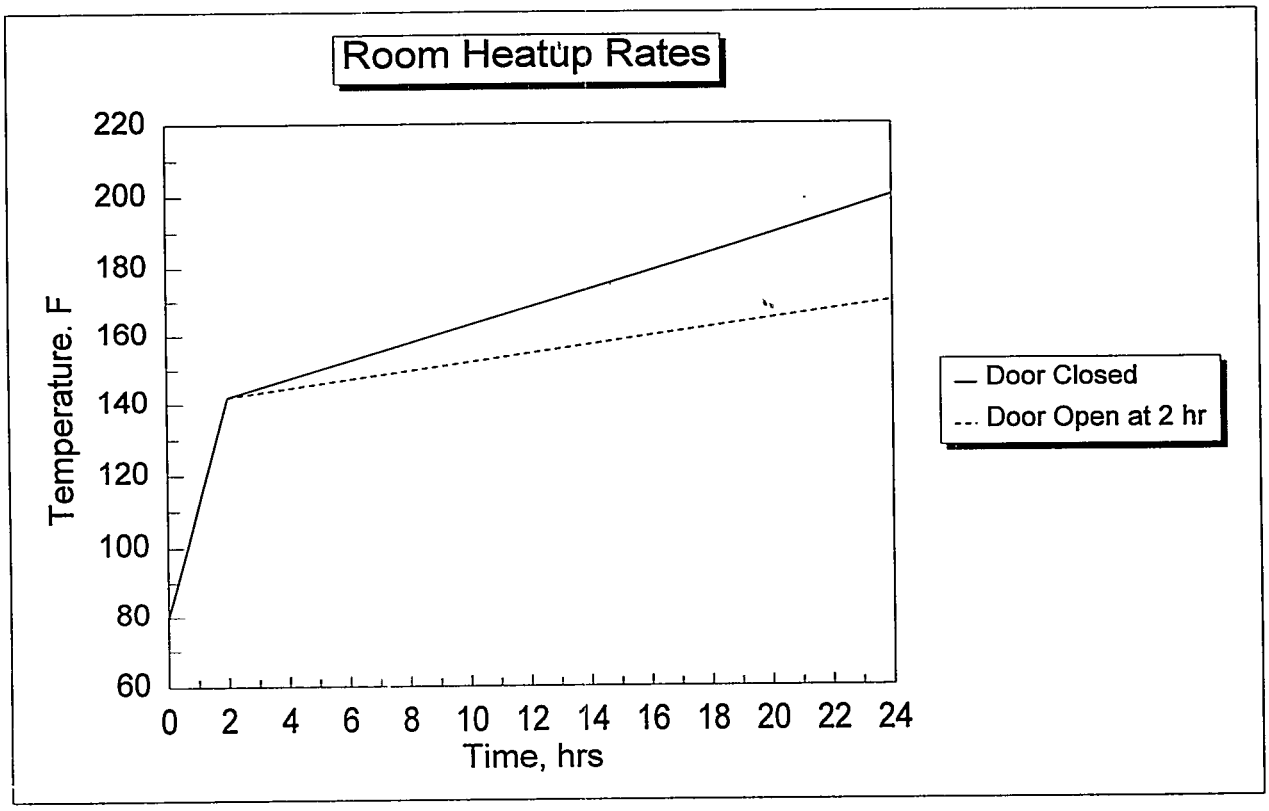


Figure 1 Illustration of transient thermal calculations: low pressure injection pump room heatup rate following loss of room cooling, including the effects of opening the LPI pump room door 2 hr. after the loss of room cooling.

Table 1 Summary of affected CDF and public risks for the representative plants in this study.(a)*

	Westinghouse PWR	General Electric BWR	Combustion Engineering PWR	Babcock & Wilcox PWR
Affected Core Damage Frequency, per RY				
- Internal Events	4.1E-06	5.4E-07	1.3E-06	1.2E-06
- External Events	8.2E-06	8.1E-06	8.2E-06	9.6E-06
Affected CDF, Total	1.2E-05	8.6E-06	9.5E-06	1.1E-05
Affected Public Risks, person-rem/RY				
- Internal Events	1.7E+00	9.9E-02	7.2E-01	1.7E-01
- External Events	1.6E+00	3.9E+00	1.6E+00	2.9E+00
Affected Public Risks, Total	3.3	4.0	2.3	3.1

(a)* All values are point estimates.

The affected CDFs for external events, summarized in Table 1, were calculated to be in the range 8E-06/RY to about 1E-05/RY. Affected public risk estimates were calculated to be between about 2 and 4 person-rem/RY. These values are significantly larger than the affected CDF and public risk values calculated for internal events. Key conclusions that were derived from the external events analysis included:

- Fires were found to dominate the affected external event CDF. Floods were also significant contributors to the external event CDF and public risk estimates.

- Room coolers and HVAC equipment are less susceptible to seismic-induced failures than other plant components such as pumps, switchgear, and water storage tanks. The resistance to damage in seismic events also provides substantial protection against tornado-generated missiles and high winds. This tends to minimize the contributions of seismic events and tornadoes to the affected CDF.

- The affected seismic CDF is dominated by a few accident sequences. The dominant accident sequences primarily involve station blackout situations, although the exact causes of station blackout are somewhat different between plant types. Seismic-induced failures of pump suction sources (tanks) and emergency switchgear were the dominant contributors. Failure of HVAC systems were not among the dominant contributors.

- Tornadoes were demonstrated to be insignificant contributors to the affected external event CDF primarily due to the protection provided by plant structures and buildings from tornado-induced high winds and missiles. Tornadoes have similar effects on HVAC components as do seismic events; i.e., it is unlikely that a tornado will fail HVAC equipment and not the safety-related equipment supported by the HVAC system (i.e., the effects of a tornado are not likely to be localized in the HVAC equipment room). Therefore, failures of components other than HVAC equipment will dominate the tornado-induced CDF.

- Approximately 10% of the affected CDF due to external events were contributed by flood-induced accident sequences. Flood effects are likely to be concentrated in a certain area of the plant. Therefore, it is possible that a flood could cause failure of HVAC equipment and not directly affect the operability of the equipment supported by the HVAC function.

These conclusions are believed to be generically applicable to all U.S. nuclear power plants. The external events analyses noted no significant plant-specific anomalies that would lead one to believe that the HVAC/room cooler systems at the plants examined in this study are any more or less vulnerable than other plants to external events. It was shown that the failure probabilities for HVAC/room coolers subjected to the various external events are lower than for other safety-system components.

IV. Generic Insights

HVAC/room cooler vulnerabilities were identified through observations made during plant visits, discussions with equipment vendors, and reviews of relevant safety-related information, such as plant safety documentation, design information, Licensee Event Reports (LERs), and other sources. Based on the information obtained, "generic" insights were developed on the applicability of the four representative plants to the rest of the plants. Generic insights were also developed on the applicability of potential high core-damage-frequency accident sequences found in the literature to the population of nuclear power plants.

The identified vulnerabilities ranged from design inadequacies to deficient operating, maintenance, and test procedures; and included:

- Installation errors during plant construction or modification were observed. A relevant example of this is an installation error that could disable water systems that remove heat from ESF pump rooms and essential chillers.

- Test and maintenance errors and inadequate test procedures, particularly failure to return safety-related HVAC components to service properly, could disable one or more trains of HVAC and significantly degrades their reliability.

- Failed or malfunctioning ventilation and fire dampers can result in a total loss of room cooling to safety-related equipment rooms.

- Inadequate ventilation airflow to certain rooms and to certain areas within rooms have resulted from design errors. One design error involving location of one train of vital DC equipment in a switchgear room for the other train was identified. Failure of room cooling to this switchgear room could cause failure of one train of switchgear and the opposite train of DC equipment.

- Examples were found of incidents in which temperatures inside electrical cabinets exceeded design basis temperatures although room ambient temperatures were below their maximum values. Design-basis assumptions that internal cabinet temperatures are 20°F higher than the room temperature for cabinets cooled by natural convection may not be valid under all conditions. The causes of higher than expected temperatures include inadequate airflow balance in the room (i.e., dead spots) and blockage of airflow through the cabinets by the equipment and equipment racks inside the cabinets. Therefore, failures of cabinet components due to high temperatures may occur even though room ambient temperatures are below design maximum values.

- Low priority appears to be placed on maintaining cabinet fans in proper working condition. Cabinet fan failures have caused internal components to overheat, leading to spurious actuations and malfunctions of safety-related equipment.

- Actual temperatures in safety-related equipment rooms are typically not monitored in control rooms. Other indications are used to detect failures of HVAC/room cooling in safety-related equipment rooms (e.g., fan start/run, chiller start/run, damper positions). The fact that there are little or no direct room temperature indications may result in operators being unable to detect room cooling failures that could lead to failure of a safety-related component.

- Fire protection system design reviews have resulted in identification of inadequate separation between electrical cables feeding redundant trains of HVAC systems.

- Heavily-loading individual divisions of vital DC control power, whose components are sensitive to elevated room temperatures, may result in failure of significant portions of safety-related systems should HVAC systems fail.

- Rapid failure of emergency onsite AC power systems will result from failure of emergency diesel generator (EDG) room ventilation systems. No recovery from failure of the EDG room ventilation systems is thought possible because the time available is less than 30 minutes and in some cases less than 15 minutes.

- In some instances, relevant emergency procedures and equipment (e.g., portable fans) were not provided to respond to losses of HVAC/room cooling functions.

- A tradeoff exists between the philosophy of isolating ESF pumps in small cubicles to protect them from fires, floods, etc., and the placement of ESF pumps in large rooms. Small rooms heat up much more rapidly than large rooms so the failure probabilities given loss of cooling are higher for small rooms than for large rooms. However, this effect competes with the increased vulnerability to external events and common cause failures that results from placement of ESF pumps in large, open rooms, collocated with other safety-related equipment.

In most cases, these vulnerabilities were concluded to be plant-specific concerns. However, potential generic vulnerabilities were identified, including the effects of ventilation and fire dampers on HVAC unavailability, potential underestimating of temperatures inside electrical cabinets, and possibly a lack of adequate test, inspection, and/or maintenance procedures and training relative to HVAC and room cooling systems. Relative to the latter vulnerability, this was not the situation observed at the LWRs visited in this study. However, due to many examples of such events described in various documents and LERs, this was judged to be a potential generic vulnerability.

To add some perspective to the affected core damage frequencies calculated for the representative plants, a review and limited-scope quantitative evaluation of five additional plants was performed. The quantification of the

affected CDFs for these additional plants was less rigorous than that performed for the representative plants. However, the affected CDFs for the additional plants are believed to be acceptable for comparison purposes.

The affected CDFs from internal events that were calculated for the nine plants (four representative plants and five additional plants) ranged from about 1E-07/RY to 1E-05/RY. The arithmetic mean affected CDF across all nine plants was calculated to be about 4E-06/RY. This was not considered to be a large variation relative to the results for the four representative plants, given the difference in the level-of-detail between the detailed assessments of the four representative plants and the approximations developed for the five additional plants. It was concluded that the four representative plants used in this study adequately represent the population of nuclear power plants in terms of the core damage frequencies and public risks attributable to HVAC and room cooler failures.

Some relatively-high core damage frequencies related to failures of HVAC and room cooling systems were observed for some plants. Although it appears that the average affected CDF for the entire population of plants is in the 1E-05 to 1E-06/RY range, there appear to be some plant-specific accident sequences at other plants that may be as high as 1E-04/RY. In most cases, the high-frequency plant-specific accident sequences were shown to be either highly-conservative estimates or have been significantly reduced through hardware and/or operational changes. The PRAs being performed in support of the Individual Plant Examinations (IPEs) have identified plant-specific anomalies and hardware configurations that have been or will be resolved. The IPEs appears to be an ideal vehicle for identification and resolution of plant-specific HVAC- and room cooler-related issues.

V. Value-Impact Assessment

A value-impact assessment was performed using the acceptable methods described in NUREG/BR-0058, Rev. 1[3] and NUREG/CR-3568.[4] The value-impact assessment was performed to identify and quantify the relevant values (benefits) and impacts (costs) of the proposed resolution strategies and to determine the cost-effectiveness of each of the three alternatives. Several attributes were quantified to characterize the consequences of the proposed resolutions. These attributes are identified and briefly described in Table 2.

Three alternative resolution strategies for GI-143 were developed in this study. The alternative resolutions are composed of several elements that are designed to prevent or mitigate one or more of the vulnerabilities discussed above. Table 3 summarizes the resolution elements and indicates the vulnerabilities addressed by each element.

The resolution elements defined in Table 3 were combined into three comprehensive resolution strategies or alternatives that were developed to bound the range of possible costs and benefits resulting from resolution of GI-143. The first alternative is the minimum improvement alternative and combines plant evaluation elements and procedural/maintenance improvements to prevent or mitigate potential losses of room cooling. The second alternative combines the elements of Alternative 1 with relatively minor hardware changes designed to improve detection and recovery capabilities. The third alternative results in maximum improvement to HVAC/room cooler reliability and combines the elements of Alternative 2 with additional testing programs and hardware changes. Each alternative addresses, to some extent, all of the vulnerabilities identified in Section 2.0. The three alternatives are summarized in Table 4.

The public risk benefits of the potential alternative resolutions to this issue are measured in terms of reductions in core damage frequency and public risks. The approach to calculating public risk benefits was to modify the risk calculation models at the cut set level to represent the potential improvements in HVAC/room cooler system reliability and the abilities of operators to recover from a loss of one of these functions. The results of the public risk reduction calculations, or the benefits of the alternative resolutions to GI-143, are summarized in Table 5. The table shows the per-plant reductions in core damage frequency and public risks associated with implementation of each of the alternatives and the total life-cycle public risk reduction. The total public risk reduction is the per-plant risk integrated over all current plants and the average remaining lifetimes of these plants.

Estimates of the onsite consequences of the three resolution alternatives were developed, including increased occupational exposures due to implementation, operation, and maintenance activities associated with each alternative. The averted occupational doses and economic consequences associated with reduced accidents were also calculated. These latter costs are treated as cost offsets and are subtracted from the total NRC and industry implementation, operation, and maintenance costs.

Table 2 Summary description of value-impact attributes quantified in this study.(a)*

Attribute	Description
Public Health	Expected changes in public exposure to radiation due to offsite radioactive releases, measured for all affected plants during the remainder of their lifetimes.
Occupational Exposure (Accidental)	Expected change in exposure to employees as a direct result of postulated accidents (summed over all affected plants for the remainder of their lifetimes).
Occupational Exposure (Routine)	Expected change in exposure to employees as a result of installation, modification, and maintenance of the proposed changes. Occupational exposures are summed over all affected plants and their remaining lifetimes.
Offsite Property	The expected total monetary savings to offsite property resulting from the proposed action; i.e., from reduced accident frequencies and consequences.
Onsite Property	The expected monetary savings to all affected licensees from the proposed action; i.e., from averted plant damage costs -- including replacement power, decontamination, and refurbishment costs.
Industry Costs	The projected net effect on the licensee; 1) to install or implement mandated changes, and 2) due to changes in routine, periodic operation and maintenance caused by the proposed changes. These are referred to as industry implementation and operation/maintenance costs, respectively. Industry costs are summed over all affected plants and their remaining lifetimes. The costs are discounted.
NRC Costs	The projected net economic effect on the NRC; 1) to prepare the proposed action for implementation, 2) to place the proposed new requirements into operation, and 3) after the proposed action takes effect (e.g., additional NRC inspections). NRC costs are summed over all affected plants and their remaining lifetimes. The costs are discounted.
(a)* Adapted from NUREG/CR-3568.[4]	

The costs of each alternative resolution are presented in Table 6. This table shows the implementation, operation/maintenance, NRC, onsite consequences, onsite property damage, and offsite property damage costs for each alternative resolution. The top half of the table presents the per-plant costs and the bottom half presents the total integrated costs. All costs are discounted to 1993 dollars.

Table 7 shows the value-impact scores for each alternative. The ratios were calculated by dividing the total discounted costs by the total public risk reduction. The table shows two value impact scores; the first was calculated without the cost offsets for onsite and public property damage and the second includes the cost offsets. These value:impact ratios were compared to $1,000/person-rem. It was concluded that all of the resolution alternatives exceed the NRC guideline, both in terms of the net and total value - impact scores.

Table 3 Matrix of vulnerabilities versus resolution elements.

Vulnerability	Resolution Elements Addressing Specific Vulnerabilities										
	1	2	3	4	5	6	7	8	9	10	11
1. Spurious actuation of control room components inside cabinets	X					X	X	X			X
2. Temperature-sensitivity of DC equipment (inverter, battery charger)	X				X	X	X	X			
3. Loss of EDG ventilation system rapidly fails EDG control equipment	X						X	X	X		
4. Installation errors disable HVAC-related water systems		X									
5. Inadequate test/maintenance procedures			X								
6. Inadequate PM Program for HVAC/room cooling equipment			X								
7. Failed/malfunctioning ventilation and fire dampers	X	X	X		X		X	X			
8. Design errors lead to inadequate airflow	X	X			X						X
9. Electrical cabinets overheat due to fan failures	X		X	X		X	X	X		X	X
10. Electrical cabinet heat rejection capabilities overestimated	X	X				X				X	X
11. Actual room temperatures not monitored in control rooms		X		X	X						
12. Essential chiller design not optimized		X									
13. Inadequate separation between electrical cables		X									
14. Heavily-loaded vital DC buses		X									
15. Inadequate emergency procedures to cope with loss of room cooling				X	X						
16. Small room versus large room	X	X				X	X	X			
17. Fires and floods dominated affected CDF due to external events	X	X		X		X	X	X			

Resolution elements are as follows: 1. Perform room heatup calculations and thermal fragility analyses. 2. Perform design review. 3. Enhanced PM and Testing Program. 4. Improve Emergency Procedures. 5. Install Remote Room Temperature Monitors/Alarms. 6. Increase Environmental Qualification Requirements. 7. Install Portable Air Cooling Equipment. 8. Install Permanent Backup Room Cooling Systems. 9. Install Thermal Barriers Between Heat-Producing Components and Thermally-Sensitive Components. 10. Electrical Cabinet Tests. 11. Airflow Test Program.

Table 4 Summary of alternative resolutions to GI-143.

Alternative	Element No.	Resolution Element Title
1	1.	Perform room heatup calculations and thermal fragility analyses.
	2.	Perform design review
	3.	Enhanced PM and Testing Program
	4.	Improve Emergency Procedures
2	1.	Perform room heatup calculations and thermal fragility analyses.
	2.	Perform design review
	3.	Enhanced PM and Testing Program
	4.	Improve Emergency Procedures
	5.	Install remote temperature monitors/alarms
	7.	Install portable room cooling equipment
3	1.	Perform room heatup calculations and thermal fragility analyses.
	2.	Perform design review
	3.	Enhanced PM and Testing Program
	4.	Improve Emergency Procedures
	5.	Install remote temperature monitors/alarms
	6.	Increase Environmental Qualification requirements
	7.	Install portable room cooling equipment
	8.	Install permanent backup room cooling systems
	9.	Install thermal barriers between heat-producing components and thermally-sensitive components
	10.	Electrical cabinet tests
	11.	Airflow test program

Table 5 Summary of values (benefits) of alternative resolutions to GI-143.(a)*

Parameter	GI-143 Resolution Alternative		
	1	2	3
Per-plant reduction in core damage frequency, per RY	2.4E-06	4.5E-06	6.7E-06
Per-plant reduction in public risks, person-rem/RY	8.4E-01	1.3E+00	2.0E+00
Total integrated risk reduction (person-rem)	2.2E+03	3.6E+03	5.2E+03

(a)* Risk reduction calculations were based on the risk models for a W PWR. Risk reduction at other plant types should not be substantially different.

Table 6 Summary of per-plant and total impacts (costs) for each resolution alternative.

Per-plant Costs ($/plant)								
Resolution Alternative	Discount Rate	Implementation Cost	Operation/ Maintenance Cost	NRC Cost	Onsite Consequences Person-rem	Onsite Consequences $/plant	Avoided Onsite Damage	Avoided Offsite Damage
1	5	2.2E+05	2.7E+04	2.2E+04	9.6E+00	9.6E+03	4.9E+04	6.5E+04
	10	2.0E+05	1.7E+04	1.7E+04	9.6E+00	9.6E+03	2.4E+04	4.0E+04
2	5	2.9E+05	3.7E+04	2.2E+04	1.9E+01	1.9E+04	7.2E+04	9.5E+04
	10	2.4E+05	2.1E+04	1.7E+04	1.9E+01	1.9E+04	3.4E+04	5.7E+04
3	5	5.6E+04	5.6E+04	4.9E+04	9.6E+01	9.6E+04	9.4E+04	1.2E+05
	10	3.1E+04	3.1E+04	3.7E+04	9.6E+01	9.6E+04	4.4E+04	7.2E+04
Total Costs ($)								
Resolution Alternative	Discount Rate	Number of plants	Implementation Cost	Include Operation/ Maintenance Cost	Include NRC Cost	Include Onsite Consequences	Include Avoided Onsite Damage	Include Avoided Offsite Damage
1	5	110	2.4E+07	2.7E+07	3.0E+07	3.1E+07	2.5E+07	1.8E+07
	10	110	2.2E+07	2.4E+07	2.6E+07	2.7E+07	2.4E+07	2.0E+07
2	5	110	3.2E+07	3.6E+07	3.8E+07	4.0E+07	3.3E+07	2.2E+07
	10	110	2.6E+07	2.9E+07	3.1E+07	3.3E+07	2.9E+07	2.3E+07
3	5	110	1.0E+08	1.1E+08	1.1E+08	1.2E+08	1.1E+08	9.9E+07
	10	110	7.9E+07	8.3E+07	8.7E+07	9.8E+07	9.3E+07	8.5E+07

Table 7 Best estimate value:impact ratios ($/person-rem).(a)*

Resolution Alternative	Discount Rate, %	Value:Impact Score - Include Cost Offsets(b)**	Value:Impact Score - Exclude Cost Offsets(c)***
1	5	1.4E+04	8.2E+03
	10	1.2E+04	8.8E+03
2	5	1.1E+04	6.2E+03
	10	9.2E+03	6.4E+03
3	5	2.3E+04	1.9E+04
	10	1.9E+04	1.6E+04

(a)* Values may contain excess significant figures for calculation purposes.
(b)** This ratio is obtained by dividing cost values in Table 6, column labeled "Include Onsite Consequences" by corresponding risk reduction value from Table 5, row labeled "Total Integrated Risk Reduction."
(c)*** This ratio is obtained by dividing cost values in Table 6, column labeled "Include Avoided Offsite Damage," by corresponding risk reduction value from Table 5, row labeled "Total Integrated Risk Reduction."

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DISCUSSION

PARKER: Two questions: (1) With regard to your transient temperature analysis, which computer code was used? (2) What boundary conditions were used in the analysis - normal maximum, abnormal maximum, design-basis accident condition, or other condition?

DALING: (1) We used transient thermal calculations performed by a utility after we (PNL) performed an independent technical review of the computer code (referred to by the utility as HVAC2), room models, assumptions, and input data. A coupled transient solution involving 23 simultaneous equations and 23 unknowns was employed to arrive at the results, as documented in Appendix F of NUREG/CR-6084.

(2) Thermal transients were calculated for both Loss of Coolant Accident (LOCA) and non-LOCA accident conditions.

LEAK TESTING OF BUBBLE-TIGHT DAMPERS USING TRACER GAS TECHNIQUES

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ABSTRACT

Recently tracer gas techniques have been applied to the problem of measuring the leakage across an installed bubble-tight damper. A significant advantage of using a tracer gas technique is that quantitative leakage data are obtained under actual operating differential pressure conditions. Another advantage is that leakage data can be obtained using relatively simple test setups that utilize inexpensive materials without the need to tear ducts apart, fabricate expensive blank-off plates, and install test connections. Also, a tracer gas technique can be used to provide an accurate field evaluation of the performance of installed bubble-tight dampers on a periodic basis.

Actual leakage flowrates were obtained at Zion Generating Station on four installed bubble-tight dampers using a tracer gas technique. Measured leakage rates ranged from 0.01 CFM to 21 CFM. After adjustment and subsequent retesting, the 21 CFM damper leakage was reduced to a leakage of 3.8 CFM.

In light of the current regulatory climate and the interest in Control Room Habitability issues, imprecise estimates of critical air boundary leakage rates--such as through bubble-tight dampers--are not acceptable. These imprecise estimates can skew radioactive dose assessments as well as chemical contaminant exposure calculations. Using a tracer gas technique, the *actual* leakage rate can be determined. This knowledge eliminates a significant source of uncertainty in both radioactive dose and/or chemical exposure assessments.

I. INTRODUCTION

After installation, the quantification of leakage through bubble-tight dampers using conventional methods has been both costly and time consuming. The pressure decay and constant pressure methods used for acceptance testing often require the installation of leak tight, temporary blank-off plates which may be impractical once duct connections have been completed. The soap bubble method used by the manufacturer as a quality control measure is not quantitative and requires access to the damper seat--often not possible in the field.

Lack of a suitable surveillance method for installed, in-service dampers has led to estimates of leakage from 0 to 20 CFM. Such estimates are not useful in evaluating damper leakage unless acceptance criteria are also approximate. This is unlikely since such loose criteria contradict the design intent of bubble-tight dampers. In light of the current regulatory climate and the interest in Control Room Habitability issues, imprecise estimates of critical air boundary leakage rates across these dampers are not acceptable.

Recently tracer gas techniques have been applied to the problem of measuring the leakage across an installed bubble-tight damper with great success. In this technique, a small amount of easily detectable tracer gas is injected upstream of a damper to be tested and the region downstream of the damper is sampled for the presence or absence of this tracer. The existence of measurable tracer downstream of the damper is evidence that the damper allows leakage. Furthermore, by carefully choosing the manner in which the tracer is introduced and sampled, data are obtained that can be used to derive actual flowrate values for leakage through the damper.

The electronegative gas, sulfur hexafluoride (SF_6), was used as a tracer. This gas is generally recognized as non-toxic, non-reactive, and inert. Since it is easily detectable in minute quantities by means of electron capture gas chromatography, SF_6 is an ideal tracer gas for bubble-tight damper leakage investigations. Analytical sensitivity to this gas ranged from 10 parts per million to approximately 50 parts per trillion although this sensitivity level is not generally required for damper leakage testing.

All tracer gas measurements were performed on-site by means of gas chromatographic instrumentation manufactured for field use. The response of a chromatographic monitor to SF_6 is not affected by the presence of other gases in the plant background such as freons and halogenated solvents. In addition, since SF_6 possesses a zero ozone depletion factor, it will not harm the ozone layer.

II. TRACER GAS TESTING

The use of a tracer gas(es) to investigate the flow, migration and dispersion of potentially harmful, noxious, or toxic gases and vapors is well established within the industrial hygiene, indoor air quality, and ventilation engineering communities (1,2,3). During the last few years tracer testing results specific to concerns within the nuclear industry have appeared in the literature (4,5,6). In simplest terms, tracer gas testing provides a means to document the actual performance of an operating ventilation system by tagging and unambiguously tracing one or more ventilation induced flows. This is done by introducing easily measurable, inert, non-toxic, non-reactive gases that are not part of the common industrial background.

The theoretical interpretation and experimental detail necessary to undertake tracer gas testing of complex ventilation systems is provided in the six prior references and will not be discussed further. Application of the principles of mass conservation to tracer injection and tracer measurement conditions allows quantitative information to be obtained on the performance of actual operating ventilation systems.

III. DAMPER LEAKAGE TESTING

In Figure 1 we show a typical experimental set-up for performing a bubble-tight damper test using tracer gas. Briefly, a uniform concentration of tracer is established upstream of the damper in a control volume. Another control volume is established downstream of the damper from which gas samples are analyzed for the presence of tracer. For low leakage rates across the damper, the tracer concentration in the downstream control volume will increase linearly as a function of time assuming that the volume of gas is well mixed. This case is illustrated in Figure 2.

If the flow through the damper is substantial the increase in downstream concentration may not be useful in inferring leakage rate due to the difficulty in assuring good mixing in a rapidly changing flow. In this case a concentration of tracer is established in the upstream test volume and the resulting exponential decay in concentration due to the air flow through the damper is measured in this test volume only. This decay rate can be used to calculate the leakage through the damper as depicted in Figure 3. Note that this technique is analogous to air change rate measurements performed according to ASTM Standard E-741 " Test Method for Determining Air Change Rate by Means of Tracer Dilution" (7).

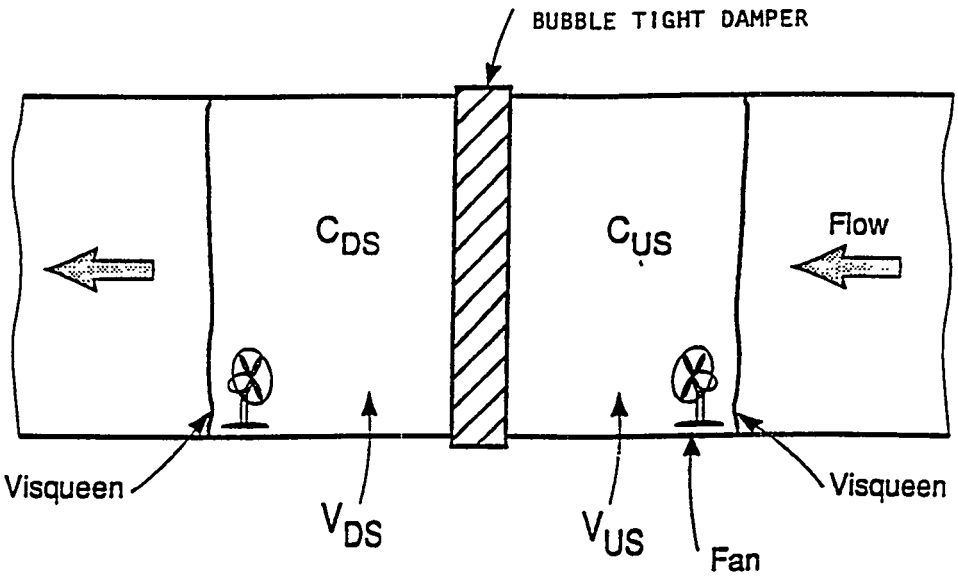


Figure 1. Bubble Tight Damper Test Using Tracer Gas

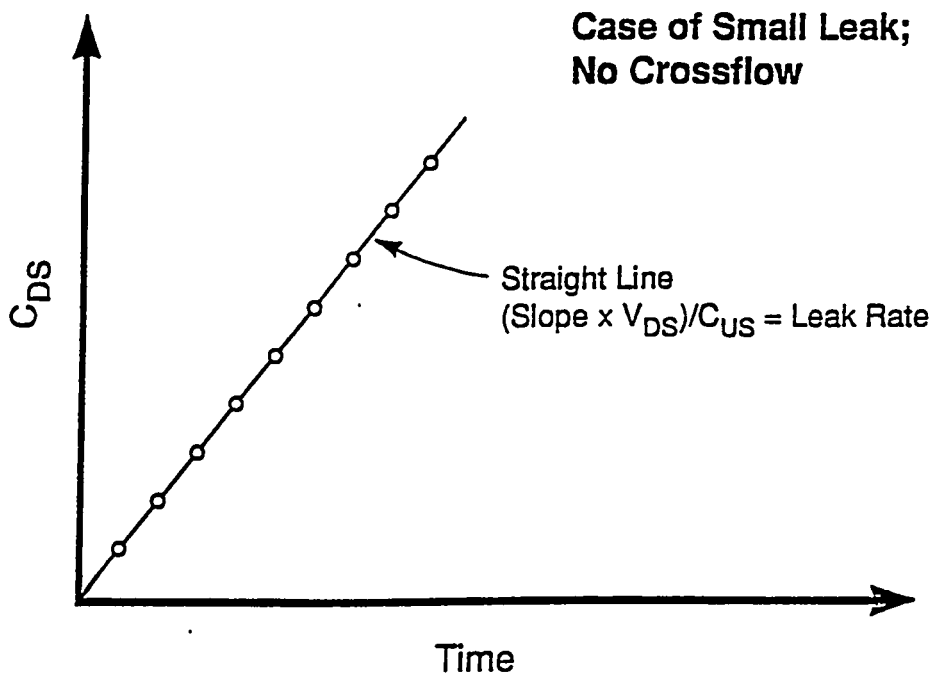


Figure 2. Tracer Concentration versus Time for Damper Test

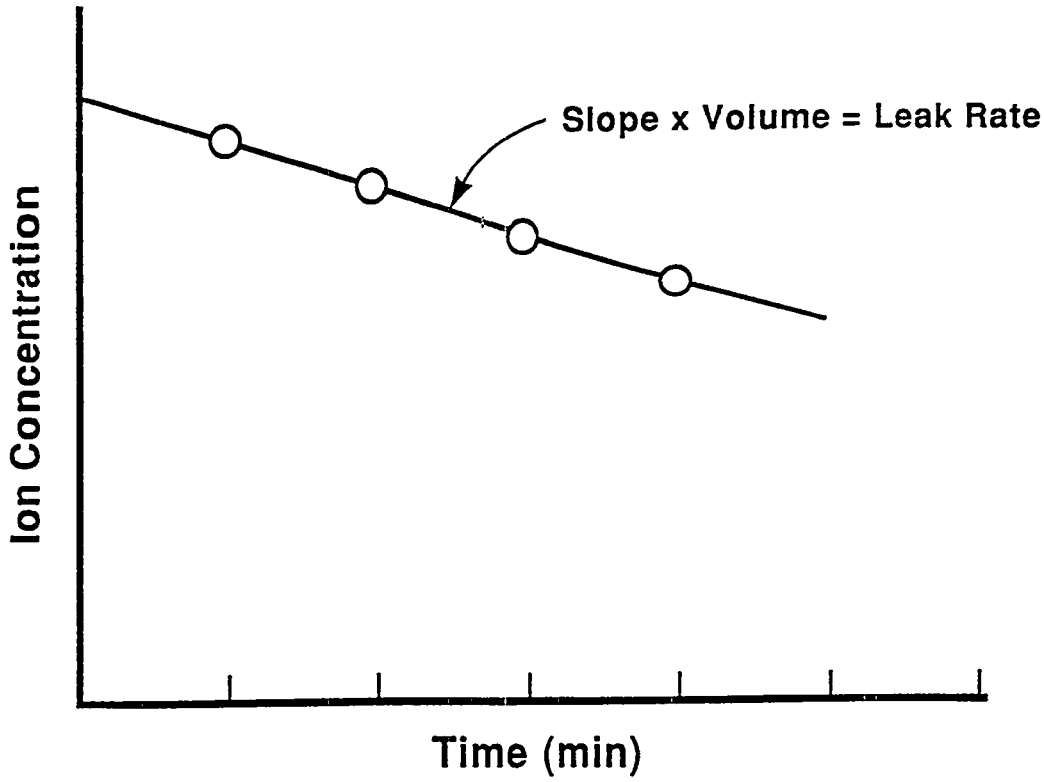


Figure 3. Tracer Decay for High Leakage Through Damper

Experimentally the upstream and downstream volumes are established by using sheets of builder's plastic (Visqueen) to wall off sections of duct. In order to allow pressure equilibrium to occur in the test volumes one or more small slits are cut into each sheet. The slit(s) is(are) then partially covered by additional plastic sheeting to minimize the amount of directed flow (such as might occur from being in the direct path of an induced airflow) that could enter the test volume.

For each test, an amount of tracer is introduced into the upstream volume sufficient to allow sensing of flow through the damper into the downstream volume of 0.005 CFM or greater. Mixing of the tracer within the upstream and downstream volumes is enhanced by placing one or more oscillating fans inside of each volume. For very large or very small test volumes the establishment of good mixing can represent a challenge. Samples of tracer laden air are obtained from the upstream and downstream volumes at timed intervals and analyzed for tracer gas concentration. The resulting data are then combined to provide flowrates using the prescriptions provided in Figures 2 and 3.

In Tables 1 and 2, actual measured leakage data from two of the four bubble-tight dampers that were tested are presented. Table 1 presents tracer concentration data obtained from a damper exhibiting relatively low leakage as well as the basic calculation of the damper leakage rate. Note that the data from test initiation until time equals 4 minutes were excluded to ensure that adequate mixing of tracer had occurred. Table 2, on the other hand, provides data obtained from a damper that exhibited a high leakage rate. A leakage rate calculation as prescribed in ASTM Standard E741 is also presented.

Table 3 summarizes the measured leakage data from all four dampers tested. Note that Dampers 1, 2 and 3 exhibited very low leakage rates. Hence for these tests the tracer buildup pattern was as shown in Figure 2. Damper No. 4 exhibited substantial leakage. The actual value of leakage through this damper was, therefore, calculated by measuring the exponential decay in concentration within the upstream volume as illustrated in Figure 3. The uncertainty estimate for each flowrate is obtained by assuming that the average upstream concentration is not the average but rather one of the extremes. This assumption is *very* conservative, but still results in a reasonable estimate of leakage rate uncertainty. After these tests were completed, Damper No. 4 was repaired by adjusting the damper actuator throw and by re-gasketing the blade seats. It was subsequently retested evidencing the reduced leakage rate shown in Table 3.

As is shown in Table 3, it is possible to accurately measure the existing leakage rate through installed bubble-tight dampers under the *actual* pressure differences that the damper experiences using the tracer gas technique. Depending on the size and ease of access to the damper one to four dampers a day can be tested with this technique.

TABLE 1

BUBBLE TIGHT DAMPER OFCV-PV 39 LEAKAGE RATE DATA

(DAMPER NO. 1)

Time (min)	Upstream Conc. (ppb*)	Downstream Conc. (ppb*)
4	116	0.245
5	97	0.345
6	89	0.513
7	79	0.568
8	72	0.744
9	62	0.856
10	53	0.991

Upstream Volume = 150.5 CU FT

Downstream Volume = 9 CU FT

Calculation: Fit points from 4 to 10 minutes to a straight line (as per Figure 2).

$$C = 0.12468 \times t - 0.26389$$

$$r^2 = 0.997$$

$$C_{av} = 81 \text{ ppb}$$

$$\text{Leak Rate} = 0.124 \times 18 / 81 = 0.03 \pm 0.01 \text{ CFM}$$

* ppb is concentration in parts per billion (10^{-9})

TABLE 2

BUBBLE TIGHT DAMPER OFCV-PV 44 LEAKAGE RATE DATA

(DAMPER NO. 4)

Time (min)	Upstream Conc. (ppb*)	Downstream Conc. (ppb*)	Difference Conc. (ppb*)
1	12	41.3	29.3
2	1.49	19.77	18.28
3	0.59	7.75	7.75
4	0.40	3.10	3.10
5	0.27	1.45	1.18
6	----	0.72	0.72
7	----	0.45	0.45

Downstream Volume = 28.65 CU FT

Calculation: Fit Difference Concentration data to exponential decay curve.

$$C = 30.3 \exp [-0.7429 t]$$

$$r^2 = 0.993$$

$$I = 44.6 \text{ ACH}$$

$$\text{Leak Rate} = I \times (\text{DS Volume}) / 60 = 21 \pm 4 \text{ CFM}$$

* ppb is concentration in parts per billion (10^{-9})

TABLE 3

MEASURED LEAKAGE ACROSS INSTALLED DAMPERS

Damper	Measured Leakage (CFM)	Size
1	0.03 ± 0.01	1.5 ft x 1.5 ft
2	0.03 ± 0.01	1.5 ft x 1.5 ft
3	0.01 ± 0.01	6 ft x 6 ft
4	21 ± 4	4 ft x 6 ft
4 (RETEST)	3.8 ± 1	4 ft x 6 ft

IV.CONCLUSIONS

Actual damper leakage through installed bubble tight dampers has been measured in the range of 0.01 to 21 CFM (after adjustment this damper showed leakage of 3.8 CFM) using a tracer gas technique. To the authors' knowledge these are the first data published on the measured leakage of installed bubble-tight dampers

A significant advantage of using a tracer gas technique to leak test bubble-tight dampers is that quantitative leakage data are obtained under actual operating differential pressure conditions. Another advantage is that leakage data can be obtained using relatively simple test setups that utilize inexpensive materials without the need to tear ducts apart, fabricate expensive blank-off plates, and install test connections.

In light of the current regulatory climate and the interest in Control Room Habitability issues, imprecise estimates of critical air boundary leakage rates are not acceptable. Such imprecise estimates can skew radioactive dose assessments as well as chemical contaminant exposure calculations. Using a tracer gas technique, the *actual* leakage rate is obtained which can then be used in these evaluations thereby eliminating a significant source of uncertainty. The technique also can provide an accurate field evaluation of the performance of bubble-tight dampers on a periodic basis.

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DISCUSSION

BELLAMY: You compared the data for damper four, that you basically categorized as failed, and dampers one, two, and three, which you characterize as passed, although you only presented the data for damper one. For dampers one, two, and three, you did not carry out the test until the upstream concentration was basically zero. Is there any theoretical or practical possibility that the slope of your line, which is basically an indication of the leak rate, is a function of the upstream concentration and that this could have affected the results? In other words, if you carried out the test longer on dampers one, two and three, until you got to a zero concentration upstream, could that have lead to higher leakage rates?

LAGUS: That is a good question and the answer is, it depends. What we tried to do in that test was to get enough data to convince ourselves that we had a good fit to a straight line. If you rely on a conservation of mass calculation, as in the plot that I showed for a small leak, the time scale is determined by the volume of the upstream concentration divided by the leak rate. It is like a flow time. If you wait long enough, flow times will begin to neck over and the equation becomes non-linear. But as long as you use the linear part, it is fairly simple to extract the leakage rate. I have done one test to completion, where the result is a curve that you have to solve iteratively. However, because the slope of the line is $1 - e^{\lambda t}$, you get the same number, within ten percent. But it is much cleaner, I think, to show it this way. It works because you are conserving mass. You can define a small leak in terms of the ratio of the leakage rate times the time divided by the volume that you are leaking into.

PORCO: Does your test method address leakage through the shaft seals when they are outside the duct work if the damper is not encapsulated?

LAGUS: Yes, the method measures leakage across the damper from whatever source. In this case the leakage as measured by the concentration decay in the upstream volume would provide a measure of the leakage through the duct seals.

TARTAGLIA: I have a practical question concerning the use of the VisQueen barrier material. For our applications, we would probably be seeing 4-5 in. w. and I suspect that the VisQueen barrier would bubble or bend, thus making it more difficult to figure out the volume. Did you see that happen in your test?

LAGUS: Good point. You are seeing 4-5 in. w. across the damper but if the damper is not leaking, the pressure is the same from the damper to where you build your VisQueen tent. As the leakage goes up, the VisQueen tent will begin to see more pressure. To take care of that particular problem, we cut a series of slits in the VisQueen tent to get pressure equilibrium. When we did the damper test that showed leakage of 22 CPM, we could see a bulge and we had to keep cutting slits until we equalized pressure. That was our first clue that we had a leaker.

CONSTANT DEPRESSION FAN SYSTEM
A NOVEL GLOVEBOX VENTILATION SYSTEM

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Abstract

In a conventional glovebox ventilation system the depression within the glovebox under normal operation is controlled by instrumentation. In the event of a breach the pressure within the box rises to atmospheric pressure, this pressure rise is detected by instrumentation which in turn operates a quick opening damper in a high depression extract to achieve a 1 metre/sec (200 fpm) inflow through the breach, which can take up to 2 seconds to establish.

This system, although widely used, suffers from two distinct drawbacks:

It takes a finite time to achieve the containment velocity of 1 metre/sec

It relies upon instrumentation to achieve its objectives

A new glovebox ventilation system has been developed by AWE to overcome these drawbacks. This is the Constant Depression Fan System (CDFS) which is based on an extract fan with a flat characteristic. This achieves all the requirements for the ventilation of gloveboxes and has the advantages that:

It has only one moving part - the extract fan

It requires NO INSTRUMENTATION to achieve its objectives

It achieves the containment velocity of 1 metre/sec in the shortest possible time - approximately 0.2 seconds - and tests have shown that containment is maintained under breach conditions

Thus the CDFS is SAFER, SIMPLER and MORE RELIABLE.

I. Introduction

The ventilation requirements for gloveboxes are that they shall be held at a safe depression at all times, usually about 500 Pa (2 ins wg), and in the event of a breach in one glovebox, eg. glove removal, a minimum inlet containment velocity of 1 metre/sec (200 fpm) must be established through the breach in the shortest possible time, with the unaffected gloveboxes on the same extract system being retained at a safe depression.

II. Conventional Glovebox Ventilation System

A conventional glovebox ventilation system such as the High Pressure Extract (HPE) is shown in Figure 1. Simply, this has a self acting control valve to control the depression within the glovebox

under normal operation. In the event of a breach the pressure within the box rises to atmospheric pressure, this pressure rise is detected by instrumentation which in turn operates a quick opening damper in the high pressure extract to achieve a 1 metre/sec inflow through the breach, which can take up to 2 seconds to establish.

This system, although widely used, suffers from two distinct drawbacks:

(i) It takes a finite time to achieve the containment velocity of 1 metre/sec

(ii) It relies upon instrumentation to achieve its objectives

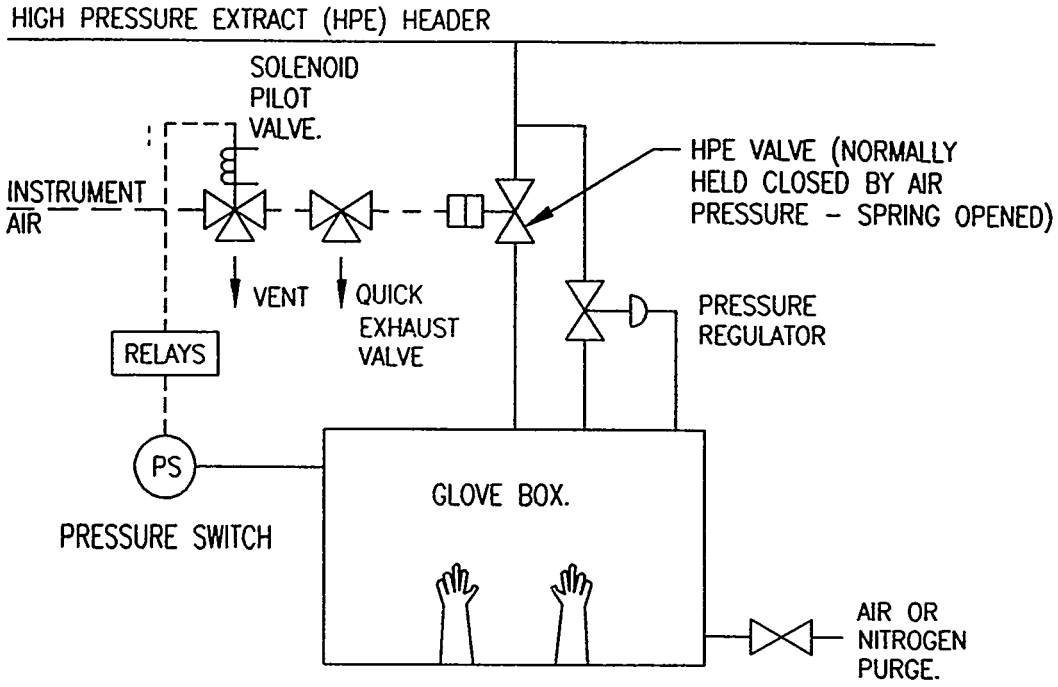


Figure 1 Simplified HPE system

III. Constant Depression Fan System (CDFS)

The basic overall ventilation system is shown in Appendix A. It comprises local HEPA filters, restriction devices, plant room HEPA filters, extract fans and interconnecting ducting. The number of gloveboxes that can be served by one system is limited to the capacity of the extract fans.

The advantages of this system are:

(i) It has only one moving part - the extract fan

(ii) It requires NO INSTRUMENTATION to achieve its objectives

(iii) It achieves the containment velocity of 1 metre/sec in the shortest possible time - approximately 0.2

seconds. This figure has been substantiated by tests carried out at AWE and high speed film has shown that any contaminant within the glovebox is contained under breach conditions

Thus this system is SAFER, SIMPLER and MORE RELIABLE.

IV. CDFS Design - How it Works

The Constant Depression Fan System (CDFS) is a glovebox extract system that provides continuous extract at a constant depression for both purge and breach flow conditions. Under purge conditions all gloveboxes are maintained at the regulatory depression with a given purge flowrate. (See Appendix A). Under breach conditions all unbreached gloveboxes maintain their regulatory depression and purge flowrate, and in the breached glovebox a 1 metre/sec inflow of air through is induced as the breached glovebox depression rises to atmospheric pressure. (See Appendix A).

This is achieved using an extract fan with a relatively flat characteristic curve that will operate in the range between purge flow (normal operating) and breach flow conditions without a significant change in head pressure (See Figure 2).

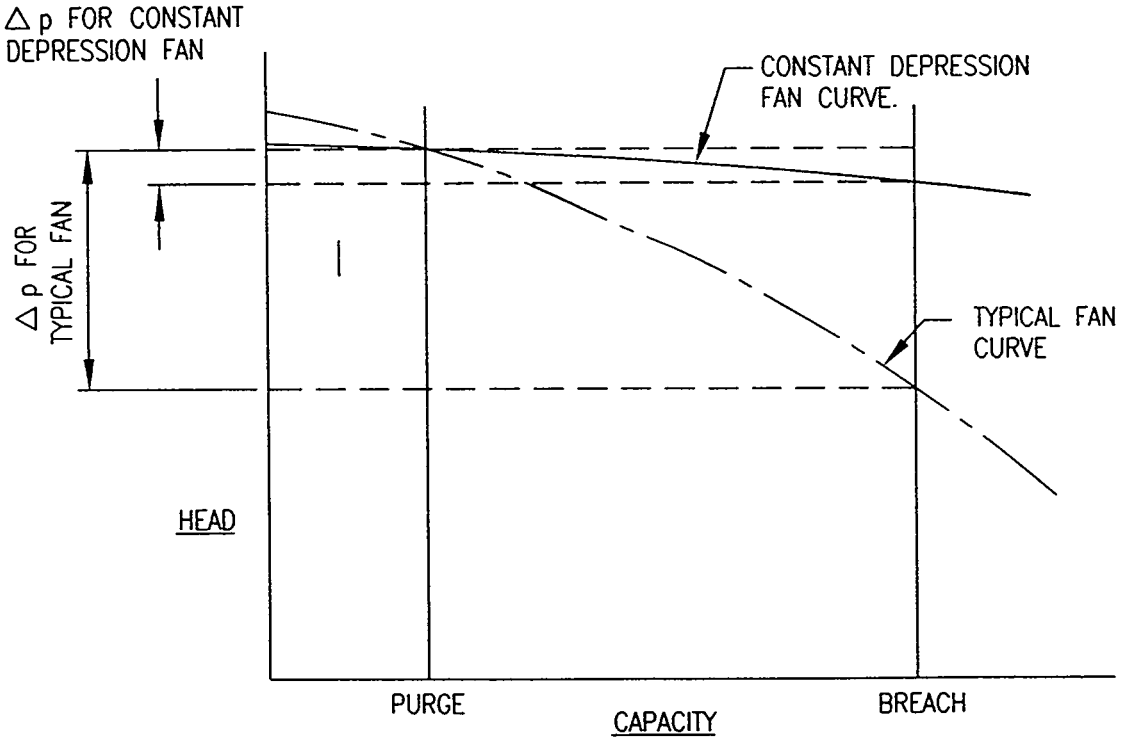


Figure 2 Characteristic of a constant depression fan

The pressure increase (to atmospheric pressure) in a breached glovebox would normally result in a pressure increase in adjoining, unbreached, gloveboxes. To prevent this, a pressure restriction device is installed in each glove box sub header. At purge flow the

pressure drop across the restriction will be negligible. However, as the flow increases (ie. under breach conditions) the pressure drop across this restriction will increase with the square of the flow rate, thereby preventing a pressure rise in the adjoining gloveboxes.

The increased flow resulting from a glovebox breach will result in an increase in the system pressure drop, which to maintain a constant header depression, would require a fan with an increased differential head at the higher flow rate. Such fans are not available.

To overcome this problem, a tolerance of typically ± 65 Pa (total 130 Pa) is allowed on the glovebox header (glovebox depression). This means, that if the depression is nominally 500 Pa, under purge conditions the depression will be 565 Pa and this is permitted to rise to 435 Pa depression under breach conditions. (See Figure 3).

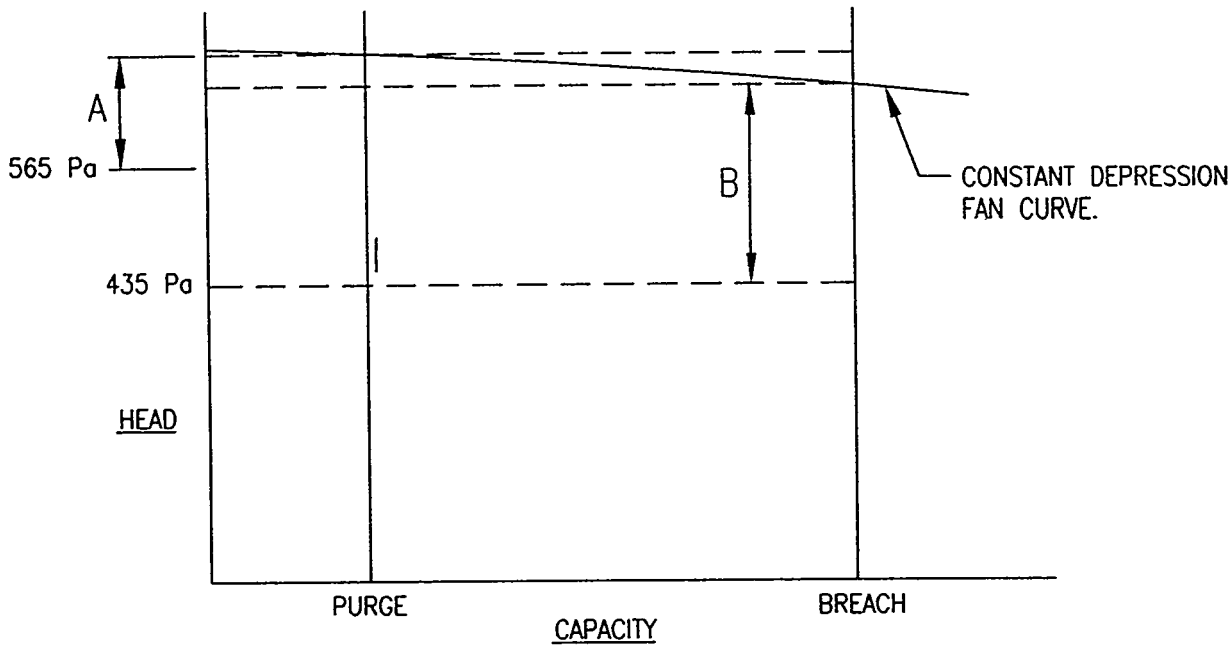


Figure 3 Fan/system characteristics
to achieve header depression tolerance

A System pressure drop under purge conditions (with header depression at 565 Pa)

B System pressure drop under breach conditions (with header depression at 435 Pa)

Note. To meet header depression tolerance $B - A \leq 130$ Pa

Therefore, the difference between system pressure drop downstream of the header in purge and breach flow conditions, must not exceed the glovebox depression tolerance (ie. 130 Pa). This may result in the need for oversized ductwork (and associated fittings) to give minimal change in pressure drop throughout the system.

Extract Fans

The heart of the CDFS is the extract fans which are designed to operate from the normal condition (glovebox purge only) to a breach condition in one glovebox with normal operation in the remaining gloveboxes. The fan differential shall be such that the glovebox header will be maintained within the required depression range under all conditions.

At least two fans are required to provide a run and standby capacity. If there is a design requirement for the system to be operational at all times, even during fan autochange, it will be necessary to provide three 100 % capacity fans. Two fans will be on line, each operating at 50% capacity, the third fan on standby. In the event of the loss of one fan the standby fan will be brought on line, but during the autochange procedure the running fan will fully cover the extract requirements.

Ducting

Ducting is normally sized to provide a duct velocity of 6 to 8 metres/sec. In this system the ducting must be oversized (to normal standards) as the ducting and ducting components must be designed such that the glovebox extract header depression is maintained within the design limits under all conditions.

Plant Room Filters

The plant room filters are two stages of bag change HEPA filters, each stage with run and standby capacity. The filters will be oversized to duty requirements so as to meet the header depression requirements, as detailed above.

Both stages of the plant room HEPA filters should be provided with in-situ filter testing points and the test conducted at the breach flow rate, ie. when the filters are most needed to provide their decontamination factor.

The in-situ test can be carried out in two ways:-

(i) An air bleed, equivalent in capacity to the breach flow, via a local HEPA filter can be provided upstream of the filters. Note! To maintain system operational during in-situ filter testing the extract fans must be sized for a breach in one glovebox + in bleed for in-situ filter testing + purge flow from all remaining gloveboxes.

(ii) A separate fan with connections upstream and downstream of each filter bank can be used to provide the test flow in the standby filter.

Sub Headers

The sub headers are sized to provide a capacity of at least ½ % of the glovebox volume it is serving. This is to ensure that when the gloveboxes are used in nitrogen service there will be no likelihood

of oxygen (air) or moisture from a breached glovebox entering an unbreached glovebox during pressure equalisation.

Local HEPA Filters

The local HEPA filters are, when possible, the push through type.

Where, due to glovebox design, push through type filters cannot be installed, then bag change HEPA filter canisters are provided and installed as close as possible to the glovebox exit.

The pressure differential across the local filters must be monitored so that the filters can be changed as necessary. As the normal (purge) flow through these filters is very low compared to the maximum (breach) flow a low differential pressure device will be required for this purpose.

Restriction Devices

A restriction device is required in each glovebox sub header to prevent excessive flow from a glovebox under breach conditions. The degree of restriction is best established during commissioning and once set it must be locked, or preferably bolted, in position.

Alarms and Trips

Pressure switches should be fitted to each glovebox to initiate alarms in the event of a glovebox breach and to isolate services to the breached glovebox.

Fan Autochange

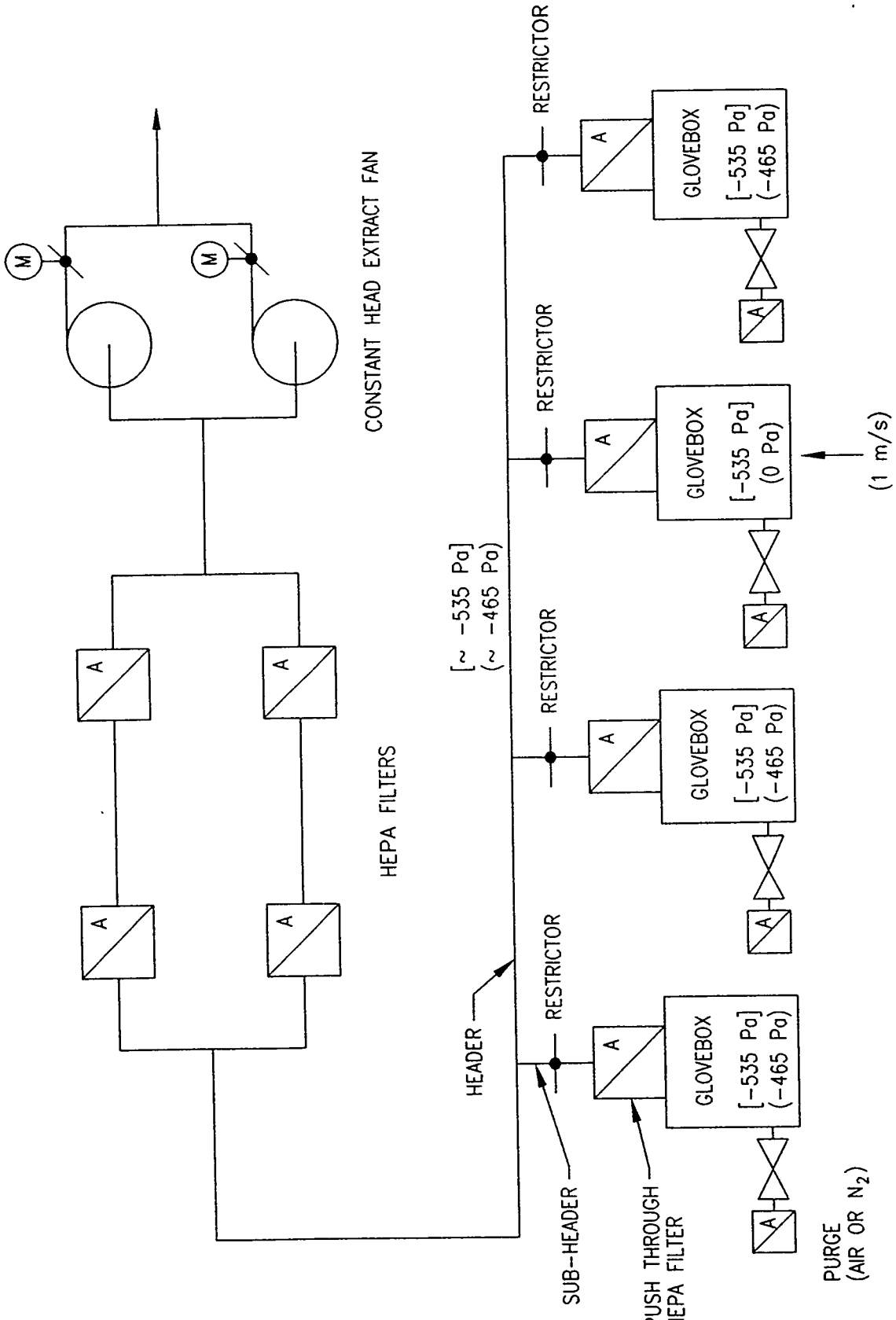
The extract fans must be fitted with a fan autochange facility. To initiate the autochange two parameters to detect fan failure should be used to avoid spurious functioning.

In a two fan operation the fan differential pressure and fan shaft speed can be used as initiating parameters. However, in a three fan operation since the change in flow per fan (upon a duty fan failure) is so small and the differential pressure is constant the only parameter easily measurable is fan shaft rotation.

V. Experience

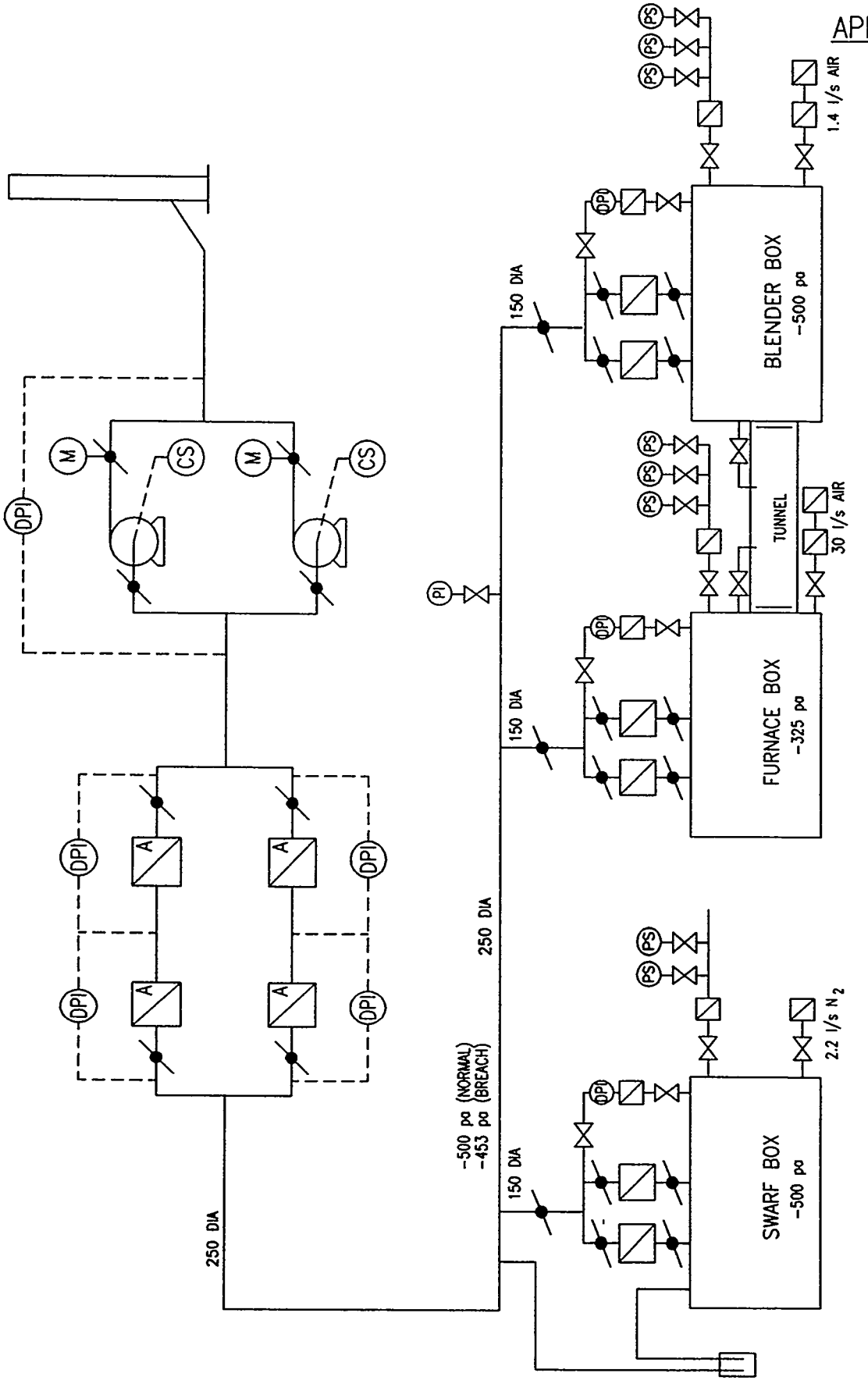
There is one system in operation which serves three gloveboxes, see Appendix B. One glovebox processes swarf under a nitrogen atmosphere with a purge rate of 2 l/s (5 cfm), another contains furnaces which is air purged at a rate of 30 l/s (63 cfm), the third a blender with an air purge of 1.4 l/s (3 cfm). The system is designed to cater for a single breach in any glovebox at a breach flow rate of 42 l/s (88 cfm). To avoid excessive flow under breach conditions in the furnace box the depression is maintained at 325 Pa (1.3 ins wg) instead of the conventional 500 Pa (2 ins wg). It is interesting to note that this one extract system serves both air and inert gas purged boxes, and that the purge rate of the furnace box is similar to the breach flow condition.

APPENDIX A



CONSTANT DEPRESSION FAN SYSTEM FOR GLOVEBOXES
[NORMAL OPERATION] (BREACH).

APPENDIX B



A CONSTANT DEPRESSION FAN SYSTEM
AS USED AT AWE.

* * *

Calculation code evaluating the confinement of a nuclear facility in case of fires

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Abstract

Accident events involving fire are quite frequent and could have a severe effect on the safety of nuclear facilities. As confinement must be maintained, the ventilation and filtration systems have to be designed to limit radioactive release to the environment. To determine and analyse the consequences of a fire on the contamination confinement, IPSN, COGEMA and SGN are participating in development of a calculation code based on introduction, in the SIMEVENT ventilation code, of various models associated to fire risk and mass transfer in the ventilation networks.

This calculation code results from the coupling of the SIMEVENT code with several models describing the temperature in a room resulting of a fire, the temperatures along the ventilation ducts, the contamination transfers through out the ventilation equipments (ducts, dampers, valves, air cleaning systems) and the High Efficiency Particulate Air (HEPA) filters clogging.

The paper proposed presents the current level of progress in development of this calculation code. It describes, in particular, the empirical model used for the clogging of HEPA filters by the aerosols derived from the combustion of standard materials used in the nuclear industry. It describes, also, the specific models used to take into account the mass transfers and resulting from the basic mechanisms of aerosols physics. In addition, an assessment of this code is given using the example of a simple laboratory installation.

I. Introduction

Within the scope of nuclear protection and safety, integration of the fire risk in a nuclear facility mainly involves:

- evaluating the consequences of fire on the confinement and determining the source term released to the environment,
- defining strategies, basically through ventilation control, aimed at reducing these consequences.

In view of the complexity of the problem raised, it is indispensable to use calculation codes integrating the descriptions of the ventilation network and fire, associated to experiment programs.

In collaboration with various teams from the CEA group, the Airborne Contamination and Confinement Research Group has participated in developing a calculation code using the SIMEVENT ventilation software which comprises various modules associated to fire risk and mass transfer in the ventilation networks.

This paper presents the current level of progress made on this code and an application example on a simple installation.

II. SIMEVENT calculation code

The SIMEVENT calculation code has already been presented in various international conferences [1], [2].

It is used to evaluate the behaviour of a complex ventilation network submitted to various disturbances of mechanical and/or thermal origin. The code is based on the division of the system in nodes linked by branches. A node represents a point of the system whose temperature and pressure can be considered as uniform. A branch is a part of the system limited by two nodes and fully defined by the relation $\Delta P = f(Q)$ which expresses the pressure difference (ΔP) as a function of the flow rate (Q) in the branch. A branch may be any kind of classical ventilation element (blower, valve, duct, filter,...). The fluid mechanics constitutive laws of these elements are memorized in the SIMEVENT library.

For a given situation, the code makes it possible to calculate the new values of pressure and temperature at the nodes and the flow rates in the branches. It can then be seen whether the simulated incident causes undesirable effects with respect to safety (reversal of air flow, room overpressure, etc).

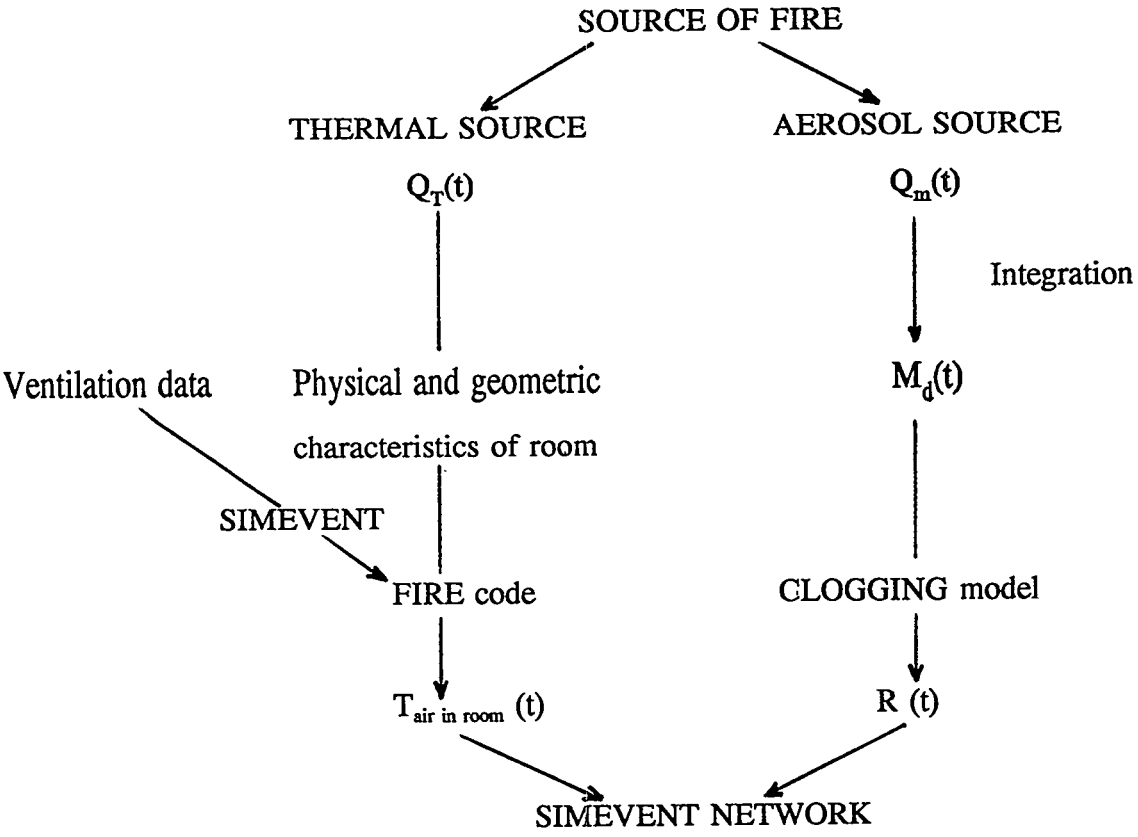
An extension to this code has been proposed to integrate the risk of fire and mass transfer (gases and aerosols) in the ventilation networks.

III. Extension of SIMEVENT code to fire

The extension of the SIMEVENT code to fire implies coupling with a code capable of describing the development of a fire in a room and introduction of models describing the effects associated to the fire (clogging of HEPA filters, thermal loss in ducts, mass transfer).

In particular, the presence of aerosols is a phenomenon which cannot be dissociated from the purely thermal effect of a fire, which specifically results in clogging the HEPA filters installed in the ventilation networks. This phenomenon must therefore be included in the SIMEVENT + FIRE code.

The thermal and mechanical effects in the SIMEVENT code can be schematized as follows:



$Q_T(t)$: heat release rate generated by combustible (variable in time),

$Q_m(t)$: mass flow of aerosols formed,

$M_d(t)$: weight of aerosols deposited on filter,

$$M_d(t) = \int_0^t Q_m(t)dt$$

R(t): aeraulic resistance of clogged filter

$Q_T(t)$ and $Q_m(t)$ are input data of the codes used.

III.1. Fire code

The approach presented here is based on a coupling of the SIMEVENT code with a fire code based on a simplified model giving the temperature rise for forced ventilated compartment fires (one-zone model). This temperature correlation has been developed at the Lawrence Livermore National Laboratory (LLNL) [3].

A more accurate analysis can be obtained if necessary by coupling SIMEVENT with a fire code based on a multi-zone model, such as the FLAMME code [4].

The empirical correlation elaborated at the LLNL estimates the upper layer temperature rise above ambient by :

$$\frac{T - T_a}{T_a} = 0.63 \left(\frac{Q_T}{m_a \, C_{pa} \, T_a} \right)^{0.72} \left(\frac{h \, A}{m \, C_{pa}} \right)^{-0.36}$$

where :
T : upper layer temperature (K)
T_a : ambient temperature (K)
Q_T : fire heat release rate (kW)
C_{pa} : gas specific heat capacity ($kJ.kg^{-1}.K^{-1}$)
m_a : mass ventilation flow rate ($kg.s^{-1}$)
A : compartment surface area (m^2)
h : effective heat transfert coefficient ($kW.m^{-2}.K^{-1}$)

$$h = \sqrt{\frac{k_m \, \rho_m \, C_{pm}}{t}} \qquad \text{if } t < t_p$$

$$h = \frac{k_m}{e} \qquad \text{if } t \geq t_p$$

where :
C_{pm} : wall specific heat capacity ($kJ.kg^{-1}.K^{-1}$)
ρ_m : wall density ($kg.m^{-3}$)
e : wall thickness (m)
t_p : wall thermal penetration time (s)

$$t_p = \left(\frac{\rho_m \, C_{pm}}{k_m} \right) \left(\frac{e}{2} \right)^2$$

t : time (s)
k_m : Wall thermal conductivity ($kW.m^{-1}.K^{-1}$)

This equation has been found to be extremely accurate in predicting upper layer temperatures in the test cell of LLNL with standard fire source of methane gas (approximately constant heat release).

Other comparisons have been made from data collected on tests cells in which polymer fires are performed [5]. These fires are characterized by heat releases Q_T variable with time.

III.2. HEPA filter clogging model

In SIMEVENT, the filter clogging is integrated using the following model:

FILTRE (A, B, C) = R_o, m_{d1}, R_1, m_{d2}, R_2, ... m_{dn}, R_n
where:

A and B are the nodes between which the filter is installed.

C is the node of the fire room (by default, C = A).

R_o is the unclogged filter resistance.

R_i is the filter resistance for a weight m_{di} of deposited aerosols.

This FILTRE model generates the following relation:

$$\Delta P_i = R_i \frac{\mu}{\mu_o} Q_v$$

ΔP_i: filter pressure drop,

μ, μ_o: dynamic viscosity of gas in filter at T and ambient temperatures, respectively,

Q_v: volumetric flow rate at the filter.

The quantity m_{di} of aerosols deposited in time is calculated using the fire source term as indicated above.

Owing to the complexity of the aerosols formed (solid particles + liquid particles due to condensation effects) during a fire and the current knowledge relative to HEPA filters clogging, it is not possible at this time to determine, through a purely theoretical approach, the value of the aeraulic resistance of a clogged filter. This is why the resistance R_i is introduced in SIMEVENT using an empirical approach from fire tests performed using several combustible materials used in the nuclear industry [6] :

PMMA : methyl acrylathe polymer
PVC : polyvinyl chloride of three types :
. pink vinyl
. opaque
. transparent

These materials are used either in the pure state or mixed.

The combustibles are samples measuring about 100 cm^2 and placed in a pan containing alcohol to start them burning. The type and weight of these combustibles are varyied from one test to the next. Most of the fire tests were performed with oxygen in excess of stoichiometric proportions, but some were performed under oxygen depletion conditions.

The HEPA filters used are of two types :

- plane filters consisting of a glass fiber medium, with a filtration area of about 100 cm^2,
- mini-pleats filters with a filtration area of 36 m^2 for a full filter cell. .

A typical example of the results obtained with PMMA fires is given in figure 1, where the variations of $R/R_0 = f(M_d)$, are shown for different types of tests.

R_0 is the filter resistance of the unclogged filter.

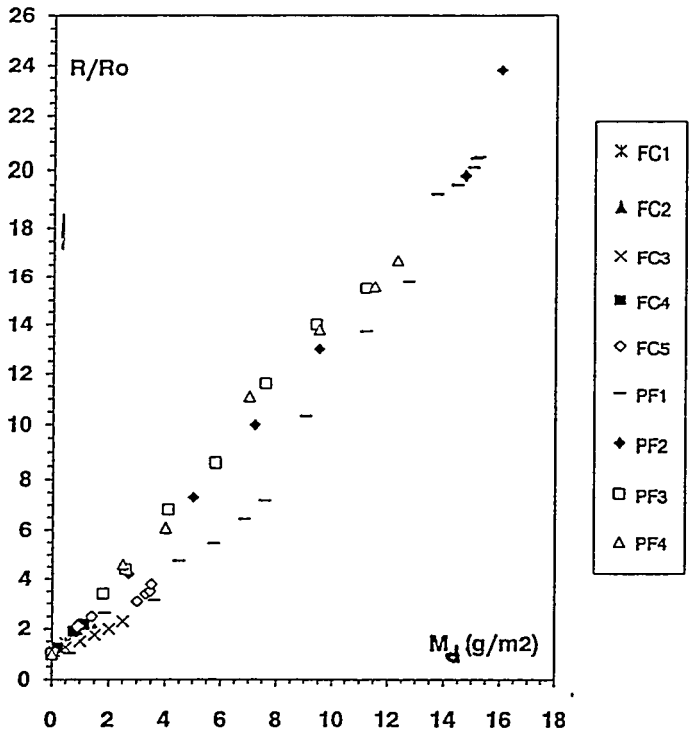


Figure 1 - Variation of R/R_0 as a function of M_d for PMMA fires

Note
FC : test with a full filter cell
PF : test with a plane filter

The clogging curves vary according to the type and composition of the material burned. As an example, figure 2 gives the variations $R/R_0 = f(M_d)$ for a mixture of PMMA and transparent PVC, in different proportions.

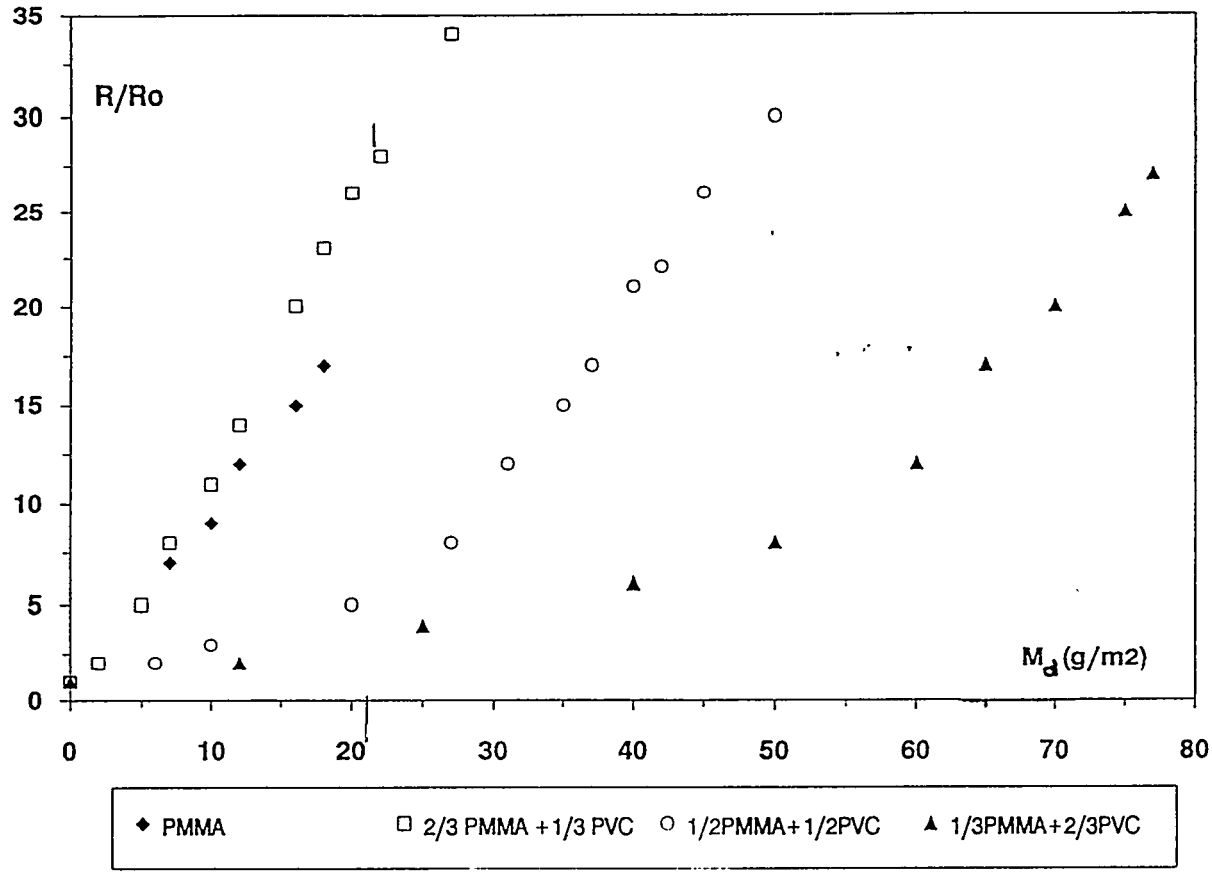


Figure 2 - Effect of the composition of the burned mixture on the variations of R/R_0

In the pure state, it is PMMA that has the greatest clogging power, and for a given mixture, the most penalizing proportions seems to be those at which PMMA is largest in quantity. In those tests where less PMMA is present, and in those where the pure material is difficult to set to flame, the thermal degradation of the material is a pyrolysis rather than open combustion because of the insufficient thermal flux added. The aerosol formation kinetics must therefore be different, which results in less clogging power, due to the nature of the aerosols formed.

Empirical clogging model

Generally speaking, for a given material, we observe that the tests on this material can be correlated with each other. The correlation function f must satisfy the initial condition :

$$\text{for} \qquad M_d = 0 \quad , \quad \frac{R}{R_0} = f(0) = 1$$

A type of function common to all of the materials, satisfying the initial condition, and integrating the variation as best possible, is :

$$\frac{R}{R_0} = 1 + a\,(M_d)^b \qquad (1)$$

in which a and b are constants characteristic of the material and of the thermal degradation considered.

Table 1 gives the constants a and b as well as the variation interval [0, M_d], in which the proposed relation was validated, for several materials tested.

Material	Thermal degradation	a	b	variation interval (g/m²)
PMMA	O_2 Excess O_2 Depletion	0,71 0,61	1,20 1,04	0 -16 0 - 16
Polystyrene	O_2 Excess O_2 Depletion	0,73 0,08	0,86 1,10	0 - 65 0 - 60
Pink PVC	O_2 Excess	0,11	0,26	0 - 10
Opaque PVC	O_2 Excess	0,06	0,73	0 - 4
2/3 PMMA + 1/3 transparent PVC	O_2 Excess	0,5	1,26	0 - 25
1/2 PMMA + 1/2 transparent PVC	O_2 Excess	0,05	1,6	0 - 50
1/3 PMMA + 2/3 transparent PVC	O_2 Excess	0,006	1,9	0 - 75
2/3 PMMA + 1/3 opaque PVC	O_2 Excess	0,32	1,22	0 - 30
1/2 PMMA + 1/2 opaque PVC	O_2 Excess	0,003	2	0 - 85
1/3 PMMA + 2/3 opaque PVC	O_2 Excess	0,018	1,65	0 - 60
9/10 PMMA + 1/10 Pink PVC	O_2 Excess	0,69	1,05	0 - 8
4/5 PMMA + 1/5 Pink PVC	O_2 Excess	0,5	0,95	0 - 5

Table 1

Note

We were not able to obtain identical M_d variation intervals for all of the tests, because some of them caused a degradation of the HEPA filter (mechanical breakage of the filtering medium element sheets for standard size filter cells) and had to be interrupted before of the material was burned.

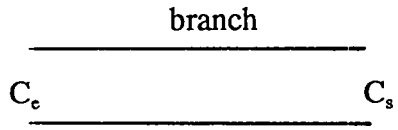
IV. Mass transfer in ventilation networks

To calculate the concentrations of contaminents transferred to various points of the installation and released to the environment, the SIMEVENT code has been modified, by introducing the required models, to take into account the mass transfers in both gas and particle form.

The modifications proposed comprise:

- weight balances at the various nodes of the network indicating the weight conservation of the concerned type,

- weight balances in the branches of the network indicating the mass deposits in the element forming the branch itself.



As a general rule: $C_s = k\ C_e$

C_s : concentration at the branch outlet
C_e : concentration at the branch inlet
k : deposit function

k = f (flow, branch, material)

flow	branch	material
- flow rate	- duct	- gas
- physical properties	- filter	- aerosol
	- register	
	- room	

The deposit function can be entered either by the user or calculated by a specific model of the element forming the branch.

At this time, two specific models are available:

- the deposit function when the branch is a duct (KCONDUIT),
- the deposit function when the branch is a filter (KFILTRE).

These two models are based on the basic mechanisms relative to the deposit of an aerosol on a wall and on the basic mechanisms relative to filtration.

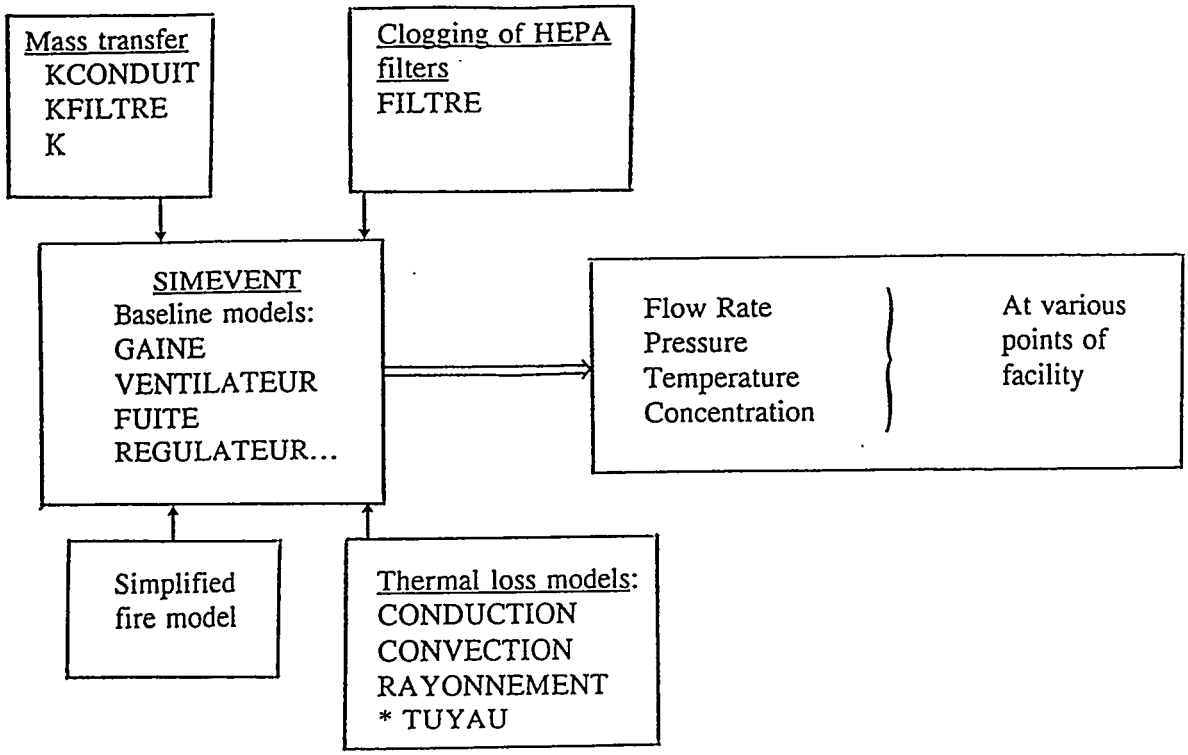
For example purposes, the input data and results provided by the KCONDUIT function are summarized below:

INLET		OUTLET
Aerosol * size distribution: . median diameter . geometric standard deviation * total numeric or mass concentration * density Flow * flow rate * temperature, pressure * physical properties of fluid Duct * shape (horizontal, vertical, slanted, elbowed) * dimensions	KCONDUIT ⟹	* total numeric or mass concentration * concentration per range of diameter

Note: At this time, the mass transfer in SIMEVENT does not take into account the possible changes in phase of the concerned material.

V. General structure of SIMEVENT code applied to fire

The general structure of the SIMEVENT code and of the various models used as part of the studies related to fire is presented below:



VI. Example of application to a simple facility

The facility used for the example (MELANIE test facility [5]) comprises a single room with its associated ventilation network (supply and extraction network with HEPA filter).

The SIMEVENT model involves a fire using a mixture of PMMA (methyl acrylate polymer) + PVC (polyvinyl chloride). The values calculated from the SIMEVENT code applied to the fire then compared to the experimental values are as follows:

- extraction flow rate (Q),
- pressures in room and upstream of HEPA extraction filter (P),
- temperature in room (T),
- concentration of aerosols upstream of HEPA extraction filter (Ca).

The results obtained are given below. Note the good correlation between the results given by the code and the experiment. The results underscore, in particular, a risk of overpressurization of the room, which is a phenomenon to be avoided in a nuclear safety program.

VII. Conclusion

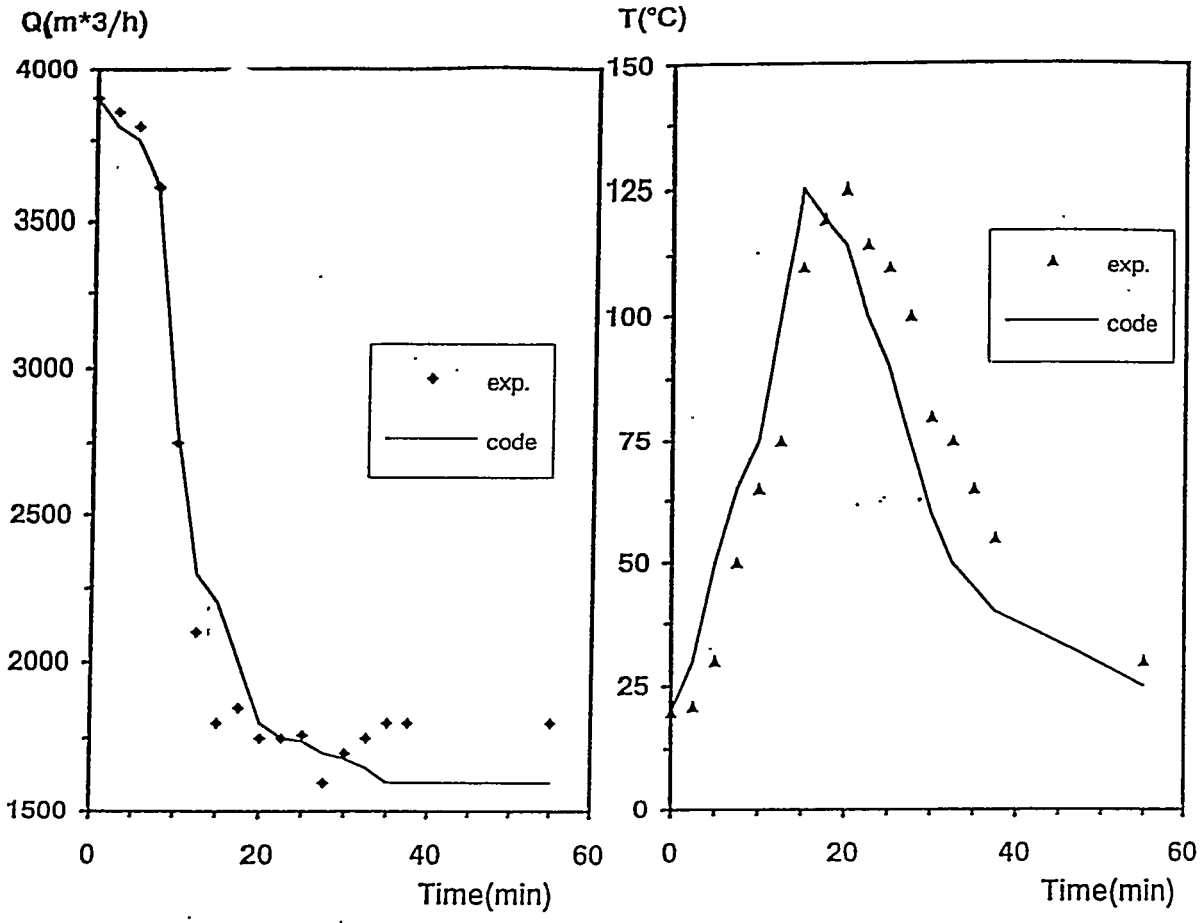
The evaluation of radioactive contamination confinement capacity requires knowledge of the parameters (in particular, temperature, pressure, pollutant concentration) defining the overall state of the ventilation network when a fire breaks out at a given point in the network.

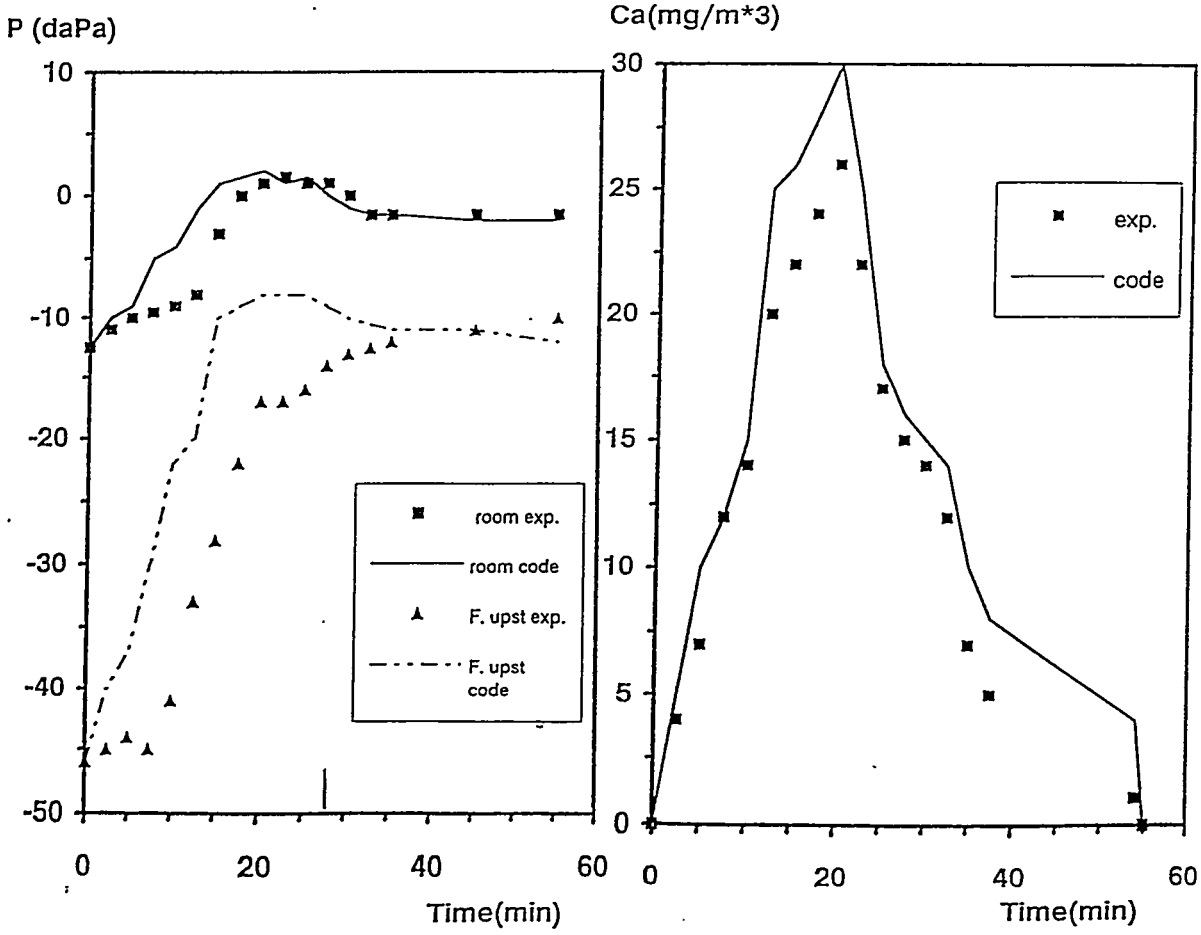
The improvements made to the initial version of the SIMEVENT calculation code provide the required information. In this respect, the SIMEVENT code has been modified by introducing models integrating the various phenomena associated to fire risks: increased temperature in room, thermal losses in ducts, clogging of HEPA filters, transfers of particles and gases in various parts of the network.

With this new version of the code, it is thus possible to verify the risk of a room becoming overpressurized, determine the thermal and mechanical stresses acting on the HEPA filters, calculate the quantities of radioactive substances released. All of these elements are of fundamental importance in evaluating the safety of a nuclear facility.

Furthermore, it is possible to study the influence of the ventilation control (opening or closing of fire dampers) on the system capacity to maintain confinement.

Development of the code is being continued, at this time, to include aerosol deposits in other elements of the network (balancing registers, fire dampers) and to introduce changes in phase of certain composites (gas condensation, for example).


Q(m*3/h)
4000
3500
3000
2500
2000
1500
0
20
40
60
Time(min)
exp.
code
T(°C)
150
125
100
75
50
25
0
20
40
60
Time(min)
exp.
code


P (daPa)
10
0
-10
-20
-30
-40
-50
0
20
40
60
Time(min)
room exp.
room code
F. upst exp.
F. upst code
Ca(mg/m*3)
30
25
20
15
10
5
0
20
40
60
Time(min)
exp.
code

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Performance of HEPA Filters at LLNL Following the 1980 and 1989 Earthquakes*

by

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ABSTRACT

The Lawrence Livermore National Laboratory has experienced two significant earthquakes for which data is available to assess the ability of HEPA filters to withstand seismic conditions. A 5.9 magnitude earthquake with an epicenter 10 miles from LLNL struck on January 24, 1980. Estimates of the peak ground accelerations ranged from 0.2 to 0.3 g. A 7.0 magnitude earthquake with an epicenter about 50 miles from LLNL struck on October 17, 1989. Measurements of the ground accelerations at LLNL averaged 0.1 g.

The results from the in-place filter tests obtained after each of the earthquakes were compiled and studied to determine if the earthquakes had caused filter leakage. Our study showed that only the 1980 earthquake resulted in a small increase in the number of HEPA filters developing leaks. In the 12 months following the 1980 and 1989 earthquakes, the in-place filter tests showed 8.0% and 4.1% of all filters respectively developed leaks . The average percentage of filters developing leaks from 1980 to 1993 was 3.3% +/-1.7%. The increase in the filter leaks is significant for the 1980 earthquake, but not for the 1989 earthquake. No contamination was detected following the earthquakes that would suggest transient releases from the filtration system.

I. Introduction

Previous studies on the ability of HEPA filtration systems to withstand seismic conditions were based on laboratory simulations on shaker tables. Yow et al (1) conducted seismic tests according to standard IEEE Standard 344 (2) and found that in a single filter housing, the separator-less filters survived the seismic tests, but the separator type HEPA filters were damaged. The separators had punctured the filter media in several locations. The authors felt that the separator filter was damaged because the filter was subjected to more vibrations than would be expected. When subjected to more realistic number of cycles, the HEPA filter was not damaged. In-place leak tests based on ASME N510 (3) on a 4 x 3 housing (12 HEPA filters) showed that the housing and both the separator and separator-less filters suffered no damage in the seismic tests. The HEPA filters used in this study had stainless steel or fire

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retardant plywood frames and aluminum separators or no separators. Since a computer analysis indicated that the 4 x 3 filter housing represented the "worst case" design, the other filter housings should also withstand seismic damage. The authors also concluded that the separator type HEPA filters have a potential for puncturing the media in a high vibration environment.

Manley et al [4] conducted seismic qualification tests on a 3,000 cfm control room emergency air cleaning system using IEEE 344. Filter penetration tests based on ASME N 510 [3] were conducted on the air cleaning system with separator type HEPA filters and showed no deterioration in performance before and after the seismic test.

Osaki and Kanagawa [5] conducted seismic tests on HEPA filters and the filter housings under conditions specified in Japanese standards. The HEPA filters used in the tests were the conventional size 5, deep pleat filters, but used steel frames, stainless steel separators and silicone rubber sealant. They found no deterioration in the HEPA efficiency after the seismic tests. They also conducted seismic tests that exceeded the design acceleration of 4 m/s^2 horizontal acceleration and 2 m/s^2 vertical acceleration. The horizontal and vertical accelerations were 45 and 17 m/s^2 respectively. Of 20 filters tested, only two had developed leaks between 0.0001% and 0.00001%. The rest of the filters had less or no leaks. These results were used to seismically qualify the HEPA filtration system.

This report is the first time, to our knowledge, that the performance of HEPA filters was evaluated under actual earthquake conditions. On January 24, 1980, LLNL experienced an earthquake with a magnitude estimated at 5.9 on the Richter Scale. The epicenter was 12 miles northwest of LLNL. Although LLNL did not have any earthquake monitoring equipment in place, the peak ground acceleration was estimated at 0.2-0.3g, with the major seismic component being in the East-West direction [6]. The estimate was computed from measurements at four facilities between 5-12 miles from LLNL [6]. LLNL suffered moderate damage as a result of the 1980 earthquake. Several buildings developed structural cracks that had to be repaired and the buildings reinforced. There was wide-spread office damage due to falling of non-secured objects such as bookcases, files, light fixtures, etc.. However, there was no measurable release of radioactivity to the environment.

LLNL experienced a second major earthquake on October 17, 1989. The earthquake was centered 45 miles southwest of LLNL and had a magnitude of 7.1 on the Richter Scale. At the time of the earthquake, LLNL had 5 operational, triaxial accelerographs that measured peak accelerations in the North-South (NS), East-West (EW), and vertical (V) directions [7]. The averages of the peak ground accelerations were 0.103g for NS, 0.055g for V, and 0.103g for EW. The peak accelerations at higher elevations in buildings and structures were much greater: 0.346g on the 7th floor of B-111, 0.332g on top of the laser spaceform, and 0.379g on top of the laser target frame. There was negligible structural damage to buildings since all upgrades had been made following the 1980 earthquake. There was also little office damage from falling objects. All bookcases and potential falling objects were secured following the 1980 earthquake. There was also no measurable release of radioactivity.

II. Analysis Of In-place Filter Tests

We determined the ability of HEPA filters to withstand earthquakes from an analysis of in-place filter leak tests that are conducted on all HEPA filters at LLNL once each year. No special tests were conducted following either of the two earthquakes since there was no measurable release of radioactivity from a large network of radiation monitors. The in-place tests followed the standard guidelines given in the standard ASME N510 [3]. LLNL used open-face HEPA filters in multi-filter plenums and encapsulated HEPA filters in single filter systems. A single probe measurement was made before and after each filter in the single filter systems while a shroud was used in the multi-filter banks to isolate individual filters. In some plenums, a traversing probe was used to measure filter leaks from individual filters. During the period in this study, 30-55% of the HEPA filters were open face HEPA filters installed in multi-filter plenums.

We assumed that any earthquake damage to a HEPA filter would be reflected in a measurable leak. Our criterion for an acceptable leak was less than 0.03% penetration. If the filter did not have less than 0.03% penetration even after attempts to tighten or reseat the seals, the filter was replaced with a new one. However, since a small percentage of filters develop leaks greater than 0.03% under normal conditions, any leaks generated by earthquakes would be seen as an increase over this baseline level. Therefore, our first step was to compute the baseline level of the fraction of filters that had leaks greater than 0.03%.

Figure 1 shows a plot of the total number of active filters and the number of filters with leaks greater than 0.03% from 1980 to 1993. The occasional decrease in the total number of filters is due to some filtration systems becoming inactive and

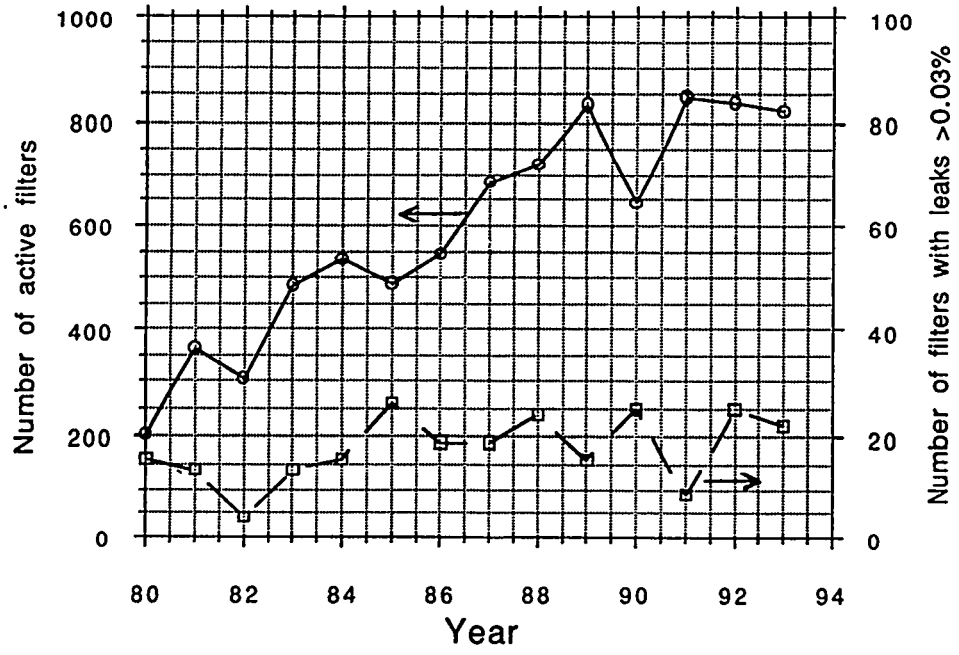


Figure 1 Graph of total number of active filters and filters with uncorrectable leaks

were therefore not included in our analysis. When the systems were reactivated, the HEPA filters were again included in our analysis. Although detailed records of the causes of the filter leaks were not made, most of the leaks were due to inadequate gasket sealing. Defective filter medium was responsible for higher aerosol penetration in a small fraction of the leaking filters. The determination of gasket leakage or media leakage was made by scanning the downstream face of the filter and around the gasket with a probe while challenging the filter with aerosols. Any contribution due to the earthquakes are included in the data in Figure 1.

We then computed the percent of active filters that have leaks greater than 0.03% from the ratio of the number of leaking filters to the total number of filters and plotted the results in Figure 2. The average and standard deviation for the data are 3.3% and 1.7% respectively. The potential contributions to the filter leaks from the 1980 and 1989 earthquakes are included in the data in Figure 2.

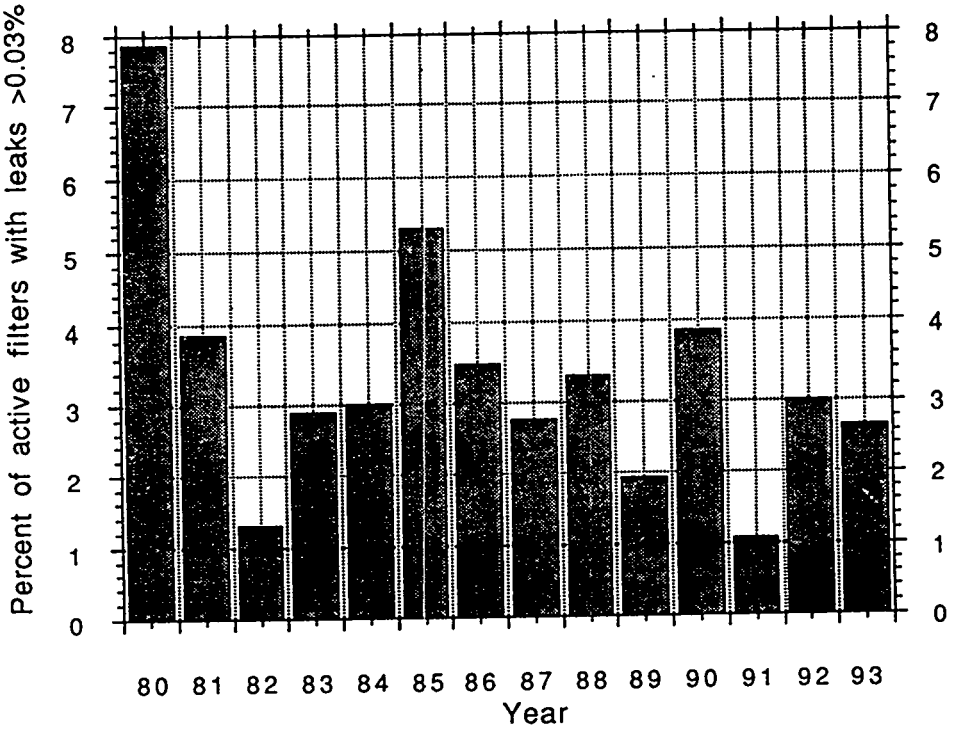


Figure 2 Percent of active filters that have leaks greater than 0.03%

A separate analysis is required to determine the percent of filters developing leaks due to earthquakes. Since the HEPA filters are tested annually, the filter test data for the 12 months following the earthquake were analyzed to determine if any of the filters had developed leaks. The earthquake data differs from the data in Figures 1 and 2, which are based on a calender year, because the 12 month period is shifted from January 1 to the date of the earthquake. Table 1 shows the number of active filters and the number and percent of filters with leaks greater than 0.03%. The time periods covered for the 1980 and 1989 earthquakes are 1/24/80 to 2/1/81 and 10/17/89 to 11/1/90 respectively.

Table 1 Annual HEPA filter leakage following the 1980 and 1989 earthquakes.

Earthquake	Number of active filters	Number of filters with leaks >.03%	Percent (%)
1980	213	17	8.0
1989	741	30	4.1

The percent of filter leakage during the 12 month period following the 1980 and 1989 earthquakes were then compared to the average leakage rate from 1980 to 1993 and tabulated in Table 2.

Table 2 Comparison of the annual HEPA filter leakage following the 1980 and 1989 earthquakes and the baseline.

	1980	1989	Baseline
filters with leaks>.03%	8.0%	4.1%	3.3% +/- 1.7%
peak ground acceleration	0.2-0.3 g	0.1 g	0 g

Table 2 shows a slight increase in filter leaks following the 1989 earthquake and a larger, but still small, increase following the 1980 earthquake. The increase in filter leaks following the 1980 earthquake is statistically significant, whereas the increase following the 1989 earthquake is not.

III. Conclusion

Our study showed that the 1980 earthquake, which had a peak ground acceleration of 0.2-0.3 g resulted in a small increase in the number of HEPA filters developing leaks. In the 12 months following the 1980 and 1989 earthquakes, the in-place filter tests showed 8.0% and 4.1% of all filters respectively developed leaks . The average percentage of filters developing leaks from 1980 to 1993 was 3.3% +/- 1.7%. The increase in the filter leaks is significant for the 1980 earthquake, but not for the 1989 earthquake. No contamination was detected following the earthquakes that would suggest transient releases from the filtration system.

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DISCUSSION

PORCO: You mentioned that the leaks were mainly in the gaskets, although there were some in the filter media. Do you have any figures?

BERGMAN: I would say about 10-20% were media leaks and the rest were gasket leaks.

ANON: Do you attribute gasket leakage to the gasket itself or to the clamping mechanism?

BERGMAN: I cannot answer that question. I can give you one example to show that there is a lack of uniform procedures and practices within DOE. We have seen one or two dozen filters where someone decided to take a regular gasketed filter and squeeze it against the knife-edge of a liquid seal mounting frame that was not designed for it. It did not work very well. I would not count that as a gasket failure, I call it an engineering error.

GREENE: You used a target penetration of 0.03%. I assume you evaluated the filters by a routine in-place test that usually results in an order of magnitude less penetration than 0.03%. Would you get the same results if you had changed the threshold to 0.005%?

BERGMAN: Yes, the results would change dramatically. Most of our filters would fail. The base line of 0.03% was selected based on the Nuclear Air Cleaning Handbook and various standards. There was no justification for using another figure. Rocky Flats and Los Alamos use a leakage criterion of 0.05% instead of 0.03%. The leak rate is not the filter efficiency, filter efficiency would be different.

FARRIS: Can you tell me if there have been any studies conducted to investigate earthquake effects for installations of this type of HEPA Filter?

BERGMAN: Rich Porco did an excellent study that was reported two air cleaning conferences ago. It was an excellent study. They used a whole bank of HEPA filters and took beautiful pictures and a movie.

FARRIS: They used a triaxial table?

BERGMAN: It was a very nice study, so the answer is, yes. They also did in-place testing.

PORCO: We had a complete filter system and we were testing with DOP at the time. We found no change in penetration.

TARTAGLIA: How soon after each earthquake was the filter train (HEPA) testing performed? Your results indicate a degradation in filter performance as a result of the earthquake. Have the results prompted the lab to update their ventilation operating procedures to require HEPA testing and repair, if needed, following any subsequent earthquakes?

BERGMAN: The filters were tested for a 12 month period after each earthquake. The report has just been prepared and it is too early to say what, if any, action will result.

THE ACTUAL PRACTICE OF AIR CLEANING IN BELGIAN NUCLEAR FACILITIES

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Abstract

With 60 % of its power generation from nuclear stations Belgium has 7 nuclear power stations in operation with a total capacity of 5.4 MWe. Enriched uranium is imported and converted to fuel assemblies. The actinides of reprocessed fuel are recycled as MOX fuel. A main waste conditioning operation has been performed in the PAMELA vitrifier.

The actual practice of nuclear air cleaning in the Belgian PWR station DOEL-4 and in the PAMELA-vitrification plant for high level liquid waste is reviewed.

I. Introduction

With 60% of its power generation from nuclear stations, Belgium is one of the most advanced nations for nuclear energy. In this way, Belgium is a leader in the peaceful application of nuclear fission in close cooperation with other European countries. The Belgian nuclear research institute SCK/CEN started its activities as early as 1952. For example, between 1970 and 1985 a Belgian team of 50 persons developed within a European framework various new methods for the cleaning of nuclear off-gases. In 1991, a handbook on the treatment of gaseous effluents at nuclear facilities (1) was edited, compiling the international knowledge in this field.

In this paper, a survey is given of the commercial activities of the Belgian nuclear industry. The actual practice of nuclear air cleaning in two important nuclear plants, e.g. a power station and a vitrifier, will be described.

II. The Belgian Nuclear Industry (2)

Located at Doel (near Antwerpen) and Tihange (near Liège), 7 nuclear power stations are operated by the Belgian utility company ELECTRABEL to generate about 60 % of the annual Belgian electricity. These PWR stations have a net power capacity of 5.4 GWe. Their ventilations systems and off-gas treatment have always been far advanced as well as the quality awareness. The actual practice will be illustrated taken the DOEL-4 power station as example.

Raw uranium at an annual generation rate of 50 THM is produced as secondary material by the local phosphate industry. Enriched uranium is imported and conditioned to fuel assemblies by FBFC at an annual rate of 400 THM. The annual quantity of about 110 THM irradiated fuel from the 7 power stations is reprocessed in France. Recovered uranium and plutonium are used by Belgonucléaire for recycling as MOX fuel at an annual throughput of 35 THM. As well the operators as the environment are suitably protected in these fuel conditioning plants by highly instrumented ventilations systems using standard prefilters and HEPA filters.

These nuclear operations results in various types of waste products. The accumulated quantities at the turn of the century are predicted to be:

- 4,000 m^3 conditioned alpha and high level waste;
- 18,600 m^3 conditioned low and medium level waste;
- 740 m^3 radioactive waste from nuclear research and the application of radioactive isotopes.

The radioactive waste collection, conditioning, storage and disposal are managed by NIRAS/ONDRAF with its technical subsidiary BELGOPROCESS. A master pilot demonstration is the vitrification of waste concentrates from the former OECD-Eurochemic reprocessing pilot plant. This demonstration has been performed at a nominal throughput of 30 kg/h in the German PAMELA plant operated as a joint Belgian-German venture at the BELGOPROCESS site. The advanced off-gas cleaning developed at Karlsruhe with partial assistance of SCK/CEN will be described later. Further, the Belgian nuclear research center SCK/CEN and NIRAS/ONDRAF are preparing the future disposal of conditoned high level waste by an experimental investigation of the burrial conditions in an underground clay layer at a depth of about 200 meters.

III. The Air and Off-gas Treatment at DOEL-4 (3)

The PWR station DOEL-4, put in operation in August 1984, has a nominal power of 1000 MWe with a corresponding thermal power of 3135 MWt. The final barrier of its multiple confinement system consists of a primary containment structure made of concrete with a steel liner. A secondary containment, also made of concrete, is designed to withstand external impacts of projectiles. Both structures are designed to withstand seismic phenomena (SSE).

As usual the primary coolant is water oversaturated with hydrogen to the level of 35 cm^3 of hydrogen per kg water in order to inhibit the radiolysis of the water. During operation this coolant get loaded with radioactive noble gases and halogens which escape during degasification and/or purging of the volume control tank, of the collecting vessel of the primary drains, of the thanks of the bore recovery loop and of the pressurizer relief tank. These hydrogeneous effluents are treated in a recycling loop consisting of a compressor, a hydrogen recombiner and a delay tank as illustrated in Figure 1.

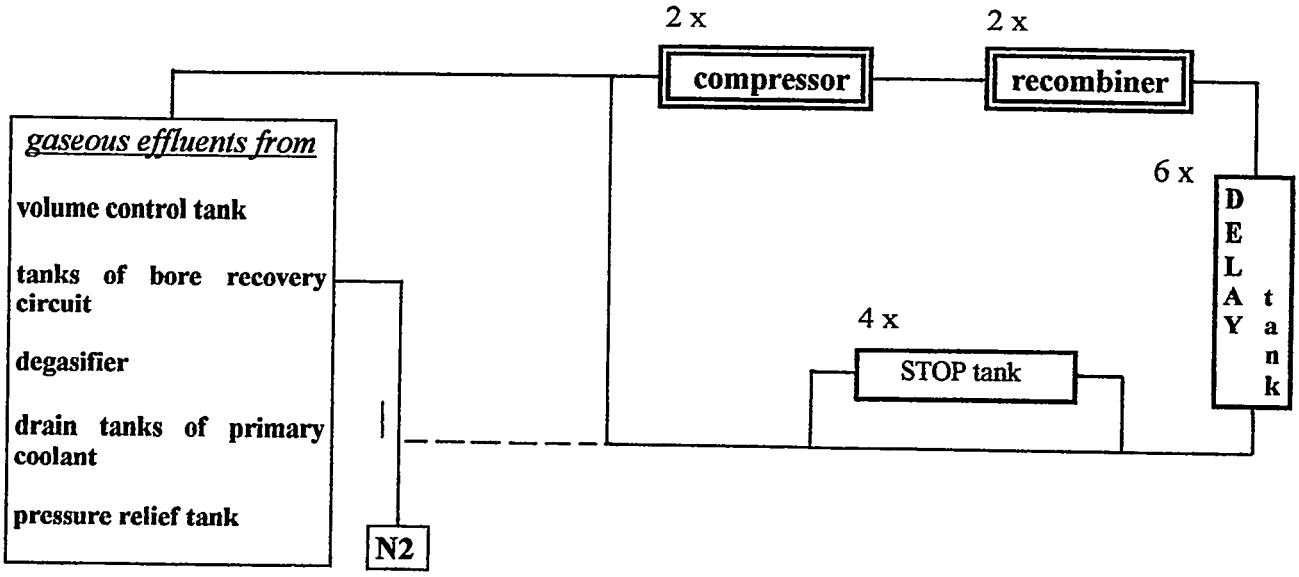


Figure 1 Block diagram of gas recycling loop of DOEL-4.

As soon as the gas pressure at the sucking-in side of the compressor exceeds 70 mbar (rel), the compressor is automatically activated during at least 15 minutes until the inlet pressure is decreased to 50 mbar (rel). A second compressor in stand-by can be started manually if the one in operation can not master. the inlet pressre increase. Continuous operation is foreseen during the purging of the volume control tank. These membrane compressors can compress the gaseous effluent to 8 bar (rel) at a nominal flow rate of 60 Nm^3/h.

The hydrogen present in the compressed gas is converted in water in one of the two recombiners in parallel. At 390 °C, the hydrogen is catalytically combined with oxygen added properly from one of two bottle racks to keep the outlet hydrogen concentration at a minimum of 0.1% hydrogen. Under these conditons the oxygen concentration at the outlet of the recombiner is nearly zero and oxidation of the construction material (austenitic steel) can thus be avoided. Alarms are foreseen when this oxygen concentration attains the levels of 15 or 60 ppm. To avoid dangerous explosive compositions in the recombiner at any time, the oxygen injection is cut as soon as the inlet hydrogen concentration exceeds 4 %; at this hydrogen level purging of the volume control tank is made impossible and the operator gets a warning. Inleakage of air is avoided by overpressurizing the recycling loop at at least 50 mbar. The standard compositionof the recycling gas is in vol % :
hydrogen: 0.1 to 4%; krypton: 0 to 2.2 %; xenon: 0 to 13.6 %; oxygen: 10 ppm; water: at saturation; nitrogen on balance: 80 to 100 % .

The outlet gas of the recombiner is accumulated in a delay tank of 20 m^3. In parallel, a similar delay tank holds up earlier accumulated gas and a third delay tank releases its content under severely controlled conditions into the venting system of the ventilation of the nuclear auxiliairies building. These three tanks are permuted sequentially every 45 days in such a way that the short-lived isotopes have sufficiently decayed before the release into the atmosphere. An additional set of three delay tanks is kept in stand-by for supplementary storage in the case of peak off-gas generation. A pressure relief valve allows to recycle the gas of the delay tank in operation either within the recycling loop or a long the blanket in the vessels of the bore recovery circuit.

In addition, this recycling loop is equipped with 4 stop tanks of 20 m^3 each filled with nitrogen. This nitrogen is reused in each stop to sweep the hydrogen blanket of the volume control tank. One of these stop tanks collect all discharges of safety valves. Further, two drain vessels of 1 m^3 each collect the condensation water from the compressors and the recombiners.

This recycling loop is highly instrumented for hydrogen safety reasons and for health physics reasons. Specific operation features are applied during the start-up of the loop in order:
- to eliminate the oxygen initially dissolved in the cooling water;
- to install safely a hydrogen blanket in the volume control tank;
- to increase the hydrogen content of the primary water to 35 cm^3/kg water.

Likewise, the loop is professionally shut down starting with a hydrogen sweep of the volume control tank to decrease the radioactivity of the gas blanket to an acceptable level and ending with a nitrogen sweep to remove the hydrogen.

At the revision stop in 1992 supplemental instrumentation and improved controling procedures were installed to increase the reliability of the operation of the drain tanks and to improve the

management of the humidity levels in any part of the recycling loop, particularly at the inlet of the recombiners.

The leaks of the radiolytic and fission product gases present in the primary coolant are collected in the various containment systems together with leaks from other nuclear related equipment. All these leaks are diluted in large ventilation masses, which are spefically treated to trap iodine isotopes and particles suspended in the air. The different ventilation systems within the reactor building are listed in Table 1. Use is made of standard prefilters and HEPA filters made from glass fibers. The iodine is retained on active carbon with a minimal layer thickness of 50 mm. The panel bed type configuration is applied and the discharging-loading of the active carbon is performed pneumatically. Each lot of impregnated carbon is checked before use in order to guarantee a minimal retention of 90 % of methyliodide and elemental iodine under the disadvantageous conditions of 70% relative humidity and 50 °C. The emergency vent, the main vent and the reactor vent are mechanically grouped for mechanical reasons.

Table 1 List of the Ventilation Systems in DOEL-4.

	Flow rate at 100 % m^3 /h	redundancy number x line capacity
A. Ventilation of the reactor containment		
at inlet: temperature conditioning of ambiant air and prefilters at outlet: HEPA filter, active carbon filter, HEPA filter	120,000	2 x 50 %
pressure equilibration system	1,000	1 x 100 %
B. Internal recirculation of air in the reactor containment		
internal cooling of the containment air using heat exchangers cooled with water	500,000	6 x 25 %
sweeping the mechanical system of the control rods	59,500	3 x 50 %
cooling the vessel wall and its support	30,000	3 x 50 %
cleaning circuit with prefilters, HEPA filters and carbon filters	40,000	2 x 50 %
C. Air circuits in the intermediary space between building and containment		
circuit for depressurization with in by-pass: prefilters, HEPA filter, active carbon filter, HEPA filter operating only under accidental conditions	300	2 x 100 %
internal recirculation circuit with temperature conditioning, HEPA filter and carbon filter	26,000	3 x 50 %

The purge ventilation system has a throughput of 120,000 m^3/h and a specific pressure equilibration line is foreseen in connection with the ventilation system of the nuclear auxiliairies building. Internal recirculation of the reactor containment air allows:
- to keep the room temperature below 60 °C;
- to cool critical instruments and the reactor vessel wall;
- and to decrease the iodine content and dust load of the local air in such a way that limited access is possible.

In the case of accident, the internal cooling of the reactor containment air is guaranteed with two of the six cooling trains available (in conjunction with the water aspersion circuit). In addition, an internal recirculation of air in the intermediary space between building and containment allows to remove iodine species and particles. Severely controlled venting is possible a long the depressurization circuit equipped with carbon filter and HEPA filter in operation.

In conclussion, the complex combination of ventilation systems and the gaseous waste treatment train are designed to satisfy the emission limits stipulated in appendix I of the 10 CFR 50 document taking as source terms the pessimistic values of a 1 % failure of the cans of the fuel elements.

IV. The Off-gas Treatment at the PAMELA Vitrification Pilot Plant. (4)

The PAMELA vitrification pilot plant was developed under the German High-Active-Waste technology program (1978-1983) and became a German-Belgian joint venture in 1986. Its construction at Dessel, Belgium, was completed in 1984. Approximately 910 m^3 of high level liquid waste concentrates from the former OECD-Eurochemic reprocessing pilot plant have been vitrified yielding 493 metric tons of borosilicate waste glass within the period 1985-1991.

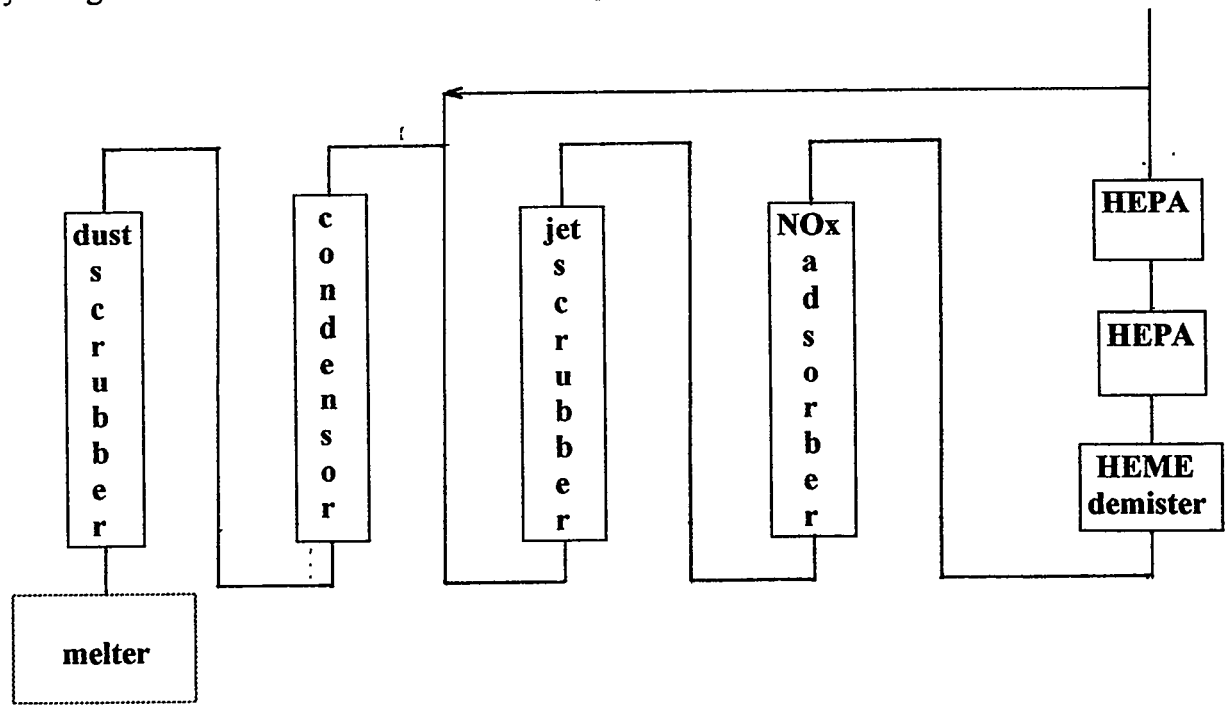


Figure 2 Block diagram of the PAMELA off-gas treatment.

The total alpha radioactivity in the feed during 6 years of operation amounted to 1.51 E15 Bq and the total beta radioactivity to 4.44 E17 Bq. As a result of the high efficiency of the off-gas treatment, illustrated in Figure 2, the accumulated radioactivity released through the stack was limited to 8.5 E3 Bq and 4. E6 Bq for respectively alpha and beta emitters. This emission corresponds to 0.01 % of the licensed value for alpha and 0.03 % for beta radioactivity.

The off-gas of the joule heated ceramic furnace consists mainly of steam (55 %), air (41 %) and NOx gases (4 %). Only 1.5 % of the material fed to the melter is evaporated into the off-gas under severely controlled operating conditions aimed to keep 90 % of the glass pool covered with a heat insulating cold cap. In these circumstances, the off-gas stream leaves the melter at a temperature of 250 °C. The bulk of the larger airborne particulates is removed in a dust scrubber operated under non-condensing conditions and located in the melter cell.

In the first off-gas cell, the steam present in the effluent is condensed in a tube-bundle heat exchanger cooling the off-gas from 110 °C to 40 °C. The condensing steam absorbs NO2 and traps partially solid particulates. The condensed steam is used to replace the high-activity scrub solution which is periodically recycled to the vitrifier. Aerosols are trapped in the subsequent jet scrubber ending in a cyclone chamber for gas/liquid separation. At this point the solid particulates load of the off-gas is lowered to less than 1 mg/Nm^3. The nitric gases are absorbed by a hydrogen peroxide liquor recirculating in a valve plate column scrubber. The secondary wastes thus generated are concentrated in an evaporator before being fed back to the vitrifier.

In a second off-gas cell the entrained nitric acid aerosols are retained on a glass fiber demister type HEME with a mat thickness of 20 cm and a fiber diameter of 4 µm. After heating the off-gas above its dew point, final treatment is performed in a two-stage HEPA filter assembly. At the end, the off-gas is cooled to 30 °C and the residual moisture is removed in a droplet separator. A fraction of the cleaned off-gas is recycled upstream of the jet scrubber in order to control the melter pressure.

Roughly, this off-gas line has a demonstrated DF of at least 10 E11, obtained by an over-all DF of 1.5 E3 for the wet part and an over-all DF of 7 E7 for the three dry cleaning steps. The glass fiber demisting mat located upstream the abolute filters appears to be an essential feature which was introduced after the first demonstration campaign. After 4 years of hot operation, the off-gas pipe between dust scrubber and condensor had to be replaced for corrosion reasons, although Incoloy 825 is the construction material. During the 6 years of hot operation the regular mechanical deplugging of the gas outlet pipe of the melter could be performed without any difficulty. It was also possible to reduce the plugging rhythm to once a month by adapting the standard operation procedure in such a way that the gas temperature at the melter outlet became as high as 600 °C. The over-all material balance of the 6 years of hot operation shows that 98 % of the radioactivity present in the high level liquid waste is embedded into the glass. The remaining 2 % are embedded into bitumen after their retention in the well-performing off-gas purification line. All high level liquid waste present at the Eurochemic site being vitrified, the successful operation of the PAMELA vitrifier has been stopped in October 1991.

V. Conclusion

The two examples of nuclear off-gas cleaning under industrial conditions confirm the expertise of the Belgian nuclear industry. The world première of vitrification of high level liquid waste has demonstrated that this critical part of the back-end of the fuel cycle can be operated reliably and safely

with negligeable burden on the environment. This Belgian experience adds to the confidence that nuclear fission can be an ecologically acceptable electricity source for industrialized countries.

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AN INTRODUCTION TO THE DESIGN, COMMISSIONING AND OPERATION OF NUCLEAR AIR CLEANING SYSTEMS FOR QINSHAN NUCLEAR POWER PLANT

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Abstract

This paper introduces the design evolution, system schemes and design and construction of main nuclear air cleaning components such as HEPA filter, charcoal adsorber and concrete housing etc. for Qinshan 300MW PWR Nuclear Power Plant (QNPP), the first indigenously designed and constructed nuclear power plant in China. The field test results and in—service test results, since the air cleaning systems were put into operation 18 months ago, are presented and evaluated.

These results demonstrate that the design and construction of the air clea ning systems and equipment manufacturing for QNPP are successful and the American codes and standards invoked in design, construction and testing of nuclear air cleaning systems for QNPP are applicable in China.

The paper explains that the leakage rate of concrete air cleaning housings can also be assured if sealing measures are taken properly and embedded parts are designed carefully in the penetration areas of the housing and that the uniformity of the airflow distribution upstream the HEPA filters can be achieved generally no matter how inlet and outlet ducts of air cleaning unit are arranged.

Ⅰ. General Description

The government of the P.R.CHINA proclaimed in 1982 that the first 300MW PWR nuclear power plant (Qinshan Nuclear Power Plant, hereinafter refer to as QNPP) in China would be designed and built indigenously. The plant site was selected in Qinshan, Zhejiang Province. The basic design was started in June 1982 and the detailed design was finished in 1985. By the time of December 15, 1991, QNPP was put into operation after testing, and connected to the main grid. All the field testing items of nuclear air cleaning systems had been completed before October 1991. The performance of nuclear air cleaning systems has been satisfactory since they were put into operation three years ago.

Ⅱ. Introduction to QNPP Ventilation Systems

1 System Description

The schematic diagram of ventilation systems is shown in Fig. 1 and the system description is as follows:

1.1 Containment cleanup system (X1—6A(B))

The system consists of two 50% capacity units located on the operating floor. Each unit is designed to deliver an air flowrate of 6400m^3/h and consists of

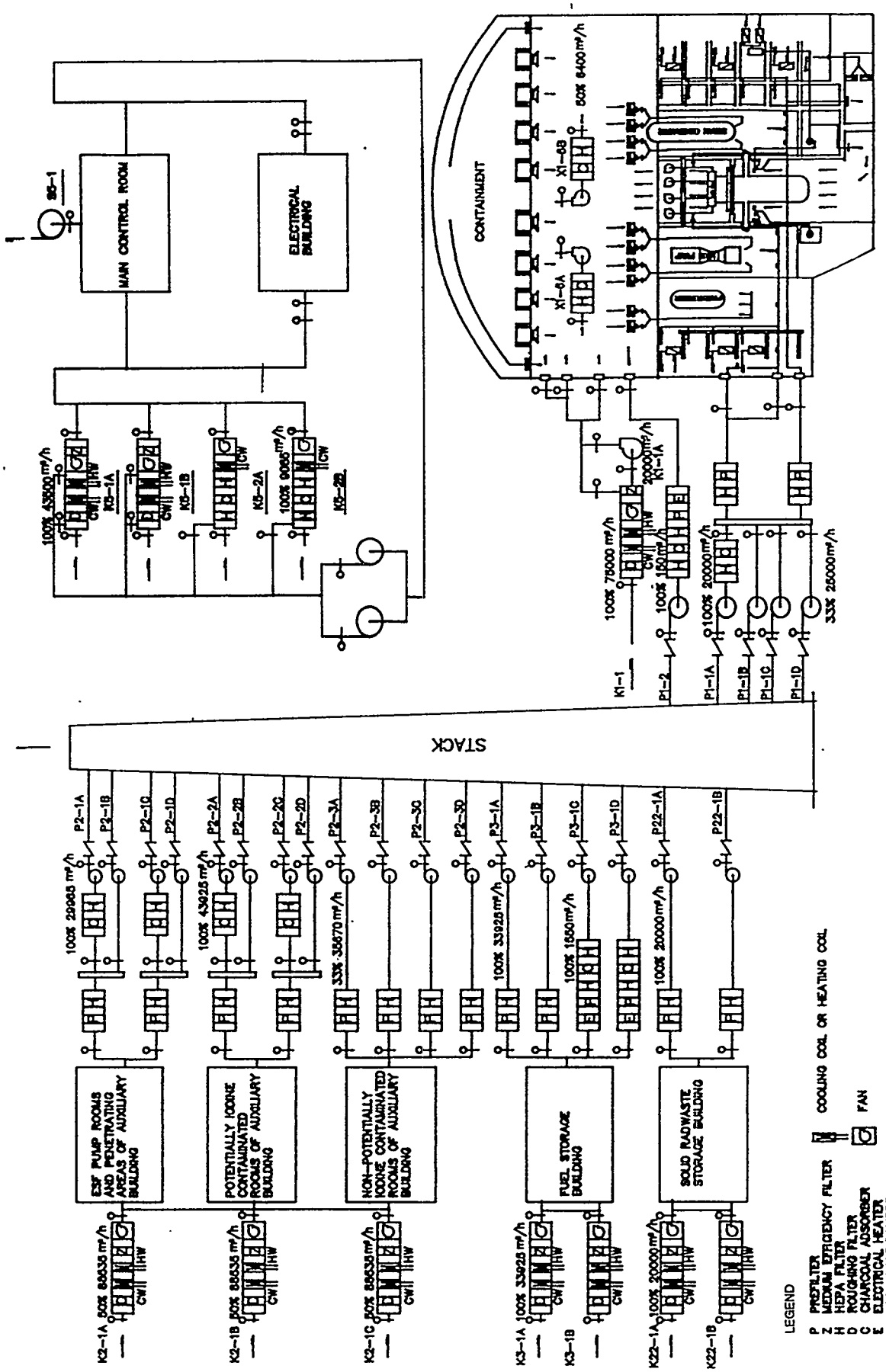


FIGURE 1 VENTILATION SYSTEMS SCHEMATIC DIAGRAM

HEPA filter, charcoal adsorber, centrifugal fan etc.

1.2 Containment purge system (K_1—1, K_1—1A, P_1—1A, P_1—1(B, C, D), P_1—2)

The containment purge system is divided into three subsystems: low volume, high volume and post—LOCA purge (Containment Hydrogen Ventilation System). These subsystems serve the containment (1) during normal plant operating conditions, (2) during planned reactor shutdown, and (3) during post—LOCA operating conditions, respectively.

(1) Low volume purge system

The system consists of the low volume purge supply system (K_1—1A) and low volume purge exhaust system (P_1—1A). The air flowrate of the system is 20000m^3/h. The air change rate is 0.4 containment volumes per hour. The exhaust air of the low volume purge system (P_1—1A) is passed through the prefilter, HEPA filter, adsorber, and afterfilter prior to being discharged to the plant vent stack.

(2) High volume purge system

The exhaust air from this system is filtered through prefilter and HEPA filter to remove radioactive particulates prior to being released to the plant vent stack. The air change rate is 1.5 containment volumes per hour.

(3) Post—LOCA purge system (Containment Hydrogen Ventilation System) (P1—2)

The Containment Hydrogen Ventilation System is designed as a backup to the hydrogen recombiners. After a LOCA, in the event that the recombiners can not be put into operation and after the activity in the containment is low enough to allow the exhaust to the plant vent stack, this system acts to purge the containment and keep the hydrogen concentration below the flammable limit. The exhaust air is passed through prefilters, HEPA filters, charcoal adsorbers and after filters before being released to the plant vent stack.

1.3 Nuclear Auxiliary Building Ventilation Systems (K2—1 A, B, C, P2—1, P2—2, P2—3)

The system is comprised of supply and exhaust subsystems. The supply system is comprised of 3× 50% air handling units (K2—1, A, B, C) The exhaust system is comprised of the following subsystems.

(1) ESF Pump Room and Penetration Area Exhaust System.

There are two redundant exhaust units in this system. Each exhaust unit consists of:

a. One 100% capacity air filter unit which is comprised of prefilter and HEPA filter.

b. One 100% capacity charcoal adsorber unit which is comprised of charcoal adsorber and after filter.

The system is designed to run continuously during all normal plant operating conditions and exhaust air after filtering through prefilter and HEPA fil-

ter banks. Provisions are also made to route the effluents from the exhaust air through prefilter, HEPA filter, charcoal adsorber and after filter on the following signals:

i) Automatically on high radiation signal
ii) Manually through a control switch in the main control room.

On LOCA concurrent with loss—of—offsite power, the other ventilation systems of auxiliary building are tripped and only this system is operated to maintain negative pressure of the nuclear auxiliary building.

(2) Potentially Iodine Contaminated Area Exhaust System

This system is similar to the ESF pump room and penetrating area exhaust system.

(3) Non—Potentially Iodine Contaminated Area Exhaust System

The exhaust air from non—potentially iodine contaminated areas is only passed through the prefilter and HEPA filter prior to the stack.

1.4 Fuel storage Building Ventilation System (K3—1 A(B), P3—1 A(B), P3—2 A(B)).

The system consists of normal ventilation system which contains the fuel storage building normal supply system K3—1A(B) and the fuel storage building normal exhaust system P3—1A(B), and the fuel storage building emergency exhaust system P3—2A(B). The normal ventilation system is designed to operate during normal operating condition. The fuel storage building emergency exhaust system is operated only following the fuel handling accident to maintain 30Pa negative pressure in the fuel storage hall. Meanwhile the normal supply and exhaust ventilation to the fuel storage hall is interrupted.

The fuel storage building emergency exhanst system is designed to route the effluent from the fuel storage building through prefilter, HEPA filter, charcoal adsorber and after filter on the following signals:

i) Automatically on high radiation signal from redundant safety—related area monitors of fuel storage hall.
ii) Manually through a control switch in the main control room.

1.5 Main Control Room Complex Ventilation System (K5—1A(B), K5—2A(B))

The main control room complex is located in the electrical building. The electrical building ventilation system is comprised of two 100% capacity redundant air handling units K5—1A(B), which are operated during the normal condition, and two 100% capacity redundant emergency air cleaning units (K5—2A, B) which are operated only during the LOCA condition.

During plant normal conditions the air handling unit is operated to maintain the room temperature and relative humidity of the main control room complex within the prescribed range.

During the LOCA condition the electrical building ventilation is stopped except for main control room complex. The emergency air cleaning unit is operated to maintain main control room complex habitable and at positive pressure. A minimum outside air is induced through outside air intake to the emergency air cleaning unit, where it is mixed with return air from main con-

trol room complex. Radioactive particulates and iodine are removed as the air is passed through the HEPA filter and charcoal adsorber. Then the mixed air is passed through chilled water coils on its way to the emergency fan inlet.

The emergency air cleaning unit is designed to start and provide outside air to main control room complex in response to any of the following signals:

i) Automatically on a high radiation signal from the radiation monitors installed in outside air intake ductwork.
ii) Manual actuation from the main control room.

In the event of a fire in the main control room, the ventilation system is stopped and the smoke exhaust fan S5—1 will be started, if necessary, to purge the main control room.

2 Design Bases

The design of the QNPP ventilation systems is to protect plant operation and maintenance personnel and the general public from exposure to radiation from airborne radioactive sources. This requirement applies to all operating conditions including refueling, maintenance, and anticipated operational occurrences.

For areas other than the control room, the design philosophy is to prevent releasing of radioactive contamination to the outside. Exhaust air from all potentially contaminated areas is filtered to meet this philosophy.

The design of the ventilation system for the main control room complex is such that, following a postulated design—basis accident (LOCA), radiation doses to main control room personnel for the duration of the accident will be within the limits set forth in 10CFR50, Appendix A, criterion 19.

3 Air Cleaning Unit Design

3.1 Design of Air Cleaning Unit Housing

The 300# reinforced concrete was selected as housing of air cleaning units. Air tightness requirement of air cleaning units prescribed in the nuclear air cleaning handbook is that at the pressure difference of 10 inch water gauge the leakage rate of the housing shall be maintained below 30% of housing volume per hour, which is equivalent to 0.1~ 0.2% of unit's rated flowrate. To meet such a high requirement for air tightness, all the penetrations such as piping, lights, doors in the air cleaning unit housing, as well as contact points between the housing and the components such as prefilters, HEPA filters and charcoal adsorbers and other penetrating components such as drains, ducts and contact points between mountings and concrete shall have proper air tightness. The heavy double hinge doors were employed in the air cleaning units for QNPP.

The mounting frames of the air cleaning units were designed based on the ERDA 76—21. The frame material was stainless steel.

3.2 Air Cleaning Components

(1) HEPA Filter

The HEPA filter used in QNPP has been manufactured in accordance with the requirements of MIL—F—51068 and ANSI/ASME—AG—1. The HEPA filter efficiency exceeded 99.99% (NaCl).

A new laboratory of nuclear grade HEPA filter testing was built in Fifth Nuclear Engineering Research and Design Institute of China in 1985. All the testing items for nuclear grade HEPA filter were finished in 1987.

(2) Medium Efficiency Prefilter

The medium of medium efficiency prefilter used in the QNPP is fiber glass which is water—resistant and fire—resistant. The medium efficiency filter efficiency exceeded 45% based on NaCl testing method.

(3) Charcoal Adsorber

The Ⅱ trap type adsorbers are used in nuclear air cleaning units for QNPP. The adsorbents are coconut activated carbon impregnated with KI and TEDA. The charcoal adsorber beds hold charcoal of 470kg/m^3 density with an ignition temperature of 340℃ . The adsorber bed is two sided (up and down). The bed thickness is 53mm. The cell of adsorber is made of stainless steel with perforation rate of about 32%, which is filled with activated carbon. The adsorber cell has the residence time of 0.25 sec. and airflow resistance of 0.31 kPa at rated capacity of 567m^3/h.

The impregnated activated carbon efficiency for removing methyl iodine at 30℃ and 95% relative humidity exceeded 97% during test.

Ⅲ. Field Testing of Air Cleaning Systems

1 Introduction

The field testing of the air cleaning systems for QNPP, performed in accordance with ANSI/ASME N510 standard, began in Oct. 1989 immediately after the systems were fully installed. The following major instruments were employed during the testing:

DOP Aerosol Generator NUCON F—1000—DG
DOP Aerosol Detector NUCON F—1000—DD
F—11 Halide Gas Generator NUCON F—1000—HG
F—11 Halide Gas Detector NUCON F—1000—HD

The tests were carried out one by one, based on the pre—established test programs and procedures, and were completed in Oct. 1991. The tests lasted nearly two years and the results show that the systems meet the design requirements fully.

When the testing was started, the nuclear power plant was not fully completed. Painting, welding and construction garbage were everywhere. If all the cleaning components had been installed to be tested under these circumstances, a great amount of cleaning components (HEPA filters and charcoal adsorbers) would have been damaged, impaired or even invalidated. Therefore, 100 medium efficiency filters were utilized repeatedly, substituting the HEPA filters when air distribution and air—aerosol mixing uniformity tests were carried out, and other components were not installed at that moment. The testing results demonstrated that same purposes could be reached.

2 Visual Inspection

The visual inspection of the air cleaning systems, which is absolutely necessary, is intended to make preparation for the succeeding test series. Through

visual inspection, many design and construction deficiencies were found (omitted here) and modifications and repairs were made in time so that the succeeding tests for air cleaning systems could be commenced smoothly.

3 Housing Leak Test

The housing leak test was the most laborious and longest lasting work for QNPP field testing. The pressure decay method was applied for the test. The inlets and discharge openings of the housing had been sealed before the test was started and a test fan was connected to the housing to produce and keep the required negative pressure. The leaks were detected and located from the rising of water column of the U—pipe manometer and the leaking sound.

The concrete air cleaning housings are used for QNPP. The field housing leak test showed that leaks of different degrees were found for initial test, mainly located around access doors, penetrating pipes and lighting fixtures etc.

As for the access doors of air cleaning units, owing to the fact that mechanical design of doors was not started when the civil engineering design had been finished, the detailed drawings of the doors with the concrete wall connections were not available. Only door openings were reserved in the walls to be poured secondarily after the installation of doors. It appears that secondary pouring is hard to meet the requirements of sealing and breaches are likely to develop between the secondary pouring concrete and door frames causing leakage. This was detected when leak test with pressure decay method was performed. However, there were no leaks detected in the concrete walls and floors themselves.

Epoxy resin was employed to seal the breaches for field repairing of leaks, which had good results. All the housing met the requirements after final leak test. The results are given in table 1. There have been no leaks reported since the system was put into operation three years ago.

We argue that the sealing of concrete cleaning housings can also be assured if sealing measures are taken and embedded parts are designed carefully in the penetration areas of the housing.

4 Airflow Capacity Test

The test results given in table 1 show that the capacities of all twenty two fans tested have margins and the design air flow rates were achieved whether the filters were under the initial resistance or under the final resistance.

5 Airflow Distribution Test

The deviation of air velocity shall not exceed ± 20% in the profiles of housings prior to the cleaning components. The sizes of some air cleaning units in QNPP are very large, e.g. the width of the housing for P2—2C system is as great as 8 meters and the air leaving the housing passes a drastic turn and enters a narrow passage (the width of which is about 1.4 meters). Refer to Fig.2. This seems unreasonable from the view point of design. However, the testing results show that the air passed through the cleaning components uniformly even for P2—2C housing. This may be because the velocity is low and the resistance is high as the air passes through filter bank which acts as an excellent flow rectifier.

The testing results are given in table 1.

Table 1 Field Testing Data Summary

System No.	Designation	Housing Leak				Airflow Capacity		
		Testing pressure (Pa)		Leakage (m^3/h)		Design Airflow Rate (m^3/h)	Deviation (%)	Deviation (%)
		Initial	Final	Max. Allowable Rate	Tested Rate		Under Initial Resistance	Under Final Resistance
P1−1	P1−1A	3500	2100	44.7	25.4	20000	−0.91	−4.99
	P1−1B	3500	2100	43.2	28.9	25000	2.67	0.13
	P1−1C	3500	2100	43.2	34.7	25000	0.40	−0.67
	P1−1D	3500	2100	43.2	34.7	25000	0.40	−0.67
P2−1	P2−1A	3500	2100	49.6	26.4	29965	1.12	−8.56
	P2−1B	3500	2100	46.5	37.8	29965	6.50	−4.22
	P2−1C	3500	2100	53.5	32.4	29965	6.50	3.10
	P2−1D	3500	2100	63.9	37.2	29965	1.12	−5.00
P3−1	P3−1A	3500	2100	49.5	34.7	33925	−0.66	−9.21
	P3−1B	3500	2100	53.4	37.3	33925	−1.54	−0.66
P2−2	P2−2A	3500	2100	84.9	29.4	43295	1.60	−6.91
	P2−2B	3500	2100	84.0	35.0	43295	9.52	4.80
	P2−2C	3500	2100	84.9	34.3	43295	3.25	−1.40
	P2−2D	3500	2100	84.0	62.6	43295	8.80	7.86
P2−3	P2−3A	3500	2100	57.9	26.6	35670	1.77	4.29
	P2−3B	3500	2100	47.1	22.0	35670	−1.03	−2.16
	P2−3C	3500	2100	46.1	19.4	35670	6.25	1.77
	P2−3D	3500	2100	48.6	21.8	35670	2.33	−7.48
P22−1	P22−1A	3500	2100	40.5	33.5	20000	9.00	3.50
	P22−1B	3500	2100	49.2	34.5	20000	4.50	−2.00
K5−2	K5−2A	2250	1350	29.7	9.6	9085	5.67	3.47
	K5−2B	2250	1350	29.7	7.1	9085	9.07	5.67

Table 1 Field Testing Data Summary (continued)

Airflow Distribution						Air—Aerosol Mixing Uniformity			In—place Leak	
HEPA filter Bank			Charcoal Adsorber Bank			HEPA filter Bank			HEPA Filter	Charcoal Adsorber
Average Velocity (m/s)	Max. Positive Deviation (%)	Max. Negative Deviation (%)	Average Velocity (m/s)	Max. Positive Deviation (%)	Max. Negative Deviation (%)	Average Concentration	Max. Positive Deviation (%)	Max. Negative Deviation (%)	Leakage Rate (%)	Leakage Rate (%)
1.43	6.99	−6.99	5.0	10.0	−12.0	32.5	9.1	−4.7	−0.019	<0.05
1.39	15.11	−17.33				10.1	8.7	−16.0	0.033	
1.33	9.02	−9.02				10.3	9.1	−16.3		
1.33	9.02	−9.02				10.3	9.1	−16.3		
1.47	12.24	−9.52	4.9	16.3	−18.4	71.0	12.7	−12.7	0.009	<0.05
1.32	6.06	−9.09				49.9	18.2	−13.8	0.027	
1.36	11.76	−13.23	5.6	19.7	−16.1	58.3	15.3	−5.7	0.008	<0.05
1.46	6.16	−9.58				1.2	4.2	−8.5	0.031	
1.32	7.57	−5.30				5.7	14.0	−12.3	0.002	
1.36	6.61	−4.41				8.6	14.0	−9.3	0.003	
1.56	5.76	5.76	5.0	12.0	−16.0	53.9	15.9	−13.7	0.015	<0.05
1.38	8.69	5.79				3.1	9.7	−16.1	0.005	
1.30	7.69	−7.69	5.90	14.3	−14.3	60.6	14.2	−17.5	0.039	<0.05
1.26	11.11	−8.73				1.1	9.1	−9.1	0.005	
1.54	5.84	−12.33				44.6	7.6	−14.8	0.011	
1.47	10.20	−10.20				19.5	23.1	−20.5	0.016	
1.50	10.00	−14.66				31.5	6.3	−11.1	0.003	
1.38	5.79	−7.24				32.4	3.7	−6.8	0.023	
1.61	11.80	−13.04				30.5	40.9	−34.4	0.019	
1.54	13.63	−12.33				43.3	27.0	−23.8	0.038	
1.34	9.70	−8.95	4.9	12.2	−12.2	14.2	27.0	−23.8	0.000/0.004	<0.05
1.36	4.41	−2.94	5.0	14.0	−12.0	50.5	14.9	−10.0	0.001/0.003	<0.05

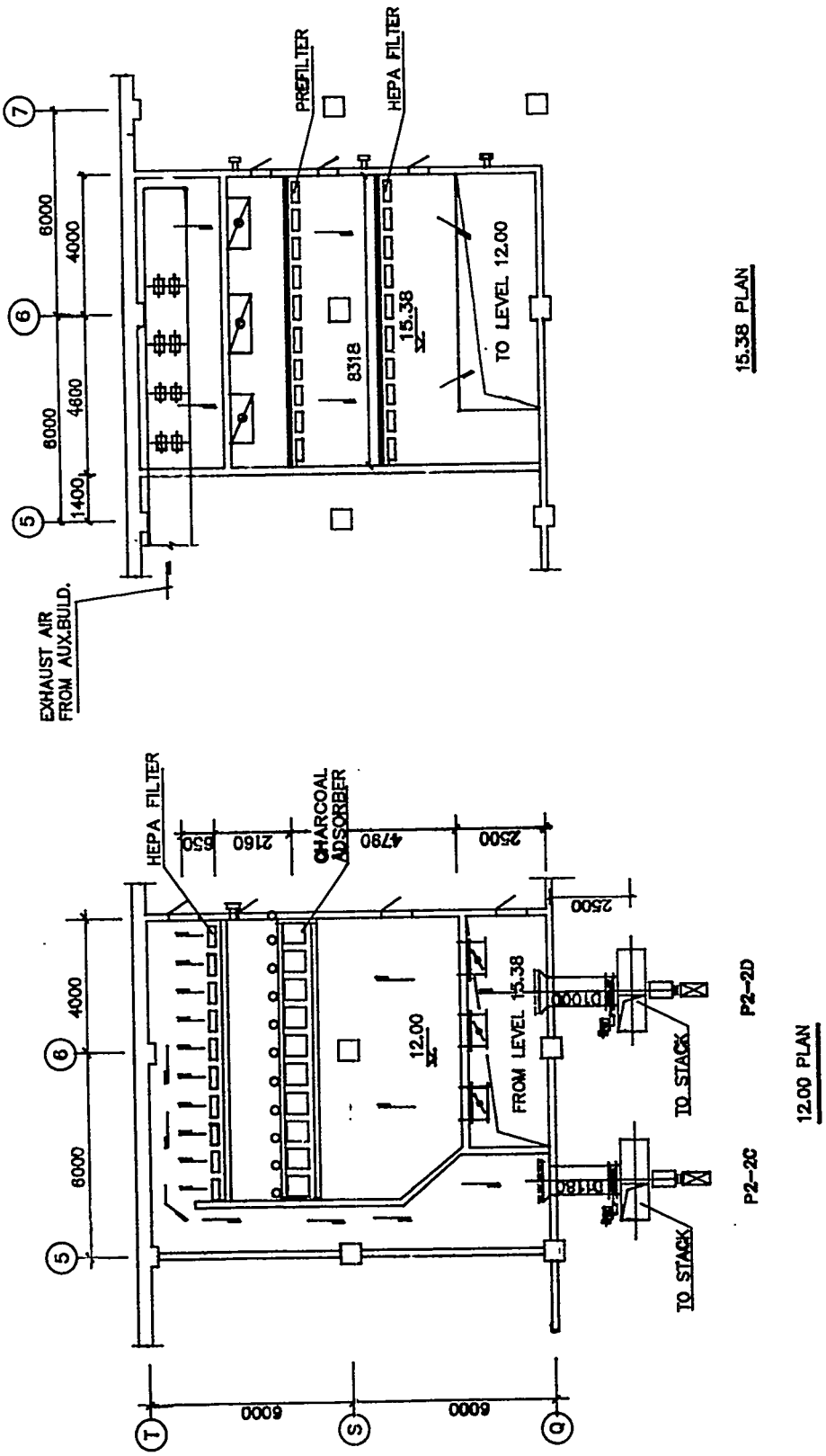


FIGURE 2 AUXILIARY BUILDING AIR CLEANING UNIT LAYOUT P2–2C(D)

6 Air—Aerosol Mixing Uniformity Test

The air—aerosol mixing uniformity test is to inject the challenge into the duct approaching the housing and test the uniformity 30cm upstream the HEPA filters. All twenty—two HEPA filter banks were tested for air—aerosol mixing uniformity. The results given in table 1 show DOP challenge and air have reached uniformity approaching most HEPA filter banks except a very small few.

7 HEPA Filter Bank In—place Test

The HEPA filter bank in—place test was carried out after housing leak test, airflow capacity and distribution and air—aerosol mixing uniformity test etc. had been completed.

For the air cleaning units of P1—2 and P3—2 which have four banks in series (medium efficiency filters, HEPA filters, charcoal adsorbers and after filters), all the banks except the medium efficiency filter bank were tested for leakage. The DOP challenge was injected into the airstream farthest to the banks in order that the air and DOP mixed uniformly in the duct. The procedure is as follows: The after filter bank was tested first for leakage while HEPA filters and charcoal adsorbers were not installed. Then the HEPA filters were tested for leakage and the charcoal adsorbers were tested last. Most HEPA filter banks passed the initial test. Only a few, e.g. those for P2—2C and P22—1B, failed because the nuts were not tightened. They passed the second test after tightening.

8 Adsorber Bank In—Place Test

A halide challenge gas R—11 was used for the field test. However, the adsorbers were tested for leakage with mass spectrometers using R—11 as a challenge gas in the factory, which lasted relatively a long time, and were not desorbed. The R—11 remaining in the activated carbon would desorb in the field testing, causing pseudo—leakage, which would affect the accuracy of the test. Thus, the adsorbers must be desorbed before field test. Laboratory sample testing showed that the desorption of R—11 was very slow when the air was in the same direction as the testing air and it was much faster with the reversed air. Incidentally, the installation direction of charcoal adsorbers in the air cleaning housing for P2—1 A/B for QNPP is reversed to that of the others: the air passes through the adsorbers reversely. All the charcoal adsorbers were desorbed in the housing of P2—1A/B. When the relative humidity of air is between 80 ~ 84%, the temperature is between 29 ~ 31 ℃ and the air velocity through the carbon bed is greater than or equal to 20cm/s, the desorption time is about 7 ~ 8 hours. After desorption, the in—place leak tests were performed for all the charcoal adsorber banks. The testing results are shown in table 1.

9 Laboratory Test of Activated Carbon Samples

The laboratory test of activated carbon was carried out in October, 1991. The testing results are given in table 2. The table shows that all the activated carbon in the adsorber banks in QNPP is acceptable.

Ⅳ. Operation

It had been 18 months since the air—cleaning systems in QNPP were put into operation from Oct. 1991 to April 1993. Periodic tests must be performed for the air cleaning sysetms, especially the leakage rate according to the requirements of ANSI/ASME N510. Therefore, the HEPA filter bank and charcoal ad-

sorber bank leak tests were performed from April 15 to May 23, 1993. The results are given in table 3. The testing results show that most air cleaning units are in good condition after operation of 18 months except for that of P2—1B, for which the HEPA filter mounting frame leakage rate exceeds 0.05%. By—pass flow in some HEPA filters was detected after scanning the banks one by one and the leaks were eliminated after the nuts were tightened.

The prefilters were changed for the first time after being put into operation for 18 months. At that time, the resistance of the changed prefilters had reached or exceeded 700Pa, which surpassed the initial resistance (100Pa) greatly and one prefilter pack in the P2—2A system had been pulled out under such pressure difference. The HEPA filters, however, had never reached their final resistance (only 300Pa after 18 months), the design initial resistance being 250Pa and the final resistance 1000Pa, 4 times the initial resistance. Because the pressure head of the fans in the cleaning systems is sized according to the final resistance of each cleaning components, it still has margin and the air flow rate of the systems can also be assured.

The resistance of the HEPA filters, protected by the medium efficiency prefilters (shop testing showed η=70~ 80% for 0.5μ particles (NaCl)), increased very slowly. It only increased from 220Pa of initial resistance to 300Pa after 18 months (May 1993) and to 340~ 350Pa (April 1994). Therefore they have not been changed yet.

The charcoal adsorbers have never had to be put into operation because of the presence of radioactive iodine detected in the exhaust air of the cleaning system. They were only put into operation for half an hour per month, performed for specified periodic test.

The laboratory tests for testing canisters after 18 months showed (see Table 4) that the charcoal in some systems had aged and the iodine removal efficiency dropped to less than specified 99% for CH_3I under 80℃ and 70%RH testing condition. Consequently, they had to be changed also.

V. Conclusion

(1) The nuclear air cleaning systems for QNPP have been in good condition since they were put into operation three years ago. This demonstrates that the design, construction and field testing of the nuclear air cleaning systems are successful.

(2) The success of the field testing of the nuclear air cleaning systems for QNPP proves that the efficiency of nuclear air cleaning systems designed and constructed indigenously can be assured.

(3) The design, construction and field testing of nuclear air cleaning systems for QNPP are carried out in accordance with American codes and standards. This demonstrates that these codes and standards are applicable in China.

Through practice, we have gained valuable experiences in design, construction and field testing of nuclear air cleaning systems and the nuclear air cl eaning industry of our country has got on a new stage.

Table 2 Initial Laboratory Testing Results of Activated Carbon (CH_3I)

System Designation	Testing Condition T(℃)	RH(%)	Required Efficiency (%)	Tested Efficiency (%)
K5—2A	80	70	>99	99.7
K5—2B	80	70	>99	99.4
P2—1A	80	70	>99	99.7
P2—1C	80	70	>99	99.7
P3—2A	80	70	>99	99.7
P3—2B	80	70	>99	99.9
P1—2	30	93—95	>97	99.4
P2—2A	30	93—95	>97	99.0
P2—2C	30	93—95	>97	99.5
P1—1A	30	93—95	>97	99.4
X1—6A	30	93—95	>97	99.6
X1—6B	30	93—95	>97	99.2

Table 3 Test Report of Leakage Rate of the Charcoal Adsorbers and HEPA Filters after 18 Months

System Designation	Leakage Rate for Charcoal Adsorber (%)	Leakage Rate for HEPA Filter (%)
P1—1A	<0.05	0.018
P1—1B,C		B: 0.045 C: 0.050
P1—2	<0.05	0.032
P2—1A	<0.05	0.0060
P2—1B		>0.050
P2—1C	<0.05	0.015
P2—1D		0.024
P2—2A	<0.05	0.035
P2—2B		0.049
P2—2C	<0.05	0.030
P2—2D		0.050
P2—3A		0.021
P2—3B		0.014
P2—3C		0.011
P2—3D		0.010
P3—1A		0.015
P3—1B		0.033
P3—2A	<0.05	<0.010
P3—2B	<0.05	0.00080
P22—1A		0.023
P22—1B		0.011
K5—2A(Bef.)	<0.05	0.018
K5—2A(Aft.)		0.005
K5—2B(Bef.)	<0.05	0.007

Table 4 Laboratory Testing Results of Activated Carbon (CH_3I) after 18 Months

System Designation	Testing Condition T(℃)	RH(%)	Tested Efficiency %, CH_3I
K5—2A	80	70	99.2
K5—2B	80	70	98.2
P2—1A	80	70	96.6
P2—1C	80	70	98.2
P3—2A	80	70	98.3
P3—2B	80	70	98.6
P1—2	30	93	99.9
P2—2A	30	93	99.6
P2—2C	30	93	99.9
P1—1A	30	93	98.9
X1—6A	30	93	99.6
X1—6B	30	93	99.1

Ⅵ. References

1. ANSI/ANS 56.6 Pressurized Water Reactor Containment Ventilation Systems
2. ANSI/ANS 59.2 Safety Criteria for HVAC Systems Located Outside Primary Containment
3. ANSI/ASME N509 Nuclear Power Plant Air Cleaning units and Components
4. ANSI/ASME N510 Testing of Nuclear Air Treatment Systems
5. ERDA 76—21 Nuclear Air Cleaning Handbook
6. ANSI/ASME AG—1 Code on Nuclear Air and Gas Treatment
7. Regulatory Guides 1.52 Design, Testing, and Maintenance Criteria for Engineered Safety Feature Atmospheric Cleanup System Air Filtration and Adsorption units of LWR Nuclear Power Plants.
8. Regulatory Guides 1.78 Assumptions for Evaluating the Habitability of Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release.
9. Regulatory Guides 1.95 Protection of Nuclear Power Plant Control Room Operators Against Accident Chlorine Release
10. Regulatory Guides 1.140 Design, Testing, and Maintenance Criteria for Normal Ventilation Exhanst System Air Filtration and Adsorption units of LWR Nuclear Power Plant.

DISCUSSION

PASCHAL: Based on the difficulty that you encountered in leak testing concrete filter housings, would you consider them to be suitable for future plants in China?

CHEN: In the future we will change the design, maybe use steel.

PORCO: You said that you sealed the concrete with an epoxy. Do you have concerns about radiation, seismic loads or aging on the epoxy sealant over time?

CHEN: Yes.

23rd DOE/NRC NUCLEAR AIR CLEANING AND TREATMENT CONFERENCE

AP600 CONTAINMENT PURGE RADIOLOGICAL ANALYSIS

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Abstract

The AP600 Project is a passive pressurized water reactor power plant which is part of the Design Certification and First-of-a-Kind Engineering effort under the Advanced Light Water Reactor program. Included in this process is the design of the containment air filtration system which will be the subject of this paper. We will compare the practice used by previous plants with the AP600 approach to meet the goals of industry standards in sizing the containment air filtration system. The radiological aspects of design are of primary significance and will be the focus of this paper.

The AP600 Project optimized the design to combine the functions of the high volumetric flow rate, low volumetric flow rate, and containment cleanup and other filtration systems into one multi-functional system. This achieves a more simplified, standardized, and lower cost design.

Studies were performed to determine the possible concentrations of radioactive material in the containment atmosphere and the effectiveness of the purge system to keep concentrations within 10CFR20 limits and within offsite dose objectives. The concentrations were determined for various reactor coolant system leakage rates and containment purge modes of operation. The resultant concentrations were used to determine the containment accessibility during various stages of normal plant operation including refueling.

The results of the parametric studies indicate that a dual train purge system with a capacity of 4,000 cfm per train is more than adequate to control the airborne radioactivity levels inside containment during normal plant operation and refueling, and satisfies the goals of ANSI/ANS-56.6-1986 and limits the amount of radioactive material released to the environment per ANSI/ANS 59.2 -1985 to provide a safe environment for plant personnel and offsite residents.

1. Introduction

1.1 ALWR Program

The Advanced Light Water Reactor Program was initiated to develop a new generation of nuclear power plants which provide a standardized design of PWR's and BWR's of both evolutionary and passive design features. One of the plant designs being developed in the ALWR program is the AP600 which is a 600 MWe PWR with passive safety features.

The advanced passive design uses simple plant systems and equipment, which minimizes operations, inspections and maintenance requirements by greatly reducing the quantity of valves, pumps, piping, HVAC fans, filtration units and other components. The design uses experience-based power generation components, to offer a high level of reliability, and it provides a high degree of public safety and licensing certainty. Construction schedules will be shortened through modularization. The AP600 passive plant will offer affordable electric power at a competitive cost.

1.2 ALWR Requirements Document

One of the primary goals of the program is to provide a document to the nuclear industry which defines a set of technical design requirements for the advanced reactor power plant designs that will enable the industry to develop an improved and standardized design for future plants. This document is called the Utility Requirements Document (URD). The URD is based on a number of key principles such as: simplification, use of proven technology, maintainability and constructability. Minimizing the number of components while adhering to proven technology reduces the space requirements and the overall plant volume, overall plant capital cost and eventual operations and maintenance costs.

This paper will outline the process of developing a simplified HVAC system design which will accommodate the majority of filtration functions required for the Radiologically Controlled Areas (RCA) of the plant. The focus of this paper will be on the development of the Containment Air Filtration System design to satisfy the RCA filtration functional requirements defined in the URD .

2. RCA HVAC requirements

The URD defines specific HVAC system performance requirements for certain areas of the RCA based on current industry practice and utility experience as follows:

2.1 Auxiliary Building Filtered Exhaust System

The URD requires the Auxiliary Building exhaust air subsystem to be capable of automatic transfer from its normal unfiltered mode of operation to a filtered mode when high radiation in the normal exhaust duct is detected. The filtered exhaust system shall provide HEPA filtration of the exhaust air to maintain the area at a negative pressure relative to the surrounding areas and the outside atmosphere. The AP600 radiologically controlled areas within the Auxiliary Building and Annex II Building have separate radiation monitoring and isolation capability for the normal exhaust paths to the plant vent. Considering the expected inleakage to the AP600 Auxiliary Building and/or Annex II Building when isolated, the required capacity of a filtered exhaust system would be 1,800 cfm.

2.2 Fuel Handling Building Filtered Exhaust System

The URD requires that upon detection of high radiation in the normally unfiltered exhaust path from the Fuel Handling Building. the normal supply and exhaust path shall be automatically isolated and a 100% redundant filtered exhaust system shall be automatically initiated to provide HEPA filtration of the exhaust air and shall maintain the area at a negative pressure relative to the surrounding areas and the outside atmosphere. The AP600 includes a Fuel Handling Area of the Auxiliary Building which has a separate exhaust path to the plant vent. Considering the expected inleakage to the AP600 Fuel Handling Area when isolated, the required capacity of the filtered exhaust system would be 500 cfm.

2.3 Containment Low Volume Purge System

The URD requires that a Containment Low Volume Purge System be provided to reduce airborne radioactivity during normal power operation. The system would consist of a single supply air handling unit and single exhaust HEPA filtration unit and fan. The URD refers to ANSI 56.6 for design guidance which recommends that the capacity of the system be equal to one volume air change every four hours. Based on the AP600 containment volume, the capacity for such a system would be 730 cfm.

The NRC Standard Review Plan 6.2.4, Branch Technical Position CSB 6-4 states that the low volume purge system can be used during normal operation for containment pressure and radiological control. The maximum size of the containment isolation valves for this system should be limited to 8 inches or an accident analysis shall be performed to verify that use of larger valves does not result in accident dose limits being exceeded. Further, the valve closure time shall be 5 seconds or less.

2.4 Containment High Volume Purge System

The URD requires that a Containment High Volume Purge System be provided to provide tempered outside air to containment for a suitable environment before and during personnel entry for cold shutdown/refueling activities. The URD refers to ANSI 56.6 design guidance that states the system should provide control of containment airborne radioactivity, control temperature in conjunction with the containment fan coolers and reduce airborne concentrations of noxious gases. The standard further recommends that the capacity of the system be between one to one and one half volume air changes per hour. Based on the AP600 containment volume, the capacity for such a system would be between 30,000 to 45,000 cfm.

The NRC Standard Review Plan 6.2.4, Branch Technical Position CSB 6-4 restricts the use of a high volume purge system until cold shutdown is reached. This is due to the inherent large size containment isolation valves (42 to 48 inches) and the potential adverse impact on accident dose analysis if the valves were open during normal operation and a design basis accident were to occur.

2.5 Containment Cleanup System

The URD requires that a Containment Cleanup System containing charcoal and HEPA filtration be provided inside containment, to minimize airborne radioactivity concentrations to permit personnel access. The system would consist of a single filtration unit which would recirculate the air inside containment. The URD refers to ANSI/ANS 56.6 for design guidance which states that the function of the Containment Cleanup System is to reduce particulates and radioiodines to facilitate personnel access and to minimize doses to personnel entering containment. The ANSI standard further states that the Containment Cleanup System may operate in conjunction with the Low Volume Containment Purge System. The URD further states that the system designer shall perform a radiological analysis to establish the functional requirements of the system.

The NRC Standard Review Plan 6.2.4, Branch Technical Position CSB 6-4 states that a containment cleanup system should be provided to minimize the need to purge the containment atmosphere.

3. Optimization Evaluations

3.1 Simplified Multifunctional System

Based on the URD and Regulatory requirements the AP600 project performed an evaluation to optimize the systems described above into one simple multifunctional system. Figure 1 illustrates the proposed design of the Containment Air Filtration System (VFS) which contains two 100% redundant 4000 cfm supply and filtered exhaust trains connected to containment through common 18 inch isolation valves and penetrations. Each supply air handling unit contains a low efficiency filter, a high efficiency filter, heating coil, cooling coil and fan. Each filtered exhaust train contains a demister, electric heater,

high efficiency filter, HEPA, charcoal adsorber, high efficiency filter and exhaust fan. Each filtered exhaust train has the capability of being aligned to containment for purging or to either one or all of the Radiologically Controlled Areas. The containment penetration for the filtered exhaust trains will also be used for pressurizing and depressurizing containment during an Integrated Leak Rate Test (ILRT).

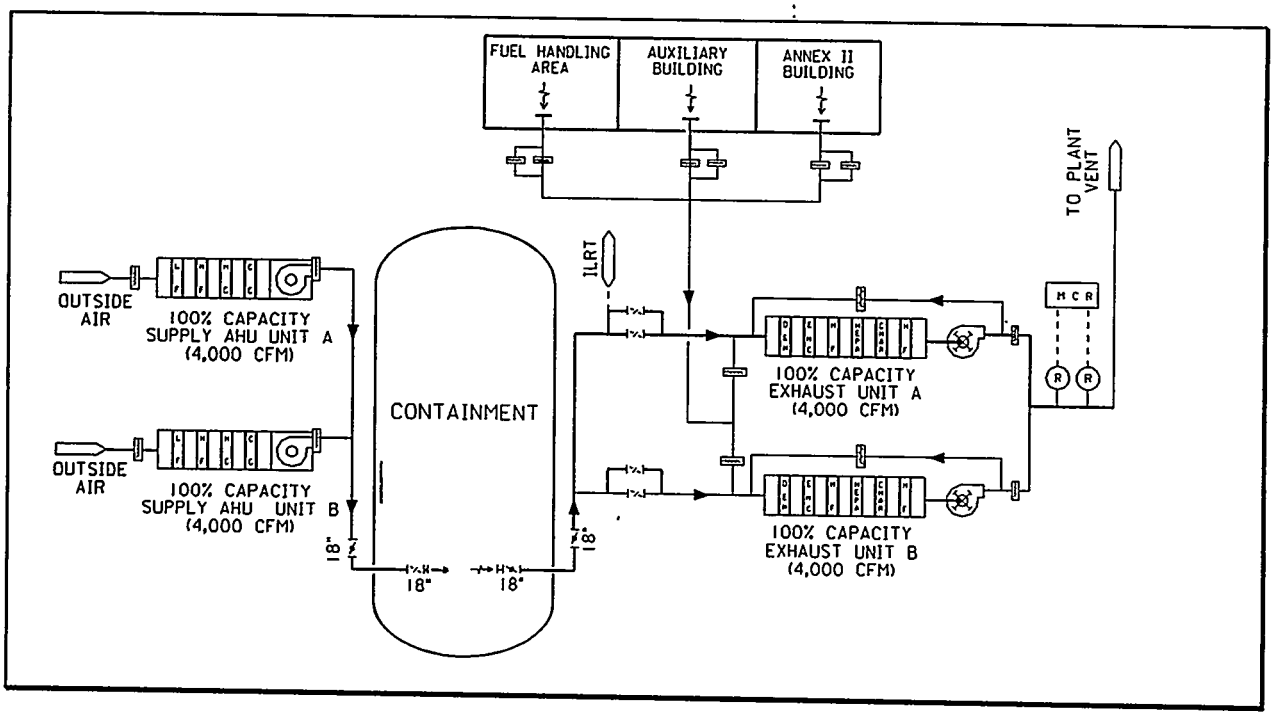


Figure 1 - AP600 Containment Air Filtration System (VFS)

The following assessment addresses each of the individual system requirements described in Section 2 to determine whether all of the requirements have been satisfied:

3.1.1 Auxiliary Building Filtered Exhaust System

The Auxiliary Building and the Annex II Building contain radiologically controlled areas which would be covered by the URD requirement. The normal ventilation system for the Auxiliary Building and Annex II Building is the Radiologically Controlled Areas Ventilation System (VAS) as shown in Figure 2. The exhaust to the plant vent is normally unfiltered but continuously monitored for high radioactivity. Interconnections and controls exist such that if high radioactivity is detected in one or both of the normal unfiltered exhaust paths for the Auxiliary Building HVAC Subsystem, the normal supply and exhaust paths are isolated and the exhaust flow is redirected to one of the VFS filtered exhaust trains. The varying exhaust air flows for this operating mode are controlled by means of a recirculation duct around the VFS filter unit.

If the Containment Air Filtration System were initially aligned to containment for purging, the system would be automatically realigned to the Auxiliary Building and/or Annex Building areas in the filtered exhaust mode. Once this alignment has been completed, the remaining VFS filtered exhaust train can be aligned to containment to continue the purge mode.

This feature of VFS adequately satisfies the URD requirements for a single failure proof, filtered exhaust system which will maintain a negative pressure in the affected areas of high radioactivity.

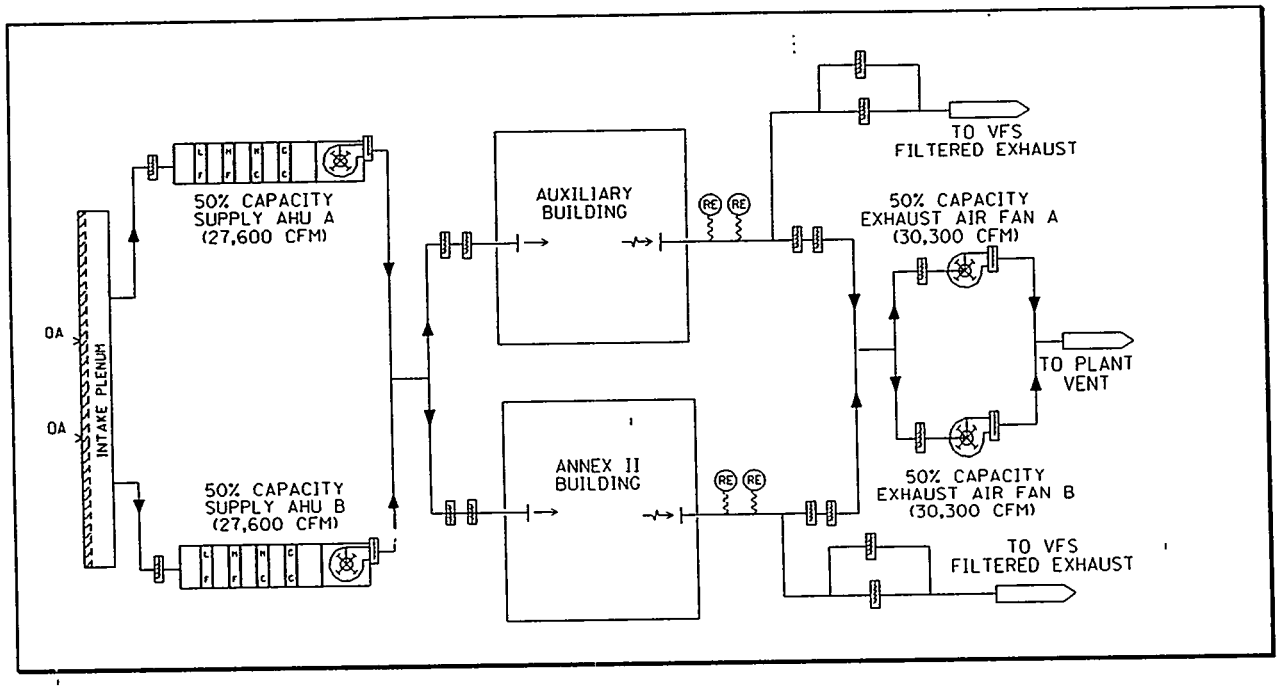


Figure 2 - Auxiliary Building HVAC Subsystem

3.1.2 Fuel Handling Area Filtered Exhaust System

The Fuel Handling Area HVAC Subsystem of the Radiologically Controlled Area Ventilation System (VAS) normally serves the Fuel Handling Area as shown in Figure 3. The exhaust to the plant vent is normally unfiltered but continuously monitored for high radioactivity. Interconnections and controls exist such that if high radioactivity is detected in the normal unfiltered exhaust path for the Fuel Handling Area, the normal supply and exhaust path is isolated and the exhaust flow is redirected to one of the VFS filtered exhaust trains.

This feature of VFS adequately satisfies the URD requirements for a single failure proof, filtered exhaust system which will maintain a negative pressure in the affected areas of high radioactivity.

Each VFS exhaust train is capable of providing exhaust flow and maintaining negative pressure in all three radiologically controlled areas simultaneously if the need arises.

3.1.3 Containment Low Volume Purge System

Each VFS supply and exhaust train provides more than sufficient flow capacity to satisfy the recommendations of ANSI 56.6 for the Containment Low Volume Purge System. A LOCA dose analysis has been performed to verify that if the 18 inch containment isolation valves were open during normal operation and a LOCA were to occur, the valves would close within 5 seconds and the resultant accident doses would be within acceptable limits in accordance with BTP CSB 6-4.

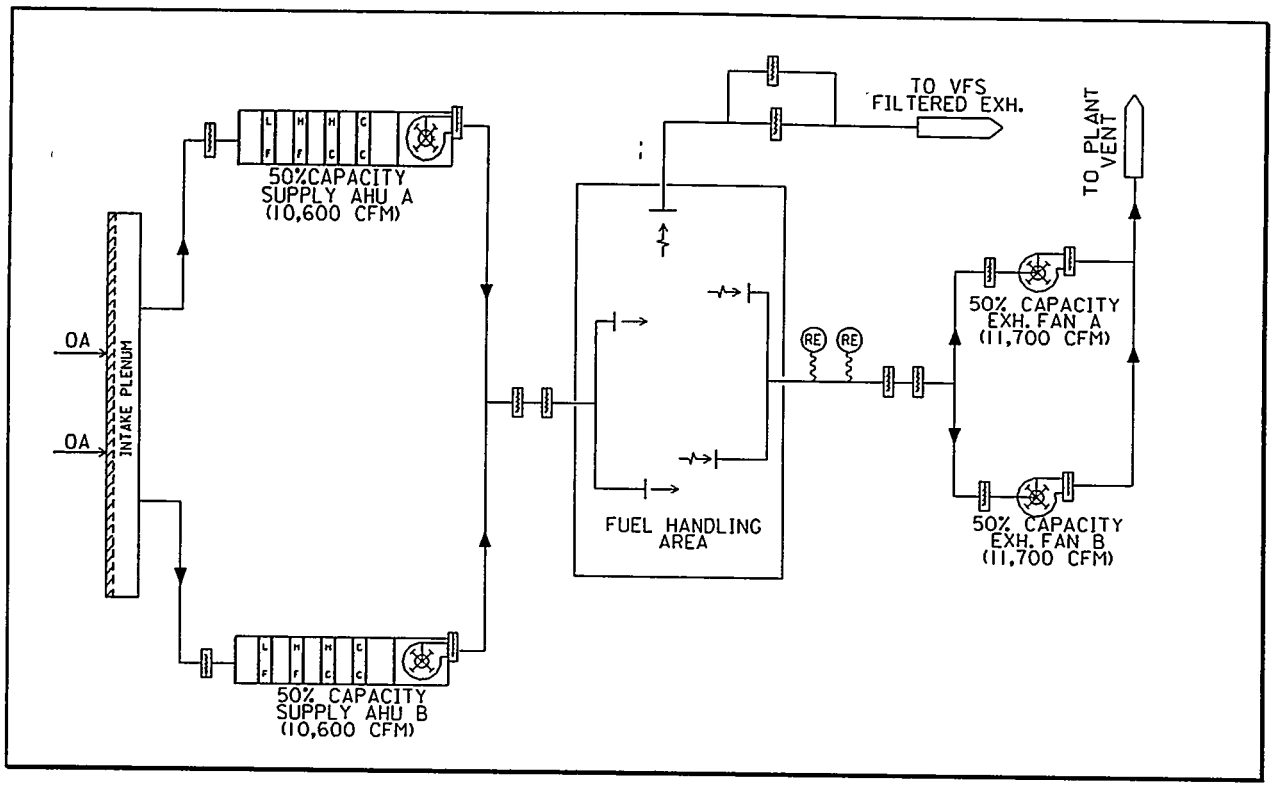


Figure 3 - Fuel Handling Area Subsystem

3.1.4 Containment High Volume Purge System

In this mode of operation the VFS has the capability to operate one supply and exhaust train at 4000 cfm or to operate both trains in parallel for a total capacity of 8000 cfm, as necessary. This total capacity is purposely much less than that recommended by ANSI 56.6 thus reducing the system space requirements, reducing maintenance problems seen historically on the large containment isolation valves, and eliminating a costly dedicated system that would only be used during shutdown.

Surveys of existing plants indicated that the sizing basis for high volume purge systems was not uniform throughout the industry. System capacities varied from as high as 50,000 cfm (TMI-1) to as low as 15,000 cfm (Vogtle).

Each of the functional requirements outlined in ANSI 56.6 will be addressed separately below relative to the adequacy of the AP600 system.

3.1.4.1 Control Airborne Radioactivity

During the shutdown period when the Containment High Volume Purge System is in operation, the sources of airborne radioactivity are primarily RCS leakage and noble gas releases when the reactor vessel head is removed. The primary concerns are the adverse impact on the refueling schedule if personnel were hindered from accessing containment or were required to wear protective clothing. Section 4 of this paper provides the details of parametric radiological analyses which address this issue.

On existing plants that have containment hatches that open directly to the environment (Vogtle, Callaway, Wolf Creek, etc.), there are further considerations that the purge exhaust air flow be sufficient to minimize unmonitored releases when the containment hatches are open. This condition

does not exist on AP600 due to the fact that the containment hatches open to the Auxiliary Building which is served by the Radiologically Controlled Area HVAC System that discharges to the monitored plant vent.

3.1.4.2 Control Containment Temperature

The URD temperature control requirement stems from the fact that existing plants primarily use the Containment High Volume Purge System for cooling to supplement the containment coolers which are served by component cooling water or plant service water at high inlet temperatures depending upon environmental conditions. This condition does not exist on AP600 due to the fact that the Containment Recirculation Cooling System (VCS) is served by the plant Chilled Water System (VWS) with a supply temperature of 44 °F. The VCS is designed to maintain 70 °F at the operating deck and other areas regularly accessible during maintenance and refueling operations without need for support from the VFS. The VFS is also designed to provide 50–70 °F supply air during shutdown conditions using chilled water cooling coils such that the operation of VCS is not adversely affected.

3.1.4.3 Reduce Airborne Concentrations of Noxious Gases

The VFS can support the cleanup of fumes from welding or painting activities in containment. However, welding and painting fumes should be controlled at the source, whenever possible, to prevent these fumes from entering the breathing zone of workers and to prevent deposition on plant equipment and building surfaces. Vapor barriers and special portable fans with filters are commercially available to control filtration of welding and painting fumes.

3.1.5 Containment Cleanup System

ANSI 56.6 states that the function of the Containment Cleanup System is to reduce particulates and radioiodines to facilitate personnel access and to minimize doses to personnel entering containment. The VFS design incorporates a charcoal adsorder bed in the filtration unit to perform this function and eliminate the necessity for a separate Containment Cleanup System. Section 4 of this paper provides the details of parametric analyses that address the performance of a Containment Cleanup System.

Surveys of existing plants that have Containment Cleanup Systems (Kidney System) indicated that the system is rarely if ever used and that outage productivity is not enhanced by use of the system.

3.2 Initial Optimization Conclusions

It can be concluded from the above discussions that the functions of a filtered exhaust system for the RCA portions of the Auxiliary Building, Annex II Building and Fuel Handling Area as well as a Containment Low Volume Purge System can be accomplished by the AP600 Containment Air Filtration System design. From the above discussions it remains to be demonstrated that a Containment High Volume Purge System and a Containment Cleanup System are not needed for strictly radiological considerations and that the VFS can adequately perform the functions of these systems. The following parametric radiological analyses will provide that justification.

4. Radiological Analysis Parametrics

A study was performed to determine the concentration of radioactive material in the containment atmosphere and the effectiveness of the VFS to keep concentrations low. The

concentrations were determined for various Reactor Coolant System (RCS) leakage rates and varied containment purge airflow rates. The efficiency of a Containment Cleanup System ("kidney" system), was also investigated. The resultant airborne concentrations were used to determine the duration of containment accessibility during the various stages of normal plant operation including refueling operations. The new 10CFR20 rule was applied in assessing personnel doses.

4.1 Methodology

The study was performed in several steps as follows:

1. The specific activity of RCS leakage within containment was combined with the production rates of various isotopes due to activation to establish an overall release rate spectrum. The RCS leakage rate was varied from a nromally expected low leakage rate beginning at 0.001 gpm to abnormally high leakage rates up to 10 gpm.

2. The effects on the concentrations due to the use of the 4,000 cfm purge on a weekly basis was determined. Various cycles of purge, no purge conditions were considered to determine the equilibrium concentrations. Various post shutdown purge rates were considered, including the design 8,000 cfm and high volume purge rates of 15,000 cfm, 30,000 cfm, and 45,000 cfm.

3. The effects on the concentration due to the use of a "Kidney" System were investigated. Comparisons of concentrations and dose rates with and without this "Kidney" System were performed.

4. Total effective dose equivalent (TEDE), skin and thyroid dose rates inside containment were calculated along with 10 CFR 20 DAC fractions.

4.1.1 Sequence of Events

For all of the cases considered, the sequence of events considered is as follows:

- Five one-week cycles consisting of 148 hours of closed containment alternating with 20 hours of 4,000 cfm purge operation.

- Sixty hours of continuous 4,000 cfm of low volume purge operation prior to reaching reactor shutdown.

- The beginning of the post shutdown high volume purge was varied to include cold shutdown, hot shutdown and 60 hours prior to cold shutdown.

- Coolant temperature decreased to 120° F prior to RPV head removal.

- Reactor vessel head removed 100 hours after cold shutdown.

The time scale is shown below:

Table 1 - Containment Purge Time History

Time Scale (hrs)	Event
-36	Beginning of pre-shutdown low volume purge operation
0	Hot shutdown
24	Cold shutdown
24	Beginning of post-shutdown high volume purge operation
124	RPV head removal

4.1.2 Parametric Assumptions

1. Reactor coolant concentrations are based on 0.025% failed fuel fraction (10% of Table 11.1-2, reference 1).

2. RCS leakage rate is constant until cold shutdown. From cold shutdown to RPV head removal, the leakage is linearly reduced from 100% to 1% of the full power level.

3. A flashing fraction of 40% is applied to the RCS leakage until hot shutdown. From hot shutdown to cold shutdown, the flashing fraction decreases with decreasing RCS temperature.

4. A decontamination factor of 100 for iodines and 1,000 for particulates is applied to the leakage after cold shutdown.

5. The AP600 reactor coolant degassification process reduces the activity accumulated in the RPV air space to 10% of the activity expected without degassification.

6. Plateout removal is considered throughout the analysis. A removal rate of 1 hr^{-1} for both iodines and particulates is assumed.

7. Dose rate limits are based on 10CFR20 yearly limits of 5 Rem whole body and 50 Rem to the skin or thyroid. Using a work year of 2,000 hours, the dose rate limits are 2.5 mRem/hour whole body, and 25 mRem/hour skin and thyroid.

8. Offsite thyroid inhalation doses and whole body cloud immersion doses are calculated assuming a χ/Q of 1.65×10^{-5} sec/m^3 for all cases considered.

4.1.3 Coolant Concentrations

The reactor coolant concentrations used in this analysis are typical of current failed fuel experience and are equivalent to a failed fuel level of 0.025%. These concentrations were derived from the design basis values provided in reference 1.

4.1.4 Containment Concentration Calculation

The containment airborne radioactivity levels are a function of fuel failure, reactor coolant leakage, power level, isotopic decay, and removal mechanisms present in the containment. This complex function is best simplified as a linear differential equation of the form:

$$\frac{dA^i(t)}{dt} = R^i - \lambda_r^i A^i(t) \quad (1)$$

where:

$A^i(t)$ is the containment activity in curies for isotope i,

R^i is the release rate from the coolant in curies/sec for isotope i,

λ_r^i is the removal rate (per second) from the containment for isotope i. This removal rate consists of decay, plateout, purge and "kidney" system filter removal. This is estimated as:

$$\lambda_r^i = \lambda_d^i + \lambda_p^i + \frac{F_p}{V_c} + \frac{F_k \varepsilon^i}{V_c} \quad (2)$$

λ_d^i is the decay constant for isotope i, sec^{-1},

λ_p^i is the plateout removal rate for isotope i, sec^{-1},

F_p is the purge exhaust rate, ft^3/sec,

V_c is the containment volume, ft^3,

F_k is the "kidney" system flow rate, ft^3/sec,

ε^i is the "kidney" system filter efficiency for isotope i.

Solving the differential equation for the concentration $C^i(t)$ between any time period t_1 to t_2, we obtain:

$$C^i(t_2) = \frac{R^i}{\lambda_r^i V_c}\left(1 - e^{-\lambda_r^i (t_2 - t_1)}\right) + C^i(t_1) e^{-\lambda_r^i (t_2 - t_1)} \quad (3)$$

The above expression was used to determine the concentrations in the containment at various times with various purge cycles. The release rate from the coolant was varied depending on the state of the plant. Different release rates were used for normal operation, hot shutdown and cold shutdown. These release rates are discussed in the following sections.

4.1.5 Release Rates into Containment

Radioactive release rates into the containment are a function of coolant leakage, coolant concentration, and operating state of the plant. Therefore, three different release rates are calculated, a normal operating release rate, hot shutdown release rate, and cold shutdown release rate.

4.1.5.1 Normal Operating Release Rate

During normal operation, the release of radioactivity into the containment atmosphere can be calculated using the following relationship:

$$R_n^i = l_c \cdot f_f \cdot C_c^i + A_n^i \quad (4)$$

where:

R_n^i is the release rate of isotope i into the containment atmosphere in μCi/sec,

l_c is the coolant leakage rate in cm^3/sec,

f_f is the flashing fraction. This parameter is 1 for noble gases and 0.4 for iodines and particulates.

C_c^i is the reactor coolant concentration for isotope i in μCi/cm^3, and

A_n^i is the neutron activation product for isotope i in μCi/sec.

4.1.5.2 Hot Shutdown Release Rate

Conservatively, the release rate during hot shutdown is assumed to be due to the same coolant leakage rate as during normal operation. Since the nuclear reaction has been terminated, the neutron activation term is no longer available. Although the reactor coolant clean-up system continually reduces the coolant concentrations, this effect is conservatively neglected. Only decay after shutdown is considered to reduce the release rate. Therefore, the release rate at any time t after hot shutdown can be characterized as follows:

$$R_{hs}^i = l_c \cdot f_f \cdot C_c^i e^{-\lambda^i (t-t_s)} \tag{5}$$

where t_s is the time that hot shutdown began.

4.1.5.3 Cold Shutdown Release Rate

After cold shutdown, the reactor coolant temperature is below boiling. Therefore no flashing of coolant will occur. The release rate of iodines and particulates to the containment atmosphere is conservatively estimated using the following expression:

$$R_{cs}^i = \frac{l_c C_c^i e^{-\lambda^i (t-t_s)}}{PF} \tag{6}$$

PF is the partition factor between water and air for each isotope species.

Noble gases are assumed to be stripped from the reactor coolant by the degassification process, therefore the levels of noble gases will be quickly reduced. After achieving cold shutdown, the release rate of noble gases from the fuel is approximately 100,000 lower than the release during normal operation. Therefore, the release rate of noble gases from the reactor coolant after cold shutdown is conservatively estimated to be:

$$R_{cs}^i = \frac{l_c C_c^i e^{-\lambda^i (t-t_s)}}{100,000} \tag{7}$$

4.1.5.4 Accumulation of Activity in the RPV Air Volume

Prior to removal of the RPV head, the coolant cleanup system operates in conjunction with the degassification process. It is expected that these processes will reduce the level of activity in the coolant

to levels that are low enough to pose no problem when the head is removed. To conservatively bound the activity that can be released, it is assumed that after shutdown a gas space in the RPV accumulates activity. This activity is then released to the containment after removal of the RPV head.

4.1.5.5 Release Rates from Open RPV

Once the RPV head is removed, the releases are due to evaporation from the reactor coolant through the open pool of water above the reactor vessel.

4.1.6 Shut-down Iodine Spike

When a pressurized water reactor containing one or more failed fuel rods has been operated for a time and the power is then decreased, the activity of certain isotopes often increases. This phenomenon is known as iodine spiking. An empirical model is used to determine the magnitude and duration of the spike.

The basic equation governing the reactor coolant radioactive material inventory is given by:

$$\frac{dA^i}{dt} = P^i - \lambda_d^i A^i - \lambda_L A^i \qquad (8)$$

where:

P^i is the rate of introduction into the coolant for isotope i,

λ_L is the CVCS letdown removal rate, given by:

$$\lambda_L = \frac{F_L}{V_{RCS}}\left(1 - \frac{1}{DF_L}\right) \qquad (9)$$

where:

F_L is the CVCS letdown flow rate,

V_{RCS} is the total reactor coolant system volume, and

DF_L is the CVCS letdown demineralizer removal efficiency.

At equilibrium conditions (e.g. steady-state power), equation 8 reduces to:

$$A^i = \frac{P^i}{\lambda_d^i + \lambda_L} \qquad (10)$$

Given that at equilibrium conditions, the RCS coolant concentration is the activity divided by the RCS volume, and defining $\lambda_R^i = \lambda_d^i + \lambda_L$, the production term can then be estimated to be:

$$P^i = V_{RCS}\lambda_R^i C_c^i \qquad (11)$$

Define a spiking factor S_f, the production term during spiking is then given by:

$$P^i = S_f V_{RCS}\lambda_R^i C_c^i \qquad (12)$$

Using the definitions presented above, and converting activity inventory to coolant concentration, the solution to equation 10 during spiking between any time interval t_1 to t_2 is given by:

$$C^i(t_2) = C_c^i S_f\left(1 - e^{-\lambda_R^i \Delta t}\right) + C^i(t_2)e^{-\lambda_R^i \Delta t} \qquad (13)$$

After spiking, the concentration can be calculated using the following equation:

$$C^i(t_2) = C_c^i\left(1 - e^{-\lambda_R^i \Delta t}\right) + C^i(t_2)e^{-\lambda_R^i \Delta t} \qquad (18)$$

Using equations 13 and 14, the ratio of the total iodine concentration to the non-spiked concentration is calculated and shown on Figure 4 below.

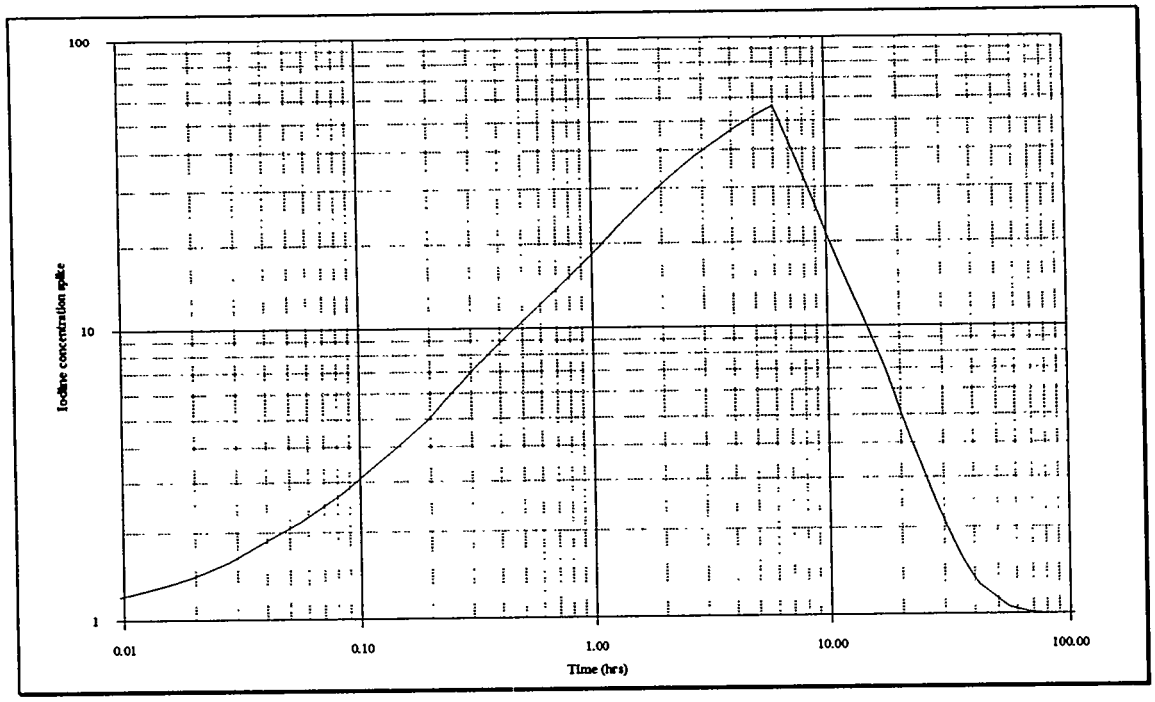


Figure 4 – Iodine Spiking Factor

4.1.7 Flashing Fraction Reduction

From hot shutdown to cold shutdown, the reactor coolant temperature decreases from 420 °F to 200 °F. From references 3 – 5, the temperature history during cooldown is shown on Table 2.

Table 2 – Cooldown Temperature History

	Time	Temp (deg.F)
Hot Shutdown	0	420
Start of Normal Heat Removal System	4	350
Cold Shutdown	24	200
End of Cooldown	100	120
Time to reach boiling	22.4	212

As shown on Table 2, the reactor coolant system is below boiling before reaching cold shutdown, therefore no flashing is expected to occur.

Using the cooldown history of Table 2, the flashing fraction as a function of time can be calculated from hot shutdown to cold shutdown. Conservatively, the flashing fraction is not allowed to fall below 10% before cold shutdown. The flashing fraction time history is shown on Figure 5.

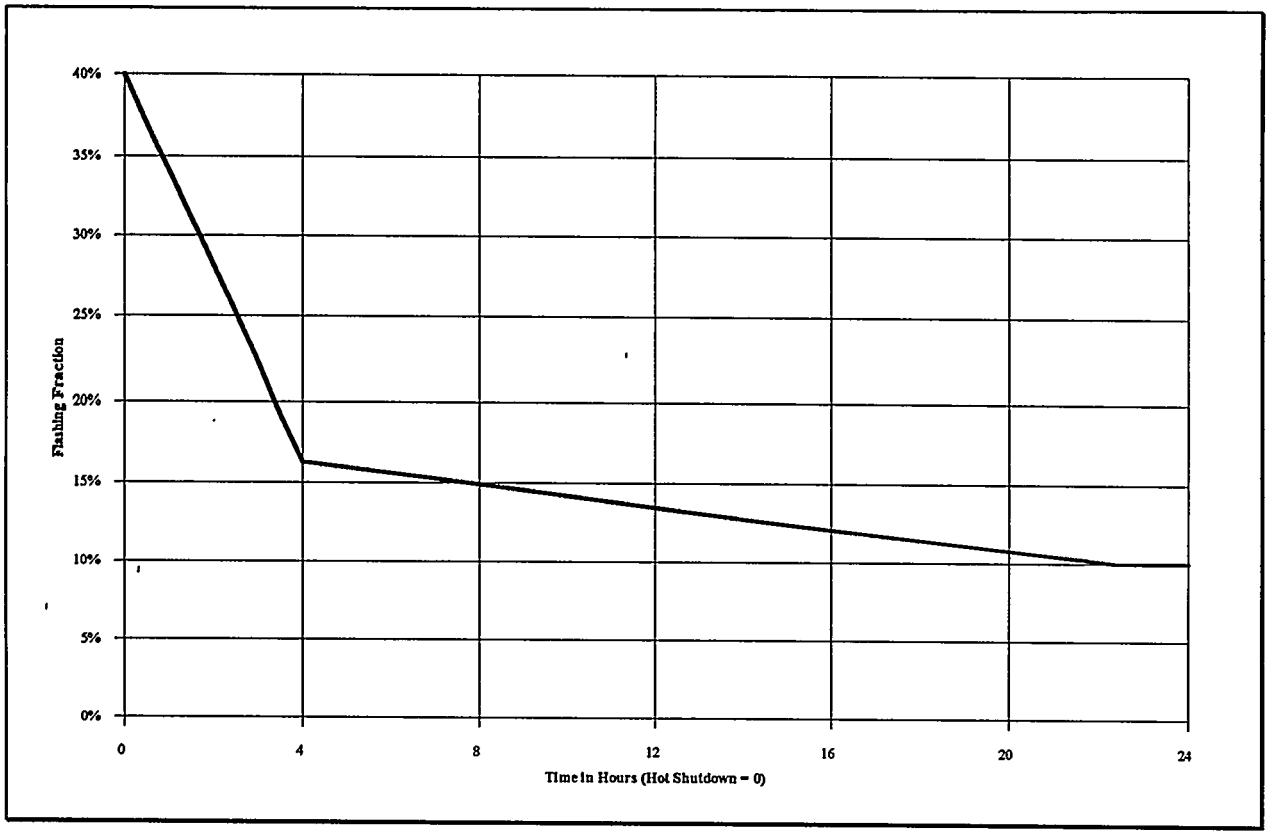


Figure 5 – Flashing Fraction Time History

4.2 Compliance with Revised 10CFR20

In January 1994, a revised 10CFR20 became effective. This revised regulation changes the dose and concentration limits and also changes the methodology used to calculate doses. From 20.1201, the annual limit is the most limiting or the total effective dose equivalent (TEDE) being equal to 5 rem; or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rem. Also, the shallow-dose equivalent of 50 rem to the skin or to any extremity.

The total effective dose equivalent is defined as the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). The committed effective dose equivalent (CEDE) is the sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues. The committed dose equivalent is the dose equivalent to organs or tissues that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.

To calculate these doses, the methodology of ICRP30 is used. Dose conversion factors using this methodology for the thyroid and the CEDE due to inhalation and external dose conversion factors are obtained from reference 6.

In addition to the new dose limits, 10CFR20 replaces the maximum permissible concentration (MPC) concept with derived air concentration (DAC). The derived air concentration is defined as the concentration of a given radionuclide in air which, if breathed by the reference man for a working year of 2,000 hours under conditions of light work, results in an intake of one ALI. The annual limit on intake (ALI) is the amount of radioactive material taken into the body in a year which would result in a CEDE of 5 rem.

4.2.1 Cases Considered

Several parametric cases were analyzed. Purge flow rates, RCS leakage rates and operation of a "kidney" system were varied. These parametrics, using a 0.025% failed fuel coolant source were performed with the following variations:

1. Effects of "kidney" system with initial purge rate of 4,000 cfm and a post-shutdown purge rate of 8,000 cfm.
   a) Continuous operation of a 4,000 cfm "kidney" system
   b) No "kidney" system operation (VFS with charcoal)
   c) RCS leak rate of 0.1, 0.5, and 1 gpm are considered.
2. Effects of containment purge sizing on concentrations:
   a) Containment Purge Volumetric Flow Rate:
      i) 4,000 cfm
      ii) 8,000 cfm
      iii) 15,000 cfm
      iv) 20,000 cfm
      v) 30,000 cfm
      vi) 45,000 cfm
   b) Baseline RCS leak rate of 0.5 gpm, 2 gpm, 5 gpm, and 10 gpm were considered.
   c) RCS leak rate remaining at baseline leak rate until cold shutdown and decreasing linearly to 1% until RPV head removal.
3. Leakage rate parametrics:
   a) Post shutdown purge rate of 8,000 cfm beginning at cold shutdown.
   b) RCS leak rate varied from 0.001 gpm to 10 gpm.

4.2.2 Results

A summary of results is presented here. The parametrics showing the effects of operating a "kidney" system vs. operating the containment air filtration system with charcoal adsorber beds are presented on Table 3. These results illustrate that the "kidney" system is not effective in minimizing the need of purging the containment during normal operation since the radiation levels are dominated by noble gases.

Results of the effects of containment purge sizing for a 0.5 gpm leak are summarized on Table 4, for a 2 gpm leak on Table 5, for a 5 gpm leak on Table 6, and for a 10 gpm leak on Table 7. A summary

of times to reach 0.5 DAC starting from hot shutdown is presented on Figure 6, page 18. A summary of times to reach the TEDE dose rate limit starting from hot shutdown is presented on Figure 7, page 18. The results of the leakage parametric analysis for the 8,000 cfm purge rate are summarized on Figure 8, page 19.

Table 3 - Containment Cleanup System (Kidney) Parametric Summary

	0.1 gpm		0.5 gpm		1 gpm		Difference		
	No Kidney	Kidney	No Kidney	Kidney	No Kidney	Kidney	0.1 gpm	0.5 gpm	1 gpm
Time to Reach Concentration (Hrs) after Hot Shutdown									
1 DAC	14.8	14.4	25.9	25.8	28.6	28.6	2.82%	0.14%	0.19%
0.5 DAC	19.3	19.0	28.6	28.6	31.5	31.4	1.57%	0.19%	0.28%
Time to Reach Dose Rate Limit (Hrs) after Hot Shutdown									
2.5 Whole Body (TEDE)	9.5	9.2	17.2	16.8	20.9	20.7	3.92%	2.09%	0.97%
25 Thyroid	#N/A	#N/A	18.4	18.1	20.4	20.1	#N/A	1.88%	1.25%
Normal Operation									
Whole Body (TEDE)									
Pre-purge Dose Rate (mR/hr)	15.53	15.44	68.38	67.96	134.44	133.60	0.54%	0.61%	0.62%
Post-purge Dose Rate (mR/hr)	3.86	3.79	12.40	12.07	23.08	22.42	1.73%	2.69%	2.89%
Hot Shutdown (mR/hr)	3.01	2.95	8.18	7.85	14.64	13.98	2.21%	4.07%	4.55%
Cold Shutdown (mR/hr)	0.17	0.16	0.84	0.82	1.68	1.64	1.98%	1.98%	1.98%
Thyroid									
Pre-purge Dose Rate (mR/hr)	24.92	23.94	114.37	109.44	226.17	216.31	3.96%	4.31%	4.36%
Post-purge Dose Rate (mR/hr)	11.75	10.97	51.14	47.23	100.37	92.55	6.66%	7.65%	7.79%
Hot Shutdown (mR/hr)	10.87	10.09	46.75	42.84	91.59	83.77	7.19%	8.37%	8.54%
Cold Shutdown (mR/hr)	0.77	0.70	3.87	3.51	7.74	7.02	9.19%	9.19%	9.19%
Prior to RPV Head Removal									
Whole Body Dose Rate (mR/hr)	1.26E-04	1.03E-04	6.28E-04	5.15E-04	1.26E-03	1.03E-03	17.97%	17.97%	17.97%
Thyroid Dose Rate (mR/hr)	3.03E-03	2.47E-03	1.52E-02	1.24E-02	3.03E-02	2.47E-02	18.48%	18.48%	18.48%
After RPV Head Removal									
Whole Body Dose Rate (mR/hr)	4.78E-03	4.05E-03	4.83E-03	4.08E-03	4.89E-03	4.13E-03	15.30%	15.44%	15.61%
Thyroid Dose Rate (mR/hr)	1.18E-01	1.00E-01	1.19E-01	1.01E-01	1.20E-01	1.01E-01	15.09%	15.26%	15.46%
Offsite Doses (mRem)									
Weekly Purging Activities									
Whole Body	1.13E-02	1.13E-02	2.25E-02	2.25E-02	3.65E-02	3.65E-02	0.01%	0.02%	0.03%
Thyroid	1.27E-02	1.27E-02	2.60E-02	2.59E-02	4.26E-02	4.23E-02	0.21%	0.52%	0.63%
Shutdown Operations									
Whole Body	2.07E-02	2.07E-02	3.03E-02	3.02E-02	4.22E-02	4.21E-02	0.05%	0.16%	0.23%
Thyroid	2.60E-02	2.57E-02	4.90E-02	4.75E-02	7.77E-02	7.48E-02	1.13%	2.99%	3.76%

Table 4 - Summary of Results for 0.5 gpm RCS Leak

Purge Flow		Time after Hot Shutdown to Reach:			
(cfm)	Purge Start	1 DAC	0.5 DAC	2.5 mR/hr TEDE	25 mR/hr Thyroid
4,000	Cold Shutdown	27.6	32.8	17.2	18.4
	Hot Shutdown	27.6	32.8	17.2	18.4
8,000	Cold Shutdown	25.9	28.6	17.2	18.4
	Hot Shutdown	21.5	26.0	15.2	18.0
15,000	Cold Shutdown	25.0	26.5	17.2	18.4
	Hot Shutdown	19.2	22.9	14.0	17.3
30,000	Cold Shutdown	24.5	25.3	17.2	18.4
	Hot Shutdown	17.3	20.1	12.6	16.3
45,000	Cold Shutdown	24.4	24.9	17.2	18.4
	Hot Shutdown	16.3	19.0	11.7	15.5

Table 5 - Summary of Results for 2 gpm RCS Leak

Purge Flow (cfm)	Purge Start	Time after Hot Shutdown to Reach: 1 DAC	0.5 DAC	2.5 mR/hr TEDE	25 mR/hr Thyroid
4,000	Cold Shutdown	38.5	44.4	25.8	21.8
	Hot Shutdown	38.5	44.4	25.8	21.8
8,000	Cold Shutdown	31.5	35.1	25.1	21.8
	Hot Shutdown	29.0	32.3	20.9	21.3
15,000	Cold Shutdown	28.1	30.0	24.6	21.8
	Hot Shutdown	25.5	27.6	19.5	20.9
30,000	Cold Shutdown	26.1	27.2	24.3	21.8
	Hot Shutdown	24.1	25.2	18.1	20.2
45,000	Cold Shutdown	25.4	26.0	24.2	21.8
	Hot Shutdown	21.4	24.6	17.2	19.7

Table 6 - Summary of Results for 5 gpm RCS Leak

Purge Flow (cfm)	Purge Start	Time after Hot Shutdown to Reach: 1 DAC	0.5 DAC	2.5 mR/hr TEDE	25 mR/hr Thyroid
4,000	Cold Shutdown	46.6	53.6	33.0	27.5
	Hot Shutdown	46.6	53.6	33.0	27.5
8,000	Cold Shutdown	36.8	41.9	28.7	26.2
	Hot Shutdown	33.5	40.0	25.3	25.3
15,000	Cold Shutdown	30.8	33.7	26.6	25.4
	Hot Shutdown	28.4	31.9	22.2	23.7
30,000	Cold Shutdown	27.5	29.3	25.4	24.9
	Hot Shutdown	25.6	27.9	20.9	21.9
45,000	Cold Shutdown	26.3	27.6	25.0	24.7
	Hot Shutdown	24.9	26.0	20.2	21.5

Table 7 – Summary of Results for 10 gpm RCS Leak

Purge Flow (cfm)	Purge Start	Time after Hot Shutdown to Reach: 1 DAC	0.5 DAC	2.5 mR/hr TEDE	25 mR/hr Thyroid
4,000	Cold Shutdown	53.6	63.6	39.3	35.8
	Hot Shutdown	53.6	63.6	39.3	35.8
8,000	Cold Shutdown	41.9	51.3	31.9	32.4
	Hot Shutdown	40.0	50.5	28.4	31.9
15,000	Cold Shutdown	33.7	44.9	28.4	29.9
	Hot Shutdown	31.9	44.9	25.3	29.6
30,000	Cold Shutdown	29.3	36.9	26.2	27.0
	Hot Shutdown	27.9	36.9	24.0	26.5
45,000	Cold Shutdown	27.6	32.3	25.5	25.8
	Hot Shutdown	26.0	32.3	21.8	24.7

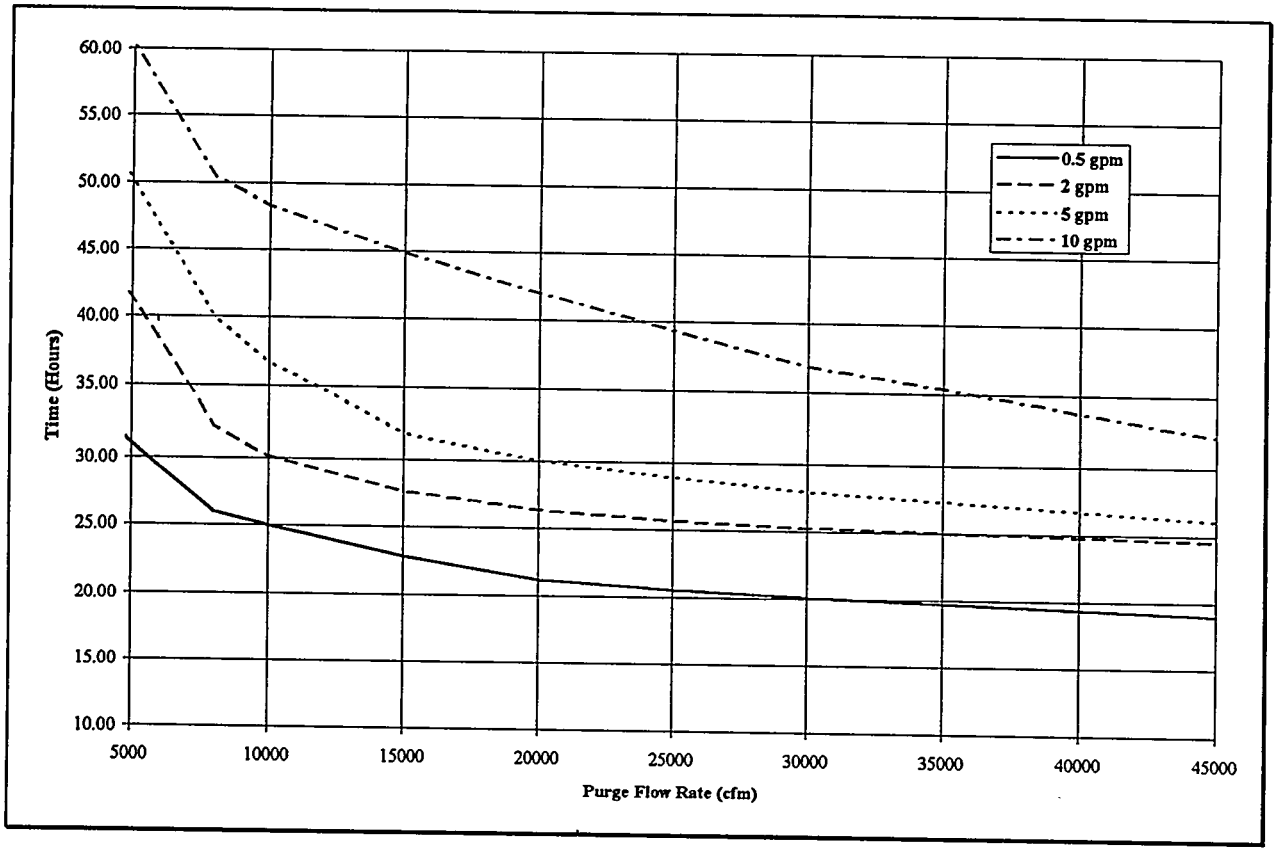


Figure 6 – Time to Reach 0.5 DAC for Various Shut-Down Purge Rates beginning at Hot Shutdown

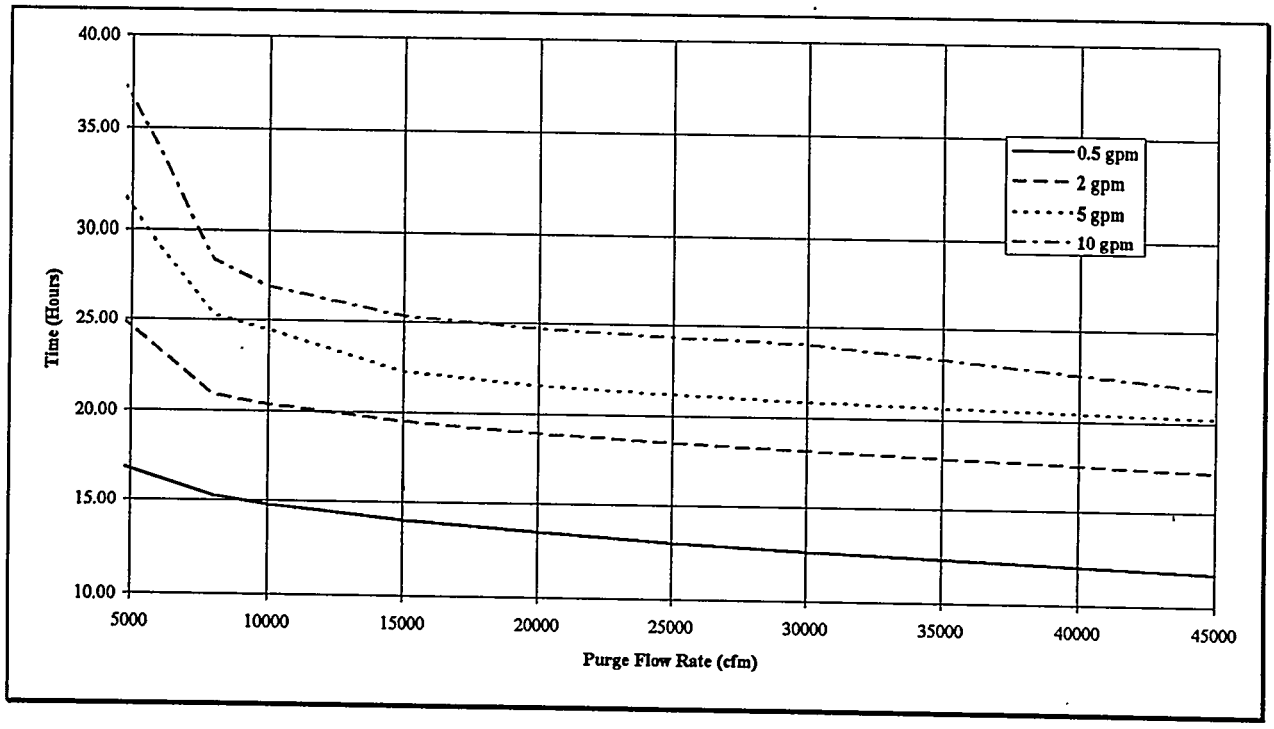


Figure 7 – Time to Reach TEDE Limit for Various Shut-Down Purge Rates beginning at Hot Shutdown

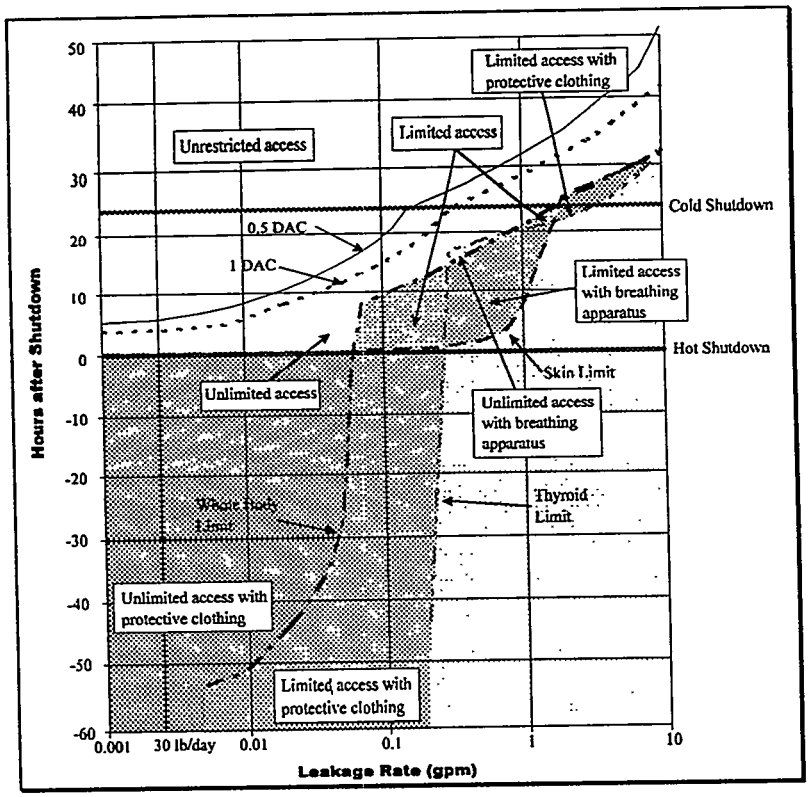


Figure 8 - Time History as a Function of RCS Leak for Post Shutdown Purge of 8,000 cfm

4.2.3 Cost-Benefit Analysis

A cost-benefit analysis is performed to determine if the dose reduction gained by increasing the purge system flow justifies the extra cost of the system. Purge flow rates of 8,000 cfm (the current AP600 design), 15,000 cfm, 30,000 cfm, and 45,000 cfm are analyzed. This analysis is performed as follows:

1. The TEDE dose rates calculated for the cases presented on Table 4 to Table 7 form the basis of the dose calculation. Dose rates as a function of time after cold shutdown for a 0.5 gpm RCS Leakage rate are shown on Figure 9. Conservatively, for times after 230 hours, the dose rate at 230 hours is used

2. The doses are calculated for each time step using a trapezoidal integration method.

3. Integrated doses for the different activities are then calculated by summing the individual doses. The following activities are considered (reference 2):

   a) HP Containment Inspection. This task begins approximately 4 hours after hot shutdown for 5 hours.

   b) Preparatory work prior to general containment access. This task begins immediately after the HP inspection is finished, approximately 9 hours after hot shutdown.

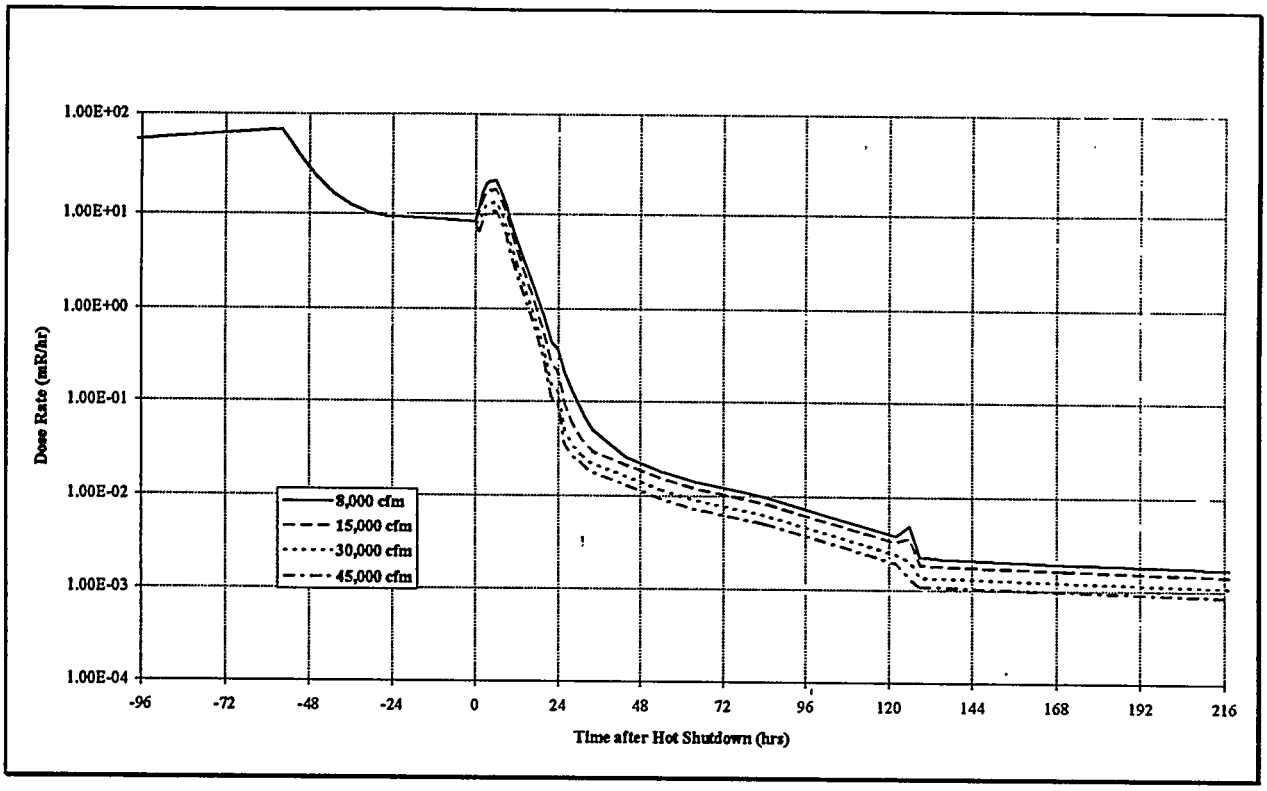


Figure 9 – TEDE Dose Rate as a Function of Time for 0.5 gpm RCS Leakage

c) General Containment Access. Beginning at cold shutdown it assumed that work crews will immediately enter the containment for the duration of the refueling outage.

The largest component of the dose occurs initially, during the HP Containment Inspection. The general access doses are low in comparison.

4. Using the integrated doses calculated above, the total Person-rem is determined by multiplying the dose with the number of workers for each of the tasks.

5. The dose averted by using a larger purge system is compared to the base 8,000 cfm purge, and the resultant costs penalties are a differential of the capital costs of the additional systems and the averted cost savings. The cost of a person-rem averted is assumed to be $10,000. Operation and maintenance (O&M) costs and the time value of money are not included for conservatism.

6. The cost penalty is derived assuming 50% of the refuelings are based on a 0.5 gpm RCS leak and 50% of the refuelings are based on a 2 gpm RCS leak.

The input parameters and assumptions used in the analysis are summarized on Table 8 below.

Table 8 – Cost Benefit Analysis Parameters

Duration of Refueling Outage	17 days
Total number of hours	408 hours
Refueling Outage Work Parameters	
HP Containment Inspection	25 workers
Preparatory Work	15 workers
Work Shift	10 hours
Number of Shifts	3
Crew per Shift	100
Plant Parameters	
Plant Life	60 years
Refueling Cycle	18 months
Number of Refuelings	40

Results of the analysis are summarized on Table 9 as Person-rem averted differential between the current AP600 purge of 8,000 cfm and proposed larger purge flow rates and the cost penalty for each.

Table 9 – Cost Benefit Analysis Results

Cost Parameters			
Shutdown Purge Size:	**15,000**	**30,000**	**45,000**
Capital Cost	$1,879,000	$2,370,000	$2,780,000
Person-Rem Cost	$10,000		
Person-Rem Estimate			
Available Person-Rem	187.9	237.0	278.0
Person-Rem per Refueling	4.7	5.9	7.0
Per Refueling Person-Rem Saved:			
Maximum (@ 10 gpm)	14.3	28.8	36.8
Normal (@ 0.5 gpm)	0.7	1.5	1.9
Normal (@ 2 gpm)	2.9	5.8	7.4
Over Plant Life (40 Refuelings)			
Normal (@ 0.5 gpm)	29.1	58.8	75.1
Normal (@ 2 gpm)	114.6	231.4	295.8
Number of Refuelings at 10 gpm Leak:	11.7	6.5	5.8
Cost Penalty			
50% @ 0.5 gpm, 50% at 2 gpm	($1,160,463)	($918,986)	($925,710)

Using a cost of $10,000 per Person-rem averted, a 15,000 cfm system would have to save approximately 188 Person-rem to be cost effective. During a typical refueling outage, this system would save 0.73 Person-rem over the current design of 8,000 cfm, for a total of 29 Person-rem over the life of the plant. If extreme leakage conditions were to exist (such as RCS leakage of 10 gpm), the 15,000 cfm purge would save 14.3 Person-rem. For this system to be cost effective, approximately 11 refuelings with high RCS leakage would have to occur during the life of the plant, a highly unlikely condition. Similarly, for a 45,000 cfm system, approximately 6 refuelings with high RCS leakage would be required to make the system cost effective.

As shown on Table 9, even in the highly unlikely event that half of the refuelings would be while a 2 gpm RCS leak was present, and the other half at 0.5 gpm RCS leak, none of the high volume purge systems would be cost effective compared to the base 8,000 cfm purge system.

4.3 Radiological Analysis Parametrics Conclusions

Several conclusions can be drawn from the results presented in this analysis:

- Twenty hours of mini purge operation during normal power operation provides essentially unlimited access from a dose limit stand point for design coolant leakage rates up to approximately 0.1 gpm and expected failed fuel.

- A shutdown purge rate of 8,000 cfm is adequate to allow unrestricted access to the containment before reaching cold shutdown for expected coolant leakage rates less than 2 gpm.

- The containment airborne concentration is highly dependent on failed fuel fraction. Current improvements in fuel design and fabrication, fuel handling, reactor coolant chemistry and purity trend towards diminished radioactive releases, thus lessening the need for atmospheric clean-up for containment access.

- RCS leakage rates greater than 2 gpm is a highly unlikely event and should occur very infrequently given improved emphasis on leakage collection techniques and preventive maintenance.

- When the leakage rates are abnormally high, iodines become dominant. In all cases, the main contributor to the increased DAC ratio are the iodines. Since iodines are primarily thyroid dose dependent, use of breathing apparatus would limit the exposure. Since the revised 10CFR20 limits consider contributions from both external and internal sources, access to the containment can be allowed without the need of using breathing apparatus. Using the tech spec limit of 0.5 gpm as an example, the containment can be accessed for up to 18 hours without purging during normal operation. The time increases to 101 hours after 20 hours of purging. With 60 hours of purging, the maximum time is 153 hours. These times would allow workers to enter the containment, find and terminate the leakage.

- If abnormally high leakage rates exist, the impact on offsite doses will necessitate actions to locate and eliminate the sources of leakage during normal operation, if at all possible, to avoid having to shutdown. If shutdown is necessary to correct the leakage situation, the difference in time to reach the dose limit after shutdown initiation is not substantial between an 8,000 cfm and 45,000 cfm purge rate (i.e., about 7 hours for a 10 gpm RCS leakage).

- Given the spectrum of RCS leakages evaluated at the expected failed fuel fraction, the advantage of a larger purge system is only marginal. An 8,000 cfm purge system provides more than adequate radiological control.

- As shown on Table 3, the maximum reduction of the containment dose rate due to operation of a "kidney" system is 18%, long after achieving cold shutdown. During normal operations,

the reduction of dose rate is less than 5%. Therefore, a "kidney" system was not considered necessary and not included in the design of the AP600 Containment Air Filtration System.

- The AP600 Containment Air Filtration System design provides the most cost effective design for containment purge. The capital and O&M expense that would be incurred if larger post-shutdown systems were added is not justified when compared to the small savings in Person-rem that these systems would provide over the life of the plant.

- The shutdown purge has no effect on the refueling outage critical path for a wide spectrum of leakage rates.

5. Conclusions

Based on the radiological analysis presented above, the proposed design of the AP600 Containment Air Filtration System adequately serves the function of a Containment Cleanup System and a Containment High Volume Purge System for strictly radiological considerations. The function of the Containment Cleanup System is served by incorporating charcoal filter in the filtration exhaust units and the function of the Containment High Volume Purge System is served by utilizing both 4,000 cfm trains during conditions of abnormal RCS leakage.

6. References

1. AP600 SSAR Table 11.1-2, "*Design Basis Reactor Coolant Activity,*" Revision 0, 26 June 1992.

2. AP600 Integrated Refueling Outage Typical Schedule.

3. AP600 SSAR Section 5.4.7.1.2.1, "*Shutdown Heat Removal,*" Revision 0, 26 June 1992.

4. AP600 SSAR Section 5.4.7.4.2, "*Plant Cooldown,*" Revision 0, 26 June 1992.

5. AP600 SSAR Technical Specification Table 1.1-1, "*Modes,*" Revision 0, 26 June 1992.

6. Nuclear Engineering Design Guide 3DG N61 002 Rev 0, "*Dose Conversion Factors for Radiation Analyses,*" Bechtel Corporation, 8 December 1993 (Bechtel Prorietary).

7. Telecon from M. L. Kasjaka, Bechtel to Henry Fong, Diablo Canyon, "*Personnel Access Requirements into the Containment,*" 22 December 1993, DCN 005723.

8. Telecon from M. L. Kasjaka, Bechtel to Skip Holman, Wolf Creek, "*Personnel Access Requirements into the Containment,*" 27 December 1993, DCN 005727.

DISCUSSION

GOOSSENS: In the safety assessment of the Belgian Power PWR-stations, a 1% failure of the fuel is presumed for calculating the concentrations of isotopes in the primary coolant. Can you comment why, in your study, the fuel failure is assumed as low as 0.025%?

SCHULZ: Current improvements in fuel design and fabrication, fuel handling, reactor coolant chemistry and purity trend towards diminished fuel failures and radioactive releases. Thus, it is appropriate to assume a lower failed fuel fraction than for the current generation of nuclear plants. It should be noted that in the radiological analysis for the Containment Purge System the expected coolant concentrations were used (based on a 0.025% failed fuel fraction). For the safety assessment of the AP600 plant, the design basis coolant concentrations were used (based on a 0.25% failed fuel fraction).

23rd DOE/NRC NUCLEAR AIR CLEANING AND TREATMENT CONFERENCE

A LOW PRESSURE FILTER SYSTEM FOR NEW CONTAINMENT CONCEPTS

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Abstract

It is demonstrated that after severe accidents the decay heat can be removed in a passive mode in a convective flow, i.e. without needing a fan. The filter components with sufficiently low pressure drop values which are required for this purpose will be described and the results indicated.

I. Introduction

Largely passive safety systems are being discussed with a view to a new generation of pressurized water reactors to increase the safety even in severe accident scenarios. This includes the management of core meltdown accidents, e.g. by providing a core catcher [1] and by guaranteeing the removal of the decay heat from the containment [2]. In this way, contamination of the neighborhood and evacuation of the population living nearby are to be avoided, even under conditions of core meltdown. First concepts were presented [3] at the preceding DOE/NRC conference.

II. The Containment Concept

The steel shell of the containment is made pressure-tight. It is planned to remove the decay heat to the ambient air through this steel shell. As leakages cannot be excluded, the vent air must be filtered in order to retain aerosols and iodine. As only convection is eligible as the driving force - assuming a passive mode solution - the maximum admissible pressure drop of the filter components inclusive of the losses in the ductwork is approx. 400 Pa.

Should a core meltdown accident occur, the energy supply could not be guaranteed. Otherwise, the problem of heat removal and filtration using proven filter components and fans could be conveniently solved in engineering terms.

Figure 1 shows the general layout of the containment with the filter components and their connection to the stack for improvement of convection. The construction of a test rig for determination of the heat transfer data and for experimental determination of the convection values is under way.

No conventional solutions will be discussed in this paper. At the outlet for convection air, above the reactor containment, temperatures of about 200 °C are anticipated which corresponds roughly to the surface temperature of the steel containment. Given these ranges of temperatures and the requirement of largely maintenance-free components, stainless steel fibers are used as filter media for the particulate air filter and silver-doped molecular sieves as iodine sorption material.

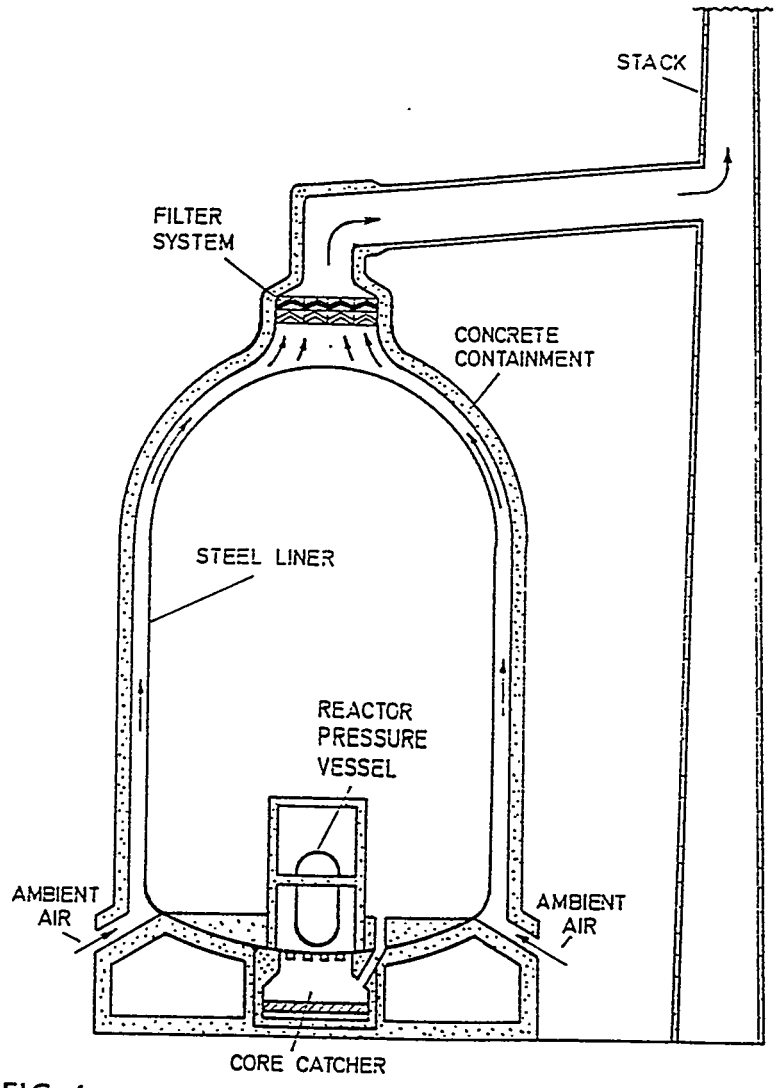


FIG. 1

CONTAINMENT 2000
COOLANT AND LEAKAGE FILTERED

III. The Filter Design

The stainless steel fiber fleeces are available at a fabricated width of 1.2 m. Therefore, a square configuration offers itself. As, in addition, the face velocities are the same in the iodine and particulate filters, a modular filter combination has been built where two fiber fleeces face each other on the inflow side and the outflow duct is enclosed by two sorption layers. These layers are introduced through a fill opening. By installing appropriate spacers a constant layer thickness can be ensured on the whole surface. Thus, the assembly can be made compact and the necessary open face area can be placed within several modular units.

The general layout is evident from Figure 2.

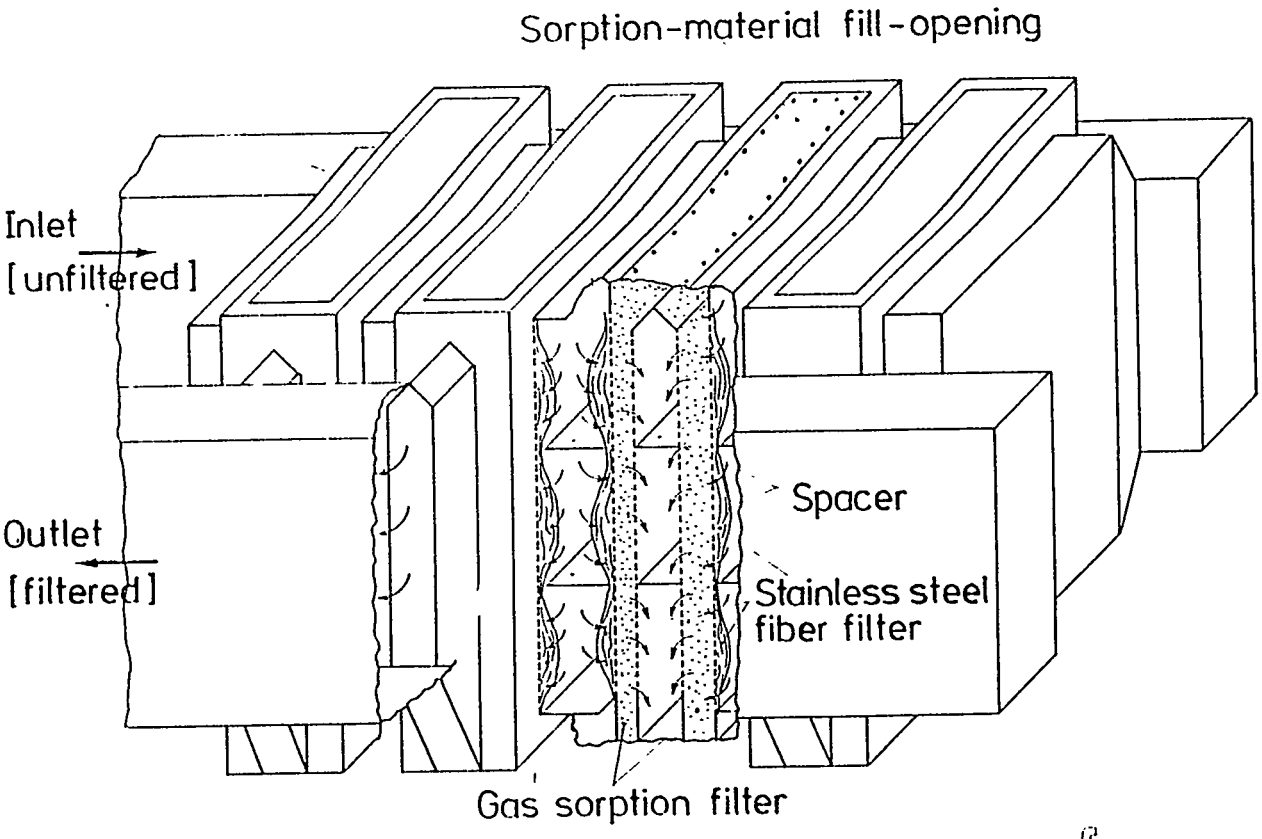


FIG. 2 Low pressure drop filter

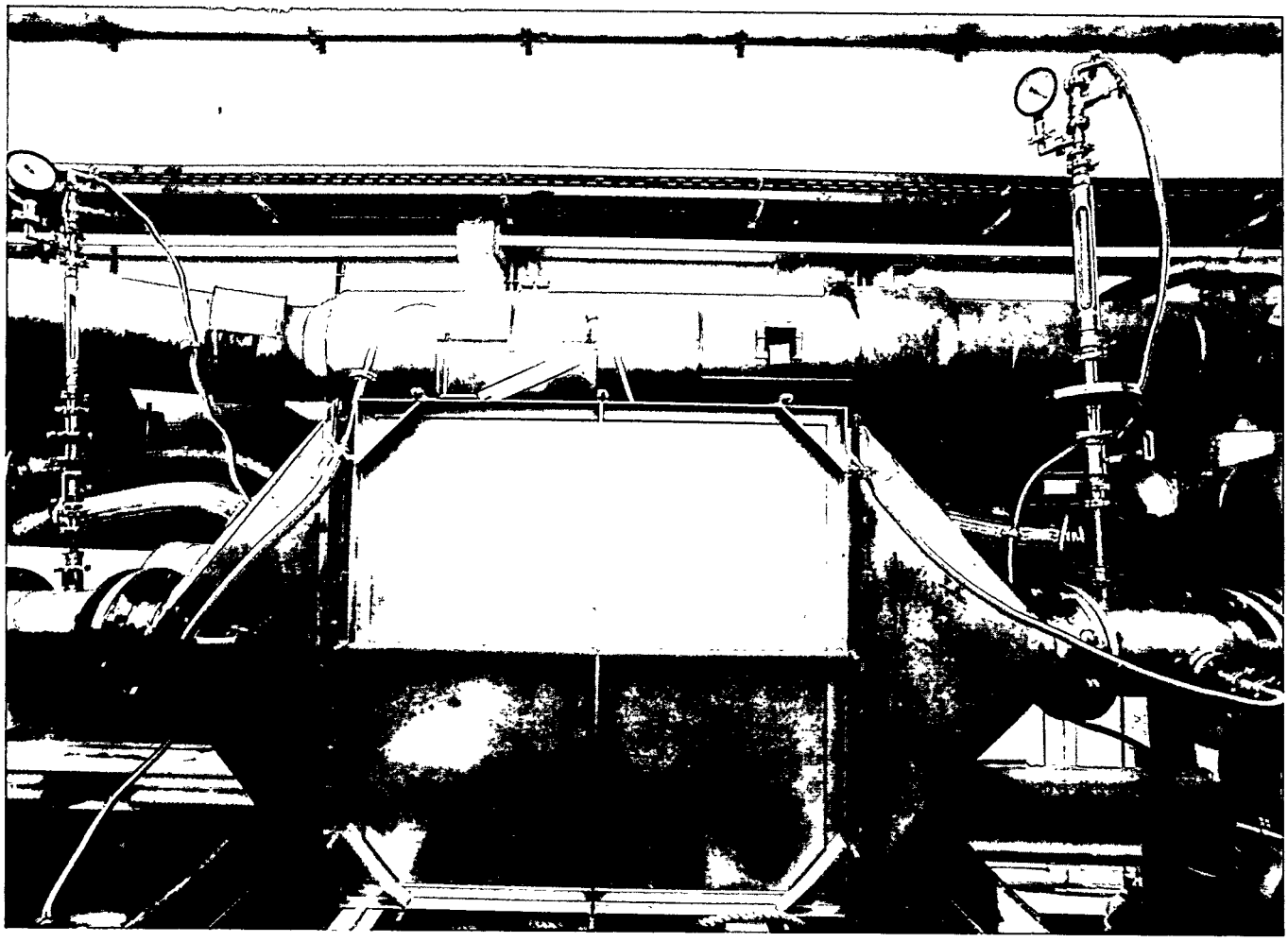


FIG. 3 The installed test filter module in the test rig

IV. The Particulate Filter

As temperatures will attain values up to approx. 200 °C, stainless steel fibers are used as well as for the venting systems. Also in this filter no organic seals are provided between the upstream and downstream side. The fibers clamped onto the edges are self-sealing. Unlike the venting filters, where high loadings had to be controlled (up to 10 kg per m^2), the amounts of aerosols to be anticipated here are not high because of the minor leakages, and therefore a plain layout can be chosen. Besides, since the expected steam leakages are also small, the evolution of condensate can be ruled out. The stainless steel fleeces are slightly sintered to improve their handling capability.

A decontamination factor (DF) of not more than approx. 1000 has been aimed at so far, because this value is also included in the radiological computations. A higher DF would give rise to a further increase in Δp of the filter and a further expansion of the open face area, respectively.

The values measured are entered in Figs. 4 and 5. It is evident that for face velocities < 5 cm/s a value Δp of 150 Pa is not exceeded.

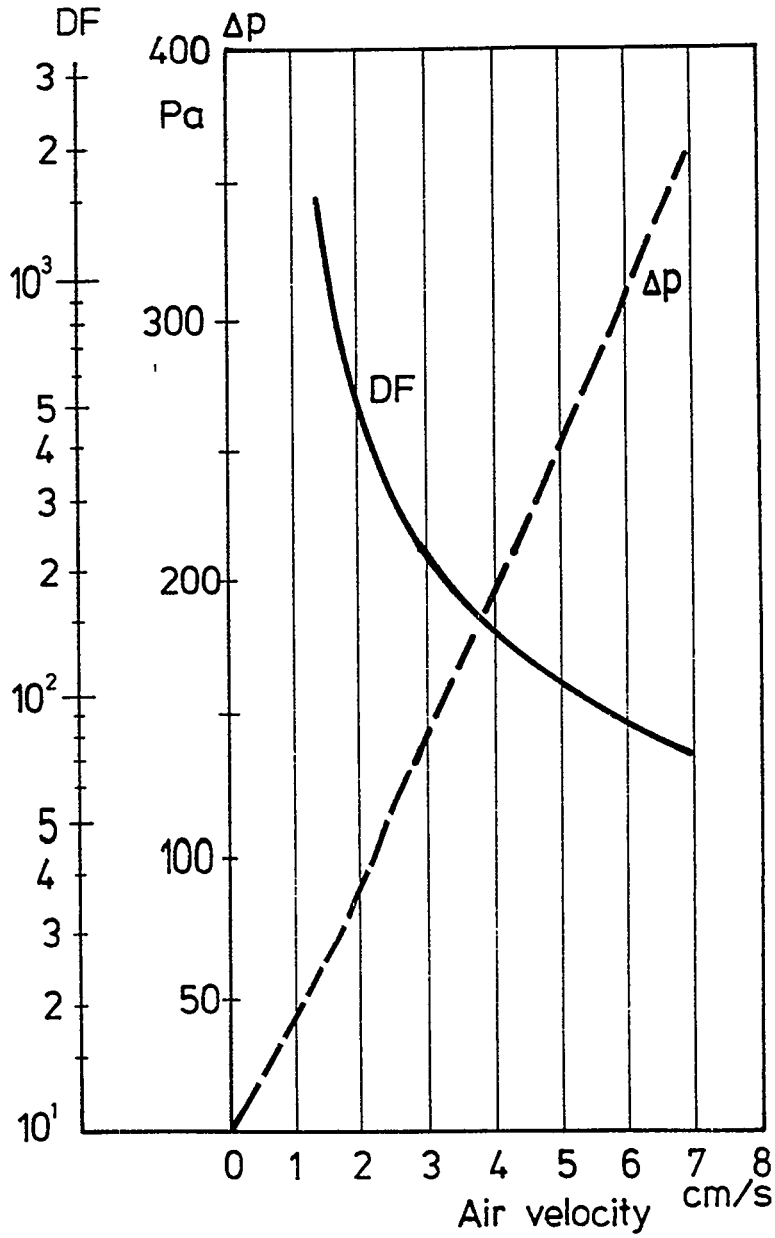


FIG. 4 DF and Δp as functions of the air velocity of a stainless steel fiber fleece filter fiber ø 2 um, 600 g/m² sintered

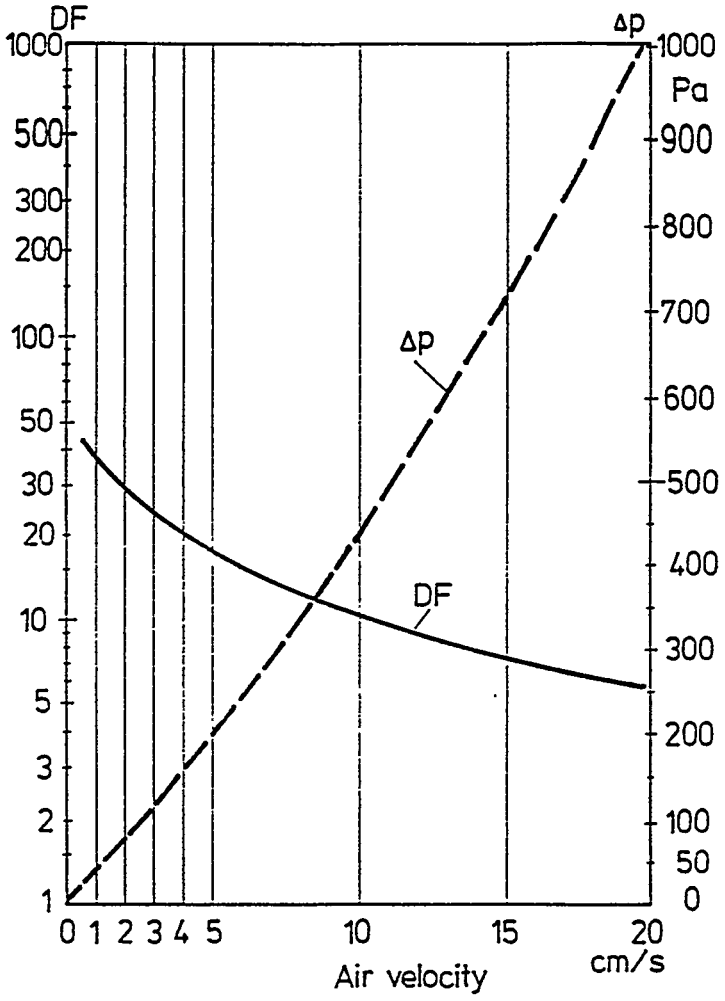


FIG. 5 DF and Δp as functions of the air velocity of a stainless steel fiber fleece filter module fiber ø 2 µm, 300 g/m² sintered

V. The Iodine Filter Stage

Figure 6 shows the pressure drop of sorption materials as a function of the face velocity. It shows that for the expected face velocities values of only ≤ 300 Pa are attained.

This corresponds to a residence time of about 0.6 s which is sufficient for a DF ≧ 1000 of the iodine filter stage. On account of the requested thermal stability, activated carbon is not eligible. However, the Ag-doped molecular sieves known from the venting systems will satisfy the requirements. Therefore, further parameter studies on the molecular sieves have been dispensed with.

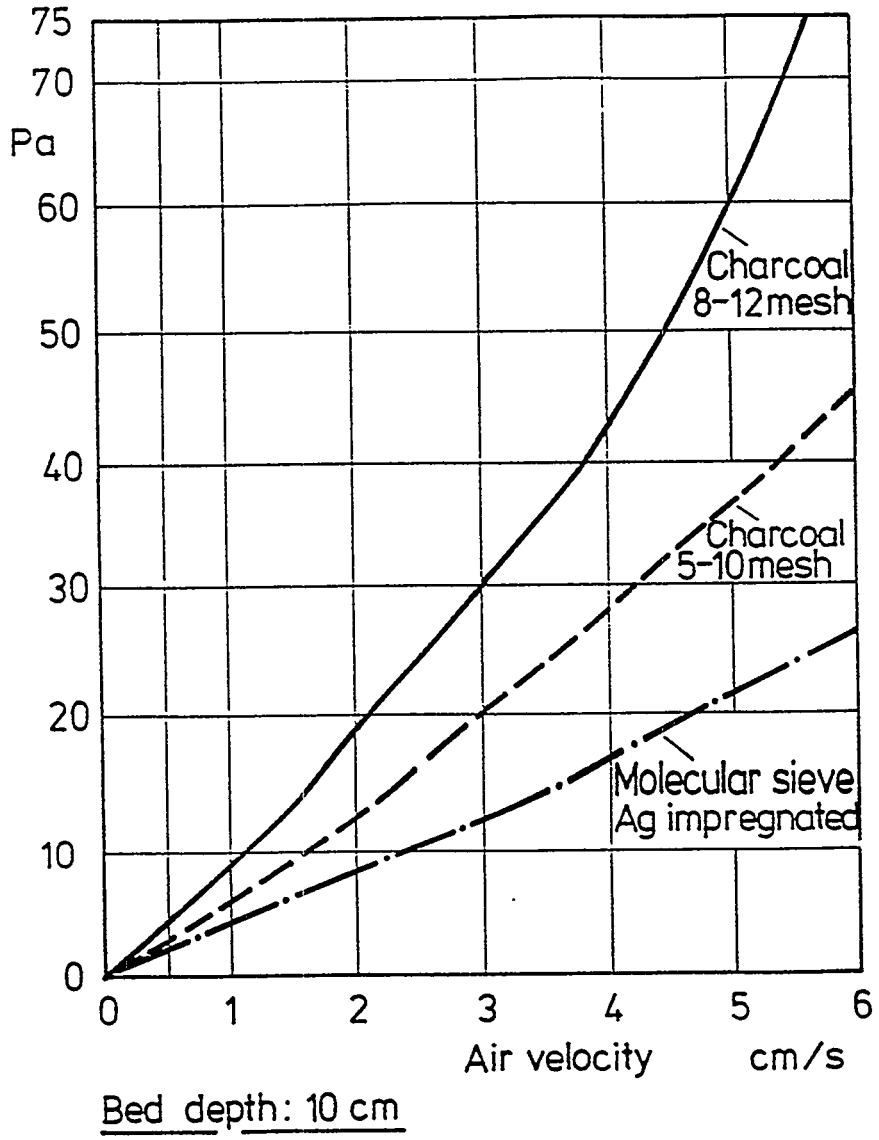


FIG. 6 Pressure drop of Iodine sorption filter materials

VI. Summary

It has been demonstrated that passive filtration is feasible in modern containment concepts. It allows filtered decay heat removal without external energy supply to be achieved after severe accidents. So the impact on the environment, even in the improbable case of a core meltdown accident, can be kept so low that there is no need for evacuation precautions to be taken. In this way, the damage can actually be confined to the facility proper.

VII. References

[1] W.Tromm, H. Alsmeyer: "Experiments for a Core Catcher Concept Based on Fragmentation", Proc. ARS '94, Topical Meeting on Advanced Reactors Safety, Pittsburgh, April 17.- 21. 1994.

[2] F. J. Erbacher, H. J. Neitzel: "Passive Containment Cooling by Natural Air Convection", Special Issue of Nuclear Technology 'A New LWR Safety Concept', to be published.

[3] J. G. Wilhelm, H.-J. Neitzel: "Concept for a Passive Heat Removal and Filtration System Under Core Meltdown Conditions", CONF-9020823, Vol 1, p. 863-874.

DISCUSSION

DILLMANN: For the past 15 years I have presented information to you from our institute. Our institute has ceased to exist and I and my coworkers will be committed from now on to gas cleaning for waste and hazardous waste incineration plants, which means that I must give up my work in the nuclear engineering field. This will probably be a farewell from the DOE conference for quite a long time.

CLOSING COMMENTS OF SESSION CO-CHAIRMAN PORCO

We have had a very long and interesting session. We have learned about value impact assessment of HVAC and chilled water systems for events including seismic, floods, fires, and tornadoes. I think we have learned that fire is the most significant event for HVAC systems. We have heard about a damper leakage method for use in installed dampers. We have heard about a novel glove box system from England. We have heard about a modification to a calculation code model to integrate the risk of fire in ventilation networks. We have also learned that there was no measurable release of radioactivity from the systems at the Lawrence Livermore National Lab during the 1980 and 1989 earthquakes. We have also learned about current air cleaning practices in Belgium and China. We learned about a new approach to the design of containment air filter systems for advanced light water reactors to satisfy radiological-controlled area filtration functional requirements. And, lastly, we have learned about a low pressure filter system from Germany that uses convection instead of a fan for circulation after a severe accident.

SESSION 13

SOURCE TERMS AND ACCIDENT ANALYSIS

Wednesday: July 27, 1994
Co-Chairmen: R. R. Bellamy
R. Zavadoski

OPENING COMMENTS OF SESSION CO-CHAIRMAN BELLAMY

REVISED ACCIDENT SOURCE TERMS FOR LIGHT-WATER REACTORS
L. Soffer

POTENTIAL RADIONUCLIDE EMISSIONS FROM STACKS ON THE HANFORD SITE, PART 1: DOSE ASSESSMENT
W. E. Davis, J. M. Barnett

THE LINK BETWEEN OFF-SITE-EMERGENCY PLANNING AND PLANT-INTERNAL ACCIDENT MANAGEMENT
H. Braun, R. Görtz

NAUAHYGROS - A CODE FOR CALCULATING AEROSOL BEHAVIOR IN NUCLEAR POWER PLANT CONTAINMENTS FOLLOWING A SEVERE ACCIDENT
R. Sher, J. Li

EXPERIMENTAL STUDY ON THE PARTICLES DEPOSITION IN THE SAMPLING DUCT
J. Vendel, J. Charuau

CLOSING COMMENTS OF SESSION CO-CHAIRMAN BELLAMY

OPENING COMMENTS BY SESSION CO-CHAIRMAN BELLAMY

In this plenary session we are going to discuss source terms and accident analysis. As you can see from the program, there are five papers in the session. Two of them are of international interest and two papers will discuss source term analyses. Another paper will get into a bit of a non-technical area. We will have a chance to hear some words this morning about the link between source terms and emergency planning. The author of the first paper is the principle author of a draft NUREG 1465. We are pleased to be able to continue our discussion of the Nuclear Regulatory Commission's revised accident source term.

REVISED ACCIDENT SOURCE TERMS FOR LIGHT-WATER REACTORS

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US Nuclear Regulatory Commission
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Abstract

This paper presents NRC revised accident source terms for light-water reactors (LWRs) which incorporate the significant severe accident research insights gained in this area over the last 15 years.

Current LWR reactor accident source terms used for licensing date from 1962 and are contained in Regulatory Guides 1.3 and 1.4. These specify that 100% of the core inventory of noble gases and 25% of the iodine fission products are assumed to be instantaneously available for release from the containment. The chemical form of the iodine fission products is also assumed to be predominantly elemental (I_2) iodine. These assumptions have strongly affected present nuclear air cleaning requirements by emphasizing rapid actuation of spray systems and filtration systems optimized to retain elemental iodine.

A proposed revision of reactor accident source terms and some implications for nuclear air cleaning requirements was presented at the 22nd DOE/NRC Nuclear Air Cleaning Conference. A draft report (NUREG-1465) was issued by the NRC for comment in July 1992. Extensive comments were received. While most have been favorable, the most significant comments involve (a) release fractions for both volatile and non-volatile species in the early in-vessel release phase, (b) gap release fractions of the noble gases, iodine and cesium, and (c) the timing and duration for the release phases. The final source term report is currently in preparation and is expected to be issued in late 1994. Although the revised source terms are intended primarily for future plants, current nuclear power plants may voluntarily request use of revised accident source term insights as well in licensing.

This paper emphasizes additional information obtained since the 22nd Conference, including contractor studies on fission product removal mechanisms, results obtained from improved severe accident code calculations and resolution of major comments, and their impact upon the revised accident source terms.

Revised accident source terms for both BWRS and PWRS in terms of fission product composition, magnitude, timing and iodine chemical form are presented. Related regulatory activities in regard to severe accident source terms are also discussed.

I. Introduction and Background

Radionuclide releases, that is, the type, quantity, timing and energy characteristics of the release of radioactive material from reactor accidents ("source terms") are deeply embedded in the regulatory policy and practices of the U.S. Nuclear Regulatory Commission (NRC). The NRC's reactor site criteria[1] require for licensing purposes that an accidental fission product release from the core into the containment be postulated to occur and that its radiological consequences be evaluated assuming that the containment remains intact but leaks at it$ maximum allowable leak rate.

The characteristics of the "source term" into the containment, which must be distinguished from a release to the environment, is contained in Regulatory Guides 1.3 and 1.4[2,3], but is derived from the 1962 report TID-14844[4], and consists of 100% of the core inventory of noble gases and 50% of the iodines (half of which are assumed to deposit on interior surfaces very rapidly).

Severe accident research insights[5] have confirmed that although the TID-14844 release is very substantial and has resulted in a very high level of plant capability, nonetheless, based upon the large amount of information obtained on severe accidents since the publication of TID-14844 more than thirty years ago, the present recipe is no longer compatible with a realistic understanding of severe accidents. It has both conservative and non-conservative aspects. For example, the assumption of an instantaneous appearance of a large fraction of the core inventory of noble gas and iodine isotopes within containment has long been recognized as highly conservative, whereas neglect of other important nuclides released into containment, such as cesium, may be non-conservative.

A major NRC research effort began about 1981 and has been under way since then to obtain a better understanding of fission product transport and release mechanisms in light water reactors (LWR) under severe accident conditions. This improved methodology on severe accident source terms has been reflected in NUREG-1150[6] which provides an updated risk assessment for five U.S. nuclear power plants. An assessment of the chemical form of iodine found within containment as a result of a severe accident has also been carried out[7].

The NRC has issued a draft report for comment[8] proposing revised LWR reactor accident source terms. These proposed accident source terms, including a discussion of some implications for nuclear air cleaning requirements, was presented at the 22nd DOE/NRC Nuclear Air Cleaning and Treatment Conference.[9]

II. Revised Accident Source Terms

The timing, magnitude in terms of fractions of the core inventory released into containment, and the composition of the proposed revision of accident source terms were based upon a range of severe accident sequences studied in NUREG-1150 and included release fractions for a complete core-melt accident, including core-concrete interactions and

releases associated with late in-vessel volatilization. A revised iodine chemical form[7] was also incorporated. The draft report also included a quantitative discussion of non fission product aerosols released into the containment from reactor vessel materials and core-concrete interactions. It must be emphasized that the formulation of the revised source term was intended to represent the generalized progression of a severe accident in order to assure that a plant design is capable of adequately mitigating such accidents. The characteristics of the revised source term were not intended to be an accurate representation of any single accident sequence, but rather combines aspects of a number of core-melt sequences.

Comments on the draft revised source term were solicited from an internationally recognized group of experts in severe accident source terms, as well as from the nuclear industry and members of the public. Over 200 separate comments were received from more than 20 individuals and organizations. The comments were generally favorable and expressed the view that the revised source term incorporated severe accident research insights and represented a major advance over the older source term employed in TID-14844. However, a number of comments expressed the view that the early in-vessel release fractions might be low, whereas the ex-vessel release fractions were believed too high. Commenters also stated that the release fractions for the low volatile nuclides, both in-vessel as well as ex-vessel, were believed to be too high. Release fractions for the gap release phase were also considered high. Comments were also received to the effect that iodine chemistry was rather complex, and that organic iodine was not addressed in the draft report. Finally, several commenters stated that the report should include a better treatment of removal mechanisms, and that showing releases for a dry reactor cavity was inappropriate for a number of advanced reactor designs that were proposing to maintain a flooded reactor cavity.

As part of the effort to finalize the source term, the NRC also sponsored contractor studies to develop more realistic models of predicting removal of fission products within containment by natural removal mechanisms as well as by engineered features such as sprays and suppression pools. At the present time, studies providing improved models of aerosol scrubbing by a water pool overlying core debris[10] and aerosol removal by containment sprays[11], have been completed and issued. Additional studies on aerosol removal by BWR suppression pools and aerosol removal in containment by natural removal processes are still in progress.

In addition, studies using the MELCOR[12] code have been performed for several sequences at the Peach Bottom[13] and Surry[14] nuclear power plants. While results from these studies showed general agreement with those of draft NUREG-1465, the duration of the gap release phase for BWR reactors was reduced from 1.0 to 0.5 hours, based on these studies[13].

The final version of NUREG-1465 is currently being readied for publication. Revisions have been incorporated as a result of the comments received as well as the additional studies noted above. The major changes in the final source term report compared with the draft version are summarized below:

(1) Release fractions for the non-volatile fission products during the early in-vessel phase, particularly cerium and lanthanum, were significantly reduced. Release fractions for the same nuclides during the ex-vessel phase were also reduced somewhat. The release fractions used in the draft source term report were based on mean values of the NUREG-1150 distributions to estimate release fractions for all nuclides. Since release estimates for the non-volatile nuclides showed a very large degree of uncertainty (spanning over four orders of magnitude) compared to the volatile nuclides, the mean value was unduly influenced by the extreme upper end of the range, and was in excess of other measures of the distribution, such as the 75th percentile. Recent research results[15] since the issuance of NUREG-1150 also indicate that release of the non-volatile fission products has been overestimated. For the final source term report, the 75th percentile of the distribution was selected to estimate the release fractions for the non-volatile nuclides, since it bounded the bulk of the data, without being unduly conservative.

(2) The duration of the BWR gap release phase was reduced from 1.0 to 0.5 hours, as noted above.

(3) A discussion of organic iodide formation, not addressed in the draft report, was incorporated in the final version. Organic iodine, while representing only a small fraction of the iodine released, can be important in the evaluation of the radiological consequences of design basis accidents because it is not readily removed by sprays or filters. It appears primarily as CH_3I, and is produced over time primarily as a result of the interaction of elemental iodine with organic compounds present in containment. An analysis and review of containment experiments[16] indicated that about 3.2 percent of the airborne iodine would be converted to organic iodides within the first two hours. The final source term report has adopted a conversion of 4 percent of the airborne iodine to organic form. Where the pH is controlled to a value of 7 or greater, elemental iodine would be no greater than about 5 percent of the total released. Hence, where pH control is maintained, organic iodide would represent about 0.20 percent of the total iodine released.

(4) A revised grouping of the nuclides, shown in Table 1, was adopted based on grouping barium and strontium together, because of similarity, rather than separately, as in the draft report.

(5) Quantitative estimates of fission product removal mechanisms, will be deleted from the final report. Since fission product removal is highly plant specific, treatment of this important area should be done on a case specific basis.

(6) Numerical tabulations of non fission product aerosols masses released into containment will be deleted from the final report, based on studies[13] indicating that estimates in the draft report may have been significantly overestimated. Since reliable data may not be presently available, non fission product aerosols will be addressed in the final report qualitatively.

Table 1 Revised Radionuclide Groups

TITLE	ELEMENTS in GROUP
Noble Gases	Xe, Kr
Halogens	I, Br
Alkali Metals	Cs, Rb
Tellurium group	Te, Sb, Se
Barium, Strontium	Ba, Sr
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
Cerium group	Ce, Pu, Np

Revised accident source terms, including composition, magnitude, timing and duration, are shown in Tables 2 and 3 for BWRs and PWRs, respectively.

Table 2 Revised BWR Releases Into Containment*

	Gap Release	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.5	1.5	3.0	10.0
Noble Gases	0.05	0.95	0	0
Halogens	0.05	0.25	0.30	0.1
Alkali Metals	0.05	0.20	0.35	0.1
Tellurium	0	0.05	0.25	0.005
Barium, Strontium	0	0.02	0.1	0
Noble metals	0	0.0025	0.0025	0
Cerium group	0	0.0005	0.005	0
Lanthanides	0	0.0002	0.005	0

* Values shown are fractions of core inventory.

The rate of release of fission products into the containment is assumed to be constant during the duration time of each phase. The releases shown for the ex-vessel phase are assumed to be for a dry reactor cavity having no water overlying any core debris. Where water is covering the core debris, aerosol scrubbing will take place and reduce the quantity of aerosols entering the containment atmosphere.

Table 3 Revised PWR Releases Into Containment*

	Gap Release	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.5	1.3	2.0	10.0
Noble Gases	0.05	0.95	0	0
Halogens	0.05	0.35	0.25	0.1
Alkali Metals	0.05	0.25	0.35	0.1
Tellurium	0	0.05	0.25	0.005
Barium, Strontium	0	0.02	0.1	0
Noble metals	0	0.0025	0.0025	0
Cerium group	0	0.0005	0.005	0
Lanthanides	0	0.0002	0.005	0

* Values shown are fractions of core inventory.

III. Regulatory Applications

When issued final, the revised source terms in NUREG-1465 are intended to be applied for future plants and will not be backfit for presently licensed plants. One advanced design (Combustion Engineering System 80+) has used the source terms in the draft NUREG-1465 for licensing purposes, and the NRC staff has approved, subject to revision when the final report is issued.

The NRC staff has also proposed[17] that for evaluation of design basis accidents (DBA) for the evolutionary and passive light-water reactor designs, only the releases associated with the gap and early in-vessel release phases will be used. The NRC staff has indicated that it considers the inclusion of the ex-vessel and late in-vessel releases to be unduly conservative for DBA purposes. Such releases would only result from core damage accidents with vessel failure and core-concrete interactions. For evolutionary and passive light-water reactors, the estimated frequencies of such scenarios are low enough that they need not be considered credible for the purpose of meeting 10 CFR Part 100.

However, the staff has also noted that equipment and features needed for severe accident prevention, mitigation, and sampling should be designed to provide a reasonable level of confidence that they will operate in a severe accident environment. Such an environment would include the ex-vessel and late in-vessel releases, in addition to those for DBAs.

After final issuance, licensees of current plants may voluntarily request use of the revised source terms in NUREG-1465 for plant applications. Several licensees have expressed interest in such usage, and the NRC staff is preparing a plan to consider such applications.

IV. Conclusions

Revised reactor accident source terms, when issued in final form, will reflect an improved understanding of fission product releases into containment for a spectrum of degraded core and severe accidents having regulatory application to light-water reactors.

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DISCUSSION

MISHIMA: Do you think these source terms are applicable to the effects of explosions on spent fuel that would severely fragment the spent fuel, therefore increasing the surface area allowing release?

SOFFER: That was not what we had in mind. The source terms in NUREG 1465 were developed from reactor accident sequences involving core heatup and fuel melting. It is unlikely, in my opinion, to be applicable for spent fuel explosions. For one thing, spent fuel does not have much in the way of noble gases or much in the way of iodines, which are, of course, the things that have traditionally been of greatest concern in reactor accidents. For another, in fragmenting and exploding fuel, you are talking about releasing aerosolized quantities of fuel particles, including uranium, transuranides, et cetera, which are not the kinds of things that one would get in large quantities in the normal heatup of fuel. So my reaction is, probably not.

MISHIMA: What about the volatile materials that are present after five year cooling?

SOFFER: To the extent that there is cesium there, it may be okay for that. But to the extent that you are going to get plutonium and neptunium, etc., I suspect that that is not the ideal way of doing it.

FIRST: You made a point about accepting applications from plants to use the new source

terms. What are the advantages of plants electing the new source term?

SOFFER: Based on the limited information we have from utilities, the industry appears to believe that there are a number of advantages in the timing aspects; that is, in getting away from an instantaneous release. For example, at the present time, secondary containments in a BWR have to turn on emergency ventilation systems immediately, and are required to get down to a negative pressure in order to receive credit for that function to be accepted as a secondary containment. This puts a rather high load on the diesels, and puts a large strain on the ventilation systems. There are people who believe this is not warranted in view of timing considerations. There are a number of people who feel the timing aspects of the source term provide advantages with regard to improved reliability, valve closure time, easier loading on diesels, things of this kind. There may be other applications as well.

HULL: As a Long Islander and someone who has been inside the business, I realize you are probably not the person to answer this question, but I will ask it anyway. Shoreham is gone, so it is too late for that, but do you see any of this leading to a relaxation of the ten mile EPZ, which effectively did in Shoreham (if any one requirement did)?

SOFFER: That is going to be the subject, I believe, of another paper in this session, and I would like to hear what the other author will say. Emergency planning has been based on a spectrum of accidents that considered not only the design basis accident, but considered very severe accidents where containments are likely to fail. The revised source terms were developed for plant design purposes where the containment is assumed not to fail to incorporate a better understanding of degraded core accidents. These source terms were not intended to be used for emergency planning.

KOVACH; L.: Is there to be any revision of fuel handling accidents, in addition to vessel accidents, in the new source term?

SOFFER: The fuel handling accident is not specifically covered in the revised source term; however, revised gap release fractions are provided. This might provide a basis for reconsidering the fuel handling accident. There is going to be a new revision of gap activity releases. And this could lead the way for a revision of fuel handling accidents.

KOVACH, L.: That would be another document?

SOFFER: That is right.

POTENTIAL RADIONUCLIDE EMISSIONS FROM STACKS ON THE HANFORD SITE, PART 1: DOSE ASSESSMENT

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Abstract

On February 3, 1993, the U.S. Department of Energy, Richland Operations Office (RL) received a Compliance Order and Information Request from the Director of the Air and Toxics Division of the U.S. Environmental Protection Agency (EPA), Region 10. The Compliance Order requires RL to evaluate all radionuclide emission points at the Hanford Site to determine which are subject to continuous emission monitoring requirements in Title 40, Code of Federal Regulations, Part 61, (40 CFR 61) Subpart H, and to continuously monitor radionuclide emissions in accordance with requirements in 40 CFR 61.93. The Information Request required RL to provide a written Compliance Plan to meet the requirements of the Compliance Order. A Compliance Plan was submitted to EPA, Region 10, on April 30, 1993.

The Compliance Plan specified that a dose assessment would be performed for 84 Westinghouse Hanford Company (WHC) stacks registered with the Washington State Department of Health (WAC 246-247) on the Hanford Site. Stacks that have the potential emissions to cause an effective dose equivalent (EDE) to a maximum exposed individual (MEI) greater than 0.1 mrem y^{-1} must be monitored continuously for radionuclide emissions. Five methods were approved by EPA, Region 10 for performing the assessments: Release Fractions from Appendix D of 40 CFR 61, Back Calculations Using A HEPA Filtration Factor, Nondestructive Assay of HEPA Filters, A Spill Release Fraction, and Upstream of HEPA Filter Air Concentrations. The first two methods were extremely conservative for estimating releases. The third method, which used a state-of-the-art portable gamma spectrometer, yielded surprising results from the distribution of radionuclides on the HEPA filters. All five methods are described.

Assessments using a HEPA Filtration Factor for back calculations identified 32 stacks that would have emissions that would cause an EDE to the MEI greater than 0.1 mrem y^{-1}. The number was reduced to 15 stacks when the other methods were applied. The paper discusses reasons for the overestimates.

I. Introduction

On December 15, 1989, EPA promulgated in the Code of Federal Regulations (CFR) the National Emission Standards for Emissions of Radionuclides Other Than Radon From Department of Energy Facilities--40 CFR 61, Subpart H[1]. Subpart H requires the emissions of radionuclides from the U.S. Department of Energy (DOE) facilities shall not exceed those amounts that would cause any member of the public to receive an effective dose equivalent (EDE) of 10 mrem in a year. In addition, potential emissions of radionuclides from a facility that could cause an EDE to a maximum exposed individual to exceed 0.1 mrem y^{-1} require continuous monitoring.

To determine compliance, section 61.93 provides requirements for monitoring of radionuclide emissions from point sources. These monitoring requirements became effective upon promulgation of the regulation on December 15, 1989. Also, DOE Richland Operations Office (RL) as defined in 40 CFR 61 is an "owner or operator" of a "facility," i.e., the Hanford Site (see Figure 1). On February 3, 1993, RL received a Compliance Order and Information Request from the Director of the Air and Toxics Division of the U.S. Environmental Protection Agency (EPA), Region 10. The Compliance Order required RL to evaluate all radionuclide emission points at the Hanford Site to determine which are subject to continuous emission measurement requirements in 40 CFR 61, Subpart H, and to continuously measure radionuclide emissions in accordance with 40 CFR 61.93. The Information Request required RL to provide a written Compliance Plan to meet the requirements of the Compliance Order. A Compliance Plan was submitted to EPA, Region 10, on April 30, 1993. The Compliance Plan set as one of the milestones: Complete assessment of the 84 Hanford Site registered stacks under management by WHC by December 17, 1993.

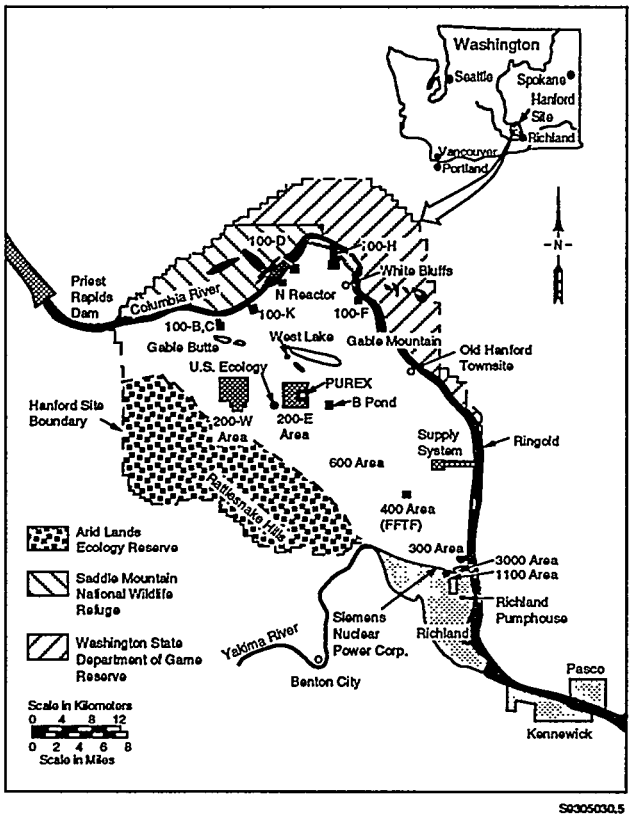


Figure 1 Hanford site.

The purpose of this document is to present the dose assessment results for the registered stacks on the Hanford Site for potential emissions (i.e., emissions with no control devices in place). Further, the document will identify those stacks requiring continuous monitoring (i.e., the EDE from potential emissions >0.1 mrem y^{-1}).

II. Scope

The stack assessment of potential emissions was performed on 84 registered stacks on the Hanford Site. These emission sources represent individual point sources presently registered under Washington Administrative Code 246-247 with the Washington Department of Health. Unfortunately, for most stacks potential emissions are not sampled. For these stacks, other methods were developed and submitted to EPA, Region 10 for approval. The methods used in assessing the potential emissions from the stacks are described below.

III. Methods for Calculating Potential Stack Emissions

The calculation of potential emissions for the registered stacks were made using five methods approved for use by the EPA, Region 10:

1. Source Term Estimation Based on release factors in Appendix D
2. Back Calculations Based on a Decontamination Factor of 3000^n, where n equals the number of HEPA filter banks in series
3. Nondestructive Assessment (NDA)
4. Upstream of HEPA Filter Air Concentration Measurements
5. Spill Release Fraction (296-Z-5)

Calculations based on the first two methods are considered extremely conservative. The last three methods should provide a realistic estimate of the actual potential releases. When the assessment of stack emissions based on Appendix D or from back calculations could indicate that the potential emissions would result in an EDE exceeding 0.1 mrem y^{-1}, while one of the last methods would result in an EDE less than 0.1 mrem y^{-1}, the more realistic method results would be accepted over the first two conservative methods.

1. Appendix D Release Factors

The potential emissions for a system without control devices can be estimated based on factors presented in 40 CFR 61 Appendix D: for gases the release factor is 1, for liquids or particulate solids the release factor is 1×10^{-3}, and for solids the release factor is 1×10^{-6}. This method is extremely conservative, because accidents involving liquids and loose particulates have a release fraction orders of magnitude less than the 1×10^{-3} release fraction. In an accident condition, a spill of powder from a 1 m height of the entire inventory would release under extremely conservative conditions 4×10^{-5} release fraction and a liquid spill from the same would release approximately 1×10^{-6} release fraction. A 1×10^{-3} release fraction is used for powders and liquids. In cases where the assessment was performed on waste tanks, if the inventory was covered by supernate, the supernate was evaluated using the release factor for liquids while the covered salt cake was estimated using a release factor for solids.

2. Back Calculations Based on a Decontamination Factor of 3000^n

This method for estimating potential emissions assumes the Nuclear Air Cleaning Handbook[2] decontamination factor (DF) of 3000 (DF~ 1/(1-0.9997)) for 0.3μ by a HEPA filter. The method assumes that each bank in series acts independently of a preceding bank and will remove contamination with the same 3000 decontamination factor. For a system with n banks of HEPA filters in series, the decontamination factor is 3000^n. This method can be conservative for a contaminated system. When processing no longer occurs, the resuspension of contamination downstream of the HEPA filters can dominate the airborne releases from a facility. Multiplying these releases by 3000^n will overestimate the potential emissions by orders of magnitude.

3. Nondestructive Assessment (NDA)

In September 1992, EPA, Region 10, concurred with a WHC proposal for a test method to measure radioactive particulate emissions (particularly gamma emitters from high-efficiency particulate air [HEPA] filters) in stack exhauster systems across the Hanford Site (see Part II, Barnett, 1994[3]). A nondestructive assessment (i.e., in situ) method was developed, tested, and implemented using portable low-resolution (NaI) gamma spectroscopy instrumentation[4] to evaluate gamma emissions from operating HEPA filters. Guidance for the developed method comes from the 40 CFR 61, Subpart H and Appendix B, Method 114[5].

4. Upstream of HEPA Filter Air Concentration Measurements

The upstream air concentrations provide direct information on the potential emissions from a facility. This method can be applied by using Continuous Air Monitoring (CAM) data, inserting air sample probes for upstream measurements, or radiological analysis of removed HEPA filters. CAM data can be used if the data is taken from the process area that contains the radionuclide inventory. Air measurements may also be collected in the ventilation system upstream of the HEPA filters. In this case, a sampling port is selected, a probe inserted, and air concentrations are measured. The final method is the sampling of the furthest upstream HEPA filters for the facility. In this case, one or more HEPA filters are removed and the radiological content is analyzed either by destructive assay or by a gamma spectrometer. The airflow during the time the HEPA filter is online is used to determine the annual release rate. These methods are technically based in that the measurements represent the potential concentrations emitted from a facility without control devices.

5. Spill Release Fraction (296-Z-5)

In the facility venting to the 296-Z-5 stack, the only potential emissions occur when a canister of PuO_2 powder is repackaged. No emissions are expected from contamination in the ventilation system for the facility because it is essentially uncontaminated. When a canister is repackaged, the double canisters containing 1,500 g of PuO_2 are opened in a hood and the powder is poured into another canister. The canister is sealed and put into another canister and it is sealed. The

pouring could cause a release of PuO_2 to the air. To estimate the potential maximum release, observed spill release fractions for depleted uranium oxide spills (Sutter et al. 1981[6]) was used. A maximum release fraction value of 4.0×10^{-5} was used. This number is extremely conservative because it represents a release fraction for a 1 m release height for the spill. The actual release height is a maximum of 0.15 m. The spill release fraction was only used for the 296-Z-5 stack.

IV. CAP88 Dose Modeling

A potential source term was calculated for the 84 registered stacks using one of the five approved methods. Once the potential source terms were determined, the source term, location, and stack height were used with Hanford Site meteorology in the CAP88 computer model (Beres 1991[7]) to predict the EDE for the maximum exposed individual.

The CAP88 model is able to incorporate plume rise from the stack based on the flow rate and stack diameter. After leaving the stack, the plume is modeled to disperses based on meteorological conditions. Modeled concentrations are decreased based on dry deposition (dry deposition velocity = 1.8 cm/s.) However, wet deposition is ignored because of the low incident of precipitation on the Hanford Site. The modeled concentrations are used to calculate an EDE for an offsite MEI by summing the product of the EDE for each isotope with its activity.

$$\dot{H}_{total} = \Sigma(EDE_{(i)} \times A_{(i)}) = mrem\ y^{-1} \qquad (1)$$

where:

$\dot{H}$ = potential unabated dose to the MEI
$EDE_{(i)}$ = effective dose equivalent for each radionuclide
$A_{(i)}$ = total activity for radioisotope i
i = radioisotope

Normalized source terms were used in CAP88 for each of the major areas on site (i.e., 100, 200, 300, 400, and 600 Areas). The 5-year average of Hanford Site meteorological data (1983-1987) taken on the site was used to compute normalized air concentrations, i.e, X/Q, where X is the air concentration (curies/m^3) normalized by the source term Q (curies/s). Two stack release heights were used: 89 m (200 ft stack with a 100 ft plume rise) and 10 m (33 ft). The following input was needed to model an individual stack: the potential radiological source terms, the stack release height was chosen, and the location of the stack in one of the major areas. Potential dose data was produced by the CAP88 model for each registered stack.

V. Results of Dose Assessment

Of the 82 registered stacks assessed, 41 were assessed using release fractions from Appendix D, 31 using back calculations, nine stacks using NDA, two using upstream air samples, and one using the powder release fraction. Fifteen stacks (see Table 1) were identified to have potential emissions that could cause an effective

Table 1 Designated stacks.

Stack Number	Facility	Number of HEPA Filters
291-A-1[b]	PUREX	3
296-A-1	PUREX	3
291-B-1[b]	B Plant	2
291-Z-12	PFP	1-3[c]
296-Z-3	PFP	2
296-A-22[b]	Tank Farms	2
296-A-27	Tank Farms	2
296-A-29	Tank Farms	2
296-A-40	Tank Farms	2
296-P-16	Tank Farms	2
296-A-22[b]	Tank Farms	2
296-P-23	Tank Farms	2
296-S-7W	ERO	2
296-S-15	Tank Farms	2
308-GL-EX	308 Building	3
340-NT-EX[b]	340 Building	2

[a]Stacks with potential emission to cause a dose >0.1 mrem y^{-1}.
[b]Original seven designated stacks.
[c]Stages of filtration depends on exhaust pathway.

dose equivalent >0.1 mrem y^{-1}. One of the original seven designated stacks, 291-T-1, was assessed by NDA and reduced to a nondesignated stack[8].

To evaluate the five different methods, each method was compared to the results of back calculations for all stacks. The back calculations were performed for all stacks and the results indicated emissions from 32 stacks would have caused an EDE that would have exceeded 0.1 mrem y^{-1} and required the stacks to provide continuous monitoring. When a stack reported a non detectable release, the minimum detection limit was used as the release for the stack. A comparison made for a subset of 30 stacks using release factors from Appendix D with back calculations showed no correlation between the two methods (Figure 2). Further, 21 of the 30 back calculations potential emission exceeded the release fractions emissions. In only four comparisons did the release fractions from Appendix D identify a designated stack when the back calculation indicated a nondesignated.

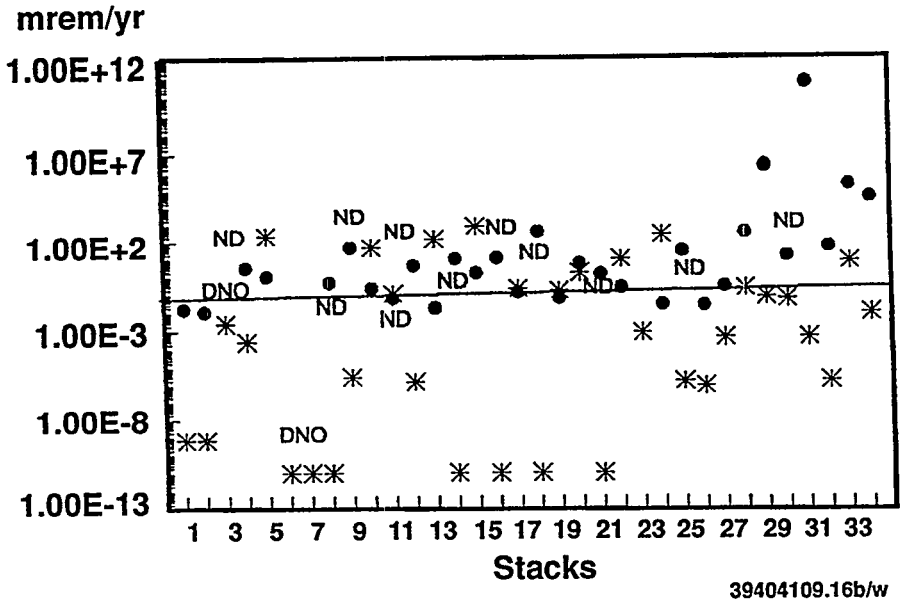


Figure 2 Doses from back calculations compared to release fractions from Appendix D (* - Release Fraction Appendix D, • - Back calculations).

The comparison between NDA and back calculations showed that back calculations overestimated the potential releases for all nine stacks by at least four orders of magnitude (Table 2 and Figure 3). (Four of the stacks [see Table 3] reported nondetection of a release. For these stacks the minimum detection limit was used for back calculations.) If back calculations had been used instead of NDA, all nine stacks would have required continuous monitoring. Upstream air sample method for two of the stacks showed that the back calculations overestimated the emissions by three orders of magnitude. For the single stack that used the powder release fraction, the back calculations overestimated the release by four orders of magnitude.

Table 2 NDA Comparison with back calculations.

Stack Number	NDA (mrem y^{-1})	Back Calculation (mrem y^{-1})	LOG_{10} (Difference (mrem y^{-1}))
291-T-1	4E-3	4E+2	5
296-T-13	3E-4	3E+1	5
296-B-13	5E-8	2E+0[a]	8
296-G-1	3E-8	3E-1[a]	7
296-S-2	1E-5	1.2E-1	4
296-S-16	2E-6	1.0E-1	5
296-W-3	2E-9	3E+0[a]	9
308-ET-EX	3E-5	1E+2	7
437-MN&ST	2E-3	2E+1[a]	4

[a]Minimum detection limit used for back calculations.

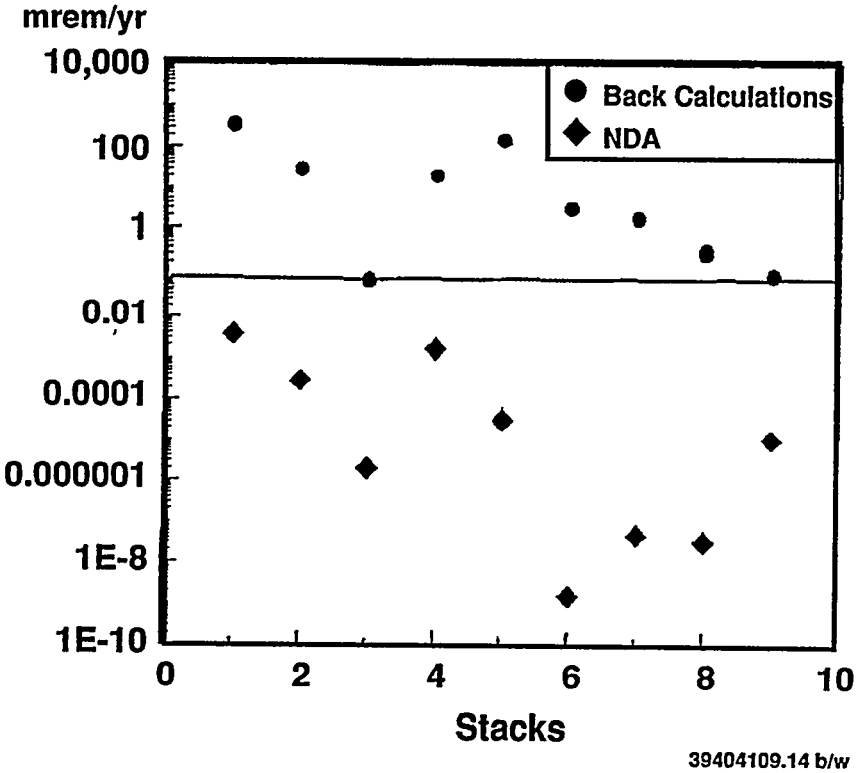


Figure 3 Doses from NDA compared to back calculations.

Table 3 Apparent decontamination factors for nondesignated stacks.

Stack Number	Method	Number of HEPA	Nuclide	Unabated (Ci)	Release (Ci)	DF
291-T-1	NDA	Prefilter, 2 HEPAs	Sr90	1.6E-3	1.2E-5	190
			Cs137	1.7E-3	1.3E-5	130
			Pu239,240	2.0E-3	2.2E-5	110
			Am241	2.0E-4	2.0E-6	100
296-T-13	NDA	2	Pu239,240	1.7E-4	2.0E-6	90
296-T-11	Upstream	2	total beta	1.5E-5	4.3E-7	30
296-S-2	NDA	2	total alpha	2.1E-6	8.4E-9	240
			total beta	3.5E-6	1.4E-8	250
296-S-7W	Upstream	2	Pu239,240	3.2E-5	7.5E-7	40
			Am241	3.9E-4	4.2E-6	90
296-S-16	NDA	1	total alpha	3.1E-7	1.7E-9	185
			total beta	9.8E-6	5.3E-8	180
308-ET-EX	NDA	2	total alpha	9.3E-7	9.2E-8	10
			total beta	1.8E-7	3.4E-7	0.5
437-MN&ST	NDA	2	total alpha	2.0E-5	1.9E-7	105
			total beta	7.8E-5	7.4E-7	106

When upstream measurements or NDA were performed, an apparent decontamination factor was computed for eight nondesignated stacks having releases above the detection limit. The term apparent has been

used because an actual decontamination factor could not be computed. An actual decontamination factor would have to be calculated for a variety of particle sizes with air concentration measurements before and after the filters. The apparent decontamination factor was formed using either the estimated annual air concentration or the loading on the filters ratioed with the measured releases. The measured releases were <2E-5 curies per year for a radioisotope for all the stacks. The unabated emissions for these stacks were less than 3E-3 curies y^{-1}. For one stack 308-ET-EX a DF of 0.5 was computed (i.e., more released than was assessed on the HEPA filters). For some of the designated stacks the DFs, if measured, would produce larger DFs than noted in Table 3.

VI. Conclusions

By using the five approved methods approved by EPA, Region 10 instead of only the original back calculation method for assessing the 84 WHC stacks, the number of stacks requiring continuous monitoring was reduced from 32 to 15.

The intercomparison between results showed that no correlation existed between back calculations and release fractions from Appendix D results. Also, the NDA, upstream air samples, and powder release fraction method results were at least three orders of magnitude lower than the back calculations results.

The most surprising results of the assessment came from NDA. NDA was found to be an easy method for assessing potential emissions. For the nine stacks assessed by NDA, all nine would have required continuous monitoring when assessed by back calculations. However, when NDA was applied, all stacks had potential emissions that would cause an effective dose equivalent below the >0.1 mrem y^{-1} standard.

Apparent decontamination factors were calculated for eight nondesignated stack emissions above the detection limit. These apparent decontamination factors ranged from 0.5 to 250.

Acknowledgments

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DISCUSSION

KOVACH, L.: The in-place permitted leak rate for a HEPA filter bank is 0.05% maximum. The efficiency of the filter itself is a minimum of 99.97%. Thus, permissible in-place efficiency is always lower than 99.97%.

DAVIS: No comment.

KOVACH, L.: Presence of contamination downstream of HEPA banks is always an indication of a bypass of HEPA filter banks (possibly during change out or before installation).

DAVIS: The presence of contamination downstream of the HEPA filters comes from: (1) older facilities (from the early 40's) operated with no filtration, causing contamination of the ducting, (2) for processing facilities, operations have now ceased but contamination was caused during the processing period, (3) during change-out some contamination could have moved downstream. Calculations made for resuspension of contamination in the ducting indicates that as little as 1×10^{-5} curies could dominate annual releases.

KOVACH, L.: At some point, decontamination of ducts downstream of existing HEPA filter banks may be more cost effective than continued "studies". Based on currently available data, do we know what the actual release is from the stacks? Based on currently available data, do we know the DF of the existing air cleaning systems?

DAVIS: The stack sampling system currently in place measures stack releases. Actual DFs are difficult to measure. However, in-place testing performed on the HEPA filters indicated that they are still performing within specifications.

KOVACH, L.: I have some familiarity with the Hanford site, so I understand the problems. I feel that, considering the time and money being spent on evaluating things, at some point you may want to look at decontaminating the downstream side of the duct because it costs so much to maintain some of these systems. You are in a decontamination mode in most of the facilities from which these stacks are supposedly releasing activity.

HULL: I cannot help but ask this question, even though it is probably not appropriate in this forum, has anybody made a cost-benefit analysis of dollars spent vs. lives saved? How much is the public paying for being protected by all this, versus the dose that they are not receiving?

DAVIS: That's a very good point, because the dose from Hanford site is something like $6x10^{-3}$ mrem/yr which is orders of magnitude below 10 mrem/yr. The regulation, 40CFR61 subpart H, requires us to evaluate it without any controls in place. This makes it necessary to upgrade a system, which is now orders of magnitude below the 10 mrem/yr limit, at a cost of about $500,000/stack.

THE LINK BETWEEN OFF-SITE-EMERGENCY PLANNING AND PLANT-INTERNAL ACCIDENT MANAGEMENT

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Abstract

A variety of accident management measures has been developed and implemented in the German nuclear power plants. They constitute a fourth level of safety in the defence-in-depth concept. The containment venting system is an important example. A functioning link with well defined lines of communication between plant-internal accident management and off-site disaster emergency planning has been established.

1. Introduction

Plant-internal accident management (AM) has been developed after the Chernobyl accident during the last years as a fourth additional level in the defence-in-depth concept. Its objective is to stabilize the plant in a long term safe state in cases of severe accidents which are beyond the design base incidents. A variety of AM measures both of the preventive and of the mitigative type have been developed and partially implemented in nuclear power plants (NPPs) in Germany. These measures are taken into account in probabilistic safety assessments (PSA) which represent a powerful investigative tool providing information on accident event sequence analyses and their probability of occurrence. The AM measures and their state of implementation is described in the following emphasizing the equipment for filtering and air cleaning.

In the case of an actual hazardous state developing in the plant, in parallel to the efforts to prevent core damage or to mitigate its consequences, precautionary off-site disaster control measures could be initiated. These measures and the criteria for initiation are discussed in the following.

At the end of this paper, a closer look at the link between plant-internal AM and off-site emergency planing is taken. Potential improvements in linking these two areas of high safety relevance in cases of severe accidents can be expected with respect to the source term, appropriate choice of emergency training scenarios and criteria for initiation of off-site measures.

2. Accident Management Measures

2.1 The Defence-in-depth Concept in Nuclear Safety

Protection against possible dangers is based on two major concepts. The first consists in preventing damage by taking appropriate precautionary measures, the second tries to limit the extent of possible damages. The greater the extent of possible damages becomes, the more important become precautionary measures which, by eliminating endangering mechanisms already in the forefield, provide for a greater distance to the manifestation of the risk. In protecting essential functions, the principle of multiple defence lines in the sense of a graduated prevention has been proven to be most effective.

This led to the development of an extensive safety concept based on the principle of graduated prevention. From the beginning, one strove to develop safety measures far ahead in the preventive area wherever this was possible and reasonable from a technical standpoint. The danger potential from radioactive materials is contained within several consecutive physical barriers. These themselves are protected by a graduated system of precautionary safety measures, the defence-in-depth concept (see Figure 1).

- **Level 1**

At the first level, high quality standards are applied to all plant components with the goal of preventing failures from occurring at all. Components and systems are designed with high safety margins and subjected to extreme quality assurance measures not only during manufacture but also during the operation of the plant.

- **Level 2**

At the second level, disturbances are intercepted before they can develop into incidents. Here, the limitation system and the reactor protection system which monitors all essential measurement values of the plant are of central importance. Whenever specific limits are exceeded protective actions such as power reduction or shutdown of the reactor are initiated. The set points are specified such that overstressing of the plant components can be excluded from consideration.

- **Level 3**

Third in line is the equipment of the safety system (that means engineered safety features) which protects the radioactivity barriers in case of damage to important systems. Here the design goal is that - in case of an incident - at least two radioactivity barriers will remain intact, thus preventing a dangerous release of radioactivity to the environment. Spatial separation and special structural measures protect the safety system from damage due to events such as fire or flooding. The engineered safeguard are designed to ensure that all design basis accidents are controlled and coped with. The analytic proof of safety required for the design basis accidents will always use assumptions for insecure data or physical models which will never overestimate, rather underestimate, the effectiveness of the safety system.

2.2 Accident Management Measures

The defence-in-depth concept as described in 2.1 has proven successful in establishing a high standard of safety in the nuclear field. On the other hand, experience has shown that the levels of defence not necessarily prevent accidents.

Even the most reliable safety system cannot guarantee that a system failure or a combination of failures occurs which is not covered by the design. It is for instance hypothetically possible that during a major incident the safety equipment itself fails. The analyses of beyond-design accidents, in the course of which a failure of the safety equipment is assumed, show that in most cases several hours remain before serious damage occurs to the reactor core. This leaves time for the prevention of damage by using remaining safety systems as well as operational or even external systems.

Since protective measures must be taken well in advance of failure development, it is important to apply these level-4 protective measures as early as possible within the plant and not to delay their initiation until effects of the accident on the environment become imminent.

Figure 1 shows the introduction of a fourth safety level into the defence-in-depth concept.

2.3 Overview over Accident Management (AM) Measures

Accident management measures can be grouped into preventive and mitigating measures. The first group of measures is aiming at the prevention of damage to the core. The second group tries to limit the release of radioactive materials as far as possible and to prevent catastrophic effects even in the case of a severe core damage.

In Table 1, an overview is provided on accident management measures that are implemented already in German NPP or which are foreseen conceptually. Two practical examples shall be presented as applied in NPPs in Germany:

- Coolant Injection to the Reactor Pressure Vessel (RPV) of a Boiling Water Reactor (BWR)

  In the case of BWR, credit can be taken from high flexibility in the use of various injection systems for core cooling, e. g. in case of a station blackout, fire fighting systems, drinking water supply and a mobile pump can feed into the reactor pressure vessel. Emergency operating procedures ensure an activation in a relatively short time to prevent a degradation of the reactor core.

- Containment Integrity

  The containment vessel is of particular in that it presents the last physical barrier in the case of beyond-design accidents. Therefore, accident management measures are considered for its protection almost independently from the probability of occurrence of corresponding damaging events. The measures concerned belong to the group of mitigation measures.

Tab. 1: Overview over Accident Management Measures

Overview over Accident Management Measures, partially implemented in German PWRs
• Emergency Manual
• Primary and Secondary Bleed and Feed
• Filtered Containment Venting
• Filtration of Air Supplied to Control Room
• System for Measuring Activity Concentration in Containment Atmosphere and Sump
• Improvement of Containment Function
• Different Measures to Improve Power Availability (Connection to Neighbouring Units, Battery Capacity, Additional Connection to Grid, Recovery of Grid Connection)

2.4 Air Cleaning AM-devices

Of direct relevance to nuclear air cleaning are the AM-measures filtered containment venting (FCV) and filtered air supply to the control room. Therefore, these measures shall be discussed here in more detail. The discussion is restricted to the pressurized water reactor (PWR). The objective of FCV is to prevent late overpressurization of the containment and to limit and control off-site releases. The design requirements include for the case without water injection to the containment pressure limitation at test pressure. (With water injection : Depressurization to half of the test pressure within two days). There are two different technical concepts that are applied:

- Combination of stainless-steel fibre filters for aerosol retention and venture scrubber for iodine- and additional aerosol retention

- Combination of stainless-steel fibre filters for aerosol retention with molecular sieves for iodine-retention

Existing penetrations through the containment shell are used.

The iodine filtration unit is located in the auxiliary building whereas the aerosol filter can be located inside the containment or in the auxiliary building as well.

These options are displayed in Figure 2. The possibility to arrange the aerosol filter inside the containment was achieved by progress in reducing the size of the unit. This option offers two advantages. Due to the higher pressure, the gas volume that has to be filtered is smaller. Furthermore, the radioactive aerosols filtered remain inside the containment. On the other hand, the equipment is exposed potentially to external forces caused by the accident and the filter capacity decreases with the size.

Under accident conditions, the filters of the containment venting system are exposed to extreme loads. For the case of a PWR, some typical key design data for the filter system are:

- 40 kg of aerosols to be filtered

- 2 kW decay heat of the aerosols

- decontamination factor > 1000
- vent gas temperatures in the range of 100° C to 200° C
- aerosol concentration ≤ 5 g/m^3
- mass flow 3,5 kg/s

Stainless steel fibre filters have been developed in Germany to meet the requirements derived from accident analyses. These modular constructed filters consist of a first layer of metal-cloth material followed by a series of layers of stainless steel fibre filters with decreasing cross section of the fibres. Then, a moisture separator is used to reduce the steam contents that affects the final stainless steel fibre filter with 2 μm thickness of the fibres. With the last stage decontamination factors of 10 000 for aerosols in the range of 0,1 to 0,2 μm are achieved.

In Germany it has been decided that the control room should be the central place for planning and performing measures of accident management. In order to ensure long term habitability of the control room in case of an emergency, appropriate filter systems will be installed to limit radiation exposure from inlet air into the control room. A slight overpressure is applied to avoid inward leakage. The filter system for this purpose is usually held in reserve and installed when required.

3. The Evaluation of AM Measures in Probabilistic Safety Assessment (PSA)

One of the objectives of the German Risk Studies / 3, 4 / has been the derivation and evaluation of accident management (AM) measures. AM means to make use of existing safety margins, human intervention and additional equipment in a plant for the early detection of plant conditions or sequences of events not explicitly foreseen in the design and to control these and to limit the consequences inside and outside the plant. According to the core-damage-state AM can be divided into preventive and mitigative measures / 5 /.

Examples for preventive AM measures derived from PSA-results comprise secondary and primary bleed and feed, which prevent core damage and stabilize the plant on the long term.

The AM measures are taken into account in PSAs on the basis of assumptions of the engineering judgement type concerning the corresponding success probability. It is estimated in / 3 / that AM measures reduce the frequency of uncontrollable events from $3 \cdot 10^{-5}$ /a to $4 \cdot 10^{-6}$ /a. The group of initiators which is mostly affected by AM measures are the transients. An impressive example for an effective AM is primary bleed and feed. The single contributions to the frequency of core damage induced by small leaks in the primary circuit or steam generator-tube rupture were reduced by two orders of magnitude, from 10^{-6} /a to 10^{-8} /a.

In an ongoing investigation project an evaluation concept for active and passive safety funtions of the containment is under development (Containment Event Tree, CET). Essential steps during development of CET are the temporal arrangements of selected practicable AM measures in the event sequence and the limitation of the time interval of assessment from the point shortly before RPV is damaged to the loss of containment - integrity. All relevant physical phenomena during ex-vessel phase are taken into account. It is expected that effectivity of AM measures can be enhanced by the results of this investigation.

4. Off-Site Disaster Control

After being informed by the plant operator of a nuclear accident, the head of the disaster control authority or his representative will decide which level of alarm (see chapter 5) shall be given. In accordance with the alarm level and the radiological condition, one or more of the following measures will be taken:

- general alert in accordance with a prescribed plan,
- identification of the endangered region,

- deployment of radiological measurement troups,
- warnings and information to the public,
- traffic restrictions,
- sheltering,
- administration of stable iodine,
- evacuation,
- alarming the drinking water supply stations,
- decontamination,
- medical care and supply.

The measures in this list that are essential to the protection of the public are the recommendations to stay sheltered, to distribute iodine tablets and to evacuate. Guidelines with respect to dose values for these recommendations are presented in the following table 2:

Tab. 2: Emergency Reference Levels for Countermeasures

Countermeasure	Reference Levels of Dose in mSv							
	Whole Body (external exposure and inhalation)		Thyroid (inhalation)		Lung or any other Dominantly Exposed Individual Organ (external exposure and inhalation)	
Sheltering	> 5	> 50	> 50	< 250	> 50	< 250
Administration of Stable Iodine	-	-	> 200	< 1 000	-	-
Evacuation	< 100	< 500	> 300	< 1 500	> 300	< 1 500

The dose values shown in this table are specified within individual bandwidths. This allows for a freedom of decision above the lower and below the upper dose value on executing the individual measure depending on the specific situation (e.g., depending on the extent of the emergency situation, daytime or night-time, weather condition).

A further measure for the protection of the population is the installation of emergency stations where affected persons can receive medical treatment.

A detailed discussion of off-site disaster control in Germany is provided in / 2 /. It covers the regulatory framework, the responsibilities of the Federal Government and of the States, surveillance and international obligations.

5. The Interface between Plant Internal Accident Management and Off-Site Emergency Measures

The link between plant internal accident management and the off-site emergency preparedness is an adequate flow of information from the plant to the concerned authorities in case of an emergency. In Germany

the operator of a nuclear power plant is bound by the nuclear licensing procedure to inform the disaster control authority when necessary. There are two alarm levels specified:

- early-warning disaster alert is given in case of an event in the nuclear facility where so far the effect on the environment is negligible with regard to the criteria for giving a disaster alert, where, however, on the basis of the plant condition it cannot be excluded that effects corresponding to the criteria for giving a disaster alert may develop.

- disaster alert is given if in case of an accident in a nuclear facility a dangerous release of radioactive materials to the environment was determined or is imminent.

The German Reactor Safety Commission (RSK) and the Radiological Protection Commission (SSK) are currently establishing specific alert criteria. These are intended to help the plant operator in deciding on the necessity for notifying the authorities and in deducing a recommendation to the authorities regarding early-warning or disaster alert from the individual accident condition. These criteria will clarify the indeterminate words in the above definitions and will allow the plant operator to differentiate between

- plant conditions due to accidents

- emissions

- immissions

in accordance with clear technical criteria or directly measured values.

Figure 3 shows a simplified scheme of the routes of information in the case of an incident or accident in a German NPP.

The operator of the NPP is obliged to inform the responsible state ministry, which is supported by a state institution performing measurements. Further state ministries have to be involved, for example the ministry which is responsible for public protection.

From the state level, the competent Federal ministry is informed (Ministry for the Environment, Nature Conservation and Nuclear Safety; BMU). BMU is responsible for communication on the international level. Support to the BMU is provided by the Federal Office for Radiation Protection (BfS) which evaluates and conducts measurements and characterizes the emergency situation radiologically.

This scheme reflects the federal structure of Germany. A more detailed description of responsibilities in the field of emergency preparedness is provided in / 2 /.

6. Conclusions and Summary

In the case of an accident in a nuclear power plant (NPP), well functioning link between off-site-emergency planning and plant-internal accident management is of vital importance.

It has been outlined that accident management (AM) measures have been introduced as a fourth level of safety into the defence-in-depth concept. A considerable variety of AM measures both of the preventive and mitigative type have been implemented in German NPP, more are being prepared for implementation. Of direct relevance to nuclear air cleaning are the filtered containment venting and the filtration of air supplied to the control room. In probabilistic safety assessment (PSA) it is demonstrated that the probabilities of core melt and of large releases are substantially reduced by AM measures.

For the purposes of off-site disaster control, a number of measures can be taken by the competent disaster control authority to protect the public. Emergency reference levels of dose for countermeasures (sheltering, administration of stable iodine, evacuation) have been defined.

The link between plant-internal accident management and the off-site emergency preparedness is an adequate flow of information from the plant to the concerned authorities in case of an emergency. The operator of the NPP has to provide information to enable the authority to decide whether or not early-warning disaster alert or disaster alert is given. In Germany, work is in progress to develop specific alert criteria to facilitate decision making.

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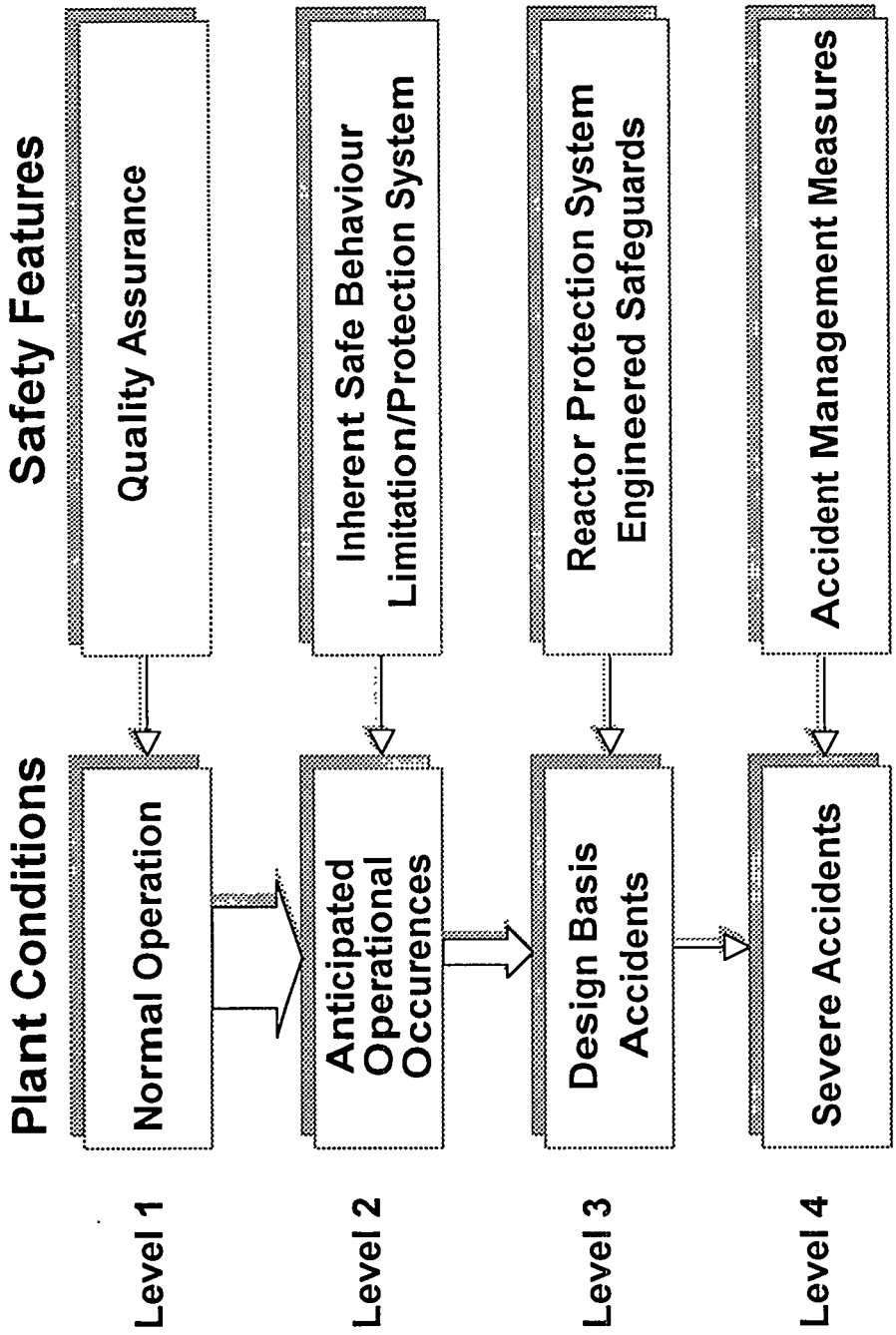
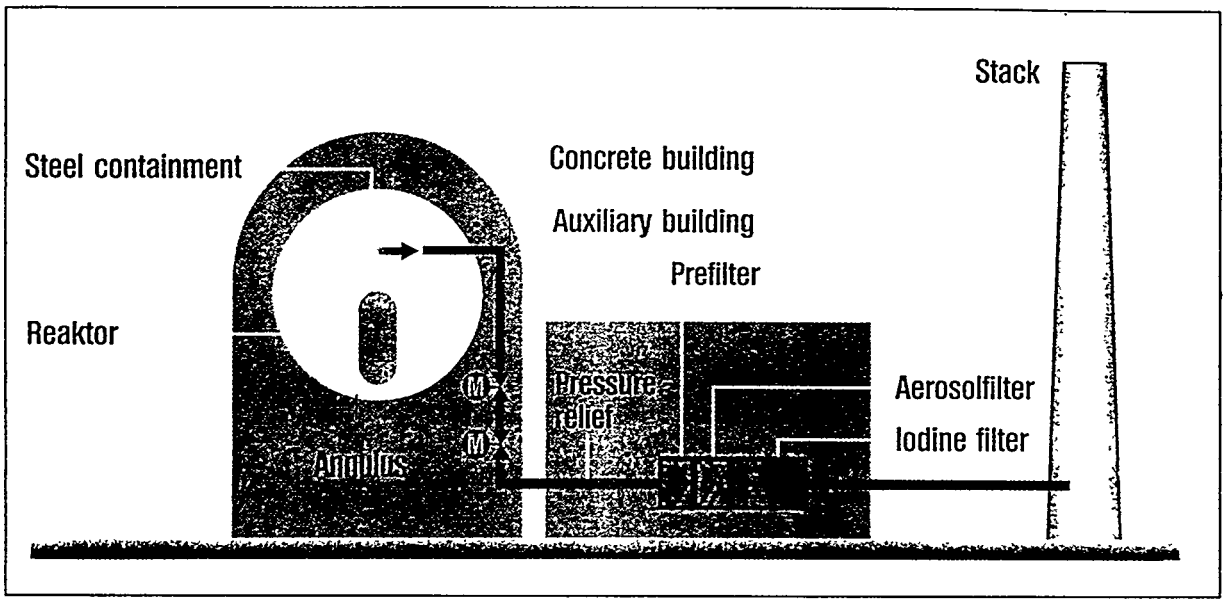


Figure 1: Defence-in-depth concept



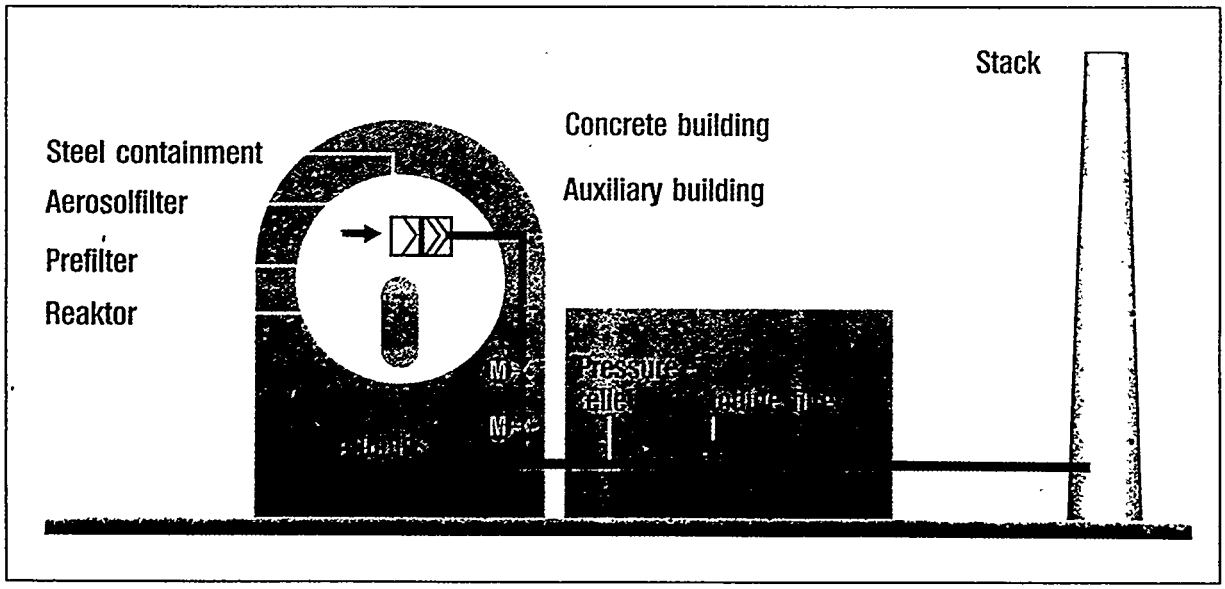
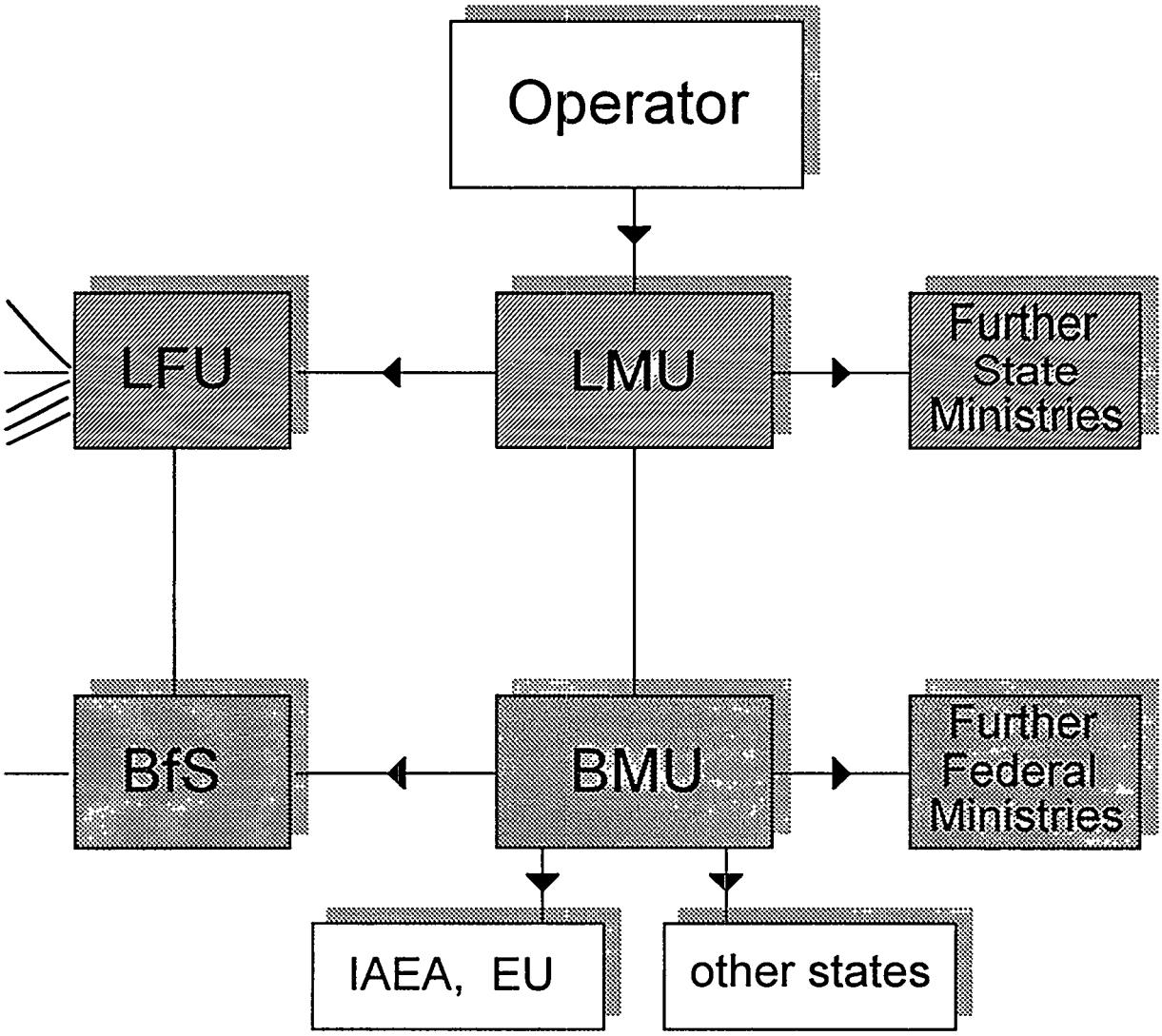


Figure 2: Two options for arranging perfilters and aerosol filters in pressurized water reactors



LMU: State Ministries responsible for Environmental Protection and

LFU: State Institution performing measurements

BMU: Federal Ministry of the Environment, Nature Conservation and Nuclear Safety

BfS: Federal Office for Radiation Protection

Figure 3: Simplified flow scheme for information in the case of an incident or accident in a German NPP / 2 /

DISCUSSION

KOVACH, L.: Have you considered releasing stable iodine at the plant, in commensurate quantity with the released radioactive iodine, to observe the same coverage as the distribution of the radioactive fallout? You would not have to relocate all these people and you would not have to go through a very complicated administrative procedure. With the varying wind directions and everything else, I think it would be a much simpler way of administering and controlling iodine than trying to track down all the people. I am not a health physicist, so I am just wondering what would be the best way to cover everybody who would be exposed to radioactive iodine.

BRAUN: I will discuss this with experts in the radiation protection commission, but I do not know whether it is possible from a technical standpoint. Too much iodine is needed as a practical reason and the public would not accept it.

NAUAHYGROS - A CODE FOR CALCULATING AEROSOL BEHAVIOR IN NUCLEAR POWER PLANT CONTAINMENTS FOLLOWING A SEVERE ACCIDENT

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Abstract

NAUAHYGROS is a computer code to calculate the behavior of fission product and other aerosol particles in the containment of a nuclear reactor following a severe accident. It is an extension of the German code NAUA, which has been in widespread use for many years. Early versions of NAUA treated various aerosol phenomena in dry atmospheres, including aerosol agglomeration, diffusion (plateout), and settling processes. Later versions added treatments of steam condensation on particles in saturated or supersaturated containment atmospheres. The importance of these condensation effects on aerosol removal rates was demonstrated in large scale simulated containment tests. The additional features incorporated in NAUAHYGROS include principally a treatment of steam condensation on hygroscopic aerosols, which can grow as a result of steam condensation even in superheated atmospheres, and improved modelling of steam condensation on the walls of the containment. The code has been validated against the LACE experiments.

I. Introduction

NAUA (German acronym for Nachunfallaerosolverhaltens, post-accident aerosol behavior) is a computer code to calculate the behavior of fission product and other aerosol particles in the containment of a nuclear reactor following a severe accident. It was developed in Germany over a period of years during the 1970s and early 1980s[1]. Two current versions of the code, NAUA mod-4 and mod-5, are in widespread use, for example in the U.S. NRC Source Term Code Package, and at ENEL(Italy), VTT(Finland), and many other organizations.

Early versions of the code treated various aerosol phenomena in dry atmospheres,including aerosol agglomeration, diffusion (plateout), and settling processes. In later versions, from mod-3 on, the phenomena associated with steam condensation on particles in saturated or supersaturated steam-air atmospheres characteristic of post-accident reactor containments were treated. The importance of these condensation effects was clearly demonstrated in the MARVIKEN, DEMONA, and LACE experiments[2]. In addition mod-5 incorporated a Stefan flow diffusiophoresis model.

In recent years modelling of steam condensation effects on particles has been extended to include the effects of hygroscopicity[3]. Hygroscopic particles can grow as a result of steam condensation even in superheated atmospheres. An improved treatment of steam condensation on interior heat sink surfaces (basically, walls) in the containment volume and its coupling to condensation on particles was developed by Li [4], who also changed the code architecture. One of the principal changes is that NAUAHYGROS uses the MAIN program as a driver for subroutines that individually contain the models for the various aerosol (and to some extent, containment) physics phenomena. (This was done to a limited degree in mod-4.) This makes it relatively easy to add new models (e.g., thermophoresis, turbulence, sprays) that are not included in mod-4 or NAUAHYGROS. These phenomena have been left out either because they were judged to be unimportant in the context of reactor accident scenarios of interest, or because required information to handle them (e.g., detailed gas flow distributions) would probably not be available.

NAUA was never conceived as a fully coupled aerosol/thermohydraulic code. The necessary thermohydraulic data had to be obtained from another source (either experiment or a containment thermohydraulic code) and input into NAUA. In mod-4 the input thermohydraulic data consisted of the bulk atmosphere temperature, the steam mass injection rate, and the volumetric leak rate, all as functions of time. NAUAHYGROS, on the other hand, because of its more detailed treatment of condensation effects, also requires wall temperatures, containment atmosphere pressure, and possibly heat fluxes to the wall. In addition, the user must also now input certain data for hygroscopic aerosol species.

A major change has been made to the way that the aerosol source data are input. In previous versions of NAUA, two size distributions (modes) in each release period were specified, and the mass fraction of each specie in each mode was specified. In NAUAHYGROS, each specie can be released in any (arbitrary) release period with its own (log normal) size distribution. As in NAUA, the source size distribution is defined by an average particle radius and the logarithm of the geometric standard deviation.

NAUA and NAUAHYGROS are so-called "nodal point" codes, that is, the aerosol size distribution is represented as a set of discrete points or "bins", rather than by a continuous function or correlation. In addition, as will be discussed later, NAUAHYGROS is a multicomponent code with moving size bins. Briefly, "multicomponent" means that to a certain extent aerosol species are tracked individually. In the original NAUA code, all suspended aerosol species are homogeneously mixed at each time step into one average composition which is independent of size. (This is not true of the condensed water on the particles - its mass fraction is size dependent.) In NAUAHYGROS, an average composition is determined for all particles in a given size bin, but this

composition varies from bin to bin. Thus, for example, if aerosol specie 1 is characterized by smaller particles than specie 2, the small size bins will have a larger fraction of specie 1 than the large bins, while the reverse situation will hold for specie 2.

"Moving" size bins arise from the fact that all particles in a given size bin will grow (or shrink) uniformly when steam condenses (or evaporates) on them, since they all have the same composition. The size of the bin can therefore be redefined after condensation or evaporation without changing its bin identification number. Doing this insures, for example, that if one condenses steam on the particles and then reverses the situation by evaporating the steam, one will recover the original size. This is not the case in NAUA, where a particle after condensation is "split" between the bins on either side of its new size. (The splitting procedure, which was originally developed to handle particle agglomeration, is still used for that purpose in NAUAHYGROS.) The use of moving size bins was first described by Gelbard[5] for the MAEROS code; it substantially reduces "numerical diffusion" effects.

II. Basic Theory and Modelling

Rate Equations

The basic aerosol equations used in NAUA (and NAUAHYGROS) will be briefly reviewed. These describe the rates at which suspended aerosol particles are added to or lost from a given size bin because of the processes of particle agglomeration (coagulation), diffusion, and gravitational settling. The particle size distribution function, which is in fact continuous, is approximated by a set of N quantities, representing the number of particles at discrete particle radius values (bins), thus forming a size histogram. The first bin corresponds to a chosen minimum radius, while the Nth bin corresponds to a chosen maximum radius. The bins are then spaced logarithmically equidistant, i.e. the ratio r_i / r_{i-1} is a constant. Clearly the larger N is, the more accurate this representation will be; on the other hand, computing time increases faster than N, approximately as N^2, (because agglomeration processes usually involve two sizes). In NAUA (and NAUAHYGROS), a value of N up to 110 can be chosen. Usually N = 30 gives sufficient accuracy with manageable computing times, although for "dry" calculations (no steam condensation on particles) larger values of N can be chosen.

It should be noted that throughout the following discussion it is assumed that the particles are spherical. Most of the applications of NAUAHYGROS are expected to involve aerosol behavior in steam/air atmospheres, in which condensation of steam on the particles will be expected to occur. It has been demonstrated[6] that under such conditions the particles will be spherical because of the surface tension of the condensed water.

User-specified shape factors to account for nonsphericity are available in NAUA and have been kept as an input option in NAUAHYGROS.

NAUA and NAUAHYGROS assume that the containment can be represented by a single volume in which the atmosphere and aerosols are uniformly distributed (well-mixed model). Thus there are no dependencies on position in the equations that follow.

In general the rate equations are of the form

$$dn_i/dt = K_l n_i \qquad (1)$$

for the processes of diffusion and sedimentation, and are of the form

$$dn_i/dt = K_{ij} n_i n_j \qquad (2)$$

for agglomeration processes.

In equations (1) and (2) n_i is the number of particles (or the number per unit volume) in size bin i, $K_l(r_i)$ is a rate constant characterizing the l^{th} removal process, and K_{ij} is an agglomeration kernel characterizing the rate of agglomeration between particles in bins i and j.

Deposition Velocities

For the phenomena described by equation (1), it is convenient to replace the rate constants K_l by their corresponding deposition velocities v_l, where $v_l = K_l V/A$. A is the surface area on which the deposition is taking place (e.g., upward-facing horizontal surfaces for sedimentation), and V is the containment volume.

The v_l's are given by the following well-known expressions:

Sedimentation:

$$v_s = 2\ \rho_p\ g\ r^2\ C(r)/9\mu \qquad (3)$$

Brownian Diffusion:

$$v_d = D(r)/\delta \qquad (4)$$

Stefan Flow Diffusion (Diffusiophoresis):

$$v_{sf} = \frac{x_s\sqrt{M_s}}{(x_s\sqrt{M_s} + x_a\sqrt{M_a})}\frac{W}{\rho_s} \qquad (5)$$

In equations (3-5) ρ_p is the particle material density, ρ_s is the steam density in the atmosphere, g is gravitational acceleration, μ is the viscosity of the atmosphere, $D(r) = kTC(r)/6\pi\mu r$ is the binary diffusion constant of the atmosphere, δ is a diffusion boundary layer thickness, C(r) is the Cunningham slip correction factor, $x_s(x_a)$ is the steam (air) mole fraction in the atmosphere, $M_s(M_a)$ is the molecular weight of steam (air), and W is the steam condensation rate per unit area on the wall surface.

The Cunningham factor is given by

$$C(r) = 1 + 1.246\ Kn + 0.42\ Kn\ \exp(-0.87/Kn) \qquad (6)$$

where Kn is the Knudsen number λ/r, λ is the molecular mean free path in the atmosphere.

In NAUA and NAUAHYGROS, the diffusion boundary layer thickness δ is user-specified; a value of 0.01 cm is usually recommended. The choice of δ is not critical, since particle removal (plateout) due to Brownian diffusion is usually quite small compared with the sedimentation or diffusiophoretic deposition rates. It should be noted that the diffusiophoretic velocity is independent of particle size.

Agglomeration Kernels

As a result of agglomeration between particles, particles are both removed from and added to size bins. If two particles of radii r_1 and r_2 coagulate, the radius of the particle thus formed is $r_3 = (r_1^3+r_2^3)^{1/3}$. If $K_{12}n_1n_2$ is the agglomeration rate between particles of radii r_1 and r_2,

$$dn_1/dt = - K_{12}n_1n_2$$
$$dn_2/dt = - K_{12}n_1n_2$$
$$dn_3/dt = + K_{12}n_1n_2$$

Note that there is a net loss of one particle per agglomeration; the total mass, however, is conserved.

Generalizing the above equations one obtains

$$dn_i/dt = - \sum_{j=1}^{N} K_{ij}n_in_j + 1/2 \sum_{(j+k=i)} K_{jk}n_jn_k \qquad (7)$$

where j+k=i means that the summation is taken over those bins for which $r_j^3 + r_k^3 = r^3$, where r^3 includes all radii which fall between bins (i-1) and (i+1); the reason for this will become apparent later. The first term in equation (7) represents the loss out of size bin i due to coagulation with all other bins

(including coagulations between particles both of which are in bin i); the second term represents the gain in bin i due to agglomerations between smaller particles.

There are three mechanisms that lead to particle agglomeration, namely, Brownian diffusion, turbulent diffusion, and gravitational "sweepout". The first two of these result in particle-particle collisions due to the random Brownian motion or random turbulence-induced motions of the particles. Gravitational coagulation results from the fact that larger or heavier particles fall with higher (Stokes) velocities than smaller or lighter ones. The faster particles can thus overtake and collide with the slower ones in the course of their fall.

In NAUA and NAUAHYGROS only the Brownian and gravitational agglomeration processes are modeled. Turbulent agglomeration is not treated since the turbulent flow in a containment atmosphere is difficult to evaluate. The effects of electric and radiation fields are also not considered.

The agglomeration kernels are given by the following expressions:

Brownian:

$$K_{ij}^{B} = (4kT/6\mu)(r_i+r_j)[C(r_i)/r_i+C(r_j)/r_j] \quad i\neq j \tag{8}$$

$$K_{ii}^{B} = (4kT/3\mu)C(r_i) \quad i=j \tag{8a}$$

These expressions are valid for particles that are large compared with the gas mean free path ($r > 10^{-5}$ cm).

Gravitational ($r_i < r_j$):

$$K_{ij}^{G} = \frac{1}{2}\frac{(r_i)^2}{(r_i+r_j)^2}(r_i+r_j)\left[v_s(r_j)-v_s(r_i)\right] \tag{9}$$

where the term $1/2\ (r_i/(r_i+r_j))^2$ is the collection efficiency.

Using equation (3), one obtains

$$K_{ij}^{G} = g\rho_p r_i^2/9\mu \bullet [C(r_j)r_j^2 - C(r_i)r_i^2] \tag{10}$$

Derivations of equation (9) are given in refs. (7) and (8). The collection efficiencies differ by a factor of three; in NAUAHYGROS the Pruppacher-Klett[8] form is used.

Size Splitting

It was noted above that the coagulation of two particles of radii r_i and r_j will result in a particle whose radius r_{ij} is equal to $(r_i^3 + r_j^3)^{1/3}$. Since NAUA uses a discrete mesh of size bins, the radius r_{ij} will in general not coincide with one of the bins, but will fall between two of them. NAUA "splits" the coagulated particle.

If a coagulated particle has a radius r that falls between bins m and n, where n = m+1, the number of particles in bin m is increased by $(r_n - r)/(r_n - r_m)$, and the number in bin n is increased by $(r - r_m)/(r_n - r_m)$. The mass in bin m is increased by $\frac{4\pi}{3}\rho_p r^3 (r_n - r)/(r_n - r_m)$, while the mass in bin n is increased by $\frac{4\pi}{3}\rho_p r^3 (r - r_m)/(r_n - r_m)$. This procedure correctly conserves the total number of particles (one) and the total mass $4\pi\rho_p r^3/3$, but does not, for example, yield a correct result for the settling rate of the coagulated particle, which is proportional to $\rho_p r^2$. This is a necessary compromise in view of the finite number of discrete size bins. If the coagulated particle has a radius larger than the largest size bin, it is totally assigned to the largest bin.

The size-splitting procedure is also used in NAUA for particles whose size changes as a result of condensation or evaporation of steam. As mentioned previously, in NAUAHYGROS the moving boundary scheme makes splitting unnecessary for condensational size changes.

The Complete Model Equation

Combining all the processes discussed above, the following integro-differential equation results for a continuous size distribution:

$$\frac{dn(r,t)}{dt} = S(r,t) - L(r,t) - (v_s(r) + v_d(r) + v_{sf})\frac{A}{V}n(r,t)$$

$$+\int_0^{\frac{r}{\sqrt[3]{2}}} K(R',r')n(R',t)n(r',t)\frac{r^2}{(R')^2}dr'$$

$$-n(r,t)\int_0^{\infty} K(r,r')n(r',t)dr' \qquad (11)$$

where $R' = (r^3 - r'^3)^{1/3}$, $S(r,t)$ is the aerosol source term, and $L(r,t)$ $(= \alpha_L n(r,t))$ represents the loss by leakage (α_L is the fractional volumetric leak rate at time t).

Equation (11) must be solved numerically - for this reason the size distribution function n(r,t) is approximated by a set of discrete quantities $n_i(r_i, t)$, and equation (11) is replaced by:

$$\frac{dn(r_k,t)}{dt} = S(r_k,t) - (v_d(r_k) + v_s(r_k) + v_{sf} + \alpha_L(r_k))n(r_k,t)\frac{A}{V}$$
$$-\sum_{i=1}^{N}(1 - \frac{1}{2}\delta_{ik})K(r_i,r_k)n(r_i,t)n(r_k,t) + \frac{1}{2}\sum_{i=1}^{N}\sum_{j=1}^{N}K(r_i,r_j)\beta_{ij}^k n(r_i,t)n(r_k,t) \qquad (12)$$

where δ_{ik} is the Kronecker δ; δ_{ik}=1 if i=k, δ_{ik}=0 otherwise.

β_{ij}^k is determined by the size-splitting formulae previously discussed if coagulation between particles in bins i and j results in a particle in bin k; otherwise it equals zero.

In NAUA there are additional terms in equations (11) and (12) that represent gains and losses due to condensational growth. These terms no longer appear in NAUAHYGROS because of the moving bins approach - it should be kept in mind that condensation does not result in a change in the number of particles in a given size bin, only a change in the radius of the particles. Nevertheless, condensational growth must still be calculated in order to determine the number and mass size distributions in terms of the new radii.

Condensation on Particles

The model used in mod-4 and mod-5 for condensational growth on particles is the Mason equation[9]. It can be written in the following form[8]:

$$r\frac{dr}{dt} = \frac{S - \exp\left(\frac{2\sigma M_w}{\rho_w RTr}\right)}{\frac{L\rho_w}{kT}\left(\frac{LM_w}{RT} - 1\right) + \frac{\rho_w RT}{M_w D(p_{sat}(T)}} \qquad (13)$$

where
- S = bulk steam saturation ratio
- σ = surface tension of water on the particle or droplet
- ρ_w = density of water on the droplet
- L = latent heat of water on the droplet
- k = heat conductivity of water vapor
- M_w = molecular weight of water (18.016 g/mol)
- D = binary diffusion coefficient of the steam/air atmosphere
- p_{sat} = steam saturation pressure
- R = gas constant (0.8314E8 erg/gm-deg.)
- T = absolute temperature

From equation (13) it is apparent that for a given S greater than one (supersaturation) there is a critical radius $r_c = 2\sigma M_w/\rho_w RT\ln(S)$ below which condensation cannot take place. For S = 1, the critical radius is infinite, so condensation cannot take place on any size particle, while for S < 1, dr/dt is negative, so only evaporation can occur. Equation (13) is derived on the basis of various assumptions, including that rdr/dt is small compared with $kT/\rho_w L$. In the derivation it is also assumed that the temperature difference between the droplet and the atmosphere is small and that the steam partial pressure is small compared with the total pressure.

Hygroscopic Particles

For hygroscopic particles equation (13) must be modified. The exponential term in equation (13) represents the lowering of the saturation pressure at the surface of the droplet due to the curvature of the droplet, the so-called Kelvin effect. If the particle is hygroscopic or soluble, there is an additional decrease in the saturation pressure due to the solute effect. This

leads to an additional term in the numerator of equation (13):

$$r\frac{dr}{dt}=\frac{S-A\exp\left(\frac{2\sigma M_w}{\rho_w RTr}\right)}{\frac{L\rho_w}{kT}\left(\frac{LM_w}{RT}-1\right)+\frac{\rho_w RT}{M_w Dp_{sat}(T)}} \qquad (14)$$

where A is the water activity of the droplet aqueous solution. For dilute solutions with many hygroscopic species, the water activity can be determined by the following equation [3]:

$$A=\sum_i \frac{1}{1+Q_i M_w m_i} \qquad (15)$$

where Q_i is the van't Hoff factor and m_i is the molality of each hygroscopic material dissolved in the droplet.

In the analysis of the growth of hygroscopic particles it is important to know the chemical composition of these particles, because the hygroscopic growth of particles depends strongly on the amount and properties of water soluble matter in the particle. Most inorganic salts, e.g. CsI, when exposed to increasing humidity, will dissolve at a certain relative humidity (deliquescent point) which depends on the properties of the electrolyte and the system temperature. At the deliquescent point particles which contain these compounds will form saturated droplets and undergo at the same time an abrupt increase in size. As the humidity increases further, the droplets will grow and become more dilute. When the humidity decreases, the droplets become smaller and the particles recrystallize at a relative humidity (RH) considerably lower than their deliquescent point. In the case where particles also contain insoluble compounds, these two humidities are near each other. On the other hand, a few electrolytes, such as CsOH and NaOH, will form an aqueous solution and start to absorb water at a very low RH (< 10 %). Particles consisting of these substances grow smoothly with increasing RH.

Equation (14) has a number of interesting features. First, there is still a critical radius, but all particles larger than this critical radius will have an equilibrium radius when RH is less than one. This equilibrium radius is reached very rapidly when RH is less than about 0.99. In NAUAHYGROS immediate equilibrium is assumed when RH is less than 0.99. Also the rate at which the particles grow is not very critical, because the final size at given RH is determined by the equilibrium radius and not by the rate steam is condensing on the particles. In contrast where only nonhygroscopic particles are present there is no equilibrium radius, and it is the condensation rate which determines the particle size.

Equation (15) is valid only for dilute solutions (i.e. high RH) and it will give conservative (too large) values at low RH. However, this is not important in reactor accident applications, because at low RH even hygroscopic particles do not grow much.

The parameters that are required to evaluate the water activity include the van't Hoff factor and the mole fraction of each hygroscopic material in the particle (equation (15)). In addition the minimum relative humidity at which deliquescence occurs must be specified. The van't Hoff factor and minimum relative humidity are user-input. Mole fractions are updated by the code at each time step. The water activity is also a function of temperature and concentration. For a given initial dry mass of hygroscopic material, the concentration will change as water condenses on the particle, thus A is continuously changing (except at the equilibrium size or concentration).

Wall Condensation Models

As shown above, condensation on particles is driven by the relative humidity (saturation ratio) in the containment atmosphere. To calculate the relative humidity and update it at each time step, NAUAHYGROS tracks the mass balance of both water and steam, and combines this information with the current atmosphere temperature and pressure. A correct evaluation of the relative humidity can only be done if condensation of steam on the containment wall is taken into account. In addition the Stefan flow diffusiophoretic effect on particle deposition depends directly on the wall condensation rate (equation (5)).

Calculation of condensation on a wall is an old problem in heat transfer and fluid mechanics, and a number of theoretical models and engineering correlations have been developed to handle it. In mod-4 there is no wall condensation model. In mod-5 a simple approach is taken: the wall condensation rate is set equal to the steam injection rate, and the saturation ratio then varies only because of condensation on the particles and changes in atmosphere temperature. In NAUAHYGROS, as has been mentioned, the wall condensation and particle condensation rates are calculated independently, as functions of the saturation ratio or other thermohydraulic quantities at the end of the preceding time step, then the saturation ratio is recalculated (updated) for use in the next time step, using mass balance to determine the new steam density in the atmosphere.

Four models are available in NAUAHYGROS for the wall condensation calculation. The user selects one by the choice of an input parameter MODEL (= 1-4).

Combined Nusselt-Heat Transfer/Mass Transfer Analogy

MODEL = 1 selects a combination of the classical Nusselt equation[10] on the liquid (condensate) side and the well-known equation that results from the heat transfer/mass transfer analogy[11] on the vapor side. The Nusselt equation is:

$$W = \frac{4}{3}\left\{\frac{k(T_{gas} - T_f)}{L}\right\}^{3/4} \bullet \left\{\frac{\rho_w(\rho_w - \rho_s)}{4\mu_w H}\right\}^{1/4} \qquad (16)$$

The heat transfer/mass transfer analogy equation is:

$$W = \frac{DM_w p}{RT_f \delta_v p_{am}}(p_{ai} - p_{a\infty}) \qquad (17)$$

In equations (16) and (17), in addition to previously defined symbols,

T_f = condensate film surface temperature
μ_w = viscosity of liquid water
ρ_s = density of steam
H = height of the wall
p = total pressure in the containment atmosphere
p_{ai} = air partial pressure at the condensate surface
p_{am} = logarithmic mean of the air partial pressure through the boundary layer at the condensate surface
$p_{a\infty}$ = air partial pressure in the bulk atmosphere
δ_v = mass transfer boundary layer thickness

δ_v is calculated according to the convective regime (forced or natural) as a function of the Grashof and Schmidt numbers, which are internally calculated.

The program solves equations (16) and (17) iteratively to find a value of the condensate film surface temperature T_f that makes the two condensation rates equal.

This method is used in a number of reactor and severe accident thermohydraulic codes such as MAAP, MELCOR, and CONTAIN. It is therefore the recommended choice when the input thermohydraulic data to NAUAHYGROS are obtained from calculations using these or similar codes.

Combined Nusselt - Kinetic Theory Model

MODEL = 2 selects the Nusselt - Kinetic Theory model[12]. This model has been shown [4] to give the best overall agreement with aerosol and condensation data from the LACE experiments[13]. The model combines the Nusselt equation (16) with the following equation derived from kinetic theory considerations[4,8,14]

$$W = \frac{\alpha_c}{(1-0.5\alpha_c)}\left(\frac{M_w}{2\pi R T_\infty}\right)^{\frac{1}{2}} p_{sat}(T_\infty)\left[S - \frac{p_{sat}(T_f)}{p_{sat}(T_\infty)}\right] \qquad (18)$$

where $S = p_s/p_{sat}(T_\infty)$, p_s is the steam partial pressure at the condensate surface (assumed equal to the bulk steam partial pressure), $p_{sat}(T_f)$ is the steam saturation pressure at the surface temperature of the condensate film, and $p_{sat}(T_\infty)$ is the bulk steam saturation pressure.

As in model 1, the film surface temperature that gives equal condensation rates from equations (16) and (18) is found by iteration. In these models the vapor side condensation rate is driven by the relative humidity S. (Equation (17) can also be cast into a form similar to equation (18) if the steam partial pressure is small compared with the total pressure.)

In equation (18) the constant α_C is the so-called condensation coefficient. Its value is somewhat uncertain. It has been measured for both "renewing" and "non-renewing" surfaces by a number of investigators [8]. For renewing surfaces

the experimental values tend to be close to one; for non-renewing surfaces they are much smaller. A thin condensate film represents a non-renewing situation (there is no fresh supply of water from deep in the condensate film), thus a value of 0.036 has been chosen for NAUAHYGROS as a reasonable average of the experimental values. However, condensation rates calculated in NAUAHYGROS are quite insensitive to the value of α_C, since the relative humidity "adjusts" to compensate for the difference in α_C[4].

One of the basic assumptions underlying equation (18) is that the steam partial pressure is constant right up to the condensate surface, i.e., S in equation (18) refers to the bulk steam partial pressure and there are no boundary layer effects. This might be expected to be the case in the presence of turbulence, and is probably the reason that this model gives the best agreement with experimental data, as mentioned earlier. It is recommended that

this model be used when experimental thermohydraulic data are supplied.

Uchida - Tagami Correlation

MODEL = 3 employs the Uchida - Tagami correlation[15,16]. The Uchida and Tagami correlations are quite similar, and the Tagami form is used in NAUAHYGROS. The correlation gives the heat transfer coefficient to the wall, h, in terms of the steam-to-air mass ratio, f_v. Multiplication by the temperature difference ($T_\infty - T_W$) and division by the latent heat of condensation yields the mass condensation rate. The correlation is:

$$h = 11.4 + 284\left(\frac{f_v}{1-f_v}\right) \qquad (19)$$

The units are SI; NAUAHYGROS internally makes the conversion to cgs units.

This choice is recommmended when the source of the input thermohydraulic data is a code that uses it.

Direct Use of Input Heat Flux Data

MODEL = 4 uses input heat fluxes to the wall to directly infer the wall condensation rate (both models 3 and 4 assume that sensible heat transfer to the wall is negligible compared with condensation heat transfer).

This model is recommended when heat transfer rates are available, either from experiment or a code calculation. The input heat fluxes must be in cgs units. If condensation rates are available from an external source, they can be used by converting them to heat fluxes, which can then be input. The choice of MODEL = 4 will then recover the condensation rates directly.

Stefan Flow Diffusiophoresis

Equation (5) gives the diffusiophoretic deposition velocity in terms of the wall condensation rate. NAUAHYGROS calculates the condensation rate W and the steam density and mole fraction at each time step, using the chosen wall condensation model to obtain W.

III. Validation of NAUAHYGROS

The principal validation of NAUAHYGROS has been against the LACE experiments LA-2, LA-4, and LA-6[13]. These tests were performed in a large (850 m^3) tank. Hygroscopic (CsOH) and non-hygroscopic (MnO) aerosols were introduced over an approximately

one hour period at a rate that resulted in maximum aerosol concentrations of 1-10 g/m^3, similar to expected accident concentrations. Steam was also injected so as to provide saturated or near-saturated conditions. (The steam and aerosol injection protocols and leakage conditions were somewhat different in the three tests). The tests were well instrumented to measure aerosol parameters such as suspended aerosol mass as a function of time, integrated settled, plated, and leaked mass, and size distribution parameters. In addition, extensive thermal hydraulic measurements were made, which provided benchmark and input data for the code calculations.

Figure 1 shows the measured and calculated suspended mass for the three tests. It should be noted that at late times, when the suspended aerosol concentrations are of the order of two orders of magnitude or more below their peak values, the calculated and measured concentrations can disagree considerably. However, the impact of this disagreement on integrated aerosol quantities is negligible.

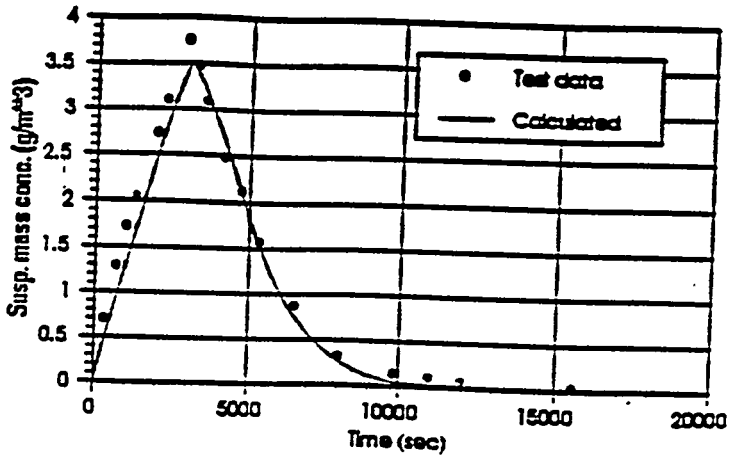
Table 1 shows the integral data for tests LA-2 and LA-4 (integral data for LA-6 were not taken). It should be emphasized again that calculations that do not properly account for diffusiophoresis and steam condensation on the particles fail quite badly in predicting the aerosol behavior in these experiments. (Thermophoresis was not significant in these tests). On the other hand these effects might not be as important in other situations, depending on the thermal hydraulic conditions.

There is a slight inconsistency between the measured suspended mass concentrations in LA-2 shown in Figure 1 and the measured integrated leakage for this test. The leakage derived from the measured suspended mass data agrees better with the NAUAHYGROS results than the measured leakage does.

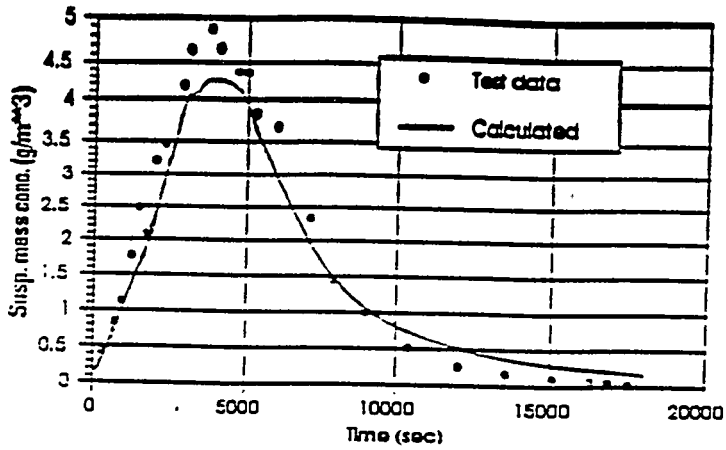
Table 1. Integrated settled, plated, and leaked aerosol, Tests LA-2 AND LA-4

	Settled (g)	Plated (g)	Leaked (g)
LA-2, test data	1973(±10%)	449(±20%)	1515(±15%)
LA-2, calculated	2366	334	1197
LA-4, test data	4490(±10%)	532(±20%)	108(±30%)
LA-4, calculated	4437	609	96
LA-4, calc. *w/o steam*	4811	10	321

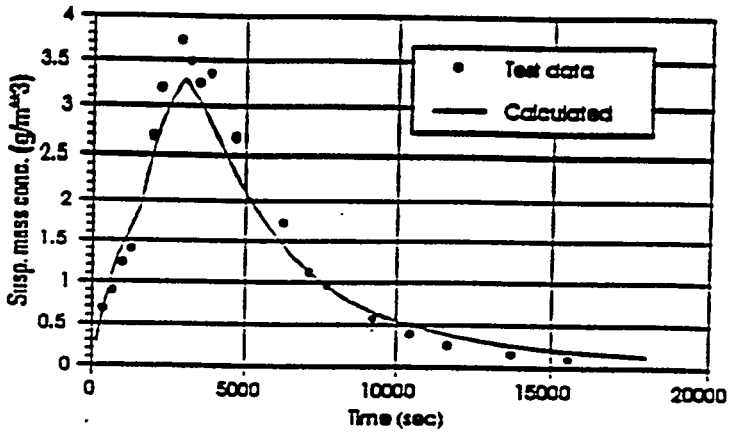
LA-2
NAUAHYGROS calculation, line



LA-4
NAUAHYGROS calculation, line



LA-6
NAUAHYGROS calculation, line



Standard deviations on the data points range from ± 5% to ± 60%.

Figure 1. Suspended Aerosol Concentrations in the LACE Tests.

IV. Summary

NAUAHYGROS is a code that simulates aerosol behavior in a nuclear power plant containment following a severe accident. It is an extension of the much used code NAUA that includes several important improvements in both the modelling of the aerosol behavior and the code architecture. Among the principal new features of the code are the ability to treat steam condensation effects, both on the aerosol particles (including hygroscopic species) and on internal heat sink surfaces, e.g. the walls of the containment; the ability to introduce up to 50 aerosol species at arbitrary times and with arbitrary (lognormal) size distributions; and a "moving bin" feature that substantially eliminates numerical diffusion effects in the treatment of steam condensation on the particles. Most of the important aerosol removal mechanisms are modelled; those that are not, e.g. thermophoresis and deposition mechanisms due to turbulence, can be readily added as new subroutines.

The code has been validated against the LACE containment experiments, giving satifactory agreement with suspended mass concentrations, integrated settled, plated, and leaked masses, and other aerosol properties (e.g. ammd). The inclusion of steam condensation effects, particularly hygroscopicity, was essential in order to obtain this agreement.

A more complete description of the code is given in ref.17, which is also a user's manual.

Acknowlegements

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DISCUSSION

WREN: Are the temperatures of walls and gas phase kept constant with time in the code? Does the aerosol behavior affect these temperatures with time? If so, how can these changes be modelled?

SHER: Temperatures are not constant - they are input data. NAUA (or NAUAHYGROS) does not calculate the effect of aerosols on temperatures. T/H codes, which can be the source of the temperature tables, may do so. There is no code that fully couples aerosols and T/H behavior.

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Experimental study on the particles deposition in the sampling duct

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Abstract

A high standard of protection against the harmful effects of radioactive aerosol dissemination requires a measurement, as representative as possible, of their concentration. This measurement depends on the techniques used for aerosol sampling and transfer to the detector, as well as on the location of the latter with respect to the potential sources. The aeraulic design of the apparatus is also an important factor.

Once collected the aerosol particles often have to travel through a variably shaped duct to the measurement apparatus. This transport is responsible for losses due to the particles deposition on the walls, leading to a distortion on the concentration measurements and a change in the particle size distribution.

To estimate and minimize measurement errors it is important to determine the optimal transport conditions when designing a duct ; its diameter and material, the radius of curvature of the bends and the flow conditions must be defined in particular.

This paper presents an experimental study in order to determine, for each deposition mechanism, the retained fraction, or the deposition velocity for different flow regimes.

This study has pointed out that it exists a favourable flow regime for the particle transport through the sampling ducts (2 500 < Re < 5 000). It has been established, for any particle diameters, equations to predict the aerosol penetration in smooth-walled cylindrical metal ducts.

I. Introduction

A high standard of protection against the harmful effects of radioactive aerosol dissemination requires a measurement as representative as possible of their concentration. This measurement is often realised after a transport, in a variably shaped duct, of the collected particles. The transport is responsible for losses due to the particles deposition on the walls, leading to a distortion on the concentration measurement and a change in the particle size distribution.

The deposition mechanisms of the particles in a duct are more or less important according to the particle size and flow regimes. The main mechanisms considered are as follows :

- brownian diffusion
- sedimentation
- impaction
- turbulent diffusion
- electrostatic precipitation

The effect of each mechanism can be quantified by the retained fraction F_R, expressed by the ratio between the concentrations measured downstream (C) and upstream (C_o) from the duct, i.e :

$$F_R = 1 - \frac{C}{C_o}$$

In laminar flow, for each case considered, an expression exists for the calculation of the retained fraction F_R.

In turbulent flow we use the notion of deposition velocity, K, considered normal and directed towards the wall.

$$K = \frac{n}{C}$$

with :
n : deposition rate
C : average aerosol concentration

The fraction hold back is then given by the general expression :

$$F_R = 1 - \exp\left(- \frac{aKL}{\bar{U} D} \right)$$

with :
L : pipe length
D : pipe diameter
U : mean flow velocity
a = 4 : deposition on the entire surface of the duct
a = 2 : deposition on a half-surface of the duct

This paper proposed some expressions for the retained fraction F_R or the deposition velocity K for each of the mechanisms considered and for different flow conditions. These expressions are derived from an experimental study [1] conducted on a cylindrical experimental duct made up of 8 assembled metallic elements. The use of monodispersed fluorescein aerosols allows a detailed measurement of the retained fraction which is determined by washing of the downstream elements and filter and by fluorimetric measurement of the solutions obtained. Certain elements may be replaced by tubes with non-conducting walls or of different diameters. The test circuit uses, to produce monodispersed particles, a Differential Mobility Analyser (DMA), in the range 0.01 to 1 μm and a Vibrating Orifice Generator (VOG) in the range 1 μm to 10 μm. The aerosol electric charge, measured by an electrometric system, is varied by ionising and neutralising systems. A bubbler gives a humidity variation of about 10 to 50 % R.H. The air is pumped through the circuit at flow rates corresponding to Reynolds numbers ranging from 0 to 10 000 for a pipe diameter of D = 2.8 cm.

II. Deposition mechanisms

Brownian diffusion

Fine particles, like gas molecules, are subject to Brownian movement owing to their thermal agitation, the particles can strike the wall of the duct and remain fixed.

- ***Laminar flow***

The fraction retained in a duct of length L may be calculated from the equations established by Townsend, Gormley, Kennedy, Demarcus [2].

If $\alpha \geq 0.01$; $F_R = 1 - 0.819 \exp(-3.657\,\alpha) - 0.097 \exp(-22.3\,\alpha) - 0.032 \exp(-57\,\alpha)$

If $\alpha < 0.01$; $F_R = 2.56\,\alpha^{2/3} - 1.2\,\alpha - 0.177\,\alpha^{4/3}$.

with : $\alpha = \dfrac{\pi\ Di\ L}{Q}$ $\quad Di = \dfrac{TC}{3\ \pi\ \mu\ dp}$

Di : aerosol diffusion coefficient
Q : flow rate
dp : particle diameter
μ : dynamic viscosity
C : Cunningham factor
K : Boltzmann constant

▸ *Turbulent flow*

If the diffusion of fine particles is assumed to take place through the laminar sub-layer of thickness, δ, and if the concentration gradient in this sub-layer is assumed linear, we get :

$$K = \frac{Di}{\delta}$$

with $\delta = A.\frac{\nu}{U_*}$, where $5 < A < 11.5$ depending the author concerned, hence

U_* : friction velocity : $U_* = \bar{U}\left(\frac{f}{2}\right)^{1/2}$

f : friction factor : $f = \frac{0.316}{4\ (Re)^{0.25}}$ for smooth turbulent flow

ν : kinematic viscosity

A single equation, relating dimensionless values, is derived from the experimental results :

$$K_+ = 0.123\ Sc^{-3/4}$$

with :

$Sc = \frac{\nu}{Di}$: Schmidt number

$K_+ = \frac{K}{U_*}$: dimensionless deposition velocity

The experiments were conducted on a metallic vertical tube (D = 2.8 cm) with monodispersed aerosols (0.025 μm < dp < 1.1 μm) in different flow conditions (3 300 < Re 10 000).

From these experiments, the dimensionless deposition velocity is calculated and ploted versus Schmidt number in figure 1. In this figure the Wells and Chamberlain equation ($K_+ = 0.2\ Sc^{-2/3}$) is also represented.

The retained fraction F_R is determined from K and a = 4 with the general expression given in introduction.

Deposition due to combined gravitational and centrifugal forces

In the case of an horizontal bend of curvature radius R, a single expression of K may be written, accounting for the relative accelerations of gravitational and centrifugal forces :

$$K = \frac{2}{\pi} \cdot \tau \left[\frac{g^2 + \left(\frac{\bar{U}^2}{R} \right)^2}{g + \frac{\bar{U}^2}{R}} \right]$$

The case of the vertical bend is more difficult to estimate since the angle between the vectors of the forces varies with the position of the particle in the bend, whereas this value is always $\pi/2$ when the bend is horizontal.

Turbulent diffusion

The gas eddies of a turbulent flow may cause certain particles to be projected towards the wall through the boundary layer. This turbulent diffusion is independant of the orientation of the duct and can sometimes be the only phenomenon responsible for deposition in vertical tubes.

Liu et Agarwal [4] have given an expression for the dimensionless deposition velocity :

if $\tau_+ < 15$ $K_+ = 6.10^{-4} \tau_+^2$

if $\tau_+ \geq 15$ $K_+ = 0.13$

where τ_+ : dimensionless relaxation time : $\tau_+ = \tau \frac{U^2_*}{\nu}$

Experiments were developed in a metallic vertical tube (D = 2.8 cm) with monodispersed aerosols ($0.6\ \mu m < dp < 9\ \mu m$) in different turbulent flow conditions ($3\,400 < Re < 10\,000$). Dimensionless deposition velocity deduced from these experimental results are presented in the figure 2 and compared with the expression given by Liu and Agarwal. This results pointed out a lower limit for this expression when $\tau_+ < 0.1$.

Deposition of charged particles due to the image force

We know the velocity of attraction, due to the image force, of a particle of charge p.e at a distance y from a wall likened to a plane :

$$\frac{dy}{dt} = \frac{- Di\ (p.e)^2}{KT\ 4y^2} \cdot \frac{1}{4\ \pi\ \epsilon_o}$$

This velocity, normal to the wall, is the deposition velocity.

Tests carried out for variable flow conditions 1 200 < Re < 10 000 have shown that deposition is minimal at the beginning of turbulent flow, Re ~ 2 500, then varies with the thickness of the laminar sub-layer δ.

If we consider the particles to undergo a constant attraction velocity at distance $y = \delta$ from the wall, we get :

$$K = \frac{Di.\ (pe)^2}{T\ 4\ \delta^2} \cdot \frac{1}{4\ \pi\ \epsilon_o}$$

Tests with 0.6 to 9 μm particles diameters of variable charge and for variable Re have yielded an approximate δ value :

$$\delta = 2.2\ \frac{\nu}{U_*}$$

This distance is inside the laminar sub-layer, of thickness :

$$5\ \frac{\nu}{U_*} < \delta < 11.5\ \frac{\nu}{U_*}$$

III. Experimental studies

▸ Influence of the Reynolds number

The variation of the particle deposition in any element of a complex circuit were studied for different flow conditions and particle sizes in tubes of 2.8 cm in diameter.

Figures 3, 4 and 5 plot the variations in the 4 μm and 9 μm particle fractions held back in the following elements :

- Straight horizontal tube : in laminar flow the measurements verify the NATANSON equation since Re = 2 500 ; in turbulent flow condition, the experiments show that the Fuchs equation is not verified because it is necessary to take into account the effects of the turbulent diffusion.

- Straight vertical tube : the measured deposits, due entirely to turbulent diffusion, line up quite well with the Liu and Agarwal equation.

- Horizontal and vertical bends for which the radius of curvature is equal to 8 times the pipe diameter ; the fraction retained is obviously greater in the horizontal bends and shows in both cases, vertical and horizontal, a minimum when the Reynolds number is around 3 000.

Figure 6 expresses the fraction retained in the whole experimental circuit which is composed of pipes of 2.8 cm in diameter with horizontal linear tube of 6 m long, vertical linear tube of 2.2 m long, two horizontal bends of 90° angle and two vertical bends of 90° angle. This figure shows that it exists for the largest particles a favourable flow conditions around Re = 3 500 to reduce the particle deposition.

▸ *Influence of the particle diameter*

The curve of figure 7, corresponding to the entire complex circuit, reveals a minimum deposition around 0.25 μm. As the particle size increases from 0.01 to 10 μm the molecular regime gives way to the inertial regime, the particle are less and less sensitive to the Brownian motion as their weight increases, while gravitational and centrifugal forces and that of diffusion due to flow eddies become stronger.

IV. Conclusion

Any transport gives rise to losses, so ducts should only be used when absolutely necessary. In such cases, the loss of aerosol can be reduced to a minimum.

As a general rule, the deposition in a transport duct involving elements of different shapes (straight, curved) and orientations (horizontal, vertical) is least when the flow is slightly turbulent : 2 500 $<$ Re $<$ 5 000.

This applies especially to large particles, and is also a factor in reducing deposition due the electric charge (image force) of aerosols. For the finest aerosol particles, it is safe to use a more turbulent flow, the deposition rate decreasing slowly as the Reynolds number rises. When the tube diameter is small (D $<$ 20 mm) it is essential to choose a flow rate such that Re = 2 500 in order to avoid turbulent diffusion, which would then become a very highly influential factor.

It is advisible to use as large pipe diameter as possible while keeping the favourable flow conditions and reduce as much as possible the horizontal tubes and curves. The bends must have a radius of curvature greater than 8 times the duct diameter and to minimize electrical effects it is better to use metallic tubes electrically conducting and earthed, because the triboelectrical effect on non-conductive walls induces very high retention of charged particles.

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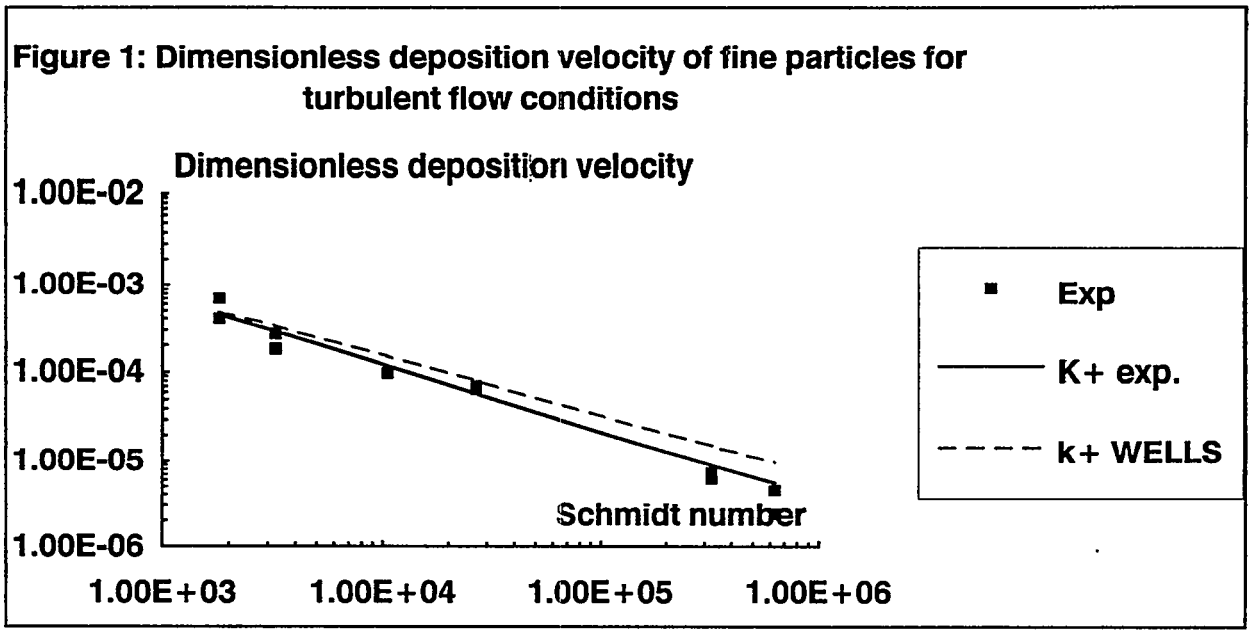


Figure 1: Dimensionless deposition velocity of fine particles for turbulent flow conditions

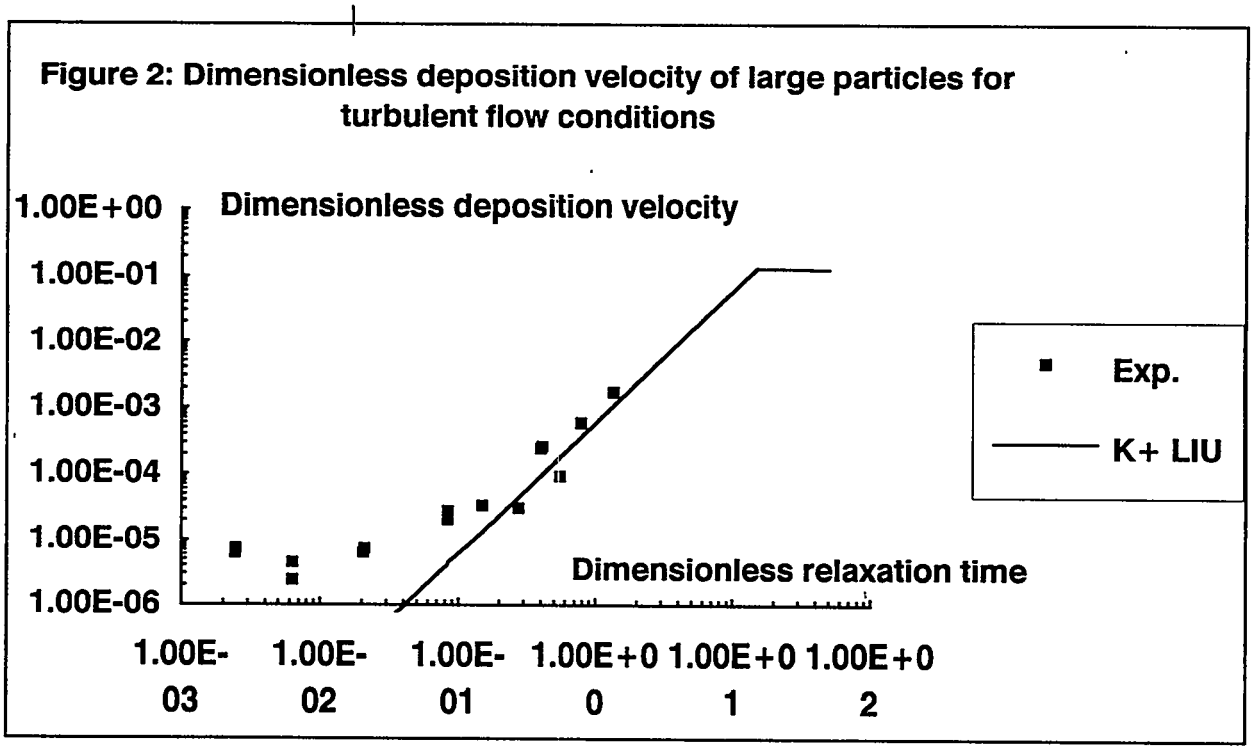


Figure 2: Dimensionless deposition velocity of large particles for turbulent flow conditions

Figure 3: Particles deposition in horizontal duct

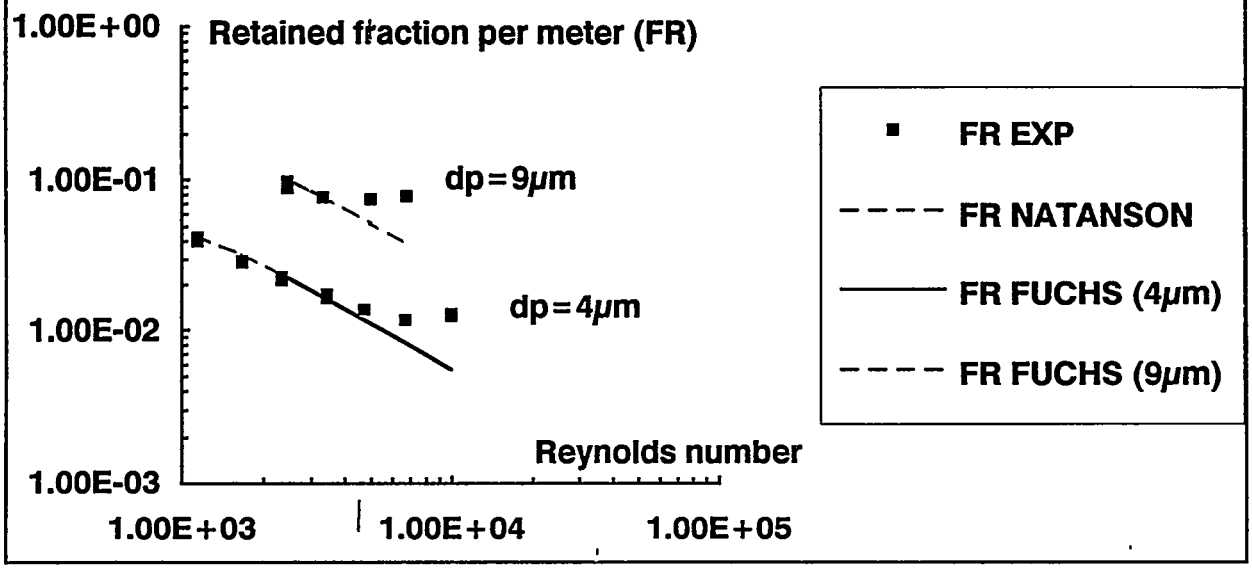


Figure 4: Particles deposition in vertical duct

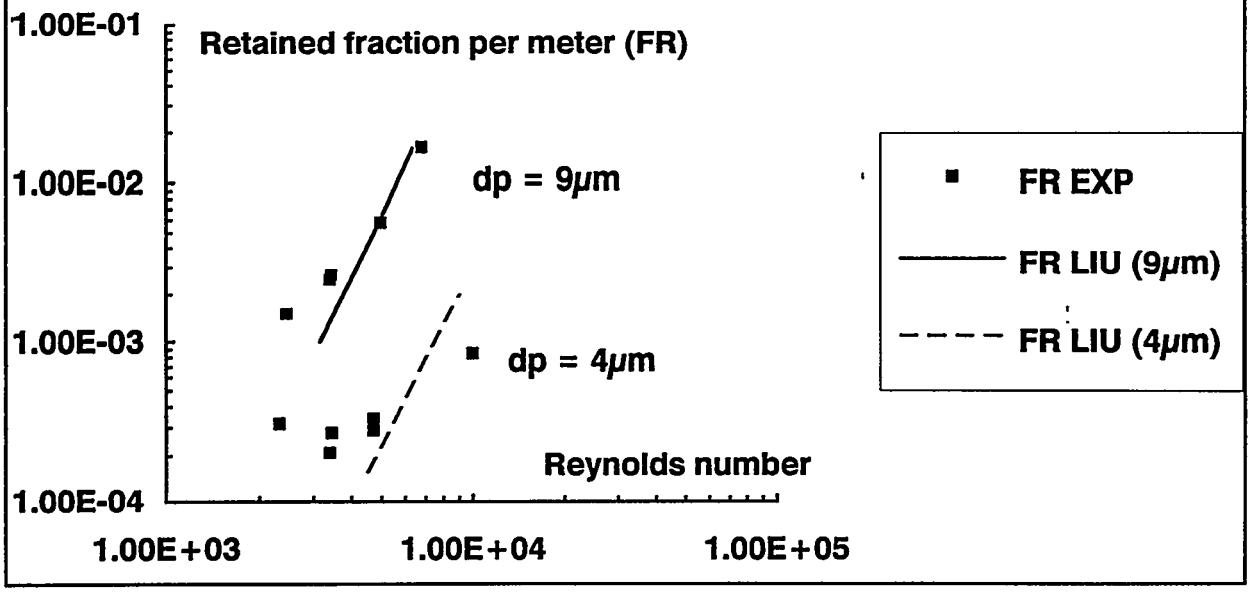


Figure 5: Particles deposition in 90° bends

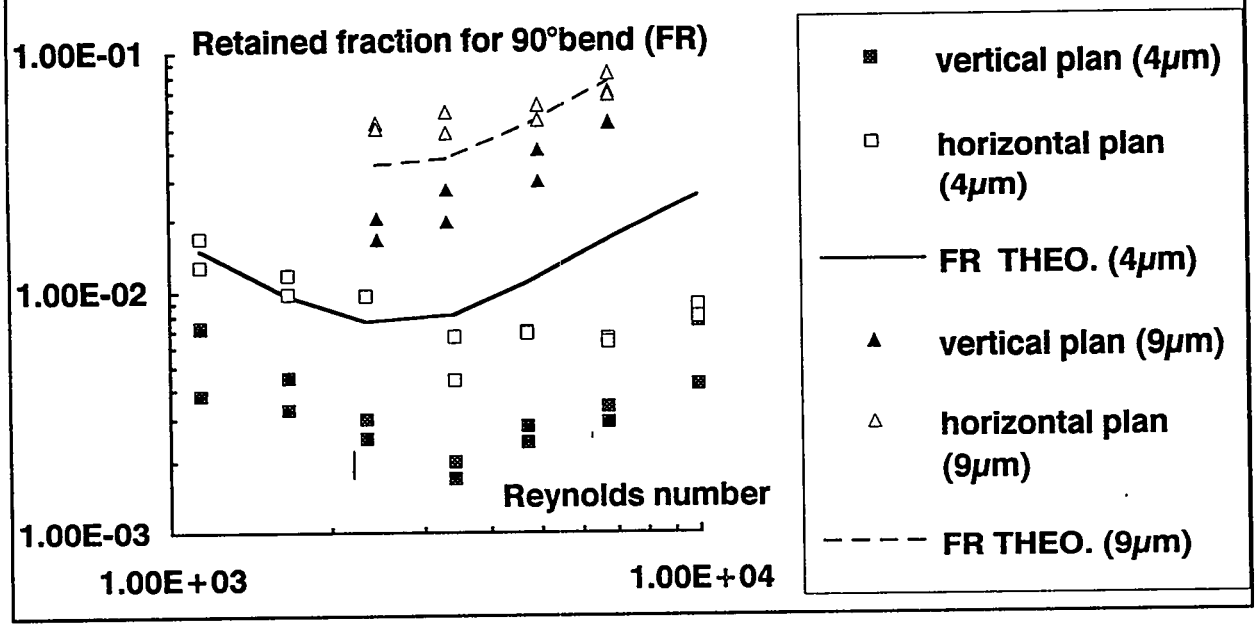
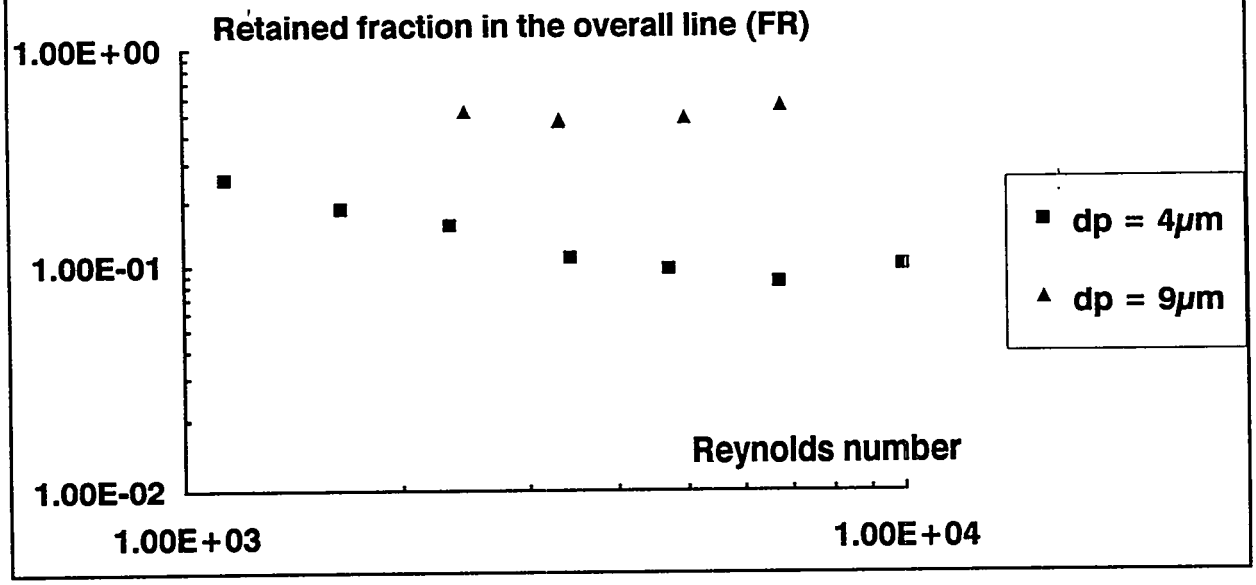
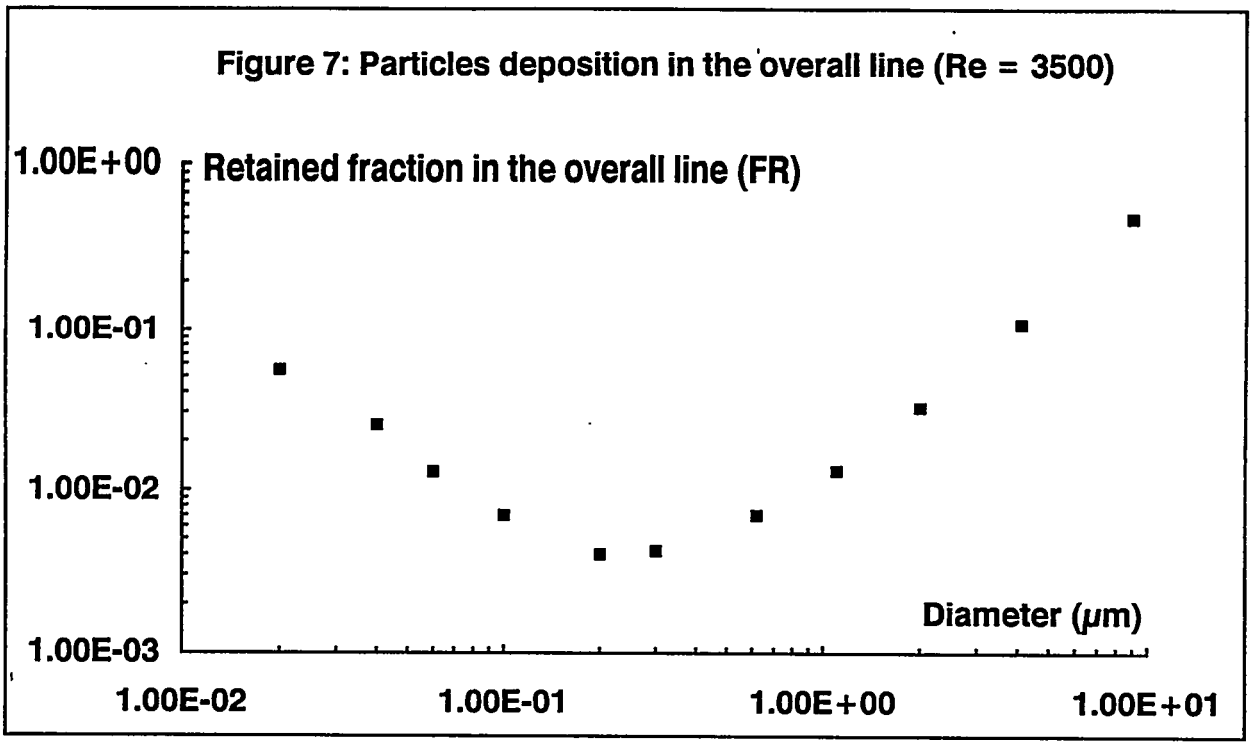


Figure 6: Particles deposition in the overall line versus Reynolds number





DISCUSSION

SHER: Were these experiments done dry, no steam?

VENDEL: Experiments were made in dry conditions and with variable relative humidity.

SHER: What was the aerosol?

VENDEL: The aerosol was sodium fluorecein.

SHER: In the LACE experiments, they did retention measurements in long pipes, and found that the results varied considerably, from a few percent to as much as 90% retention, depending on the nature of the aerosol (composition, hygroscopicity). I wonder if you observed anything like that. These experiments were in steam conditions, i.e., high relative humidity.

VENDEL: Retention does not depend strongly on the nature of the aerosol when the conditions are dry air. In the case of hygroscopic particles in wet conditions, the rate of deposition can be affected, because the particle sizes vary according to the retention time, and the rate of deposition must be evaluated during the entire time.

CLOSING COMMENTS OF SESSION CO-CHAIRMAN BELLAMY

In this session we have learned where the NRC stands with its revised accident source terms. We have heard a little bit about the effect of EPA regulations on monitoring needs at the Westinghouse Hanford facility. We have discussed emergency planning in Germany, aerosol behavior inside containments after severe accidents, and, finally, some concerns about particle deposition inside sampling ducts. We want to thank the presenters and the commenters for their interest this morning.

SESSION 14

AGING EFFECTS ON AIR CLEANING COMPONENTS

Wednesday: July 27, 1994
Co-Chairmen: R. R. Weidler
W. Bergman

OPENING COMMENTS OF SESSION CO-CHAIRMAN WEIDLER

FILTER-ADSORBER AGING ASSESSMENT
W. K. Winegardner

CRITERIA FOR CALCULATING THE EFFICIENCY OF HEPA FILTERS DURING AND AFTER DESIGN BASIS ACCIDENTS
W. Bergman, M. W. First, W. L. Anderson, H. Gilbert, J. W. Jacox

PERFORMANCE OF HEPA FILTERS UNDER HOT DYNAMIC CONDITIONS
D. P. Frankum, G. Costigan

PRELIMINARY STUDIES TO DETERMINE THE SHELF LIFE OF HEPA FILTERS
H. Gilbert, J. K. Fretthold, F. Rainer, W. Bergman, D. Beason

EFFECTS ON THE EFFICIENCY OF ACTIVATED CARBON ON EXPOSURE TO WELDING FUMES
D. Ghosh

CLOSING COMMENTS OF SESSION CO-CHAIRMAN BERGMAN

OPENING COMMENTS OF SESSION CO-CHAIRMAN WEIDLER

As nuclear power plants and facilities age, one of the primary concerns that needs to be addressed, and repeatedly assessed in future conferences, is the effect of aging on component systems and structures. At today's session, we will hear presentations on filter and adsorber aging assessments, results of a study to determine the shelf life of HEPA filters, and the effect on efficiency of activated carbon after exposure to welding fumes. Two related topics will also be addressed; a method for estimating the efficiency of HEPA filters during and after a design basis accident and the performance of HEPA filters under hot dynamic conditions. I expect all of these topics to be of extreme interest to all members participating in the conference.

FILTER-ADSORBER AGING ASSESSMENT

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Abstract

An aging assessment of high-efficiency particulate (HEPA) air filters and activated carbon gas adsorption units was performed by the Pacific Northwest Laboratory as part of the U.S. Nuclear Regulatory Commission's (USNRC) Nuclear Plant Aging Research (NPAR) Program. This evaluation of the general process in which characteristics of these two components gradually change with time or use included the compilation of information concerning failure experience, stressors, aging mechanisms and effects, and inspection, surveillance, and monitoring methods (ISMM).

Stressors, the agents or stimuli that can produce aging degradation, include heat, radiation, volatile contaminants, and even normal concentrations of aerosol particles and gasses. In an experimental evaluation of degradation in terms of the tensile breaking strength of aged filter media specimens, over forty percent of the samples did not meet specifications for new material. Chemical and physical reactions can gradually embrittle sealants and gaskets as well as filter media. Mechanisms that can lead to impaired adsorber performance are associated with the loss of potentially available active sites as a result of the exposure of the carbon to airborne moisture or volatile organic compounds.

Inspection, surveillance, and monitoring methods have been established to observe filter pressure drop buildup, check HEPA filters and adsorbers for bypass, and determine the retention effectiveness of aged carbon. These evaluations of installed filters do not reveal degradation in terms of reduced media strength but that under normal conditions aged media can continue to effectively retain particles. However, this degradation may be important when considering the likelihood of moisture, steam, and higher particle loadings during severe accidents and the fact it is probable that the filters have been in use for an extended period.

I. Introduction and Background

The Phase I aging assessment of HEPA filters and gas adsorption units (adsorbers) described in this paper is part of an evaluation of the effects of aging on engineered safety feature (ESF) systems, one of the groups of systems of current interest in the NPAR Program.[1] The identification of potential safety-related aging issues, coupled with the need to avoid duplication, were the key considerations in identifying candidate ESF systems for initial study. Air-treatment or cleaning and ventilation systems were ultimately selected because failure of these systems can impact both plant and public safety. The first step in the Phase I study was to identify a boundary to isolate system

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components that would be assessed. The fundamental components in terms of providing plant and public safety are those designed to capture radioactive gaseous and particulate contaminants (namely, the adsorbers and HEPA filters, respectively). The particles could be radioactive chemical compounds or otherwise inert airborne material contaminated by radioactive species. Gasses of primary interest include the elemental and organic forms of radioiodine. Activated carbon is used to remove the gaseous or volatile forms of iodine and is usually impregnated with other chemicals to enhance the removal of the organic species. The adsorbers and filters can be the last barrier in preventing the release of radioactivity to the public following an accident, including that associated with the airborne iodine and cesium radionuclides which can provide a substantial contribution to total dose. Further, satisfactory performance is essential to ensure a controlled atmosphere for plant personnel, including control room habitability, during normal operations as well as accidents.

Activated carbon adsorption units and HEPA filters were selected as the focus for initial Phase I assessment not only because they are essential for safety, but in the case of the adsorbers, it has long been established, and was evidenced again during the Three Mile Island (TMI) accident, that aging mechanisms in these components can lead to impaired performance.[2,3] Aging is defined as the general process in which characteristics, including the performance of the two components, gradually change with time or use.[4] Related mechanisms and negative effects are discussed in the paper in conjunction with stressors, the agents or stimuli that stem from pre-service and service conditions which can result in the degradation. Aging mechanisms are the specific processes that gradually change characteristics of the components with time and use and include wear, embrittlement, corrosion, and other physical and chemical processes. Aging degradation is a negative effect or net change in a characteristic that can impair the ability to function within acceptance criteria. For filters, degradation includes loss of media strength, but for both filters and adsorbers, focus is ultimately on reduced performance in terms of particle and volatile iodine retention.

Failure experience, based on analyses of licensee event reports (LERs) and a survey of U.S. Department of Energy (USDOE) sites, is summarized in the report and, where possible, related to aging. The discussion of failure experience also includes exhaust-air treatment experience associated with the TMI accident.

Inspection, surveillance, and monitoring methods including specific surveillance tests that are performed periodically are reviewed and described. These methods and tests are used to establish the condition of the filters and adsorbers once the components have been placed in operation.

Information concerning the design and construction of the two air-treatment system components is briefly reviewed before the discussion of aging and failure experience. References to the documents that combine to provide basic physical, chemical, test, and performance standards for filter media and impregnated activated carbon as well as for assembled components are also included in the report.

II. Component Design and Construction

The particle filtration and gas (vapor) adsorption units (adsorbers) included in nuclear air-treatment systems are designed to remove radioactive materials. The materials, airborne particles or volatile species, may be suspended in or exist as gas phase constituents of recirculating aerosols, gaseous effluents, or accidental releases. The adsorbers, activated carbon beds, can effectively remove elemental radioiodine and are often impregnated with other chemicals to enhance the retention of organic species. The fine particle filters, HEPA filters, may also be significantly involved in radioiodine removal because iodine can be in the form of solid cesium iodide (CsI).

HEPA filters clean or treat aerosols by separating suspended particles from an essentially atmospheric pressure gas stream. The filter media is made from a mixture of glass fibers and is in the form of filter "paper." Particles are collected by interception, impaction, and Brownian diffusion. The continuous gas phase passes through virtually unchanged. A HEPA filter is defined as a "high-efficiency particulate air filter having a fibrous medium with a particle removal efficiency of at least 99.97% for 0.3 μm particles of dioctyl phthalate [DOP]."[5] An excellent review of the construction and service characteristics of HEPA filters has been prepared by First.[6] As indicated by First, "Filters constructed with paper pleated the full depth of the rigid outer frame and with adjacent pleats held apart by full-depth corrugated separators are the most widely used." The glass fiber filter paper is usually in the form of a continuous sheet, pleated vertically.

As indicated above, adsorbers are used to remove gaseous radioiodine species from effluents. Various standards and specifications permit the use of any adsorbent medium that has been demonstrated to be equal to or better than activated carbon in terms of radioiodine retention. However, only activated carbon is discussed because it is essentially the only material used and supporting testing procedures and acceptance criteria are specific to its use. Gasses are directed through tightly packed beds of activated carbon granules. The carbon or charcoal is prepared by controlled heating in a steam environment to remove volatile organic material. The heating generates a material with a large surface-to-mass ratio and internal surfaces or sites where adsorption of iodine molecules can take place. The active media is usually "...activated charcoal that is derived from either coal or coconut shells. Although elemental radioiodine is retained efficiently by activated carbon alone..., the charcoal is often impregnated with additional chemicals to improve its retention of organic species (e.g., methyl iodide)."[3]

Basic qualifications for the filter media and assembled HEPA particulate filters are contained in two military specifications.[7,8] Primary standards for adsorbent media, assembled adsorbers, and related testing methods are presented in Sections FD, FE, and FF of ASME/ANSI code AG-1 and two ASTM specifications.[9,10,11] Numerous documents supplement and/or support the preceding five references, including USNRC and other ASME and ASTM publications. One of the most closely related USNRC publications is Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."[12]

III. Stressors, Aging Mechanisms, and Degradation

The obvious effect of service associated with HEPA filter use is the increased pressure drop (resistance to air flow) through the filter media arising from particle retention (loading). In a relatively clean environment, filters may be used for several years before replacement is required. Dust loading, along with heat and radiation, also have the potential to reduce the effectiveness of the organic materials used to strengthen the filter medium and provide water repellency. High moisture content is another potential normal stressor, an agent or stimuli stemming from normal conditions and potentially producing aging mechanisms and effects which can lead to increased pressure drop as well as reduced filter medium strength. Filter media may embrittle from prolonged exposure to air containing normal concentrations of oxygen and oxides of nitrogen and sulfur.

Aging mechanisms or processes that can gradually change the physical characteristics of HEPA filter components other than the media include corrosion of metal members and physicochemical reactions that alter the properties of sealants, gaskets, and water repellents. Metal components that could be affected include the frames and corrugated separators. Heat and radiation are stressors associated with physicochemical reactions potentially affecting face gaskets and the adhesives and sealants that are used to splice the medium, fasten the gaskets to the filter face, and seal the filter pack to the frame.

Exposure to air containing stressors (namely, moisture and contaminants or pollutants) can slowly and continuously degrade the performance of gas adsorbers. This aging, also termed weathering, is inherent because of the nature of the material used, in that it has been "activated" to dramatically increase surface to mass ratio and provide countless reaction sites, and because of the essentially ubiquitous nature of the contaminants. Many airborne constituents, including moisture, can readily react or be adsorbed by carbon beds, reducing the number of active "sites" that otherwise would be available for radioiodine adsorption. During normal operation, charcoal may be exposed to air flows containing such contaminants as volatile organic solvents, sulfur dioxide, nitrogen oxides, and carbon monoxide. Because of system flow rates, even traces of pollutants can have a significant, cumulative effect. Oxidation as well as competitive loading can impair bed performance, including decreasing the efficiency of the impregnant. Because airborne constituents can vary dramatically with time and among sites, it is essentially impossible to provide criteria for the useful life of impregnated activated carbon.

It should be emphasized that the possibility of impaired performance in terms of radiodine retention as a result of the aging or weathering of impregnated activated carbon has been a concern almost from the start of the nuclear industry's existence. In fact, concern about the useful life of carbon could be considered one of the first, if not the first, aging issues.

IV. Failure Experience

An unusually large amount of potentially useful failure data were found in the literature. Analyses of LERs submitted by commercial nuclear power plant operators, covering the periods from January 1, 1975 through June 30,

1978, 1978-1981, 1985-1987, and 1988-1991, to identify those pertaining to air-monitoring, air-treatment, and ventilation systems, have been prepared by.[13,14,15,16] In summary, 15% of 43,500 reports were associated with the subject systems while an estimated roughly 100, or about 2%, of the system reports appeared to be associated with HEPA filters and carbon adsorbers. No relationship to aging was reported. Moeller and Kotra[17] and Moeller and Sun,[18] reviewed LERs for the periods 1981 through 1983 and 1984 through 1986, respectively, to isolate those pertaining to control room habitability. Of the approximately 20,800 reports, 4% were related to this subject. About 60 of the LERs were associated with impaired filter or adsorber efficiency. Several instances of the premature aging of carbon from error-induced conditions were reported.

Using a survey, Carbaugh[19] collected data on the numbers of and reasons for HEPA filter changeouts and failures at USDOE sites for the years 1977 through 1979. A total of 1,105 filter failures, or 12% of the 9,154 filter applications, were reported. A majority of the failures occurred for unknown or unreported reasons. Johnson et al.[20] explained the results of the above survey: "If one examines the categories reported for filter failure, it is evident that the effects of aging could contribute to 81% of these failures, except for the 19% resulting from handling or installation damage." These same investigators reported results of an experimental evaluation of the tensile breaking strength of aged filter media specimens that revealed that 42% of the samples did not meet the specifications for new material. A significant loss in water repellency from values specified for new media was also measured.[20]

Selection of adsorbers for Phase I study was reinforced by literature references revealing that the low radioiodine decontamination factors associated with the TMI accident were attributed to the premature aging of charcoal. Before presenting details related to aging issues, it should be emphasized that building ventilation systems played an important role in decreasing the radionuclide release associated with this incident and, in fact, prevented the release of most of the radioiodine. However, the terms "somewhat dismal results" and "rather unsatisfactory" have been used to describe exhaust air filtering experience during the TMI accident.[21,3] In summary, it was determined that the filtering systems installed at the time of the accident provided a decontamination factor (DF) of 9.5 for all species of radioiodine (corresponds to a retention efficiency of 89.5%) and the radioiodine releases were higher by about a factor of five than they would have been if there had been no system deterioration. The preceding values and reasons for the degradation are detailed by Rogovin and Frampton[22] and discussed by Bellamy[23] in a paper that was prepared to summarize pertinent (related to air-cleaning technology) efforts and recommendations of various investigative groups. Briefly, system degradation and the fact that radioiodine retention was associated with an efficiency of 89.5% were ultimately attributed to pre-accident aging of the impregnated activated carbon. For a number of reasons, ventilation air was passed through exhaust systems from the time of carbon installation to just shortly after the accident. This approximately one-year time frame included periods of exposure of the carbon beds to fumes from painting and cleanup efforts. Furthermore, exemptions to technical specifications postponed surveillance tests that could have revealed the extent of aging.

Review indicated that releases of radioactive material in particulate form during the TMI accident were negligible. Postaccident efforts included changeout of HEPA filters as well as carbon. As indicated by Rogovin and Frampton,[22] "These HEPAs were visually examined before changeout and were intact and in satisfactory condition, but were damaged during changeout....Unfortunately, no used HEPA filters or sections of filter media were retained for analysis."

V. Inspection, Surveillance, and Monitoring Methods

The use of surveillance tests, series of tests periodically performed to monitor component condition, is emphasized. Surveillance tests include in-place leak tests and visual inspections to evaluate filter banks and adsorbers in terms of component damage and bypass. Also included, and specifically related to aging concerns, are laboratory tests of used or aged carbon samples to determine remaining radioiodine adsorption capacity. In addition to the surveillance tests, HEPA filter pressure drop is continuously monitored (alarmed and indicated).

Surveillance leak testing of installed HEPA filters and adsorbers is specified because gradual deterioration and leaks could develop under standby as well as service conditions. These tests are based on a DOP or halide challenge aerosol or gas introduced upstream of the filters or the adsorbers, respectively. Concentrations upstream and downstream are then measured. It should be emphasized that these are leak rather than efficiency tests. For adsorbers, however, the leak test is supplemented with laboratory tests of aged or used carbon samples to determine system efficiency and remaining capacity for methyl iodide. It is stated in ASME N509 that "Sufficient test canisters or other means of obtaining samples... of used adsorbent shall be installed in the adsorber system to provide a representative determination of the response of the adsorbent to the service environment over the predicted life of the adsorbent. Test canisters shall be installed in a location where they will be exposed to the same air flow conditions as the adsorbent in the system, shall have the same adsorbent bed-depth as the adsorbent in the system, and shall be filled with representative adsorbent from the same batch of adsorbent as that of the system."[5]

VI. Conclusions

There appear to be adequate inspection, surveillance, and monitoring methods for normal conditions. It is recognized that, excluding local ruptures and tears, neither pressure drop monitoring nor surveillance leak testing of installed HEPA filters will provide indications of aging in terms of reduced filter media strength. However, even aged, intact filters can still effectively remove particles under normal conditions. On the other hand, this lack of indication may be important when considering reactor accident conditions. Regulations require atmospheric cleanup systems to mitigate the consequences of postulated accidents and, as indicated earlier, system components can be the last barrier against the release of radioactivity to the public.

The Phase I study established that the HEPA filters and adsorbers are considered to have a long service life, especially the filters. Thus, if a severe accident happens it is likely to occur at a time when these two final

confinement barriers have been in use for an extended period, even years. Even with existing ISMM, aged and possibly degraded components could fail to provide the radiation protection needed for safe shutdown or be the weak link that allows the release of radionuclides to the environment. Furthermore, the assessment has revealed the need for an improved definition of accident conditions and the possible need for additional information to evaluate the performance of aged components under such conditions. Past design basis evaluations based on the assumption that iodine is primarily in the volatile form may be neither correct nor conservative when evaluating air-cleaning system performance. It is recognized that improving the definition of accident conditions is outside the scope of the NPAR program. It is also recognized that comprehensive assessments limited solely to the development of information for performance evaluation could be made. However, for such studies to be definitive, knowledge concerning challenges must ultimately be obtained, as will be discussed below.

Some information concerning the range of challenges expected for filters and adsorbers during accidents is already available as a result of source term reassessment and reactor risk studies.[24,25] Subsequent work has revealed further insights into iodine chemical form. Although this follow-on effort did not address ultimate disposition, using calculated data from seven severe accident sequences, it was found that "in most of the calculations for the seven sequences, iodine entering containment from the reactor cooling system was almost entirely in the form of CsI with very small contributions of I or HI."[26] Additional insights concerning accident conditions should also become available as a result of recent regulatory efforts to provide improved source term definitions. As part of several regulatory activities to incorporate severe accident insights into the safety assessment of future plants, the NRC has issued a proposed revision of the reactor accident source terms. The proposed revision is in terms of fission product composition, magnitude, timing, and iodine chemical form, for release into containment.[27] Utilization of these revised source terms should provide improved estimates of accident conditions and the associated challenges to filters and adsorbers. For example, evaluation of iodine chemistry during the portion of the transient that follows release into containment will be needed, resulting in better estimates of whether iodine remains primarily in the particulate form or is converted to a volatile species before it reaches filters and adsorbers.

Even if details concerning accident conditions were available, there might be insufficient information to develop reliable predictions concerning the performance of aged components. Experiments to evaluate the behavior of filters and adsorbers under non-standard conditions including high humidity airflows, steam, and high temperature and radiation levels have been conducted. Nevertheless, statistically designed engineering-scale experiments, involving aged components and using the improved estimates of the ranges and combinations of challenges, may ultimately be justified. It is also possible that improved estimates of conditions could negate the need for further work, because either moderate or extremely severe stressors were predicted. In the case of moderate stressors, additional study would be unwarranted if near-normal operating conditions were identified. In the case of severe conditions, HEPA filters are simply not sized to handle the massive loading of nonradioactive particles that conceivably could result from molten core-concrete interactions.

An improved definition of accident conditions may also reveal that aging concerns associated with the filters and adsorbers of "normal" treatment systems may equal those of ESF systems and that the designations of primary and secondary used in RG 1.52[12] for ESF systems may be trivial in terms of accident considerations. One of the findings resulting from investigations into the TMI accident was the fact that "...a nonengineered safety feature filter system designed for normal operation only, i.e., the auxiliary building exhaust ventilation filtration system, greatly reduced the quantity of radioiodine release to the environment...the safety grade versus nonsafety grade designation [for the two air treatment systems operating at the time of the accident] was meaningless..."[22] An improved understanding of severe accident conditions could also further emphasize the importance of other air-treatment system components. For example, along with HEPA filters, greater emphasis would be placed on coarse or roughing pre-filters and even demisters if it were determined that high particle concentrations were involved.

Finally, it should be emphasized that information used to estimate the performance of aged air-treatment system components can have a significant impact on the calculated consequences of postulated accidents. This is illustrated by the analysis of a postulated loss-of-decay-heat-removal accident at Browns Ferry. The standby gas treatment system (SGTS) failure model used, listed as the area of greatest uncertainty with respect to informal sensitivity analyses, includes best estimates concerning the filter loading that will cause failure, the type of failure, and the functioning of the adsorber under the projected accident environment. It is noted that "the SGTS failure model assumes prime importance because the SGTS is the last barrier to the atmosphere in this accident sequence."[28]

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23rd DOE/NRC NUCLEAR AIR CLEANING AND TREATMENT CONFERENCE

CRITERIA FOR CALCULATING THE EFFICIENCY OF DEEP-PLEATED HEPA FILTERS WITH ALUMINUM SEPARATORS DURING AND AFTER DESIGN BASIS ACCIDENTS*

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ABSTRACT

We have reviewed the literature on the performance of high efficiency particulate air (HEPA) filters under normal and abnormal conditions to establish criteria for calculating the efficiency of HEPA filters in a DOE nonreactor nuclear facility during and after a Design Basis Accident (DBA). This study is only applicable to the standard deep-pleated HEPA filter with aluminum separators as specified in ASME N509 [1]. Other HEPA filter designs such as the mini-pleat and separatorless filters are not included in this study. The literature review included the performance of new filters and parameters that may cause deterioration in the filter performance such as filter age, radiation, corrosive chemicals, seismic and rough handling, high temperature, moisture, particle clogging, high air flow and pressure pulses. The deterioration of the filter efficiency depends on the exposure parameters; in severe exposure conditions the filter will be structurally damaged and have a residual efficiency of 0%. Despite the many studies on HEPA filter performance under adverse conditions, there are large gaps and limitations in the data that introduce significant error in the estimates of HEPA filter efficiencies under DBA conditions. Because of this limitation, conservative values of filter efficiency were chosen when there was insufficient data.

The estimation of the efficiency of the HEPA filters under DBA conditions involves three steps. In the first step, the filter pressure drop and environmental parameters such as temperature and moisture are determined during and after the DBA. The second step consists of comparing the filter pressure drop to a set of threshold values above which the filter is structurally damaged. There is a different threshold value for each combination of environmental parameters. The filter efficiency is determined in the third step. If the filter pressure drop is greater than the threshold value, the filter is structurally damaged and is assigned 0% efficiency. If the pressure drop is less, then the filter is not structurally damaged; and the efficiency is determined from literature values of the efficiency at the environmental conditions. The efficiency of the HEPA filters within DOE facilities should be determined on a case-by-case basis.

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I. Introduction

The primary standard that governs the use of HEPA filtration systems in DOE facilities is DOE Order 6430.1A [2], which in turn refers to ASME N509 [1]. However, the standards do not provide guidance for determining their efficiency under abnormal or accident conditions. Under normal operating conditions, a HEPA filter will have a minimum efficiency of 99.90% [3]. This is the minimum filter efficiency at the most penetrating particle size of 0.15 μm diameter for a HEPA filter that has a minimum efficiency of 99.97% for 0.3 μm DOP particles. Previous publications had assigned a single value for the efficiency of HEPA filtration systems under all accident conditions. However the publications reviewed in this paper demonstrate that the efficiency of HEPA filters will vary greatly depending on the operating conditions, thereby requiring a case-by-case analysis.

Elder et al [4] prepared a guide for analyzing the accidental release of radioactive material from nonreactor nuclear facilities. This guide reviews the applicable DOE orders, provides a description of the design basis accidents, and evaluates the consequences of the accidents. In his section on reduction and removal factors, Elder discusses the efficiency of HEPA filters under DBAs. Elder recommended that the first stage HEPA be credited with an efficiency of 99.9% and each subsequent stage with 99.8%. These values were selected from an unofficial HEPA filtration guideline that was established in 1971 during a meeting between officials from the Atomic Energy Agency and Albuquerque Operations Office [5]. These guidelines represented the opinions of the meeting attendees, and were not supported by technical data.

Walker [6] tabulated the available data on HEPA filter efficiency as of 1978 and recommended efficiencies of 99.9% for the first stage, 99.0% for the second and third stages, and 83.0% for the fourth stage. Although Walker cited experimental data, the recommended values were based on his opinion. Walker further stated in the summary that further study of HEPA filter efficiency is needed "to better establish relationships with relative humidity and temperature of sweep gas, service aging, material loading, etc."

The Nuclear Regulatory Commission in Regulatory Guide 1.52, recommended an efficiency of 99% for a filtration system consisting of a two stages of HEPA filters with an adsorber in between [7]. The guide is applicable to light-water-cooled nuclear power plants, where the filtration system is off-line during normal operations and only activated during accident conditions. Table 1 of the code shows that environmental conditions to which the HEPA filters would be exposed are approximately atmospheric pressure, 180°F, 100% relative humidity, and 105 rads/hr. There is no explanation for setting the HEPA

filter efficiency at 99%, when the new filter is tested at 99.97% for 0.3 μm DOP particles and is also tested in-place for leaks at 99.95% as prescribed in ASME N510. [8]

The Nuclear Air Cleaning Handbook [9] recommended 99.8% efficiency for the first stage and 99.9% to 99.95% for each of the remaining stages. The Handbook presumably recommends a lower efficiency for the first HEPA filter and a higher efficiency for the second stage HEPA than the AEC 1971 draft because of the assumption that the first filter would take the brunt of any adverse effect from accidents. In discussing multistage HEPA filtration, the Handbook stated on page 38, "The purpose is to increase the reliability of the system by providing backup filters in the event of damage, deterioration, or failure of the first filters."

The major deficiency in the previous guidelines on HEPA filter efficiencies under DBA conditions is that the HEPA filter efficiency in DOE facilities can vary from 99.9% to 0% depending on which of the large number of different DBA conditions is applicable. In contrast, for light water reactors, there is a single DBA condition for which the HEPA filter efficiency is assigned 99% [7] We have proposed a method for computing the efficiency of HEPA filters on a case-by-case basis using the available data reported in the literature. In this paper we describe the development of criteria for calculating the efficiency of HEPA filters during and after a DBA. These criteria are intended to be used in a future DOE Standard.

The goal of this paper is to provide guidance for computing the efficiency of HEPA filters during and after a DBA. The computed filter efficiency can be used in determining off-site doses from postulated releases of airborne radioactivity for both existing and future facilities. However, this study is not intended to define criteria for HEPA filter survival or provide guidelines on how to construct a HEPA filter that survives a DBA. Since there are a large number of different DBAs that are applicable to the many different DOE facilities, the HEPA filter efficiency will be computed on a case-by-case basis.

II. Approaches for Computing HEPA Filter Efficiency Under DBA

Single Efficiency Value for All DBA Conditions

We considered various approaches for developing the criteria for computing HEPA filter efficiency under DBA conditions. The previous approach was to specify a single efficiency value for all conditions. We considered this to be unrealistic since the efficiencies of the available HEPA filters vary widely with different environmental parameters. Since there are no commercially available HEPA filters certified for

use in DOE facilities that can survive all of the postulated DBA conditions it is not possible to assign a single efficiency value to HEPA filters.

For the currently available HEPA filters, the only efficiency value that can be applied for all DBA conditions is 0%. However, since the HEPA filters in most DOE facilities will survive the applicable DBA, assigning 0% efficiency to these filters is unrealistic. We concluded that until a high-strength HEPA filter is developed, it will not be possible to assign a single efficiency to HEPA filters that will apply for all DBA conditions.

Efficiency Determined From Matrix of Three Severity Levels of Fire, Explosions, and Tornados

We then reviewed various matrix approaches in which there are multiple filter stresses and multiple responses. A filter efficiency would be assigned for each of the filter stresses. In our initial review, there appeared to be only a few realistic DBA scenarios that include combinations of anticipated stresses that reflect graduated levels of severity. For example, a remote fire usually represents a filter loading episode without a very high temperature or a pressure pulse; an explosion usually represents a temperature and pressure pulse, and it may be the pressure resulting from rapidly increased airflow that does the damage and rapidly dies away to be followed by a fire without excessive pressure or airflow volume increase; a tornado results in a rapid pressure increase and/or decrease but no fire or particle/droplet loading. Each scenario can occur in varying degrees of severity that can probably be classified as "below concern", "moderate stress", and "extreme stress". This represents a 3 x 3 matrix and should not be excessively complex.

Although the concept is great, we were not able to assign an efficiency value for each of the proposed nine elements of the matrix because the three stresses of fire, tornado, and explosions were not single stress quantities, but could very widely in terms of more fundamental parameters such as temperature, pressure, and aerosol quantity and composition. For example estimating the efficiency of a HEPA filtration system after a fire cannot be derived from categories such as below concern, moderate stress or extreme stress. In order to assess the response to the fire, it will be necessary to estimate the temperature, quantity of smoke, type of smoke, water spray from a fire suppression system, the system flow rate, and the pressure drop across the HEPA filter. The pressure drop across the filter is, in turn, a function of the flow rate, water contact, and the quantity and type of smoke.

There is not a single DBA sequence that represents a fire. For example the remote fire scenario only involves filter clogging with little temperature increase or pressure pulse. In reality, there are a large

number of different fire scenarios: a well ventilated fire generating slow plugging aerosols, an under ventilated fire generating rapidly plugging aerosols, a fire with high temperature flames, use or non-use of fire suppression system based on water sprays and demisters, etc..

If the fire is remote and well ventilated, then the only consequence would be a small or moderate increase in pressure drop and no loss in efficiency. However a remote fire that is under ventilated (oxygen starved) would cause rapid filter plugging. If the air blower does not exceed the breaking point of new filters (37 inches) then the effect would be a near or total shut down of the ventilation system. The efficiency of the clogged filter would be higher than the efficiency of the clean filter. If the smoke is diluted prior to reaching the HEPA filter, then the filter plugging will be reduced proportionately to the air dilution.

As another example dealing with fire, assume the fire is localized at the filter and reaches 400°C. If no water suppression system is activated and the filter pressure drop is below 15 inches then the filter will be structurally undamaged but have a penetration about of about 3% . However if the water spray system is activated, then the filter will rapidly plug due to water accumulation. If the air blower can pull greater than 10 inches of water then the filter will be structurally damaged and have dramatically reduced efficiency (conservatively set at 0%). However if the air blower cannot pull greater than 10 inches of water, then the plugged filter will shut down the ventilation system but still have the high efficiency.

Thus from these few examples we have shown that estimating the efficiency of the HEPA filters requires a more complex approach than categorizing a stress as below concern, moderate stress or extreme stress. There are also many other potential stresses than fire, tornado and explosion: for example, steam from a ruptured steam line, fire suppression system that sprays water, chemical effluents, and seismic stress.

In addition, the DBA stresses cannot be limited to a small number of different conditions because new DBA conditions will be established for new facilities and operations as DOE missions and directions change. New programs in weapons dismantlement, waste clean up, decontamination and decommission operations will bring their own set of stress factors.

Efficiency Determined from Generalized Matrix of Operating Parameters and Parameters Causing Structural Damage

We then evaluated a generalized matrix in which the efficiency is established at each level of stress for a series of fundamental parameters at increasing parameter level. Table 1 illustrates the

generalized matrix for tabulating the filter efficiency at each parameter level for a series of different parameters. These are the known parameters that have an effect on filter efficiency.

Table 1. Generalized matrix for establishing filter efficiency for operating parameters

	Efficiency at Different Parameter Levels		
Operating Parameter	Low	Moderate	High
temperature			
solid particle loading			
water/liquid particle loading			
air flow			

Although Table 1 shows only three parameter levels (low, moderate, and high), the complete table could consist of many parameter levels, each level defined by specific values of the operating parameter. The primary advantage of the generalized parameter matrix over the three-stress matrix is that the parameters are fundamental parameters that can be uniquely defined. It is also possible to use semi-empirical equations for computing the efficiency due to the different parameters. Since the four parameters can be considered to act independently on the filter, the filter efficiency for combination of parameters can be computed from the efficiency for individual parameters.

The effect of the four operating parameters on the HEPA filter efficiency can be explained in terms of particle capture and particle loading theory and experiments. For HEPA filters, the maximum particle penetration is determined by the Brownian motion and interception capture mechanisms [10]. As the temperature is increased, the Brownian motion will increase while the interception mechanism remains constant. Thus higher temperatures will result in higher filter efficiencies. As the air flow increases, the capture by Brownian motion will decrease because of the decreased particle residence time in the filter. Since air flow has no effect on the interception mechanism, the net effect of increased air flow will be a lowering of filter efficiency. At exceedingly high air flows, the particle inertia becomes important and increases the filter efficiency. For solid particle loading, the particles form dendritic structures that capture additional particles, thereby increasing filter efficiency [11]. For liquid particles, the particles coalesces within the filter and decrease the void volume within the filter medium. The decreased void volume results in higher internal velocity and thus less residence time for Brownian motion [12]. Thus, as liquid deposits form in the filter, the aerosol penetration will increase.

Unfortunately the direct approach for computing the efficiency of HEPA filters under DBA conditions described above is not possible because the HEPA filters are frequently structurally damaged, which lowers the filter efficiency. For example as the temperature is increased, the efficiency increases. Figure 1 shows the decontamination factor (DF = 1/penetration) as a function of particle diameter for a deep-pleated HEPA filter with aluminum separators at increasing temperatures up to 200 C.

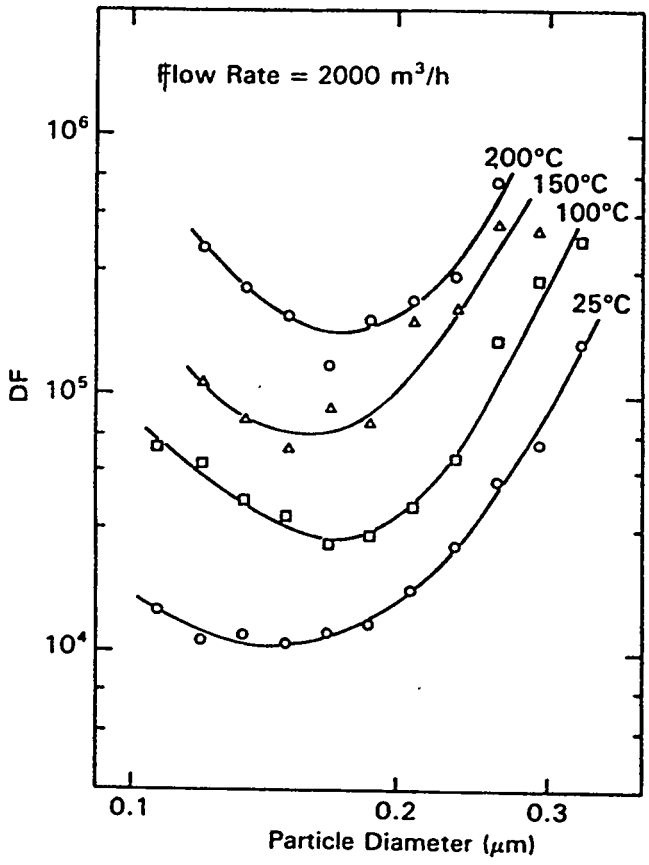


Figure 1 Decontamination factor (DF =1/penetration) for HEPA filter as a function particle diameter at increasing temperatures. Data from Osaki and Kanagawa [13].

With increasing temperature the organic components begin degrading and introduce leak paths through the filter. Figure 2 illustrates the effect of increasing temperature on the decontamination factor for HEPA filters with different frame materials.

The effect of increasing air flow provides another example of a stress having a direct effect on the filter efficiency due to the basic filtration mechanism and an indirect effect due to structural damage to the filter. Figure 3 shows the increasing DOP aerosol penetration as a function of particle size for a HEPA medium tested at increasing air velocities.

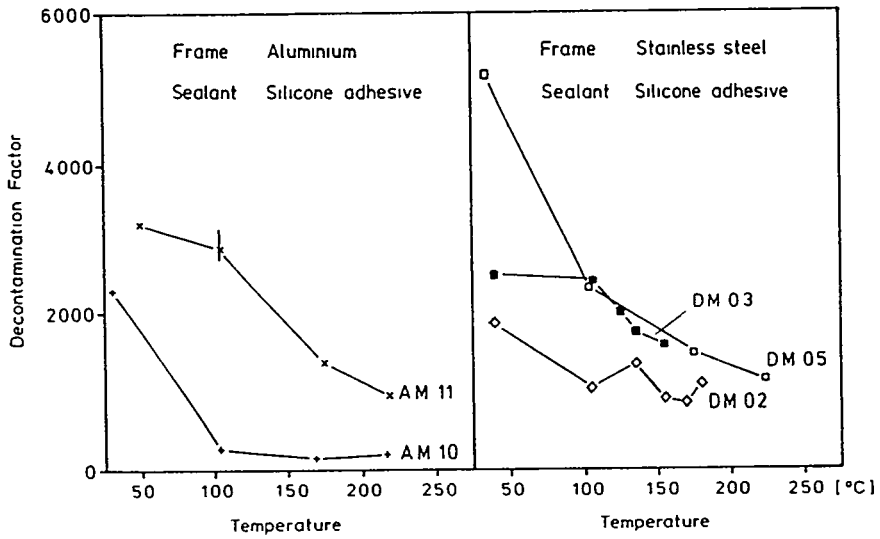


Figure 2 Decontamination factor for HEPA filters as a function of increasing temperatures. Data from Ensinger et al [14].

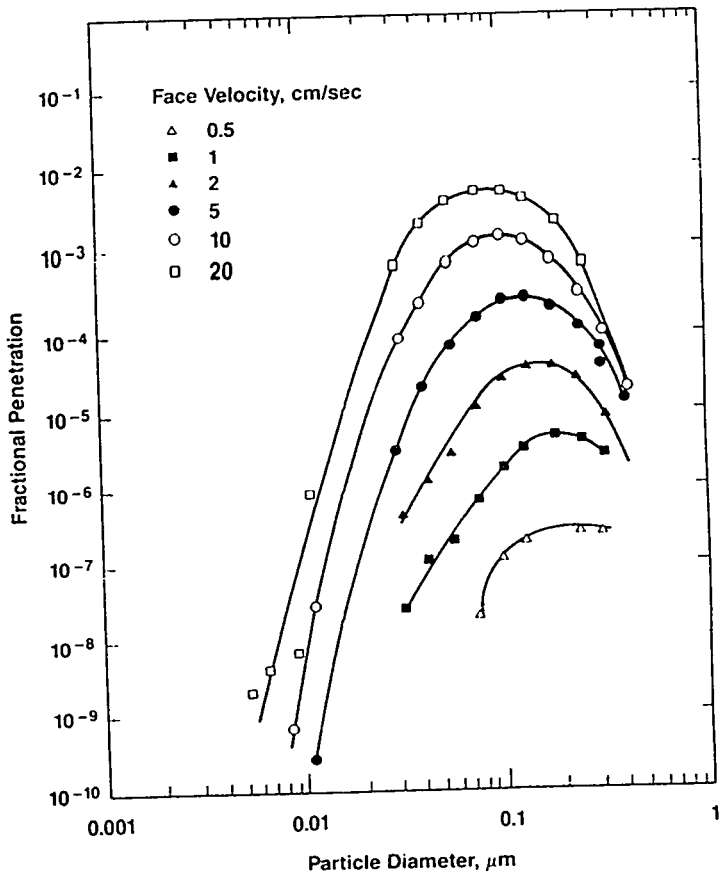


Figure 3. Penetration of DOP aerosols as a function of particle diameter at increasing air velocities for a HEPA medium, Lydall grade 213. Data from VanOsdell et al [15].

As the air flow is increased the increasing pressure drop on the filter stresses the filter medium and eventually causes medium tears At high air flows, the entire medium pack is blown out of the housing as shown in Figure 4. It is obvious that the filter has 0% efficiency.

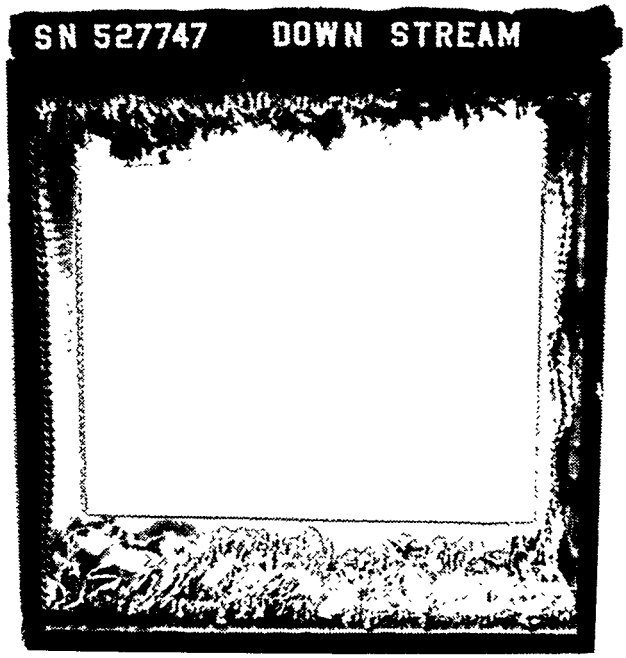


Figure 4 Photograph of 15-19 year old HEPA filter after subjected to a 13 inch differential pressure Photograph from Johnson et al [16].

There are additional stresses such as age, radiation exposure, water/liquid loading, seismic or rough handling, and chemical attack that also can weaken or cause structural damage to the HEPA filter. Structural damage creates leak paths that lower the HEPA filter efficiency, whereas a weakened filter suffers structural damage at lower stress levels. Thus in order to determine the efficiency of HEPA filters under DBA conditions, it is necessary to include indirect effects due to HEPA filter damage in addition to the direct effects from the parameters listed in Table 1. However, incorporating structural damage and a weakened filter into a general method for determining filter efficiency is extremely difficult because a number of interacting parameters must be incorporated in the efficiency calculation. We will use data obtained from the literature to establish a method for computing the efficiency of HEPA filters under normal and DBA conditions.

III. Literature Review of HEPA Filter Efficiency Under Normal and DBA Conditions

The Minimum Baseline Efficiency for Each Stage of HEPA Filtration is 99.9%

Single HEPA Stage According to ASME N509 each HEPA filter must have a minimum efficiency of 99.97% for 0.3 μm DOP aerosols. However, recent studies have shown that the 0.3 μm DOP aerosol is not the most penetrating size and therefore is not the most conservative efficiency. The particle penetration through HEPA filters is a strong function of particle size as shown in Figure 5 taken from Bergman et al [17]. The maximum particle penetration at 0.15 μm diameter is about four times the penetration measured using 0.3 μm DOP particles. The penetration decreases at smaller and larger particles due to competing particle capture mechanisms [18]. Particles less than 0.15 μm have increasing Brownian motion with smaller sizes and therefore have lower penetration through the filter. Particles larger than 0.15 μm have greater inertia with increasing size and are unable to follow the tortuous path through the filter. This results in a lower penetration with increasing size. The two capture mechanisms result in the experimental penetration curve seen in Figure 5.

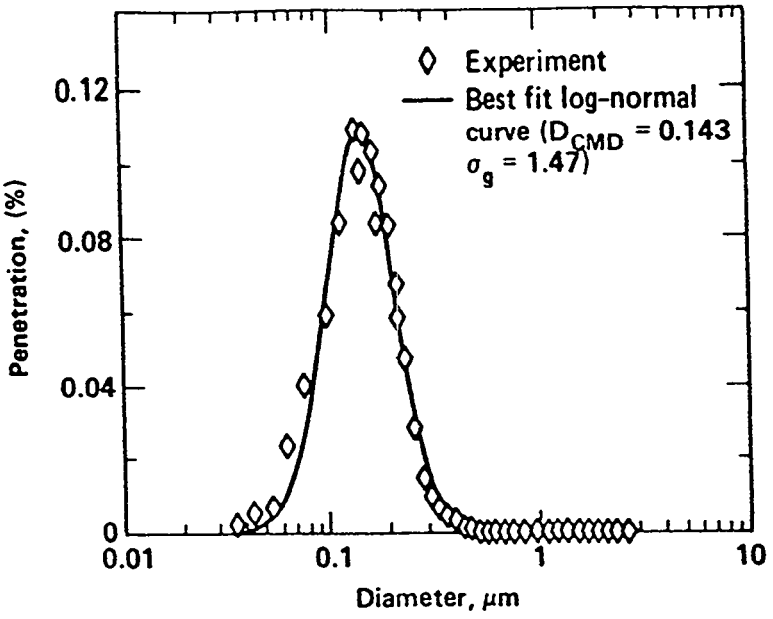


Figure 5 Penetration of DOS aerosols through a standard nuclear grade HEPA filter [17].

Scripsick et al [3] conducted a comparison study between the standard DOP test using 0.3 μm particles and the maximum penetration test in which penetration is measured as a function of particle size. They found that of 849 filters tested that had greater than 99.97% efficiency for 0.3 μm particles, 18% of the filters had less than 99.97% but greater than 99.90% efficiency for 0.15 μm particles. Thus the minimum efficiency for a new, undamaged HEPA filter is 99.90% at any particle size. The efficiency in most applications will be much greater, not only because the particle size is usually greater that 0.15 μm, but also because the particles generally have a greater density than the density of DOP (0.98g/cc). Anderson [19] and Tillery [20] have shown higher density aerosols have significantly higher efficiencies. However, if the aerosols consist of agglomerates of smaller particles, then the density will be significantly smaller than the bulk material density.

Multiple HEPA Filtration Stages. If the HEPA filter penetration is measured at the most penetrating particle size then the total maximum penetration for multiple stages of HEPA filters is the product of the penetration from each of the HEPA filters in series. There is no limit to the number of stages for which this computation will apply. However, each stage or combined stages must be leak-tested as specified by ASME N510 [8]. The penetration for each new, undamaged HEPA filter is 0.1% (99.9% efficiency). Untested HEPA filters with no visual damage are assigned a penetration of 10% (90% efficiency) based on the following field data using radiological measurements: Hetland and Russell [21] measured HEPA filter efficiencies of 94% and 83% for Pu aerosols in the second and third stages respectively. Frigerio and Stowe [22] measured an average efficiency of 99.4% for Pu and 92.9% for U in a third stage HEPA filter. Previously, untested HEPA filters were arbitrarily assigned a penetration of 0.2% (99.8% efficiency) [9].

The determination of total penetration for multiple HEPA filters from the product of the individual filter penetrations appears to contradict the widely held belief that the filter penetration increases (i.e. efficiency decreases) with each filter stage. Previous studies that have shown a decrease in filter efficiency with multiple stages of filtration were generally based on average efficiencies for heterodisperse aerosols and/or have had significant background measurements. If filter efficiencies are based on average measurements over a particle size distribution, then the apparent filter efficiency will change with successive stages of filtration because the average particle size has changed. Ortiz [23] showed that even heterodisperse aerosols with narrow distributions will be altered when passing through HEPA filters and thereby cause a shift in the average particle size and the filter penetration.

Gonzales et al [24] conducted experiments with plutonium aerosols that showed the penetration of each HEPA filter in a system consisting of three filter stages is 99.99% for the first two stages and 99.84% for the third stage. They measured the HEPA penetration as a function of particle size for the first

two stages. The extremely low concentration of plutonium particles challenging the third filter made measurements extremely difficult and is most likely responsible for the lower filter efficiency.

Osaki and Kanagawa [13] showed that the penetration of 0.15 μm DOP particles for the second HEPA filter in a two-stage filter system yielded identical results as the first HEPA filter. They also verified that the two filters continued to have the same efficiencies at various air flows.

The study by Gonzales et al [24] and filtration theory were used in ERDA 76-21 [9] to conclude that the particle penetration through multiple HEPA filters equals the product of the penetrations of individual HEPA filters. ERDA 76-21 [9] points out that the multiplication of penetrations is valid for any particle particle size

Literature Review of Parameters That Cause a Deterioration in HEPA Filter Efficiency

A number of laboratory studies have shown that the efficiency of HEPA filters will degrade under certain environmental conditions. In extreme cases, the HEPA filter will be structurally damaged and result in 0% efficiency. The identified parameters that can cause a deterioration in the filter include age, moisture, chemical attack, high temperature, overpressure, shock and rough handling, particle loading, and radiation. The effect of these parameters on HEPA filters is dependant on the materials used and the design of the HEPA filters.

We summarized the literature data for the differential pressure required to cause structural damage on HEPA filters for various parameters in Table 2 In some cases only media data were available, and we had to compute the pressure drop for an equivalent HEPA filter. The percent reduction in media tensile strength or bursting strength from before and after exposure to a given parameter was used in the computation. This percent was then multiplied by the measured pressure drop required to blow out the medium in a new HEPA filter as measured by Gregory et al [25]. For example , Breschi et al [26] measured a 33% reduction in medium burst pressure after exposure to 200 C for one hour. The medium thus had 67% of its original strength. Multiplying 67% by 37 inches of water (the minimum pressure required to blow out the medium from a new HEPA by Gregory et al [25]) yields 25 inches of water. Although no data was available on the structural damage from acid exposure, the filter is degraded with increasing exposure and will even collapse by its own weight with prolonged exposure.

Table 2 Threshold Values of Differential Pressure Required to Structurally Damage the Standard HEPA Filter

ΔP threshold, inches Avg.	Range	Parameter	Reference
66	(37-81)	Baseline (high air flow)	Gregory et al [25]
57	(38-72)	Baseline (high air flow)	Osaki et al [13]
63	(47-90)	Baseline (high air flow)	Ruedinger et al [27]
52	(29-70)	Explosion shock	Gregory et al [25]
38	(13-78)	Age (15-19 year old filters with Asbestos separators)	Johnson et al [16]
33[1]	(18-40)[1]	Radiation (5×10^7 rad)	Jones [28]
X[2]		HNO_3, HF exposure (variable)	Woodard et al [29]
		Temperature	
44[3]	(25-54)[3]	200°C (392°F), 1 hr.	Breschi et al [26]
33[4]	(19-41)[4]	300°C (572°F), 10 min.	Hamblin et al [30]
26[5]	(15-32)[5]	400°C (752°F), 1 hr.	Breschi et al [26] and Hamblin et al (30)
13[6]	(8-16)[6]	500°C (932°F), 10 min.	Pratt et al [31]
	(8-20)	500°C (952°F), 10 min.	Pratt [32]
23	(10-36)	Clean filter, water spray	Ruedinger et al [33]
20	(16-25)	Loaded filter, 100% humidity	Ruedinger et al [33]
18	(7-36)	Clean filter, water spray	Ricketts et al [34]
16	(3.6-25)	Loaded filter, 99% RH	Ricketts et al [34]
40[7]	(22-49)[7]	Clean dry filter, prev. wet	Ricketts et al [34]

Footnotes

1. Values computed from a measured 50% reduction in media tensile strength and base line values from Gregory et al [25].
2. No available data relating differential pressure threshold and acid challenge. Observations of HEPA filter after acid challenge show the HEPA media collapses and may fall out of its housing by its own weight.
3. Values computed from a measured 33% reduction in media rupture pressure and baseline values from Gregory et al [25].
4. Values computed from a measured 50% reduction in media rupture pressure and baseline values from Gregory et al [25].
5. Values computed from a measured 60% reduction in media rupture pressure and baseline values from Gregory et al [25]
6. Values computed from measurement of 80% reduction in tensile strength and baseline filter values from Gregory et al [25].
7. Values computed from measurement of 40% reduction in tensile strength and baseline filter values from Gregory et al [25].

Note: This table applies to HEPA filters having the deep-pleat design, organic sealant, and conventional glass fiber media. Other commercially available HEPA filters have lower threshold values for differential pressure.

Table 3 Effect of Parameters on the Penetration of HEPA Filters

Parameter	Effect on Filter Penetration	Reference
Baseline	0.1%	Scripsick et al [3]
HF Corrosion		
1,500 ppm-hr	0.1% increase	Brassel et al [35]
Temperature		
increase from 25 to 200 C	decreases penetration from 0.01 to 0.001%	Osaki et al [13]
200°C	.03-0.01%	Pratt et al [31]
240°C for 6 hours	0.01%	Osaki et al [13]
300°C	0.12-0.01%	Pratt et al [31]
350 C	0.4-0.03%	Pratt et al [31]
500°C	0.9-0.2%	Pratt et al [31]
500°C for 10-45 min.	0.9-0.1%	Hackney [36]
538°C	1.2-0.5%	First [37]
Moisture		
Up to 100% RH	Negligible effect	Osaki et al [13]
Water spray loaded to 8 in.	Increase by 10 times	Osaki et al [13]
Filter clogging		
Solid particle loading	Decreases penetration	Bergman [11]
NaCl deposits to 1.9 in.	Decreases penetration from 0.003 to 0.000001%	Osaki et al [13]
Liquid DOP loaded to 4 in.	Penetration increases by factor of 10	Osaki et al [13]
Oil aerosols	Penetration increase is 1.3 $P_i\Delta P/\Delta P_i$ increase	Payet et al [12]
Air Flow		
Increasing velocity from 0.5 cm/s to 20 cm/s	Penetration increases from 0.00003% to 0.5%	VanOsdell et al [15]
Increasing air flow by 10 times	Penetration of 0.1 μm particles increases by100 times	Osaki et al [13]
Air Pulse		
1 psi pulse	Penetration of 0.46 μm latex particles is 0.1%	Gregory et al [25]
Shock tests on filters preloaded with .46 μm latex	Penetration is 0.9%	Gregory et al [25]
Seismic (0.2- 0.3 g)	negligible effect	Bergman et al [38]

We also reviewed the literature for the effect of different parameters on the efficiency of HEPA filters and have summarized the data in Table 3. The HEPA filters in these studies did not suffer structural damage except for the higher temperature and acid exposure tests. Although the HF exposure shows only a single entry, the penetration will increase with increasing exposure due to chemical attack

on the filter medium. Table 3 also shows a range of penetration values that were observed for different HEPA filters after the high temperature exposures.

IV. Developing a Method for Determining Filter Efficiency Under Normal and DBA Conditions

Generalized Matrix Method is Not Feasible

Our initial approach was to incorporate the literature data on filter deterioration into the generalized matrix shown in Table 1. Unfortunately it is not possible to simply add additional parameters that weaken the filter or cause structural damage such as age and radiation exposure to the existing parameter list because the parameters are not independent. For independent parameters, it is not necessary to consider parameter interaction in computing filter efficiency. For example, when determining the filter efficiency under specific conditions of temperature, solid particle loading, water/liquid loading and air flow where there is no structural damage, one can add the contributions to filter efficiency by each of the three parameters. In practice the change in filter penetration (1 - efficiency) due to each parameter is added to the baseline penetration to yield the total penetration.

For dependent parameters it is necessary to consider the parameter interaction when computing filter efficiency. All of the parameters that affect filter strength or cause structural damage are dependent parameters because the filter damage or weakening from one parameter affects the amount of damage or weakening from a second parameter. For example the minimum pressure drop (or flow) required to rupture a new HEPA filter is 37 inches [25] whereas the minimum pressure drop for a 15-19 year-old filter is 13 inches [16] Both filters had 0% efficiency after the test. Intermediate pressure levels are required to burst HEPA filters less than 15 years old. A different series of pressure and age parameters define the limits for each efficiency level between 0% and 99.9%. In his study of water effects, Ricketts [34] showed that a combination of factors leads to a greater loss in filter efficiency than the cumulative effect of the individual factors alone. This illustrates that filter efficiencies derived from studies on filter structural failures require combination of parameters and cannot be determined by individual parameters.

The primary difference in a matrix consisting of independent parameters for filter efficiency and dependant parameters for structural damage is the number of matrix elements required. For N independent parameters and 3 values of each parameter, the matrix consists of 3N elements. For N dependent parameters and 3 values of each parameter, the matrix consists of 3^N elements. The parameters that affect the HEPA filter efficiency directly are temperature, solid particle loading, water/liquid particle loading, and air flow. Of these, the solid particle loading is the only independent

parameter since the other parameters can also cause structural damage and hence are dependent parameters. The remaining dependent parameters are age, radiation exposure, seismic or rough handling, and chemical attack.. Thus for one independent and seven dependent parameters, the matrix consists of $3 + 3^7 = 1{,}380$ elements. For a matrix with 4 values for each parameter, the matrix would consist of $4 + 4^7 = 16{,}388$ elements. Even if the matrix were reduced to the minimum parameters of temperature, water loading, air flow, and age, the 3 value matrix would consist of 81 elements, and the 4 value matrix 256 elements. These are obviously very large matrixes.

We believe that it is not practical to pursue the development of a generalized matrix for computing the efficiency of HEPA filters under normal and DBA conditions because there are far too many elements, the experimental data is highly variable, and a separate matrix would be required for each design and construction of HEPA filter. Although a large number of studies are reported in the literature on the efficiency of HEPA filters under adverse conditions, all of the studies have been limited to one or two parameters, whereas most DBAs involve several parameters. There are few studies on evaluating filter efficiency for combination of parameters . As shown in our matrix examples, the large number of matrix elements is due to the combinations of different parameters.

In addition most of the literature data pertains to the standard deep-pleated HEPA filter using the standard glass fiber medium with aluminum separators. If other filter designs and/or other materials are used, then a different filter efficiency matrix is needed. There are a large number of different combinations of filter media, filter pleat configuration, sealant, separators, frame and gaskets. Each combination of components yield a HEPA filter with a unique set of responses to applied stress. Unfortunately, there exists very little data for other filter designs or other materials. The other filter designs occasionally used are the mini pleat, the cylindrical cartridge with radial pleats, the cylindrical cartridge made with standard deep pleats and media and the deep pleated separatorless filter. The use of glass cloth reinforced filter media dramatically increases the filter strength and the value of differential pressure required to structurally damage the HEPA filter.

The final problem for computing filter efficiency that applies to the matrix and all other methods is the variability of the data. For example, Johnson et al [16] exposed 15-19 year-old HEPA filters to simulated tornados and measured the differential pressure across the filters during the approximately 5 second air pulses. They tabulated the differential pressure at which the filter showed the first signs of damage and the pressure drop for total medium blow-out The data is retabulated here in Table 4

All of the six filters tested had the media blown out as shown in Figure 4 except for the filter in test 17. Although no efficiency measurements were made on the filters, we added a column in Table 4

for the estimated efficiency at the end of the test. Since the filters in tests 18-23 had no media left in the filter housing, the efficiency was 0%. The filter in test 17 had the medium pack torn loose from the filter housing, but did not blow out at 77.8 inches pressure drop, which was the maximum applied stress during the test. A more powerful air pulse was needed to blow out the medium. We estimated the efficiency of the partially damaged filter at 98% based on efficiency measurements made on filters showing similar structural damage.

Table 4. Differential pressure for initial and total damage to aged HEPA filers from Johnson et al [16]

	Differential Pressure (in. water)		Estimated Efficiency (%)
Test No.	Initial	Total	
17	25.5	77.8	98
18	--	15.5	0
19	3.9	18.8	0
21	3.3	56.8	0
22	3.3	13.6	0
23	2.8	45.7	0
Average	7.8	38.0	

Table 4 illustrates the problem with trying to establish an efficiency value for a HEPA filter subjected to a given stress that causes structural damage. There is an initial stress value at which the filter shows the first indication of structural damage. For the six filters tested, the initial stress values varies from 2.8 to 25.5 inches of water and has an average of 7.8 inches of water. The initial damage consisted of a small segment of a filter pleat torn open. Based on efficiency tests on other filters with similar damage, the efficiency at the initial damage was about 99%. As increasing stress was applied to the filter, the efficiency would decrease until it reached 0% at total medium blow out. The differential pressure at total medium blow out ranges from 13.6 inches to greater than 77.8 inches with an average somewhat greater than 38 inches.

Experimental Measurements Under DBA Conditions Is the Preferred Method, But There Are No Facilities

The preferred method for determining HEPA filter efficiency under DBA conditions is to conduct experiments on a scaled HEPA filtration system under controlled test conditions that simulate the DBA. The scaled down, 1,000 cfm HEPA filtration system would be mounted in a test duct that would contain the desired air flow or air pulse, temperature, and water content. All of the components of the HEPA filtration system would be included such as prefilters, water deluge system, mist eliminator, and one or

more stages of single HEPA filters. The test would consist of a series of exposures that simulate the specific DBA. For example a simulation of a fire DBA might consist of a smoke challenge followed by a temperature increase and then a water deluge spray. The test filter could be preconditioned with particle loading, chemical exposure, seismic stress, etc. as needed to represent the exposure history for the DBA. Several repeat test would be conducted to establish the variability in the data.

The experimental approach has many advantages over a system based on determining the filter efficiency from a match with a catalog of test conditions and the corresponding efficiencies. The most important advantage is that the experimental approach will yield efficiencies for experimental conditions that match the DBA conditions as close as is experimentally possible. The problem with the catalog approach is that most of the available data is limited to HEPA filter efficiency for a single exposure parameter and a single or limited range of parameter values. This is in contrast to most DBAs that consist of a series of concurrent and sequential parameters. The catalog matching approach will undoubtedly require approximations and assumptions because the catalog will not contain the large number of matrix elements needed to fully describe the particular DBA. Moreover the exact filtration components and HEPA filters can be used in the simulation tests. In contrast, the catalog approach would undoubtedly contain data for different HEPA filters and system components. Another advantage is that conducting simulations of DBAs on HEPA filtration systems would be less costly than an attempt to develop catalogs of all the possible matrix elements in each of the many different matrixes that correspond to specific HEPA filtration systems.

Unfortunately, there is no test apparatus in the U.S. that can conduct DBA simulations or generate DBA matrix elements for HEPA filtration systems. A special test apparatus would have to be built for either application. The existing U.S. test equipment is limited to studying single parameters such as high air flow (New Mexico State U.), smoke loading (LLNL), and the quality assurance tests of heated air, overpressure, and rough handling (Rocky Flats). The quality assurance test equipment at Rocky Flats is designed for specific test conditions as part of the filter qualification prescribed in MIL-F-51068 [39] and cannot be varied to provide a range of DBA parameter values. Although the European countries have test equipment that can evaluate HEPA filters under a wide range of values for specific parameters, they do not have an apparatus that can expose HEPA filters to multiple parameters that simulate DBA conditions. Germany has a test rig that can expose HEPA filters to high humidities and water sprays at temperatures up to 151 C and another system that can expose HEPA filters to high air flows. England and France have test rigs that can expose HEPA filters to various temperatures up to 400 C. Much of the data on HEPA performance under off-normal conditions was generated in these European facilities.

We Selected the Method Based on First Determining Structural Damage and Then Efficiency

Since it is not feasible to determine HEPA filter efficiency under DBA conditions using the generalized matrix method, and since no experimental facilities are available, we examined alternative approaches that could make use of the data base of HEPA filter performance under off-normal conditions. We focused our efforts on a method for separating out the many dependent parameters that affect filter strength or cause structural damage since these parameters are responsible for the large matrix for establishing HEPA filter efficiency . The filter efficiency would then be determined in two steps: first determine the filter efficiency for the parameters that cause structural damage and then add any contribution to the efficiency by the parameters in Table 1 for an undamaged filter.

The parameters that affect the filter strength and structure are temperature, water/liquid loading, air flow (pressure drop), age, radiation exposure, seismic or rough handling, and chemical attack. We have placed pressure drop in brackets following the air flow because the two parameters are directly proportional. Of these parameters only air flow (pressure drop) and seismic or rough handling can cause structural damage directly. The other parameters only weaken the filter. Thus higher air flows and resulting pressure drops will produce greater structural damage in the filter.

Particle loading is an important factor in determining the filter pressure drop, but is is not important for the threshold value of pressure drop at which the HEPA filter shows structural damage. A HEPA filter will suffer structural damage at the same pressure drop, whether or not particles are present. Particle deposits will increase the filter pressure drop and cause the threshold pressure drop to be reached at lower flow rates. However, the particle deposits will not lower the threshold pressure drop. Particle deposits were therefore not included in Table 5. None of the studies reviewed in this paper show that particle deposits have much of an effect on the structural strength of HEPA filters. Ricketts et al observed a 5% reduction in tensile strength of creased media [34]. Particle deposits also significantly increase the amount of water absorption on a filter, which subsequently decreases filter strength. However, this decreased strength is due to water, not particle deposits. Ricketts et al [34] found no significant difference in structural strength between HEPA filters with or without particle deposits during exposure to humid air and water sprays. Smith et al [40] observed a small increase in the pressure drop required to cause structural damage with HEPA filters loaded with salt particles. The salt deposits had formed a crust over the protective wire screen, thereby protecting the HEPA filter medium.

We wanted to construct a table showing filter efficiency at increasing levels of structural damage produced by increasing pressure drop for the parameters that weaken the filter. Unfortunately, the literature data contained little or no information on the efficiency of partially damaged HEPA filters. Most

of the studies pertaining to HEPA filter damage only measured the pressure drop as shown in Table 2 and not the efficiency . The literature data was also limited to a single pressure drop measurement. In most cases, the pressure drop data corresponded to the threshold or first intermediate damage level. However, in a few cases, the pressure drop corresponded to the to total medium blow-out.

The level of structural damage is important since the efficiency can vary from 99% for a partially damaged filter to 0% for a totally damaged filter. Generally, the reported pressure drop data on structural damage due to high air flow corresponded to a partially damaged HEPA filter. Johnson et al [16] reported the pressure drop corresponding to both the threshold and the medium blow-out as shown in Table 4. The threshold and medium blow-out data were derived from a motion picture camera synchronized with the pressure pulse of a simulated tornado. In all of the other studies, only the final condition of the filter was available. Unfortunately, except for the high temperature studies, few efficiency measurements were made on the partially damaged filters, and few quantitative descriptions of the filter damage were made.

Because of the limitations in the literature data, we selected the method of first determining the structural damage on the HEPA filter and then the efficiency. If the HEPA filter suffered significant structural damage , the filter would be assigned 0% efficiency. If the filter was not significantly damaged, then the efficiency would be determined from literature values of the efficiency.

Justification for Assigning 0% Efficiency To Partially Damaged Filters: The selection of the filter efficiency that corresponds to the reported pressure drop for partially damaged HEPA filters in Table 2 was a major step in the development of a method for computing filter efficiency under DBA conditions. We decided to set the efficiency for all damaged HEPA filters, whether partially or fully, to 0% for several reasons. First, there is very little data on the efficiency of partially damaged HEPA filters. The few reported efficiency measurements show partially damaged HEPA filters have high efficiencies . For example, a HEPA filter with a 5 inch slit along one pleat of the HEPA medium has an efficiency of 99%, while a filter that has the media pack loose within the frame and slightly pushed out of the frame has an efficiency of 94% [41]. The primary reason for selecting 0% efficiency for partially damaged HEPA filters is that there is a large variability in the differential pressure required to blow out the filter medium as seen in Table 4 for high air flow (from 13.6 to greater than 77.8 inches). This variability makes it prudent to select a conservative value of differential pressure at which the filter medium blows out of the filter.

Another important reason for selecting conservative values is that most DBA s consist of multiple challenges, whereas the available data generally applies to one parameter, not counting the differential pressure, that is responsible for the damage. We have assumed that using conservative values for the

pressure drop that is required to blow out the filter medium due to a single parameter will compensate for the additional damage caused by multiple parameters.

Establishing Threshold Values of Differential Pressure for Structural Damage and 0% Efficiency:
The threshold value of the differential pressure for structural damage depends on the structural strength of the filter, which in turn is a function of filter age, radiation exposure, corrosive chemical exposure, temperature and moisture exposure. These parameters cause a lowering of the threshold differential pressure.

Other parameters such as high air flow or pressure pulses define the filter differential pressure. Filter clogging by aerosols and water also contribute to an increased differential pressure by reducing the filter permeability in addition to lowering of the threshold differential pressure.

The effect of the various parameters on the threshold differential pressure for causing structural damage is estimated from the literature data in Table 2. From these values it is possible to generate Table 5 relating the threshold differential pressure for each environmental exposure that can structurally damage the filter. The data in Table 5 is applicable to the standard HEPA filter used in DOE facilities: a deep pleated filter with aluminum separators, and an organic sealant. Although other filter designs are available and periodically used in DOE facilities, they have lower threshold values for differential pressure.

Conservative values for the different parameters have been included in Table 5 to simplify the determination of structural damage. In most cases, the conservative value is the lowest experimental or computed value believed to cause structural damage. However for moisture exposure, the lowest values were not used because they seemed unrealistically low. Ricketts et al [34] found three filters out of 29 tested that had structural failures when wet and exposed to less than 10 inches of water: one failed at 3.6 inches and two at 7 inches. Although the authors did not state the extent of the damage at 3.6 and 7 inches, they were most likely tears in the media pleats based on the photographs shown in the paper. The authors did state they measured the "differential pressure at which the first visible structural damage to the filter medium appeared.": This damage would correspond to the initial damage shown in Table 4. Note that the differential pressure at which medium blow out occurs is more than 4 times the value at the initial damage. If a similar ratio applies for the wet filter, then the lowest pressure drop for blowing out the wet filter is 14.4 inches. The conservative value for the threshold differential pressure for wet filters was therefore set at 10 inches of water.

Table 5 Threshold Values* of Differential Pressure Required to Structurally Damage the Standard HEPA Filter

Parameter	ΔP Threshold*, inches w.g.
Baseline (new filter, normal conditions)	37
Age (15 years or older)	13
Radiation (6×10^7 Rad)	18
Chemical (HNO_3, HF)	0-37
Temperature	
less than 200°C, (392°F)	37
200-300°C, (192-572°F)	
10 minutes	33
1 hour	30
10 hours	22
300-400°C, (572-752°F)	15
400-500°C, (752-932°F)	10
Moisture	
wet filter, (greater than 95% relative humidity)	10
dry filter, previously wet	22
Air pulse from explosion	29

* These values represent the most conservative values (except for moisture) taken from an analysis of experimental studies reviewed in this report and summarized in Table 2

We also did not select the lowest value for structural damage at 500 C because 8 inches only produced tears in the media, but did not cause total filter blow-out [32]. In a second test, Pratt [32] observed a total filter collapse at 20 inches after exposure to 500 C. We selected 10 inches of water as a compromise between the conservative value based on partial filter damage in one of the two tests conducted by Pratt [32] and the wide range in values observed in Table 2 for HEPA structural damage.

It is important to note that the parameters in Table 5 represent exposures to a HEPA filter during its life, not just during the DBA. All of the parameters must be taken into account throughout the life of the filter in addition to the parameters applicable to the specific DBA. The damage caused by the different parameters is not reversible except for water, and that is only partially reversible. A previously wet HEPA filter will have a 40% reduction in its tensile strength [34].

A major concern with the use of Table 5 for determining filter survivability is the question of whether multiple stresses act independently and are additive. Unfortunately there are not sufficient data available to answer this question definitively. Nearly all previous studies on HEPA stress were done with single stresses. However, it is possible to make general observations. Age, radiation and temperature

all reduce the tensile strength of the filter medium. If a combination of these stresses were applied to the medium, the tensile strength of the medium would decrease up to the limiting value where the binder that holds the glass fibers together is no longer effective. Thus the three parameters can be viewed as acting independently and can be added up to the limiting strength of no binder. The three stresses also cause a loss of water repellency, which makes the filter absorb water and loose strength. Since all of the stresses in Table 5 degrade the medium strength by attacking the binder, there is a lower strength limit for these parameters. An approximate value is 10 inches of water.

In contrast, the attack by chemicals such as HNO_3 and HF attack the glass fibers directly and can continuously degrade the media strength to the point where it collapses under its own weight. Chemical attack must be treated as a separate case from the other stresses.

The effect of temperature on the structural damage to the HEPA filter depends on both the temperature value and the exposure time. Figure 6 shows the percent of the initial tensile strength and burst strength for HEPA filter media samples after ten minute exposures at the indicated temperatures [30]. The burst strength and tensile strength follow a common curve because the filter media are held together with similar acrylic binders [30]. Breschi et al [26] found that at 200 C, the filter medium becomes

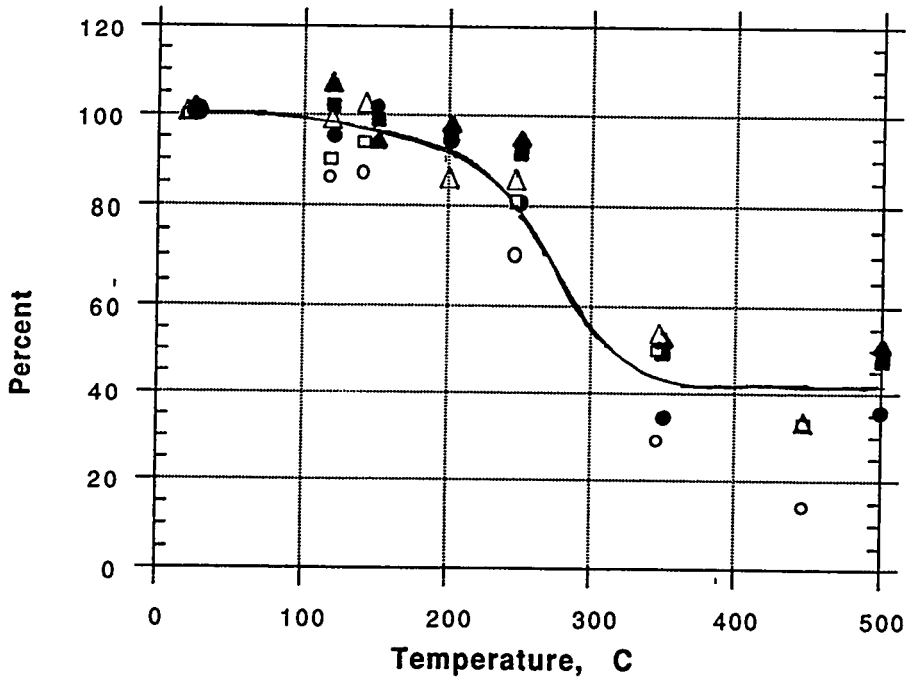


Figure 6. Percent of initial tensile strength and burst strength for three different HEPA media as a function of temperature. Open points represent tensile strength and solid points represent burst strength. Data replotted from Hamblin and Goodchild [30].

brown, and electron micrographs show the microfibers are covered with small lumps and deposits due to melting of the binder. Raising the temperature to 300-400 C cleans the microfibers from all deposits [26]. Gilbert et al [41] also found that the acrylic weight loss occurs between 300-400 C.

Figure 7 shows the percent of the initial tensile strength and burst strength as a function of time for HEPA filter media samples exposed to 120 C and 250 C [30]. At 120 C, the media show a negligible decrease in strength over time. However, at 250 C, the media show a gradual, but significant, loss in strength. Hamblin and Goodchild [30] also found that all the binder is lost in less than 10 minutes at 400 C.

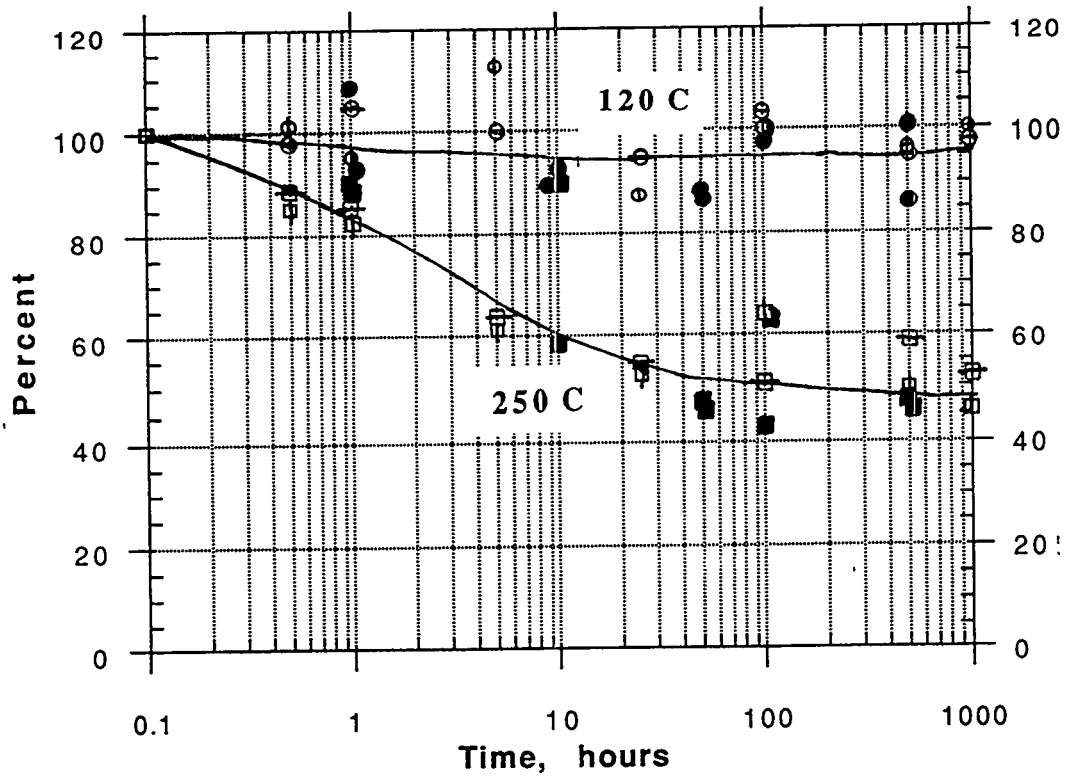


Figure 7. Percent of initial tensile strength and burst strength for three different HEPA media as a function of time after exposure to 120 C or 250 C. Open points represent tensile strength and solid points represent burst strength. Data replotted from Hamblin and Goodchild [30].

We have used these studies to generate the temperature and time data in Table 5. Since there is no significant deterioration at temperatures below 200 C, the threshold value of differential pressure will be the same as the baseline filter. For temperatures between 200-300 C, we used the curve for 250 C in Figure 7. The estimated percent of initial strength at 10 minutes, 1 hour, and 10 hours are approximately 90%, 80%, and 60% respectively. These percent values were multiplied by the baseline differential pressure of 37 inches and yield 33, 30, and 22 inches respectively. For temperatures above 300 C the deterioration in medium strength is essentially complete within ten minutes as seen in Figure 6. Thus, the differential pressure for temperature exposures between 300-400 C is computed from 40% residual strength and the baseline 37 inches to yield 15 inches. Since the deterioration occurs in less then 10 minutes, there is no time associated with this exposure.

The temperature can also cause the sealant that holds the media pack in the frame to char and sometimes burn and break loose from the filter frame. High temperature can also cause the filter medium to develop tears, especially when using steel frames due to the difference in thermal expansion between the steel frame and the medium pack. The additional failure modes lower the differential pressure for structural failure. These type of failure modes are not significantly affected by the other parameters in Table 5

Thus for determining the threshold differential pressure for structural damage, we can use the following prescription: For single stresses, the threshold differential pressure is read from the applicable entries in Table 5. For multiple stresses, whether sequential or simultaneous, the lowest differential pressure threshold is selected. If the environmental parameters are not well defined for a given DBA then the threshold value of 10 inches is used.

In order to use Table 5 for estimating structural damage, the pressure drop for a particular filter must first be determined for the environmental exposure in the DBA. The driving force for the filter differential pressure may be a fan, a tornado, an explosion, a pressurized gas release, or other sources of negative or positive pressures. For a plugged filter, the differential pressure can be approximated by the source pressure since nearly the full load will be applied across the filter. Cases were the filter may approach full plugging are during high smoke concentrations from oxygen starved fires [42] and water exposure on clogged filters [33,34].

For a clean or partially plugged filter, the differential pressure will be less than the source pressure and must be estimated. The most common source of partial filter plugging is particle deposits from atmospheric dust or plant operations. The data compiled by First and Price [43] and Novick [44] can be used to approximate the particle loading on HEPA filters. All new HEPA filters must have a pressure

drop less than one inch of water at 1,000 cfm. At higher air flows the pressure drop increases as shown in Figure 8. This figure can be used directly to determine the pressure drop across a clean filter if the air flow is known. Figure 8 can also be used for computing the pressure drop of partially plugged filters at a second flow if the filter pressure drop is known at one flow.

$$\Delta P_2 = \Delta P_1 \times \Delta P_{C2}/\Delta P_{C1}, \quad (1)$$

where ΔP_2 is the pressure drop of the partially plugged filter at the second flow

ΔP_1 is the pressure drop of the partially plugged filter at the first flow

ΔP_{C1} is the pressure drop of the clean filter at the first flow from Figure 8

ΔP_{C2} is the pressure drop of the clean filter at the second flow from Figure 8

For an illustration, assume that a HEPA filter is loaded with particles and has a pressure drop of 4 inches at the standard flow of 1,000 cfm. If the flow suddenly increases to 10,000 cfm, the resulting pressure drop would be 48 inches (4 X 12/1). We obtained the 12 inches and 1 inch from Figure 8.

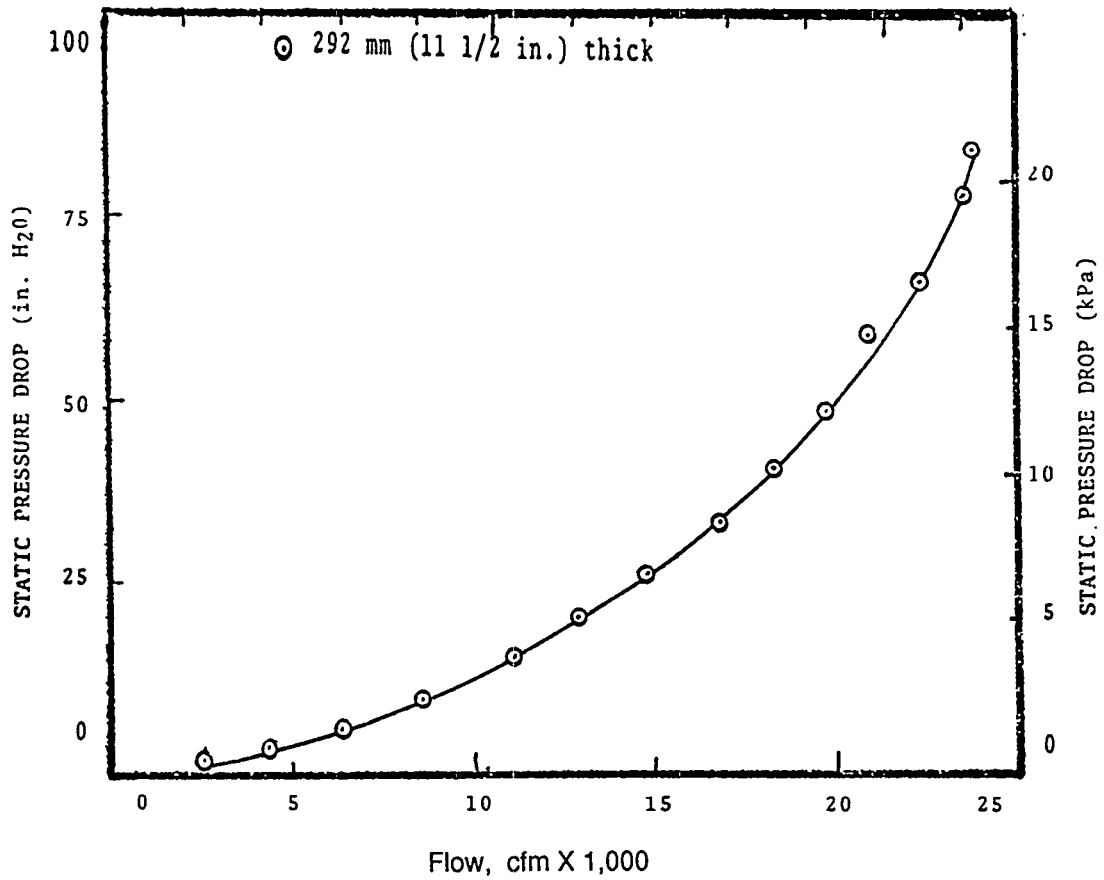


Figure 8. Pressure drop as a function of flow rate for a clean HEPA filter from Gregory et al [45].

Once the appropriate pressure drop is determined for the HEPA filter under DBA conditions, the value can be compared to the applicable threshold values listed in Table 5. If the estimated differential pressure is less than the threshold value, then the HEPA filter will not be structurally damaged, and the filter efficiency can be determined in the next section. However, if the differential pressure exceeds the threshold values, then the filter will be structurally damaged and will be assigned a 0% efficiency under DBA conditions.

We can also use Figure 8 and Equation 1 to compute the air flow through a partially plugged filter if we know the initial pressure drop and flow rate and the final pressure drop. This computation is made easier by rearranging Equation 1 as follows:

$$\Delta P_{C2} = \Delta P_{C1} X \Delta P_2 / \Delta P_1 , \qquad (2)$$

For example, if the partially plugged HEPA filter has a pressure drop of 4 inches at 1,000 cfm and is then subjected to a pressure drop of 48 inches, the pressure drop for the clean filter will be 12 inches (1 x 48/4). The corresponding air flow is then read from Figure 8 to be 10,000 cfm.

Once the appropriate pressure drop is determined for the HEPA filter under DBA conditions, the value can be compared to the applicable threshold values listed in Table 5. If the estimated differential pressure is less than the threshold value, then the HEPA filter will not be structurally damaged, and the filter efficiency can be determined in the next section. However, if the differential pressure exceeds the threshold values, then the filter may be structurally damaged and will be assigned a 0% efficiency under DBA conditions.

Determination of HEPA Filter Efficiency Under DBA. If the HEPA filter is not structurally damaged, then the filter efficiency can be estimated from the filter efficiency tables and from the conditions during the DBA. Table 3 summarizes the effect of various parameters on the penetration of HEPA filters obtained from our literature review. However, before the data can be used for determining filter efficiency under DBA conditions it must be consolidated and simplified.

One consolidation of Table 3 is to treat the air flow and air pulse as one parameter since they are not independent. This is possible because the particle capture mechanism is the same for both processes. In our combined air flow and pulse, the air pulse is viewed as a higher flow for a short time. We were also able to derive an empirical equation for the maximum filter penetration as a function of air flow based on the data from VanOsdell et al [15]. The maximum penetration point for velocities of 0.5, 1,

2, 5, 10, and 20 cm/s were taken from Figure 3 and regraphed in Figure 9. The straight line through the data points fit the following equation:

$$P = 3.65 \times 10^{-4} V^{2.58} \ \%, \qquad (3)$$

where P is the penetration in percent and V is the velocity in cm/s.

Since air flow is a more practical unit of measure than air velocity through the medium, we can substitute air flow/filter area for velocity in Equation 3. If we assume the average deep-pleated HEPA filter has 200 square feet of media then Equation 3 becomes

$$P = 3.65 \times 10^{-4} (\text{flow}/400)^{2.58} \ \%, \qquad (4)$$

where flow is in cfm.

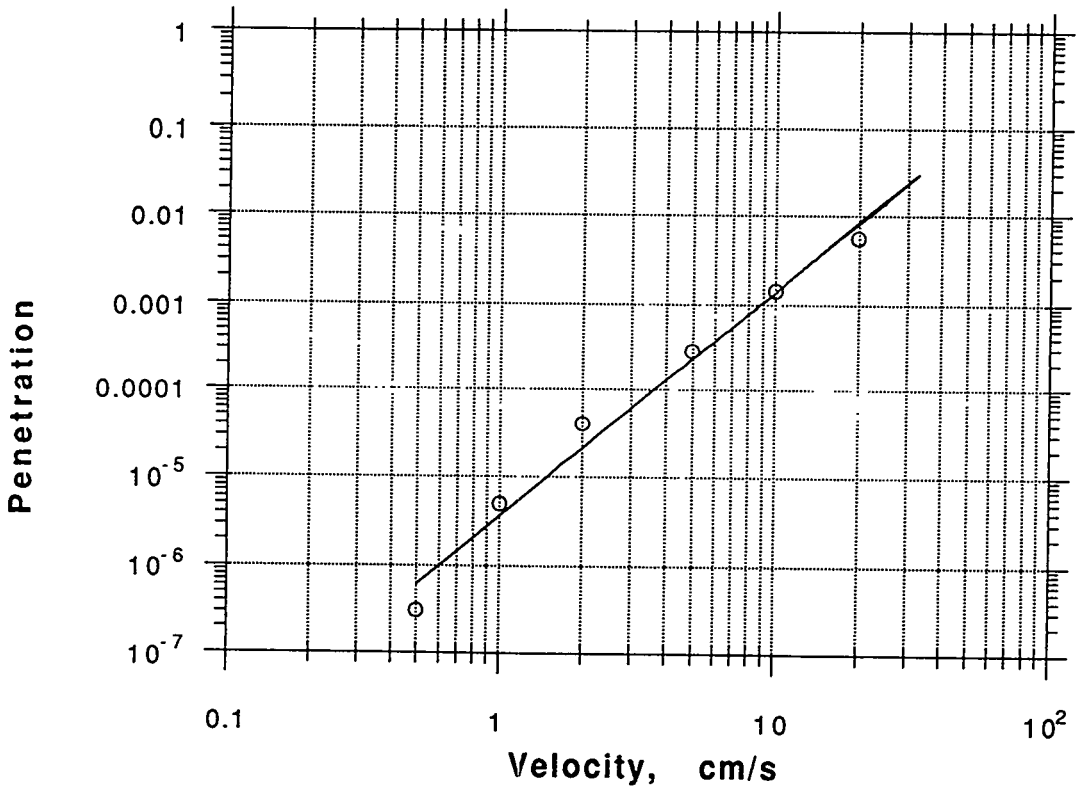


Figure 9. Maximum filter penetration as a function of air velocity replotted from Figure 3

Equation 4 also applies to an air pulse. However, in order to obtain filter penetration from the equation, it is necessary to determine the air flow through the filter that corresponds to the applied differential pressure. For example, a 1 psi (27.7 inches) air pulse applied to a clean filter corresponds to

an air flow of 15,000 cfm as determined from Figure 8. Computing the penetration at 15,000 cfm with Equation 4 yields a penetration of 4.2%. Gregory measured 0.1% penetration for 0.46 μm latex particles in a 1 psi pulse simulating a tornado [25]. He also measured 0.9% penetration for previously deposited 0.46 μm latex particles in a simulated explosion shock (1 psi estimated). Since the penetration for 0.1 μm particles is more than ten times the penetration for 0.46 μm particles (see Figure 5), Equation 4 is in general agreement with experiment.

We should point out that the increasing penetration with increasing air flow is not in disagreement with previous studies that show an initial increase and then a decrease in penetration with increasing air velocity [18]. In the previous studies, the particle size remained constant as the air flow increased. The penetration initially increased because of decreasing time for diffusional capture. With increasing air flow, the particle inertia increased and resulted in decreasing penetration. The observed penetration increased and then decreased as the dominant capture mechanism switched from diffusion to inertia. However, for the maximum penetration, the particle size decreases with increasing air velocity and therefore extends the range where diffusion dominates and results in increasing penetration with increasing air flow. The increasing contribution due to inertia is also seen in Figure 9 by a small deviation from the straight line to lower penetration levels.

We then prepared Table 6 for computing filter efficiencies under DBA conditions by consolidating the data from Table 3 and making simplifying assumptions. The primary assumption was to use the most conservative penetration value for each parameter. We also combined the penetration increase due to water accumulation and oil accumulation into a single liquid accumulation term. All of the entries in Table 6 represent penetration values to be added to the baseline 0.1% penetration. Equation 4 was used in place of the data for air flow and air pulses. Although the entry for HF corrosion shows only a single data, the penetration increases continuously with increasing HF exposure. Unfortunately there were no additional data available. For the temperature parameter, we reduced the number of temperatures and used the highest penetration value. No credit was given for the initial decrease in penetration from 25 to 200 C, as shown in Figure 1, because the change in penetration was small compared to 0.1%. We also used the maximum allowed penetration of 3% for exposure to 371 C (700 F) for 5 minutes as specified in MIL-F-51068 [39].

The final entries to complete Table 6 were solid and liquid loading. For solid loading we indicated -0.05% since it improves filter efficiency. For liquid aerosols, the penetration increases by 1.3 times the product of the initial penetration and the ratio of the final pressure drop to the initial pressure drop. The increased penetration is due to liquid build-up that blinds portions of the medium and hence increases the

air flow through the remaining unblocked portions of the medium. The higher air flow causes the increased particle penetration.

Table 6 Effect of Environmental and Filter Parameters on the Aerosol Penetration through HEPA filters.

Parameter	Filter Penetration*
Baseline	0.1%
HF corrosion	
1,500 ppm-hr.	0.1%
Temperature	
less than 200°C (392°F)	0.0%
200-300°C (392-572°F)	0.1%
300-350°C (572-662°F)	0.4%
350-500°C (662-932°F)	3.0%
Particle Deposits	
Solids	-0.05%
Liquids	$1.3\ P_i\Delta P/\Delta P_i$ %
Air Flow (Air Pulse)	$3.65 \times 10^{-4}(\text{flow}/400)^{2.58}$ %

* These values represent the most conservative values taken from an analysis of experimental studies reviewed in this report and summarized in Table 3.

Table 6 can be used to estimate the aerosol penetration of a HEPA filter under the conditions expected in a given DBA. Once the parameter values that apply for a given DBA are established, the corresponding penetrations can be used for determining the HEPA filter penetration. As we have previously discussed, since the penetrations due to the different parameters are independent of each other, then the filter penetration for a combination of parameters will equal the sum of the penetrations for the individual parameters.

Thus for determining the aerosol penetration from the data in Table 6, we use the following prescription: For single stresses, the penetration is read from the applicable entries or computed from the flow. For multiple stresses, the penetration from each of the stresses are added to yield a total filter penetration.

V. Computing the Efficiency of HEPA Filters Under DBA Conditions

General Procedure

The determination of the efficiency of the HEPA filters under DBA conditions is a three step process.

(1) Determine the values of the temperature, moisture, radiation, and chemicals to which the final exhaust HEPA filters are exposed during normal operations and during and after the DBA. Standard engineering practices should be used in determining these parameter values. Also assume a filter age that represents the maximum age of filters in the facility. The most important parameter to be determined is the filter pressure drop during and after the DBA. Guides for estimating the pressure drop are presented in this paper.

(2) Compare the environmental and filter parameters listed in step (1) to the corresponding values in Table 5 that show the threshold value of differential pressures that would cause structural damage to the HEPA filter. For multiple parameters, use the lowest differential pressure. Compute the differential pressure for the HEPA filter under the DBA conditions assuming maximum particle loading. If the filter differential pressure under DBA conditions is greater than the threshold value shown in Table 5, then the HEPA filter is assumed to be damaged, and the allowed efficiency credit is 0%.

(3) If the differential pressure is less than the threshold value shown in Table 5, then the filter is considered to be undamaged, and the filter penetration is determined from the filter penetration values shown in Table 6 that correspond to the environmental conditions in the DBA. For a DBA challenge with multiple environmental parameters, the total HEPA penetration is obtained by adding the penetrations from the individual environmental parameters.

Examples

Example 1: Assume a two-stage HEPA filter system with a demister system, a blower with a 15 inch w.g. vacuum and a DBA that consists of a hot fire followed by activation of a fire protection water deluge system to protect the first stage HEPA filter. The temperature at the first filter is determined to be 250°C and the second 200°C. In this scenario, the first filter would be partially loaded, but not completely plugged, due to the smoke. Using the data in Table 5, each filter has a maximum differential pressure threshold of 33 inches, beyond which value the filters will rupture. Since the fan can only pull 15 inches w.g., there is no damage to the filters at this point. However, once the water deluge is turned on, the first filter becomes wet and becomes plugged. The threshold value for filter rupture is now reduced to 10

inches w.g. because the filter is wet. Since the blower vacuum exceeds the minimum value of differential pressure required to damage the filter, the first filter will be structurally damaged. This follows because the full vacuum of the blower will be applied across the first stage HEPA filter. If the smoke from the fire and the water spray continues, then the second filter will soon plug and also rupture. The assumed efficiency for each ruptured filter is 0%, thus resulting in 0% efficiency for the two filters.

Example 2: Assume the same conditions as in example 1, but the blower can only pull a 10 inch w.g. vacuum. In this case the threshold value for the differential pressure is still 33 inches w.g. for each of the two filters during the high temperature exposure. After the water spray is turned on, the differential pressure threshold for the first filter is 10 inches w.g. Since the fan can only pull 10 inches w.g., neither filter is damaged, but the first filter is plugged and the air flow is reduced to a very low flow. The first HEPA filter has a penetration of 1.4% according to Table 6 because 1.3% penetration due to moisture accumulation is computed from the liquid particle deposits and added to the baseline penetration of 0.1%. Since the second HEPA filter has the baseline 0.1% penetration, the total penetration for the two filters is 0.0014%, or an efficiency of 99.9986%. However, the process air ventilation system is effectively shut down, and contaminants may leak out into the building rooms from the containment structure, ducts and housings.

Example 3: A two-stage HEPA filter that has a flow of 1,000 cfm and an initial pressure drop of 4 inches w.g. for the first stage and 1 inch w.g. for the second stage is subjected to a negative 75 inch w.g. tornado condition. Assume that the calculations from the EVENT code [45] predict a pressure drop of 50 inches w.g. across both filters and a flow of 10,000 cfm. Under these conditions the first filter will have a pressure drop of 40 inches and the second filter 10 inches w.g. According to the data in Table 5, the first filter will exceed the threshold differential pressure of 37 inches w.g. and will be structurally damaged. After the first filter is destroyed, it will no longer impose a restriction on the air flow and a new flow computation is required. Assume that the new computations using EVENT show the remaining HEPA filter has a pressure drop of 25 inches w.g. and a flow rate of 15,000 cfm. Under these conditions, the second HEPA filter will remain undamaged. The damaged filter will be assigned 0% efficiency, and the efficiency of the second filter determined from Table 6. Assuming the relationship between air flow and penetration in Table 6, we estimate that 15,000 cfm air flow will result in a penetration of 4.2%. After the tornado pulse has passed, the filter penetration will return to the baseline value 0.1% (99.9% efficiency). Thus the efficiency of the HEPA filter system is 95.8% during the tornado and 99.9% after.

VI. Conclusion

We have established criteria for calculating the efficiency of HEPA filters in a DOE nonreactor nuclear facility during and after a Design Basis Accident (DBA). Previous efforts of selecting a single efficiency value for all accidents were unsatisfactory because the potential DBAs vary significantly among different facilities and operations and result in large variations in computed filter efficiencies, primarily due to the deterioration of the HEPA filter. The deterioration of the filter efficiency depends on the exposure parameters; in severe exposure conditions the filter will be structurally damaged and have a residual efficiency of 0%. The large variation in filter performance makes it necessary to determine filter efficiency on a case-by-case basis. It also prevents the use of a limited number of tests to estimate the efficiency for a given DBA.

Despite a significant amount of data in the literature on the performance of HEPA filters under off-normal conditions, it is not sufficient to cover the full matrix of different values and combinations of the following parameters that weaken the filter or cause structural damage: age, moisture, chemical attack, high temperature, overpressure, seismic and rough handling, liquid and solid particle loading, and radiation. The primary reason for the large number of combinations of parameters is that the parameters are highly dependent and cannot be separated. Because of the large number of possible DBA conditions, the preferred method for determining the efficiency of HEPA filters under DBA conditions is to conduct experiments in a test stand that can simulate the DBA conditions. Unfortunately since this test stand is not available, a secondary method is required

The method for computing filter efficiencies that we selected was chosen to be consistent with the limited data available. We approximated the performance of HEPA filters under DBA conditions by separating the effect of filter structural damage from filter efficiency. In this fashion, we were able to significantly reduce the number of required test parameters to match the available literature data. Even with this approximation, there are large gaps and limitations in the data that introduce significant error in the estimates of HEPA filter efficiencies under DBA conditions. Because of this limitation, conservative values of filter efficiency were chosen when there was insufficient data.

The method we selected for estimating the efficiency of the HEPA filters under DBA conditions involves three steps. In the first step, the filter pressure drop and environmental parameters such as temperature and moisture are determined during and after the DBA.. The second step consists of comparing the filter pressure drop to a set of threshold values above which the filter is structurally damaged. If the filter pressure drop is greater than the threshold value, the filter is structurally damaged and is assigned 0% efficiency. If the pressure drop is less, then the filter efficiency is determined in a third

step where the efficiency is determined from literature data that matches the same exposure conditions. The efficiency of the HEPA filters within DOE facilities should be determined on a case-by-case basis.

VII. Acknowledgement

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DISCUSSION

WEIDLER: I heard you say it is going to be a DOE proposed standard. Is the standard proposed only for DOE, or would it be for the rest of the industry?

BERGMAN: I believe it could be for whomever feels it is applicable, but I will let Jim Leonard respond further.

LEONARD: The intent is that it would apply to DOE.

MISHIMA: I noticed that you have a value for air pulse, and you also have a value for failure. Am I to infer that the failure pressure, e.g., 37 in.w., is a static load and the air pulse is over some time duration?

BERGMAN: What Joe Mishima has just referred to is air flow and air pulse. Professor First was very gentle with me in pointing out that I am counting too many marbles at the same time; all of these things are the same phenomenon. What you have is air that creates a force that is identical whether it is over a few milliseconds or over a longer period of time. I have not made the point in this presentation because of time, but the equation applies over a millisecond. There is only one penetration episode reported in the literature. The calculation came up with 30% penetration across the HEPA filter during a tornado pulse. Another, reported by the Las Cruces people, reported 32-35%. Therefore, the damage applies to both pressure types.

MISHIMA: I have seen the data you are talking about by Paul Smith at Las Cruces. They seem to indicate that there is a difference between the failure mechanism from a pulse, essentially a shock, and one from a deflected blast.

BERGMAN: I understand that you are saying that I left out the air pulse and you are suggesting I should put it in. I am not quite sure that is correct. I will have to look at the details. Let's compare, for example, a shock pulse that gives a pressure differential of 30 in.w. and a static flow that gives a differential of 30 in.w. I am not sure that the shock flow will have a different result with respect to rupturing the filter than the steady flow of 30 in.w. Logically, you could figure you have to account for inertia, and you may have some other factors. We are at such a crude stage in the study that I did not see any difference. That is to say it is within the same order of magnitude. Refinements are called for. I will make sure I include that in the final paper.

JENKINS: I am sure this is interesting from a theoretical point of view, but I am trying to think of practical applications for our plant, given some accident scenario that we might be in. By that I mean, I am not sure what degree of accuracy you might achieve. You are looking to predict off-site doses under accident conditions. Since we generally have actual post-filter radiation levels, what is the application of this theoretical approach?

BERGMAN: The application of the paper is for computing the efficiencies of HEPA filters in nonreactor nuclear facilities during and after a design basis accident. The purpose

of the method described in this paper is to make the computations before an accident occurs. You are correct in implying that the method has no purpose after an actual accident occurs. Actual measurements are always better than calculations. Of course, it would be instructive to compare the predicted and actual efficiencies in order to improve the predictions. The method was intended for nonreactor applications where there are many different DBAs, in contrast to reactors that have a single DBA defined by Nuclear Regulatory Guide 1.52.

WEBER: Is there a cumulative effect of certain excessive exposure parameters, such as temperature or moisture, that might accelerate aging?

BERGMAN: Concerning cumulative effects, one of the most common events is when people repeatedly wet the filter and dry it. They think no detrimental effect occurs. I should acknowledge that 90% of the data on this comes from Mr. Wilhelm's group at KfK. The U.S. does not have appropriate facilities for conducting these experiments. I am sad to learn the facilities in Germany have been dismantled. But Mr. Wilhelm pointed out many years ago that after you wet a filter, dry it, wet it again, it will retain only half its strength.

WILHELM: There is a very strong effect of operational time on the behavior of HEPA filters under wet conditions. We found that the binder starts to get soft and dissolves at high differential pressures. For longer times, there is also a time dependency, but it may not be very much. There is a very strange time dependency on the influence of water on HEPA filters. After I heard what you said, I would be very much interested to assist you. If you would send me your papers, I think I could send results that are not published. We worked for years and years on that project with one American scientist, Dr. C. I. Rickerts, who is now at Las Cruces.

BERGMAN: Thank you for that input. We have incorporated many of your outstanding research findings in our analysis. Water exposure is unquestionably an important factor leading to the deterioration of HEPA filters and their degradation to 0% efficiency when coupled with higher pressure drops. Unfortunately, this condition is precisely what may occur during accidental fires when the water fire-suppression system is used on the filters. The filters become weak and plug with water. The full vacuum of the blower is then applied across the HEPA medium which may then blow out. I recall a fire at Rocky Flats in 1980 where an entire bank of filters was blown out in such a fashion.

WEBER: Is there a cumulative effect of transient conditions such as excessive temperature or moisture which could accelerate aging and affect pressure drop thresholds? In applying your results to filters in service, is the existing detection equipment adequate to monitor and record transients significant to pressure drop and other thresholds?

BERGMAN: Regarding the first question, there are indications that there may be cumulative effects, but the data is insufficient to make definitive statements. For example, studies have shown that a dry filter medium that was previously wet has 40% lower tensile strength. We do not know what a sequence of repeated applications would do. For

temperature, the effects appear to be both time and temperature dependent. It takes a given exposure time at a given temperature to have a complete effect. For example, applying 350° C for a short time would show less damage than the same temperature applied for a long time. In this case, multiple applications of the same temperature would appear to be cumulative, at least until the complete damage is reached. After the complete damage is reached, there will be no further effect. As the temperature is raised, there is a new level of damage. The greater the damage to the filter, the lower will be the differential pressure for medium blow out.

Regarding the second question, the existing detection equipment generally consists of radiation detection equipment that can alarm quickly and temperature sensors that activate fire suppression systems quickly. Although HEPA filters generally have a Magnehelic differential pressure sensor, they are generally not instrumented for alarm or other emergency responses. The temperature sensors have a strong effect on the HEPA performance by activating the water suppression system that makes the HEPA filters wet. Some facilities do not have the water suppression automatically turned on because of the concern with HEPA filter damage.

TARTAGLIA: An example was given on how to apply the results of the paper. The example examined the estimated filter efficiency drop following a fire and subsequent water inundation of the filter. Nuclear power plants do not consider a fire as a design basis accident (DBA), nor is a fire considered to occur concurrent with a DBA. Can you provide an example where this paper applies to nuclear power plants?

BERGMAN: The paper is intended for non-reactor facilities. Nuclear Regulatory Guide 1.52 applies to nuclear power plants. It is not our intention to compete with or supplant the R. G. 1.52.

PERFORMANCE OF HEPA FILTERS UNDER HOT DYNAMIC CONDITIONS

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ABSTRACT

Accidents in nuclear facilities involving fires may have implications upon the ventilation systems where high efficiency particulate air (HEPA) filters are used to minimise the airborne release of radioactive or toxic particles.

The Filter Development Section at Harwell Laboratory has been investigating the effect of temperature on the performance of HEPA filters under hot dynamic conditions [1] for a number of years. The test rig is capable of delivering air flows of 1000l/s (at ambient conditions) at temperatures up to 500°C, where measurements of the penetration and pressure drop across the filter are obtained.

This paper reports the experiments on different constructions of HEPA filters; rectangular and circular. The filters were tested at an air temperature of 200°C for up to 48 hours at the rated airflow to assess their performance. The penetration measurements for rectangular filters were observed to be below 0.02% after prolonged operation. In a number of cases, holes appeared along the pleat creases of circular filters although the penetration remained below 1%. The sealing gasket for these filters was noted to deform with temperature, permitting a leakage path. A prototype high strength circular filter was evaluated at temperatures of up to 400°C with a penetration less than 0.65%.

1. INTRODUCTION

High Efficiency Particulate Air (HEPA) filters are used in Nuclear plants to minimise the airborne release of radioactive or toxic particulate to the environment. Most applications involve the filtration of air at ambient temperature. Fires may occur either in the plant or the ventilation system and, under these circumstances, the filters may be subjected to high air temperatures.

The Atomic Energy Standard Specification (AESS) [2,3,4] for rectangular and circular filters requires a static oven test, where, for a high temperature (Type II) filter, the penetration measured using the sodium flame test, [5] must be no greater than 2% after being heated to 500°C for 10 minutes. The filter is then subjected to a burst strength test, in which it is dust loaded to a pressure drop of 3kPa at the rated flow. However, this test does not accurately simulate the type of accident condition likely to occur in practice, where a filter may be subjected to hot gas containing a significant amount of smoke.

Investigations of the mechanisms of filter failure [6] with dry air have revealed failures due to either buckling of the filter pack or swelling of individual pleat channels leading to splitting. However, at elevated temperatures, degradation of the paper binder [7] can lead to dramatic reductions in the tensile strength of the medium and therefore failure may occur at reduced pressure drops.

The performance of HEPA filters under hot dynamic conditions have been experimentally studied by a number of workers [8,9] who noted that the filters performance deteriorated with increasing temperature, due primarily to cracks in the sealant between the pack and casing.

This report describes the series of tests performed on clean HEPA filters under hot dynamic conditions for operation at prolonged periods. The results are also presented for a prototype filter using a high strength medium, derived from the addition of a glass fibre scrim.

2. APPARATUS AND TEST PROCEDURE

2.1 Apparatus

A flow diagram of the Hot Dynamic Test Rig is shown in Figure 1. The rig has been designed as a "once-through" system where air, after passing through the filter, is exhausted to the environment. Air is drawn through an inlet leg which incorporates an orifice plate for measuring the air flow. The air then passes through a 750 kW direct heating gas burner and then via a thermally insulated 35cm duct to the filter housing. This duct section contains an aerosol injection point just after the gas burner, and a Stairmand disc to aid mixing of the test aerosol at the sampling point before the filter housing. The filter housing may be either rectangular, or circular and these were uninsulated to enable realistic temperature gradients to be established in the filter housing. Instrumentation is provided to measure the inlet air temperature to the filter housing and pressure drop.

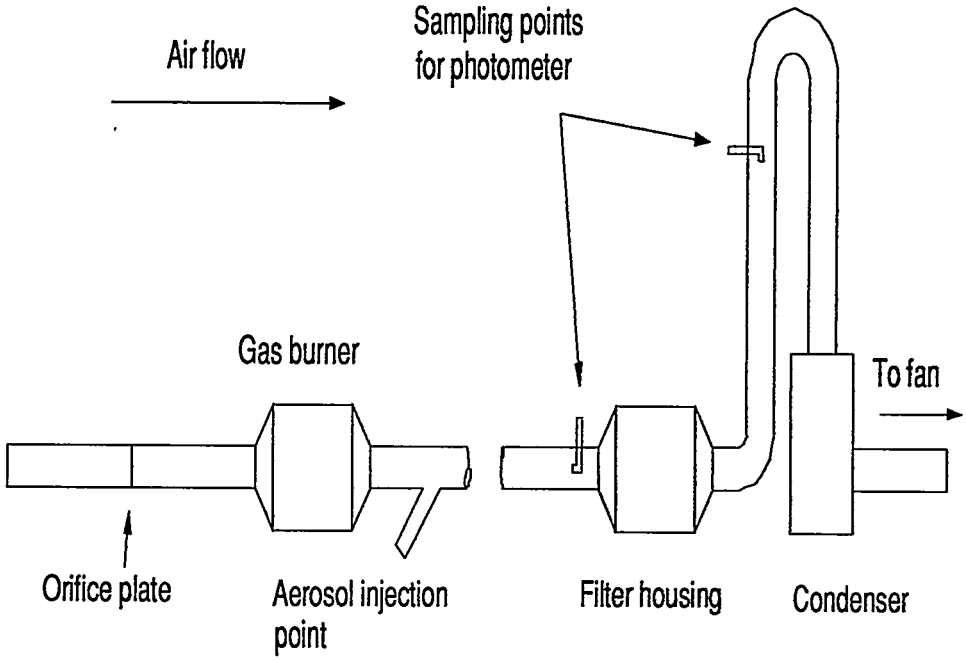
The hot air then passes through a vertical section of duct which includes a Stairmand disc to aid mixing of any test aerosol passing through defects in the filter before reaching the downstream sampling port. The hot air then passes through a water cooled finned tube heat exchanger to reduce the air temperature to 50°C or less before being expelled to the atmosphere. Between the heat exchanger and the fan, additional instrumentation permits the measurement of volumetric flow rate and air temperature.

A programmable microprocessor unit provided three term control for both the fan speed and gas burner. This allowed flow rates of up to 3400m^3/hr with temperature not exceeding 500°C to pass through the filter under test, where the pressure drop and penetration were recorded.

The penetration of the filter was measured using a modified sodium chloride technique. A Harwell design sodium flame photometer was used to measure the sodium aerosol concentration, which was generated using a sodium chloride stick thermal generator. The generator is designed for in-situ testing and provides an aerosol of 0.35μm mass mean diameter by vaporising a stick of sodium chloride in an oxy-propane flame. The concentration of the aerosol cloud could be altered by varying the feed rate of the salt stick, which for the tests performed was 1.9mm/min. The penetration was determined from the ratio of the downstream to upstream concentration as

$$Penetration(\%) = \left(\frac{downstream concentration}{upstream concentration} \right) \times 100 \quad (1)$$

Figure 1. Test Apparatus



2.2 Test procedure

The filters were tested at the rated volumetric flow rate, which was measured before the gas heater using an orifice plate as shown in Figure 1. The flow rate was maintained constant, such that after the gas burner, the reduction in density of the air (due to change in temperature) caused the volumetric flow rate through the filter to increase while the mass flow rate remained constant. The high temperature filters, were operated at 200°C for up to 48 hours. During this time the penetration and pressure drops were recorded, when at temperature and after the system had returned to ambient conditions.

For the tests performed with rectangular filters, a constant fan speed was used to give the rated flow at ambient conditions. Consequently, when the air was heated, the increased volumetric flow rate after the gas burner caused increased frictional pressure drop in the pipe work and across the filter and housing, thus reducing the flow rate.

3. RESULTS AND DISCUSSION

3.1 Mini-pleat filters

The value of the penetration and pressure drop measurements are shown in Figure 2. Figure 2(a) shows the penetration value for the filter at different temperatures. The penetration values are shown for ambient temperature and for an air temperature of 200°C, against the time that the

filter has been operated at temperature. The initial penetration was below 0.01% and was observed to steadily increase with time to a final penetration value of just below 0.02%, with very little difference between the value recorded at ambient or elevated temperature conditions.

At the same time the penetration was measured, the flow rate and pressure drop across the filter were recorded, as shown in Figure 2(b). The operation procedure for rectangular filters, at a constant fan speed, resulted in the flow rate through the filter dropping in response to an increase in pressure drop due to the increased local velocity and deformation of the packing. The initial pressure drop of the filter at ambient conditions was 280Pa which remained constant throughout the test, but the volumetric flow rate through the filter, after 48 hours at temperature, decreased by 10%. The pressure drop when operating at elevated temperature was greater than that at ambient temperature due to the reduced air density and increased local velocity. The pressure drop remained steady with time, but the flow rate was observed to decrease from 500 to 450l/s during this period. Closure of the medium pleats was observed, which may account for the lower volumetric flow rate.

3.2 Deep pleat filters

The penetration and pressure drop values for the deep pleat type filters are shown in Figure 3. From Figure 3(a) it may be observed that over the first four hours of operation the penetration rose sharply and thereafter slightly increased. In all cases the penetration remained below 0.02%, with little difference between the penetration values recorded at ambient or elevated temperature.

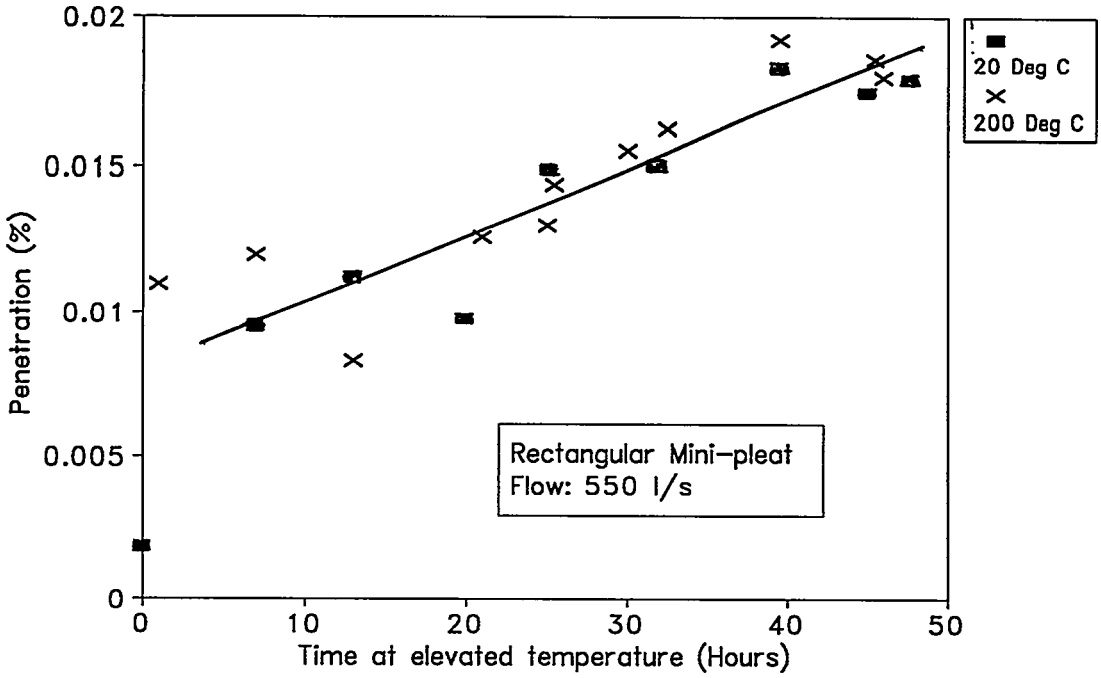
The pressure drop and flow rate data are shown in Figure 3(b), where in the first couple of hours of operation at temperature there was an increase in the pressure drop which was attributed to shifting of the packing and closure of individual flow channels. This was noted in the pressure drop measurements at ambient conditions, where from an initial value of 210Pa at 500l/s the final pressure drop was 600Pa at a flow rate of 470l/s.

3.3 Internal seal circular filters

The tests on the cirqular filters were performed under constant mass flow rate conditions. The results are shown in Figure 4, with the penetration values shown in Figure 4(a). The initial penetration value recorded at an air temperature of 200°C was about three times the initial ambient value suggesting that the dwell time of the test aerosol through the filter matrix was sufficiently reduced such that the collection by the diffusion mechanism was diminished. Thereafter, the penetration values at temperature remained relatively constant. However, after 10 hours at temperature the penetration measured at ambient conditions exceeded those at temperature. Examination of the sealing ring [10] after the tests revealed deformation that was believed to allow a seal leakage path when cool and was reduced at temperature due to thermal expansion of the locating spigot. The filter was replaced in the hot dynamic rig 22 days after the test had ceased. The penetration value at ambient conditions was measured at 0.0176%, indicating that the sealing ring did not retain its original shape.

Figure 2. Mini pleat test results (AESS 30/93401)

(a) Penetration measurements



(b) Pressure drop and volumetric flow rate measurements

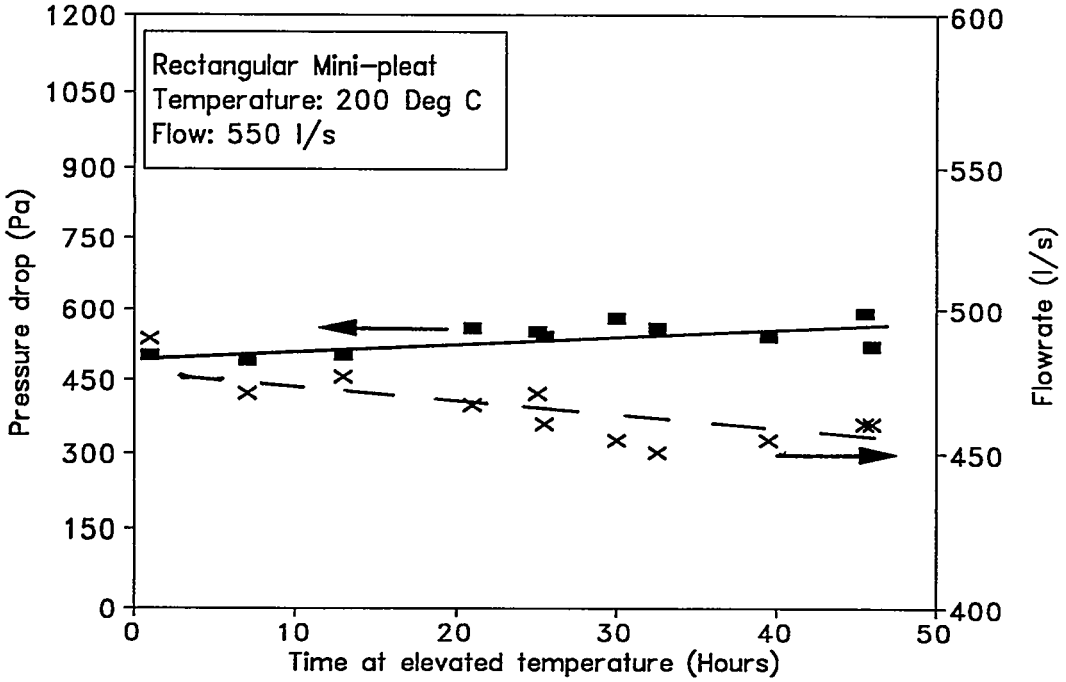
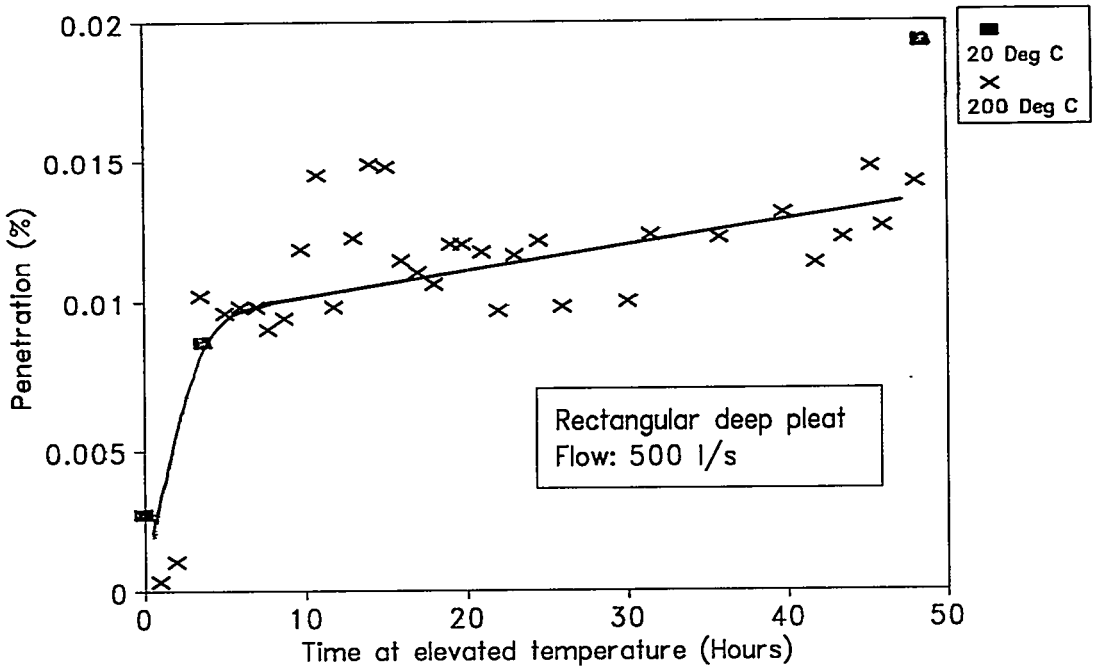


Figure 3. Deep pleat test results (AESS 30/93402)

(a) Penetration measurements



(b) Pressure drop and volumetric flow rate measurements

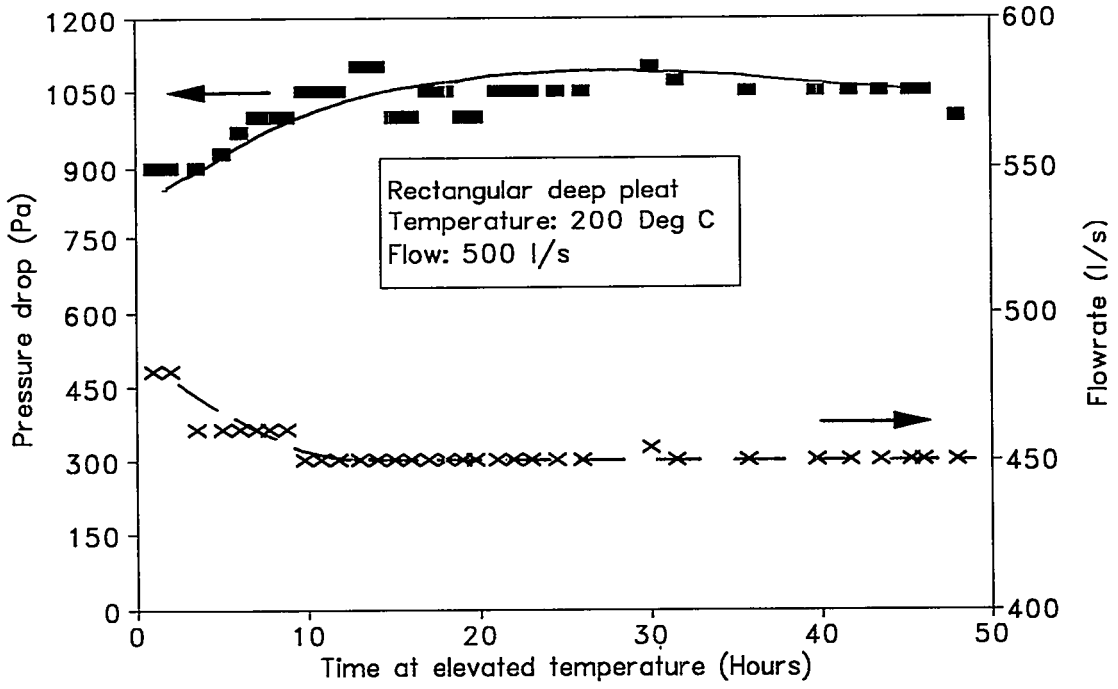
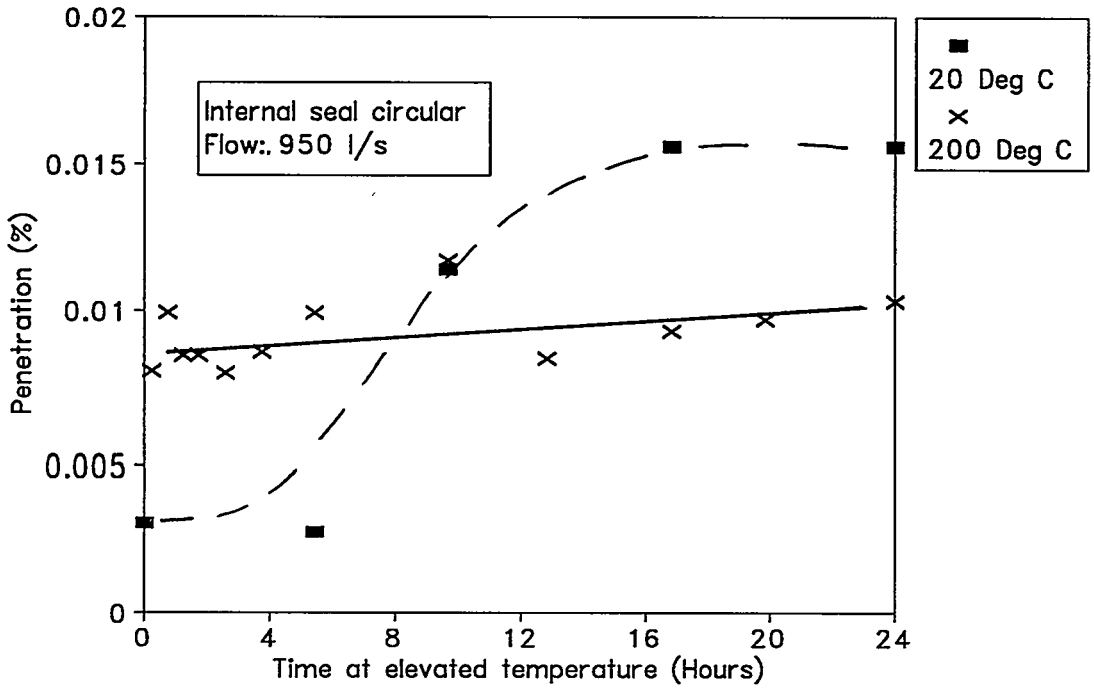
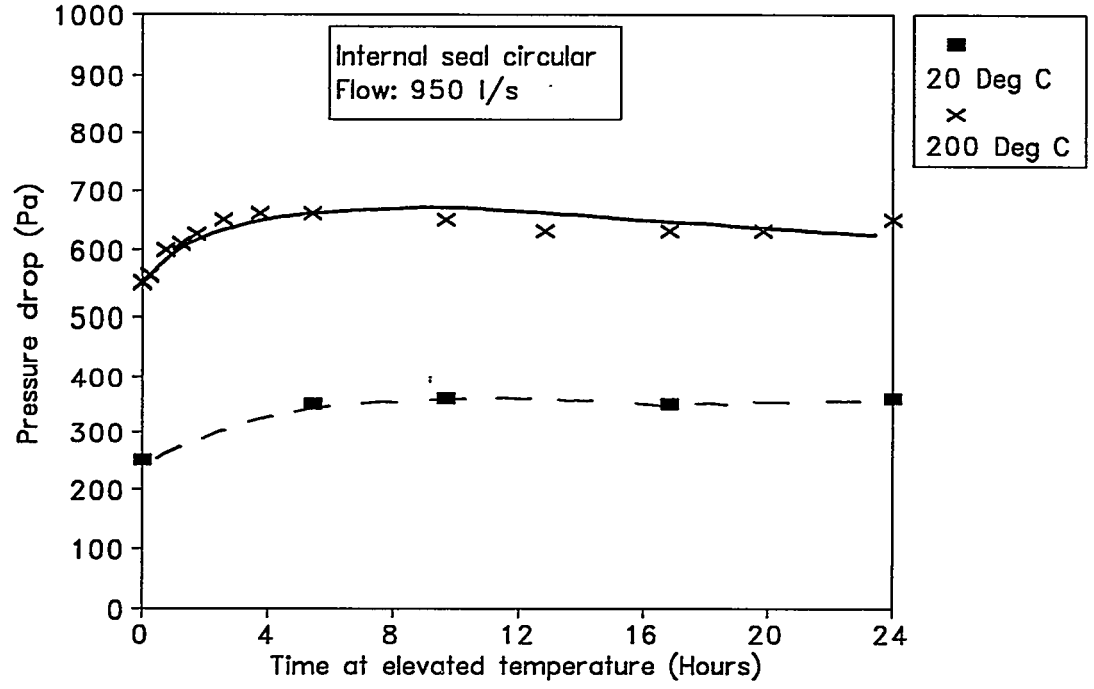


Figure 4. Internal seal circular test results (AESS 30/95100)

(a) Penetration measurements



(b) Pressure drop measurements



The pressure drop values, given in Figure 4(b), indicates that they increase in the first few hours of operation at temperature, and thereafter remain relatively constant. The reduction in the density of air between the temperatures of 20 to 200°C is of the order of 40%, which with the mass flow rate remaining constant will result in an equivalent increase in the air velocity through the filter. Previous experience has indicated that the pressure drop across the filter, ΔP, is proportional to the volumetric flow rate, Q, to a small power, n,

$$\Delta P \alpha Q^n \tag{2}$$

where a typical value for n is 1.2. The volumetric flow rate may be expressed in terms of the mass flow rate, $\dot{m}$, and the density, ρ, such that,

$$\Delta P \alpha \left(\frac{\dot{m}}{\rho} \right)^n \tag{3}$$

Hence for a reduction of 40% in the density of air due to the temperature change, the pressure drop should increase by 85%, which is close to the increase observed in Figure 4(b).

The pressure drop increased during the first 6 hours of operation and thereafter decreased slightly with time. This reduction in the pressure drop may be due to the degradation of the paper binder.

A number of failure points were observed in circular filters, where in particular cases the penetration rose dramatically in the first few minutes of operation, but was less than 1%. It was observed that holes appeared randomly along the pleated medium crease. A possible explanation of the cause of these holes may be due to local melting of the glass fibre medium by incandescent particles [11]. The pressure drop for the filters steadily increased with time, even though holes appeared, indicating closure of the flow channels.

3.4 High strength prototype circular filter

A prototype circular internal seal filter was constructed using a high strength medium, Lydall 3255-LIW in which the filter medium was supported by a glass fibre scrim.

Experiments were carried out at the rated flow, but in this case they included operation at different temperatures and the penetration and pressure drop were recorded with time as shown in Figure 5. Referring to Figure 5(a), there was an initial increase in the penetration when operated at 250°C which thereafter remained relatively constant at 0.005%. After 24 hours the temperature was raised to 300°C for four hours. During this time the penetration remained at 0.005%. After the total time of 28 hours at temperature the air temperature was again raised, to 350°C, where the first penetration measurement increased to 0.024%. Penetration increased steadily to 0.028% over the next 6 hours of operation. The air temperature was then raised to 400°C and over the next 6 hours of operation the penetration increased to 0.65%.

Visual examination of the insert after operation at temperature for 38 hours revealed that the silicon gasket material was cracked, allowing a leakage path for the test aerosol. Similarly, the penetration measurements at ambient conditions were greater than those recorded at temperature due to deformation of the gasket. The penetration after testing was measured using the sodium flame technique [5] at 3.25%. The original gasket was brittle and disintegrated when touched, was replaced before the penetration was again measured and found to be 0.09% at a pressure drop of 1490 Pa under ambient conditions. Examination of the filter revealed cracks within the sealant between the end cap and the medium, with the medium touching the outer support grille.

The pressure drop measurements, shown in Figure 5(b), revealed an increase in the first few hours of operation at temperature but thereafter remained relatively constant. With each increase in temperature there was a step increase in the pressure drop due to the reduction in density of the air and an increase in the velocity of the air passing through the filter. The final pressure drop at ambient temperature was three times the initial value, indicating that considerable shifting of the packing medium had occurred. The pressure drop across the filter was sufficiently large that the flow rate dropped during testing from 950 to 750l/s, when at temperature, due to the limitations of the fan performance.

4. CONCLUSIONS

1. Rectangular mini-pleat filter when tested at an air temperature of 200°C for 48 hours was found be give penetration values of less than 0.02%. No deterioration was observed of the gasket material.

2. Similar observations were observed with the deep pleat filter. Swelling of individual pleat channels resulted in a significant pressure drop increase during the test period although physical damage of the medium was not observed.

3. The penetration value of the internal seal circular filter was observed to be higher when measured at ambient conditions than at temperature. Deformation of the silicone gasket was responsible.

4. A number of failures were observed with circular filters where holes appeared along the pleated crease, but the penetration remained below 1%. Damage to the filters occurred in the first few minutes of operation.

5. The high strength internal seal circular filter performed well at temperatures below 300°C. At 350°C the gasket material failed resulting in a final penetration of 0.65% when operated at 400°C. Replacing the gasket of the high strength filter revealed a final penetration of 0.09%, with no obvious damage to the filter medium, but cracking of the high temperature sealant between the filter pack and the casing was evident.

6. Further investigations are needed to determine an alternative circular gasket to withstand operation at high temperatures.

ACKNOWLEDGEMENTS

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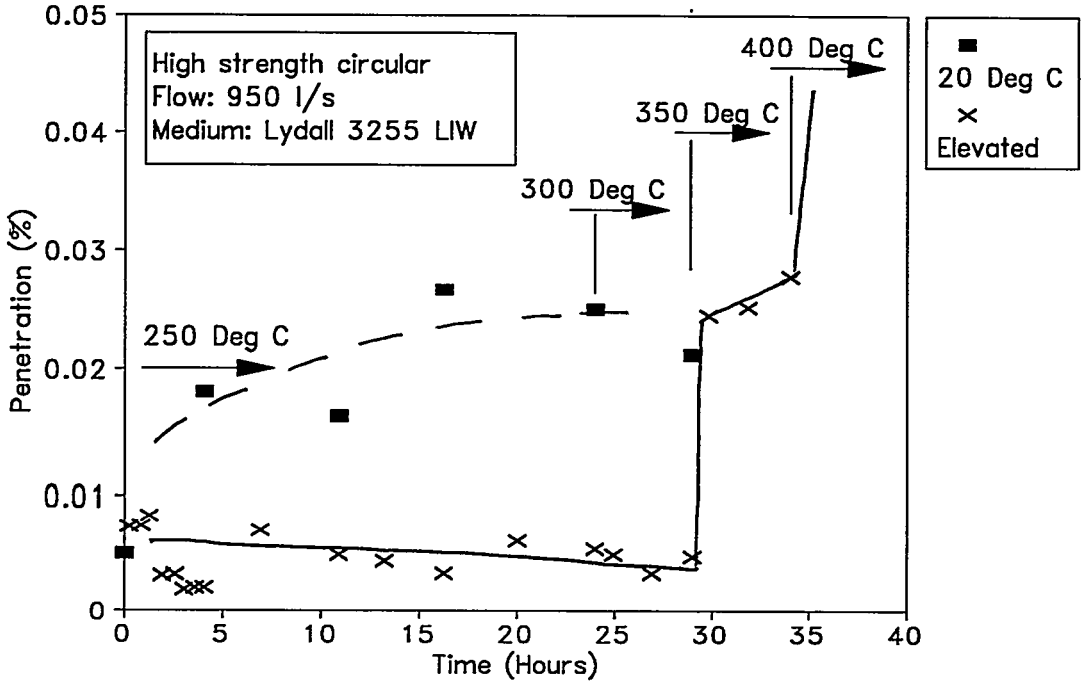
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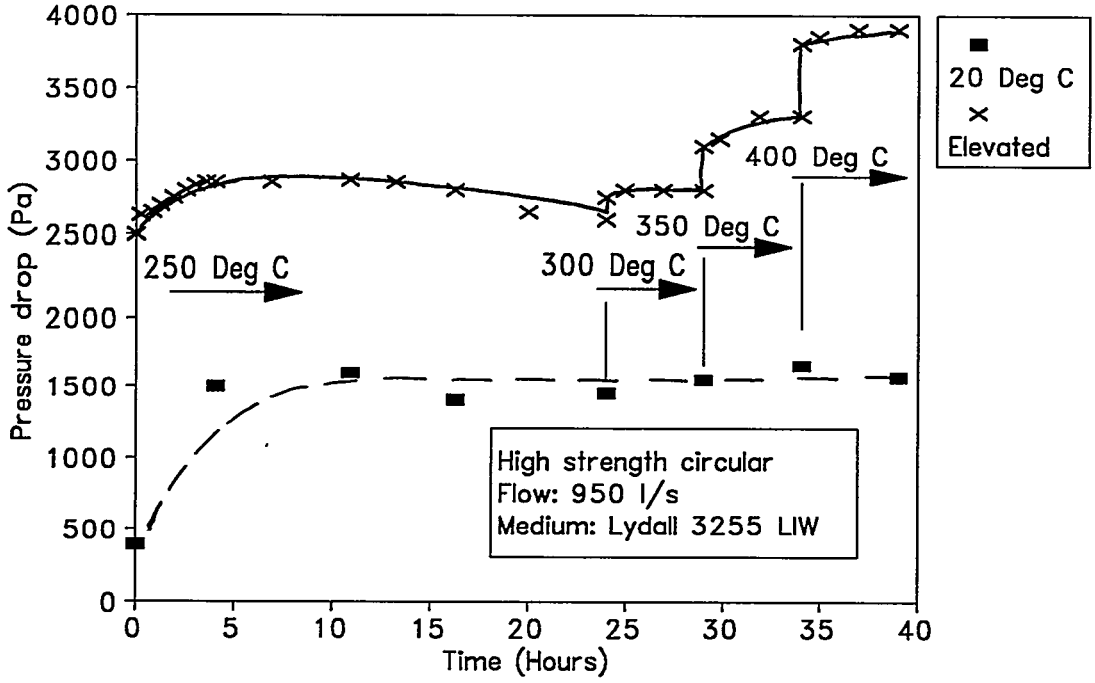
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Figure 5. Internal seal circular test results (High strength)

(a) Penetration measurements



(b) Pressure drop measurements



DISCUSSION

FIRST: Were your efficiency measurements made at the temperature to which the filter was being heated, or were they done after the filter had been cooled down again?

FRANKUM: There were two temperatures shown on the graphs, one at 20° C, ambient conditions, and one at temperature. The ones at temperature were actually made at the temperature the filter was being operated at. And the ones at 20° C were made after the rig had been shut down.

PRELIMINARY STUDIES TO DETERMINE THE SHELF LIFE OF HEPA FILTERS

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Abstract

We have completed a preliminary study using filter media tests and filter qualification tests to investigate the effect of shelf-life on HEPA filter performance. Our media studies showed that the tensile strength decreased with age, but the data were not sufficient to establish a shelf-life. Thermogravimetric analyses demonstrated that one manufacturer had media with low tensile strength due to insufficient binder. The filter qualification tests (heated air and overpressure) conducted on different aged filters showed that filter age is not the primary factor affecting filter performance; materials and the construction design have a greater effect. An unexpected finding of our study was that sub-standard HEPA filters have been installed in DOE facilities despite existing regulations and filter qualification tests. We found that the filter with low tensile strength failed the overpressure test. The same filter had passed the heated air test, but left the filter so structurally weak, it was prone to blow-out. We recommend that DOE initiate a filter qualification program to prevent this occurrence.

I. Introduction

Although it is a widely held belief that HEPA filters degrade with age, there is little technical information on which to establish a shelf life. The components of a HEPA filter such as the organic

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*This work was performed under the auspices of the U.S. Department of Energy by the Lawrence Livermore National Laboratory under Contract No. W-7405-ENG.48.

binder that holds the glass fibers together in a sheet, the organic sealant that glues the filter pack into the frame and forms a seal, the organic gaskets, and even the glass fibers themselves are subject to ageing. An informal statement [1] made more than two decades ago that glass fibers, used in medium manufacture, start to age and weaken at the moment it emerges from the platinum orifice of the melting pot from which it is extruded. Further private information from a significant manufacturer of continuous - strand glass fiber, employed as the reinforcing agent in fiberglass and plastic lay-ups, indicates that his product is placed in a conditioned storage space after manufacture to inhibit loss of strength.

Previous studies on the related in-service ageing have shown significant reductions in filter strength [2,3]. Johnson et al [2] studied the effect of age on the structural integrity of HEPA filters. They conducted simulated tornado pulse and shock overpressure tests on HEPA filters that were in service for 13-14 years and found no loss in efficiency, but a significant decrease in filter strength. The HEPA media pack was blown out of the filter housing at a differential pressure as low as 13 inches of water. The HEPA filters used glass fiber media that was folded in a deep pleat configuration with asbestos separators and glued into the plywood frame with a urethane foam sealant. Tests on the filter media showed it was weak and brittle; 42% of the samples failed the tensile strength; and 100% of the samples failed the water repellent top test and 57% the water repellent bottom test. The media was so brittle that it was not possible to test the tensile strength across the bends.

Robinson et al [3] examined filters with a service life of up to ten years and observed significant reductions in the paper strength and severe corrosion of aluminum separators. They found that the corroded aluminum separators formed a strong bond to the HEPA paper and would therefore have an increased tendency to tear the filter media when subjected to pressure stresses. They also found the aged HEPA media had lost its water repellency.

These previous studies imply that ageing and weakening continue even after the fiber has been made into medium and the medium fabricated into a filter. Although the previous studies were conducted on in-service ageing, and not shelf-life ageing, they suggest the type of effects one can expect for shelf-life ageing. The

because the filters were second-stage filters that were effectively only exposed to clean air and were housed inside a building characterized by moderate temperature and humidity conditions.

II. Experimental Plan

The experimental plan that we selected for this study was to first accumulate unused HEPA filters that had been stored for various years in warehouses. We wanted a minimum of four filters from the same manufacturer for each filter age; one filter would be cut open and provide samples of filter media and other components, while the remaining three filters would be tested separately for resistance to overpressure, heated air and rough handling. After we accumulated the available filters, we would first conduct tests on the filter media to identify candidate parameters that show a relation to filter age. Following the media tests we would conduct the overpressure, heated air and rough handling tests on those filters for which a trend is established between a media parameter and the filter age. The experiments conducted in this study were limited in both scope and depth to keep the costs within the available budget. The study should therefore be viewed as a preliminary effort for a more in-depth investigation to follow.

III. Test Results

We were able to locate a number of size 5 (2 x 2 x 1 ft.) HEPA filters that had been stored at the Rocky Flats Plant (RFP) and at Lawrence Livermore National Laboratory (LLNL) The filters ranged in age from 0.2 to 18 years. All of the filters came from Rocky Flats except the June 1984 filter of Manufacturer A. In addition to the size 5 filters, we obtained media samples of Manufacturer A for 1976 and also cut media samples from a 1991, 50 cfm HEPA filter (Manufacturer A) found at LLNL. All of the size 5 filters were open-faced with the media folded in a deep pleat configuration and were made with organic sealant and plywood frame. Except for the January 1976 filter of Manufacturer B , all were equipped with aluminum alloy separators. The exception had crysotile asbestos separators.

Filter Media Tests

The laboratory tests on the filter media selected in this study consisted of measuring the tensile strength, quantity of binder and Nomex (du Pont trade name) fibers, media thickness and weight. We carefully cut open the filters and extracted several layers of media with the separators in-tact. The media was unfolded and appropriate samples cut for the different measurements. We used a thermogravimetric analyzer (TGA) for determining the amount of binder and Nomex fiber in the media and an Instron tester for measuring the tensile strength. We omitted many additional tests that would be included in a more comprehensive study. The primary tests omitted are air flow resistance and DOP penetration for newly cut samples and samples after flexing; tensile strength in across-machine direction, after heated air, and after wetting; and water repellency [4]. Since the previous studies [2,3] suggest that aged filters loose their water repellency treatment and become brittle, it is important that the additional tests be conducted in a more complete study.

The medium thickness was determined with a Scherr-Tumico micrometer having a 0.5 inch diameter anvil and foot and is designed for measuring the thickness of sheet stock like paper and rubber. The filter medium was compressed in the micrometer until the first click of the tension ratchet assembly.

To determine the amount of binder and Nomex, the TGA was used rather than the conventional TAPPI T-413 method of weighing the sample before and after heating in an oven at specific temperatures [5]. We found the TAPPI method prone to error, time consuming, and requiring modification to evaluate Nomex. Temperature settings are required @ 107 C. to drive off adsorbed water, at 400 C. to drive off the binder, and at 550 C. to drive off the Nomex. Preliminary tests with the oven method showed rapid weight gain, due to moisture adsorption, during the weighing operation after removing the filter medium from the oven.

We determined the medium tensile strength with an Instron tester in accordance to MIL-F-51979 [4]. We cut several 1 x 6 inch samples in the flat portion of the medium and additional samples with a pleat in the center. The 1 x 6 inch samples were cut with the length in the machine direction (the direction in which the medium moves forward during the paper manufacturing).

A TGA model 2950 from TA Instruments, New Castle, DE, was used to determine the percent by weight of binder and Nomex fiber in 1/4 inch disks punched from the media samples. Nomex was added to the filter media for RFP in the early 80's to make it more resistant to attack by nitric and hydrofluoric acid. Asbestos was formerly used to provide the acid resistance. Although other DOE facilities do not specify the addition of Nomex, it sometimes appears in the media.

After inserting the sample into the TGA, the instrument automatically measures the amount and rate of change in the sample weight as a function of temperature in a nitrogen atmosphere. The amount of weight loss in known temperature regions is used to determine the percentage of the acrylic binder and du Pont's Nomex in the filter medium. Figure 1 shows the TGA curve for the 17.5-year-old sample of Manufacturer A. The sample shows the weight loss begins at about 250 C and ends at about 420 C. The 4.5% weight loss is due to the binder. The derivative of the weight loss curve is also shown in Figure 1 as a single peak and is useful for identifying the beginning and end temperatures of the weight loss from a component. The sample in Figure 1 shows a single peak in the derivative curve and thus indicates only a single component, the binder, accounts for the lost weight.

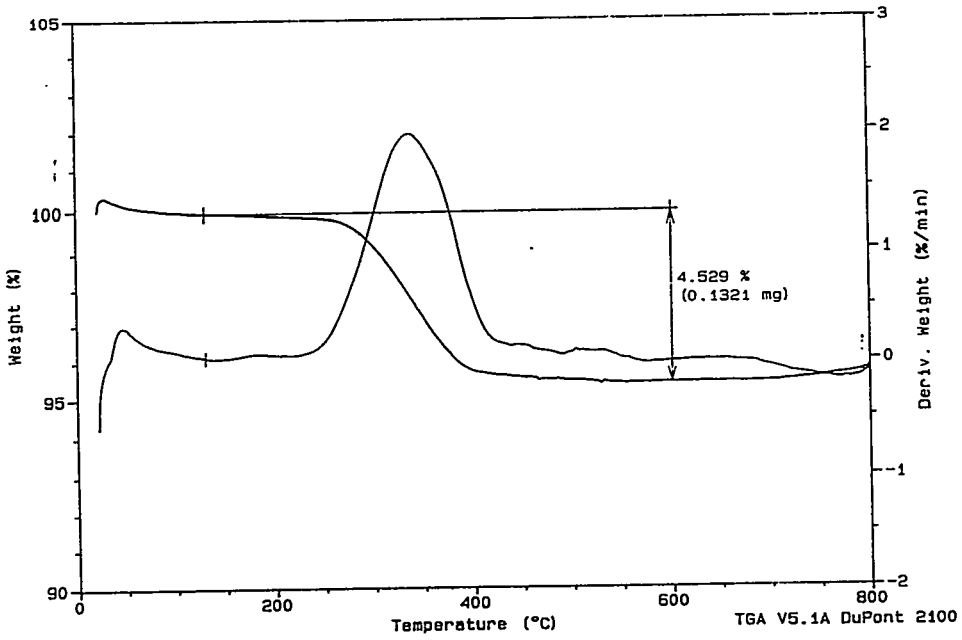


Figure 1. TGA results for filter media taken from 17.5-year-old filter (Manuf. A) showing binder loss.

Figure 2 shows the TGA curve for the 2.7-year-old sample of Manufacturer A. We see two peaks in the derivative curve, one for the binder and one for the Nomex. In this case the binder begins to loose weight at about 320 C and is completely vaporized at 429 C. The second peak for the Nomex begins at 429 C and is through at about 580 C. The weight loss for the binder and Nomex peaks are 1.5% and 1.6% respectively.

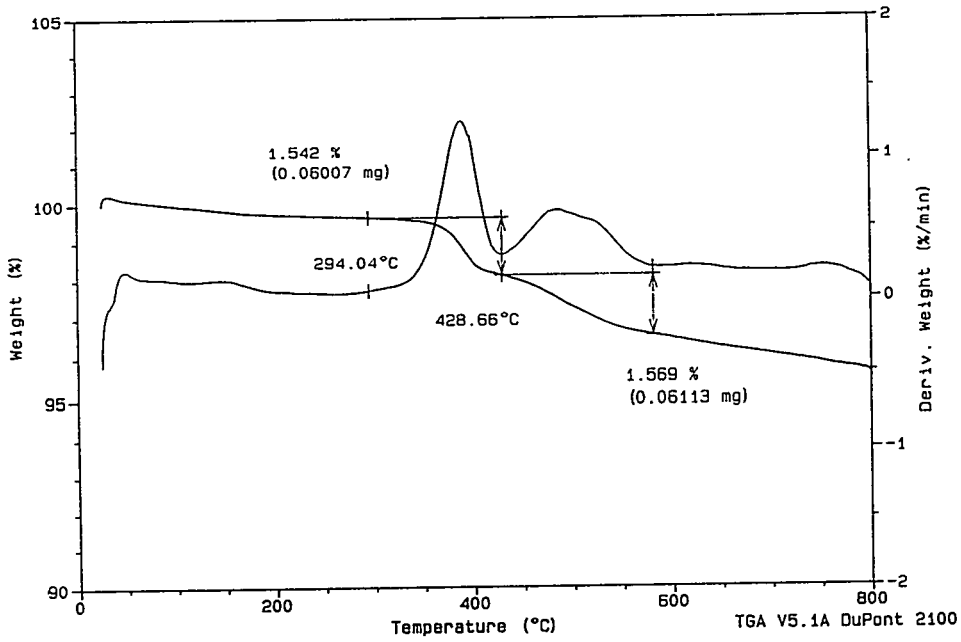


Figure 2. TGA results for filter media taken from 2.7-year-old filter (Manuf. A) showing loss of binder and Nomex.

We encountered more complicated TGA curves as shown in Figure 3, where four derivative peaks are seen for the sample taken from a 12.1-year-old filter (Manuf. D). The four derivative peaks indicate four components are present, although not necessarily independent. The first peak is due to adsorbed water removal. Water adsorption on the filter media suggests that the organic water-proofing has degraded. The second and third peaks are due to loss of binder and Nomex respectively. The weight loss due to binder is 1.6% beginning at 290 C and ending at 424 C. The weight loss due to Nomex is 1.8% beginning at 424 C and ending at 581 C. We do not know what the fourth peak is due to.

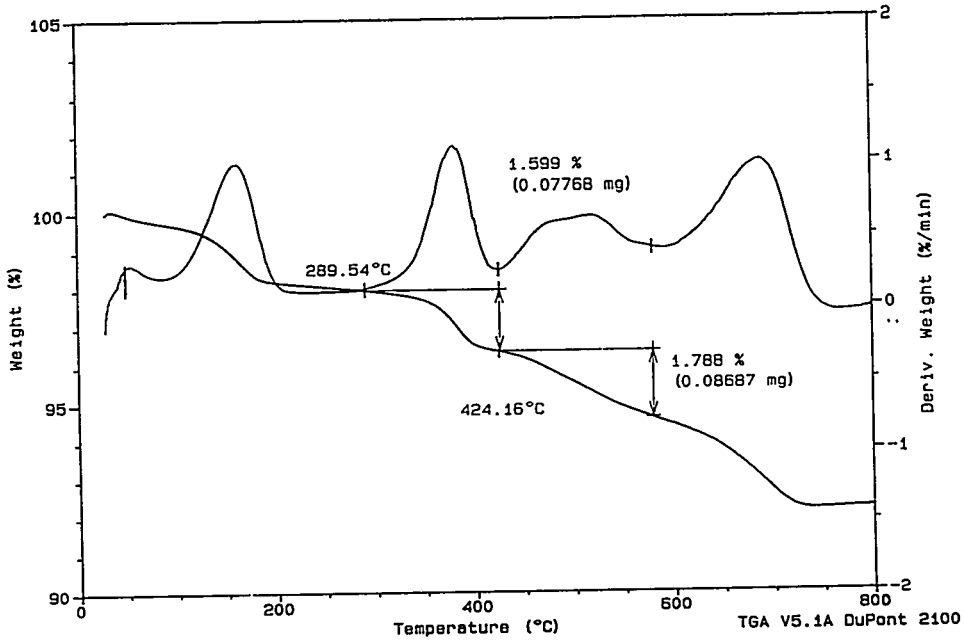


Figure 3. TGA results for filter media taken from 12.1-year-old filter (Manuf. D) showing loss of water, binder, Nomex and an unknown component.

In order to relate the weight loss in the TGA curves to the weight percent of binder and Nomex fibers in the filter media, we conducted baseline TGA tests using pure acrylic binder and Nomex fibers (6). The results for the acrylic binder and Nomex fibers are shown in Figures 4 and 5 respectively. The baseline tests show the weight loss for acrylic binder is 81.2% in the temperature limits between 250 C and 420 C and 34.7% for Nomex fibers in the temperature limits between 420 C and 580 C. Since the pure samples do not show 100% weight loss in the temperature limits, the measured weight loss for binder and Nomex have to be corrected. For the acrylic binder, the measured weight loss has to be multiplied by 1.23 to obtain the percent of binder in the medium. For the Nomex fibers, an additional correction has to be made since a small portion of binder looses weight in the temperature region attributed to Nomex fibers. This overlap is corrected by subtracting 11% of the measured binder weight loss from the Nomex weight loss. The net measured weight loss for Nomex fibers is then multiplied by 2.67 to obtain the percent of Nomex fibers in the medium.

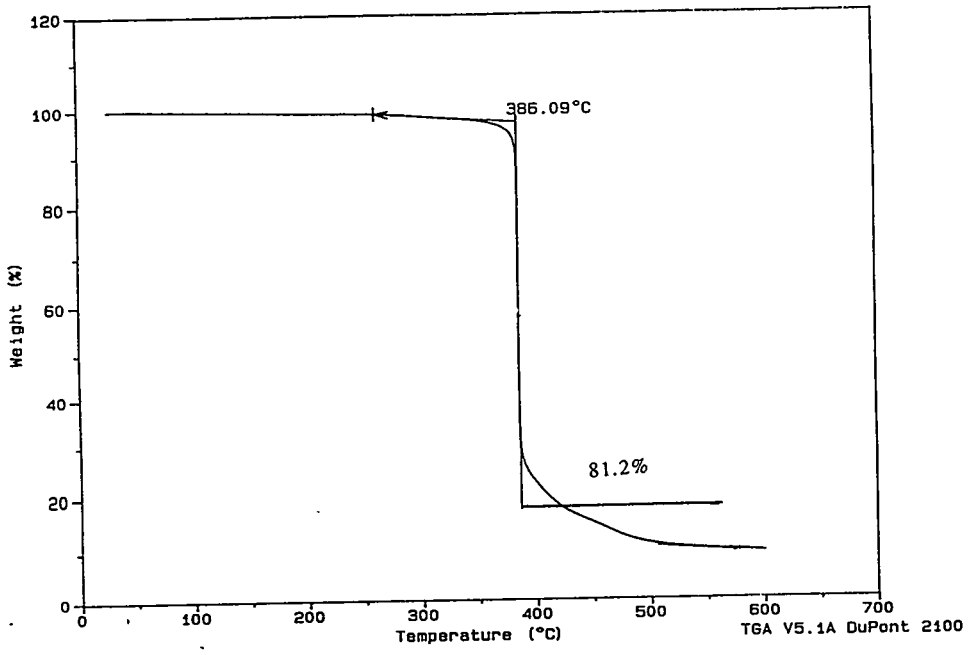


Figure 4. Baseline TGA curve for 100% acrylic binder.

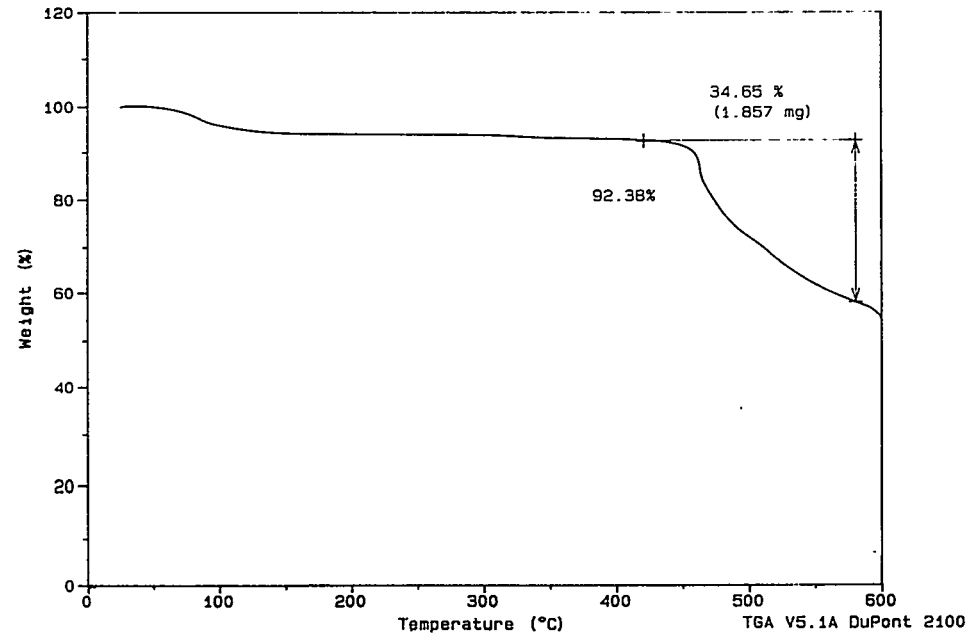


Figure 5. Baseline TGA curve for 100% Nomex fibers.

The results from the laboratory tests are summarized in Table 1, where we have tabulated the percent by weight of binder and Nomex, the tensile strength of the unfolded and folded samples, the thickness, and the weight of the media samples. We computed the correlation of filter age to the binder composition, tensile strength, thickness and ream weight, but did not find any significant correlation. Figure 6 shows the least squares fit between the tensile strength of the unfolded media and the filter age. We expected to see a strong correlation in tensile strength with age based on previous studies on in-service aging [2,3].

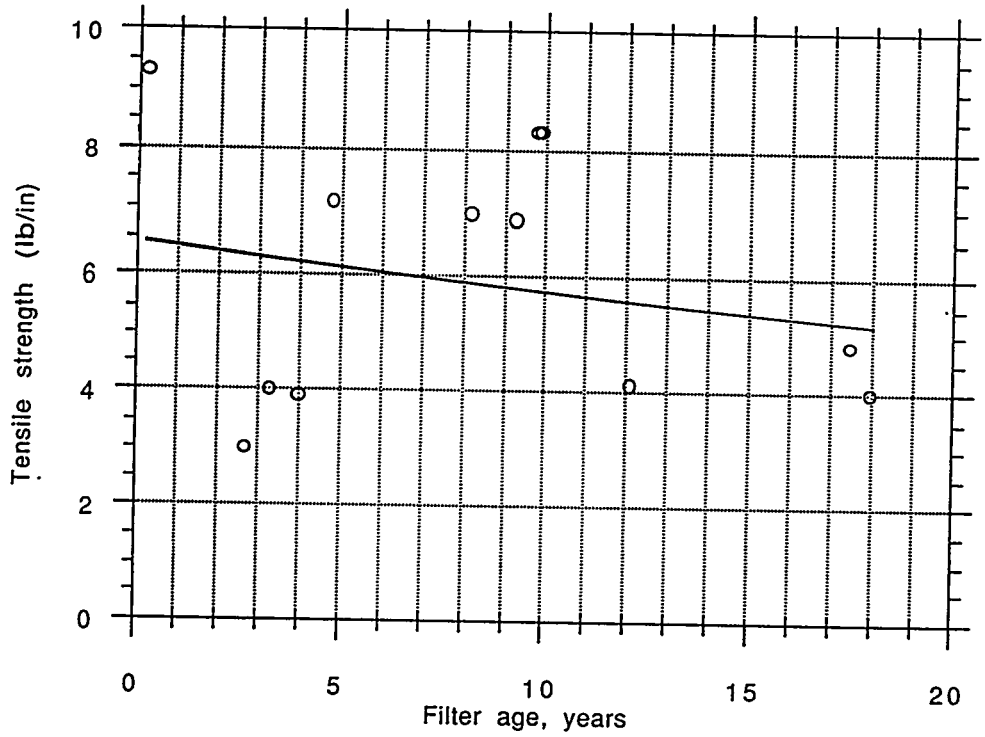


Figure 6. Correlation of media tensile strength to filter age.

During our shelf life study, we learned that the filter medium in the 1990 and 1991 filters of Manufacturer A currently was developing cracks during transportation and installation. Since the problem seemed to be related to manufacturing and not age, we examined the effect of removing the questionable data from our correlations. We removed the 1990 and 1991 data on Manufacturer A and replotted the data in Figure 7. The data now show a strong correlation of decreasing tensile strength with age. The regression coefficient of the linear least squares fit is R= 0.81. A similar trend is seen for the tensile strength of the folded media in Figure 8, although the values are much smaller.

Table 1 Media Test Data

Age (Years)	Manu-facturers	Purchase Date	Composition (% by wt.) Binder	Composition (% by wt.) Nomex	Tensile Strength (lb./in., Mach. dir.) Unfolded[1]	Tensile Strength (lb./in., Mach. dir.) Folded	Thickness (in.)	Weight lb/3000ft^2
18	A	1976	1.4	0.0	4.0	–	.019	42.7
17.5	B	Jan '76	5.5	0.0	4.8	1.1	.021	58.1
12.1	D	Aug '82	2.0	4.3	4.1	1.9	.022	59.3
9.8	C	Apr. '84	5.8	4.3	8.3	2.7	.024	58.6
9.9	A	Jun '84	3.0	2.4*	8.3	1.6	.023	48.0
9.3	E	Nov. '85	4.3	3.7	6.8	1.9	.020	53.8
8.2	A	Oct. '86	3.8	2.9	6.9	1.6	.025	56.6
4.8	E	May '89	4.7	4.0	7.1	2.7	.020	57.4
4	A	1990	2.0	2.9	3.9	–	.024	52.3
3.3	A	Mar '91	2.5	1.9*	4.0	0.9	.023	46.8
2.7	A	Oct '91	1.8	3.7	3.0	0.9	.020	43.7
0.2	A	May '94	3.0	3.2	9.3	2.3	.024	50.5

[1] Standard is 2.5 lbs. minimum
* Nomex not requested in LLNL purchase order

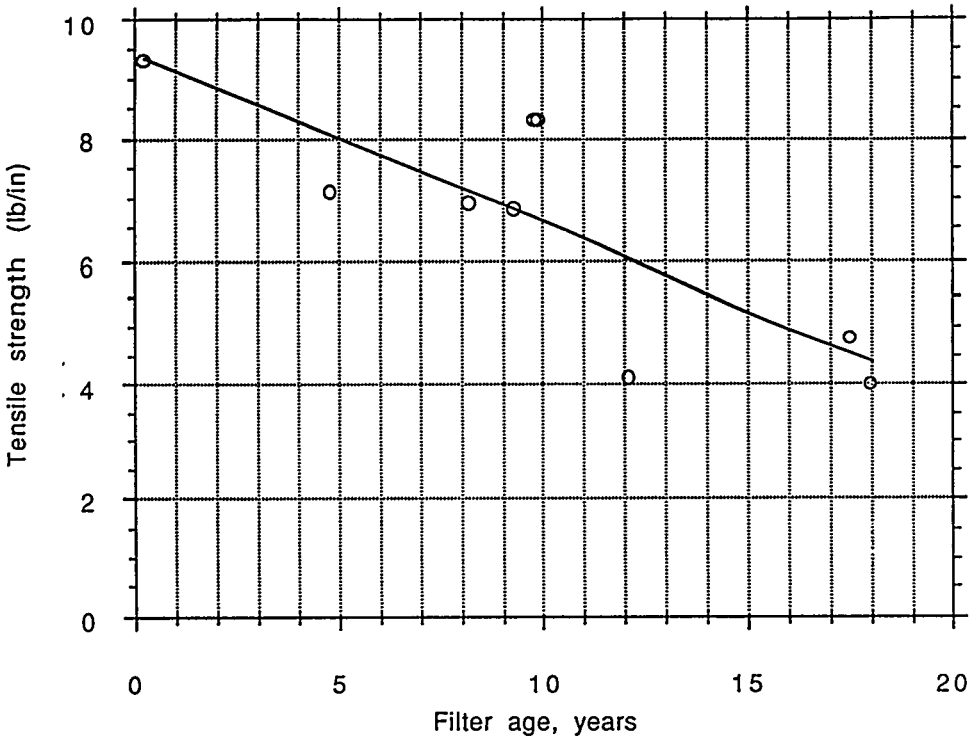


Figure 7. Correlation of media tensile strength to age with 1990 and 1991 data on Manufacturer A removed.

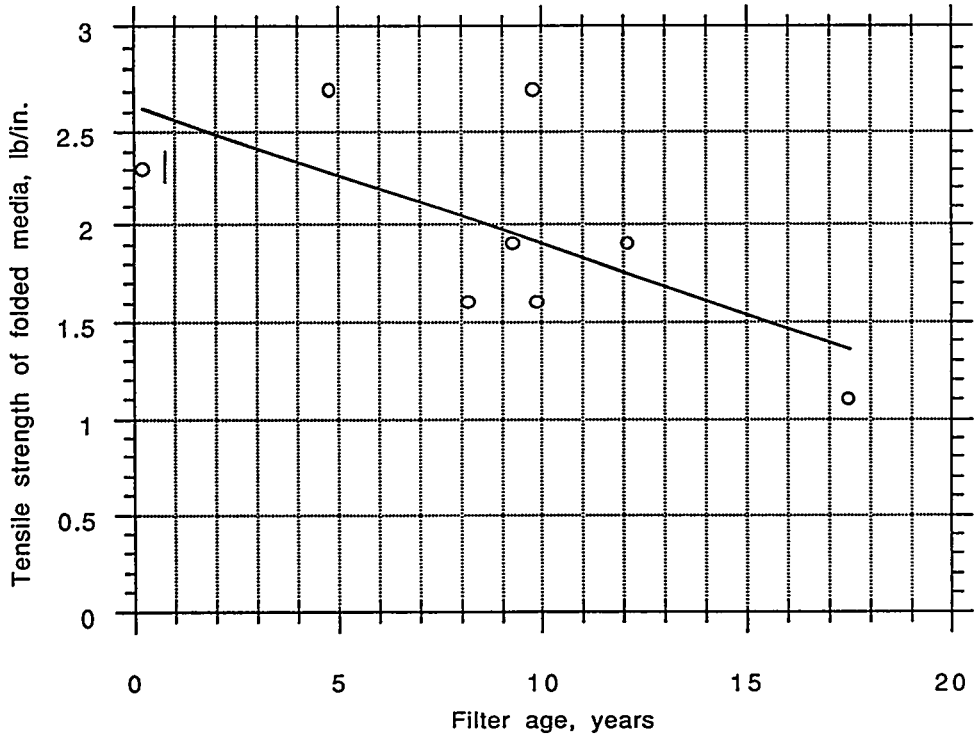


Figure 8. Correlation of tensile strength to age of folded media with 1990 and 1991 data on Manufacturer A removed.

The results from our media tests show that the tensile strength of the media decreases with age, but does not provide sufficient information to establish a shelf life for filters. All of the filter media tested in our study meet the requirements in MIL-F-51079 that specifies the tensile strength of unfolded media be greater than the minimum value of 2.5 lb./in.(4). There is no requirement for the tensile strength of the folded media, although that is the critical parameter determining media failure. Note that the average tensile strength of the folded media is about 1/3 the strength of the unfolded media.

The media tests do suggest that the low, although acceptable, tensile strengths of the 1990 and 1991 media of Manufacturer A is due to insufficient binder to hold the glass fibers together in a sheet. We have plotted the tensile strength versus the percent of binder for all of the media samples of Manufacturer A in Figure 9 and show good correlation. The straight line correlation was obtained by least squares analysis. The correlation is not as good when the data of all the manufacturers are combined. This is not surprising since the manufacturers do not use the identical materials and processes.

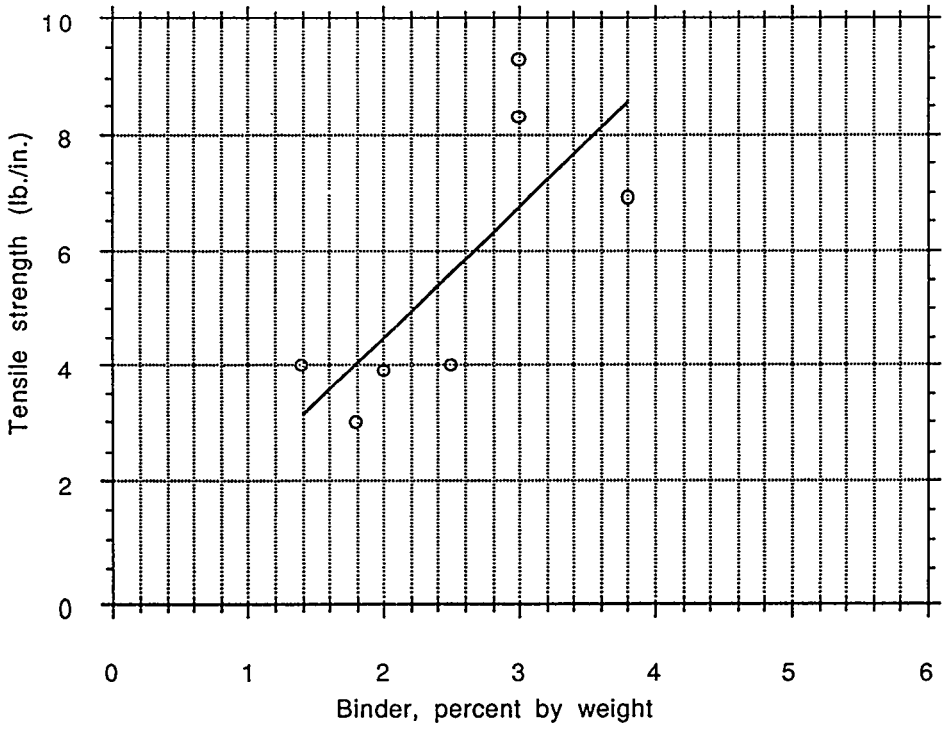


Figure 9. Correlation between the percent of binder and the media tensile strength for Manufacturer A.

Filter Tests

We conducted a limited number of non-destructive quality assurance (QA) tests and destructive qualification tests on three different aged filters we had available. These tests were selected because they specify the minimum acceptable performance for new filters. We reasoned that aged filters should also meet these requirements. A previous study on in-service filter life used simulations of tornado pulses and shock wave overpressures to evaluate aged filters.

The quality assurance tests on assembled filters consist of visual inspection for damage, penetration, and resistance (7). Because all of the filters under study are Size 5, designated 24 by 24 by 11-1/2 inches, the latter two tests involve use of the Q107 penetrometer to measure penetration of dioctyl phthalate (DOP) particles through the filter and resistance to air flowing at 1000 cfm. A second penetration test is made with air flow reduced to 20% (200 cfm.) and is primarily targeted at identifying unwanted holes in the media and leakage at those seams where frame and gasket pieces are joined together. The upper limit for DOP penetration is 0.03% at either flow rate. The maximum allowed resistance (pressure drop) is 1.0 inch, water gauge, at the 1000-cfm. flow rate.

Qualification testing has reliability of the design of the assembled filter as its primary aim. Testing is abnormally severe and is so intended to ferret out any component weakness by which the filter might fail. This dictates that a filter not be used after a qualification test; therefore, qualification testing is classified as destructive. A detailed explanation of the tests was included in the Proceedings of the 22nd DOE-NRC Nuclear Air Cleaning conference (8) and listed below by function:

a. Q160 overpressure apparatus. A 60-minute test which subjects a previously humidified filter to steam droplets in air flowing at an elevated resistance of 10.0 inches, water gauge. Post-test penetration must not exceed 0.03% at 20% air flow.

b. Heated air device. Subjects a filter to air while heating to 700 degrees F. and operation at that temperature for five minutes. Post-test penetration must not exceed 3.0% at 100% flow.

c. Rough handling machine. The filter, in its shipping carton, is strapped to platform which subjects the packaged filter to an amplitude of 1/2 inch at a frequency of 200 cycles per minute. Post-test penetration must not exceed 0.03% at both 100% and 20% flow. No rough handling tests were conducted in this preliminary study.

We conducted QA and qualification tests on only three different ages filters in this preliminary study. The 0.2- and 2.7-year-old filters of Manufacturer A were selected to provide information on an immediate problem with filters used at RFP and other DOE facilities. The 9.8-year-old filter of Manufacturer C was selected at random. In a more comprehensive study, all of the filters we collected would be tested.

We conducted the heated air and pressure resistance tests using a different filter for each test as specified in MIL-F-51068 [7]. We also conducted tests using the same filter in a test sequence consisting of a heated air test followed by a pressure resistance test. The latter test sequence was previously used to evaluate a high-strength HEPA filter to better simulate the multiple stresses placed on HEPA filters during accident conditions[8]. Ruedinger et al (9) reported that nuclear reactors in Germany require filter qualification tests consisting of elevated temperature in still air, pressure resistance in high air flow, and humid air resistance.

Table 2 shows the test results for the pressure resistance tests. The 0.2-year-old filter of Manufacturer A and the 9.8-year-old filter of Manufacturer C passed the test and showed no structural damage. In contrast, the 2.7-year-old filter of Manufacturer A failed the test. Examination of the filter after the test showed a slit about 5 inches along a pleat (Figure 10) and small media tears near the housing (Figure 11), both defects on the downstream side.

Table 2. Pressure resistance tests.

			Before			After		
Age (Years)	Manufacturer	Purchase Date	ΔP in.	Penetration 100%	20%	ΔP in.	Penetration 100%	20%
9.8	C	April '84	.86	.030%	.033%	.86	.026	.032
2.7	A	Oct '91	.80	.01%	.01%	.74	.51%	.81%
0.2	A	May '94	.80	.016%	.018%	.78	.005%	.011%

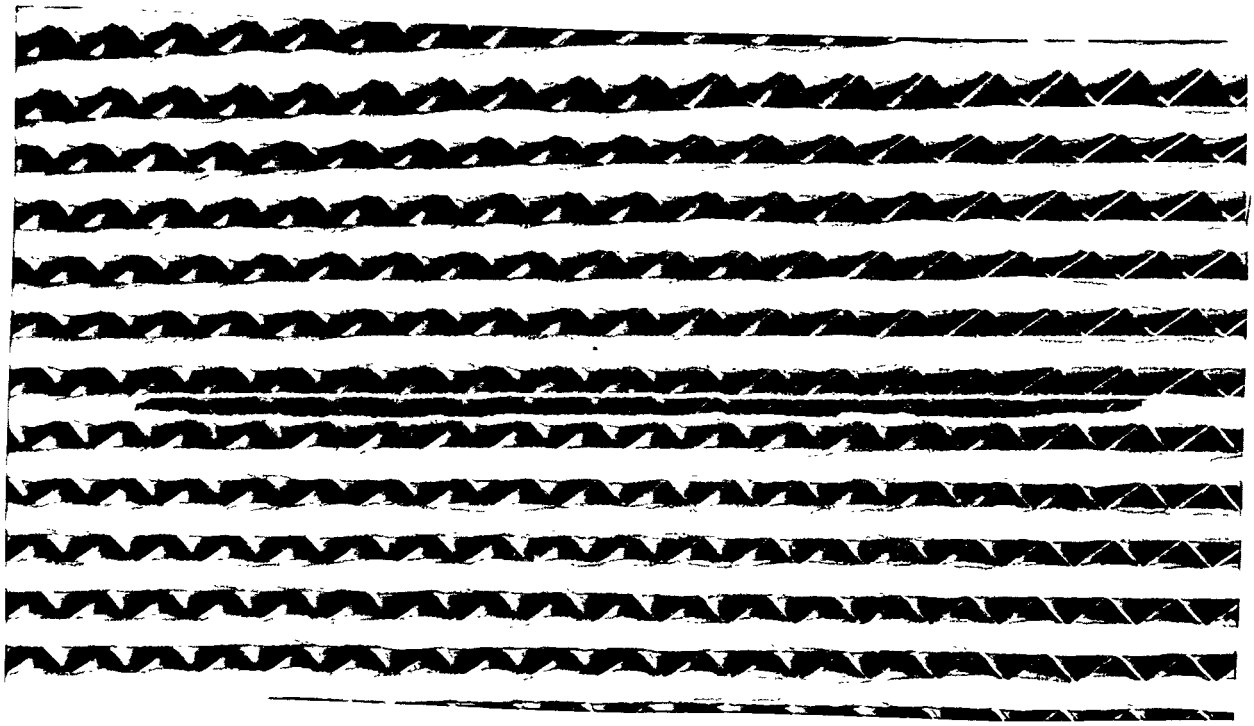


Figure 10. Slit in media pleat on the downstream side of a 2.7-year-old filter of Manufacturer A following the pressure test.



Figure 11. Small tears on downstream side of 2.7-year-old filter of Manufacturer A following the pressure test.

The failure of the 2.7-year-old filter in Table 2 can be attributed to the low tensile strength of the medium which resulted from insufficient binder. The other two filters show no deterioration after the pressure test despite the large age difference. Filter construction properties clearly have a much greater effect on resistance to pressure than filter age.

We then evaluated two additional filters in a test sequence consisting of a heated air test followed by a pressure test and a third filter in the heated air test only. Filter penetration measurements were made before and after each test. The test results are summarized in Table 3.

Table 3. Test results following heated air and pressure resistance.

			Before			After Heated Air			After Heated Air and Pressure		
Age (Years)	Manu-facturer	Purchase Date	ΔP in.	Penetration 100%	20%	ΔP in.	Penetration 100%	20%	ΔP in.	Penetration 100%	20%
9.8	C	April '84	.86	.012%	.013%	.90	.015%	.017%	–	–	
2.7	A	Oct '91	.76	.018%	.023%	.74	1.24%	4.52%	0.64	5.86%	7.51%
0.2	A	May '94	.74	.008%	.008%	.76	0.77%	1.91%	.80	2.05%	4.12%

The three filters meet the requirement of less than 3% penetration at 100% flow after the heated air test. Since there are no standards for a sequence of tests on the same filter, there is no established penetration level defined as a failure.

During the heated air test on the 2.7-year-old filter, the sealant was burning and continued to burn even when the heat was removed. Examination of the filter after test showed the sealant had burned through the entire depth of the filter, leaving the filter pack loose in the wooden frame. Figure 12 shows a close up of the sealant on the exit side of the filter after the heated air test. The gap between the wooden frame and the filter pack is visible in the photograph. The filter pack also showed a slight buckling of the separators and media pleats into an "s" shape near the bottom frame of the filter. This buckling was presumably due to the softening of the media and the weight of the filter pack. No tears or holes were found in the filter media. The penetration at 100% flow is 1.24% and

is within the allowed 3% limit. Note that the filter would have failed if the penetration at 20% flow were used as done in the overpressure test.

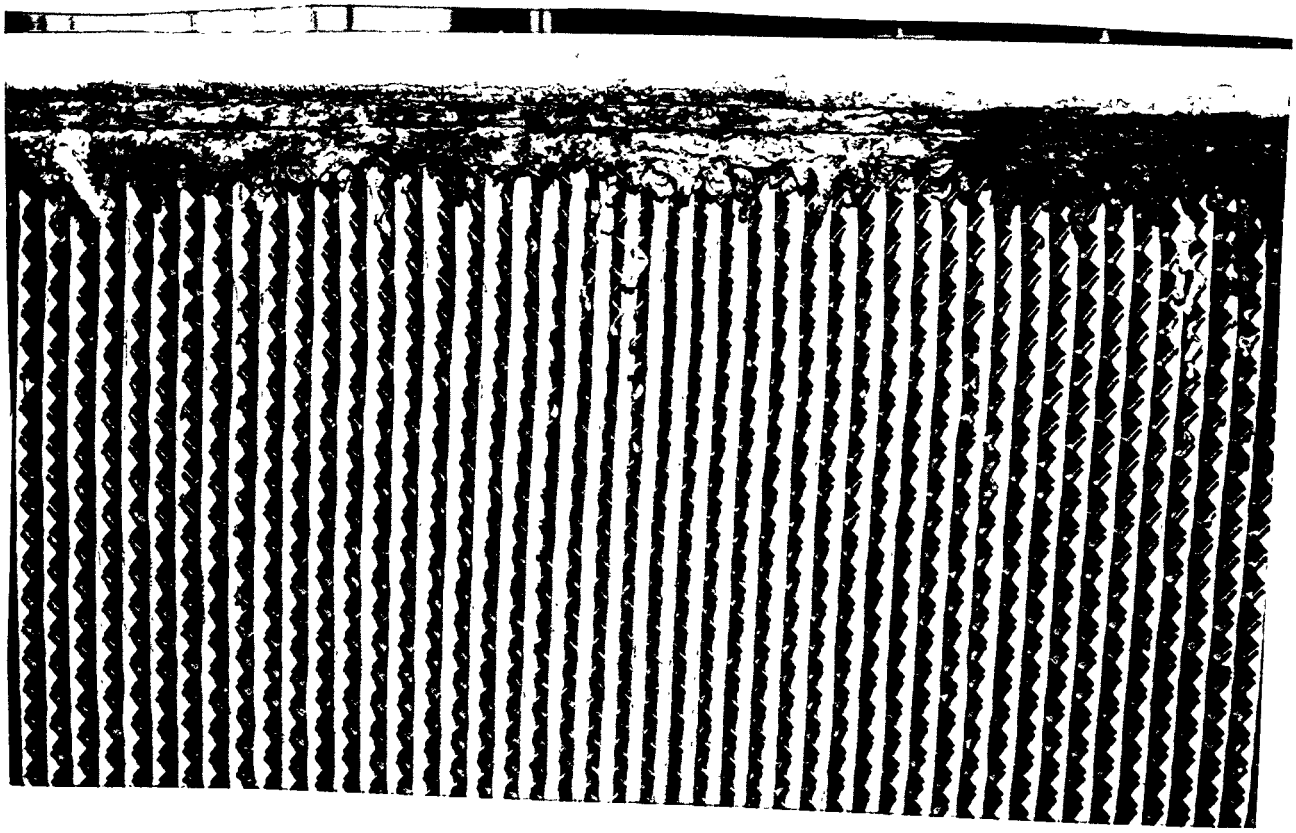


Figure 12. Photograph of 2.7-year-old filter of Manufacturer A showing the crack where the sealant had burned away.

The damaged 2.7-year-old filter of Manufacturer A was then subjected to the overpressure test. During the test, a major portion of the filter pack was pushed out of the frame about one inch, but was not blown out. Figure 13 shows the exit side of the filter with portions of the filter pack pushed out. (Note that the gasketed side of the single gasket filter must be faced upstream in the heated air test and reversed to face downstream in the overpressure test.) The displacement of the filter pack can be seen by the extension of the aluminum separators beyond the gasket in the right side of the photograph in Figure 13. The central portion of the filter pack was pushed out even farther after the seventh aluminum separator from

the right. The step dislocation is seen by the exposed filter medium separating the two portions of the filter pack. The filter also showed more extensive buckling near the bottom frame and had media tears in that region as seen in Figure 14.

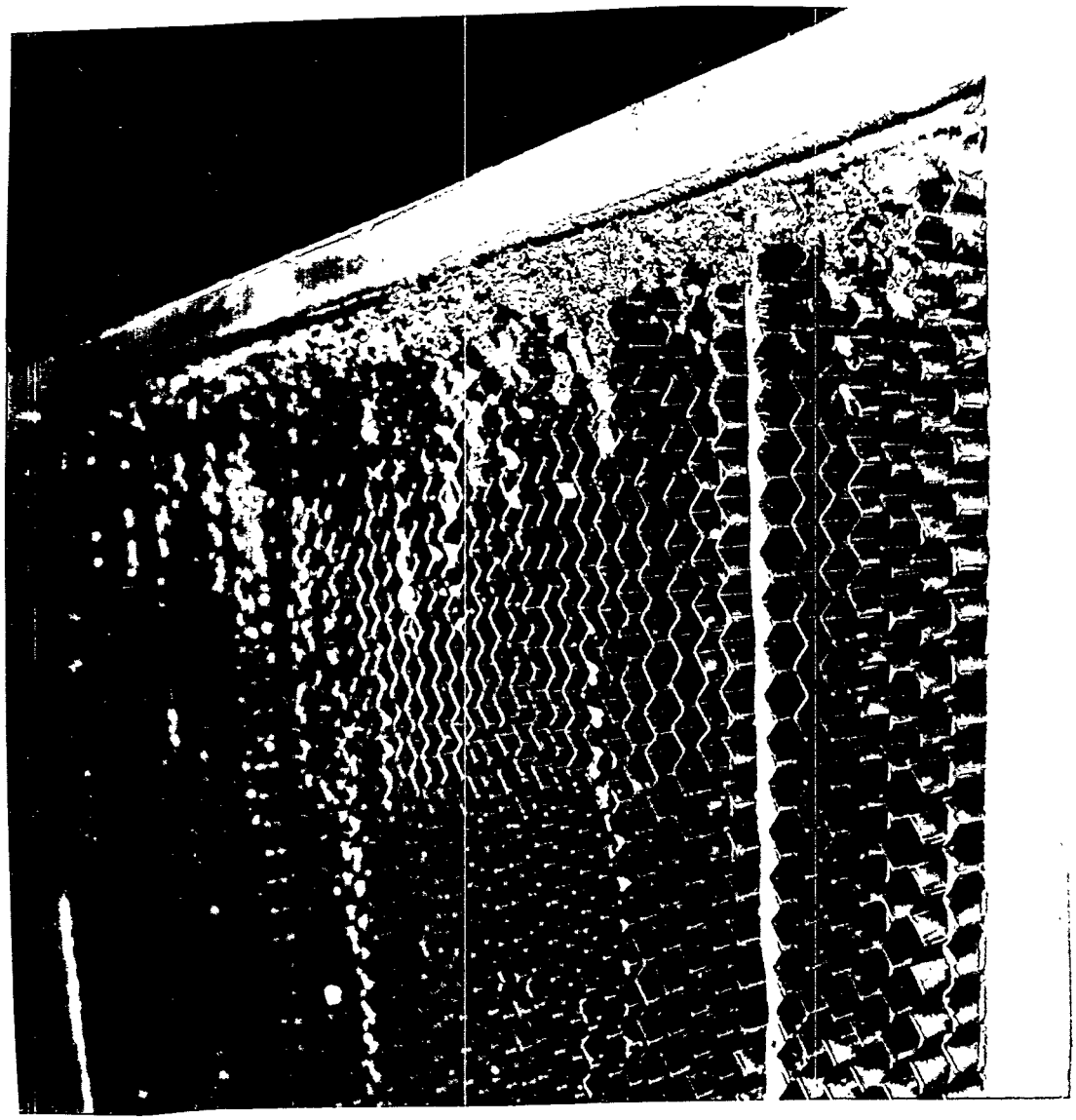


Figure 13. Photograph of the exit side of the 2.7-year-old filter of Manufacturer A showing filter pack pushed out of its frame after the overpressure test.

Penetration tests on the filter showed 5.86% and 7.51% at 100% and 20% flow respectively. Under slightly more severe conditions, the media pack would have completely blown out of its frame and resulted in 0% efficiency.

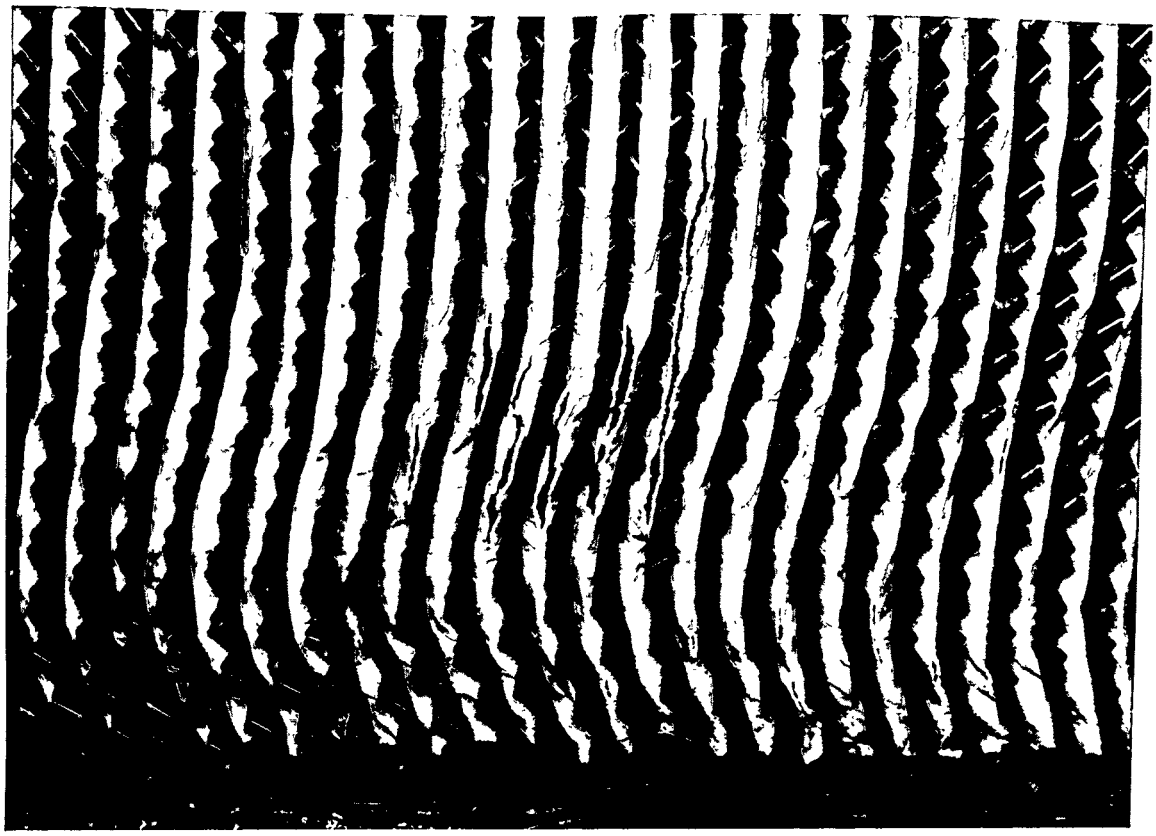


Figure 14. Photograph of the exit side of the 2.7-year-old filter of Manufacturer A showing media buckling and tearing after the overpressure test.

The 0.2-year-old filter of Manufacturer A had far less damage after the heated air and overpressure tests than the 2.7-year-old filter. The filter sealant had charred but held the media pack tight to the frame. There were four slits along the media pleats, about 3-4 inches long, on the exit side of the filter along with slight media buckling. During the heated air test, smoke was jetting out from the joints in the plywood frame. Bubbles were forming at the cracks indicating that the glue used to seal the frame was melting and burning.

The large difference in the fitler penetration measurements for the three filters in Table 3 after the heated air test is due to the type of material used to seal the filter pack into the filter frame. Table 4 shows the filter penetration data befofe and after the heated air test retabulated from Table 3 along with filter age, medium tensile strength and sealant type. Filter age and medium tensile strength do not correlate well with the penetration data. However there is a

strong correlation between the sealant type and the penetration. The filter using a flame inhibited rubber is unaffected by the heated air, while the filters using a urethane sealant show significant degredation to heated air.

Table 4. Effect of filter parameters on heated air test results

Parameter			Heated Air Test	
Age (years)	Medium Tensile Strength (lb/in)	Sealant	Penetration (%)	
			Before	After
9.8	8.3	inhib. rubber	.012	.015
2.7	3.0	urethane	.018	1.24
0.2	9.3	urethane	.008	.77

Based on the limited filter tests summarized in Tables 2-4, we can conclude that filter age is not the primary factor affecting filter performance in the heated air and overpressure qualification tests. The 2.7-year-old filter of Manufacturer A had failed the overpressure test due to tears in the filter media. We have shown this filter has poor tensile strength due to insufficient binder. The filter also had severe structural damage during the heated air test, although it met the maximum penetration requirements. The sealant was completely burned out around the filter pack, leaving it susceptible to blow-out. In contrast, the filter of Manufacturer C that used a sealant made from flame inhibited rubber showed no deterioration after the heated air test. Thus, our preliminary conclusion is that the type of materials and the construction design used in HEPA filters have a greater effect of HEPA performance than age.

IV. Rocky Flats HEPA Failures

Our study to determine the shelf-life of HEPA filters coincided with a problem encountered at the Rocky Flats Plant with aged HEPA filters of Manufacturer A. The volume of filters purchased by Rocky Flats during 1990 and 1991 exceeded the capacity of the local storage facility. The filters were found acceptable upon quality assurance tests following delivery and were stored in containers which were outside and exposed to weather changes. During 1993 and 1994 many of these filters were withdrawn from containers, re-certified with quality assurance tests, and issued for installation. Some filters at the installation site were found with damaged

medium; and following installation of others, in-place testing identified installed HEPAs with damage not prominently discernible to visual inspection.

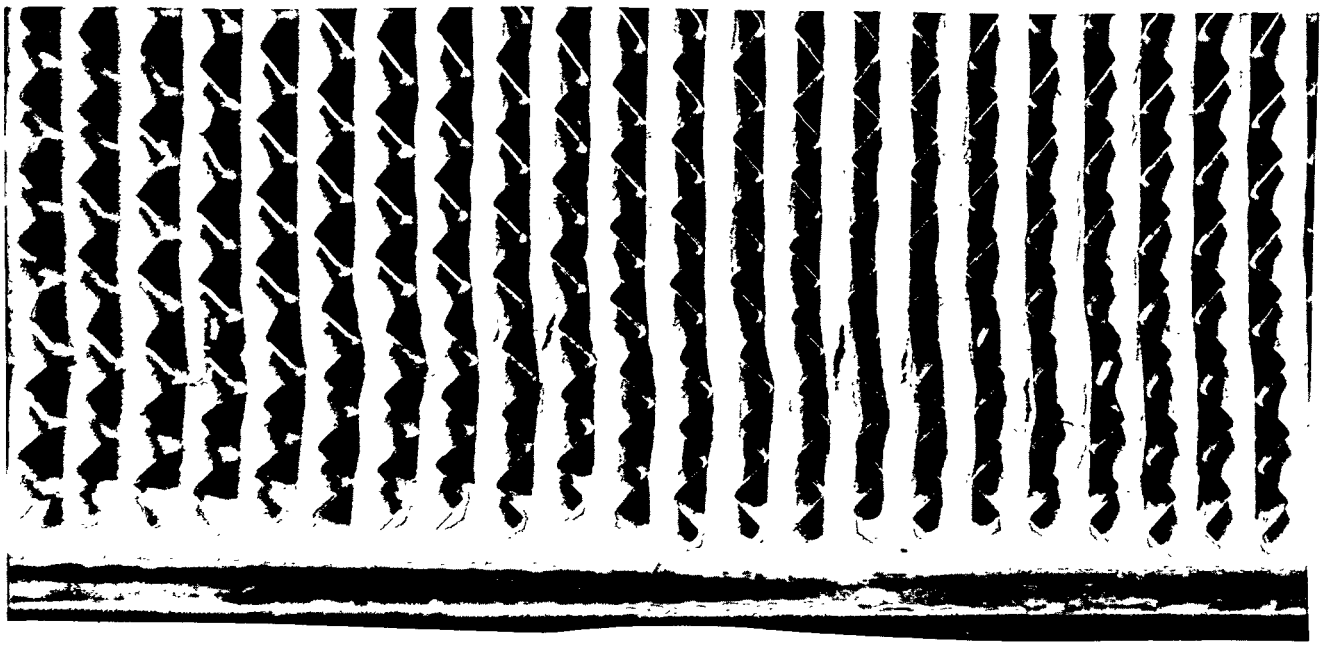


Figure 15. Photograph of filter media tears encountered at the Rocky Flats Plant during the installation of 2- to 3-year-old filters of Manufacturer A.

An independent study by some of us (J.K.F. and F.R.) showed that the filter failures were due to small tears in the media that occurred during the transportation and installation of the filters. The mechanical stresses placed on the filter were evidently sufficient to damage the media. Figure 15 shows the typical damage that was seen with these filters. Our present study showed the damage was due to low tensile strength as a result of insufficient binder.

V. Conclusions

We have completed a preliminary study using filter media tests and filter qualification tests to investigate the effect of shelf-life on HEPA filter performance. The filter media was extracted from filters stored up to 18 years. Our media studies showed that the tensile strength decreased with age, but the data was not sufficient to establish a shelf-life. The media was also analyzed for the percent of binder and Nomex fibers using a TGA. These analyses

demonstrated that the low tensile strength of the 1990 and 1991 media samples of Manufacturer A were due to insufficient binder.

The filter qualification tests (heated air and overpressure) conducted on different aged filters showed that filter age is not the primary factor affecting filter performance. The type of materials and the construction design have a greater effect on HEPA performance than age. We found that the 1991 filters of Manufacturer A had failed the overpressure test. The filter had passed the heated air test, but left the filter so structurally weak, it was prone to blow-out. The unsatisfactory performance of these filters is partially due to the lack of sufficient binder in the medium. The use of a combustible sealant that burns through the filter depth and leaves the filter pack loose inside the filter frame is also not acceptable.

This is a preliminary study of some of the factors involved in establishing a shelf-life for HEPA filters. The study was limited in both the type of parameters studied and the number of samples. Future studies should also include all of the filter components, e.g. sealants, gaskets, frame, in addition to the media. Other filter tests such as simulation of tornado pulses and smoke plugging should also be conducted in addition to the qualification tests to better differentiate the age effects.

The discovery that sub-standard HEPA filters have found their way into DOE facilities despite an existing system of regulations and quality control laboratories suggests that the present system must be improved. All HEPA filter manufacturers that sell their filters to DOE facilities must have every type of size 5 filter they sell pass the qualification tests (heated air, overpressure, and rough handling) every five years [7, 10]. The enforcement of the requirements is typically left to buyers in the DOE facilities. Some manufacturers do not requalify their filters every five years. Our present study has shown that even if a manufacturer qualifies their filters every five years as required, the manufacturing process or materials used may change and lead to sub-standard filters. We recommend that DOE initiate a filter qualification program to prevent this occurrence.

Our study also suggests that the Nuclear Regulatory Commission (NRC) should re-evaluate its policy of not requiring HEPA filter qualification tests at the DOE filter test stations [11]. The NRC study cited the low rejection rate (about 6%) at the filter test stations

and the cost of the certification tests as the basis for their decision to discontinue certification tests. Unfortunately, the study was incomplete because it did not address the risk from the low probability event of rejected filters. The study ignored the consequences of installing unqualified filters in radioactive facilities. Once installed, these filters have a high disposal cost (about $5K) and potential decreased performance (12). Based only on the cost to test and dispose of contaminated HEPA filters, the filter certification test has a break-even cost at 0.7% HEPA failure rate. The NRC study also assumed that the low rejection rate was not related to the requirement for filter certification, an assumption strongly denied by filter manufacturers and HEPA experts. A classic risk analysis based on the product of the probability of failure (% failure) times the effect or consequence of the failure would justify having HEPA filters certified in the filter test stations.

VI. Acknowledgment

The authors thank the Department of Energy, Office of Engineering and Operations Support, Defense Programs, for supporting this study and for providing helpful suggestions and guidance.

We also thank Messrs. Rick Caulfield, Gerald Rivera and Wayne Gilbert of EG&G Rocky Flats Plant for their assistance in conducting the tests.

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DISCUSSION

FIRST: You and I have talked about this before, but I want to make sure that the point gets brought out for everyone. It is very troubling that manufacturers make unannounced changes in the composition of the materials that go into the filters, and change manufacturing methods. My question to you is, how do you account for changes in technology over time, as well as the aging effect? In other words, were filters just less strong 18 years ago than they are now, and are we now seeing deterioration effects that will no longer occur?

GILBERT: I do not think there is any one factor involved. In some cases, during paper making, a new binder is put in, but I do not think it results in any change in the aging effect. I hold to the opinion that everything about the filter does age. However, I do not think that great care is taken to avoid it. For example, the fiber is made from time to time and may be stored a long time before delivery to the paper maker. After the paper making process, paper may be stored for long periods by the filter manufacturer. Since each paper manufacturer seems to use a recipe which is unique, maybe somebody slipped. So, filter failures have a number of sources. Some aging effects merely aggravate what started out as a deficiency.

WEBER: Glass is a water soluble material, and I do not know whether aging, as seen in tensile degradation, could be accelerated by humidity. The solubility of glass is not that great in neutral or acidic water. In our laboratories, when we subject fresh resin-bonded glass fiber media to short-term wetting, there is an immediate very large loss of tensile strength that appears to be associated with the way the fibers are adhering to one another, rather than to some short-term degradation of the fiber. It could have something to do with a change in lubricity affecting the resin binder held on the glass fibers. I am curious to know if you are considering using microscopy or some other examination of failed media to find out which components are the most affected.

GILBERT: No, our study was very limited in scope. A variety of laboratory studies still are

needed to assess the response of the fibers and medium to age and various environmental conditions. However, your question illustrates the need for a cooperative endeavor involving filter users and the filter industry, including fiber and medium manufacturers. This, in turn, might lead to a voluntary agreement for controlled storage at the source to preserve the tensile strength of fibers after they are produced. Medium fabricators also will want to review their similar responsibility for the period before converting the fibers into HEPA and ULPA media. Such a cooperative effort would benefit both users and producers.

MALIK: I have a comment on the tensile strength of glass media. Glass fibers do not have natural hydrogen-bonding, like cellulose fibers do. The amount and type of binder in glass media is very important. Measuring internal bonding strength of glass paper may be more appropriate than testing tensile strength alone or performing both tests can give a better picture for the strength of glass media.

GILBERT: I am not sure that there is a floor that could be put on the amount of binder because each glass paper manufacturer makes a product that is slightly different.

BERGMAN: I want to make the point that the greatest effect on filter strength I have found is related to the type of materials used in construction. Aging has a much lower effect, overall. Even if this information came across earlier, I want to emphasize the point.

TARTAGLIA: What was the basis for the test temperature (200°-600° C) and test pressure, 10 in.w.? Most nuclear plants would challenge the filters design basis accident (DBA) only to 200° F maximum and 2½ in.w.!

GILBERT: The standard is not intended for nuclear power plants. The test temperature and pressure are specified in the military specification for the heated air and overpressure tests, respectively.

YETTER: HEPA filters are the main boundary of protection to keep both chemical and radiation particles from the air we breathe. I am troubled by what I heard in the talks, that the rules and regulations regarding the assembly of HEPA filters may not have the enforcement that at one time was in place. I sincerely hope that we maintain high standards for these products.

GILBERT: DOE has maintained strict requirements for DOP tests of every HEPA filter used in DOE facilities. A sample of filters also is required to pass qualification tests for exposure to heated air, overpressure and rough handling every five years or whenever new filter designs or filter materials are used. Although these requirements generally have insured that high quality filters are supplied to DOE, our study has shown that substandard filters can pass the DOE penetration and resistance tests but not meet the required qualification tests. The problem is that the existing qualification tests do not catch deficiencies due to production changes between the five-year intervals. In our paper we showed that the strength of the medium of Manufacturer A had decreased significantly in 1990 and 1991 due to insufficient binder. Filters fabricated with this medium were weak structurally and failed the overpressure test. The problem with the current qualification test requirements can be solved by adopting a standard practice of testing representative samples periodically, not just the testing of custom manufactured filters initially and every five years. The DOE facility at Rocky Flats has the capability to perform the required qualification tests.

EFFECTS ON THE EFFICIENCY OF ACTIVATED CARBON ON EXPOSURE TO WELDING FUMES

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Southern Company Services, Inc., Birmingham, Alabama

1.0 Abstract

There exists in the nuclear power industry the belief that welding fumes have a detrimental effect on activated charcoal used for air cleaning and filtration processes. It is the intention of this paper to document that certain types of welding fumes have little or no effect on the effectiveness of the carbon filter air filtration efficiency when directly exposed to a controlled amount of welding fumes for a short-term period. No effort has been made to determine the effect of long-term welding fume exposure on activated carbon methyl iodide removal efficiency.

The welding processes studied were restricted to shielded metal arc welding (SMAW), flux cored arc welding (FCAW), gas tungsten arc welding (GTAW) and gas metal arc welding (GMAW) processes. These were some of the most common welding processes used at a nuclear site. Contrary to the SMAW and FCAW processes, the GTAW (or TIG) and the GMAW (or MIG) welding processes do not require the use of flux as part of the overall process. Credit was taken for these processes occurring in inert gas environments and producting minimal amount of smoke. It was concluded that a study involving the SMAW process would also envelop the effects of the TIG and MIG welding processes.

The quantity of welding fumes generated during the arc welding process is a function of the particular process, the size and type of electrode, welding machine amperage, and operator proficiency. For this study, the amount of welding for specific testing was equated to the amount of welding normally conducted during plant unit outages. Different welding electrodes were also evaluated, and the subsequent testing was limited to an E7018 electrode which was judged to be representative of all carbon and stainless steel electrodes commonly used at the site.

The effect of welding fumes on activated charcoal was tested using a filtration unit complete with prefilters, upstream and downstream high efficiency particulate air (HEPA) filters, and a carbon adsorber section. The complete system was field tested in accordance with ANSI N510 standards prior to exposing the filters and the adsorber bed to welding fumes. The carbon samples were tested at an established laboratory using ASTM D3803-1989 standards.

2.0 Introduction

The most common welding processes performed at the nuclear site are the SMAW, FCAW, GTAW, and GMAW processes (figure 1). Contrary to SMAW and FCAW processes, the GTAW (or TIG) process does not require the use of a flux as a part of the overall process. In this process an arc is established between a nonconsumable tungsten electrode and the piece being welded. Once the arc is established, bare filler metal is fed into the weld area where it coalesces with molten base metal to form the weld. Argon, helium, or a combination of the two is used as shielding gas and serves as current carrying plasma. Smoke and fumes produced are minimal since there is no metal to produce them and the gases used are inert. Smoke and fume generation during the GMAW process is also minimal since this process uses inert gas as the shield gas and consumable filler metal electrodes that leave no slag.

The SMAW process, commonly referred to as the "stick" welding process, is widely used and, at present, is the most dominant of various weld processes. This process uses an electric arc between a covered metal electrode and the base metal to obtain fusion. The electrode provides shielding and filler metal for the weld. This weld process generates the most smoke and fumes. It is assumed that a study involving the SMAW process will also envelop the FCAW process.

3.0 Purpose of the Study

This study was developed as a result of a statement in the Nuclear Regulatory Commission Regulatory Guide 1.52 which specified retesting of the filtration system following events of chemical release. The welding process was inferred as a form of chemical release and, consequently, it was understood that charcoal in a filtration system may be poisoned or contaminated following exposure to smoke and fumes generated during the welding process.

At the nuclear site, the standby gas treatment (SGT) system filter unit is normally operated twice a day for about an hour to purge the primary containment. The site configuration requires the SGT system to be operated during all refueling outages. This has required stoppage of all work involving welding for the duration of the purge. This restriction resulted in wasted manhours during outages. It was, therefore, desirable to establish the basis for the concern and investigate the most cost effective solution.

4.0 Analysis

The American Welding Society (AWS) has conducted numerous studies involving generation of smoke and fumes during various weld processes. All the

studies were restricted to concerns of health and the impact on the welders from short- and long-term exposure to the fumes. Table 1 lists the various welding contaminants and their effect on different parts of the human body. The welding process studies have concluded that the various gaseous byproducts do not constitute a major environmental problem.

The smoke and fumes generate particulate with average diameters from 0.3 to 0.7 microns. The particle behavior is influenced by the air movement in the immediate vicinity of the weld process. For a fume control ventilation design process, capture velocities in excess of 2000 ft/min are commonly used in order for the particle to generate sufficient momentum for transportation with the air stream. The building ventilation systems that normally exist in most plants are designed with terminal velocities less than 1500 ft/min and often do not provide enough momentum for transportation of the smoke/fume particles. It was concluded that any particulate that gains access to the plant exhaust system will eventually be trapped by either the prefilters or the upstream HEPA filters.

For the specific welding fume study, the type of welding processes commonly used in that plant were considered along with various electrodes. The predominant types of welding electrodes used were E7018, E308-16, E309-16, E316-16, and E71T-1. Excerpts from the AWS publication addressing E7018, E308-16, E70S-3, and E70T-1 are reproduced for reference, and a summary of results is presented in tables 2 through 6. The E7018 is a carbon steel low-alloy electrode. The E308-16 is considered to be representative of the E309-16 and E316-16 electrodes, with the major differences being the increase of chrome and nickel contents in the E309 and an increase in the molybdenum content in the case of the E316. The E70T-1 is basically the same as the E71T-1 electrode with the difference being that it is a consumable type of electrode. The E70S-3 is a consumable electrode used in the GTAW process.

Reviewing representative data from the AWS publication did not support any definitive conclusion regarding detrimental effects of welding fumes on activated charcoal; however, it was determined from the review that the possibility existed that activated charcoal may not be affected by short-term exposure to welding fumes for welding processes commonly performed at that plant site.

A test to determine the effect of welding fumes on activated charcoal was performed at Charcoal Service Corporation, Bath, North Carolina. The SMAW process was selected for the study since the fumes generated by this process are considered to be representative of the other weld processes. Welding electrode E7018-3/32 in. is commonly used at the nuclear site; therefore electrode type E7018 was selected for the test. The weld fume generated by the use of the carbon steel E7018 electrode was expected to be "dirtier" than stainless steel electrodes table 2. For the test, the largest diameter (5/32 in.) was selected. The test setup is shown in figure 2.

The number of electrodes for the prototype test was determined based on the air change per hour in the reactor building with SGT filtration train operating and the maximum number of electrode consumption rate per hour (i.e., during any given plant unit outage).

4.1 Test case

No. of electrodes= Building air change/hr (0.22) x 70 electrodes /hr $\simeq$ 15 electrodes.

4.2 Test Parameters

A filtration unit was utilized complete with a prefilter, a HEPA filter, carbon adsorber beds, dampers, necessary instrumentation, and a fan. The design features of the filtration unit are as follows:

- Two - 2 in. charcoal adsorber beds with sampling canisters.bed velocity: 40 fpm max; residence time >0.4 sec.
- One prefilter (45% efficiency based on ASHRAE 52-76).
- One 99.97% HEPA filter.
- One fan with provision for air flow adjustments: 0-2000 cfm.
- Flow control dampers.
- Differential Pressure gauges across each filter banks.
- Relative humidity indicator.

4.3 Test Setup

An air filtration system complete with an exhaust (capture) hood and ductwork was setup as shown in figure 2.

4.4 Test Conditions

The filtration system setup was functionally tested in accordance with ANSI N510-1989. The following tests were performed on the filter unit prior to testing the system with welding fumes:

- Visual inspection (check gaskets, etc.).
- Housing leak test (leak rate <3%/hr of housing volume at 10 in. wg).
- Air flow capacity and distribution test (verify that fan capacity is within $\pm$10% design flow; verify that airflow distribution through the HEPA filter and adsorber beds is fairly uniform at the designated flow rate).
- Air aerosol mixing uniformity test (verify proper mixing, e.g., max min readings do not vary more than $\pm$ 20%).

- In-place filter testing (verify HEPA efficiency >99.97%; adsorber bed efficiency >99.9%; the two 2-in. beds are tested as one 4-in. bed)

Two tests were conducted. Each test resulted in the complete consumption of 15 E7018 electrodes.

- Test No.1:"Protected Test". Setup is shown in figure 2

- Test No. 2:"Unprotected Test". Setup is shown in figure 2, but without the prefilter and the HEPA filter.

The two 2-in. charcoal banks were identified as banks "A" (upstream) and "B"(downstream). Nineteen samples were generated from this test: nine were from the "Protected Test", nine were from the "Unprotected Test", and one sample of new carbon was included from the original drum.

The sequence of sample removal during each test was as follows:

- Samples 1A and 1B after consumption of 5 electrodes.
- Samples 2A and 2B after consumption of 10 electrodes.
- Samples 3A and 3B after consumption of 15 electrodes.
- Samples 4A and 4B after consumption of 15 electrodes.
- Sample 5A was taken directly from the first adsorber bed after running the fan for 30 minutes to account for any offgas.

Test samples, including new carbon samples, were sent to Nuclear Consulting Services, Inc., (NUCON) for laboratory testing in accordance with ASTM D3803-1989. . The carbon samples were tested for methyl iodide removal efficiency at test condition of 30°C/95% RH.

Samples from the "Protected" and "Unprotected" tests were tested as follows:

1. 1A + 1B - combined as a 4-in bed
2. 2A + 2B - combined as a 4-in bed
3. 3A + 3B - combined as a 4-in bed
4. 1A, 2A, 3A - tested separately as a 2-in bed
5. 4A - tested separately as a 2-in bed
6. 4B - tested separately as a 2-in bed
7. 5A - tested separately as a 2-in bed

It should be noted that the established test conditions were more severe than those experienced under real conditions because the charcoal was subjected to the total amount of welding fumes generated during the test period.

5. 0 Conclusion

Laboratory results are included in table 7. The results indicate that no detrimental effects were observed on the efficiencies of the charcoal test samples when exposed to a controlled amount of welding.

As a result of this test, credit was taken for not retesting the filtration unit when exposed to welding fumes. However, for added conservatism, charcoal in the filtration units were laboratory tested following plant outages for the next plant cycles. Charcoal test data following outages have shown no adverse impact on the charcoal efficiencies.

Acknowledgments

The author wishes to acknowledge the services of Mr. Latif Kaid and Mr. Al Dunbar of Charcoal Service Corporation Inc. who assisted with the tests. Further appreciation is extended to Mr. Mike Richards of Southern Company Services, Inc., for furnishing information on the various welding processes.

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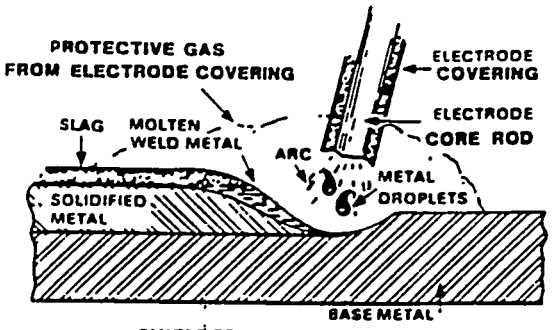
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FIGURE 1

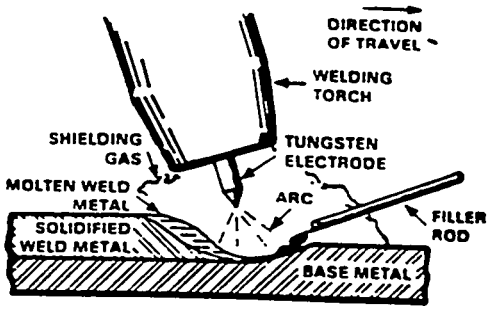
WELDING PROCESSES

Shielded Metal Arc Welding (SMAW)



SHIELDED METAL ARC WELDING

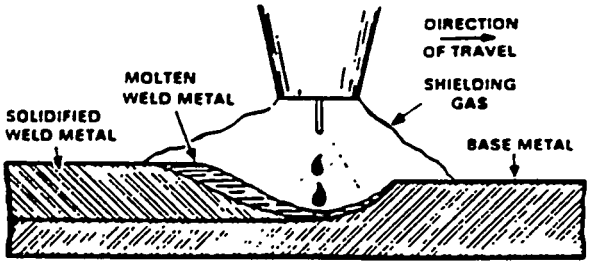
Gas Tungsten Arc Welding (GTAW)



GAS TUNGSTEN ARC WELDING

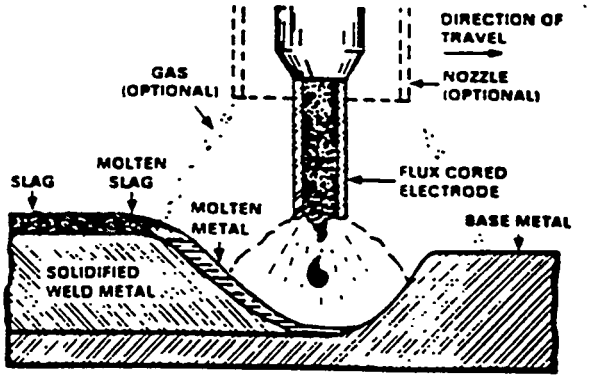
Gas Metal Arc Welding (GMAW)

Short circuiting, globular or sprav transfer modes possible.



GAS METAL ARC WELDING

Flux Cored Arc Welding (FCAW)



FLUX CORED ARC WELDING

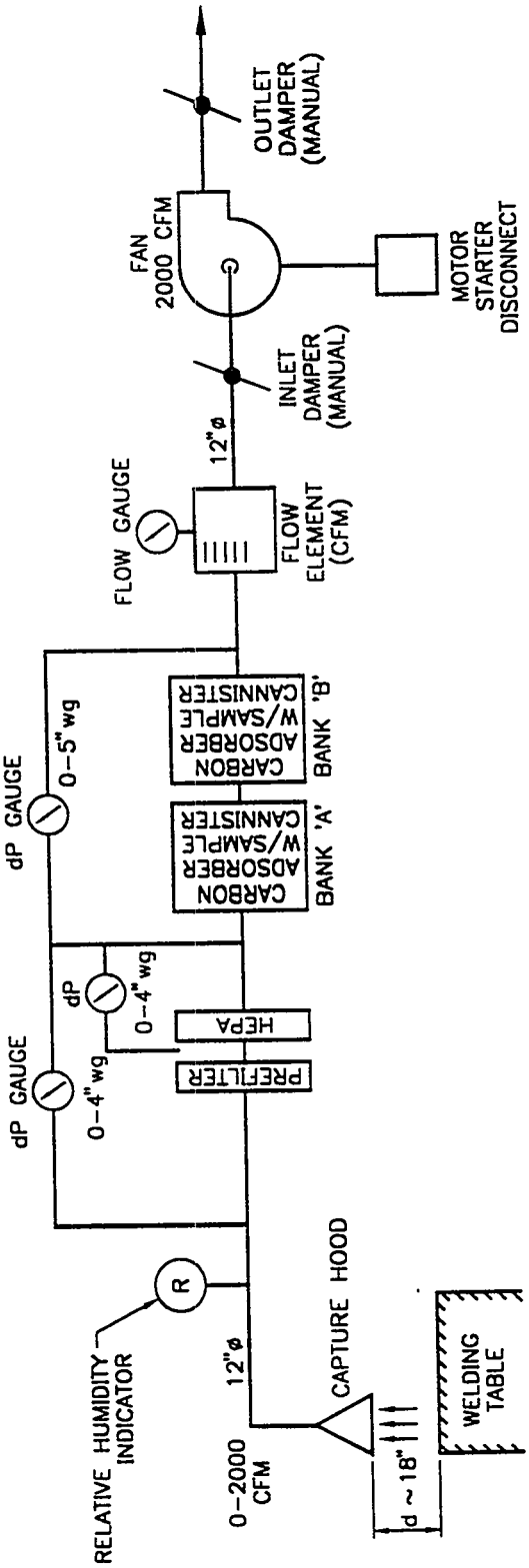


FIGURE 2

TEST SETUP

TABLE 1

TYPICAL WELDING CONTAMINANTS

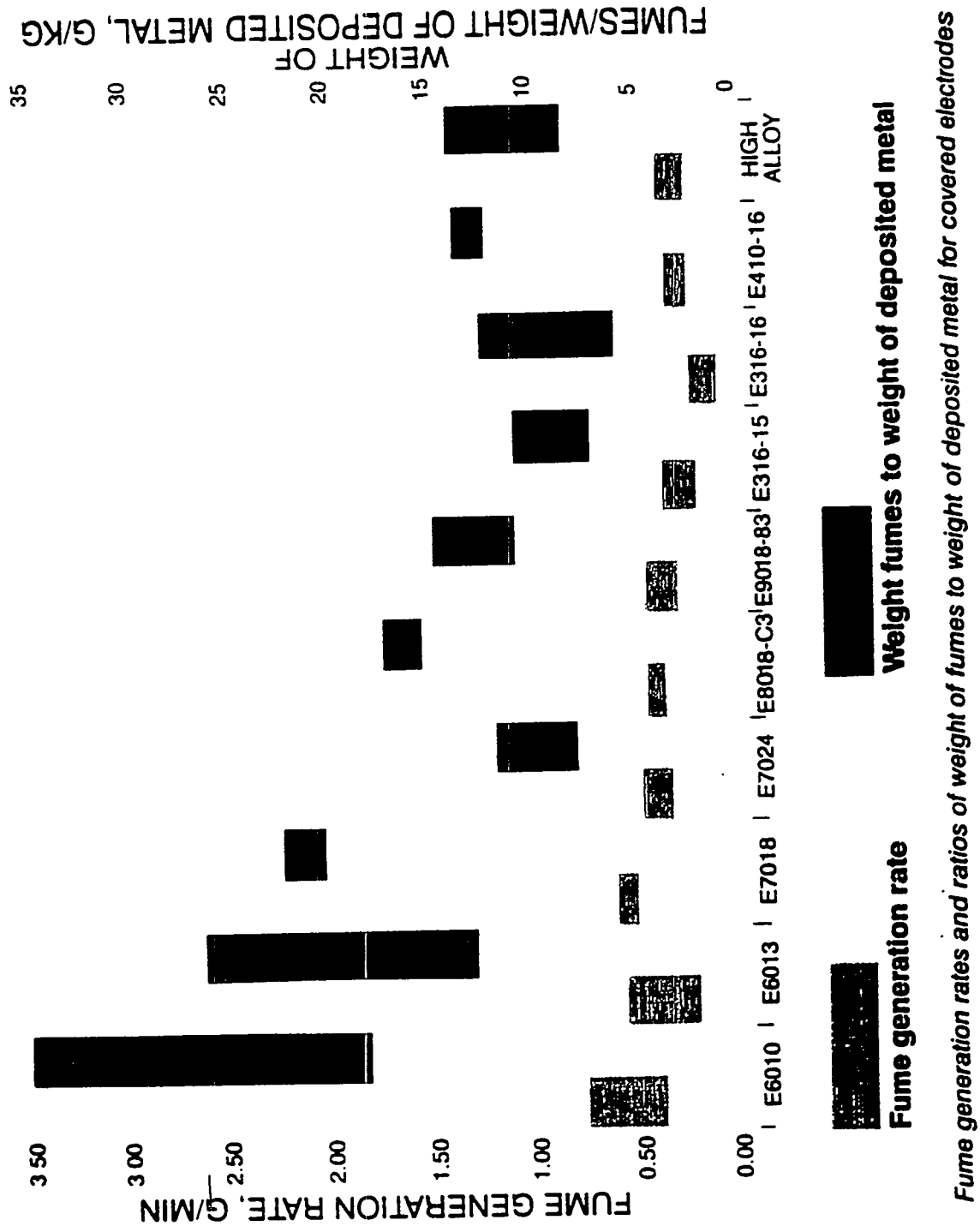
MATERIAL	THRESHOLD LIMIT VALUES[a]						POSSIBLE HEALTH HAZARD WHEN TLV IS EXCEEDED
	Time-weighted averages (TWA), mg/m^3 in 8 hours			Short-term exposure limits (STEL), mg/m^3 in 15 min			
FUMES	**ACGIH[b]**	**OSHA**	**NIOSH**	**ACGIH[b]**	**OSHA**	**NIOSH**	
Alundum (Al_2O_3)	10	–	–	20	–	–	Nuisance dust
Arsenic (As)	0.2	0.5	–	–		0.002	Lung & lymphatic cancer, respiratory irritant
Beryllium (Be)	0.002	0.002	–	–	0.005 (30 min)	0.005 (130 min)	Lung cancer, pneumonitis
Cadmium Oxide (Cd)	0.05	0.1	0 4	–	0.3	0 2	Lung and kidney effects, pulmonary edema
Chromium (Cr)	0.5	0.5	0.001[c]	–	–	0.05[c]	Lung cancer, skin ulcers, nasal perforation
	–	–	0.001[d]	–	–	–	
Cobalt (Co)	0.05	.05	–	–	–	–	Chemical pneumonitis
Copper Fume (Cu)	0.2	0.1	–	–	–	–	Irritant, fume fever
Fluoride (F)	2.5	2.5	2.5	–	–	–	Kidney and bone effects
Iron Oxide (Fe_2O_3)	5	10	–	–	–	–	Nuisance dust
Lead, Inorganic (Pb)	0.15	0.2	<0.05	–	–	–	Systemic poisoning
Magnesium Oxide (MgO)	10	10	–	–	–	–	Fume fever
Manganese (Mn)	5	–	–	–	5	–	Nervous system disorders
Molybdenum (Mo)	5[e]	5[e]	–	–	–	–	Respiratory Irritant
	10[f]	10[f]	–	–	–	–	
Nickel (Ni)	1	.007	0.015	–	–	–	Lung and nasal cancer, skin effects
Tin Oxide (SnO_2)	2	2	–	–	–	–	Nuisance dust
Titanium Dioxide (TiO_2)	10	10	–	–	–	–	Nuisance dust
Vanadium (V)	0.05	–	–	–	0 1	0.05	Eye, skin, and lung effects
Zinc Oxide (ZnO)	5	5	5	10	–	15	Fume fever
Welding fumes	5	5	–	–	–	–	Depends on constituents
GASES							
Carbon Dioxide (CO_2)	9,000	10,000	10,000	54,000	30,000	30,000 (10 min)	Respiratory effects
Carbon Monoxide (CO)	55	35	35	440	200	200	Asphyxiant
Hydrogen Fluoride (HF)	2.5	3	2.5[g]	–	6	5[g]	Skin, eye irritation, bone effects
Nitrogen Dioxide (NO_2)	6	5	–	–	1	1	Airway effects
Nitrogen Monoxide (NO)	–	25	25	–	–	–	Blood effects
Ozone (O_3)	0.2	0.1	–	0.6	0.3	–	Pulmonary edema
Phosgene ($CoCl_2$)	0.4	0.1	0.1	–	–	0.2	Airway effects

a. Threshold limit values for the measured elements and compounds, given in parentheses by their chemical symbols, are listed in mg/m^3 (milligrams per cubic metre) for particular fumes and in ppm (parts per million) for gasses
b. ACGIH = American Conference of Governmental Industrial Hygienists.
c. Carcinogenic chromium.
d. Noncarcinogenic chromium.
e. Soluble molybdenum.
f. Insoluble molybdenum.
g Reported as mg/m^3, milligrams of fluorine per cubic metre.

Source: National Institute of Occupational Safety and Health (NIOSH)
Note: OSHA values are mandated; ACGIH and NIOSH values are recommendations only.

TABLE 2

Electrode Classification



Fume generation rates and ratios of weight of fumes to weight of deposited metal for covered electrodes

TABLE 3

Table 5b
E7018 — Percent Composition

	Percent by number	Percent by surface area	Percent by volume	Percent by mass
1 C/Organic	—	—	—	—
2 Fe	0.7	0.5	0.3	0.5
3 Fe/Low Si	1.0	0.4	0.1	0.2
4 Fe/High Si	0.2	0.0	0.0	0.0
5 Fe-Mn	0.9	0.4	0.2	0.5
6 Fe-Cr	0.1	0.0	0.0	0.0
7 Fe-Cr-Mn	—	—	—	—
8 Fe-Cu	0.4	0.9	0.6	1.3
9 Fe-V	0.6	0.2	0.1	0.1
10 Fe-Zn	—	—	—	—
11 Fe-Ti	0.2	0.4	0.5	0.5
12 Fe-Al	0.5	0.2	0.0	0.1
13 Fe-S	0.2	0.1	0.0	0.0
14 Fe-Cl	0.2	0.1	0.1	0.0
15 K-Cr	0.9	0.6	0.3	0.3
16 K-Fe	14.2	13.8	12.3	11.7
17 K-Cr-Fe	1.7	1.52	0.8	0.7
18 Ca-Fe	15.0	11.7	9.3	9.1
19 K-Ca-Fe	48.0	49.3	42.6	42.0
20 K-Mn	2.1	2.1	1.6	1.5
21 K-Ti	0.8	0.5	0.3	0.3
22 Ca-Cr	0.2	0.0	0.0	0.0
23 Ca-Mn	0.4	0.5	0.6	0.5
24 Ca-Ti	0.3	0.2	0.1	0.1
25 K-Ca	6.4	11.6	27.2	26.8
26 K-Rich	2.1	1.7	1.0	0.8
27 Ca-Rich	1.5	1.5	0.8	0.9
28 Pb-Rich	0.1	0.1	0.0	0.1
29 Cr-Rich	0.1	0.0	0.0	0.0
30 Mn-Rich	0.1	0.1	0.0	0.1
31 Cu-Rich	0.2	0.2	0.1	0.2
32 Ti-Rich	—	—	—	—
33 V-Rich	0.1	0.1	0.0	0.1
34 Fe-Rich	0.5	0.9	0.5	0.9
35 Na-Rich	—	—	—	—
36 Mg-Rich	—	—	—	—
37 Al-Rich	0.2	0.1	0.0	0.1
38 Si-Rich	0.2	0.4	0.6	0.5
39 Misc	0.4	0.1	0.1	0.0

TABLE 4

Table 5d
E70T-1 — Percent Composition

	Percent by number	Percent by surface area	Percent by volume	Percent by mass
1 C/Organic	0.8	0.5	0.2	0.1
2 Fe	14.4	9.2	2.7	2.5
3 Fe/Low Si	24.9	19.9	6.0	4.6
4 Fe/High Si	4.9	3.4	0.9	0.4
5 Fe-Mn	36.7	51.1	86.0	89.5
6 Fe-Cr	1.7	1.2	0.3	0.3
7 Fe-Cr-Mn	0.8	0.9	0.2	0.2
8 Fe-Cu	0.4	0.4	0.1	0.1
9 Fe-V	1.2	1.5	0.5	0.4
10 Fe-Zn	0.2	0.3	0.1	0.1
11 Fe-Ti	2.8	2.3	0.4	0.2
12 Fe-Al	0.1	0.0	0.0	0.0
13 Fe-S	0.2	0.2	0.0	0.0
14 Fe-Cl	0.5	0.4	0.1	0.0
15 K-Cr	—	—	—	—
16 K-Fe	1.3	1.0	0.4	0.2
17 K-Cr-Fe	0.1	0.1	0.0	0.0
18 Ca-Fe	2.5	3.0	1.0	0.5
19 K-Ca-Fe	0.2	0.5	0.1	0.1
20 K-Mn	—	—	—	—
21 K-Ti	—	—	—	—
22 Ca-Cr	—	—	—	—
23 Ca-Mn	0.1	0.2	0.1	0.0
24 Ca-Ti	0.1	0.0	0.0	0.0
25 K-Ca	—	—	—	—
26 K-Rich	—	—	—	—
27 Ca-Rich	—	—	—	—
28 Pb-Rich	0.1	0.0	0.0	0.0
29 Cr-Rich	0.1	0.0	0.0	0.0
30 Mn-Rich	2.8	1.8	0.5	0.4
31 Cu-Rich	—	—	—	—
32 Ti-Rich	—	—	—	—
33 V-Rich	—	—	—	—
34 Fe-Rich	1.0	0.8	0.2	0.2
35 Na-Rich	—	—	—	—
36 Mg-Rich	—	—	—	—
37 Al-Rich	—	—	—	—
38 Si-Rich	1.8	1.3	0.2	0.1
39 Misc	—	—	—	—

TABLE 5

Table 5c
E70S-3 — Percent Composition

	Percent by number	Percent by surface area	Percent by volume	Percent by mass
1 C/Organic	0.1	0.1	0.0	0.0
2 Fe	9.0	13.2	18.3	20.3
3 Fe/Low Si	14.4	32.9	58.9	56.7
4 Fe/High Si	0.3	0.2	0.1	0.0
5 Fe-Mn	16.9	12.5	6.0	7.7
6 Fe-Cr	5.1	3.9	1.2	1.8
7 Fe-Cr-Mn	0.9	0.9	0.3	0.4
8 Fe-Cu	4.2	2.4	0.7	0.9
9 Fe-V	3.7	3.4	1.7	1.5
10 Fe-Zn	2.6	1.9	0.7	0.8
11 Fe-Ti	3.0	2.2	0.8	0.4
12 Fe-Al	5.8	2.4	0.6	0.5
13 Fe-S	3.3	1.9	0.5	0.5
14 Fe-Cl	3.3	4.8	3.6	2.2
15 K-Cr	—	—	—	—
16 K-Fe	5.8	2.9	0.8	0.5
17 K-Cr-Fe	0.9	0.2	0.0	0.0
18 Ca-Fe	9.8	5.5	1.7	1.0
19 K-Ca-Fe	1.1	0.4	0.1	0.1
20 K-Mn	0.2	0.1	0.0	0.0
21 K-Ti	—	—	—	—
22 Ca-Cr	0.1	0.1	0.0	0.0
23 Ca-Mn	—	—	—	—
24 Ca-Ti	—	—	—	—
25 K-Ca	—	—	—	—
26 K-Rich	0.1	0.0	0.0	0.0
27 Ca-Rich	0.1	0.0	0.0	0.0
28 Pb-Rich	0.1	0.2	0.1	0.2
29 Cr-Rich	0.3	0.1	0.0	0.0
30 Mn-Rich	0.4	0.1	0.0	0.0
31 Cu-Rich	0.1	0.0	0.0	0.0
32 Ti-Rich	—	—	—	—
33 V-Rich	0.1	0.0	0.0	0.0
34 Fe-Rich	8.0	7.6	4.0	4.5
35 Na-Rich	—	—	—	—
36 Mg-Rich	—	—	—	—
37 Al-Rich	0.2	0.0	0.0	0.0
38 Si-Rich	0.3	0.1	0.0	0.0
39 Misc	0.3	0.1	0.0	0.0

TABLE 6

Table 5e
E308-16 — Percent Composition

	Percent by number	Percent by surface area	Percent by volume	Percent by mass
1 C/Organic	0.5	1.1	0.3	0.2
2 Fe	0.8	0.8	1.1	1.6
3 Fe/Low Si	1.2	0.5	0.2	0.2
4 Fe/High Si	1.2	1.0	0.7	0.5
5 Fe-Mn	5.1	5.2	4.2	7.0
6 Fe-Cr	1.5	1.8	1.4	2.7
7 Fe-Cr-Mn	1.4	0.8	0.5	0.8
8 Fe-Cu	0.6	0.5	0.3	0.5
9 Fe-V	0.9	0.8	0.6	0.7
10 Fe-Zn	0.8	0.6	0.5	0.8
11 Fe-Ti	1.0	0.4	0.2	0.1
12 Fe-Al	0.9	0.4	0.2	0.2
13 Fe-S	1.0	4.4	11.9	13.4
14 Fe-Cl	0.9	0.6	0.8	0.6
15 K-Cr	5.3	4.9	4.1	3.1
16 K-Fe	18.1	18.7	14.3	10.8
17 K-Cr-Fe	5.5	7.5	14.3	10.9
18 Ca-Fe	7.0	4.9	3.0	2.4
19 K-Ca-Fe	7.1	6.7	4.8	3.8
20 K-Mn	7.5	6.8	5.2	4.0
21 K-Ti	2.9	2.5	2.0	2.0
22 Ca-Cr	0.7	0.5	0.4	0.3
23 Ca-Mn	1.7	1.1	0.7	0.5
24 Ca-Ti	0.5	0.6	0.5	0.6
25 K-Ca	1.5	1.8	1.4	1.1
26 K-Rich	6.5	6.8	5.4	3.8
27 Ca-Rich	1.2	0.9	0.5	0.5
28 Pb-Rich	0.1	0.0	0.0	0.0
29 Cr-Rich	1.9	2.5	1.6	2.4
30 Mn-Rich	2.8	1.8	0.9	1.2
31 Cu-Rich	0.9	1.2	0.9	1.6
32 Ti-Rich	0.9	0.5	0.3	0.4
33 V-Rich	0.5	0.3	0.2	0.2
34 Fe-Rich	1.2	3.8	12.2	17.7
35 Na-Rich	—	—	—	—
36 Mg-Rich	0.1	0.2	0.1	0.1
37 Al-Rich	0.9	0.8	0.5	0.4
38 Si-Rich	6.2	6.2	3.9	2.8
39 Misc	1.3	0.5	0.2	0.1

TABLE 7

LABORATORY TEST RESULTS SUMMARY

Test Standard: ASTM D3803-1989

Test Condition: 30°C/95% RH

SAMPLE NO.	EFFICIENCY "PROTECTED" MODE	EFFICIENCY "UNPROTECTED" MODE
1A	98.627	98.827
1A +1B	99.983	99.977
2A	99.018	99.619
2A + 2B	99.970	99.982
3A	99.092	99.181
3A + 3B	99.982	99.971
4A	99.269	99.641
4B	99.461	99.222
5A "OFF GAS"	99.421	99.412
NEW CARBON	99.548	

DISCUSSION

JENKINS: You mentioned that your test results for "protected" charcoal (charcoal with installed upstream prefilters and HEPA filters) showed less impact than charcoal without upstream filters. I want to comment that my experience with TM1-Unit 2 cleanup supports your observation that upstream filters help. We noted that plasma-arc cutting of steel sections generated significant smoke that rapidly loaded up exhaust system prefilters. I noticed that the prefilters, which normally would operate for months at a time without having to be replaced or without showing any significant pressure increase, loaded up within weeks after we started doing extensive plasma arc cutting. Prefilters for HEPA filters make a big difference.

GHOSH: I agree with that.

WEIDLER: What is your charcoal impregnated within the system?

GHOSH: It was 2.5% TEDA.

ADAMS: What is the difference in length of duct run that you used in your test compared to the length of your normal standby gas system from containment?

GHOSH: The difference in the length? The plant that you are talking about is a two unit plant. Unit one uses the normal ventilation ducting system. Unit two has a separate duct, which takes direct suction from the secondary containment.

ADAMS: Do you have an estimate of the length?

GHOSH: I'd be guessing, but I would say unit two has about 200 ft., and unit one at least 300 ft.

ADAMS: In the test set-up that you had, was there ever a check to determine if any particulate was in the corrugated ducting that you used in your filter suction?

GHOSH: No. We did not look at the ducting, we were monitoring the smoke.

ZAVADOSKI: Were you successful in submitting a change in your technical specifications to the NRC to have welding fumes removed?

GHOSH: We did not go to NRC. We made a clarification to our technical specifications because we had a valid test to back it up. With that added assurance, we went back and did another test following a refueling outage. We did not see any degradation of the charcoal, and took that into consideration.

HOLTORP: What is the impregnant and impregnation level of the carbon in this system?

GHOSH: 2.5% TEDA.

GREENE: I have been debating for quite some time whether to make these points about the

effects of uncertainties in all aspects of the presentations that have been made at this conference and previous conferences. I believe that the attention given to errors in experiments, uncertainties in analyses and statistics, as well as the propagation and reflection of the uncertainties through regulations and enforcement is completely inadequate in this industry. Because it has been found to be unacceptable in all other fields of scientific research, product development, and government regulation, I have been reluctant to bring it up at this point. I am sure there are a lot of people who do not want to hear this, but I have been keeping a list while attending a lot of presentations and I have found that air filtration people apparently are unable to quantify uncertainties with respect to manufacturing standards and manufacturing practice, testing standards and testing practice, regulatory standards and regulatory guidance, compliance with performance standards with respect to trends in data for performance of filters as well as filter materials, uncertainties in the ranges of data and error bounds of data. Analysis of uncertainties is not propagated to consequences of risk, margins to performance limits apparently are not addressed. Thresholds for failure are portrayed as single point variables. Needs for risk- based regulation have not been recognized, and the statistics of historical performance are not adequately portrayed in order to evaluate trends.

Recently, we have recognized difficulties in predicting the performance of nuclear air filtration practice. I believe a lot of the difficulty in being able to forecast the performance of nuclear air filtration systems under projected harsh environmental conditions is due to a chronic failure to address the effects of uncertainties in all of these aspects. I believe that progress in filtration performance will continue to be hampered by such an approach. Most important, it is apparent that the people doing the research and making the regulations cannot tell the manufacturers, with any degree of certainty, what is causing degraded performance. A special difficulty that this field is going to have to address, not only DOE but NRC as well, is a lack of reliable and defensible estimates of expected performance, consequences, and risk in continuing operations in degraded facilities, especially under accident conditions and projected decommissioning and decontamination operations. I would like to hope that, in the future, people who do this work and who make presentations in this forum will give more attention to the effects of uncertainties on the extrapolation of their work so it can be more accurately put into practice.

CLOSING COMMENTS OF SESSION CO-CHAIRMAN BERGMAN

All five papers at this session have dealt with aging effects on air cleaning equipment. Mr. Winegardner discussed aging of HEPA filters and carbon adsorbers. He showed that stress factors such as heat, radiation, volatile compounds, and aerosols add to the aging effect. He specifically cited tensile strength and water repellency as two items that decrease with age. With regard to carbon, he attributes the loss of activity in adsorption sites to moisture and volatile organics. His conclusion is that degradation of HEPA filters and carbon adsorbers occurs.

A paper was presented that outlines a three step process to estimate the residual efficiency of HEPA filters during and after design basis accidents. The first step is determining what the environmental parameters are and what the pressure will be during the design basis accident. The next step compares the pressure drop under accident conditions to a threshold value. When the filter has a lower pressure drop than the threshold value, it survives and efficiency can be determined. When it does not survive because high pressure drop causes structural damage, it is assigned zero efficiency. When it survives, efficiency is determined from a tabulated summary of values that reflect efficiencies under different conditions.

Another paper described the performance of HEPA filters under dynamic conditions at temperatures up to 500° C. It was very interesting to learn that the rectangular filters, when heated up to 200° C for 48 hours showed less than 0.02% penetration, whereas circular filters showed 1% penetration. Slits and cracks were found in the medium plus deformation of the sealing gasket. The presentation was concluded with a description of a new prototype high-strength filter that showed 0.65% penetration at temperatures up to 400° C.

Preliminary studies made to determine the shelf life of HEPA filters found that media lost tensile strength with age, but the work was not yet sufficient to establish a shelf lifetime. The critical factors for determining filter performance are the type of materials used and construction methods. They are much more important than filter age. As an example, low media strength resulting in structural damage in a filter was due to insufficient binder, rather than age.

It was shown that short-term exposure to welding fumes had no effect on the effectiveness of activated carbon adsorbers. Long-term studies were not conducted. The welding fume exposures occurred in a complete system consisting of prefilters, upstream and downstream HEPA filters, and a carbon adsorber. The important finding was that HEPA filters prevent welding fumes from reaching the carbon adsorber.

SESSION 15

WORKING LUNCHEON

Wednesday: July 27, 1994
Co-Chairmen: L. Soffer
R. R. Bellamy

Invited Speaker: The Honorable Kenneth C. Rogers, Commissioner

CLEAN AIR AND CLEAR RESPONSIBILITY
Commissioner Kenneth C. Rogers

CLEAN AIR AND CLEAR RESPONSIBILITY

PRESENTED BY

COMMISSIONER KENNETH C. ROGERS
U. S. NUCLEAR REGULATORY COMMISSION

AT THE

23RD DOE/NRC
NUCLEAR AIR CLEANING AND TREATMENT
CONFERENCE

BUFFALO, NEW YORK - JULY 27,1994

Good afternoon ladies and gentleman. I am pleased to have this opportunity to talk with you today about an issue that could have a significant effect on the future of nuclear air cleaning treatment and technology.

In this conference, you have been focussing on a variety of technological considerations related to nuclear air cleaning and treatment. I would encourage you to step back from the "how-tos" of this technology for the moment and to reflect instead on the regulatory framework in which you work and, more importantly, on the foundation of that framework. There are important changes that may be coming and, as professionals, your understanding and input will be needed to help guide these changes.

Today, I will discuss the regulatory regime for controlling emissions of radioactivity into the air. I will start with a brief review of the statutory basis for regulating radioactive air emissions. I will then discuss the NRC's regulatory approach to controlling such emissions, the principles which underlie that approach, and, in particular, what I believe to be the great value of the "as low as is reasonably achievable" or ALARA principle. Finally, I will talk about the current regulatory regime, some potential changes to it, and the issues that relate to how they are made.

Statutory Basis

Various acts and their amendments provide the statutory basis for regulating emissions of radioactivity into the air. Both NRC and the Environmental Protection Agency (EPA) are given regulatory authority by these acts. Under the Atomic Energy Act of 1954 (AEA) and Title II of the Energy Reorganization Act of 1974, NRC is authorized to license and regulate a range of activities connected with the use of source material, byproduct material, and special nuclear material. This authority includes the regulation of air emissions of these materials. Under the Clean Air Act and its amendments, EPA was given the authority to

regulate air emissions of radioactive materials including those that are covered by the Atomic Energy Act. Thus, two agencies have regulatory authority over the same air emissions. The statutory bases are different, however, and the two agencies exercise their authority in different ways. I would now like to review some important features of the NRC's approach.

The NRC's Regulatory Approach

NRC's approach for protecting people from radioactivity applies radiation protection principles that are used worldwide. It is based on the radiation protection recommendations of the International Commission on Radiation Protection, the ICRP and its U.S. counterpart, the National Council on Radiation Protection and Measurements. NRC's radiation protection standards are codified principally in 10 CFR Part 20 although some dose limiting requirements for particular classes of licensees are codified in other parts of Title 10. For example, Appendix I to 10 CFR Part 50 contains numerical guides for power reactor design objectives that are intended to keep radioactive effluents as low as is reasonably achievable.

Three basic principles underlie NRC's approach and these same principles govern radiation protection in most of the world. The first of these principles is that any activity that causes people to be exposed to ionizing radiation must have sufficient benefit to justify the exposure. This is the principle of "justification". The second principle is that exposures should be kept as low as is reasonably achievable -- the ALARA or "optimization" principle. The third principle is that the maximum dose to any individual should be limited so that an acceptable level of risk is not exceeded -- the "limitation" principle.

The principle of justification is not explicitly stated in NRC's regulations. Instead it is manifest in the choice of activities for which NRC grants licenses. The principles of limitation and optimization have explicit expression in Part 20, the standards for protection against radiation that apply to all licensees.

10 CFR Part 20 -- Standards for Protection Against Radiation

It is not my intent today to give you a detailed description of the requirements of 10 CFR Part 20; however, I would like to mention briefly some of its key features. I believe that you are familiar with them and I am confident that you will recognize their connection with the basic principles that I spoke about just a few moments ago. First of all, Part 20 includes occupational limits in Section 20.1201. These limits establish

the acceptable level of risk for those people who work with materials controlled by the Atomic Energy Act.

Second, Part 20 includes limits for exposures of members of the public by activities which use Atomic Energy Act materials. These are codified in Section 20.1301 and the principal requirement is that, with certain exceptions, exposures of members of the public may be no greater than 100 mrem per year. These limits establish the acceptable level of risk for members of the public and are fully consistent with the recommendation of the ICRP.

An essential aspect of radiation protection for members of the public is the control of radioactivity released in liquid and gaseous effluents. This control is addressed in Section 20.1302 which allows a licensee two approaches for demonstrating that its operation does not exceed the annual dose limit. The more commonly used approach involves a demonstration that concentrations of radioactivity in effluents do not exceed values specified in Table 2 of Appendix B to Part 20. These values assure that doses to an individual from radioactive emissions to the air do not exceed 50 mrem per year.

Finally, I would mention that Part 20 explicitly requires that licensees implement the ALARA principle as part of their radiation protection program. This requirement appears in Section 20.1101 and is to be applied for both occupational and public exposures. An important feature and a great strength of the ALARA principle is that it gives explicit recognition to real world factors that must be faced in providing radiation protection.

As used in Part 20, ALARA means, "making every reasonable effort to maintain exposures to radiation as far below the dose limits in this part as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest."

In the next few minutes I would like to share my thoughts about the value of the ALARA principle and the great benefits that I believe it has brought and can continue to bring to radiation protection.

ALARA -- A Win-Win-Win Principle

ALARA encourages reductions in exposures to radioactivity by weighing the value of such reductions against several countervailing, but equally important considerations. As such, I believe it should hold an important place in any licensee's radiation protection program.

At a workshop, about which I will say more in a few minutes, one participant suggested that the role of ALARA in regulation is troublesome because it has different meanings to different people. To him, one meaning involved the formal cost-benefit balancing and weighing of alternatives that is used in establishing regulatory limits. The other involved the less formal, good-judgement, good-faith implementation of technology and procedures that is used by licensees to reduce exposures. He believed it difficult for a regulator to rely on the latter.

I must say that I see things a bit differently. I see these as manifestations of the ALARA principle that represent opposite ends of a spectrum. At one end is a formal cost-benefit balancing of alternatives. Certainly, it can be used by regulators to establish limits. But, I would hasten to add that it can also be used by licensees. In complex situations, involving costly technology or practices, a licensee might well choose to do a formal ALARA analysis. Our regulations certainly allow for this.

More often though, the formal analysis itself would be more costly than implementation of a better technology or practice. In such situations, the licensee skips the analysis and "just does it". I see this as the other end of the same spectrum. Moreover, I do not see a lack of formality as troublesome. In NRC's regulatory approach, ALARA is used to achieve reductions below the radiation protection limits that establish an already acceptable level of risk. Rather than characterize licensees' implementation of ALARA as troublesome for a regulator, I would characterize it as realistic, flexible, and effective.

Let me be more specific. I would call it realistic because it does not make reduction of exposures the end-all and be-all of a licensee's operation. True, it recognizes the importance of such reductions, but it also recognizes the importance of the technical, social, and economic factors that a licensee must confront. I would call it flexible because it allows licensees broad discretion in determining how to achieve reductions. This offers licensees the opportunity to use their creativity and intimate knowledge of their own operations to get the best possible reduction in exposures given the other factors they must also consider.

I would call it effective because its practice results in doses that are generally lower than the allowable limits and often much lower. Our regulations require that licensees measure and record occupational exposures. NUREG-0713 is an annual NRC publication that compiles statistics for these exposures. When I look at these statistics, I am struck by a downward trend in occupational exposures over the years. ALARA may not be the only explanation, but I am confident that it has been a significant factor in creating this trend.

Because doses to individual members of the public are generally not measured directly, discerning any effect that practice of the ALARA principle may have had is somewhat more problematic. However, I think there is evidence that the principle has been at work here too. EPA used its COMPLY code to calculate maximum individual doses for 367 randomly chosen NRC licensees. The input data were obtained by an EPA questionnaire. The sample included the full range of non-power-reactor licensees.

I find the results impressive. The calculated doses for all 367 licensees were less than 10 mrem. The highest calculated dose was 8 mrem and only 7 calculated doses were in the range of 1 mrem to 10 mrem. The rest, 360 out of 367, were less than 1 mrem. Considering that revised Part 20 had not yet gone into effect when EPA did its survey and that, at the time, the Part 20 dose limit was 500 mrem I have to think that ALARA was a significant factor here too.

The NRC's ALARA requirement has another very important characteristic. Specifically, it places the burden for a top-notch radiation protection squarely on the licensee -- where it belongs. I believe that what evidence we have shows that, in general, NRC licensees have done well at shouldering this burden. What is more, they also derive the satisfaction of a job well done.

In April 1993, I had the good fortune of observing one of the workshops that NRC sponsored in connection with its enhanced participatory approach to developing site decommissioning standards. This particular workshop was held at King of Prussia, Pennsylvania and included an extensive and stimulating discussion of ALARA. In listening to this discussion, I was quite impressed by the spirited defense that the industry participants made for the ALARA principle as an effective driver of their activities. Their defense was obviously heartfelt and they seemed to take real satisfaction from the accomplishment it compelled. However, they were insistent that ALARA's realistic, flexible nature was essential to their commitment.

I believe that the ALARA principle has served the NRC well, the nuclear industry well, and the public well. I think of it as

a Win-Win-Win principle and believe that it must remain a central part of NRC's approach to radiation protection.

The Regulatory Regime for Air Emissions

There is currently a dual regulatory regime for emissions of radioactivity into the air from Atomic Energy Act materials. NRC regulates these emissions under the Atomic Energy Act. Its regulations require that radioactivity in these emissions result in doses no greater than 50 mrem and require further that licensees attempt to reduce doses below that level consistent with the ALARA principle.

EPA regulates these same emissions under its Clean Air Act authority and has established a dose standard for members of the public of 10 mrem per year. Under current EPA rules, NRC licensees must meet this standard. EPA has also established a reporting requirement for operations which result in radioactive air emissions that result in calculated doses in excess of 1 mrem. NRC licensees are required by EPA to comply with this requirement too.

There are obvious advantages to having a single regulatory regime. Specifically, there would be one Federal interface for NRC licensees, one set of requirements to meet, and one set of inspections. These are advantages for both licensees and the government. Consequently, when the Clean Air Act Amendments of 1990 were passed, they included a provision known as the Simpson Amendment which allows EPA to rescind its authority for these emissions if it determines that the NRC program provides protection of the public health with an ample margin of safety.

Consistent with the Simpson Amendment, the two agencies have been working toward rescission. There have been some successes -- primarily in connection with radon emissions from uranium mill tailings. I believe we are also close to rescission for power reactors. However, there have been some difficulties in reaching agreement about rescission for materials licensees.

The outstanding issue centers around the 10 mrem standard in EPA's Subpart I to 40 CFR Part 61. NRC is reluctant to adopt this value as a dose limit. To do so would be inconsistent with our reasons for revising 10 CFR Part 20. Furthermore, speaking as one Commissioner, I would be very reluctant, indeed, to undermine, in any way, the ALARA principle that has served us so well.

Thank you for your attention. I would be pleased to address any questions or comments that you might have.

DISCUSSION

HULL: What is the value of EPA's involvement in this same area? I am going to ask a philosophic question, based on the fact that I have an increasingly uncomfortable feeling that we are operating in a vacuum of our own professional activity. I cannot help but wonder about a justification for expending the resources devoted to this activity, compared to the possibility that these same resources could be allocated to other areas. My own feeling is that they could be more effective in promoting the public health.

ROGERS: I certainly agree with you. We would be delighted to have the whole thing close down and just proceed along. I think that our experience has been that, for materials licensees that we are responsible for, we feel very comfortable that our regulations and applications, the ALARA principle, has worked extremely well. Nevertheless, a problem was created by Congress when it gave both organizations authority over the same kinds of materials. They approach them from quite different points of view, and now, trying to bring them together has been rather difficult. Our problem seems to be that EPA feels a lack of confidence with respect to enforceability issues, i.e., what tools do we have to enforce our point of view when we do not have an absolute number that must be met? Our point of view is that use of absolute numbers is not the best way to get the least exposure. When you have a single number that everybody must meet, you will get that number and that is all you will get. You will never get any less than that. The ALARA principle encourages, actively encourages, licensees to try to drive toward much lower levels. And, in fact, I think the data seem to show that it is being achieved, and I believe very strongly that it will continue to be achieved in the future under the ALARA principle. I quite agree with you; I too think the taxpayers' money for EPA's work and licensees' money for our work (because we get our whole budget from our licensees) could be better spent on other things.

KARHNAK: I am from EPA. I do not know where to start, but let me offer just one point regarding your last comment. If, indeed, the 10 mrem limit were the only one to be met, all the DOE facilities would be at 10 mrem. In fact, we find two or three of them are above. So, I am not sure that your last point is valid.

SHER: Maybe it is irrelevant now, but is there any real conflict between the ALARA principle and the 10 mrem limit? There would only be a problem if ALARA did not allow you to get down to 10 mrem, not to get up to it.

ROGERS: The problem is enforceability. The question is whether you can enforce an ALARA goal or not. With all of our licensees, our experience has been that NRC's point of view, when conveyed seriously to licensees, prevails, and that the legal argument about whether something is enforceable or not, plays much less of a role than licensees deciding that it is very much in their interest to try to follow NRC's lead on where to go on things. This certainly has been true in the power reactor area, and I am quite sure it can be true in the materials area.

KOVACH, B.: Since I am a health physicist, I would like to express my acclamation. I am very glad to hear that the radiation limits have been so low lately. I had the feeling that it was due to my good work that NUCON radiation limits dropped, but it seems to me

that everybody met the 10 mrem limit. When I started to work in the nuclear field, the limits were much higher, and I did not know how to do reduce them, while today it is possible and looks like everybody is successful in reducing the radiation limits.

As a matter of fact, we canceled all the x-ray checks for NUCON field test personnel, since the exposure of one single x-ray check was ten times higher than a yearly exposure from nuclear power plants. On the same subject, one single medical test with radioisotope application will deposit enough activity into your body to trigger the alarm on the entrance gates of most nuclear power plants for months. The time has come when we have to praise the nuclear industry for its safety records and explain to the public that the high radon level in houses was discovered on workers' clothing, while coming from home and entering into the nuclear reactors. We actually reached the natural background levels and I would not spend more money to reduce them, but would maintain this level of consciousness and inform the public properly.

ROGERS: I agree with you on the public needing to know about it. Nevertheless, it has been my experience as a regulator over the last seven years that regulatory attention is absolutely essential. I believe that it is easy to backslide, and situations change with licensees, managements change, economic circumstances change, personnel change. And what is now a very good program may not always be a good program unless there is outside attention checking on it from time to time. So, I have to say that I do believe that regulation is important and regulatory attention is important. But it should not get in the way of encouraging licensees to do the best they possibly can on their own. I think that is part of the issue that is being discussed here philosophically.

ANON: My question is, do you have insight to determine whether regulation is leaning toward NRC or toward the EPA side?

ROGERS: At NRC, we are working very closely with EPA on this to try to decide what will be sufficient for them to agree that NRC can take authority here; that they can rescind their oversight and allow us to apply our own regulations to our licensees. We have not come to an agreement yet with the EPA. It is something we are working on, and we hope that we can come to an agreement on it in the near future. But, it has been a sticky issue for us, and I really cannot predict when or how it will come about. It is conceivable that we cannot come to an agreement, and dual regulation will persist for material licensees. I hope that is not the case. The President has issued directives to reduce regulatory burdens as much as possible. I think that we have had direction from the White House that we should try to avoid things like dual regulation. Congress passed the Simpson amendment to the act that I mentioned to encourage the two agencies to get together and find some way to avoid dual regulation. But at the moment, we have not come to a meeting of minds on this between NRC and EPA, because the approach that EPA would like us to take would pose some very serious problems for us in a number of ways. I'd rather not try to go into all of them one by one, but we are hopeful that we will be able to come together. We have worked with EPA quite well on a number of other issues and have come to agreements. This particular one, I think, is more difficult because it has specifically received a considerable amount of attention from some Congressional committees. And when you get into something like that, everybody gets a little antsy. We have not been able to come to an agreement yet. I hope that it

will take place, but I think we must face the possibility that we won't be able to agree.

LAWTON: My question is, how do NRC and EPA work together with materials that are mixed with radioactivity and chemicals?

ROGERS: Mixed waste situations?

LAWTON: Yes.

ROGERS: I really cannot say much about mixed waste. It is a very difficult area for us. We are much relieved when we find that we are dealing with just radioactive waste. Mixed waste, toxic chemicals mixed with radioactive waste, has given us all some very, very difficult problems to deal with. And I would say, at the moment, I do not really have anything very helpful to say. It is an area that NRC is working on, but I must say I cannot offer anything constructive on this very tough problem area.

KARHNAK: In regard to your question about dual regulation, I just want to add that the 1990 Simpson amendment had certain requirements and certain findings that the EPA has to make in order to be able to withdraw their rule. We face a legal challenge in the courts if we are not able to make those findings successfully. Those are the things the Commissioner has mentioned as things we are trying to work out, but we are both bound by them. The EPA is bound by having to meet the findings before we can go forward with the rescission. We would certainly like to do that. Exactly how to meet those findings, or an agreement in terms of what is sufficient to meet those findings, is part of what we are discussing right now.

WREN: I have not kept up with all the changes in the regulations in Canada, but I do not think that we have the ALARA principle in our regulation. Because Canada is a neighboring country, I am wondering whether there is a broad agreement between these two governments regarding nuclear regulations?

ROGERS: Not that I am aware of. We have informal exchanges quite frequently with Canadian regulators. I am not familiar with the materials area in Canada as much as I am with the power reactor area. I know that we do have a somewhat different approach to safety regulation of nuclear power reactors. Canada's reactor designs are quite different from any that we have in the United States, so their approach is somewhat different. Some of the problems we have, with respect to physical security questions and things of this sort, they do not have. But in the materials area, I am not familiar enough with the entire program. As you know, we have a rather special situation in the United States in that the Nuclear Regulatory Commission regulates only a very limited class of sources of radiation. We do not regulate x-ray machines or linear accelerators. We do not regulate naturally occurring radioactive products. We really regulate a very narrow class of materials. It is quite a few, but it isn't everything. In other countries, that is usually not the case. In many countries, health and safety regulation tends to be all radiation, whatever the sources are. We have the absurd situation in the United States that a cobalt source that produces a gamma ray of a certain energy is regulated by the NRC. The same kind of radiation, the same wave length and quantum energy radiation coming from, say, a linear accelerator or some other source, is not regulated. The health

effects of this radiation, when it finds its way into living material, are exactly the same, and yet we regulate the cobalt source, but we do not regulate other sources of the same kind of radiation. That is typically done at the state level, not at the federal level. It is a very complex situation in the United States with respect to regulation of radiation. The sources of radiation, if they are not reactor-produced materials, are regulated by the states, rather then the federal government.

SESSION 16

OPEN END

Wednesday: July 27, 1994
Co-Chairmen: M. W. First
K. Duvall

OPENING COMMENTS OF SESSION CO-CHAIRMAN FIRST

BAR CODED AIR SAMPLE TRACKING: HOW IT CAN ASSIST IN COMPLIANCE WITH THE DOE'S NEW 10 CFR 835 AND THE NRC'S 10 CFR 20 REGULATION
W. H. Bailey

EXPERIMENTAL STUDY OF ELEMENTARY COLLECTION EFFICIENCY OF AEROSOLS BY SPRAY: DESIGN OF THE EXPERIMENTAL DEVICE
D. Ducret, S. LeGarrec, J. Vendel

DEGRADATION OF HEPA FILTERS EXPOSED TO DMSO
W. Bergman, K. Wilson, G. Larsen, R. Lopez, J. LeMay

PROPOSED RETROFIT OF HEPA FILTER PLENUMS WITH INJECTION AND SAMPLING MANIFOLDS FOR IN-PLACE FILTER TESTING
J. K. Fretthold

OPENING COMMENTS OF SESSION CO-CHAIRMAN FIRST

Welcome to the session that we call "open end". This session follows a long tradition at the air cleaning conferences. The significance of the session is that it is a place for short presentations, for progress reports on research underway, for studies of operations that are in progress and that have already produced enough interesting significant information to make a progress report worthwhile. In addition to that, we encourage people who have problems of a puzzling nature to present them here and ask the assembled audience if there is somebody in the auditorium who might have had some experience or special knowledge that would be helpful in solving the problem. These are intended to be short presentations with a great deal of commentary following each one. This afternoon we have a number of presentations.

BAR CODED AIR SAMPLE TRACKING

HOW IT CAN ASSIST IN COMPLIANCE WITH THE DOE'S NEW 10 CFR 835 AND THE NRC'S 10 CFR 20 REGULATION

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1.0 Introduction

With new government regulations on the horizon, advanced technology will become a necessity for the accurate assessment of air sampling data and maintaining internal dose ALARA. Bar coding is one advanced technology which has provided significant improvements in radiological air sample tracking and accountability. When combined with specific hardware and software, bar code technology can be used to automate procedures and enhance the data accuracy associated with air sampling in the workplace. This paper discusses some of the regulatory issues regarding radiological air sampling and describes how improved bar coded accountability techniques can assist in regulatory compliance.

2.0 Air Sampling Regulations

Regulations specify that ambient air sampling be performed as a means of monitoring airborne radioactivity in and around work areas. Government regulations such as the Nuclear Regulatory Commission's (NRC) revised 10 CFR 20, the Department of Energy's (DOE) draft 10 CFR 835, and DOE Order 5480.11 state that occupational workers and the workplace be routinely monitored. Many facilities will need to modify their air sampling programs in order to comply with regulations.

2.1 Revised NRC 10 CFR 20 Regulations

The main impact of the revised 10 CFR 20 (which is scheduled to be implemented on January 1, 1994) is a requirement for facilities to sum both internal and external exposure. This requirement is based on ICRP methodology which states the sum of the risks from a given radionuclide to an organ or tissue should not exceed the risk of exposing the whole body uniformly to 5 rems per year. Because the 10 CFR 20 revision includes a more detailed system of dose limitations, many licensees will need to gather more detailed information than presently collected in order to accurately perform dose assessment. This will greatly effect how many facilities will implement air sampling to monitor for the potential of internal exposure. The NRC currently has issued, for comment, a draft regulatory guide titled "Air Sampling In The Workplace" as one method for meeting certain survey and dose assessment requirements proposed in the revised 10 CFR 20, "Standards for Protection

Against Radiation." This document suggests that air sampling can be used to:

(1) Ensure effective radioactive material confinement,
(2) Measure airborne radioactive material concentrations in the workplace,
(3) Estimate worker intakes,
(4) Determine posting requirements,
(5) Determine protective equipment requirements,
(6) Serve as an early warning for significantly elevated levels of airborne activity, and
(7) Determine the time of an exposure incident, a requirement for interpretation of bioassay data.

"Air Sampling in the Workplace" suggests that the extent of air sampling provided should be based on accurately documented historical air sample data and bioassay results. If such data is not available or is inaccurate, potential intakes and concentrations should be estimated based on consideration of the following:

(1) The quantity of radioactive material being handled,
(2) The material's Annual Limit on Intake (ALI),
(3) The release fraction of radioactive material (based on the material's physical form or use),
(4) The type of material confinement, and
(5) Other factors appropriate for the specific facility.

The need for accurately documented historical air sampling data will become more stringent upon implementation of the revised NRC and DOE regulations. Several basic components are needed in order to establish an accurate air sampling program. Draft Regulatory Guide DG-8003 "Air Sampling In The Workplace" discusses the basic aspects which must be considered in order to implement an effective air sampling program. These aspects include:

(1) The type of air sampling which should be used,
(2) The location of air samples,
(3) Demonstration that air samples are representative of air breathed by workers,
(4) Adjustments to Derived Air Concentrations (DAC),
(5) Measurement of the volume cf air sampled, and
(6) Evaluation of air sampling data.[1]

2.2 DOE Order 5480.11 and Draft 10 CFR 835 Regulations

DOE Order 5480.11, "Radiation Protection for Occupational Workers", was implemented to establish radiation protection standards and program requirements for DOE and DOE contractor operations. Proposed draft regulations, titled 10 CFR 835, could replace DOE Order 5480.11 in the near future. As currently written, the proposed 10 CFR 835 is very similar to the revised NRC 10 CFR 20 regulations in all aspects including air sampling.

The proposed rule would require occupational workers and the workplace to be routinely monitored. Retrospective monitoring will be required in areas where workers are likely to exceed an annual intake of 2% (or more) of the ALI. In addition, real-time monitoring will be required in areas where workers are likely to be exposed to concentrations exceeding 10% of DAC.

Monitoring of individuals and areas should be performed to:

(1) Demonstrate compliance with the regulations,
(2) Document radiological conditions in the workplace,
(3) Detect changes in radiological conditions,
(4) Detect the gradual buildup of radioactive materials in the workplace,
(5) Verify the effectiveness of engineering and process controls, and
(6) Reduce radiation exposure.

In the event of upset conditions, analysis of data obtained from representative air sampling can be used to alert health physics personnel to possible internal exposure of an individual from the intake of radioactive material. Actual intake can then be measured using whole body counting and/or other bioassay techniques, as appropriate. According to the new rule, estimations of internal dose equivalent shall be based on bioassay data rather than air concentration values. However, air concentration values may be used for estimating internal dose if bioassay data is unavailable or inadequate.[2]

3.0 Effects of Regulations on Air Sampling Information Systems

New NRC and DOE requirements will likely effect air sampling information systems in many ways. Because many radionuclide airborne limits will be dramatically reduced by new regulations, additional data may be required in order to more accurately assess worker exposure. In addition, analysis trending and quality control methods may also be effected.

3.1 Data Requirements for Air Sampling

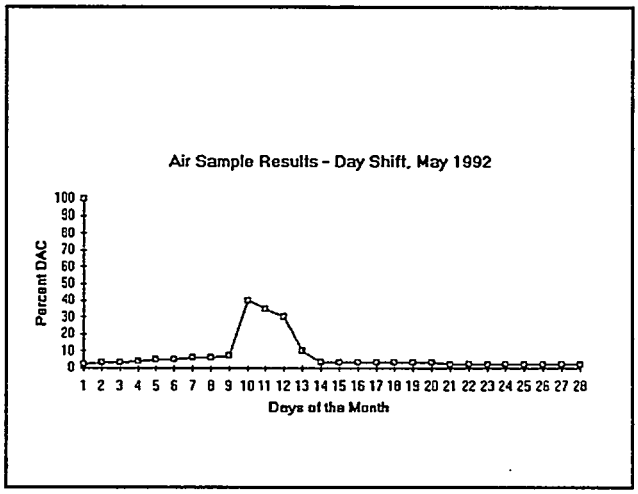
Proposed air sampling regulations could increase the amount of data required for air sampling due to the need to record additional information, such as particle size information, lung solubility studies, and air-flow rate correction calculations. Facilities which deal with alpha emitting radionuclides such as uranium and thorium will be most effected.

Without controls, as the amount of information gathered increases, so does the potential for more errors associated with air sample data collection. It is estimated that a facility which maintains and analyzes 100 area sampling locations and 20 lapel

samples at a rate of three shifts per day will require approximately 100 megabytes of annual computer storage for associated air sample data.[3]

3.2 Air Sample Analysis Trending and Quality Control

Air sample data can be used for analysis trending and quality control charts. This is highly desirable for demonstrating an effective ALARA program. These analysis functions can be easily communicated by charts and graphs which show air sampling results for various locations over a period of time. Analysis trending is also particularly useful when analyzed with respect to individual stations, areas, or buildings and associated activities performed during a specified period of time.



3.3 Accurate Record Keeping

A quality control method to ensure more accurate record keeping for air sampling begins by linking each sample filter to the sampling location. Without a high degree of confidence that air sample accountability is maintained, data regarding air sample locations should not be considered representative. For this reason, many licensees are evaluating new ways to collect data to insure a high degree of air sampling accountability.

4.0 Data Collection with Bar Codes

Bar codes offer one of the most accurate methods for data collection within the workplace. Air sample data collection can be greatly improved by implementing bar codes as a method of automatic identification. Through the bar coding of air sample filters, sampling locations, and personnel badges, technicians can use portable bar code readers to quickly and accurately collect air sample filters and corresponding data. Data entry using bar codes is one million times more accurate than manual key punch entry.[4] Therefore, air sample records created through use of bar codes ensures greater accuracy and reliability.

4.1 Bar Code Techniques

There are many types of bar coding formats (called symbologies)

available today. Depending upon the application or industry, a specific symbology is often required. For example, the retail industry standardized on the Universal Product Code (UPC) symbology. Some symbologies are designed for specific applications, such as Interleaved 2 of 5, which only contains numeric data.

For air sampling, symbology Code 3 of 9 (CODE 39) is recommended. The CODE 39 format is capable of handling the full range of alpha and numeric data and includes self checking characters built into the code itself. These capabilities provide a wide variety of air sample identification and error checking features to ensure better data accuracy and sample accountability.

4.2 Portable Bar Code Readers

Devices for reading bar codes are becoming smaller and smaller. Presently, a bar code decoder integrated into a powerful computer can be carried in the palm of the hand. These machines are capable of storing over one million bytes of information in non-volatile memory. Some models even contain fixed disk drives.

Sophisticated programs can turn these portable readers into devices which promote "Automated Procedures". Automated procedures refer to advances in portable bar code reader software which allows only valid data to be collected in the proper procedural sequence. Most portable readers contain a full keyboard, a 40- to 80-character display, an audible speaker, and enough battery power to collect data for an eight-hour period. Bar code scanning devices, such as lasers, can be integrated into portable bar code readers. In addition, most units can support external scanning devices such as wand scanners.

5.0 Implementing Bar Codes for Air Sampling

By implementing bar codes for air sampling, technicians can utilize the portable bar code reader to automatically perform many operations. For example, each air sample station can be checked for activation and to ensure that technicians perform proper procedures when bar codes are scanned. In addition, the portable reader can be used to automatically record the time and date of sample installation and collection via an internal clock.

5.1 General Air Sampling

By using specialized software operating on a personal computer, technicians can store data about each sampling area. This data can then be uploaded to the portable bar code reader. When linked with RWP's or personnel access records, general air sample information can be used to monitor for elevated airborne activity and to determine the need for bioassay analysis.

5.2 Representative Air Sampling

Specialized software and bar coding can also be used to provide dose estimates from stationary representative air samplers. When linked with a "time-in-area" system, Health Physics personnel can estimate an individual's internal exposure by correlating the air sample results with the time-in-area records. By bar coding the location at each workstation, each worker can scan their bar coded identification badge or card to record the time spent at each specific workstation. Air sample collection and analysis data can then be linked to each individual to assist with representative breathing zone sampling.

Lapel samples, another form of representative air sampling, can also be tracked using bar codes. In this case, the individual can be issued a bar coded identification such as a badge or card. Portable sampling pumps are bar coded to provide a more accurate record of lapel sampling activities. Portable bar code readers are used to obtain lapel sample information such as an individual's identification and lapel sample pump number. PC software can be utilized to download lapel sampling information.

5.3 Early Warning Air Samplers

Early warning samplers perform two functions simultaneously. They provide general area sampling while warning workers in the area of elevated airborne activity. Early warning air samples are collected in the same manner as general area samples and can be used to determine the need for bioassay analysis.

5.4 Tracking Air Sample Media

Tracking of air sampling media can be performed using several bar coding methods. These methods consist of applying bar codes to storage envelopes or attaching bar codes to the filter media itself. These methods have been found to be highly effective when combined with specialized PC software and portable bar code readers. By using software which operates on a microcomputer, technicians can eliminate the vast majority of manual data entry required for collecting air samples.

5.4.1 Bar Coded Envelopes

One of the easiest ways to improve air sample accountability is to directly apply bar coded labels to an air sample collection container such as an envelope. Portable bar code readers (loaded with special software) speed air sample collection. Bar code readers can be used to prompt technicians to scan an individual air sample location bar code and enter the ending vacuum flow rate. Technicians can then be prompted to remove the air sample filter, scan the corresponding envelope bar code and place the filter into the envelope. This operation links the air sample filter to the air sample location and the portable reader automatically records the time and date of sample collection. Technicians are prompted

by the bar code reader to place a new filter on the air sample station and adjust the beginning flow rate as instructed by the portable reader. Portable reader data is then downloaded to a microcomputer where it is merged with analysis data for reports and graphs.

5.4.2 Quick-Disconnects

Another method similar to bar coded envelopes involves quick-disconnect filter holders. Each filter holder is bar coded and equipped with a special fitting which allows the technician to quickly remove the entire air sample holder unit. This allows the air sample filter to be contained within the filter holder until it is brought back to the lab for analysis. This method works well for decommissioning projects and maintenance activities of short duration where the number of samples is small. By using portable bar code readers, bar coded air sample locations, and quick disconnect holders, technicians can improve sample accountability.

Portable readers prompt the technician to remove the air sample filter holder, scan the bar code and install the new holder and sample. As with the bar coded envelope method, the portable reader automatically records the time and date of sample collection and prompts the technician to adjust beginning flow rates for eventual flow rate averaging. Portable reader data is then downloaded to a microcomputer where it is merged with analysis data for reports and graphs.

5.4.3 Filter Cards

The previous two methods for tracking air sample filter media are very effective for tracking sample filters from the field to the lab for analysis. However, once in the lab, separating the sample media from it's identification in order to perform analysis is a weak link in the accountability chain. One way to ensure complete accountability throughout the sampling and analysis process is to bar code the sample itself. Placing a bar code label directly on the sample media will restrict the air flow through the filter. Therefore, bar coded filter cards are normally used to address the accountability issue. The filter media is held in place by special adhesives and uniquely numbered bar codes are printed on each card.

Portable readers prompt the technician to scan the air sample location bar code and enter the ending vacuum flow rate as previously discussed. Technicians are then prompted to remove the air sample filter card, scan the bar code located on the card and place the card in a protective envelope. This operation ensures that the correct filter card is linked to the air sample location.

Next, the portable reader prompts the technician to scan and install a new bar coded filter card. By doing so, the technician links the unique filter card with the sample location for complete

accountability throughout the sampling and analysis process. As with the bar coded envelopes and the quick disconnect holders, the portable reader automatically records the time and date of sample collection and prompts the technician to adjust beginning flow rates for flow rate averaging. Portable reader data is then downloaded to a microcomputer where it is merged with analysis data for reports and graphs. Bar code scanners mounted on analysis equipment can be used to read each sample as it is counted, so as to link the results with the sample identification.

5.4.4 Adhesive Cards

Another device similar to the filter card is a bar coded adhesive card designed for lapel, early warning, and other air sample applications where the special filter holders needed for the filter card cannot currently be installed. This device allows technicians to place filter media onto an exposed adhesive area located in the center of the card.

Portable readers prompt the technician to scan the air sample location bar code and enter the ending vacuum flow rate as previously discussed. Technicians are then prompted to remove the air sample media, place the media on the adhesive card, scan the bar code located on the card and place the card in a protective envelope. This operation ensures that the correct filter card is linked to the air sample location.

Next, the portable reader prompts the technician to scan and install a new air sample filter. As with the bar coded filter cards, the portable reader automatically records the time and date of sample collection and prompts the technician to adjust beginning flow rates for flow rate averaging. Portable reader data is then downloaded to a microcomputer where it is merged with analysis data for reports and graphs. As with the bar coded filter card, analysis equipment containing a bar code scanner can read each sample as it is counted, so as to link the results with the sample identification.

6.0 Case Studies

Many DOE and NRC licensed facilities have implemented one or more of the bar coded air sampling techniques discussed above. Improved data accuracy, along with rapid analysis results were the main reasons for implementing a bar code air sampling system. However, these facilities also received the added benefit of substantial cost savings and improved worker efficiency through implementation of bar coded air sampling programs. The following represents case studies of four such facilities.

6.1 DOE Uranium Materials Facility

A uranium materials facility implemented a bar coded air sample tracking system in 1991 to track approximately 50 general area air

samples utilizing a bar coded envelope method. The purpose of the system was to address the following areas:

1) Provide better accountability for air samples,
2) Reduce errors associated with air sample collection, analysis, and reporting, and
3) Improve the efficiency associated with general area air sampling.

The facility utilizes an automatic, low-background, alpha/beta counter for air sample analysis. Portable bar code readers with hand-held laser scanners are used for air sample data collection. The portable readers are carried in protective cases which can be easily decontaminated. The portable readers contain software that helps to automate procedures and permits the air sample data to be downloaded to a microcomputer which merges analysis data with the bar code reader data to produce radioactive airborne concentration reports. The analysis data is also exported to the site's minicomputer to supplement personnel exposure data.

The bar coded air sampling system has allowed the uranium materials facility to increase the number of air samplers in use, while at the same time improving the accountability associated with air sample data. The site's bar coded system promotes standardized collection procedures and allows reporting of consistent, legally defensible air sample results regardless of the technician performing the work.

Site personnel estimate that the bar coded air sample tracking system has significantly improved air sample accountability and reduced data entry errors by 75%. In addition, microcomputer software designed for bar coded air sampling has reduced man-hour requirements for air sampling by 40%. Man-power previously allocated for data entry is now being used to perform a greater amount of data analysis and corrective actions, thus lowering the potential risk to workers.

6.2 Plutonium Fuel Facility Decommissioning Project

A plutonium fuel facility decommissioning project implemented a bar coded air sampling system in 1990 to track approximately 30 general area air samples utilizing the quick-disconnect method. The purpose of the system is to address the following areas:

1) Provide better accountability for air samples,
2) Provide accurate data to ensure worker safety during decommissioning, and
3) Improve the efficiency associated with routine and non-routine air sampling.

The facility utilizes an automatic, low-background, alpha/beta counter for air sample analysis. Portable bar code readers with hand-held laser scanners are used for air sample data collection.

The portable readers contain software that helps to automate procedures and permits the air sample data to be downloaded to a microcomputer which merges analysis data with the bar code reader data to produce radioactive airborne concentration reports. The analysis data is also exported to the site mainframe computer to supplement personnel exposure data. To date, it is estimated that the bar coded air sample tracking system has maintained sample accountability and saved approximately 50% in man-hours when compared to manual data collection and reporting methods.

Bar coding has provided the decommissioning project with the capability to collect non-routine (as well as routine) air samples. For example, four or five samples may be non-routinely collected in a specific area to evaluate airborne radioactive conditions which may be elevated due to an increase in decommissioning activities. The data associated with the non-routine samples is also managed by the project's bar code tracking system. In addition, many glove box leaks were rapidly detected and repaired throughout the life of the 5-year project as a result of the speed associated with bar coded sampling systems.

6.3 Environmental Remediation Project

An environmental remediation project involving uranium contaminated sludge implemented a bar coded air sampling system in 1991 to track approximately 12 general area air samples utilizing the quick-disconnect method. The purpose of the system is to address the following areas:

1) Provide better accountability for air samples,
2) Provide accurate data to ensure worker safety during remediation, and
3) Improve the efficiency associated with routine and non-routine air sampling.

The facility utilizes an automatic, low-background, alpha/beta counter for air sample analysis and portable bar code readers with built-in laser scanners. The air sample collection and analysis methods are similar to those implemented for the decommissioning project described in Section 6.2 above. The analysis data is also exported to the site mainframe computer to supplement personnel exposure data. To date, it is estimated that the bar coded air sample tracking system has maintained sample accountability and saved approximately 25% in man-hours compared to manual data collection and reporting methods.

Bar coding has provided the remediation project with the capability to collect non-routine (as well as routine) air samples while maintaining data integrity. For example, three or four samples may be collected in a specific area to evaluate airborne radioactive conditions which may be elevated due to an increase caused by dewatering activities. The data associated with the non-routine samples is also managed by the bar code tracking system.

6.4 NRC Licensed Uranium Fuel Facility

An NRC licensed uranium fuel facility implemented a bar coded air sampling system in 1987 to track approximately 250 representative air samples utilizing the bar coded filter card method. The purpose of the system is to address the following areas:

1) Provide very high accountability of air samples,
2) Ensure accurate data to ensure worker safety during fuel production, and
3) Provide efficient air sample collection for large number of air samples.

Because 250 air samples are collected each shift (750 per day), maintaining a high degree of sample accountability is very important. It is estimated that the bar coded air sample tracking system provides a 95% confidence level of air sample accountability and data accuracy. In addition, the man-hours saved by utilizing the system is estimated to be about 75% less than manual methods of data collection. An automated time and attendance system (also implemented at the facility) allows each worker's time-in area to be correlated with area air sample results to estimate worker exposure.

Automatic, alpha counters equipped with bar code scanners are used for air sample analysis. Portable bar code readers with wands are used for data collection. Special software loaded into the portable readers helps to automate air sampling procedures and provides the technicians with an efficient method for air sample collection. The software also permits air sample data to be downloaded to a microcomputer which merges analysis data with the bar code reader data to produce radioactive airborne concentration reports. The analysis data is exported to the site mainframe computer to be correlated with time and attendance data for personnel exposure tracking and air sample analysis trending.

7.0 Conclusion

New regulations will drive the need for more accurate and efficient data collection. Air sampling is an area where improved data accuracy and collection efficiency can increase worker safety. In addition, air sample accountability will become more important as internal exposure monitoring requirements become more stringent.

Case studies indicate that bar coded systems can greatly enhance air sample accountability, and enhance data accuracy. In addition, they can also promote timely review of data, promote analysis trending, and improve collection efficiency in the workplace. Man-power savings achieved though automation can be redirected towards a closer review of air sample results to ensure compliance with expected reductions in airborne activity limits.

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DISCUSSION

FIRST: Where did you come up with a million times less chance for error with bar codes than with manual tracking? Is this something quantitative or is it somebody's imagination?

BAILEY: It was obtained from the bar code symbology book published by Intermec Corporation.

FIRST: The people who supply the bar codes?

BAILEY: The people who supply the bar codes.

FIRST: It is certainly an unbiased source. Let me ask you a serious question. The examples you showed were for radioactive counting. How would the bar code work if you were going to take the filter off and weigh it by an electrobalance, because, practically, you could not weigh the card with the filter, or if you wanted to subject the sample to a chemical analysis? How do you have assurance that the bar code stays with the sample?

BAILEY: One of the ways that you can do this is by setting up a bar code scanner that links to the analysis equipment, such as a chemical analytical device, and then scanning the bar code at the time that the sample is removed from its holder and placed in solution.

Experimental study of elementary collection efficiency of aerosols by spray : design of the experimental device

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Abstract

The safety of a nuclear power plant containment building, in which pressure and temperature could increase because of a overheating reactor accident, can be achieved by spraying water drops. The spray reduces the pressure and the temperature levels by condensation of steam on cold water drops. The more stringent thermodynamic conditions are a pressure of 5.10^5 Pa (due to steam emission) and a temperature of 413 K. Moreover its energy dissipation function, the spray leads to the washout of fission product particles emitted in the reactor building atmosphere. The present study includes a large program devoted to the evaluation of realistic washout rates. The aim of this work is to develop experiments in order to determine the collection efficiency of aerosols by a single drop. To do this, the experimental device has to be designed with fundamental criteria :

- Thermodynamic conditions have to be representative of post-accident atmosphere.
- Thermodynamic equilibrium has to be attained between the water drops and the gaseous phase.
- Thermophoretic, diffusiophoretic and mechanical effects have to be studied independently.
- Operating conditions have to be homogenous and constant during each experiment.

This paper presents the design of the experimental device. In practice, the consequences on the design of each of the criteria given previously and the necessity of being representative of the real conditions will be described.

NOMENCLATURE

Particle characteristics:

Cu	: Cunningham slip-flow correction	(-)
D_p	: particle diameter	(m)
Kn	: Knudsen number	(-)
k_p	: thermal conductivity	($W.m^{-1}.K^{-1}$)
Stk	: Stokes number	(-)
g	: gravity acceleration	($m.s^{-2}$)

Drop characteristics:

Cp_d	: specific heat	($J.kg^{-1}.K^{-1}$)
D_d	: drop diameter	(m)
Heq	: equilibrium height	(m)
L	: heat of vaporization	($J.kg^{-1}$)
m_d	: drop mass	(kg)
M_w	: water molar mass	(kg)
Re	: Reynolds number	(-)
T_d	: drop temperature	(K)
V_d	: settling drop velocity	($m.s^{-1}$)
ρ_d	: drop density	($kg.m^{-3}$)

Gas characteristics:

D_s	: steam diffusivity	($m^2.s^{-1}$)
k_F	: thermal conductivity	($W.m^{-1}.K^{-1}$)
M_a	: dry air molar mass	(kg)
M_S	: steam molar mass	(kg)
Pr	: Prandtl number	(-)
P	: total pressure	(Pa)
P_S	: partial steam pressure	(Pa)
P_{SV}	: saturation vapor pressure	(Pa)
R	: molar gas constant	($J.mol^{-1}.K^{-1}$)
S	: saturation rate	(-)
Sc	: Schmidt number	(-)
T_F	: gas temperature	(K)
X_a	: dry gas fraction in moist gas	(-)
X_s	: steam fraction in moist gas	(-)
μ_F	: gas viscosity	(Pa.s)
$\rho_{s,F}$	: density of steam in gas	($kg.m^{-3}$)
$\rho_{s,d}$	: density of steam in gas arround drop	($kg.m^{-3}$)
ρ_F	: gas density	($kg.m^{-3}$)

I. INTRODUCTION

The safety of a nuclear power plant containment building can be achieved by spraying water drops. Severe accident scenarios (rupture of steam pipe or coolant system loss) provide for pressure and temperature increase. The spray reduces the pressure and the temperature levels inside the containment building by condensation of steam on cold water drops. The most stringent thermodynamic conditions are a pressure of 5.10^5 Pa and a temperature of 413 K in present reactors. In the future, these conditions will be $7.7.10^5$ Pa and 433 K. Moreover its energy dissipation function, the spray leads to the washout of fission products emitted in the reactor building atmosphere. Safety studies show that the median diameter of aerosols in case of overheating accident would vary between 0.1 and 5 μm. At present time, washout rates provide conservative assumptions in safety studies. This study includes a large program devoted to :

. determine realistic washout rates for present reactor
. optimize washout function of spray in futur reactor

The first step, which is the aim of this work, is to develop experiments in order to determine the collection efficiency of aerosols by falling drops in severe accident representative conditions.

II. WASHOUT MECHANISMS

Aerosols fission product washout results of mechanical and phoretic effects.

Mechanical effects: differential settling velocity between drops and aerosols induces collisions. There are three mechanical effects and their relative importances are function of drop and aerosol diameters.

▸ *impaction* : falling drop velocity induces fluid flow variations. High inertia particles turn off flow lines and then are collected on drops. This mechanism increases with drop velocity and particle diameter.

▸ *interception* : this mechanism is just a geometric effect and then independent of falling drop velocity. This mechanism increases with $\frac{D_p}{D_d}$.

▸ *wake-capture* : small particles collection efficiency increases for drops which Reynolds numbers are between 30 and 300. Small particles, without significant inertia, trace the outline of drop and are collected in drop wake by eddy[1,2,3]. For Reynolds numbers larger than 300, collection efficiency decreases swiftly. So there is an optimum for Re $\simeq$ 300.

Phoretic effects: temperature and steam concentration gradients arround drops induce aerosol collection by thermophoresis and diffusiophoresis.

▸ *Diffusiophoresis*: steam condensation on cold water drops induces molecular steam flow rate towards drops. This flow drags along particles. PRUPPACHER and KLETT[4] propose a theoretical collection efficiency :

$$E = \frac{C_D \; 2 \; M_a \; D_s \; (1 + \sigma_{s/a} \; . \; X_a)}{D_d \; V_d \; \rho_F \; M_w} \quad (\rho_{s,F} - \rho_{s,d})$$

with :

$$C_D = 1 + 0.108 \; (Sc^{1/3} \; Re^{1/2})^2 \quad \text{for} \quad Sc^{1/3} \; Re^{1/2} < 1.4$$

$$C_D = 0.78 + 0.308 \; Sc^{1/3} \; Re^{1/2} \quad \text{for} \quad Sc^{1/3} \; Re^{1/2} \geq 1.4$$

$$\text{and :} \qquad \sigma_{s/a} = \frac{M_w^{1/2} - M_a^{1/2}}{X_s M_w^{1/2} - X_a M_a^{1/2}}$$

Collection efficiency is proportional to local concentration gradient and independent of drop size. Diffusiophoresis effect is very important for small particles ($D_p \leq 2 \; \mu m$) for which impaction is not significant and particularly for small drops ($D_d \leq 180 \; \mu m$) because wake-capture doesn't exist.

▸ *Thermophoresis*: local temperature gradient around drops induces thermophoresis force. PRUPPACHER and KLETT[4] propose a theoretical collection efficiency :

$$E = \frac{C_T \; 2 \; C \; k_F}{D_d \; V_d \; P} \quad (T_F - T_d)$$

$$\text{with} \qquad C = \frac{0.4 \; Cu \; (k_F + 2.5 \; k_p \; Kn)}{(1 + 3 \; Kn) \; (k_p + 2 \; k_F + 5 \; k_p \; Kn)}$$

and

$$C_T = 1 + 0.108\ (Pr^{1/3}\ Re^{1/2})^2 \quad \text{for} \quad Pr^{1/3}\ Re^{1/2} < 1.4$$

$$C_T = 0.78 + 0.308\ Pr^{1/3}\ Re^{1/2} \quad \text{for} \quad Pr^{1/3}\ Re^{1/2} \geq 1.4$$

Thermophoresis collection efficiency is proportional to temperature gradient and in inverse ratio to pressure. This efficiency decreases swiftly for larger drops.

All washout mechanisms and their possible coupling can be represented, for each drop, by an elementary collection coefficient. This coefficient is function of drops, aerosols and gaseous characteristics.We can see on figure 1 from PRUPPACHER and KLETT[5], the evolution of this coefficient versus drop radius for several relative humidities. These theoritical and experimental results show a great variation of the collection coefficient (factor 100).

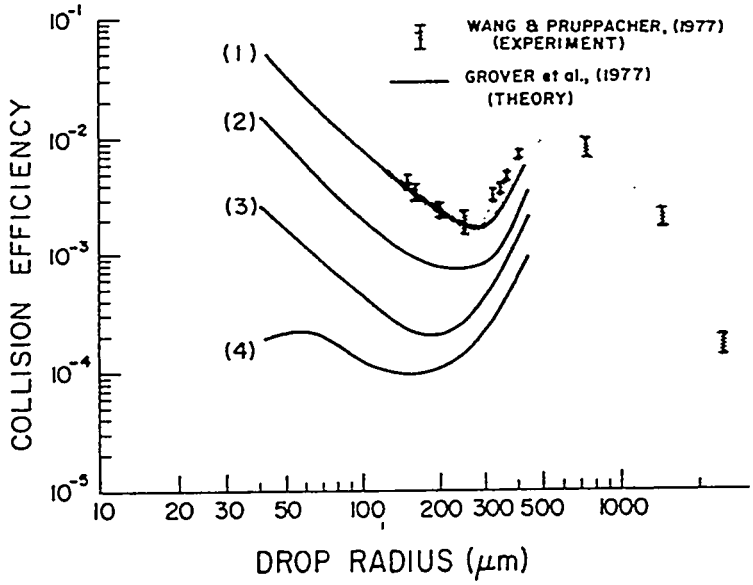


Figure 1 : Collision efficiency of particles (D_p = 0.5 μm) as function of collection drop size for relative humidities of (1) 20 %, (2) 75 %, (3) 95 % and (4) 100 %

III. DESIGN CRITERIA

There are theories and experiments but without specific thermodynamic conditions of spray system. So the experimental device is designed for a study of elementary collection coefficient in spray working conditions. The aim of this work is to develop analytical experiments to determine elementary collection coefficient of aerosols by a single drop. These results will be used to compute realistic washout rate of spray system.

To do this, the experimental device have to be designed with fundamental criteria :

▸ **Representative conditions:** in accident scenarios, with using spray system, thermodynamic conditions of reactor building atmosphere could change in a wide range. Moreover, prior results (figure 1) show a very large range of collection coefficient in function of different thermodynamic

conditions, aerosol and drop diameters. So experimental device would be able to reproduce the most of accident scenarios conditions.

▸ **Thermodynamic drop-gas equilibrium:** in order to determine precisely phoretic effects on collection, falling drop height must be enough for drops reach their thermodynamic drop-gas equilibrium. In fact, real equilibrium height is infinite. Moreover, phoretic effect is function of temperature gradient or steam concentration gradient, so we consider that equilibrium is reached when these gradients are insignificant. This equilibrium height is also very dependent of initial drop diameter. So we have determined experimentaly drop size distribution of an industrial spray nozzle (SPRACO 1713A). Results (figure 2) show that drop diameters are in range from 20 μm to 600 μm, with a numeric median diameter of roughly 200 μm.

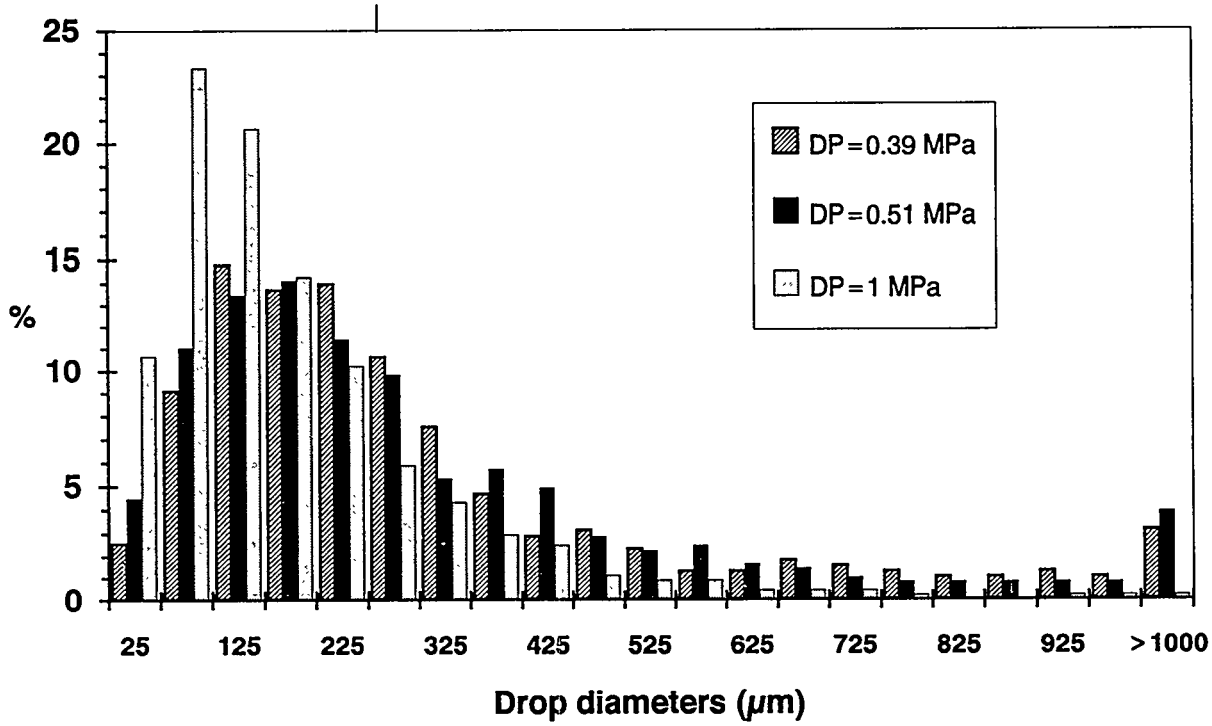


Figure 2 : Drop size distribution for different water pressures (DP)

▸ **Separate each washout mechanisms:** the study objective is to determine elementary collection efficiency of each effects independently. Then, experimental device must allow a great aptitude for different working conditions.

▸ **Steady-state and homogeneous working conditions:** the operating parameters are different from one experiment to an other, so as to represent all the spraying process conditions. But, in order to determine collection efficiency, we measure aerosol mass collected by drops, so during each experiment, these working conditions (pressure, temperature, saturation rate and aerosol concentration) have to be homogeneous and constant.

IV. DESIGN CRITERIA CONSEQUENCES

Experimental device is designed with these criteria:

▸ **Representative conditions:** from accident scenarios and calculation code, working conditions, which would be reproduce in experiments, are :

pressure	: from 10^5 to 5.10^5 Pa
temperature	: from 293 to 413 K
steam saturation rate	: from 0 to 1
aerosol diameters	: from 0.1 to 5 μm
drop diameters	: from 100 to 500 μm

Futur reactor conditions will be study, so experimental device is also designed for a pressure of $7,7.10^5$ Pa and a temperature of 433 K.

In order to separate each washout mechanism and to modelize, each working condition must be adjust indepentently.

▸ **Thermodynamic drop-gas equilibrium:** steam mass transfert flow is proportional to local steam concentration gradient. We consider that equilibrium is reached when drop temperature is equal to fluid saturation temperature, taking into account the second decimal. For large drops (> 100 μm) curvature effect of the drop surface on equilibrium vapor pressure is insignificant. So, kinetic condensation equations are :

drop diameter variation (dD_d) during Δt :

$$dD_d = \frac{4.\ D_s.\ M_s}{D_d\ .\ \rho_d\ .\ R}\left[\frac{P_s}{T_F} - \frac{P_{so}}{T_d}\right]\ \Delta t$$

$$\text{with : } P_s = P_{sv}(T_F)\ .\ S \quad \text{and} \quad P_{so} = P_{sv}(T_d)$$

drop mass variation is (dm_d) then :

$$dm_d = \frac{\pi}{6}\ \rho_d\ \left((D_d + dD_d)^3 - D_d^3\right)$$

and thermal balance is :

$$m_d\ .\ Cp_d\ .\ T_d + dm_d\ .\ L + 2\pi\ D_d\ k_F\ (T_F - T_d)\ \Delta t = (m_d + dm_d)\ Cp_d\ (T_d + dT_d)$$

Using numerical integration, we determine equilibrium height as function of initial drop diameter for more stringent thermodynamic conditions (figure 3).

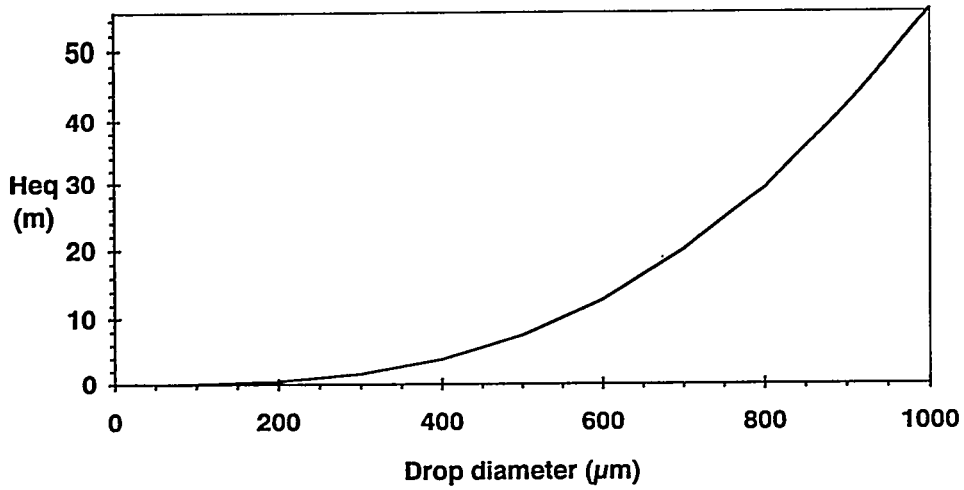


Figure 3: Equilibrium height as function of initial drop diameter
$P = 5.10^5$ Pa, $T_F = 413$K, $S = 1$, $T_d = 293$ K

Equilibrium height increases rapidly with initial drop diameter. But, in fact, phoretic effects are more important in the first falling meters. We can use the condensed water mass as equilibrium indicator. Equilibrium rate (EQUI), in function of drop position (y), is defined as :

$$EQUI\ (y) = \frac{D_d^3\ (y) - D_d^3\ (0)}{D_d^3\ (Heq) - D_d^3\ (0)}$$

(1-EQUI) represents the fraction of water which has not condensed after y meters of fall (figure 4).

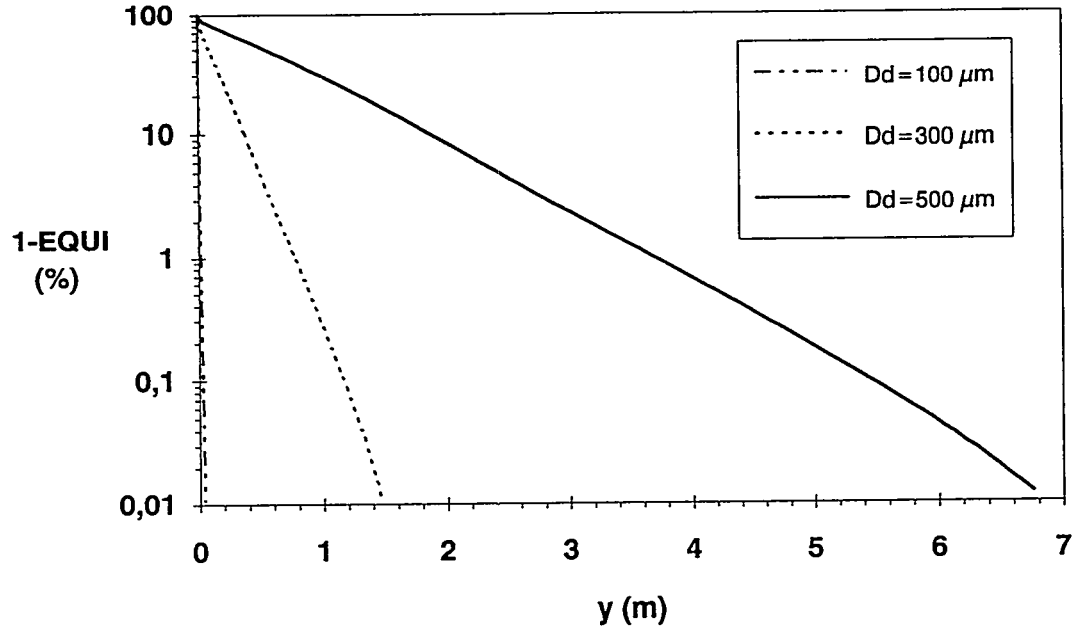


Figure 4: no condensed mass as function of falling height
$P = 5.10^5$ Pa, $T_F = 413$ K, $S = 1$, $T_d = 293$ K

For drop diameters ranging from 100 μm to 500 μm, a falling height of five meters is enough to condense more than 99 % of the water mass. So, we fixe our experimental device height at five meters.

▸ **Separate each washout mechanisms:** we have seen that with our experimental device each working conditions could be adjust independently and that phoretic effect can be study with a height of five meters. For mechanical effect determination, WANG and PRUPPACHER[4]advise that drop velocity is about 99 % of terminal settling velocity.

Basic dynamic equations of drop fall are :

$$m_d \frac{dV}{dt} = \frac{\pi}{6} D_d^3 (\rho_d - \rho_F)\, g - C_t\, \rho_F \frac{\pi}{8} D_d^2\, V^2$$

Drag coefficient (C_t) is function of drop Reynolds number:

$$Re_d = \frac{\rho_F \cdot V \cdot D_d}{\mu_F}$$

$$\text{for} \quad Re_d < 1 \Rightarrow C_t = \frac{24}{Re_d}$$

$$\text{for} \quad 1 < Re_d < 905 \Rightarrow C_t = \frac{24}{Re_d}\left[1 + \frac{Re_d^{2/3}}{6}\right]$$

$$\text{for} \quad Re_d > 905 \Rightarrow C_t = 0{,}44$$

terminal settling velocity (V_l) is obtained for $\frac{dV}{dt} = 0$

Drop velocity is computed by numeric integration :

$$V(t + \Delta t) = V(t) + \frac{dV}{dt}\Delta t$$

We can determine the drop fall height required for reach 99% of terminal velocity in function of drop diameter. For drop diameter of 500 μm, these results (figure 5) show that without steam the height required is about 0.5 meter and with steam, this height is about two meters. In order to determine mechanical collection coefficient, a drop acceleration area without aerosol is needed. The height of this area is fixed at two meters.

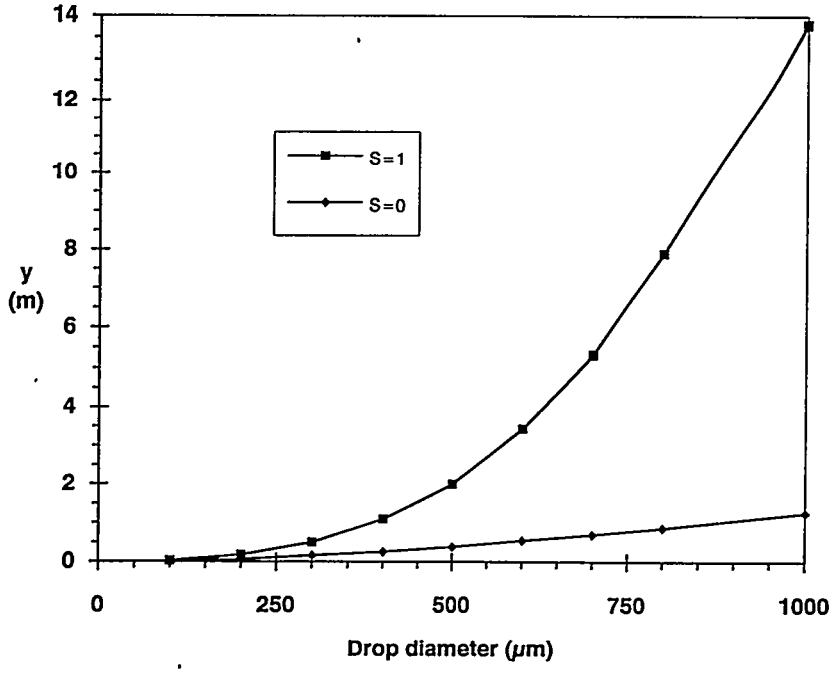


Figure 5: Falling height for reach 99% of V_l as function of drop diameter
$P = 5.10^5$ Pa, $T_F = 413$ K, $T_d = 293$ K

▸ **Steady-state and homogeneous working conditions:** homogeneity of working conditions (temperature and aerosol concentration) has been studied by numerical simulations of natural convection motion. The turbulence is modelized by using a k-ϵ model in a code (TRIO-EF) developed in CEA. The simulations show that an efficient mixing can be achieved after 20 minutes by applying a temperature difference of only a few degrees between the walls and the bottom of the experimental enclosure (figure 6).

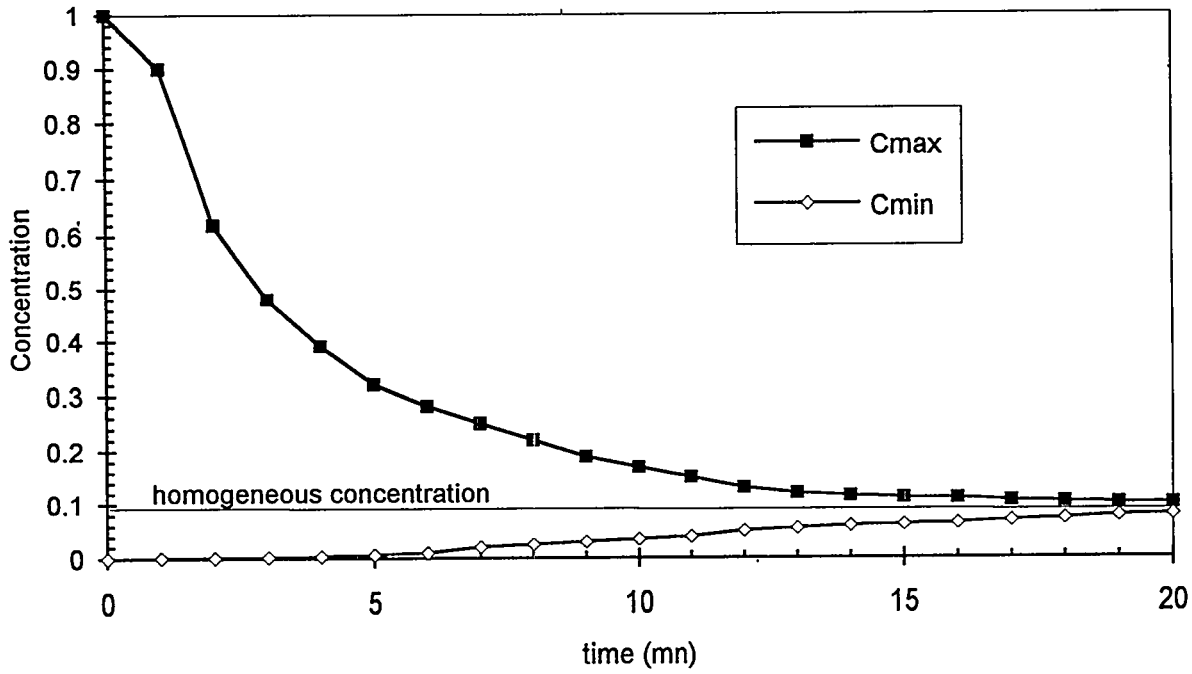


Figure 6: Evolution of maximal and minimal concentrations as a function of time

Several simulations, with a range of Grashof numbers between 10^8 and 5.10^{11} and a range of Prandtl numbers between 0.9 and 1.1, have been done in order to represent all possible thermodynamic working conditions.

Steady-state working conditions during each experiment is an important characteristic of this study. Using kinetic condensation equations, it is possible to comput steam volume removal by a drop. The most unsteady-state working conditions appear with an initial drop diameter of 500 μm. In this case, each drop removes $2.10^{-8}m^3$ of steam. So, if we accept a pressure variation of 1 %, the number of drops injected in the device must be less than 10^6 drops. By using the greatest washout coefficient which can be expected (about 10^{-1}), the injection of 10^6 drops during an experiment induces a variation of aerosol concentration less than 7 %. Then we can consider a constant concentration.

V. EXPERIMENTAL DEVICE DESCRIPTION

Using all design criteria described above, we have realized an experimental device.

The main characteristic is that these experiments will be analytic in order to modelize each collection mechanism in steady-state conditions. This concept induces short-time experiments and few drops injected.

The experimental device (figure 7) working conditions are :

- geometry: cylindrical
- drop falling height: 5 m
- enclosure diameter: 0.6 m
- temperature: from 293 K to 433 K
- total pressure: from 10^5 to 8.10^5 Pa
- steam saturation rate: from 0 to 1
- monodispersed drops diameter: from 100 to 500 μm
- aerosols aerodynamic diameter: from 0.1 to 5 μm
- aerosols' nature: CsI

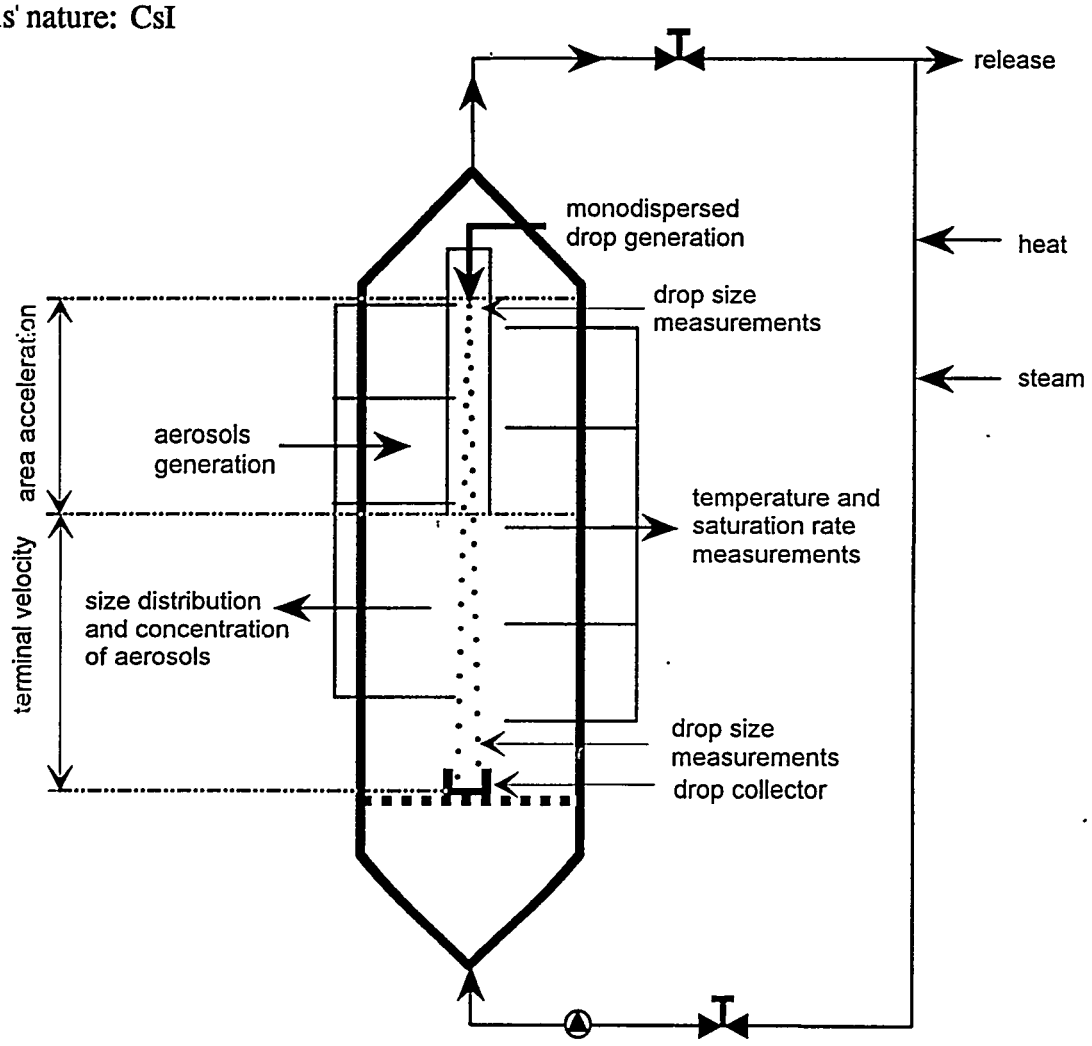


Figure 7: Scheme of experimental device

VI. EXPERIMENTAL SET-UP

In order to determine the collection coefficient, each experiment, for fixed working conditions is in fact a succession of four tests. Experimental set up (figure 8) is based on the assumption that the different mechanisms are additive and independent.

Test 1: all washout mechanisms are taken into account. These test working conditions are representative of the experiment. Aerosol mass collected during this test is noted M_1.

Test 2: only mechanical collection occurs during acceleration falling velocity and terminal velocity. Although thermodynamic conditions are different from test 1, mechanical effect parameters (viscosity and density) are comparable. Aerosol mass collected during this test is noted M_2.

Test 3: using acceleration area of drops in which there is no aerosol, we can determine mechanical collection with a constant velocity. Aerosol mass collected during this test is noted M_3.

Test 4: thermophoretic effect importance will be determine with this test. Steam saturation rate will be adjusted in order to have no evaporation. In fact, this collection effect will certainly be insignificant. Aerosol mass collected during this test is noted M_4.

By using aerosol mass collected by drops in each test, we can determine each collection effect :

diffusiophoretic effect : $M_1 - M_4$
mechanical effect : M_3
thermophoretic effect : $M_4 - M_2$

TEST NUMBER	1	2	3	4
washout mechanisms	mechanical diffusiophoresis thermophoresis	mechanical	mechanical with terminal settling velocity	mechanical thermophoresis
acceleration area	no	no	yes	no
T_F (K)	413	293	293	413
P (Pa)	5.10^5	$2.5.10^5$	$2.5.10^5$	$3.5.10^5$
S (-)	0.95	≈ 0	≈ 0	≈ 0
T_d (K)	293	293	293	293
D_d (µm)	100 to 500 µm	100 to 500 µm	100 to 500 µm	100 to 500 µm
ρ_F (kg.m^{-3})	3	3	3	3
μ_F (Pa.s)	$1.45.10^{-5}$	$1.8.10^{-5}$	$1.8.10^{-5}$	$2.25.10^{-5}$

Figure 8 : experimental set-up

CONCLUSIONS

This study, about design of the experimental device, shows the feasibility to determine experimentaly collection coefficients of aerosols by falling drops. This experimental device, which is going to be realized, has two great specific characteristics:

- the most important is the concept of analytical experiments: we will measure elementary collection efficiencies of each collection mechanism,
- the second is that these experiments will be done with working conditions representative of nuclear power plant severe accident scenarios.

Using these experimental results, more accuracy washout rates will be computed in safety studies.

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D. Reider Publishing C

DISCUSSION

FIRST: I had a question regarding the relevance of a 0.6 m tower in relation to a full-scale nuclear containment vessel. Have you taken scale into account? It is obvious that wall effects in a vessel of the diameter you used will have an important bearing on the final results. I am raising a question regarding the relevance of data in a 2 ft diameter cylinder to effects in a full-scale nuclear power plant containment vessel that will be perhaps 75-100 m in diameter. It would seem to me that a small vessel has important wall effects.

DUCRET: In this experiment we will analyze only the effects of drops without wall effects. The objective of this study is to measure only the collection by drops.

SHER: I have a related question. Is the experimental vessel completely filled with drops and completely filled with aerosol? Is there good mixing of the aerosol and droplet streams?

DUCRET: There is aerosol in the entire enclosure and a few drops are injected in the middle of the enclosure. Aerosol mixing will be achieved by natural motion and we will measure aerosol concentration in four samples. Because few drops will be injected, the aerosol and drop mixing will be good.

SHER: Do you have to worry about mixing of the aerosols with the drops?

DUCRET: It is a big problem but it is not the objective of the study. In our study, very few drops are injected so we assume a good mixing.

Degradation of HEPA Filters Exposed to DMSO*

by

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Abstract

Dimethyl sulfoxide (DMSO) sprays are being used to remove the high explosive (HE) from nuclear weapons in the process of their dismantlement. A boxed 50 cfm HEPA filter with an integral prefilter was exposed to DMSO vapor and aerosols that were generated by a spray nozzle to simulate conditions expected in the HE dissolution operation. After 198 hours of operation, the pressure drop of the filter had increased from 1.15 inches to 2.85 inches, and the efficiency for 0.3 μm dioctyl sebacate (DOS) aerosols decreased from 99.992% to 98.6%. Most of the DMSO aerosols had collected as a liquid pool inside the boxed HEPA. The liquidwas blown out of the filter exit with 100 cfm air flow at the end of the test. Since the filter still met the minimum allowed efficiency of 99.97% after 166 hours of exposure, we recommend replacing the filter every 160 hours of operation or sooner if the pressure drop increases by 50%. Examination of the filter showed that visible cracks appeared at the joints of the wooden frame and a portion of the sealant had pulled away from the frame.

Since all of the DMSO will be trapped in the first HEPA filter, the second HEPA filter should not suffer from DMSO degradation. Thus the combined efficiency for the first filter (98.6%) and the second filter (99.97%) is 99.99996% for 0.3μm particles. If the first filter is replaced prior to its degradation, each of the filters will have 99.97% efficiency, and the combined efficiency will be 99.999991%. The collection efficiency for DMSO/HE aerosols will be much higher because the particle size is much greater.

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I. Introduction

This study was initiated in support of the nuclear weapons dismantlement program in which the HE, HMX, is being removed from the weapon by dissolution with DMSO. Heated DMSO is sprayed against the exposed HE to dissolve it and aid in breaking up the HE layer. The DMSO spray generates a high concentration of fine aerosols that is contained in a workstation. This aerosol may contain up to 25% by weight dissolved HE. The exhaust from the workstation is filtered through two, 135 cfm HEPA filters to prevent potential radioactive contamination or HE from being released to the environment.

A schematic of the ventilation system used for the HE dissolution workstation is shown in Figure 1. The ventilation system consists of a 35 cfm inlet HEPA filter with a butterfly valve to seal off the inlet during the DMSO spraying. The exhaust consists of two,135 cfm HEPA filters in series followed by a Fischer valve (Associated Process Controls, Pleasanton,CA, 94566), and a flow meter. Although a standard 50 cfm HEPA filter would be sufficient to meet the required exhaust flow capacity of 55 cfm for safety considerations, we selected the 135 cfm HEPA filter to reduce the exhaust flow restrictions. The Fischer valve opens when the vacuum in the 16 cubic feet workstation drops below -1 inch of water and closes when the vacuum increases greater than -1 inch of water. During the dissolution process, there is no ventilation through the workstation except for short intervals when the Fischer valve opens to maintain a -1 inch of vacuum inside the workstation. After the dissolution operation is completed, the inlet butterfly valve is opened, and filtered room air sweeps the workstation clean and exhausts through the exhaust HEPA filter.

Since DMSO is an excellent solvent for many polymers, we were concerned that the solvent would dissolve the components of the HEPA filter and degrade the filter performance. The primary components of the HEPA filter that we suspected as being susceptible to attack from DMSO are the acrylic binder that holds the glass fibers together in the media and the polyurethane foam sealant that seals the pleated filter pack into the frame. The filter type that we tested was a combined, prefilter-HEPA filter unit boxed in a single plywood housing: Flanders, Dual Pack, 50 cfm, model 007-C-04-00-NL. The prefilter medium was a glass fiber medium with an atmospheric dust spot efficiency of 60% according to ASHRAE Standard 52-76. The HEPA medium met the requirements in MIL-F-51079.

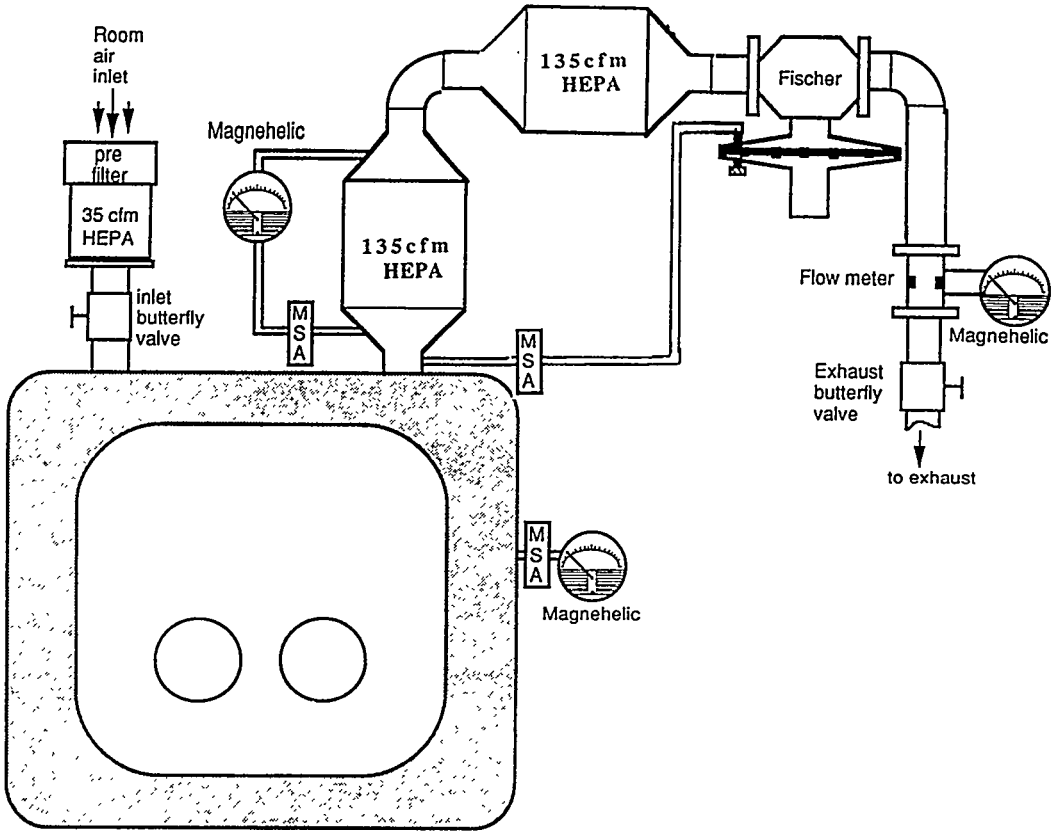


Figure 1. Ventilation system for the HE dissolution workstation.

II. Characterizing DMSO challenge to HEPA filters

In order to evaluate the effect of the DMSO exposure on HEPA filters, we first had to characterize the challenge to the filters. Although the challenge consisted of both DMSO vapor and aerosols, we only measured the aerosol component. Since the spray operation generates a high concentration of aerosols, we could assume that the vapor concentration would be the saturated concentration, which is easily computed for each temperature. Table 1 shows the saturated vapor pressure and concentration of DMSO at various temperatures. Two of the data sets are experimental measurements, while the remainder were computed using the Clapeyron-Clausius equation

Table 1 Saturated DMSO vapor pressure and concentration

Temperature		Vapor Pressure	Concentration	
°F	°K	Torr	%	g/m^3
72*	293	0.40*	0.05	1.7
100	311	1.39	0.18	5.3
122*	323	3.00*	0.39	11.7
150	339	7.43	0.98	27.6
189	360	22.8	3.00	79.5

*experimental data

We measured the concentration and size distribution of the DMSO aerosols produced under simulated operating conditions using the apparatus shown in Figure 2.

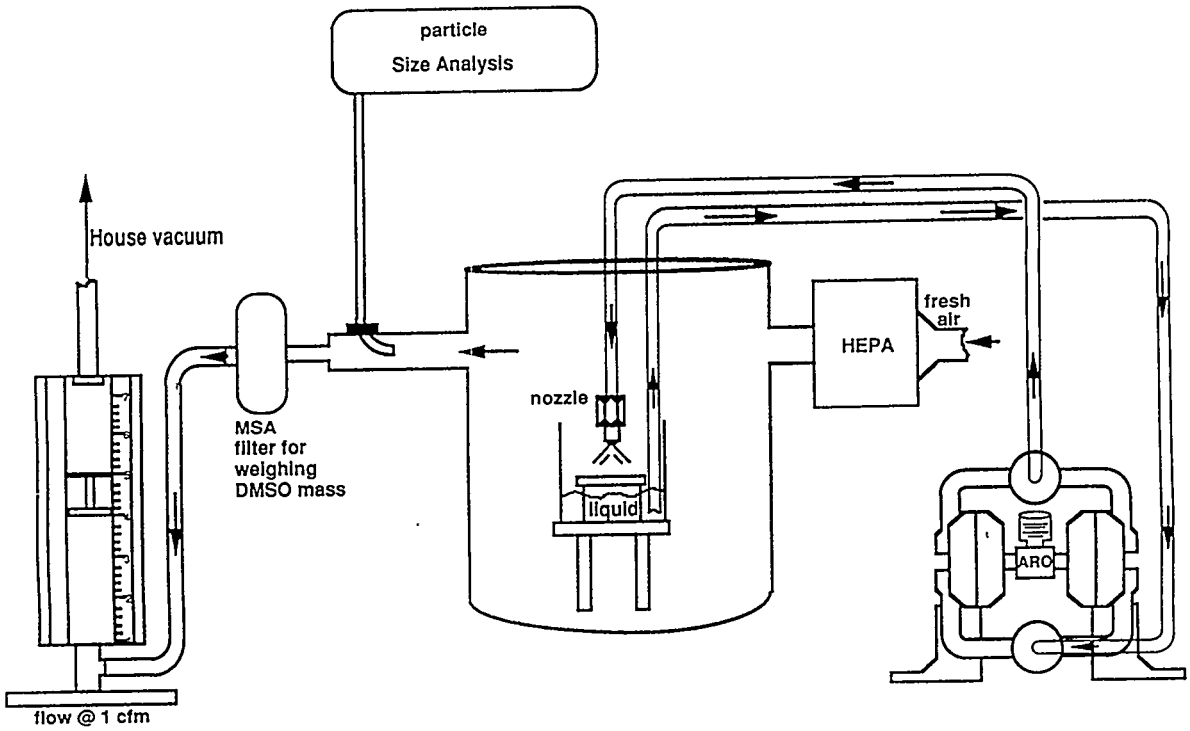


Figure 2. Experimental apparatus used for measuring the mass concentration and size distribution of DMSO aerosols generated from a single nozzle.

We simulated the work station and spray operation with a sealed Plexiglass chamber (0.87 cu. ft.) in which a spray nozzle was positioned 1.5 inches from a flat plate. In practice, multiple nozzles would be directed against the exposed HE. A piston pump (ARO) fed liquid DMSO at 100 psi to the spray nozzle and pulled liquid from a can that served both as a collector for the liquid run-off and as a reservoir. The temperature in the test chamber was at room temperature, about 70°F. The Plexiglass chamber also had a HEPA filtered inlet and a vacuum exhaust to simulate the ventilation system. We used a MSA respirator cartridge filter to filter the DMSO aerosols from the exhaust. The mass concentration of DMSO aerosols was determined by weighing the filter before and after the test and dividing by the air flow volume. The average concentration for four measurements is 3.3 g/m^3 with a standard deviation of ± 1.1 g/m^3. We also measured the aerosol size distribution with a Climet 208 particle counter. Figure 3 shows the size distribution of DMSO aerosols plotted as a function of particle diameter. The peak number concentration occurs at 1.6 μm diameter.

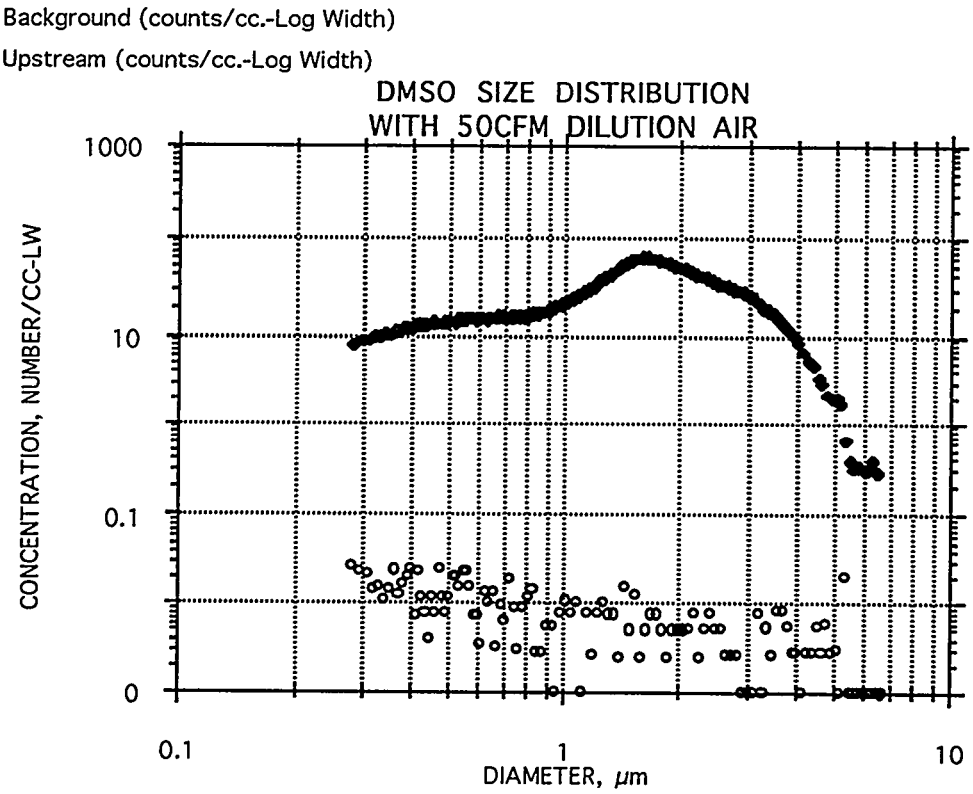


Figure 3. Size distribution of DMSO aerosols.

To verify that the DMSO concentration used in our filter experiments was realistic, we measured the mass concentration of DMSO/HE in a full-scale dissolution test. The tests were conducted on a simulated warhead that was sprayed with a solution of DMSO/HE. The temperature within the workstation was 120°F. Thirty-two nozzles (12 jets and 20 fans) were used in a manifold spray fixture. The aerosol mass was determined at twelve positions within a workstation that was 46 inches long, 34 inches wide, and 32 inches high.

Figure 4 shows a schematic of the workstation and sampling train used in the measurements along with the location of the twelve sampling locations. Each of the twelve sampling locations was six inches from the nearest wall. For example the top corner samples were taken six inches from the top wall and six inches from each side wall. The side samples were taken in the middle of the side wall, six inches from the side wall, and six inches from either the top or bottom walls. The aerosols were sampled at 1 cfm (0.0283 m^3/min) for three minutes through eight feet of 0.625 inch ID nylon tubing prior to collection on a 2 inch filter disk that was cut from a standard HEPA media. The aerosol mass was determined by weighing the filter disks before and after the sample. No tests were conducted to establish the particle line loss since potential losses would be negligible compared to the variation in concentration found in the box. The concentration measurements were computed from the DMSO/HE mass collected divided by 0.0849 m^3 (0.0283 m^3/min x 3 min.= 0.0849 m^3) and are tabulated in Table 2.

The average DMSO/HE concentration from Table 2 is 1.87 g/m^3 with a standard deviation of $\pm$0.57 g/m3. This concentration is less than what was measured with a single nozzle in the apparatus shown in Figure 2. We suspect that the larger particles were quickly settling out in the workstation yielding a smaller average particle size and hence a lower mass concentration. The larger aerosols in the single nozzle apparatus could not settle out because all of the air was being swept out and filtered for mass determinations. In contrast there was no air flow in the workstation during the spraying operation to prevent the larger particles from settling out. We did not measure the aerosol size distribution in the workstation to confirm our hypothesis because our instruments would be contaminated with HE and could not be cleaned. However, during the purge cycle the large particles will also be swept out into the filtration system. Thus, the single nozzle measurements provide a more conservative measurement of aerosol mass concentration.

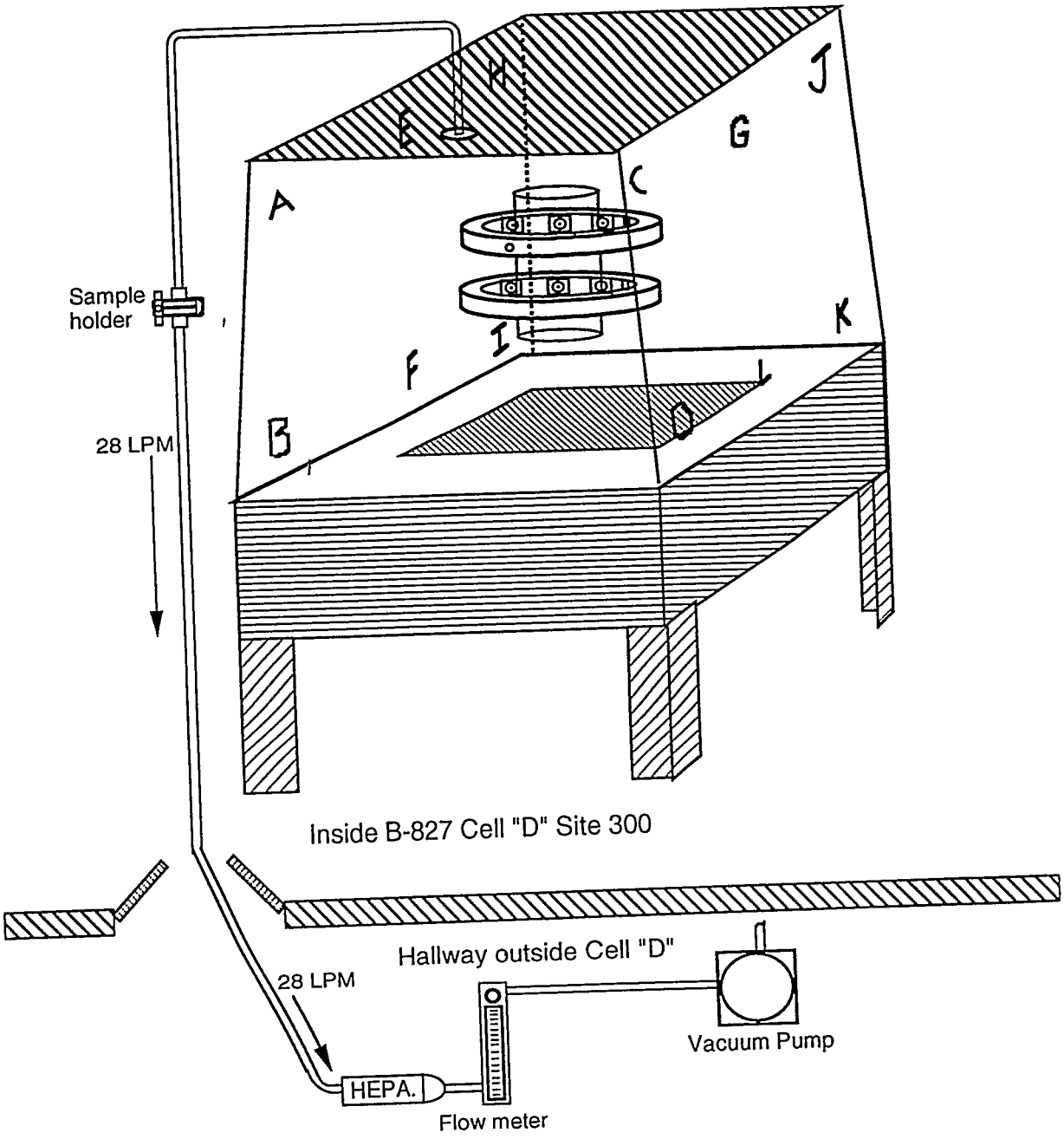


Figure 4. Schematic of the sampling train and the location of the sample probes in the DMSO/HE dissolution workstation

In addition to the DMSO aerosol challenge, the HEPA filters would also be exposed to a vapor challenge. As seen in Table 1, the vapor concentration is comparable to the aerosol challenge. At higher temperatures, the vapor concentration is much higher than the aerosol concentration. However, most of the vapor DMSO will pass through the HEPA filter whereas more than 99.97% of the DMSO aerosols will be trapped on the first HEPA filter. In practice, since the DMSO/HE particles are greater than 1 μm diameter, more than 99.9999% of the DMSO/HE particles will be trapped on the first filter. Although we have not measured the size distribution of DMSO/HE aerosols, the dissolved HE should not significantly affect the size distribution. Thus based on the single nozzle and full-scale tests, we estimate that about 3 g/m^3 of aerosol and 1.7 to 35 g/m^3 of vapor will be generated in the DMSO workstation depending on the temperature between 72 F and 160 F. This aerosol and vapor will challenge the exhaust HEPA filter. At the higher temperature, a significant fraction of the DMSO vapor will penetrate the first filter and condense on the second filter.

Table 2 Concentration of DMSO aerosols at various locations in a HE dissolution workstation.

Location	Concentration g/m^3
Top Corner	
A	1.79
C	2.89
H	1.24
J	2.04
Bottom Corner	
B	1.62
D	2.76
I	1.31
K	1.58
Top Side	
E	1.30
G	2.48
Bottom Side	
F	2.16
L	1.41

III. Filter Media Tests

We exposed samples of HEPA media cut from a new HEPA filter (MSA 50 cfm, part 464891) to saturated DMSO vapor and then conducted tensile strength tests on the exposed media. The precut media samples were suspended over a pool of DMSO contained in a sealed container. Since DMSO readily absorbs water, the samples were preconditioned at 50 C in a vacuum oven for 65 hours to remove any water absorbed on the media. We also conducted baseline tests on samples with and without preconditioning. The tensile strengths were measured using an Instron tester in both the machine direction (MD) and in the cross direction (CD). The results of our tests are shown in Table 3. The number of samples tested at each condition are shown in the table and provide a measure of the standard deviation.

Table 3 Media tensile strength after DMSO vapor exposure

Exposure	Pretest Conditioning	Maximum tensile load at failure (lbs)	
		Machine direction	Cross direction
Baseline (ambient air)	As-is	1.03±0.08	0.74±0.06
"	Dried under vac @ 50°C for 65 hrs	1.48±0.01	0.99±0.09
"	Dried under vac @ 50°C for 65 hrs	1.30±0.11	0.92±0.20
Sat'd DMSO vapor @ RT for 483 hours	Dried under vac @ 50°C for 43 hrs	1.32±0.04	0.99
Sat'd DMSO vapor @ 50°C for 343 hrs + 149 hrs @ RT	Dried under vac @ 50°C for 43 hrs	1.38±0.05	1.04

Note: The "±" values represent a single standard deviation.

The test results show that exposure to saturated DMSO vapor has no effect on the HEPA media strength. Table 3 also shows that the preconditioning increases the media strength by about 30%. We suspect that the pretreatment provided additional curing of the acrylic binder used to hold the fibers together in the media.

IV Filter Tests

We conducted a series of DMSO exposure tests on a boxed 50 cfm HEPA filter with an integral prefilter. This is the typical first stage filter used in the exhaust from glove boxes at LLNL. A photograph of the HEPA filter with flanges removed after the exposure test is shown in Figure 5.

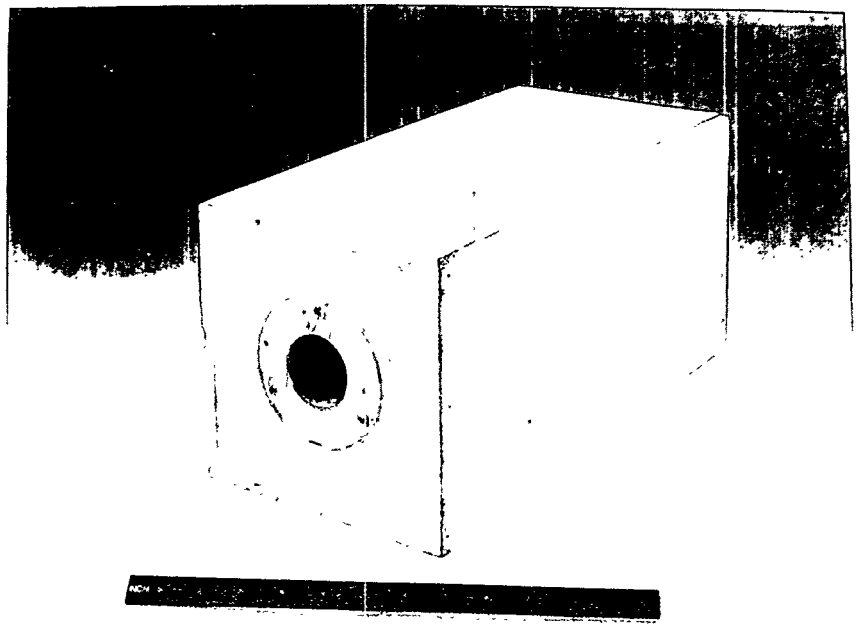


Figure 5. Boxed HEPA filter that was used in our simulation exposure test

The exposure test was designed to simulate conditions in the HE dissolution operation which consists of spraying heated DMSO for six hours to dissolve the HE. The inlet valve to the workstation is closed during the spraying operation to contain the DMSO/HE solution. The Fischer valve maintains a -1 inch pressure within the workstation during the dissolution by intermittently opening for short periods when the box vacuum drops below -1 inch due to thermal expansion of the air, leaks in the workstation, and evaporation of the solvent. We estimate that an exhaust flow of about 1 cfm is pulled through the HEPA filter during the six hour dissolution process. After the dissolution is complete, the DMSO spray is turned off and the inlet valve to the workstation is opened and the workstation is swept clean with room air for about 30 minutes.

We set up two different flow systems to simulate the HE dissolution operation. Figure 6 shows the experimental apparatus used to simulate conditions during the dissolution spraying operation. An air stream containing DMSO aerosols is heated and then passed through the HEPA

filter. Heating tape wrapped around the inlet tube can raise the temperature from room temperature up to 150 F at the inlet to the HEPA filter. Two different size Laskin nozzle generators were used to generate the DMSO aerosols: a single nozzle generator and a six nozzle generator. The air flow through the small and large generators was 1.5 and 6.5 cfm respectively. All of the air passing through the filter came from the aerosol generator. The concentration of DMSO aerosols for the small and large generators was computed from filter mass measurements to be 0.083 g/cu. ft. and 0.022 g/cu. ft. respectively.

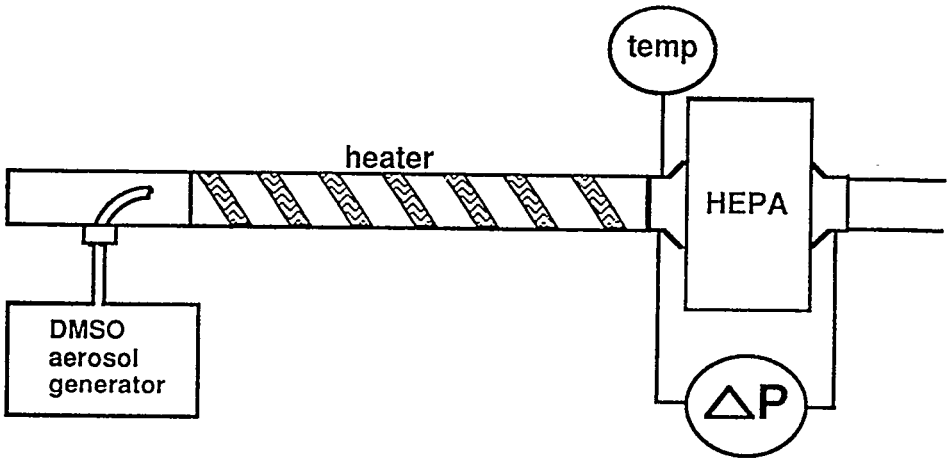


Figure 6. Schematic of apparatus used for simulating HEPA exposure to DMSO during the HE dissolution mode.

Following the 6 hour exposure to DMSO in the test apparatus shown in Figure 6, the HEPA was connected to a blower and had 30 cfm of air pass through the filter for 30 minutes. This operation represents the purge mode of the filter exposure.

The filter efficiency was also determined at the beginning of the HEPA exposure and at the end of each week of exposure for the six week test. We used a laser particle counter (Particle Measuring Systems HS-LPS) to measure the penetration of DOS aerosols throughout the HEPA filter at 50 cfm as a function of particle size from 0.065 to 1.0 μm diameter. The DOS aerosols were generated with a Laskin nozzle generator. The upstream concentration measurement was diluted to prevent coincidence counting. A schematic of the filter efficiency test apparatus is shown in Figure 7

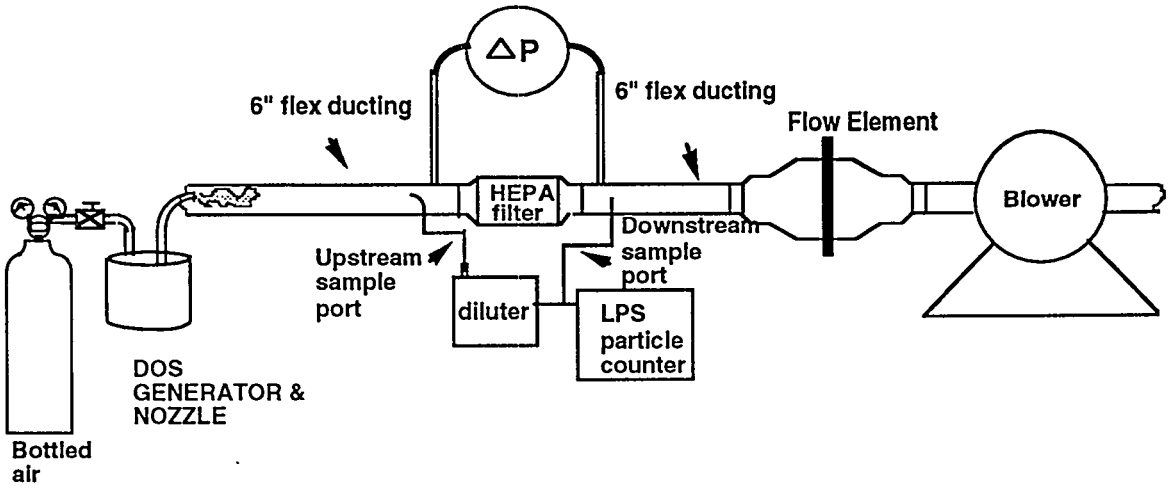


Figure 7. Schematic of filter efficiency test apparatus

For the first 81 hours of the 198 hour test, the HEPA filter was exposed daily to 2.9 g/m^3 of DMSO at 1.5 cfm for 6 hours followed by an air purge at 30 cfm for 0.5 hours. For the next 85 hours the HEPA filter was exposed to 0.78 g/m^3 of DMSO at 6.5 cfm for 6 hours, but with no air purges. The HEPA filter had no DMSO challenge or air flow during evenings and on weekends. This allowed liquid DMSO trapped on the filter to drain from the media and/or evaporate. During the final 32 hours, the HEPA filter was exposed to a 0.78 g/m^3 of aerosol generated from a solution of plastic EPON dissolved in DMSO to simulate the HE dissolved in DMSO. At the end of the test program, the filter was subjected to 100 cfm air flow (twice its rated flow) to see if the filter would develop structural damage.

V. Simulation Test Results

A summary of all of the filter exposure tests are given in Table 4., where the exposure time, initial and final pressure drop, flow rate, and computed mass of DMSO deposit is tabulated. The filter was exposed to DMSO aerosols for 198 hours over a six week period. Note that although the pressure drop of the HEPA filter increased during loading, it generally

Table 4. Summary of HEPA filter exposure tests

Date	Exposure time (hrs.)	ΔP initial (w.g.)	ΔP final (w.g.)	Test	Flowrate (CFM)	Mass of DMSO (gms)
8/20/93	5.50	0.020	0.041	Loading (1)	1.52	41.349
8/25/93	8.00	0.035	0.046	Loading	1.50	60.144
	0.50	0.580	0.680	Purge	30.00	0.000
8/26/93	8.00	0.250	0.055	Loading	1.50	60.144
	0.50	0.700	0.650	Purge	30.00	0.000
8/30/93	8.00	0.025	0.050	Loading	1.50	60.144
	0.50	0.061	0.061	Purge	30.00	0.000
8/31/93	9.50	0.025	0.046	Loading	1.50	71.421
	0.50	0.640	0.680	Purge	30.00	0.000
9/1/93	9.00	0.020	0.052	Loading	1.50	67.662
	0.50	0.660	0.710	Purge	30.00	0.000
9/2/93	8.00	0.031	0.057	Loading	1.50	60.144
	0.50	0.660	0.680	Purge	30.00	0.000
9/3/93	8.00	0.025	0.047	Loading	1.50	60.144
	0.50	0.640	0.670	Purge	30.00	0.000
9/7/93	8.50	0.024	0.054	Loading	1.50	63.903
	0.50	0.670	0.690	Purge	30.00	0.000
9/8/93	8.00	0.030	0.061	Loading	1.50	60.144
9/9/93	4.50	0.031	0.038	Loading	1.50	33.831
	4.00	0.165	0.165	Loading (2)	6.50	0.000
9/13/93	10.00	0.110	0.192	Loading	6.50	85.700
9/14/93	10.00	0.120	0.179	Loading	6.50	85.700
9/15/93	10.00	0.125	0.195	Loading	6.50	85.700
9/16/93	6.00	0.170	0.190	Loading	6.50	51.420
9/17/93	8.00	0.134	0.180	Loading	6.50	68.560
9/20/93	10.00	0.130	0.195	Loading	6.50	85.700
9/21/93	9.50	0.123	0.190	Loading	6.50	81.415
9/22/93	8.00	0.135	0.190	Loading	6.50	68.560
9/23/93	4.00	0.320	0.350	Loading	6.50	34.280
	1.50	2.300	2.600	Loading (3)	42.00	12.855
9/24/93	7.00	0.250	0.390	Loading (4)	6.50	59.990
	2.00	0.390	0.430	Loading (5)	6.50	17.140
9/27/93	8.00	0.320	0.480	Loading	6.50	68.560
9/28/93	9.00	0.400	0.670	Loading	6.50	77.130
9/29/93	5.50	0.600	0.980	Loading	6.50	47.135
		5.000		Purge (6)	100.00	0.000
					Total DMSO Collected	
Notes:	(1) DMSO is generated at 7.518 gms/hr.				on filter (gms)=	1568.88
	(2) DMSO is generated at 8.570 gms/hr.					
	(3) DMSO is generated at 8.570 gms/hr. and is put into a 42 CFM airstream					
	(4) DMSO is generated at 8.570 gms/hr.					
	(5) 8.570 gms/hr. of DMSO is collected on filter. 72.0 gms of EPON 1002F					
	and 92.0 gms of EPON 1004F was added to 2.20 liters of DMSO					
	(6) Purge test with 100 CFM. Approximately 2.0 pints of DMSO spewed out of the					
	downstream side of the filter and blower.					

returned to its original value at the beginning of the next test on the following day. Figure 8 shows the typical increase in filter pressure drop at 1.5 cfm during the 6 hour DMSO exposure. The elevated pressure drop had decreased to its initial value by the start of the next day's test. This increase in filter pressure drop and the subsequent decrease on sitting overnight was typical throughout the test program, even for the DMSO loadings that were not followed by an air purge.

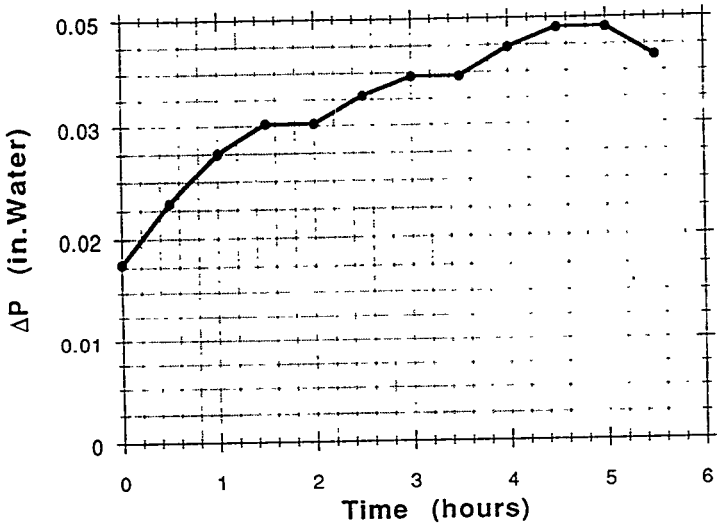


Figure 8 Typical increase in filter pressure drop during DMSO loading during the simulated HE dissolution process.

Filter penetration measurements with DOS aerosols were also conducted prior to DMSO exposure and after each week of DMSO exposures. All of the tests were conducted at the rated flow of 50 cfm. Figure 9 shows the filter penetration prior to DMSO exposure.

The filter penetration measurement after approximately every forty hours of DMSO exposure up to 166 hours was similar to the data for the fresh, unexposed HEPA filter. Although the filter pressure drop had increased from 1.15 inches to 1.55 inches after 166 hours, this increase is due to the atmospheric dust that deposited on the filter. Figure 10 shows that the filter penetration after 166 hours of DMSO exposure is comparable to the penetration for the new filter in Figure 9. It is clear from the data that exposure to DMSO will not cause a rapid degradation in HEPA filter performance.

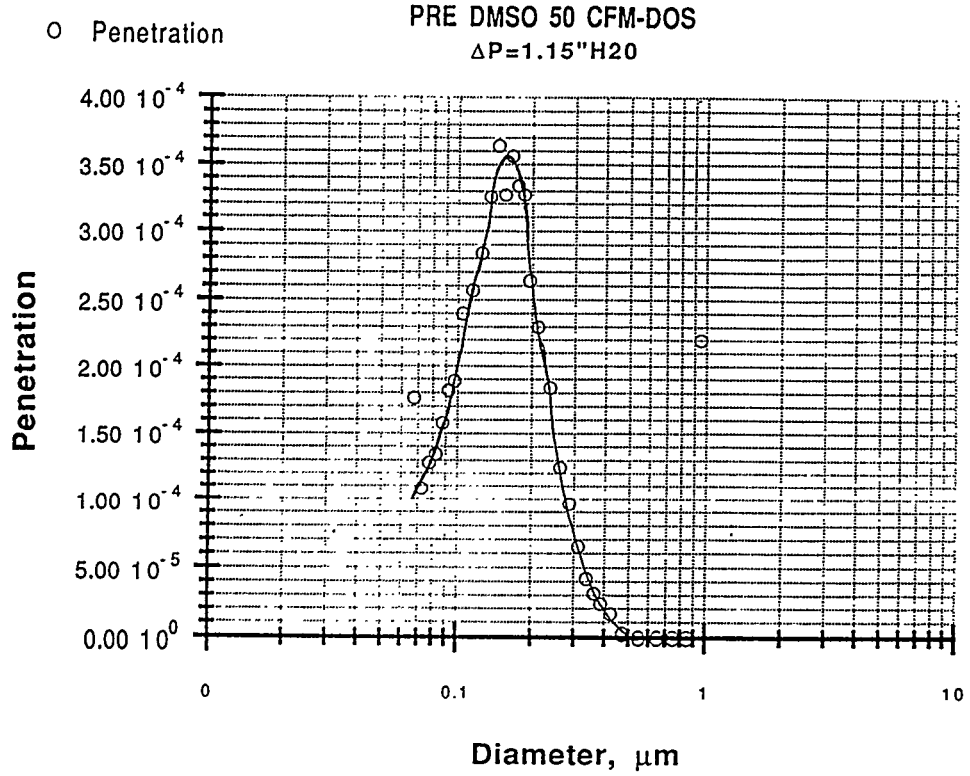


Figure 9. Penetration of DOS aerosols through HEPA filter prior to DMSO exposure.

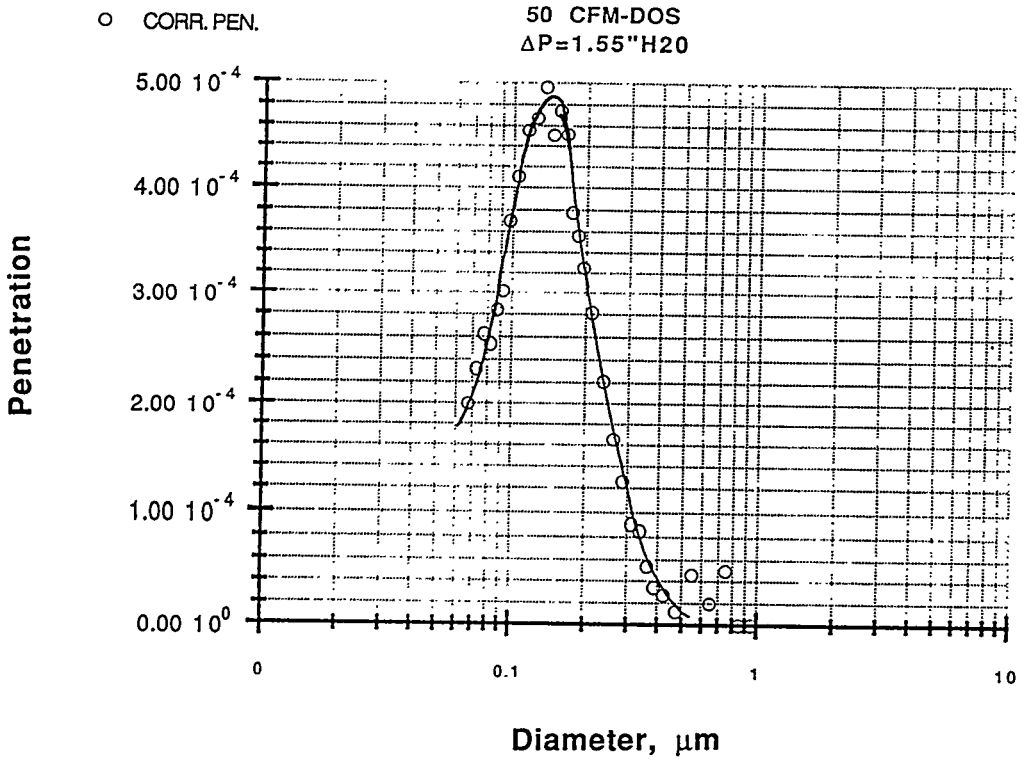


Figure 10. Penetration of DOS aerosols through HEPA filter after 166 hours of DMSO exposure.

Since the DMSO will have up to 25% dissolved HE, we conducted exposure tests on the HEPA filter with 6.3% dissolved plastic to see the effect of dissolved solids. After 32 hours of loading with the DMSO/plastic solution, the HEPA filter pressure drop increased from 0.32 inches to 0.98 inches (See Table 4) at a flow rate of 6.5 cfm. We then tried to pass 100 cfm through the loaded filter to see if the filter would be structurally damaged.

However, we quickly found that liquid DMSO was spraying out from the downstream side of the HEPA filter. By a combination of high air flow (100 cfm) and pouring out standing DMSO, we were able to remove about 1 liter of DMSO that was trapped inside the filter. According to Table 4, the computed mass of DMSO injected into the filter is 1.59 kg or 1.4 liters of liquid. Since we recovered 1 liter out of 1.4 liters, most of the DMSO aerosol injected into the filter will remain in the filter as a liquid. The HEPA filter apparently acts as a sponge soaking up liquid DMSO until the filter becomes fully saturated

After we removed the liquid DMSO from the filter, we conducted a DOS penetration test at 50 cfm and found the pressure drop had increased to 2.85 inches, and the aerosol penetration increased to 1.4%. The penetration measurements are shown in Figure 11 The significantly increased width of the penetration curve as well as the higher penetration values in Figure 11 compared to that in Figures 9 and 10 is characteristic of a filter leak. It is unlikely that a filter leak was caused by the high air flow since the total pressure drop across the HEPA was only 5 inches, and that pressure was divided between the prefilter and the HEPA filter.

We then tried to remove any residual DMSO remaining in the filter by passing heated air at 150 F through the filter. After four hours, the filter pressure drop decreased from 2.85 to 2.1 inches, and the filter penetration decreased from 1.4% to 1.1%. Three additional days of passing 50 cfm of 150 F heated air through the filter reduced the pressure drop to 1.68 inches and the penetration to 0.4%. The DOS penetration curve for the HEPA filter after the three day purge is shown in Figure 12.

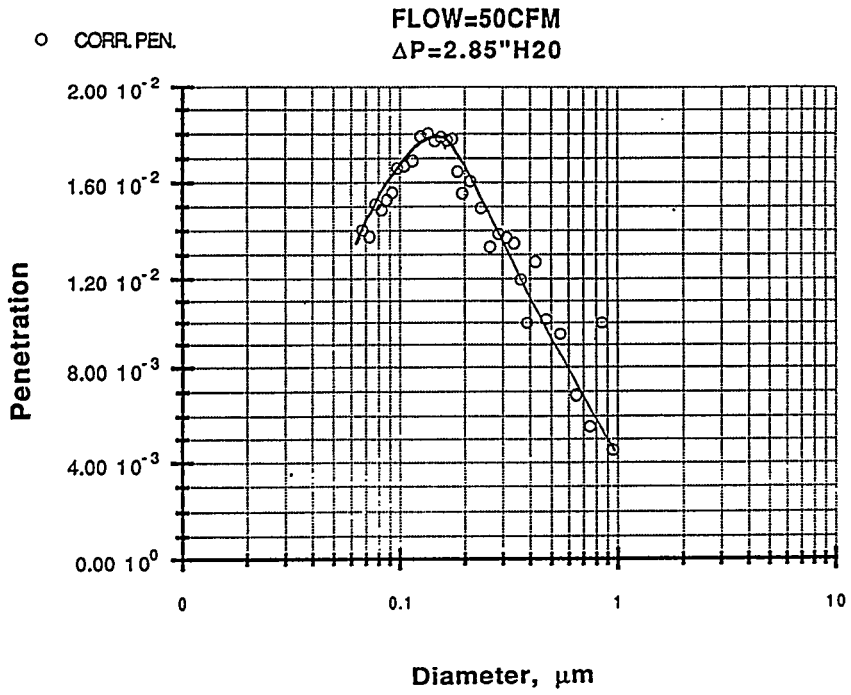


Figure 11. Penetration of DOS aerosols through HEPA filter exposed to 198 hours of DMSO aerosols followed by 32 hours of DMSO/dissolved plastic aerosols.

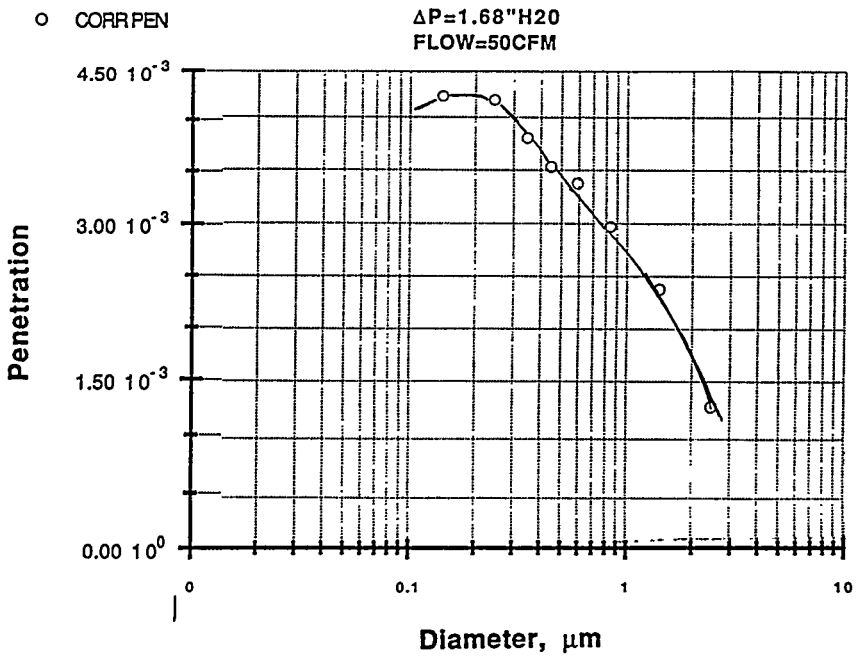


Figure 12. Penetration of DOS aerosols through HEPA filter after three days of purging with 50 cfm of heated air

A summary of all the penetration measurements made in this evaluation is given in Table 5.

Table 5. Summary of penetration and pressure drop tests.

Exposure (Hours)	Pressure Drop (Inches Water)	Penetration at 0.3 μm
0	1.15	8×10^{-5}
6	1.25	8×10^{-5}
56	1.32	6×10^{-5}
81	1.25	9×10^{-5}
120	---	---
166	1.55	1.1×10^{-4}
198	2.85	1.4×10^{-2}
+	2.10	1.1×10^{-2}
++	1.68	4.0×10^{-3}
+++	1.30	3.0×10^{-4}

\+ After purging with 50 cfm air at 150°F for 4 hours
++ After purging with 50 cfm air at 150°f for 3 days
+++ HEPA requirements

We can attribute the changes in filter performance to the residual DMSO in the filter that was driven off by heated air. The filter pressure drop and the penetration decreased from 2.85 inches to 1.68 inches and from 1.4% to 0.4% respectively by passing heated air through the filter. The remaining effects are due to the atmospheric dust deposits and the permanent structural damage from the DMSO attack on the HEPA filter. Table 5 shows that the pressure drop increased by about 0.1 inch of water for each 40 hour week of operation except for the final week when liquid DMSO had saturated the filter. However, once the DMSO was driven off by heat, the increase in pressure drop for the final week was also about 0.1 inch. This pressure drop increase is most likely due to the atmospheric dust that has accumulated on the filter. The accumulation of atmospheric dust is seen as the darkened center of the filter pack in Figure 13. The front end of the wooden box was cut open to expose the filter.

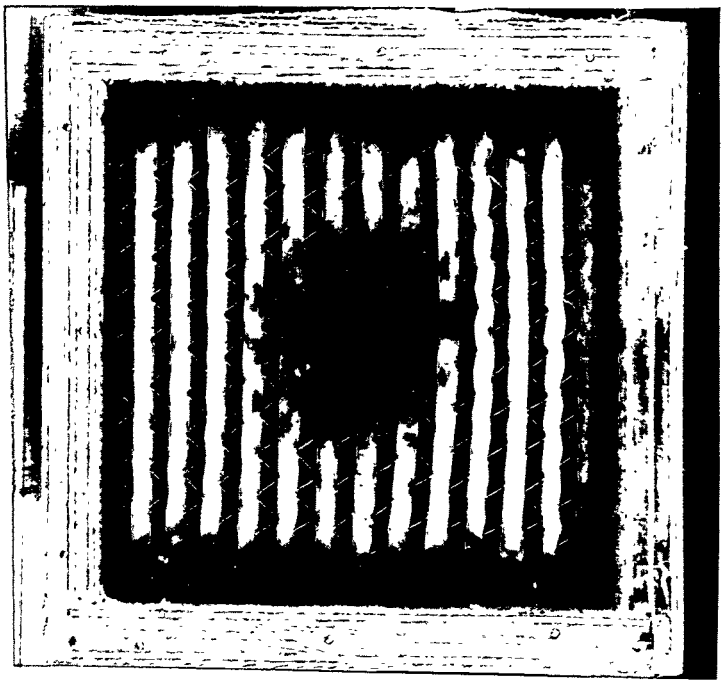


Figure 13. Inlet side of the prefilter-HEPA filter with end cut open to expose the media. Note the dust accumulation in the center of the media pack.

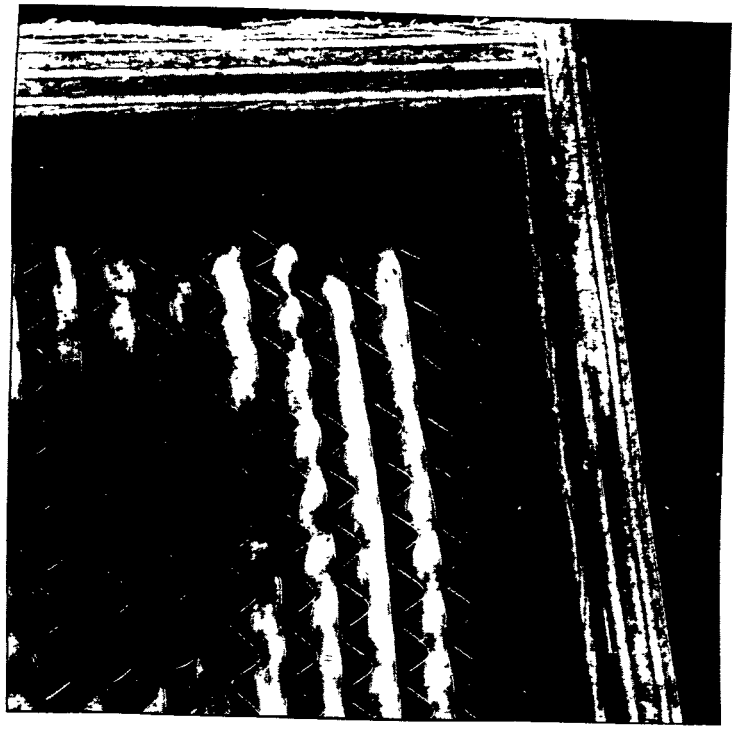


Figure 14. Close-up of upper right corner of Figure 13 showing a crack in the urathane sealant between the wooden frame and the media pack. This crack is a leak path.

We cut the HEPA filter open to expose the inlet and exit filter faces to determine if we could see signs of physical degradation that could explain the increased filter penetration. Figure 14 shows that there is a small crack in the sealant between the wooden frame and the filter pack. This crack was found to be one of two leaks that produced the higher aerosol penetration

We also found that the DMSO exposure created cracks and warped the plywood frame of the HEPA. Figure 15 shows the exterior corner of the plywood frame with the paint blistering and the edges separating due to warpage. Figure 16 is a cut-away portion of the same corner and shows a direct leak path from the interior to the exterior of the HEPA box. Since the leak path is on the inlet side of the HEPA, it can not be the cause of the increased aerosol penetration because any particles leaking into the box would be filtered.



Figure 15. Photograph of a corner of the HEPA filter showing box warpage and joint separation.

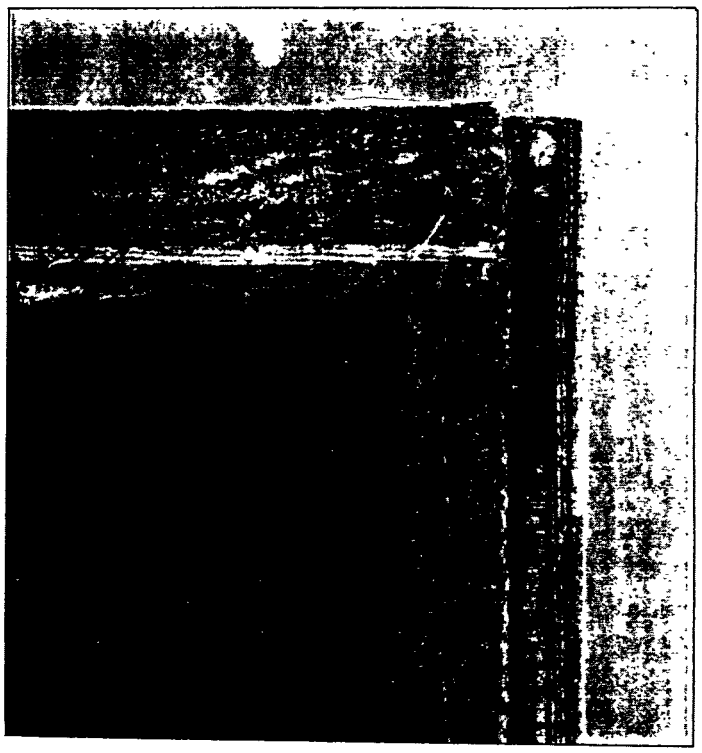


Figure 16. Photograph of cut-away portion of the corner in Figure 15. Note the direct leak path at the corner.

We conducted a series of scan tests to verify that the cause of the filter failure was due to the crack in the sealant between the filter pack and the plywood frame shown in the upper right corner of Figure 14. The scan test consisted of scanning across the open filter face (inlet side with prefilter) with a 1/4 inch probe and measuring the DOS concentration with a light scattering photometer (Phonex Precision Instruments, model JM-7000). DOS aerosols were were injected in the opposite filter face with a 5 cfm air stream. Measurements confirmed that aerosols were leaking through the visible crack in the upper right corner. A similar leak was also detected on the bottom right corner, although no visible crack was seen. The filter was then reversed and another scan made across the filter face (exit side with HEPA filter). The scan measurements showed leaks on the upper and lower left corners between the sealant and the plywood frame, and directly opposite from the leaks on the other side. There were no visible cracks. It appears that the leak path between the sealant and frame runs along the entire filter depth.

VI. Preliminary Field Test Results

We evaluated the HEPA filter that was used in trial tests for removing HE from mock warheads to see if there was any deterioration in performance. The filter was a dual-stage prefilter-HEPA filter unit manufactured by Donaldson and similar to the boxed unit in Figure 5, but with a 135 cfm capacity . The larger capacity HEPA filter was used to reduce the restriction in the exhaust flow compared to the standard 50 cfm filter used in our laboratory tests. The filter had been exposed to a total of 50 hours of DMSO/HE exposure: 5 hours of the venting phase when the exhaust valve was open and 45 hours of the dissolution phase in which hot DMSO is sprayed against the HE. We estimate the exhaust flow was about 5 cfm during this dissolution process. Unfortunately, the work station was not instrumented to provide a measure of the exhaust flow or the pressure drop across the HEPA filter.

The test apparatus for measuring the filter penetration and the pressure drop is shown in Figure 8. Prior to conducting the penetration tests, we monitored the downstream flow for potential HE release, but found none. The results of the penetration measurements at 135 cfm and 27 cfm are shown in Figures 17 and 18 respectively. The filter penetration values at 0.3 μm are .006% and .0004% at 135 cfm and 27 cfm respectively. The corresponding pressure drop measurements are 1.80 inches and 0.30 inches at 135 cfm and 27 cfm respectively. For comparison the filter certification tests on the new filter at the Rocky Flats Filter Test Station showed the penetration was less than .01% for both 135 cfm and 27 cfm. The only pressure drop recorded was 2.2 inches at 135 cfm. We suspect that restriction in the adapters used to connect to the boxed HEPA filter was responsible for the higher initial pressure drop.

The preliminary data suggests that there was no degradation in filter efficiency or pressure drop within the 50 hours of DMSO/HE exposure. If the HEPA degradation for the 135 cfm HEPA follows the same trend as seen for the 50 cfm HEPA filter, then no degradation would be seen until about 200 hours of DMSO exposure. The field evaluation was clearly too short to see any loss in HEPA performance.

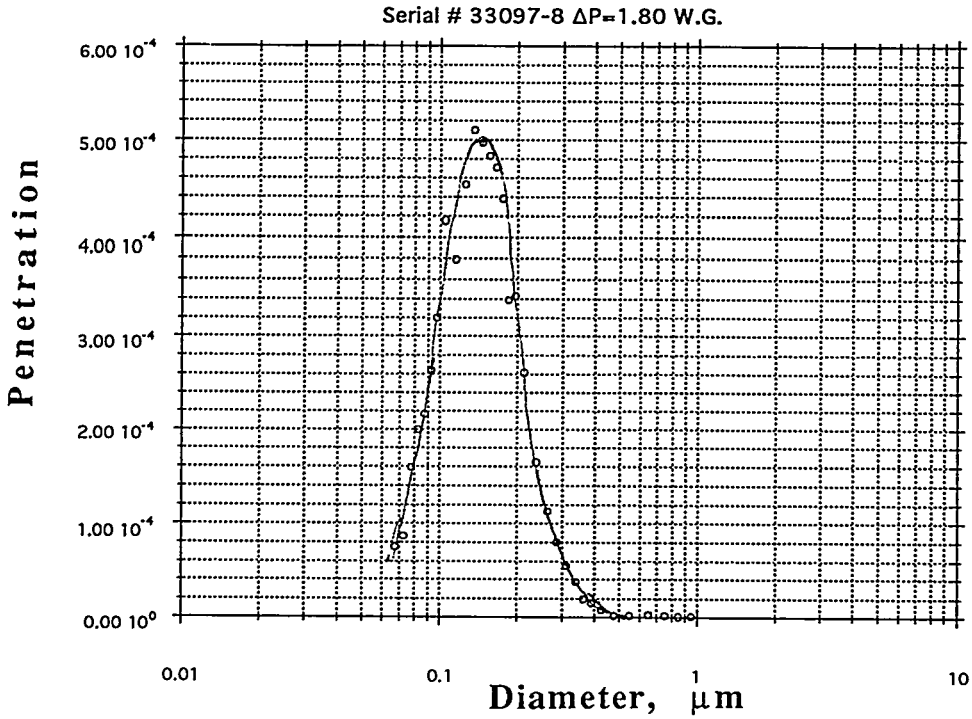


Figure 17. Penetration of DOS aerosols through 135 cfm HEPA filter at 135 cfm after 50 hours exposure to DMSO/HE

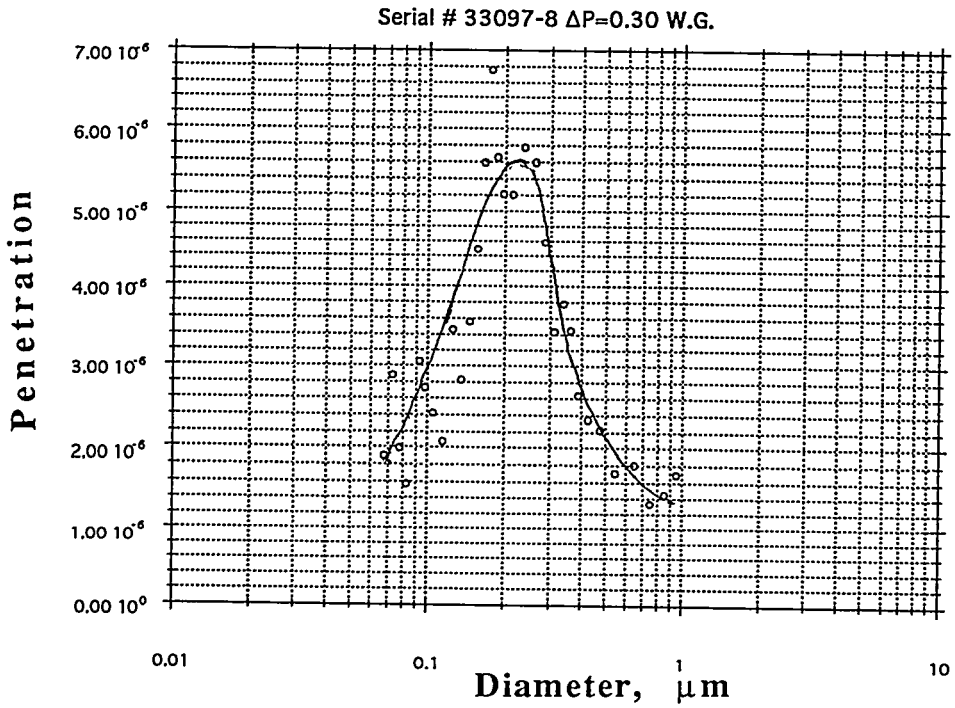


Figure 18. Penetration of DOS aerosols through 135 cfm HEPA filter at 27 cfm after 50 hours exposure to DMSO/HE

V. Conclusions

Preliminary tests show that DMSO aerosols degrade the performance of HEPA filters over time by chemical attack to create leak paths and by saturating the filter media to increase both the pressure drop and the aerosol penetration. A 50 cfm HEPA filter was exposed to DMSO aerosols for 198 hours under conditions that simulate the HE dissolution operations in weapons dismantlement. The efficiency for 0.3 μm particles decreased from 99.992% to 98.6%, and the pressure drop increased from 1.15 inches to 2.85 inches during the exposure. We found that most of the DMSO aerosol that was captured by the HEPA filter had condensed in the filter as a pool of liquid. Since the efficiency of the HEPA filter drops below the minimum allowed efficiency of 99.97%, it has to be replaced with a new filter.

The larger portion of the decreased efficiency and increased pressure drop is due to the DMSO saturation of the HEPA filter media. Of the 1.39% decrease in efficiency, 1.00% is due to the saturation, while 0.39% is due to the leak paths. Of the 1.70 inches increase in pressure drop, 1.17 inches is due to the saturation, while 0.53 inches is due to the atmospheric dust that was not prefiltered in our experiments. We were able to isolate the effect of the DMSO saturation by driving off the liquid with heated air. The difference in efficiency and pressure drop before and after the DMSO is driven off can be attributed to the DMSO saturation. Loss in filter performance due to liquid saturation or partial saturation is not unique to DMSO, but applies to all liquids.

The smaller portion of the decreased efficiency in these preliminary tests is due to the leak paths generated by DMSO attack. The DMSO had produced two leak paths between the polyurethane sealant and the plywood frame. One of the leak paths was visible in the photograph in Figure 14 where the sealant was separated from the plywood frame. Although the penetration due to the leak path is about one third of the total penetration, the 0.39% penetration due to the leaks is still more than ten times the allowed 0.03% penetration. The filter would have to be replaced.

The DMSO exposure also produced visible cracks and warpage on the inlet portion of the plywood frame as shown in Figures 15 and 16. Although the cracks formed a leak path to the exterior of the filter, the leak was not measured as an increase in aerosol penetration because the filter was under vacuum, and the ambient aerosols would be filtered. We

believe that the accumulated DMSO liquid in the boxed HEPA filter is responsible for both the plywood warpage and the creation of leak paths due to separation of the sealant from the plywood frame.

The efficiency of the two exhaust HEPA filters in Figure 1 to remove potential radioactive or DMSO/HE aerosols depends on the exhaust flow rate, the particle size distributions of the aerosols, and any degradation resulting from DMSO exposure. Although most of the DMSO aerosols will be trapped in the first HEPA filter, a large fraction of DMSO vapor may pass through the first filter and condense on the second HEPA filter and cause degradation. Since the HE is not volatile, it will remain on the first filter except for the small fraction that passes through as an aerosol. Thus the combined efficiency for a degraded first filter (98.6%) and a standard second filter (99.97%) is 99.99996% for 0.3μm particles. If both filters are degraded (98.6%) by DMSO attack, then the combined efficiency will be 99.98%. If the first filter is replaced prior to its degradation, each of the filters will have 99.97% efficiency, and the combined efficiency will be 99.999991%. The collection efficiency for DMSO/HE aerosols will be much higher because the particle size is much greater. However, if liquid DMSO/HE is allowed to accumulate in the filter, it can flow unobstructed through the filter, thereby leaving only the second HEPA filter for stopping the DMSO/HE aerosols. In this case the overall efficiency for the two filters is 99.97% for 0.3 μm particles.

VI Recommendations

We recommend that the first stage HEPA filter from the DMSO dissolution workstation be replaced after 160 hours of exposure. If the pressure drop across the HEPA filter increases by more than 50% of its initial value, then the filter should also be replaced even if the exposure is less than 160 hours. This will assure that the HEPA filter will not be degraded if different operating conditions cause accelerated filter saturation. The first stage HEPA filter should also be mounted in a vertical configuration to promote liquid drainage into the workstation. The second stage HEPA filter also should be monitored (pressure drop increase) for DMSO accumulation due to condensation from the vapor. If the pressure drop increases by 50% of its initial value, the second stage HEPA should be replaced.

We also recommend that the filters removed from the dissolution workstations be evaluated for filter efficiency and examined for any

structural damage. This will provide a data base for defining the HEPA degradation under actual field conditions.

VII. Acknowledgement

The authors thank Mr. Derek Wapman, Project Manager of the LLNL HE Dissolution Workstation Project, for his support and for many helpful discussions and suggestions.

DISCUSSION

GOOSSENS: I wonder why you do not use a high efficient mist eliminator (HEME) as we have introduced in the PAMELA off-gas treatment circuit.

VENDEL: Our mist eliminators remove all particles of 1.5 μm.

BERGMAN: Thank you for your suggestion. We will certainly take a closer look at mist eliminators in our future work. We initially considered installing a mist eliminator but decided not to use one because of the added space requirement, the small size of DMSO droplets, and the limited ventilation system. We were worried that the existing ventilation system only had 6 inches of vacuum and could not accommodate a high efficiency mist eliminator in addition to the two HEPA filters.

VENDEL: It was manufactured by an American company?

BERGMAN: Yes.

FIRST: The US-approved mist eliminator that has gone into a number of nuclear power plants is tested at a droplet size of about 3 μm. When particle size decreases, it still has significant efficiency. In other words, it does not get below 80% efficiency until you get well down into the fractional micrometers. Therefore, when you have a very deep, dense eliminator, such has been put into nuclear plants, I think it would do a good eliminator job, even on 1.5 μm; it would protect the HEPA filter for quite a long time. On the other hand, if you think of a prefilter as one of these half-inch thick mesh or fiber units, of course it will not be very efficient.

BERGMAN: We were thinking of the half-inch mesh kind that they put in front of HEPA filters. You made an excellent suggestion and I think I will amend the paper accordingly.

PROPOSED RETROFIT OF HEPA FILTER PLENUMS WITH INJECTION AND SAMPLING MANIFOLDS FOR IN-PLACE FILTER TESTING

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Abstract

The importance of testing HEPA filter exhaust plenums with consideration for As Low as Reasonably Achievable (ALARA) will require that new technology be applied to existing plenum designs.

I. Introduction

HEPA filter in-place testing at Rocky Flats has evolved slowly due to a number of reasons. The first plenums were built in the 1950's preceding many standards. The plenums were large which caused air dispersal problems. The systems were variable air flow. Access to the filters was difficult. The test methods became extremely conservative. Changes in methods were difficult to make. The acceptance of new test methods has been made in recent years with the change in plant mission and the emphasis on worker safety.

II. History of In-place Testing at Rocky Flats Plant

The Rocky Flats Environmental Technology Site, Department of Energy (DOE), utilizes approximately eleven thousand, size 5, HEPA filters in supply and exhaust systems. Eight thousand filters are used for PU Operations. The filter plenums that house the HEPA filters contain from three filters per stage of filtration to six hundred and twenty filters per stage. The filter plenums have from 2 stages of filtration to four stages.

Early in the history of Rocky Flats (e.g., the 1950s and 1960s), in-situ mass flow testing of filter plenums was performed. This method flooded the upstream filter plenum with DOP, and measured the concentration downstream of the HEPA filters. In the 1970s, this method was changed to one of testing individual HEPA filters because the size of the filter plenums precluded full plenum flooding with DOP. In this revised testing, personnel entered the filter plenum upstream of the HEPA filter being tested and sprayed DOP aerosol at the filter. Concurrently, personnel in the filter plenum downstream of the HEPA filter being tested scanned the filter to measure the individual filter efficiency. (See Figure 1). The average of all the filters was used for the stage efficiency.

In the 1980s, further improvements were made. A separate Filter Test Group was formed to provide single-point accountability for HEPA filter testing.

These personnel received detailed training in HEPA filter testing. Management personnel attended formal off-site training. The Filter Test Group formalized procedures for in-place testing of HEPA filters. To ensure consistency of measurements, enhancements were made to existing filter test practices, including utilization of a shroud to direct DOP to the specific filter being tested. (See Figure 2).

In 1994 a test shroud, (as in Figure 3), was put into use on the downstream of the HEPA filter. The use of the shroud greatly reduced the filter test time but still required the test personnel to enter the filter plenum.

III. Project Description

The goal of the project is to determine the type of in-place test manifold which can be best utilized in existing filter plenums at the Rocky Flats Site. We will use an existing "New" filter plenum for testing manifold systems; we will evaluate existing manifold system designs in use at other sites: and we will try to minimize the physical modifications required to install a manifold system in the existing filter plenums.

IV. Facility

The test site selected is a building that was slated for use as a waste packaging facility, (as in Figure 4). The process requirements changed, leaving the building open for other uses.

A new addition had been added to the building just prior to shut down of the process. The new section included two filter plenums which were never put on line. (See Figure 5). The plenum selected for the testing is located on the ground floor. (See Figure 6). It is a two (2) stage walk-in unit with nine (9) filters per stage, air locks, (as in Figure 7), and a deminster section.

The modifications required to this system are minimal. The two system fans will require control modifications to enable them to be run simultaneously for CFM variations. The inlet duct system will require modification. The duct will be disconnected from the process area and routed to bring air in from outside the building.

V. Manifolds

Three styles of injection manifolds will be installed and tested. Modifications to these will be made to optimize there design. Style One will be a duct inject type installed in the inlet duct, using the plenum inlet deflector plate and the deminster section to mix the challenge. Style Two, a pipe grid manifold (see Figures 11 and 12) and Style Three, an air injected deflector (see Figure 13) will be installed between stages of filtration, air flow will supply the mixing.

Sample manifolds will be installed upstream and downstream of the filter stages. The manifold will take a sample at each filter. (See Figure 14). Additionally a exhaust duct sample point will be evaluated.

The plenum window ports will be modified for hose connections to introduce the test challenge and take samples. (See Figure 15).

Qualification of the test method will follow ASME N-509 and N-510 Standards. The test plenum is similar to on-line plenums allowing for simulation of air flows and mixing patterns.

VI. Additional Test Plenum Uses

The filter plenum manifold development project will involve a number of side projects. The evaluation of a Los Alamos National Laboratory laser spectrometer in-place test system and it's use with the manifold system (this system allows the testing of two filter stages at the same time). (See Figure 16).

The testing of the effect of a fire water spray system on HEPA filters. (See Figure 8).

The sample line loss for various material and lengths of run.

The evaluation of a filter hold down system to replace the multiple filter hold downs now in use. (See Figure 9).

The development of a X-Y scan method for material collected on filters.

The plenum will also provide a training location. The system can be expanded to provide simulation D & D mock ups.

VII. Program Status

The project was begun in fiscal year 1993. Equipment was purchased and Engineering was completed. The project was unfunded for fiscal year 1994. The request for funding was submitted for fiscal year 1995.

VIII. Conclusion

This project is a research and development project with two driving forces, ALARA and Cost Savings. We are open to any suggestions.



Figure 1



Figure 2

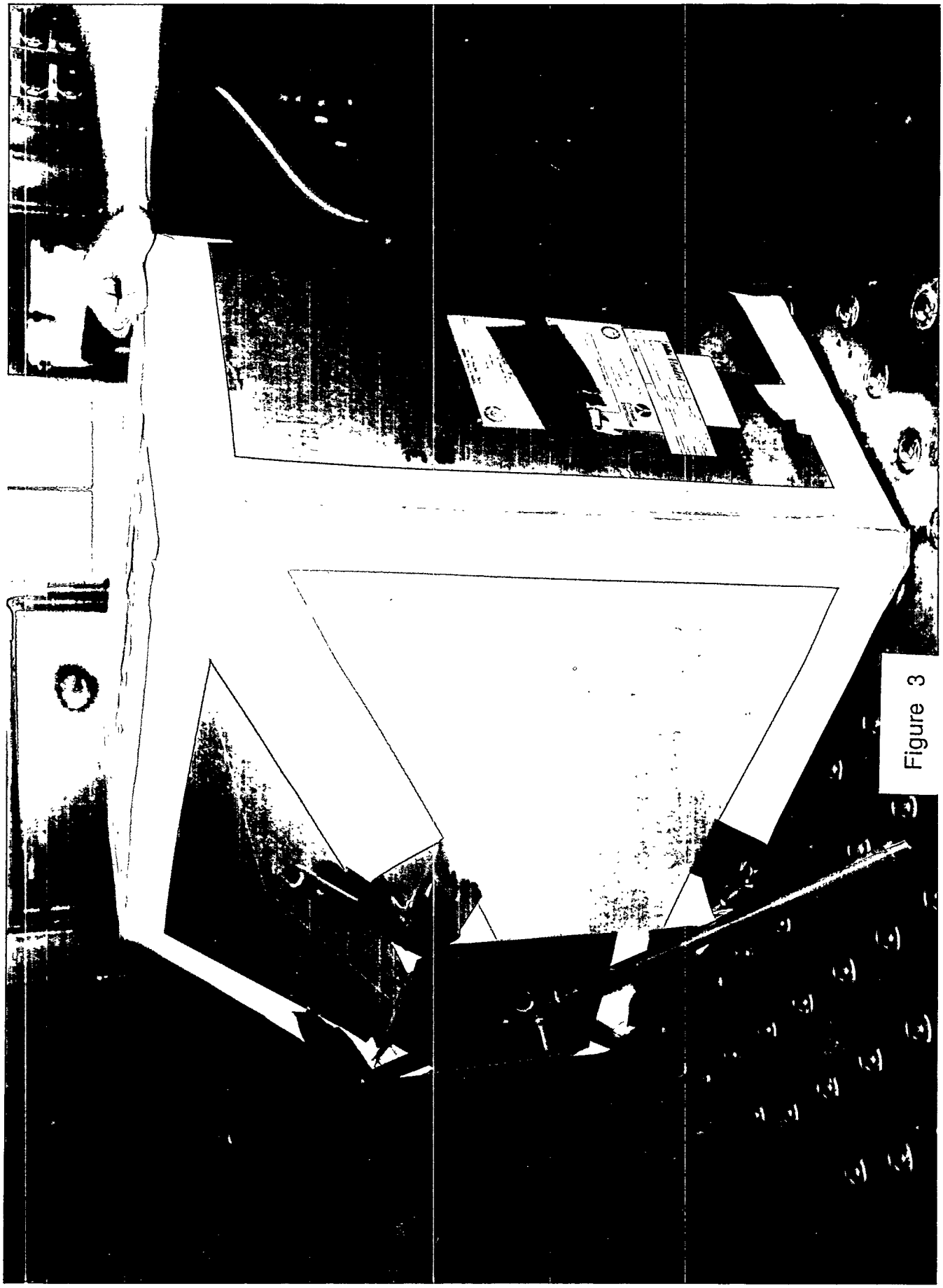


Figure 3

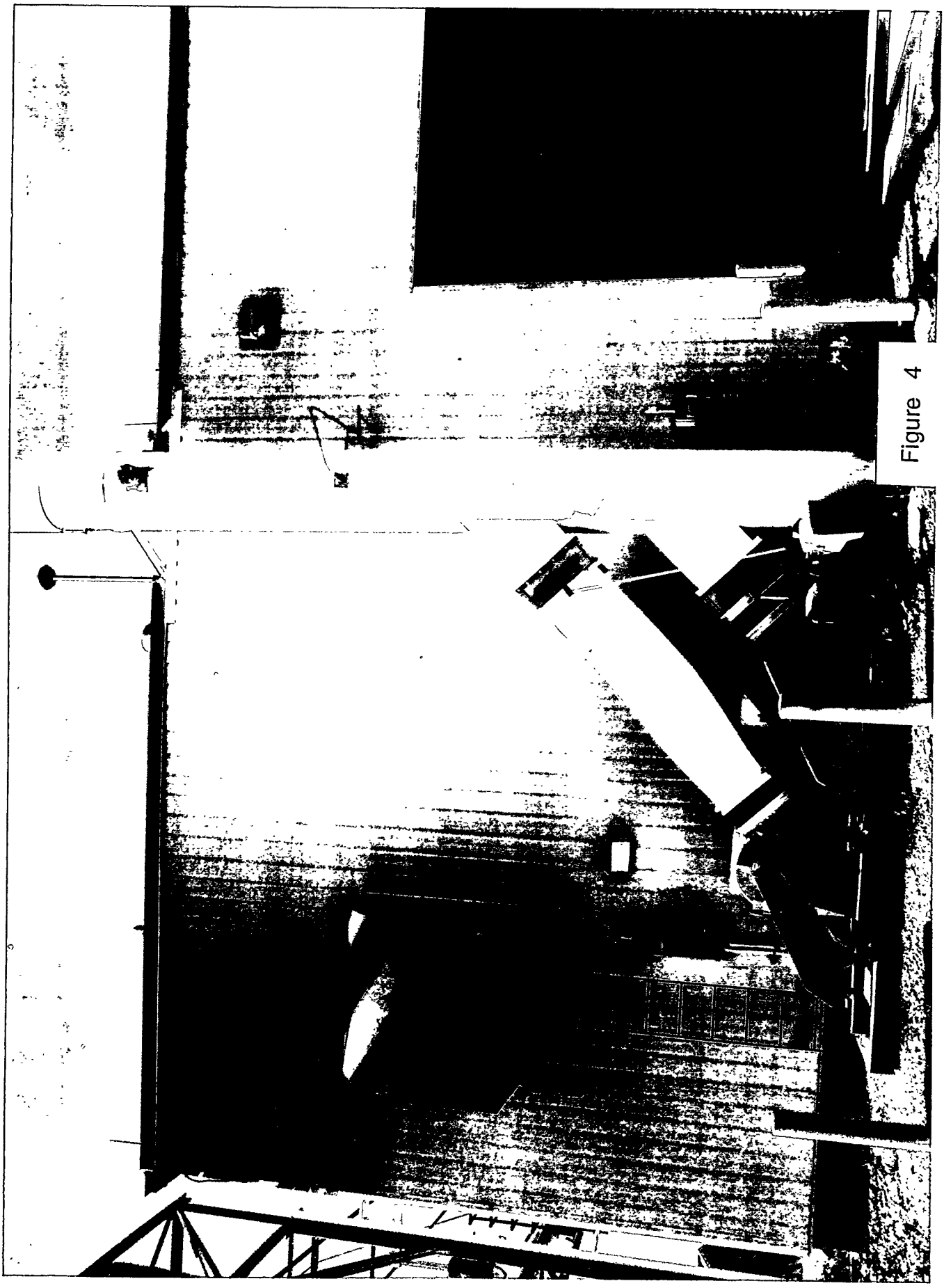


Figure 4

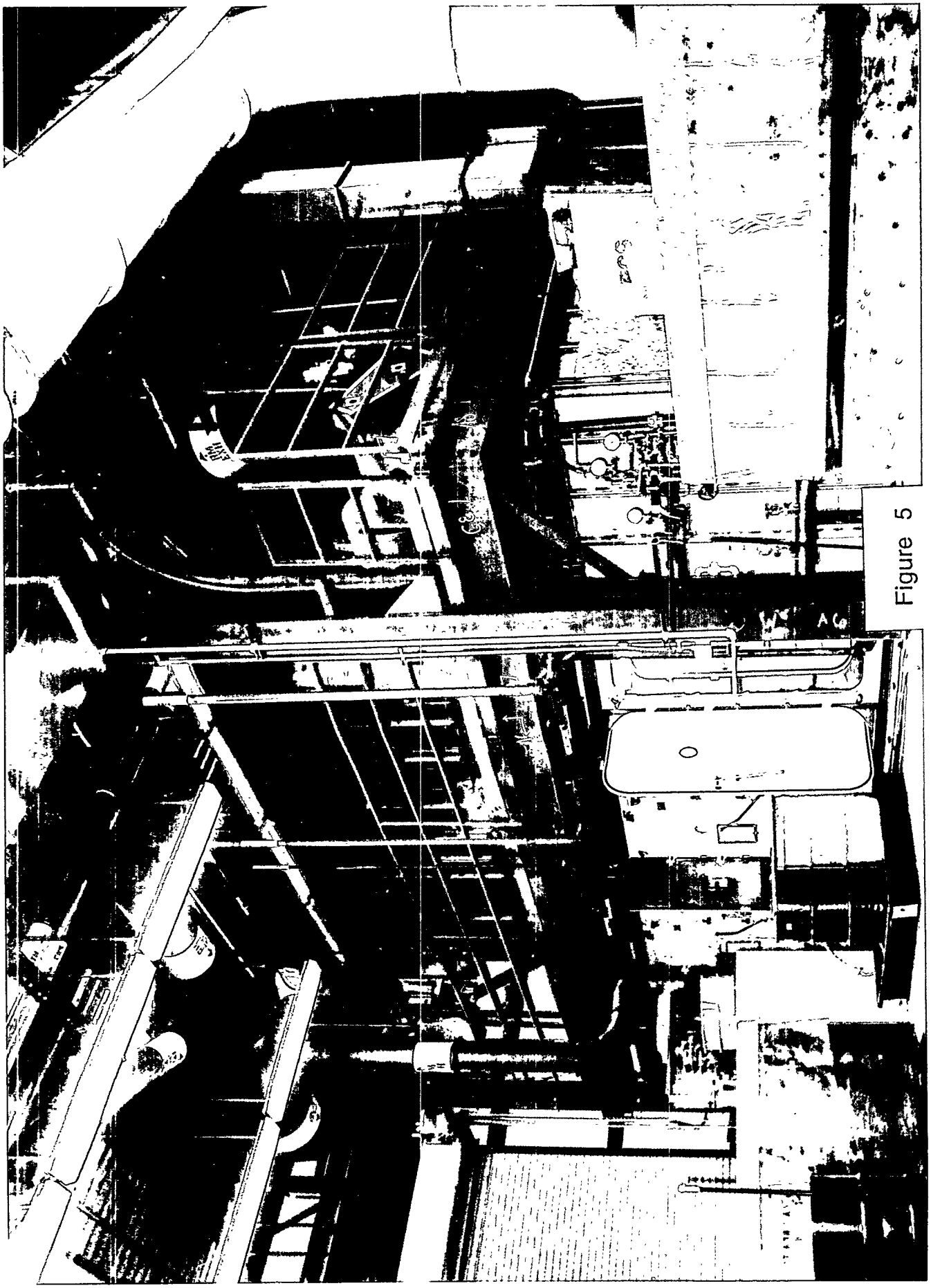


Figure 5

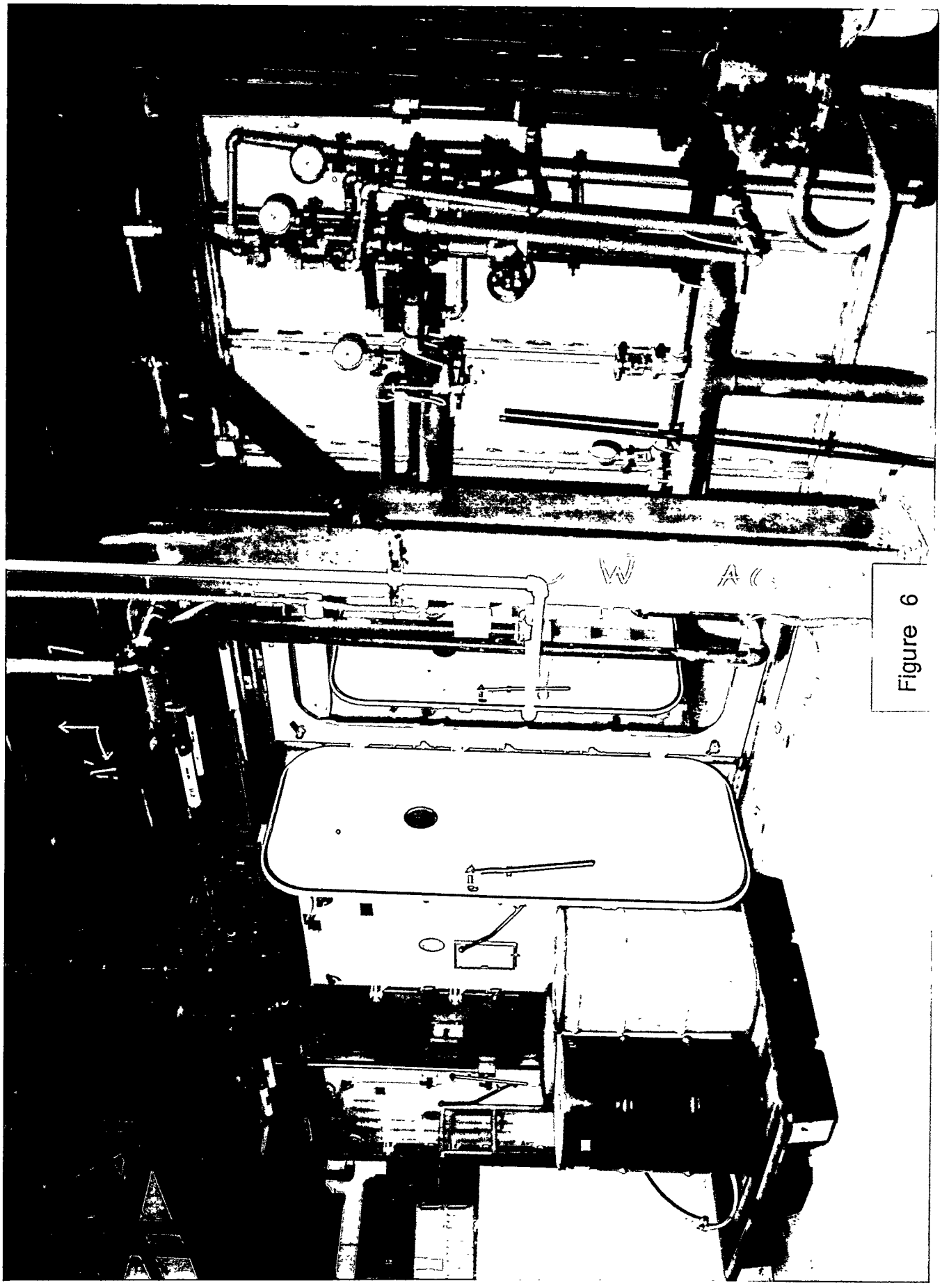


Figure 6

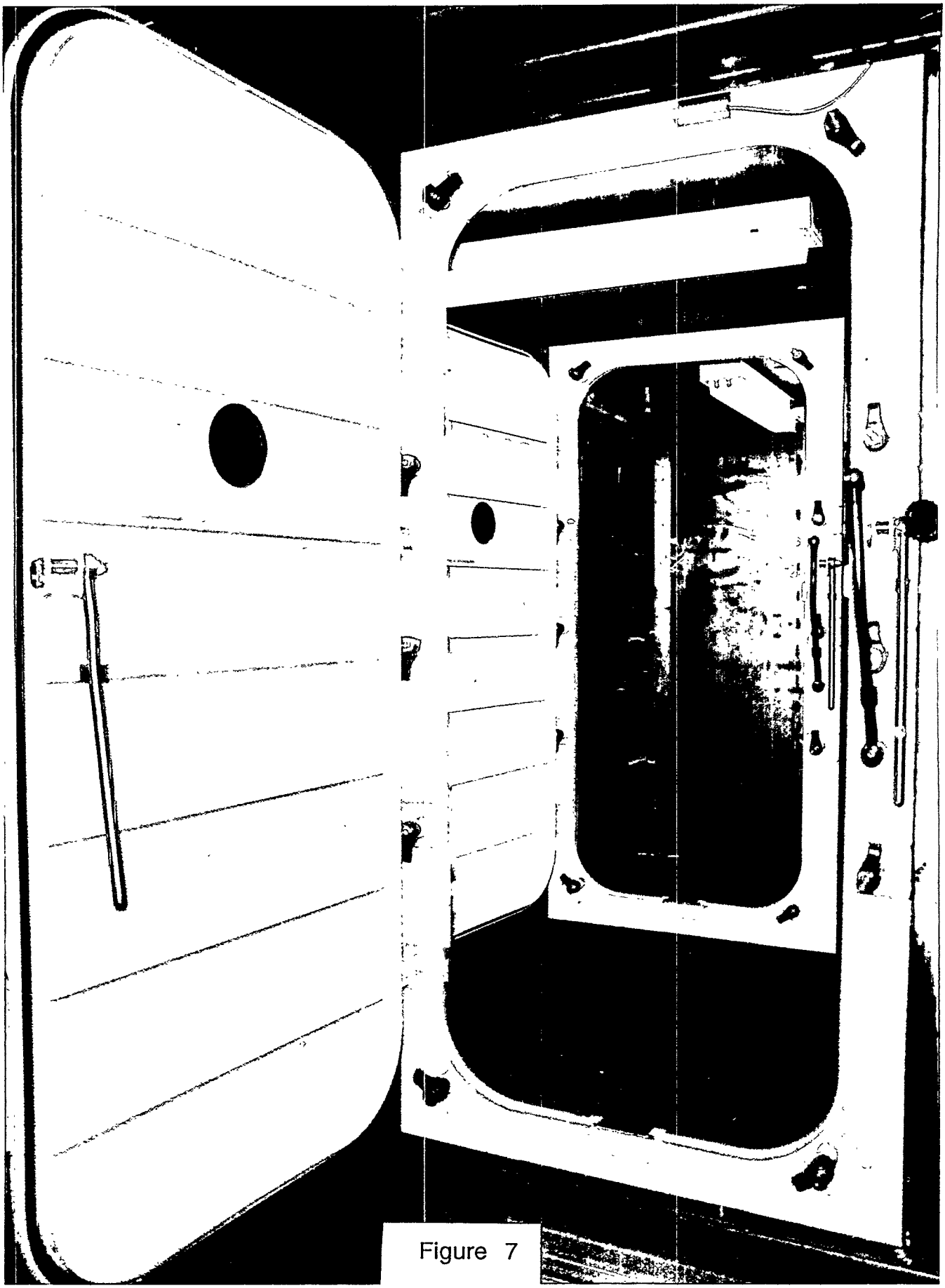


Figure 7



Figure 8

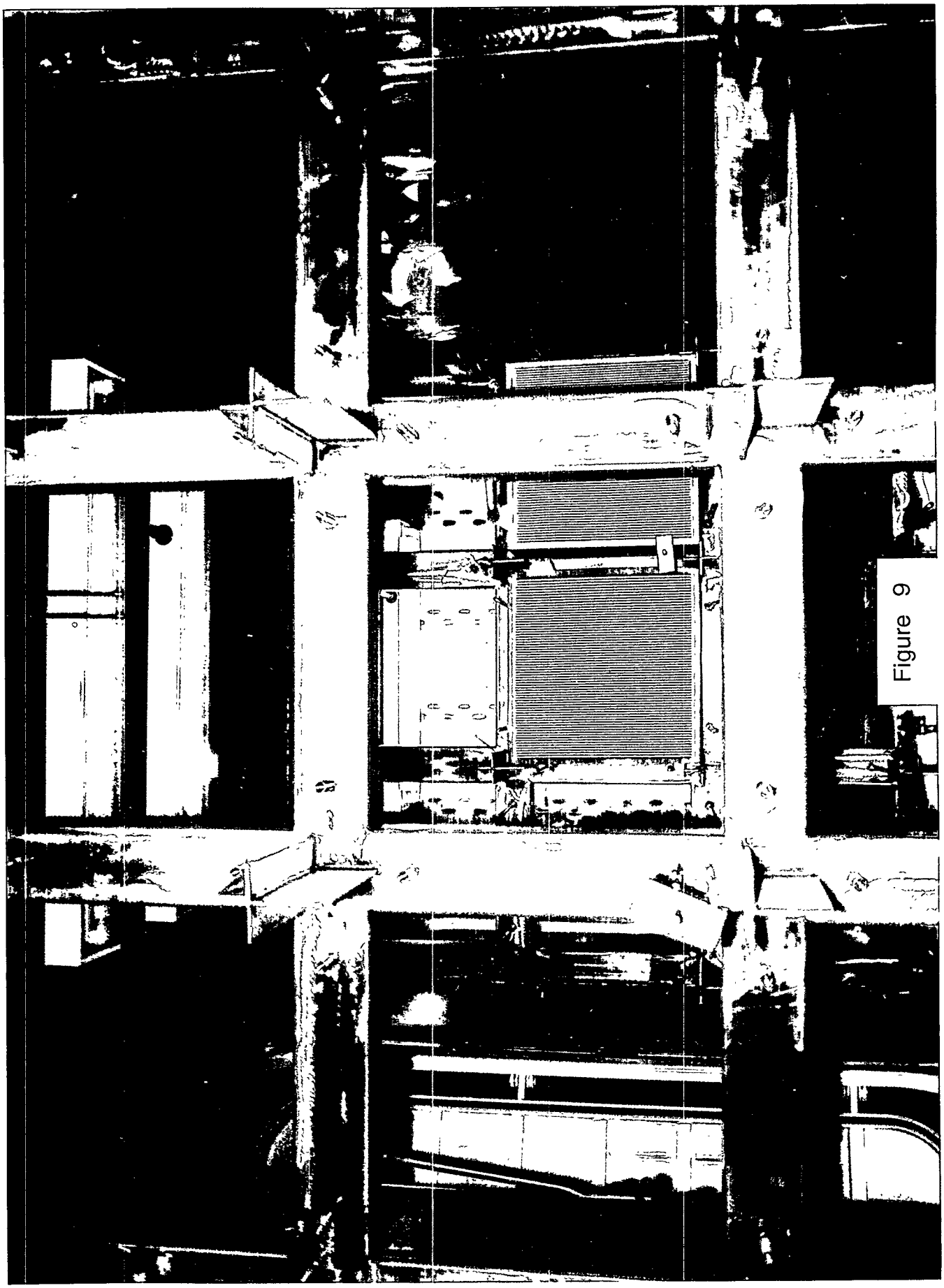


Figure 9

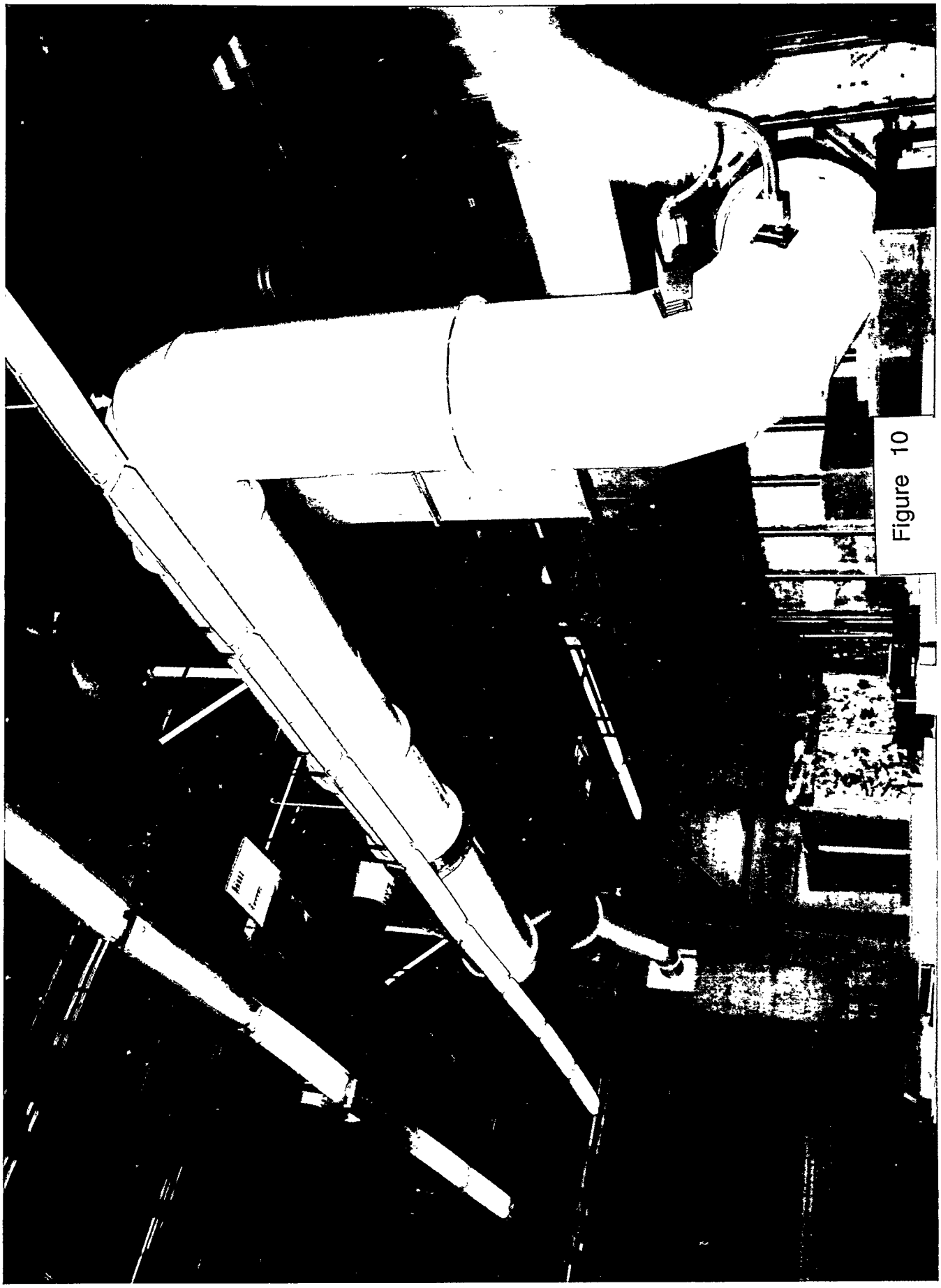


Figure 10

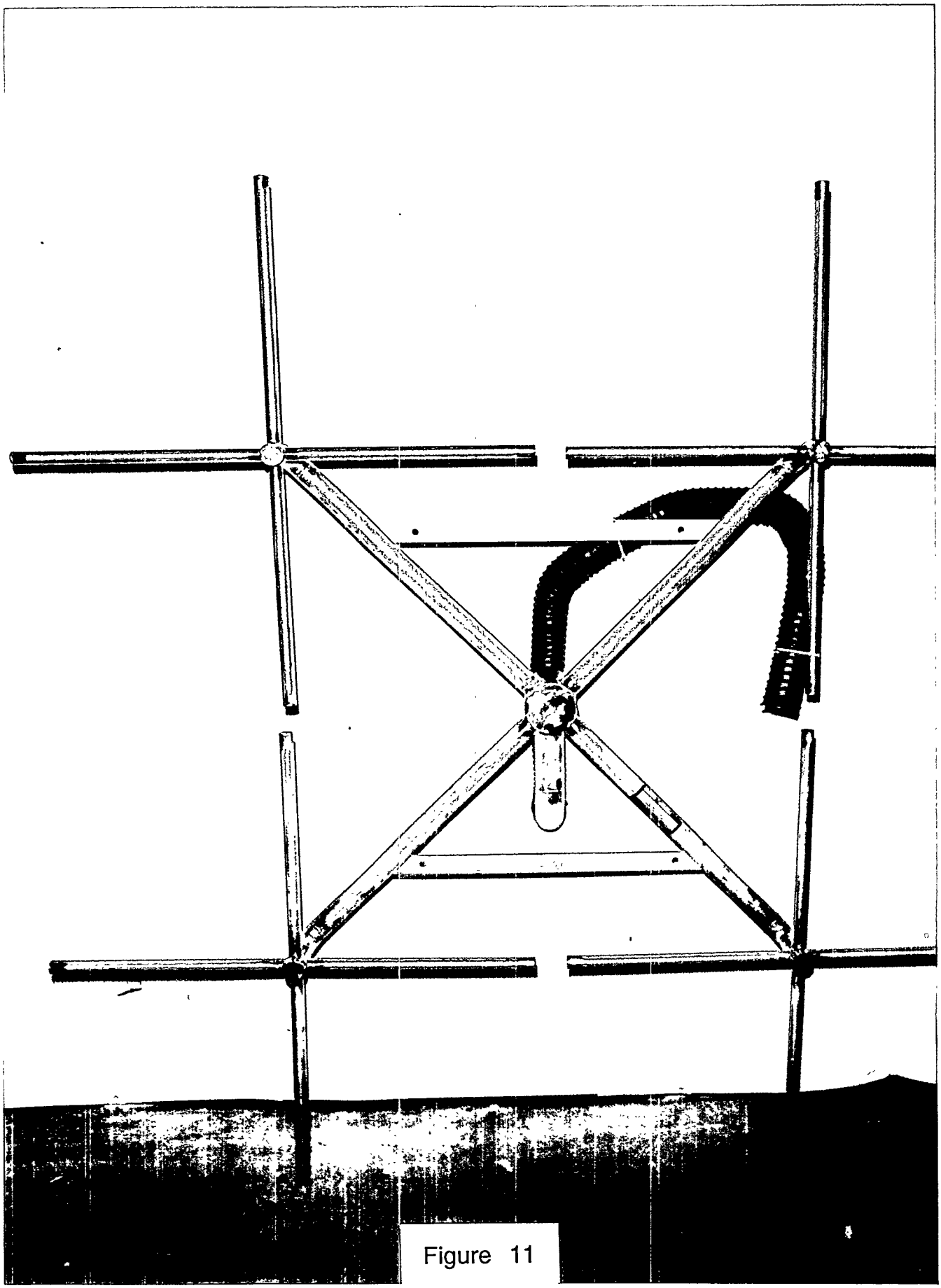


Figure 11

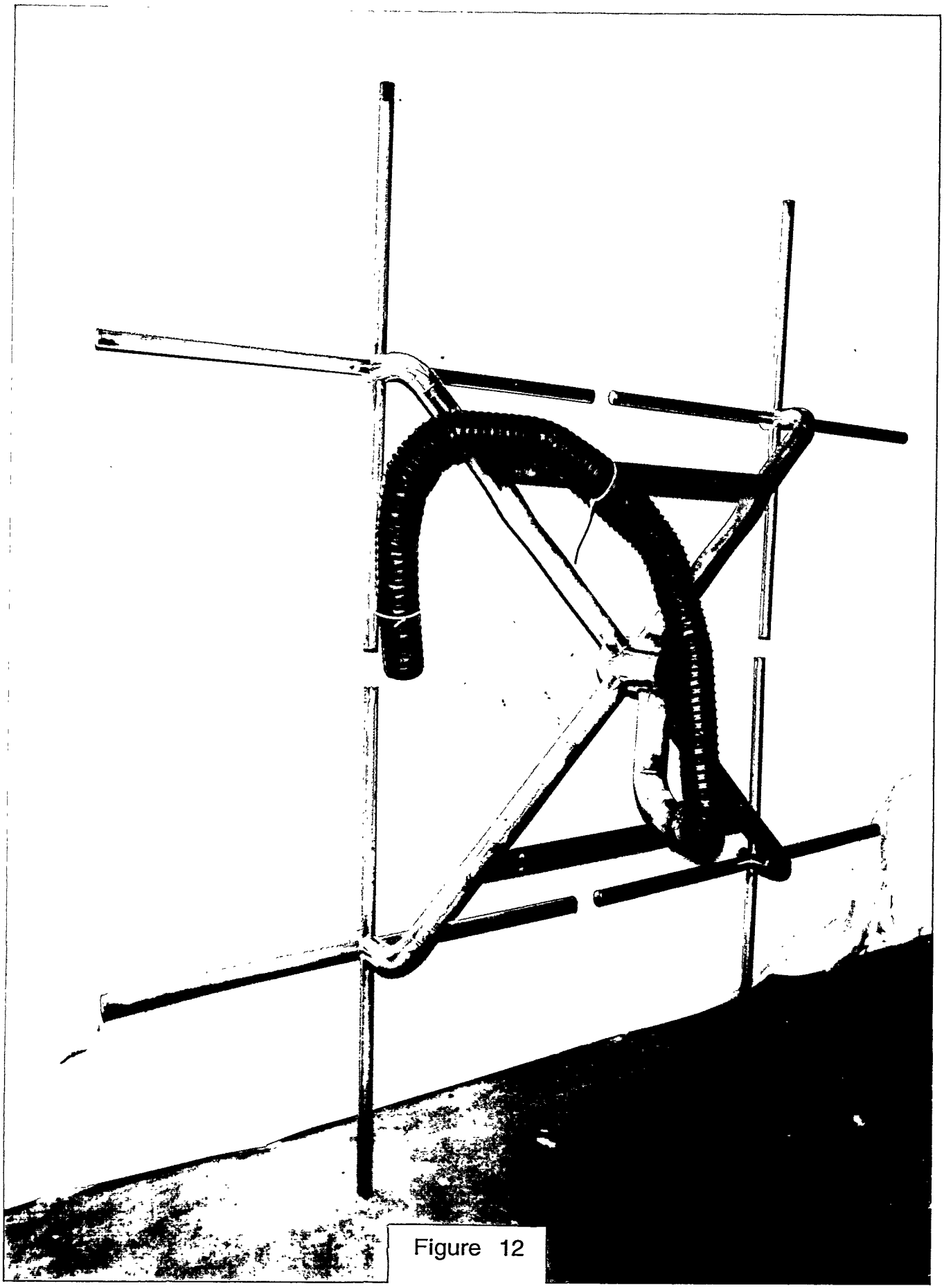


Figure 12

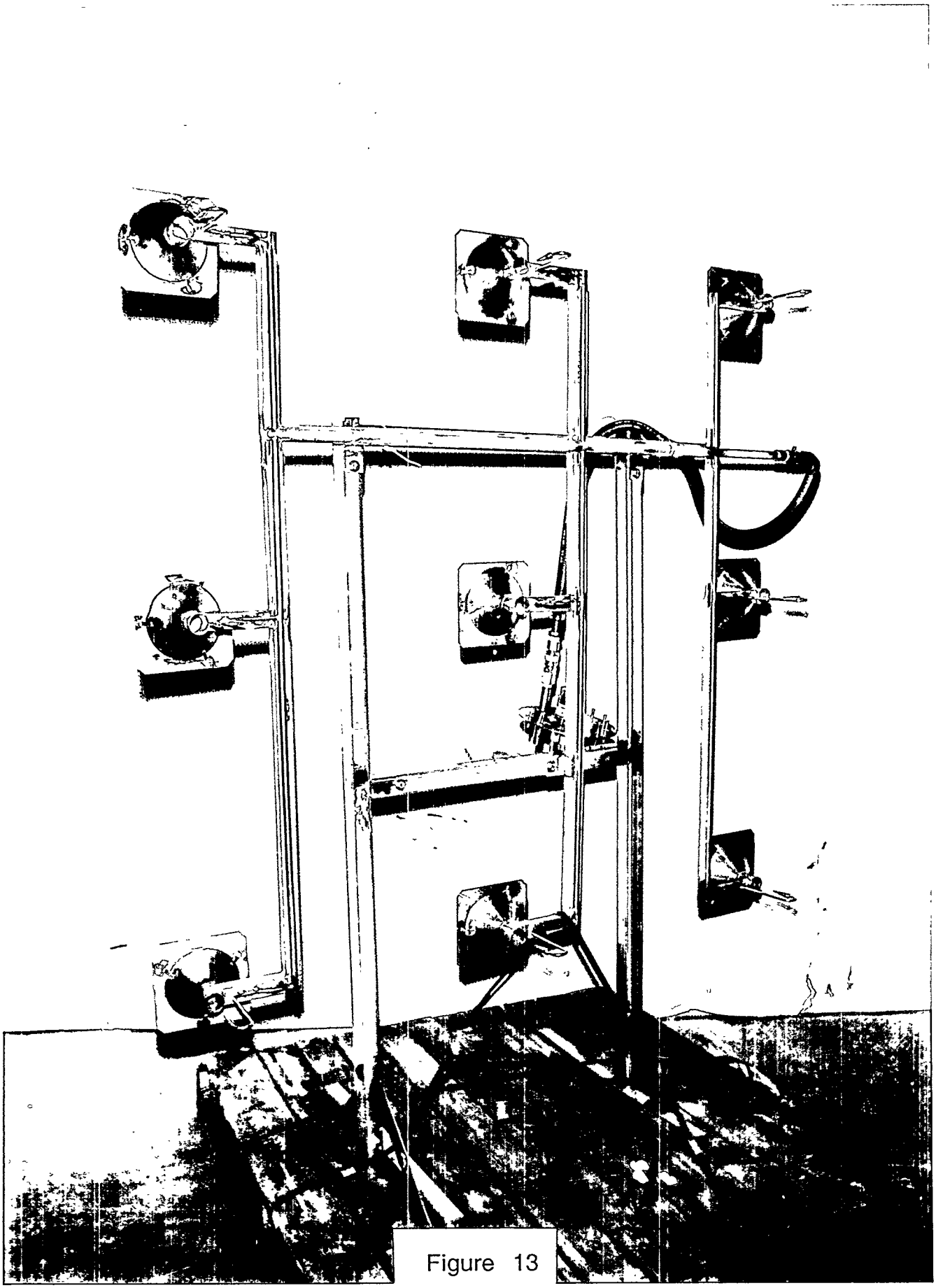


Figure 13

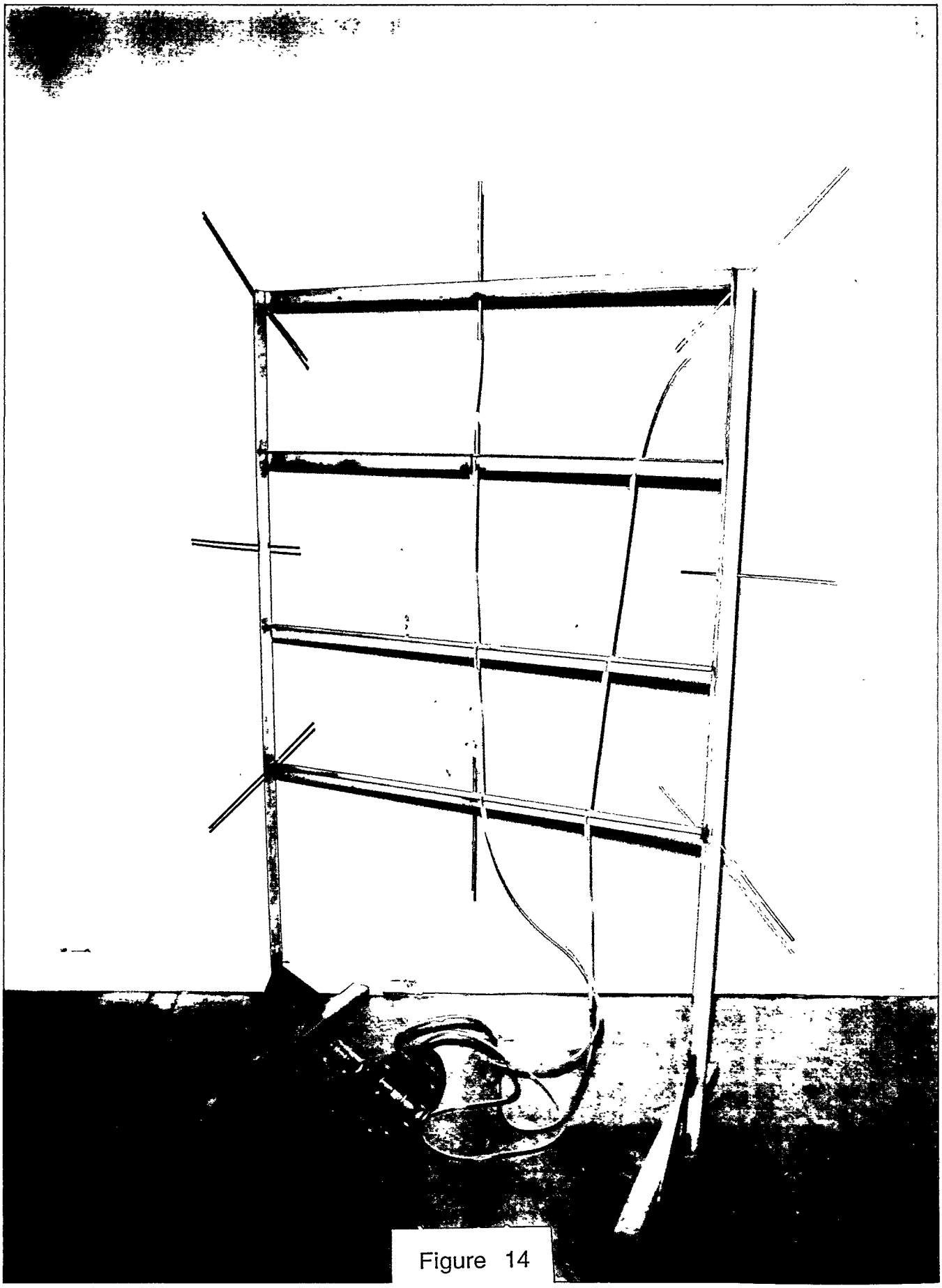


Figure 14



Figure 15

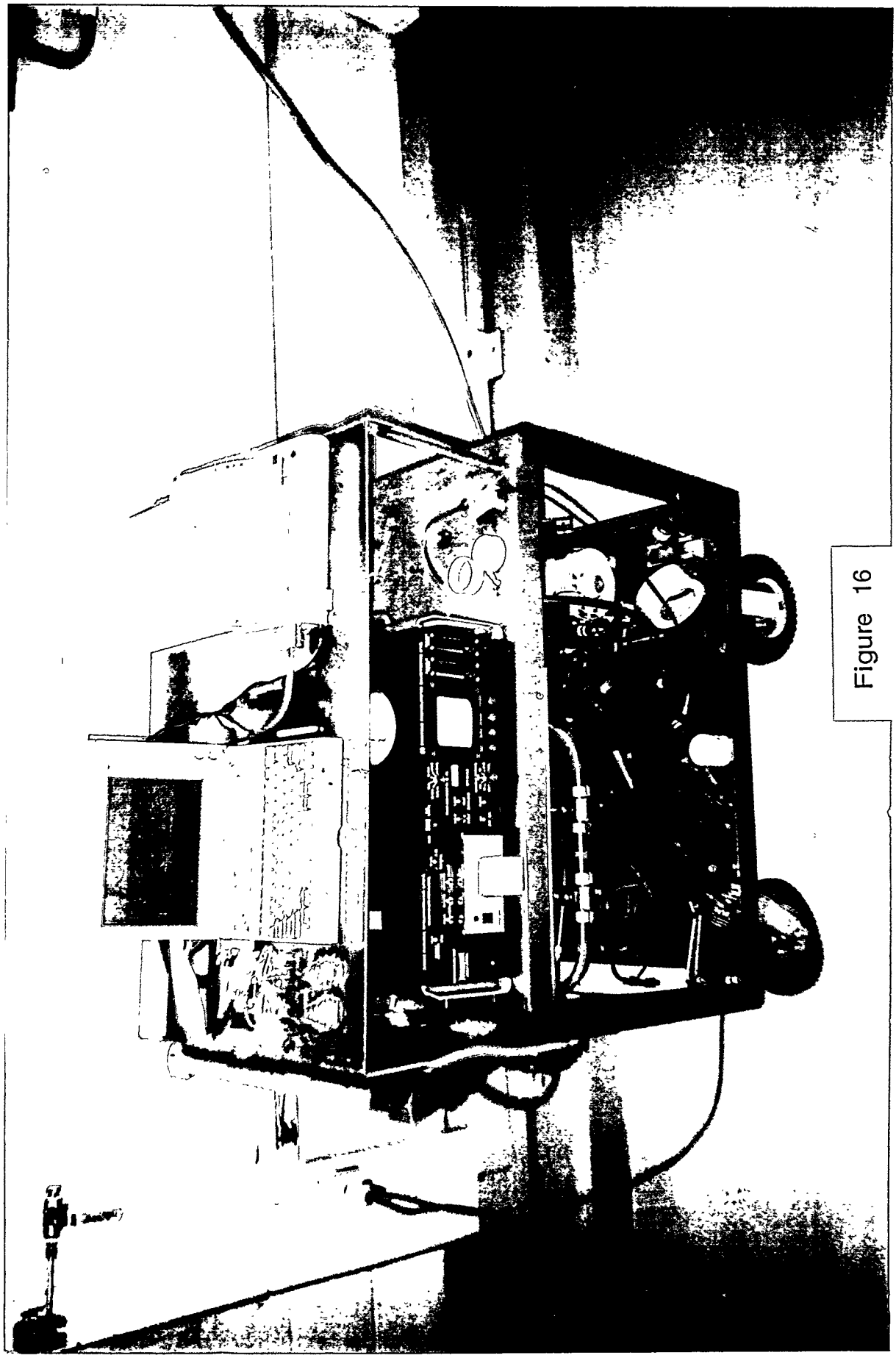


Figure 16

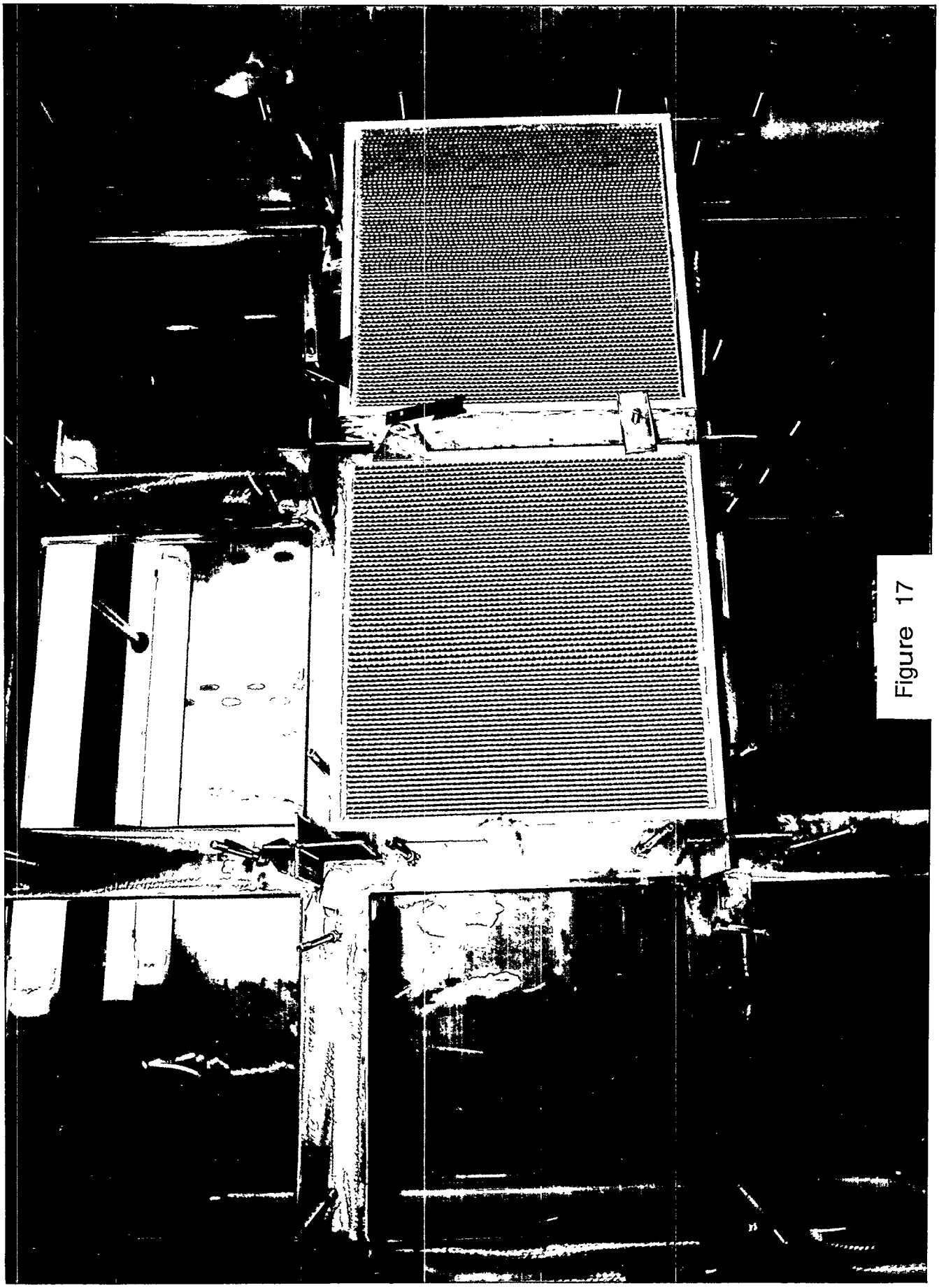


Figure 17

DISCUSSION

FIRST: You have described a very exciting project, and I look forward to hearing a full report at the next Air Cleaning Conference. I have a question regarding the selection of your equipment, did you build it according to the directions in N-509?

FRETTHOLD: These were manifolds that were already built to N-509 specifications and used in other plenums. We contacted a couple of manufacturers and asked them what they had that we could use. Better than trying to reinvent the wheel, we thought we could take what was already out there and analyze it for our use.

FIRST: I guess the diagrams are not as revealing as they might be, because as I looked at them I could not make the connection between what you showed and what is in N-509. But I am, indeed, very happy to learn that they were built according to N-509, because I think questions remain as to how good that design is in practice. Let me make one more comment, it concers the matter you brought up of having one clamp holding two filters. I believe that is not approved according to the Nuclear Air Cleaning Handbook. How did it come about?

FRETTHOLD: I think most of these installations were built during the 50s and there were space restrictions. Engineering tried to cram everything they could into the smallest amount of space with no concern for service requirements. It was a learning process.

FIRST: I had the impression that you had rebuilt it, that is why I made the comment.

JACOX: First, when you bought these manifolds that had been built for some other application, did you get sufficient documentation to show how qualification was proven? Second, the way N-509 is written, the manifolds are system-specific, at least to a very large extent. So I think that part of your research ought to be to see that they are requalified properly. The design of a manifold for testing is as much an art as a science, so they are very specific to a particular type of housing and filter bank. I recommend that you look at this very carefully and also get what documentation you can from whomever you purchased them.

FRETTHOLD: I did ask for documentation and did receive it. I also had the design set up for our array of filters. It was arranged specifically for that plenum, with the idea that it would have the coverage needed. We will be verifying that we do indeed have the coverage needed and looking at which one will give us the best performance. This is a starting point. Then, we will specify modifications. As far as retrofitting other plenums, this one is a typical plenum. We have many approximately that size, with similar air flow patterns, entries, distance between demisters, distance between filter banks, etc. Therefore, we will be looking at the basic pattern, first of all. Then, when we retrofit existing plenums in the hot areas, we will go in, do air flow calculations, and cover all of the requirements. We will try to keep the work to a minimum, this is why we are starting out with something that looks like it will work.

SESSION 17

FILTER TESTING

Thursday: July 28, 1994
Co-Chairmen: W. L. Anderson
J. F. Leonard

NEW PERFORMANCE DATA FOR "EMERY 3002" AND "EMERY 3004," TWO ARMY-APPROVED SAFE MATERIALS TO REPLACE DOP IN MASK AND FILTER TESTING
H. R. Carlon, M. A. Guelta

COMPARISON OF EMERY 3004 AND 3006 CHARACTERISTICS WITH DOP FOR POSSIBLE USE IN HEPA FILTER LEAK TESTS
B. J. Kovach, E. M. Banks, G. Kovács

REVIEW OF IN-PLACE HEPA FILTER TESTING AT SEVERAL DOE FACILITIES
B. V. Mokler, R. C. Scripsick

CLOSING COMMENTS OF SESSION CO-CHAIRMAN ANDERSON

NEW PERFORMANCE DATA FOR "EMERY 3002" AND "EMERY 3004," TWO ARMY-APPROVED SAFE MATERIALS TO REPLACE DOP IN MASK AND FILTER TESTING

Hugh R. Carlon, U.S. Army Fellow, and Mark A. Guelta
Research and Technology Directorate
U.S. Army Edgewood Research, Development and Engineering Center
Aberdeen Proving Ground, MD 21010-5423

Abstract

At the 22nd Conference in Denver, we reported that the U.S. Army Surgeon General (SG) had approved our developmental material "Emery 3004" as a safe replacement for the suspected carcinogen DOP (dioctyl phthalate) in mask and filter testing throughout the Army. Subsequently the SG approved a second, less viscous material, "Emery 3002," for similar applications. We have measured the viscosities and surface tensions of these liquids over a wide range of temperatures, and have initiated liquid breakup studies through Laskin and two-fluid nozzles. New measurements have been carried out with both liquids, e.g. using the ATI, Inc., TDA-4A cold generator to disperse aerosols for which droplet size distributions were measured using the TSI, Inc., Differential Mobility Particle Sizer (DMPS). Among the findings were that Emery 3004 performs very much like DOP in the TDA-4A, with some possible advantages, while Emery 3002 in the TDA-4A produces mean droplet diameters about one-half those of Emery 3004 or DOP. This suggests that Emery 3002 could yield more rigorous filter tests with a smaller consumption of material. New laboratory results will be summarized. Sources of the "Emery" materials will be discussed since the production facility formerly operated by Emery is now run by the Ethyl Corporation and the source products are now known as "Ethylflo 192" (Emery 3002) and "Ethylflo 194" (Emery 3004).

I. Background

The U.S. Army routinely performs 100% quality control testing of filter canisters manufactured for use with field-issue gas masks, and periodic sampling and testing of filters stored in its supply depots. In April, 1986, the U.S. Army's Office of the Surgeon General (OTSG) announced that DOP posed potentially serious health risks to workers, and placed severe restrictions upon testing with it; agencies were also informed that dioctyl sebacate (DOS) would no longer be acceptable as a DOP replacement material, and that similar restrictions would apply for both. These restrictions included occupational exposure monitoring of workers exposed to DOP aerosols and liquid, medical surveillance, issue of personal protective equipment, formal notification to workers of associated risks, and labeling of work areas as "cancer suspect agent areas."

II. Summary of the Successful DOP Replacement Program

The above actions placed severe restrictions upon routine, 100% quality assurance testing of filters and other equipment. For this reason, in 1987 the U.S. Army initiated a detailed study of the problem of finding an acceptable substitute material for DOP that could meet all stan-

dard military test specifications while itself being a non-carcinogen and, ideally, having other attributes including acceptable acute inhalation toxicity, low cost, ready availability, and the ability to replace DOP directly in machines at test installations without retrofit or other modification of these machines.

The Army had used DOP for many decades in non-destructive servicability testing of respirator canisters and protective filters, and in a variety of aerosol penetration studies including mask leakage and face fit. The program initiated in September 1987 to find a safe replacement material for DOP was sponsored by the Product Assurance Directorate (PAD).[1,2] A synthetic lubricant named "Emery 3004," from the class of compounds called poly-alpha olefins (PAOs), was approved by the OTSG on 8 January 1992 for use Army-wide as a safe replacement for DOP in "hot smoke" and "cold smoke" testing machines. Emery 3004 was approved after successfully passing three tiers of mutagenicity testing that included the Ames system assay, the sex-linked recessive lethal test in fruit flies, and the rodent bone marrow micronucleus assay performed with rats.

A less-viscous PAO, "Emery 3002," also was also approved by the OTSG on 24 February 1993 for Army-wide use in cold-smoke applications. The Ethyl Corporation[3] is the primary manufacturer of these PAOs, under the trade names "Ethylflo 162" (repackaged as Emery 3002), and "Ethylflo 164" (repackaged as Emery 3004). These materials are extremely useful DOP replacements. They perform at least as well as DOP in various testing machines. They can replace DOP directly in existing machines without modification. They are inexpensive, readily available, and should continue to remain so in the future. Ethylflo 162/Emery 3002 and Ethylflo 164/Emery 3004 are readily specifiable, non-corrosive, free of natural impurities, thermally and chemically stable, and safe to work with. They are recommended to replace DOP in Army-wide testing as soon as is practicable.

III. Properties of the Army-Approved Materials

With two safe DOP replacement materials now approved by the OTSG for Army-wide use, a choice is available between them for specific applications. As will be seen, Emery 3004/Ethylflo 164 works very well in both "hot smoke" and "cold smoke" machines, while Emery 3002/Ethylflo 162 is recommended only for cold smokes or machines operating at moderate temperatures. This is because 3004/164 has a flash point of 225°C, well above typical "hot pot" operating temperatures of 170-180°C, while 3002/162 has a flash point of only 164°C. Auto-ignition occurs in either material only at much higher temperatures; nevertheless, we recommend that only 3004/164 be used in "hot smoke" machines.

The PAOs have many unique and interesting properties. They represent a family of synthetic lubricating oils that were unknown a few decades ago. Yet they have replaced natural petroleum products in a great variety of applications, primarily because they work better, they are purer, and they are inexpensive.

Because Ethyl Corp. (and thus Emery) PAOs are produced by combining 10-carbon decene molecules, the resulting products do not contain distributions of sequential carbon chain lengths, like those found in common petroleum products, which are difficult to separate by distillation. Instead, each PAO consists of chain lengths in multiples of 10 which, in principle, should allow easier distillation to obtain samples of a single kind of molecule of high purity. Approximate analyses for the PAOs of interest here are

listed in Table 1. The last digit of the "3000" or "160" numbering scheme indicates the viscosity in centistokes of that particular mixture at 100°C.

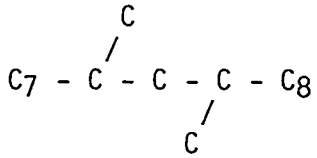
Table 1. Approximate Analyses by Carbon Chain Length for Emery 3002/Ethylflo 162 and Emery 3004/Ethylflo 164.

Ethyl/ Emery Product	Percentages by Carbon Chain Length					
	C20	C30	C40	C50	C60	C70
162/3002	97-99	1.0				
164/3004	0.60	82.1	16.0	1.0	2.0	

Table 1 shows that Ethylflo 164/Emery 3004 is a mixture of of C_{30} and C_{40} polyolefins with molecular weights (MW) of 420 and 560, respectively, and traces of other carbon chain lengths. But the Ethylflo 162/Emery 3002 is nearly all "dimer," i.e., 20-carbon chains each formed from the combination of two decene-1 (10 carbon) molecules.

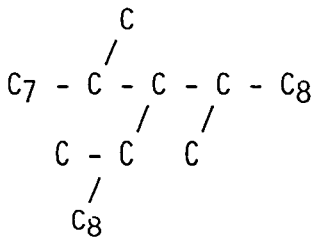
The molecules comprising the PAOs become progressively non-linear with increasing size. Thus, for larger sizes, the molecules increasingly become isomers of one another. The manufacturing process begins with decene-1, a linear hydrocarbon molecule of 10 carbons with one double bond at the first position. The decene-1 is polymerized in the presence of a BF_3 catalyst, hydrogenated to saturate any remaining double bonds, and distilled into fractions (dimer C_{20}, trimer C_{30}, quatrimer C_{40}, ...). These are blended to produce desired product mixtures for various applications, as indicated in Table 1. There are many other PAOs of higher viscosities, which are not shown.

Chemists generally agree that the C_{20} dimer is nearly straight-chained but, because polymerization occurs at the site of the decene-1 double bond, a "bending" occurs such that the dimer molecule has the form:



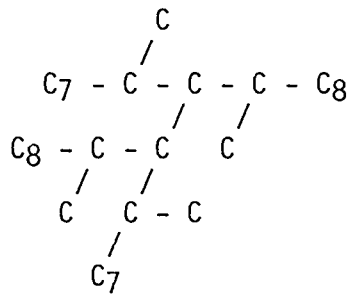
(1)

Structural theories of the PAOs suggest that as the molecules become larger chain attachment occurs at selected sites such that the C_{30} trimer has the form:



(2)

while the C_{40} quatrimer has a form resembling:



(3)

Materials other than decene-1 can be used to make PAOs, but decene-1 gives products with the most desirable range of viscosities for lubricants. To prevent discoloration due to oxidation catalyzed by contact with metals during use, antioxidants can be added to PAOs in concentrations well below one percent.

Some physical properties of the poly-alpha olefins (PAOs) discussed here are given in Table 2.

Table 2. Properties of Poly-Alpha Olefins (PAOs).

Ethyl/ Emery Product	Pour Point, °C	Flash Point, °C	Fire Point, °C	Auto-Ignition Point, °C	Specific Gravity
162/3002	-65	164	178	324	0.80
164/3004	-69	225	250	343	0.82

The viscosities, in centistokes (cSt), of Ethylflo 162/Emery 3002 and Ethylflo 164/Emery 3004 are plotted versus Centigrade temperature in Figure 1. These data were obtained from manufacturer's specification sheets. At 100°C, the viscosity of 162/3002 is 1.80 cSt, approximating the value "2" indicated by the last digits of the product numbers. For 164/3004 at 100°C, the viscosity is 3.90 cSt, approximating "4" in the product numbers.

Table 3 contains manufacturer's[3] viscosity data for the Ethylflo/Emery PAOs, as well as values measured at ERDEC over a limited temperature range for dioctyl phthalate (DOP).

Table 3. Viscosities of PAOs and DOP.

	Viscosity, Centistokes (cSt)		
Temperature, °C	Ethylflo 162/Emery 3002	Ethylflo 164/Emery 3004	DOP
-40	310.0	2460.0	
-18	62.0	341.0	
15			109.80
25			58.36
35			33.90
40	5.54	16.8	
50			17.55
100	1.80	3.90	

Viscosity is an important parameter when liquids are sprayed through two-fluid atomizers to produce fine aerosols such as those used in a variety of laboratory research applications. As temperature is reduced, the ratio of viscosities of 164/162 or 3004/3002 increases. At 100°C this ratio is u_4/u_2 = 3.90/1.80 = 2.17. At 40°C it is 16.8/5.54 = 3.03, and at -18°C the ratio is 341/62.0 = 5.50.

Thus, at lower temperatures, the viscosity of Ethylflo 162/Emery 3002 increases more slowly than that of Ethylflo 164/Emery 3004. If room temperature is taken as 20°C, the viscosity ratio (Figure 1) becomes approximately 35/10 = 3.5. When "cold smokes" are generated by spraying at room temperature, these numbers indicate that 162/3002, being much less viscous than 164/3004, should "break up" more readily to form the smaller droplets of the two liquids. Laskin nozzles, which use pressurized gas injected through jets beneath the surface of the liquid being aerosolized, represent a more complex case.

Another important parameter in liquid breakup through nozzles is surface tension (ST). Values of ST over a limited temperature range were measured for the PAOs and for DOP at ERDEC, with the results shown in Table 4.

The data presented in this report for hot and cold smokes produced from Ethylflo 164/Emery 3004, and for cold smokes (only) produced from Ethylflo 162/Emery 3002 are empirical (experimental). Limited progress has been made in the accurate prediction of droplet size distributions from spray nozzles. A well-

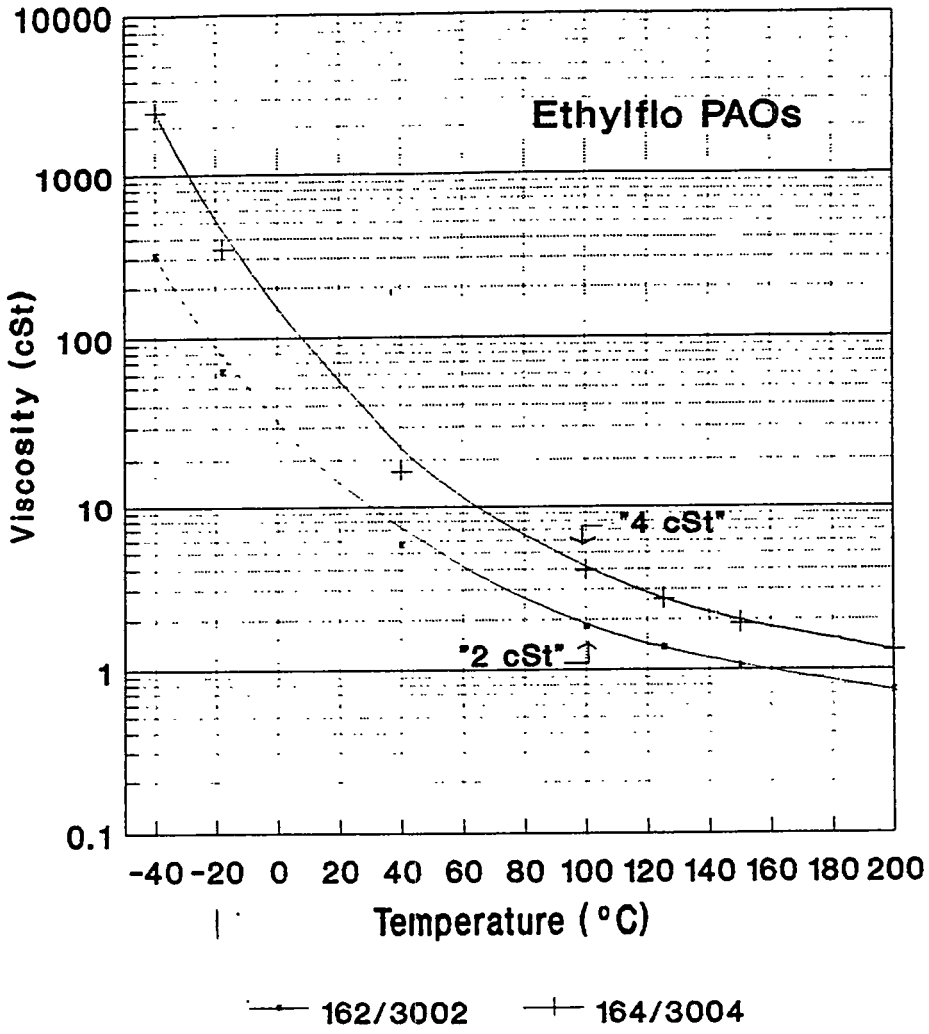


Figure 1. Viscosity (cSt) of PAOs Versus Temperature (°C).

Table 4. Surface Tensions of PAOs and DOP.

Temperature, °C	Surface Tension (dynes/cm) Ethylflo 162/Emery 3002	Ethylflo 164/Emery 3004	DOP
15	28.8	29.9	32.1
25	28.0	29.0	31.3
35	27.1	28.2	30.5
50	26.0	27.0	29.2

known equation for liquid breakup was published in 1939 by Nukiyama and Tanasawa[4], but this was deficient in that it contains no factors relating to nozzle design. Also, except for sonic flow, there is no dependence upon the physical properties of the gas which is used to disperse the liquid. These factors were pointed out by Gretzinger and Marshall[5], who proved their importance.

An empirical equation which answers some of the objections of the Nukiyama-Tanasawa expression, and which appears to fit experimental data more closely, has been given by Kim[6] for convergent type atomizers. The mixture of English and metric units is Kim's:

$$\bar{x}_m = \frac{249}{A^{0.365}\, v_{rel}^{1.144}} \left(\frac{\sigma_l}{\rho_a}\right)^{0.572} \left(\frac{\mu_l^2}{\sigma_l \rho_l}\right)^{0.161} + \frac{1260}{v_{rel}^{0.54}} \left(\frac{\mu_l^2}{\sigma_l \rho_l}\right)^{0.17} \left(\frac{M_l}{M_a}\right)^m \quad (4)$$

where

$\bar{x}_m$ = mass mean diameter (microns)

μ_l = liquid viscosity (centipoises)

ρ_l = liquid density (lb/ft^3)

ρ_a = gas density (lb/ft^3)

σ_l = liquid surface tension (dynes/cm)

v_{rel} = difference in gas and liquid velocity (ft/sec)

A = flow area for the gas ($in.^2$)

M_a and M_l = mass flow rates for gas and liquid respectively

m = 1 for $M_l/M_A > 1/3$ and m = 0.5 for $M_l/M_A < 1/3$

At large gas-to-liquid flow rates, a limiting small particle size is achieved and the value for this particle size is given by the first term on the right side of the equation. At small gas-to-liquid flow rates, particle size will be appreciably larger and controlled primarily by the second term on the right of the equation. Significantly, factors are present which include the design of the nozzle, namely, its area, as well as the density of the gas stream.

It can be seen in Equation 4 that liquid viscosity and surface tension are both well represented. Droplet diameter increases as the 0.572 power of surface tension. Offsetting this are the terms in parentheses taken to the small powers 0.161-0.17, where the square of viscosity is divided by liquid surface tension and density. Thus at face value, the equation appears to support the view that increasing either viscosity or surface tension, or both, will tend to give larger droplet size distributions from sprayers.

IV. Typical Performance Data

There are two general types of smoke generators: those for "hot" smokes used in mask and filter testing, and those for "cold" smokes used in filter testing as well as for tests such as mask face fit or leakage other than through the filter canister.

Often a hot smoke is generated by vaporization/recondensation of a liquid, particularly when a "monodispersed" aerosol is required by test directives. Monodispersity implies a very narrow droplet size distribution. For example, U.S. Army mask canisters must be challenged by smokes of 0.2-0.3 um mean droplet diameter, with a geometric standard deviation equal to or less than 1.3.

There are other designs for hot smoke generators. For example, the Los Alamos National Laboratory (LANL) has built a prototype system[1] called "LAMAPP," an acronym for Los Alamos Monodispersed Aerosol Prototype Penetrometer. This system uses a cold pot to hold the test liquid, which is finely aerosolized by a Laskin nozzle. The aerosol then passes through a heated column and into a mixing chamber. The droplets are vaporized, and recondensation then occurs on very tiny salt (NaCl) nuclei that are much smaller than the resulting, monodispersed, smoke droplets. In our tests, Ethylflo 162/Emery 3002 was found to be the best-performing of all liquids in the LAMAPP system. But in the standard TDA-100 hot pot, hot smoke machine, the best performer was Ethylflo 164/Emery 3004. Due to the vaporization and recondensation temperatures, 162/3002 works poorly in the TDA-100.

Both materials, 162/3002 and 164/3004, can be used in cold smoke applications, and both have attained full approval for use by the U.S. Army's Office of the Surgeon General (OTSG). Thus the choice of one or the other for a given application will include consideration of factors such as operating temperature, size distributions and concentrations obtained from various nozzles, and parameters including flow rates, pressures, and liquid/gas mass flow ratios.

The one restriction has been noted earlier and can be rephrased here: Ethylflo 162/Emery 3002 should not be used in hot pot machines (above 100°C).

V. Hot Smoke (Vaporization/Recondensation) Applications

Until the recent approval from the OTSG of Ethylflo 162/Emery 3002 for cold smoke testing, the authors had concentrated their efforts upon Ethylflo 164/Emery 3004, which had OTSG approval a year earlier, for both hot and cold smoke applications. Thus, many existing data were taken using 164/3004 to generate cold smokes. 164/3004 was intended primarily for hot smoke generation against stringent military specifications.

Great success in both hot and cold smoke testing with Emery 3004/Ethylflo 164 has been achieved at the Hanford (WA) Nuclear Site, which is now operated by the Westinghouse Hanford Company (WHC) for the U.S. Department of Energy (DOE). WHC scientists reported their own test data, which confirmed ERDEC's conclusions that 3004/164 performs at least as well as DOP, and somewhat better than DOS (dioctyl sebacate, another replacement material), in a variety of machines.

WHC had been using DOS in their Q-76, Q-107, and Q-127 (TDA-100) testers because the use of DOP had been curtailed due to its possible carcinogenicity. DOS was causing gumming and other fouling problems not unlike those encountered with corn oil in other machines. DOE was dealing with the problem of deciding whether to convert to a suitable DOP substitute such as one of

the authors' materials, or to replace their machines with new units that incorporate completely different "cold smoke" technology. These would generate smokes with much poorer "monodispersity," i.e., wider particle size distributions, than currently are achievable using hot smoke technology.

WHC reported that "Emery 3004 has several advantages over approved performance testing chemicals, including that it is not considered a carcinogen or suspect carcinogen; therefore, respiratory protection is not required during testing.' 'Additionally, Emery 3004 does not cause buildup on or plugging of the test equipment like DOP or DOS.' 'By reducing the maintenance required on equipment, use of Emery 3004 increases the efficiency of.... operations." Hazardous waste is greatly reduced using 3004/164 with nuclear dust filters. If DOP is used, the waste is classed as both nuclear and carcinogen contaminated; disposal is much more complex.

As a result of this work, DOE Headquarters has notified WHC that they have approved Emery 3004/Ethylflo 164 as a challenge aerosol for the in-place testing of high-efficiency particulate air filter systems, effective September 1992. This should lead swiftly to approval for similar use throughout DOE.

These results and many additional data taken by independent investigators confirm that Ethylflo 164/Emery 3004 performs extremely well in hot smoke applications[1,2]. It possesses virtually every attribute that could be desired in a direct replacement material for DOP and DOS. It is also very effective in cold smoke applications, as will be discussed below.

Ethylflo 162/Emery 3002 outperforms 164/3004 only in the "hybrid" of hot smoke applications mentioned earlier: the LAMAPP machine. In the LAMAPP, because the liquid is first aerosolized using Laskin nozzles in a pot at room temperature, the 162/3002 gives a fine spray of liquid droplets which are then evaporated to allow recondensation upon tiny salt nuclei.

VI. Cold Smoke (Room Temperature) Applications

3004/164 has been more extensively tested in cold smoke applications than has 3002/162, since the former was approved by the U.S. Army's OTSG a year earlier than the latter.

The TSI, Inc., Model 8110 Automated Filter Tester (AFT) is a cold smoke machine that generates an aerosol which is used to challenge a test filter; filter penetration is calculated by a microprocessor. Once a filter or filter canister is installed in the holder, two buttons are pushed and the test runs automatically with results displayed at the conclusion. The AFT has a "low" and a "high" aerosol concentration mode, enabling challenge aerosols to be generated covering the range 15-100 mg/m^3.

In other tests, the TSI Model 3932 Differential Mobility Particle Sizing System (DMPS), which prints a bar chart of particle size distribution versus mobility channel, was utilized. Emery 3004/Ethylflo 164 was found to give an aerosol mass concentration from the Model 8110 that was greater than the DOP yield by 52 percent in the "high" mode and 60 percent in the "low" mode.

Other tests were conducted in the Edgewood Area of Aberdeen Proving Ground using a new prototype machine developed by the Los Alamos National

Laboratory (LANL) which is known as the High Flow Alternative Test System, ATS. Aerosol is generated in a cold pot through a selectable number of Laskin nozzles (to control mass concentration), and is analyzed before (with dilution) and after filter penetration using a Laser Aerosol Spectrometer (LAS). Tests were run with Emersol 875 (isostearic acid, another candidate DOP replacement material), and with Emery 3004/Ethylflo 164 at 1500 CFM air flow rate under various conditions.

The ATS operates on a new principle, in that a broad array of particle (droplet) sizes is generated, but only those size ranges that are of interest need to be monitored to determine filter penetration. The system does, however, print out the full range of penetrations in each size interval or "bin."

The final cold smoke generator that was utilized and will be discussed here is the ATI, Inc., Model TDA-4A. This generator features a total of eight Laskin nozzles, which are selectable by opening or closing a pair of valves so that, e.g., one, three, six, or eight nozzles can be used. The liquid to be aerosolized in placed in the bottom of the housing, submerging the nozzle outlets. Compressed air creates the liquid aerosol by shearing the liquid.

Most cold smoke generators use some configuration of Laskin nozzles, which produce rather wide particle size distributions. To limit the upper size range, thus conserving most of the liquid which is sprayed in droplets too large to be useful for test purposes, some sort of crude pre-sizing is used. All of the generators discussed in this section use a system of pads, filters, impactors, or baffles to limit mean particle sizes of their output aerosols to values typically near 0.7 um.

These resulting size distributions are still much broader than those from hot smoke, "monodispersed" aerosol machines, which produce aerosols having geometric standard deviations (GSDs) of 1.30 or less; these GSDs can be smaller than 1.20 in well-tuned machines. By contrast, cold smoke machines typically produce aerosols with GSDs of 1.60 - 1.90, or even larger values.

Experiments were conducted in our laboratory, using the Model TDA-4A cold smoke aerosol generator, to allow direct comparisons of the performances of of Emery 3002/Ethylflo 162, Emery 3004/Ethylflo 164, and DOP with each other in true cold smoke applications. We have shown[2] that 3004/164 is superior in hot smoke, monodispersed machines, and that 3002/162 works best in the hybrid (cold pot, hot smoke) LAMAPP machine.

The TDA-4A output was connected through a pair of aerosol diluters to a TSI, Inc., Differential Mobility Particle Sizer (DMPS) with a Condensation Nucleus Counter (CNC). Air pressure was 20 psi. The liquids tested were DOP, Ethylflo 162/Emery 3002, and Ethylflo 164/Emery 3004. One Laskin nozzle was operated to limit TDA-4A output and conserve liquids. The air flow from one nozzle at 20 psi was 0.391 liters per minute. Mass concentrations and other data for the three liquids are shown in Table 5.

Table 5. Data for Three Liquids in the Model TDA-4A Aerosol Generator, Operating with One Nozzle at 20 psi Pressurization.

Liquid	Mass Concentration, g/m^3	Geometric Mean Droplet Diameter, um		
		Count	Surface	Volume
Ethylflo 162/Emery 3002	5.03	0.202	0.287	0.333
Ethylflo 164/Emery 3004	5.61	0.374	0.482	0.532
Dioctyl Phthalate (DOP)	4.79	0.384	0.503	0.548

Before looking at the droplet size distributions for aerosols of these three liquids, some interesting observations can be made concerning the data in Table 5. First, note that the mass concentration yield for DOP is the smallest

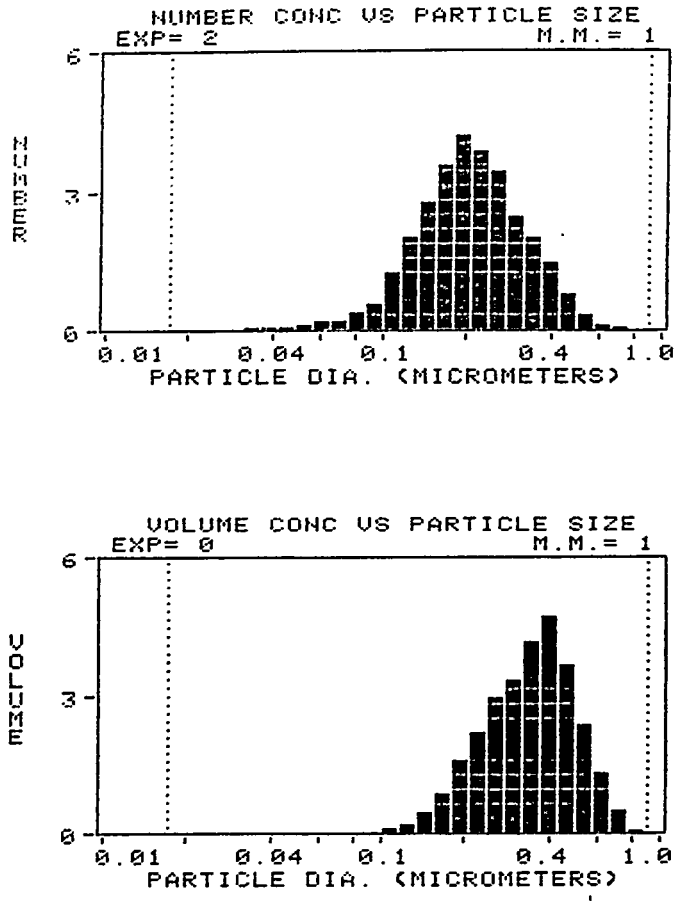


Figure 2. DMPS Particle Size Distributions for Ethylflo 162/Emery 3002.

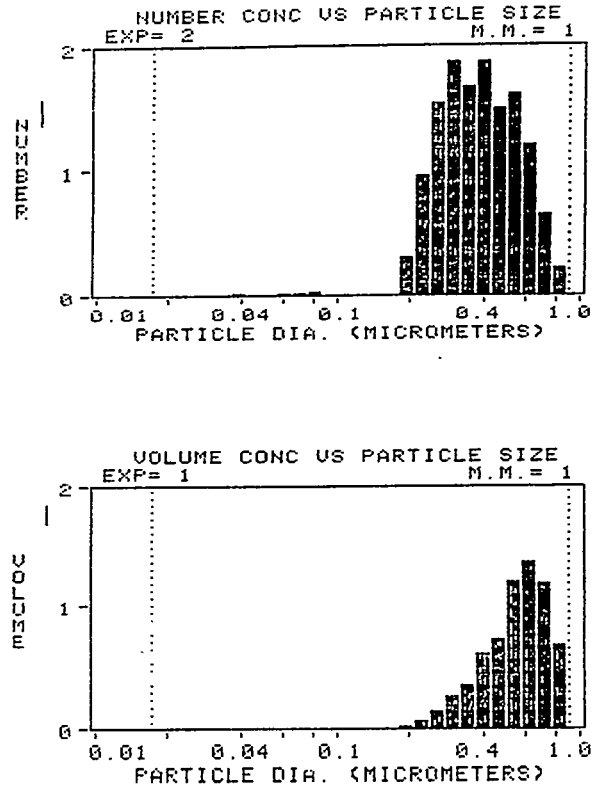


Figure 3. DMPS Particle Size Distributions for Ethylflo 164/Emery 3004.

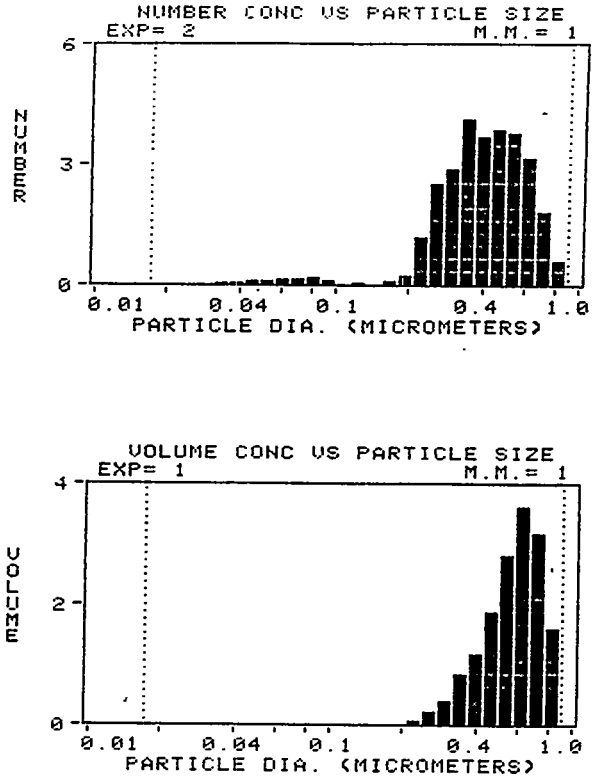


Figure 4. DMPS Particle Size Distributions for Dioctyl Phthalate (DOP).

of the three liquids. It is about 15 percent smaller than that of 164/3004. Second, it is seen that the mean droplet diameters of DOP and 164/3004 are quite similar, with the DOP droplets being the slightly larger ones. But the 162/3002 droplets are by far the smallest of any, and lie in a size range that is nearly optimum for filter penetration. Thus Ethylflo 162/Emery 3002 seems capable of producing much higher counts of more penetrating particles with similar mass flow rates or concentrations to those of Ethylflo 164/Emery 3004 and/or DOP.

Droplet size distributions and other data for aerosols of these three liquids in the TDA-4A are shown in Figures 2-4. Number or count distributions (top) and volume or mass distributions (bottom) are shown in each figure. In general, the number distributions are quite symmetrical, while the volume distributions are skewed toward the right. In Figure 2, both the number and volume peaks are seen to lie far to the left of those in Figures 3 and 4 for 164/3004 and DOP, respectively.

VII. Conclusions

The information and discussions presented in this technical report lead to the following conclusions:

- There are now two liquids approved by the U.S. Army's Office of the Surgeon General (OTSG) for use in mask and filter testing and in general aerosol testing to replace the suspected carcinogen DOP; these are Emery 3004/Ethylflo 164 (approved by OTSG on 8 January 1992), and Emery 3002/Ethylflo 162 (approved by OTSG on 24 February 1993);

- Emery 3002/Ethylflo 162 should be used only in cold pot machines since its flash point of 164 ^{o}C is close to that of temperatures used in "hot pot" machines;

- The physical and chemical properties of both liquids are well known (Tables 1,4, Figure 1);

- The molecular structures of the two liquids are completely different (Table 1); Emery 3002/Ethylflo 162 is almost pure "dimer," its molecules contain 20 carbon atoms; Emery 3004/Ethylflo 162 is a mixture of mostly "trimer" (30 carbons) and some "tetramer" (40 carbons);

- Liquid breakup behavior by submerged Laskin nozzles cannot be expected to be the same as that for two-fluid pneumatic nozzles in gas or air; Equation 4 can be validated only by experiment;

- Emery 3004/Ethylflo 164 has been found to be highly useful by the Westinghouse Hanford Company (WHC), in many kinds of testing and in cost savings realized.

VIII. Recommendations

We recommend that:

- Emery 3004/Ethylflo 164 be used in all hot-pot, hot smoke machines that require replacements for DOP or other materials to meet safety criteria;

- Emery 3002/Ethylflo 162 be used in hybrid cold-pot, hot smoke machines such as the LANL LAMAPP (see text) to meet safety criteria;

- Emery 3004/Ethylflo 164 or Emery 3002/Ethylflo 162 be used in cold-pot, cold smoke machines such as the TSI Model 8110, the LANL HFATS, or the ATI TDA-4A, to meet safety criteria;

- Further studies are needed of the performance of both liquids in a variety of Laskin nozzle and two-fluid pneumatic nozzle systems;

- A mathematical modeling capability needs to be validated to predict droplet size distributions expected from a given nozzle under given operating parameters (e.g., Equation 4). this will require in-house measurements of surface tension versus temperature for Emery 3002/Ethylflo 162 and Emery 3004/Ethylflo 164.

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DISCUSSION

FRANKLIN: In filter testing we know that something like 0.1 μm is much more challenging to a HEPA filter than 0.3 μm. I am wondering if that size would not be prefered as a challenge. I do not know what size particles you are really trying to collect at DOE, whether it is more toward the smaller size particles.

BERGMAN: To answer the question about whether or not a smaller particle size would be better, all of the tests reported here were conducted with cold DOP, which is typically about 0.7-0.8 μm median diameter. None of the tests are valid for filter efficiency certification. These aerosols are used for leak testing. What you are asking is, how do Emery oils perform as substitutes for the hot DOP aerosol. I have no data to be able to say anything about that. My impression is that this report should be viewed as preliminary since there are no filter test data to answer that question.

COMPARISON OF EMERY 3004 AND 3006 CHARACTERISTICS WITH DOP FOR POSSIBLE USE IN HEPA FILTER LEAK TESTS

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Abstract

The particle size distribution, concentration, liquid to aerosol conversion rate and ignition properties of DOP, Emery 3004 and Emery 3006 aerosols generated by the NUCON Aerosol Generators Models SN-10 and DG-F were obtained. Results demonstrate the Emery products are acceptable replacements for DOP in performing leak testing of HEPA filters.

I. Introduction

After DOP (di-octyl-phthalate) was identified by the National Cancer Institute's National Toxicology Program as a carcinogen, as stated in the MSDS[3] various groups initiated a search for an acceptable aerosol substitute[1,2]. Two of the frequently suggested substitutes are polyalphaolefins, or (PAO); trade-names "Emery 3004"[4] and "Emery 3006"[5]. Emery 3004 [polyalphaolefin 4 centiStokes (cSt) viscosity (PAO 4 cSt)] and Emery 3006 (PAO 6 cSt) are non mutagenic, inexpensive and thermally stable liquids. The US. Army's Office of the Surgeon General (OTSG) has approved Emery 3004 for aerosol leak testing of filters and respirators. According to Carlon and Guelta[2], the performance characteristic of the PAOs, as aerosol challenge agents are similar to, or better than DOP. This suggests the possibility of these compounds being used with the NUCON F-1000 series aerosol instruments.

Various NUCON testing procedures were followed using DOP, Emery 3004, and Emery 3006 respectively as aerosol challenge agents. Testing included particle size distribution measurements, aerosol concentration measurements, ignition properties and liquid-to-aerosol conversion rates.

II. Description of Materials and Equipment

DOP was obtained from the Ashland Chemical Company; Emery 3004 and Emery 3006 were obtained from the Henkel Corporation. Aerosols were generated with standard production NUCON Aerosol Generators models SN-10[6] and DG-F[7]. The SN-10 is an air operated "pneumatic" aerosol generator, and the DG-F is a large capacity electrically heated "thermal" aerosol generator. For particle size and distribution analysis a CLIMET

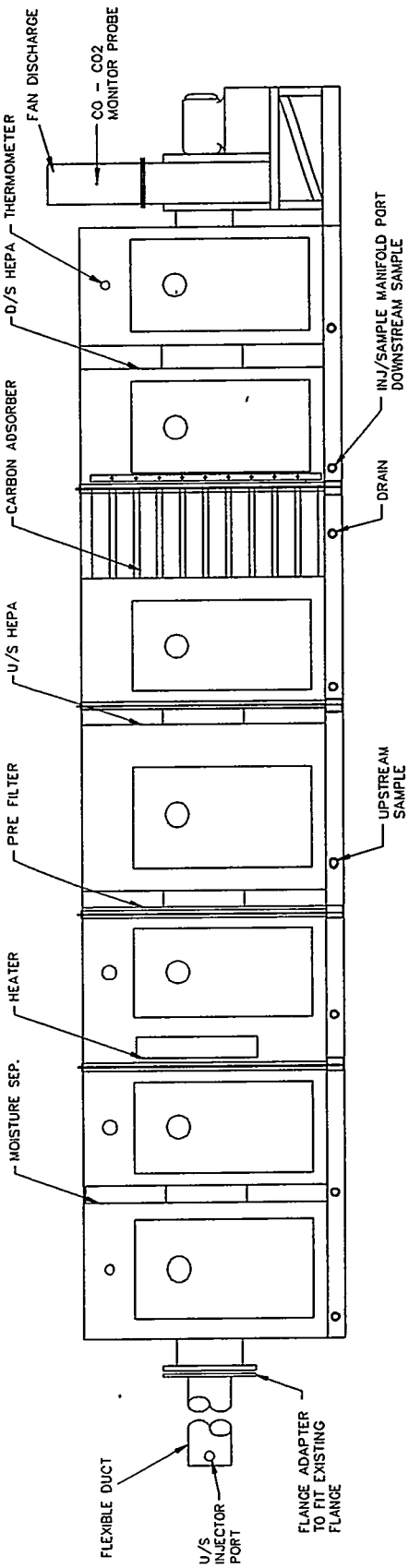


Figure 1. NUCON Air Cleaning Unit (ACU)

model CI-225 detector with a CI-8040 analyzer was used. Tests were performed on a typical air cleaning unit (ACU) consisting of a 6 filter HEPA bank (Figure 1) operating at 5000 SCFM (2.36 m^3/s). Output capacity of the aerosols generated by the SN-10 were measured using a NUCON F-1000-DD-F Aerosol Detector[8].

III. Procedure And Methods

Three identical Model SN-10 Aerosol Generators were filled with DOP, Emery 3004, and Emery 3006 respectively. The generated aerosol was injected into the inlet of the ACU (Figure 1). To provide a diluted sample and prevent the saturation of the CLIMET detection equipment, a bypass around the HEPA filter bank was created using a 1/4 in. (6.35 mm) diameter hole. The bypassed mixture of air and aerosol was then analyzed for particle size distribution and particle count by the CLIMET detection equipment. The test was first performed using DOP as the challenge agent, then repeated using the Emery products. The operating air pressure of the SN-10 generators was maintained at 20 psi for all three tests.

The performance of the thermal aerosol generator (DG-F) was tested using the same ACU arrangement. A consistent aerosol output was maintained by adjusting the generator's built in flow meter to the same value for the three tests. Only one generator was used for all three challenge agents to assure like comparisons. The generator was thoroughly cleaned after each test series.

The aerosol output capacity of NUCON aerosol generator SN-10 was measured according to NUCON procedure 12-67[9] using a NUCON aerosol detector, model DD-F.

Ignition tests were performed only on the thermally generated aerosol produced by the NUCON F-1000-DG generator. With compressed air as the carrier gas, and operating according to the instrument manual, the generated aerosol stream was exposed to an open propane flame placed approximately twelve inches from the generator's aerosol outlet. A spark ignition test was done using a welders' electronic igniter (such as used to ignite oxyacetylene torches). All ignition tests were repeated using bottled nitrogen as the carrier gas.

Liquid to aerosol conversion was performed by using the weight loss method. Each liquid was put into an SN-10 aerosol generator operated for two hours. The beginning weight and the ending weight were compared, and the liquid to aerosol conversion rate was calculated in grams/minute units.

IV. Results

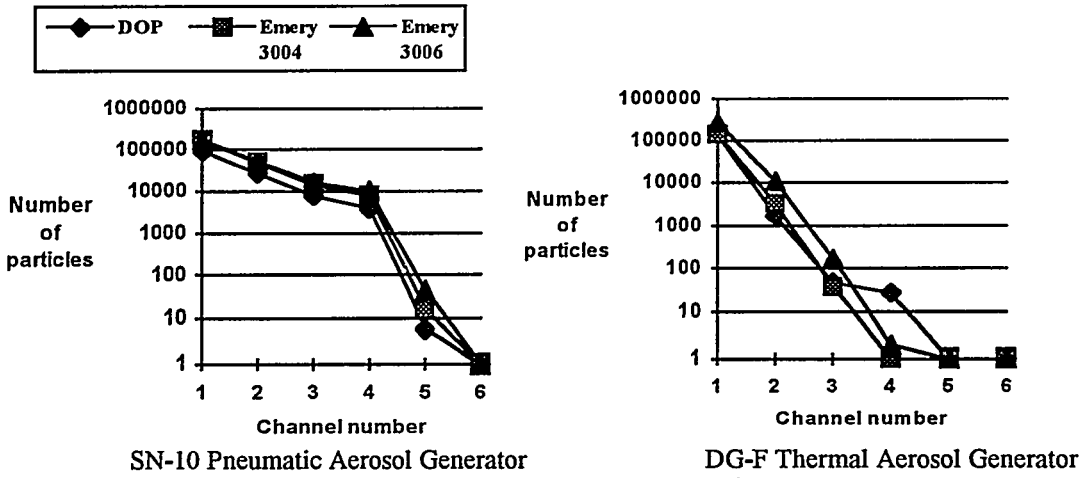
Particle size distribution

The CLIMET aerosol detection/measurement equipment were arranged to sample the air aerosol stream bypassing the first HEPA bank. Before beginning injection, background count readings were obtained. A total of 5 sample counts were made and averaged which then established the average background count.

The CLIMET was operated so the six channels were analyzed simultaneously. The data presented in Table 1 represents the total average count, minus the average background count, per cubic foot of air. A total of 5 samples were taken and averaged.

Table 1 : Particle Size Distribution

Channel number	Particle size micrometer	SN-10						DG-F					
		DOP	%	Emery 3004	%	Emery 3006	%	DOP	%	Emery 3004	%	Emery 3006	%
	Total	131875	100	244419	100	250392	100	138539	100	144071	100	284215	100
1	0.3-0.4	93710	71.1	174600	71.4	172400	68.9	136700	98.7	140800	97.7	273000	96.1
2	0.4-0.5	26340	20.0	47910	19.6	51620	20.6	1765	1.3	3231	2.2	11040	3.9
3	0.5-0.6	7818	5.9	14160	5.8	16430	6.6	46	.030	39	.02	172	.060
4	0.6-0.8	4001	3.0	7733	3.2	9896	2.0	27	.010	1		2	
5	0.8-1.0	6	.004	16	.006	45	.020	1		0		1	
6	1.0-2.0	0		0		1		0		0		0	



SN-10 Pneumatic Aerosol Generator

DG-F Thermal Aerosol Generator

Figure 2. Aerosol Particle Size Distribution

NUCON F-1000-DD-F Aerosol Detector response to DOP and Emery 3004 aerosols

A comparison test was performed to determine if the NUCON aerosol detector equipment possessed similar sensitivity characteristics to the Emery products as compared to DOP. A The upstream injection port was used for aerosol injection (using NUCON SN-10 generator) - while the NUCON DD-F aerosol detector sampled the airstream through the upstream and downstream sample ports respectively (see Figure 1). DOP aerosol measurements were recorded first to establish a baseline. With the SN-10 generator operating at 20 psi, the calibrated DD-F detector was operated to read the upstream sample then the downstream sample respectively.

This method was performed using Emery 3004. The generator was operated at 20 psi and no adjustments were made to the aerosol detector. The data collected are given in Table 2.

Table 2: Output capacity of SN-10

	Upstream concentration		Downstream concentration	
	DOP	Emery 3004	DOP	Emery 3004
Concentration	60	200	0.02	0.06

Ignition test of DOP, Emery 3004 and Emery 3006

This test revealed that no possibility exists of self ignition under normal operating conditions, however; by exposing the generated aerosols of the thermal aerosol generator to an open flame it is possible to ignite the DOP aerosol as well as Emery 3004 and Emery 3006.

To demonstrate this, compressed air was connected to the DG-F thermal generator to perform as the carrier gas. When an aerosol plume was generated it was exposed to an open flame. The source of the fuel for this flame was propane gas generated from a hand held gas torch. The flame produced from the ignition of the aerosol propagated back to approximately one inch from the aerosol outlet of the generator. From that point it expanded outward approximately eighteen inches and engulfed the entire aerosol plume. Past this point the flame was no longer self-supporting. Performing this test and using an electronic spark igniter instead of the open flame did not result in the ignition of the aerosol plume.

The flame size and closeness to the generator's aerosol outlet could be changed (within limits) along this axis by varying the liquid or carrier gas outputs. Identical conditions were observed when this test was repeated using nitrogen as the carrier gas.

Liquid to aerosol conversion rate, using an SN-10 aerosol generator

A comparison test was performed to compare the liquid to aerosol conversion rate of DOP, Emery 3004 and Emery 3006, to establish a baseline. An SN-10 generator was filled with DOP. The filled generator was weighed initially and then operated at 20 psi for 2 hours. At the end of the 2 hours the generator was re weighed. The difference between the beginning and the ending weight was determined as the liquid-aerosol conversion rate, and is reported in units grams/minute. (Table 3). This method was then repeated using Emery 3004 and Emery 3006.

Table 3.: Liquid to aerosol conversion rate of the SN-10 aerosol generator

Conversion rate	DOP	Emery 3004	Emery 3006
grams/minute	0.533	1.921	1.678

V. Evaluation Of Results

Particle size distribution and aerosol output

The data (Figure 2, Table 1) indicate that the particle size distribution curves for the aerosols generated by using Emery 3004 and Emery 3006 are essentially the same as the data generated by using DOP and are well within the ANSI/ASME-N510-75, 80, 89 size distribution requirements (Table 4).

Table 4.: Particle size comparison, by count

ANSI-N510 Particle Size Requirements	SN-10			DG-F		
	DOP	3004	3006	DOP	3004	3006
99% less than 3.0 microns	100%	100%	100%	100%	100%	100%
50% less than 0.7 microns	97%	97%	96%	>99%	>99%	>99%
10% less than 0.4 microns	71%	71%	69%	99%	98%	96%

The output capacity test performed by the SN-10 aerosol generator with Emery 3004 and the DD-F aerosol detector showed an increased amount for generated aerosols of approximately 3:1 compared to DOP under the same operating conditions. This increase is due in part to the lower viscosity of Emery 3004. This was also confirmed with the liquid to aerosol conversion test.

Ignition test of DOP, Emery 3004 and Emery 3006

The ignition temperature of Emery 3006 is higher than that of Emery 3004, therefore it is somewhat harder to ignite. If an ignition occurs, the flame can be extinguished simply by turning off the liquid flow. Tests performed during the original design process of the DG-F, demonstrated that the 1/4 inch outlet tube acts as a flame arrester, therefore it is not necessary to use nitrogen as propelling gas [10].

VI. Conclusions

The test results indicate that Emery 3004 and Emery 3006 are acceptable substitutes for DOP in aerosol generation with NUCON F-1000 instrumentation. The particle size distribution is almost identical for either method of aerosol generation.

Results of the ignition test demonstrated that there is no advantage of the use of nitrogen gas as propellant instead of compressed air with the DG-F.

The thermally generated aerosols have a much larger population (in excess of 96%) of the particles generated at 0.4 micrometer or smaller. The increase in aerosol output of the SN-10 for the Emery products means that a larger air system can be tested without thermally generated aerosols.

VII. References

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4. Emery MSDS Emery 3004 CAS No.: 68649-12-7 Henkel Corporation December 3 1992

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6. Bulletin 9B11-3.91 Model SN-10 Aerosol Generator

7. Bulletin 9B9-12.91 Model F Thermal Aerosol Generator

8. Bulletin 9B7-6.89 Model F Aerosol Detector

9. NUCON Procedure 12-67: In-place Leak Test HEPA Stage

10. Original design for DG-F

DISCUSSION

CROSBY: Why did you use a detector with a cut off of 0.3 μm diameter when a sizeable amount of the aerosol is <0.3 μm in size? It seems to make your aerosol distribution data incomplete.

KOVACH, B.: Correct - it is incomplete. We had no detector for a particle size distribution below 0.3 μm. I wish I had had one at the time of the experiments. However, it makes little or no difference for leak testing.

SIGLI: The in-place DOP test is a leak test. As far as a leak test is concerned, the size distribution obtained with a substitute can be slightly different than with DOP. You will not see any difference if properly measured. If you want to look to efficiency, it is another problem, the size distribution having a direct influence on the result. Have you a comment about your experience using Emery vs. DOP?

KOVACH, B.: I thank you for confirming that the leak test and the efficiency test do not require an identical particle size distribution to accomplish the ANSI-N510 requirement.

REVIEW OF IN-PLACE HEPA FILTER TESTING AT SEVERAL DOE FACILITIES[1]

by

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ABSTRACT

The Office of Nuclear Energy Self-Assessment recently sponsored reviews of HEPA filter systems at several DOE facilities. One aspect emphasized in these reviews was in-place filter testing practices. Although in-place testing was generally performed as required in facility specifications, we noted several areas in which improvements were possible. Examples of some common problems and approaches to their solution will be presented. Areas of suggested improvement include: 1) ensuring the validity of test results; 2) recognizing and quantifying the uncertainty in penetration measurements; 3) expanding the analysis and reporting of test results to provide more than pass/fail information; 4) addressing the special problems of multiple stage systems; and 5) increasing the technical support and training provided in-place testing personnel. Ensuring the validity of test results, for example, requires more careful attention to the operation of test equipment, checking test measurements and system operating parameters for internal consistency, and more attention to documentation of system geometry and operation. Some issues will require additional study before the results can be incorporated into decision making on filter bank testing requirements and performance specifications.

I. INTRODUCTION AND BACKGROUND

In-place testing of HEPA filter installations has been part of the routine operating procedures at facilities supported by the US Atomic Energy Commission and its successor organizations for approximately 30 years. The importance of these tests in ensuring the performance of HEPA filter systems is recognized world-wide. The proper performance of these systems has become increasingly important as allowable emissions and exposures have been progressively lowered. For example, many DOE nuclear air cleaning systems were designed and built when the public dose limit was 500 mrem/year. This limit was reduced to 10 mrem/year in 1989 when DOE facilities came under the jurisdiction of the EPA and may be reduced even further in the future. These changes have not been addressed by comparable changes in in-place testing practices or requirements.

[1] Work performed under U.S. Department of Energy Contract No. W-7405-ENG-36 with primary support from the Office of Nuclear Energy Self Assessment and additional support from the Defense Programs Office.
[2] Environmental Health Sciences, Inc., contractor to Los Alamos National Laboratory.

Partly as a response to the reductions in allowable public exposures, the DOE Office of Nuclear Energy Self-Assessment (formerly NE-80 now NE-1.2, Quality Assurance Staff) initiated a series of selected environment, safety, and health reviews in late FY 1992. These reviews considered air emission control systems at several NE sites and facilities. One aspect of this effort was a survey of HEPA filter installations and testing procedures initiated as a DOE Safety and Health Issue (1). Another aspect was on-site reviews of specific HEPA filtration systems. The emphasis during these reviews was generally on the largest or most critical systems at each NE-supported facility. Consultants from Los Alamos and Sandia National Laboratories and The University of Michigan participated in those reviews concerned with air emission controls and associated monitoring. The reviews were not intended to be comprehensive audits of facilities or programs. Rather, they were to help describe program elements, bring attention to those elements that might benefit from changes or additional effort, and provide information to site personnel. The reviews have been summarized in a formal Los Alamos report (2).

The types and designs of air emission control systems, the extent of in-place filter testing, and testing practices varied widely between the sites. Within a given site, however, policies and practices were generally applied to all facilities regardless of the DOE office providing support and/or direct oversight for a specific facility. Similar observations at many of the sites implied common problems concerning technical and information or policy aspects of in-place filter testing. Many of the filter systems were designed and installed several decades ago and reflect outdated approaches and practices. To the extent the facilities and systems considered in these reviews are representative of those supported by other DOE offices, the observations presented here may also apply to other DOE facilities.

II. HISTORICAL NOTE

In-place testing of HEPA filters has been discussed at almost every Air Cleaning Conference since 1960. The first presentations on the technique and results of in-place testing were made at the Seventh Conference in 1961. The original work drew heavily on the expertise of the Naval Research Laboratory. Since those initial efforts, there has been considerable discussion on both the technical aspects of in-place testing and the role of in-place testing in ensuring that air emission control systems perform adequately. Many of these previous presentations were concerned with the problems of providing a uniform challenge aerosol concentration at the upstream face of a filter bank and obtaining downstream samples that correctly represent the average concentration of aerosol in the filtered air. These problems are still very much with us.

In reviewing the proceedings of previous conferences for papers concerned with in-place filter testing, we realized that many papers of interest were not found when using the key·word (subject) index published as a volume of the Proceedings of the 20th Conference (3). We therefore compiled a list of conference papers we believe are of interest to individuals concerned with in-place HEPA filter testing. The subjects of the listed papers are primarily in the areas of testing methodology, field experience and observations, and discussions of the role of in-place testing. The papers are listed, by conference, in the Appendix. The listing does not include papers concerned primarily with HEPA filter quality, quality assurance testing, or the development of ANSI standards.

III. OBSERVATIONS AND RECOMMENDATIONS

In this paper we will depart from normal technical writing practice and present both the major problems observed during the reviews and our associated recommendations or solutions in a single section. The problems have been divided into two broad categories -- those that are essentially technical in nature and those that can be described roughly as "administrative" or organizational. As is often typical with such categorization, there is some overlap in the details of the problems and their solutions.

A. Technical Problems

1. Injection and Sampling Locations

The air emission control systems included in these reviews were not new. Most of them had been designed and built before publication of the first editions of current design and testing documents. Not surprisingly, these systems do not conform to currently recommended and accepted design and construction practices. In fact, recognition of the problems posed by these systems was part of the impetus behind development of the current recommended practices.

A frequent problem in these older systems is the location of ports for test aerosol injection and sampling. An example of such a system is shown in Figure 1. The test aerosol inlet (not shown in the figure) is more than 20 diameters upstream of the filter plenum inlet. The test aerosol is

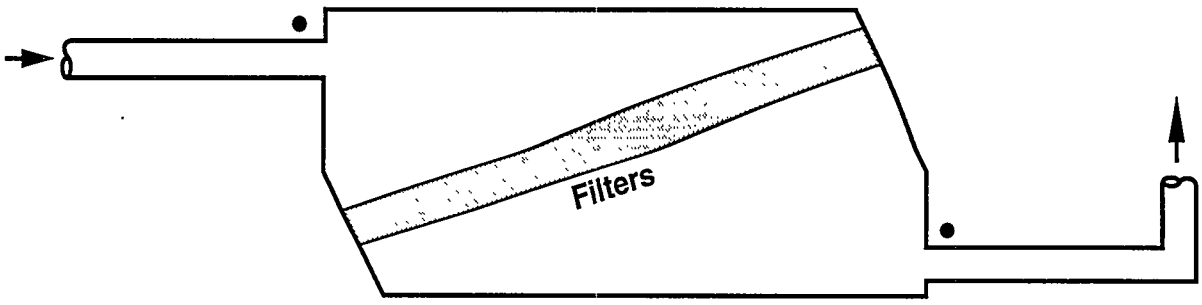


Figure 1. Schematic illustration of a HEPA filter housing with one bank of several filters. The solid dots indicate the aerosol sampling locations for in-place testing.

expected to be well mixed both at the upstream sampling point and over the upstream face of the filter bank. The downstream sampling point is on the duct center line approximately one duct diameter from the abrupt transition from the plenum to the duct. A sample taken at this point is unlikely to be representative of the average aerosol concentration in the discharge. In addition to the length of duct being too short to provide good mixing, the plenum itself appears to have been carefully designed to provide smooth, approximately constant velocity flow. Thus, aerosol penetrating through a leak at the periphery of the bank will not be uniformly mixed into the flow until it has passed through a considerable length of exhaust duct beyond the sampling point. The primary problem with this system is that it is very unlikely the present sampling approach can provide a representative downstream sample.

Poorly located injection and sampling points were encountered in nearly all the emission control systems reviewed. This observation reflects, at least partially, the old design of these systems. It was not clear in any of these cases, however, that consideration had been given to making changes that would correct the deficiencies. The prevailing attitude appeared to be that the injection or

sampling points had been specified and, no matter whether they were recognized as inadequate or not, they would be used without further consideration of their effect on the validity of performance measurements. It seems likely that in some cases the test points are so poorly located that the test data cannot represent the average performance of the filter bank. If the conditions are such that it is unlikely that accurate aerosol measurements can be made, it would be reasonable to consider such systems untestable unless modified.

Correction of these problems will require engineering review of injection and sampling locations and assessment of the degree to which they permit good mixing and representative sampling. If adequate injection and sampling locations are not possible, it will be necessary to assess the influence of test conditions on in-place test data. This assessment should include consideration of the measurement uncertainty introduced by the locations of the available injection and sampling points and other system characteristics. In some cases it may also be necessary to consider alternative approaches for establishing filter bank integrity.

2. Testing Practices

In-place filter testing is a deceptively simple appearing task --even if the problems of uniform challenge aerosol distribution and representative sampling have been recognized. We believe three changes in testing procedures would enhance the reliability of test results. A fourth change is essential if the uncertainty of test results is to be quantified and considered in system evaluation and the establishment of system performance criteria.

The first change in testing practice focuses on the magnitude of the challenge aerosol concentration. During the on-site reviews we found that tests are often performed with an essentially arbitrary challenge concentration. This is unfortunate because it means the concentration may be outside the optimum range of the photometer response characteristics. If the challenge aerosol concentration is too low and the common practice of adjusting the photometer gain to give a full scale reading is followed, excessive gain will unnecessarily increase electronic noise in the downstream reading. On the other hand, too high a challenge concentration will force the photometer beyond its linear range for the challenge concentration and result in an overestimation of the penetration. Aerosol generators should be operated so that challenge concentrations are in the recommended range of 20 to 100 micrograms per liter of a DOP (or suitable alternative material) aerosol with a size distribution that has a count median diameter of approximately 0.25 μm and a geometric standard deviation of approximately 1.5. The generator operating conditions to reach the desired challenge concentration should be a part of the test record for each filtration system.

The second change concerns photometer operation. Modern photometers are equipped with an internal reference or "calibration" light. The manufacturers recommend its use for setting instrument gain rather than the older practice of adjusting gain so that a full scale reading corresponds to the challenge concentration. The recommended practice ensures operation of the photometer electronics at a useful sensitivity without introducing excessive electronic noise. Setting the gain in this way also allows the photometer to be used as an approximately direct reading aerosol concentration instrument. Although operating in this manner will require calculation of filter penetration, the practice provides a simple method of verifying the reasonableness of test results as discussed in the next paragraph.

The third suggested change involves comparison of the photometer measurement of challenge aerosol concentration with the concentration calculated from system air flow rate and generator output. One of the shortcomings of the present approach of only comparing upstream and downstream concentrations is that there is no check on system operation or instrument performance. Using the photometer as a direct reading instrument, however, provides a method of verifying test equipment and system operation. From the air flow rate through a filter system and generator output rate, one can calculate an expected challenge aerosol concentration. This should agree, within some tolerance, with the photometer measured concentration. The tolerance will need to be determined after gaining field experience with the method but initially it seems reasonable to expect the two values to agree within a factor of two. On the other hand, it seems unreasonable to expect routine agreement to better than 10 - 20 per cent. Exact agreement between the observed and calculated concentrations is not the goal. The important point is that it is quite easy to determine that system and test instrument operation are consistent with each other. Some of the examples of inaccurate tests encountered during our reviews would have been avoided if this testing procedure had been in use.

Although it is generally assumed that an air emission control system will be operating normally during an in-place filter test, it is important to verify this. In this way testing personnel will be more certain that no changes have been made since previous tests and that flow and pressure conditions are within the expected range. Configuration and operating conditions, both expected and actual, should be part of the record from all tests. These should be verified on-site before a test is started. The field testing aid described later would be very useful for this purpose and would avoid reliance on the tester's memory of past test results.

The fourth suggested change involves how concentrations are read. Current practice is to observe an analog (or sometimes a digital) display for a short period of time and estimate the "average" reading. There is no attempt to or means of quantifying the variability of the meter readings. Penetration is therefore reported as single number with no indication of the uncertainty in the measurements used in the calculation. A more technically sound approach would be to record several (10 to 15) readings and calculate their mean and standard deviation. Although difficult and time consuming if done manually, the process is simple if done electronically. A device to make the readings and calculations and display the results would not be expensive and could be added to current photometers, many of which already have output connections suitable for use with a data logger or other external recording or display device. The initial use of such statistical information would be to describe temporal uncertainty in reported penetrations. The information may also be useful for monitoring changes in a system or comparing different systems.

3. Knowledge of Systems Being Tested

The depth of knowledge in-place testing personnel have of the systems they test is an example of a problem with both technical and organizational aspects and solutions. Testers frequently appear to simply follow management-provided-test procedures with little concern for understanding or evaluating the applicability of the procedures. This strict conformance to procedures appears, at least in part, to be a consequence of a major emphasis on quickly completing tests according to a tight schedule. In some cases this adherence to procedures had no impact on the testing but this was not always the case. For example, major problems were discovered with the procedures for a filter housing similar to the illustration in Figure 2. This housing is approximately 7 feet tall and 9 feet wide. The inlet duct, entering from the left of the figure, is 12 inches in diameter and is used to introduce the

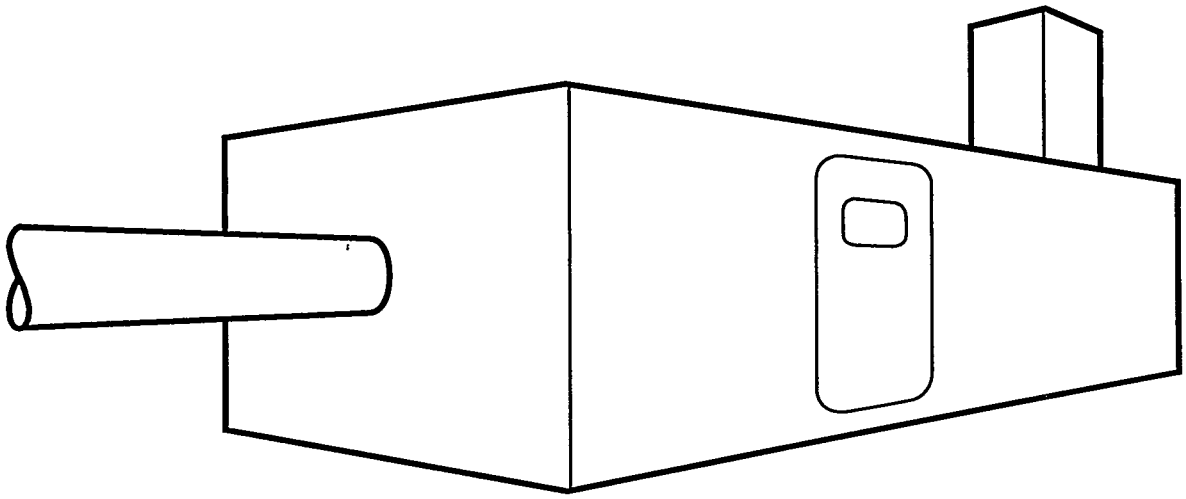


Figure 2. View of a medium size filter HEPA filter housing.

test aerosol. The upstream aerosol sample is taken near the entry to the filter housing. The downstream sample is taken near the top of the short duct on the fan discharge. Both samples are well mixed **BUT** the penetration calculated from them does not represent the performance of the HEPA filters in the housing. Two factors, as shown in the system schematic in Figure 3, cause this. First, the primary inlet, which is approximately 30 inches in diameter, is not apparent to anyone working around the filter housing. Second, there is a bank of prefilters that, in the case encountered during the review, testing personnel were not aware of. One way to ensure that test personnel know about all relevant system features would be to provide them a schematic of the exhaust system. The schematic should also include information on anticipated system operating conditions and test point locations.

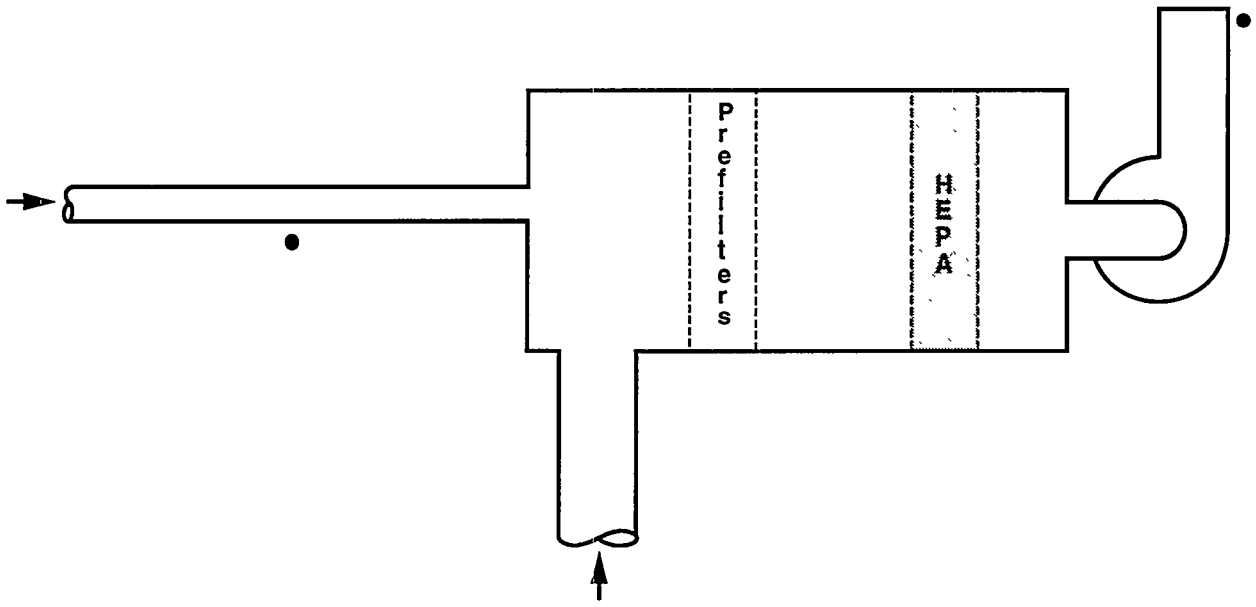


Figure 3. Schematic illustration of the HEPA filter housing shown in Figure 2. The solid dots indicate the aerosol sampling locations for in-place testing.

What is the extent of the engineering background and knowledge it is reasonable to expect of in-place testing personnel? In the situation described above, technical common sense would have shown that the obvious inlet duct is much too small for the filter housing served. Often the job of testing personnel is to follow established procedures and conduct the tests as rapidly as possible. Meeting the testing schedule is considered a primary goal. Under these circumstances, there is little opportunity or expectation that the suitability of the measurement points will be evaluated. This is a reasonable expectation in that such an evaluation should be carried out before routine in-place testing is initiated. This is one example of the need for the application of engineering judgment to the in-place testing of HEPA filter systems.

We believe it would be useful to have a simplified schematic diagram for every system that is to be in-place tested. This schematic should be in a form that can easily be taken into the field for each test. In addition to showing the arrangement of the system, including all inlets and how the different flows combine before entering the filter housing, information on expected operating conditions (flow rates and pressure drops) should be included. Such a diagram will allow testers to verify that system operation is in the expected range before starting a test. The schematic could also serve as the data sheet for recording test results as well as test equipment operating conditions.

4. Test Result Reporting and Analysis

In-place filter testing results appear to be considered only as indicators of the current acceptability of an emission control system. Test results are typically reported to system "owners" in a memo specifying the filter banks tested, the efficiencies (or occasionally the penetrations), and the test date. Although records of test results are maintained, we did not find any indication during these reviews that new data are considered in the context of previous results. It would be a relatively simple additional step to prepare and distribute test results in graphical form as an aid to comparing new results to earlier data and more readily noting trends or sudden changes in performance before specifications are exceeded.

While it seems appropriate to expect those relying on a filter system to control emissions from their facility to watch for trends in test data, we believe the responsibility should rest with the in-place filter testing group. There are three principal reasons for this. First, the testing group is specifically -- and sometime exclusively -- concerned with HEPA filter system performance. Second, what appears to be a change in system performance may be the result of a change in test method or instrument performance. Comparison of current data to previous results provides a further check on test validity beyond that described earlier. Third, in-place filter testing personnel should have a broader perspective on systems and their performance than those operating the processes that rely on the filter systems for control of emissions. This broader perspective gives the testing group the ability to interpret specific test results in the general context of the capabilities and limitations of in-place testing and the influence of air cleaning system design features on test results.

Another aspect of test result reporting and analysis we believe requires attention is that of the reliability of results. The technical aspect of the issue of measurement uncertainty was discussed briefly earlier in the section on testing practices. The influence of different sources of uncertainty is discussed in detail in our recent report (2). System design, construction, and operation can also affect the reliability of penetration measurements. In some instances we found that results were reported that we cannot be certain are representative of the filter banks tested. System designs that exclude provisions for in-place testing can make it impossible to obtain a uniform challenge aerosol concentration or representative samples. The reports for such systems, however, did not qualify or limit the results in any way or distinguish them from those obtained for systems with proper provisions for in-place testing.

5. Multiple-Stage Systems

Filter systems that incorporate more than one stage of HEPA filtration represent a particular problem for in-place testing. The most obvious problem is that many of the older systems were not designed for in-place testing of each filtration stage. Sometimes it is possible to extract a sample after the first stage of filtration but even this may not represent the average concentration of penetrant aerosol. In other cases it is impossible to properly test any of the stages.

One of the less obvious forms of multiple-stage filtration was found during some of the reviews of older systems. The system on which Figures 2 and 3 are based was only one of several systems encountered during the reviews in which prefilters were included in the test measurements for a single stage of HEPA filtration. Such testing is often forced by a filter mounting design that makes it impracticable to properly introduce and sample the challenge aerosol between the pre- and HEPA filters. When this was done, however, we did not observe any recognition in either the penetration requirements or test reports that the filters were being tested in this manner.

The problems of testing multiple-stages of HEPA filters have been discussed by several authors at previous conferences. The discussions and proposed solutions have been primarily technical in nature. Although some of the solutions appear quite practical, during these reviews we encountered several older multiple-stage systems in which the individual stages could not be tested. One obstacle is that of whether the anticipated remaining useful life of a facility warrants the expenditure to install and verify the performance of improved in-place testing features. A second is that of how performance of multiple-stage systems is specified. The specifications often seem to ignore the issue of whether it is even possible to perform a reliable in-place test.

B. Administrative/Organizational Problems

1. Technical Support for In-Place Testing

The numbers and backgrounds of in-place filter testing personnel, as well as the continuity of their testing work, varied widely between the sites visited during these reviews. In-place testing staff sizes range from one or two people performing filter tests less than 10% of the working year to groups with several full-time testers. The educational backgrounds of the testers span a similarly broad range from nondegreed maintenance and laboratory technicians to industrial hygienists with undergraduate technical degrees. Preparation of in-place testing personnel to perform their work generally emphasizes on-the-job training supplemented by attendance at one of the available specialized courses. In all cases the testing personnel made a serious effort to perform their work efficiently and according to the procedures they had been directed to use.

The variety of group sizes and backgrounds also lead to a wide range of capabilities to recognize and deal with unusual or nonideal conditions at the different sites. This range points to one of the shortcomings of most in-place filter testing programs. Although test personnel generally followed specified test procedures, we found that testing was often done without consideration of the reasonableness of the data obtained. This reflects, in part, the tendency to treat measurements as point values and evaluate them only in comparison to a specific pass/fail criterion. It is also indicative of the general emphasis on completing tests quickly and with minimal professional technical involvement with the details of routine in-place testing. Increased professional involvement should help reassert the importance of in-place testing and the need to ensure the validity of test data.

In-place filter testing groups do not always have the staff capability to perform an engineering review and evaluation of the systems they must test. At least one of the larger groups has recently added a professional engineer to its staff but this appears an exception rather than a normal practice. It is clearly out of the question for the smallest groups to have someone with such a background routinely available. Regardless of whether done by a permanent member of the in-place testing group or by an engineer temporarily assigned to the task, review and evaluation of HEPA filter systems is essential for three reasons. First, the review will provide the information necessary to prepare the schematics and basic information needed for the testing aid recommended previously. Second, there is a need to establish whether current testing locations and practices are providing accurate measurements of filter system performance. Third, the evaluation is an essential step in quantifying the influence of system design on measurement uncertainty.

In addition to more emphasis on engineering support for in-place testing personnel, they also should have their own "laboratory" system for testing, calibration, and instrument comparison. This is necessary both as a training aid and as a tool for monitoring instrument performance. This attention to ensuring correct functioning of test instruments will enhance the accuracy of filter system measurements. Availability and use of a system for this type of quality assurance effort was not common at the sites reviewed.

2. Communication/Coordination

The topics considered in this section provide a reminder of the many possible interactions and relations between all the groups specifying, working with, and benefiting from HEPA filters. Each of the topics was encountered during these reviews and, regardless of the positive or negative aspects encountered, emphasizes the need for communication between all those involved with HEPA filters.

Exhaust systems that include HEPA filters are normally used to control a recognized potential hazard identified during an engineering and safety analysis. In one instance, however, we found HEPA filters being used inappropriately. An exhaust system had been installed over a hot work area in a large room to provide local control of the heat and the fumes and vapors that were sometimes released in this area. Sampling showed the exposures of unprotected workers at the hot location were within applicable occupational exposure limits but it was decided to filter the exhaust and "just to be on the safe side" HEPA filters were specified. The filters, of course, have no effect on vapors, perhaps the most serious of the materials entering this exhaust system. There were also no criteria established for filter performance based on the need to control process releases. Nevertheless, the filters, because they are HEPA filters, must be tested annually to satisfy facility requirements. The common penetration limit of 0.05% is applied. Here the lack of communication and coordination has led to the use of HEPA filters to "control" -- sometimes -- an exhaust that is not truly hazardous and the commitment of in-place testing resources to verify the performance of a system without a rational performance specification.

Use of HEPA filters to control laboratory or process exhaust emissions normally includes consideration of the chemical nature of the exhaust stream. The susceptibility of standard HEPA filters to damage by acid aerosols, gases, and vapors is well known to those who routinely work with HEPA filters. The knowledge is not always possessed by those performing experiments or operating processes in areas served by a HEPA filtered exhaust. Thus, we have encountered systems that showed signs of corrosion because of changes in the chemical nature of the emissions being controlled. The changes were made without discussion with those responsible for the exhaust system. Fortunately the problem was observed before there was a serious failure of the filters. The problem will be corrected by the replacement of the present filters with acid resistant ones. Better coordination between the operating groups served by the exhaust system and those knowledgeable about HEPA filters would have prevented this incompatibility and the damage to the filters that might have led to an unacceptable discharge of dangerous materials.

Problems may also arise as the usage of an exhaust system changes and total flow rates are changed or the number, size, and flow rates of specific inlets are changed. Although it is likely that those responsible for maintaining the performance of the HEPA filters will be aware of these changes, it should not be assumed. Perhaps the most serious problem is the influence of changes in flow rate and pattern on the ability to conduct in-place tests that provide a realistic evaluation of filter performance.

Coordination of exhaust system operation -- or nonoperation -- with maintenance and construction activities is also important. Here the problem is the damaging effect of dusts, fumes, and vapors generated during these activities on filter or adsorber capacity. Most operators appear to be aware of the potential problems and some have formal procedures to minimize the effects.

Establishing the necessary schedule for in-place testing appears to have been principally an arbitrary choice of a convenient interval. Ideally testing frequency would be based on the likelihood of HEPA filter failure and the severity of the consequences of such a failure. (Note that failure here refers to leakage and not a catastrophic failure such as fire or blow out.) The possibility of developing a formal approach for converting this ideal to a practical in-place testing schedule is not known. A practical testing schedule must also recognize factors other than physical or risk related ones. Annual testing is a practical minimum frequency, for example, based primarily on the administrative and record keeping convenience of annual tasks. Near the other conceptual extreme, daily testing would

probably present too severe a burden both in terms of testing personnel effort and the loading of the filter medium with test aerosol material. The issue is the need to have a rational, documented assessment for setting the testing frequency that includes consideration of the potential hazard and the effects of testing on filter service life and operation of the facility or process served by the filters. Guidelines for performing such assessments should be developed.

Although it might appear that exhaust air monitoring should be considered in establishing an in-place testing schedule, we do not believe this is appropriate. The two efforts have very different purposes and philosophies. The fundamental purpose of emissions monitoring is to document that emissions are below specified limits. In some respects the monitoring serves as proof to others that a HEPA filter equipped system is controlling emissions. Emission levels above planned limits are certainly cause to reevaluate the controls. However, other independent programs such as in-place testing should also be the first line of defense to ensure air cleaning system performance and minimize the likelihood of excess releases. The effectiveness of air emission controls cannot be determined by air monitoring and must be demonstrated by an independent method. This strategy should ensure that routine emissions are always below specified limits.

3. Recognition of Uncertainty in Performance and Testing Specifications

The issue of uncertainty in penetration measurements has already been mentioned as a technical problem. It is, however, also an administrative one in the sense that developing performance criteria and setting standards involve administrative action. The present approach fails to incorporate recognition of either the uncertainty inherent in the measurement process or the influence of departures from ideal aerosol mixing and sampling conditions. The nearly universal application of a 0.03% or 0.05% measured penetration limit is inappropriate given the wide range of measurement uncertainty that exists for penetration measurements at DOE facilities. At the minimum, it seems reasonable to rewrite filter performance requirements to require that the measured penetration and its experimental uncertainty be such that there is a 95% probability that the actual penetration is less than a particular value. Whether this value is 0.05% or some other value should be based on an analysis of the contributions of system design and construction limitations to the uncertainty of test results.

IV. CONCLUSION

Although in-place testing has been practiced for more than 30 years, we still face many of the same problems encountered three decades ago. Because of the need to demonstrate that air emission control systems can meet increasingly stringent emission restrictions, we need to improve our test procedures and how we use the test results. We must recognize the uncertainty in our measurements in both our procedures and our performance criteria. Many filter systems, especially the older ones, are at best difficult to test and the reliability of test data should be considered in establishing filtration performance requirements.

REFERENCES

(1) O.W. Lowe, Safety and Health Issue No. ISS-92-6, October 22, 1992 and responses from various facilities through January, 1993.

(2) B.V. Mokler and R.C. Scripsick, "High Efficiency Filter Systems - General Observations, 1992-1993," Los Alamos National Laboratory report LA-12763-SR, 1994.

(3) S.D. Jennings (Preparer), "An Index to the 1st through 20th AEC/ERDA/DOE and DOE/NRC Nuclear Air Cleaning Conferences," Volume III of CONF-880822.

APPENDIX

Compilation of Air Cleaning Conference presentations concerned with in-place testing of HEPA filters. The publication number for each proceeding is shown in parentheses; publication was generally in the year following the conference.

Seventh Air Cleaning Conference, Brookhaven National Lab, Oct 10- 12, 1961 (TID-7627)

A.L. Breslin, "Simple Apparatus for the In-Place Testing of Unit Filters" (Title not shown), pp 56-57

J.A. Young, "Evaluation of High-Efficiency Air Filter Systems," pp 84-97

R.W. Schneider, "Procedures for Testing High Performance Filters at ORNL," pp 187-203

B.J. Held, "High Efficiency Filter Program - National Reactor Testing Station," pp 215-219

H.W. Heacock and C.E. Jones, "The Hanford Reactor Confinement Program," pp 284-306

Eighth AEC Air Cleaning Conference, Oak Ridge National Laboratory, Oct 22-25, 1963 (TID-7677)

Round Table -- In-Place Filter Testing, pp 426-438

E.C. Parrish and R.W. Schneider, "In-Place Testing of High- Efficiency Filters at ORNL," pp 484-493

Ninth AEC Air Cleaning Conference, Boston, MA, Sept 13-16, 1966 (CONF-660904)

R.H. Knuth, "An Evaluation of Two Portable Thermal Aerosol Generators for In-Place Filter Testing," pp 763-780

F. Panchuk and W. Rachuk, "Filter Testing Program Atomic Energy of Canada Limited," pp 979-1020

F. Billard and J. Brion, "Testing of Air Cleaning Systems - Testing of the Components - In-Place Tests," pp 1021-1031

H. Flyger and H.C. Rosenbaum, "On-Site Testing of Filters with Special Reference to Solid Aerosols," pp 1032-1062

IAEA Symposium - Treatment of Airborne Radioactive Wastes, New York City, NY, Aug 26-30, 1968

E.C. Parrish and R.W. Schneider, "Review of Inspection and Testing of Installed High-Efficiency Particulate Air Filters at ORNL," pp 243-264

Twelfth Air Cleaning Conference, Oak Ridge, TN, Aug 28-31, 1972 (CONF-720823)

B.J. Grady, Jr and K.H. Henry, "In-Place Efficiency Tests of a Large Scale Ventilation Exhaust Filter," pp 646-654

A.G. Rolie and F.E. Adley, "High Capacity DOP Aerosol Generator," pp 718-730

Thirteenth Air Cleaning Conference, San Francisco, CA, Aug 12-15, 1974 (CONF-740807)
A. O'Nan, R.P. Williams, and J.M. Goldsmith, "Reliability and Testing Considerations in the Design of Nuclear Reactor Filtration Systems," pp 30-40

F.J. Linck and J.A. Geer, "In-Place Testing of Multiple Stage HEPA Filter Plenums," pp 526-539

Fifteenth Air Cleaning Conference, Boston, MA, Aug 7-10, 1978 (CONF-780819)
B.G. Schuster and D.J. Osetek, "In Situ Testing of Tandem HEPA Filter Installations with a Laser Single Particle Spectrometer System," pp 838-847

L.G. Musen, "A Remote Sampling System for HEPA (1500 cfm) In- Place Filters with a Fluid (Channel) Seal System," pp 987-997

Sixteenth Air Cleaning Conference, San Diego, CA, Oct 20-23, 1980 (CONF-801038)
J. Dupoux, "In Situ Measurement of the Efficiency of Filtration Installation in the Nuclear Industry by the Soda-Fluorescein (Uranin) Aerosol Method - AFNOR Standard NFX 44.011," pp 17-34

M. Marshall and D.C. Stevens, "A Comparative Study of In-Situ Filter Test Methods," pp 35-55

L.P. Murphy, S.J. Fernandez, and B.G. Motes, "Comparison of HEPA Filter Test Methods in Corrosive Environments," pp 67-85

J.C. Elder, T.G. Kyle, M.I. Tillery, and H.J. Ettinger, "In-Place Testing of Tandem HEPA Filter Stages Using Fluorescent Aerosols," pp 86-103

M.W. First and H. Gilbert, "Aerosol Filtration," pp 638-666

Seventeenth Air Cleaning Conference, Denver, CO, Aug 2-5, 1982 (CONF-820806)
J. Furrer and A. Linek, "A New Method of Determining the Overall Particle Decontamination Factor for Multiple Offgas Cleaning Components in Reprocessing Plants," pp 576-590

M. Dupoux, "In-Situ Control of Filtration Systems in France: 5 Years Experience," pp 591-601
R.L. Herman, "In-Place HEPA Filter Aerosol Test System," pp 847- 866

F.A. Hohorst and S.J. Fernandez, "In-Place Realtime HEPA Filter Test Method," pp 867-881

W.D. Hanson, "DOP Testing HEPA Filter Banks in Series," pp 882- 894

Eighteenth Air Cleaning Conference, Baltimore, MD, Aug 12-16, 1984 (CONF-840806)
K.N. Kirchner, C.M. Johnson, J.J. Lucerna, and R.L. Barnett, "In- Situ Continuous Scanning High Efficiency Particulate Air (HEPA) Filter Monitoring System," pp 277-298

J.P. Ortiz, "In-Place Testing of Multiple Stage Filter Systems without Disruption of Plant Operation in the Plutonium Facility at Los Alamos," pp 299-310

L. Dauber, J. Barnes, and W. Appel, "Two-Detector Dioctylphthalate (DOP) Filter Testing Method and Statistical Interpretation of Data," pp 1399-1416

Nineteenth Air Cleaning Conference, Seattle, WA, Aug 18-21, 1986 (CONF-860820)
J.W. Jacox, "Review of Current Halide and Aerosol Leak Testing Methods of Nuclear Air Treatment Systems, Components and Banks," pp 486-510

Twentieth Air Cleaning Conference, Boston, MA, Aug 22-25, 1988 (CONF-880822)
J.R. Edwards, "In-Place Testing of Close-Coupled Redundant Series Filters," pp 1094-1098

G.Q. Kirk, C.E. Childress, and D.D. Schmoyer, "Methods of Testing HEPA Filters with Short Upstream Approaches," pp 1210-1215

D. Loughborough, "Development of Filter Systems: Injection Systems and Mixing Device Evaluation," pp 1216-1230

Twenty First Air Cleaning Conference, San Diego, CA, Aug 13-16, 1990 (CONF-900813)
Ph. Cassette, N. Dupoux, and J.C. Laborde, "New Injection System for a Short Mixing of Test Aerosols and Tracers Inside Ventilation Ducts," pp 80-94

D. Loughborough, D. Morris, and S. Capon, "Development of Filter Systems - Part 2 -Injection Systems and Multi-Point Sampler Evaluation," pp 139-154

Panel: How to Use N510 Testing Methods and Acceptance Criteria for Air Treatment Systems Not Constructed According to N509, pp 317-342

A.E. Dunbar, Jr, J.R. Edwards, and K.W. Heffley, "Progress Report on the In-Place Acceptance Tests of Walk-In Plenum Filter Housings with Close Couple Redundant Series Filters," pp 360-365

Twenty Second Air Cleaning Conference, Denver, CO, Aug 24-27, 1992 (CONF-920823)
R.C. Parker, M. Marshall, and R.B. Bosley, "A New Method of In-Situ Filter Testing Using Pulses of Aerosol and Photometric Detection with Computer Control," pp 154-162

C. Costigan and D Loughborough, "Development of the Quickmix Injector for In-Situ Filter Testing," pp 163-176

J.D. Paul, "Effects of Filter Housing and Ductwork Configuration on Air Flow Uniformity Inside Air Cleaning Filter Housings," pp 205-210

DISCUSSION

ANON: You mentioned the need to consider test equipment accuracy. Is there a method to incorporate inaccuracy into results which is commonly used?

MOKLER: Uncertainty - or inaccuracy - of the results is presently not considered in making or performing HEPA filter penetration measurements. As noted in the paper, obtaining the average and standard deviation of a series of concentration measurements would require a simple addition to or modification of existing photometers. Standard error propagation calculations could then be used to quantify the uncertainty of the penetration. A more complete description of the effect of measurement uncertainty and nonuniform concentration and flow distributions on the uncertainty of calculated penetrations is available in the Los Alamos report referenced in the paper.

BERGMAN: Is there a technical report available that describes the result of the in-place filter test survey? The present report is too general and does not provide the tester with useful information. How do the different groups compare with the N510 requirements? Do the sites comply and how far off are they? Also, it is important that an in-place test accommodate production schedules and not turn into a research project. A users group should be formed with research support, and not have a research technique forced on the users.

MOKLER: The report summarizing the survey observations is referenced in the paper. Facility specific observations and recommendations, which would be excess detail here, were included in the reports to each site surveyed. As noted in the paper, the majority of the reviewed systems predate N509-1076 and compliance with any version of N510 is inappropriate and often impossible. An in-place testing users group has been considered as a potentially useful way of sharing and disseminating the experience of the different testing groups.

HOLTORP: What are your views on including the prefilter in with the HEPA filter test provided this is done consistently within a particular system?

MOKLER: No problem, as long as the procedures and test report do not give the impression that just the HEPA bank is being tested.

HOLTORP: Can you give a general example of an "unnecessary" HEPA filter installation?

MOKLER: Exhaust from a room with a low and occupationally acceptable exposure level that passes through a HEPA filter is an example of an unnecessary installation.

DYMENT: Do the regulatory/licensing authorities need to satisfy themselves of the technical viability of test procedures in order to issue or maintain a license?

MOKLER: Because my experience is all with DOE, I haven't had to deal with licensing situations.

CLOSING COMMENTS OF SESSION CO-CHAIRMAN ANDERSON

The first two papers in our session this morning dealt with the replacement of di-octyl phthalate (DOP) with a poly alpha olifin (PAO) in the QA/QC area and in-place testing. This action was deemed necessary when safety factors questioned the use of DOP in the plant environments. The reported efforts have determined that PAO can be substituted for DOP and is acceptable from the aerosol generation and safety standpoint. Because of viscosity similarities, Emery 3004 (Ethylflo 194) appears to be the PAO of choice. It should be pointed out, however, that the PAO selected is not a direct drop-in replacement for DOP. Because of vapor pressure differences, a temperature adjustment for the "hot" generators must be made to maintain challenge concentration levels. Since the densities of the two materials are different, one can expect differences in the penetration/velocity and penetration/size profiles. At identical test conditions, a difference in absolute penetration and maximum penetrating size will be observed for standard media. The refractive index of the materials is different, thus influencing the light scattering response of the photometer; it is surmised that separate photometer calibrations will be required for the challenge aerosol concentration. Of primary concern is that to preserve the 50 years of DOP performance data, some relationship of the new PAO data must be found so that the PAO information can be used as an extension to the existing DOP database.

The third paper reviewed in-place HEPA filter testing procedures at several DOE sites. Areas of potential concern were identified and possible changes/improvements suggested. This paper provides field support to a paper presented by Mr. Leonard at an earlier session of this conference where he addressed numerous problems in the filtration area and gave action items for their solution.

We would like to thank the presenters for their excellent papers and the attendees for your interest as reflected by your rapt attention and inquisitive questions.

SESSION 18

PANEL SESSION: VENTILATION AND AIR CLEANING DESIGN AND APPLICATION FOR CONTROL OF RADIOACTIVE CONTAMINANTS DURING DECONTAMINATION AND DISMANTLEMENT OPERATIONS

Thursday: July 28, 1994
Co-Chairmen: J. E. Williams III
R. Palmer

Panel Members: D. Edwards
R. D. McCullough
K. Tschaenn

OPENING COMMENTS OF SESSION CO-CHAIRMAN WILLIAMS

VENTILATION SYSTEM DESIGN FOR CONTROL OF RADIOACTIVE AIRBORNE PARTICULATES DURING THE DECONTAMINATION AND DISMANTLEMENT (D&D) OF THE PLANT ONE ORE SILOS

J. E. Williams

PANEL DISCUSSION

OPENING COMMENTS OF SESSION CO-CHAIRMAN WILLIAMS

Good morning. My name is James E. Williams, I am with Parsons Environmental Services, Inc. We are an architect-engineering firm under contract to Fermco, manager of the Frenald project. The panel, from Fermco, is here to talk about ventilation system design for control of radioactive airborne particulates during the decontamination and dismantlement of the plant one ore silos.

Ventilation System Design for Control of Radioactive Airborne Particulates During the Decontamination and Dismantlement (D&D) of the Plant One Ore Silos

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ABSTRACT

Fernald Environmental Management Project (FEMP) is owned by the Department Of Energy (DOE) and was used for the processing of uranium metal products for the nations' defense programs. The facility is a 1,050-acre site located in southwest Ohio. In July 1989, production was discontinued. The Fernald site has completed its site wide Remedial Investigation/Feasibility Study (RI/FS) which is pursuant to the Comprehensive Environmental Responsibility Compensation and Liability Act (CERCLA) and the Amended Consent Agreement between the US EPA and the DOE. In the RI/FS, a variety of response actions where identified for various Operable Unit (OU) areas at the FEMP. Of the many response actions in progress at the FEMP, removal action #17 (in OU-3), known as the Decontamination & Dismantlement (D&D) of the Plant One Ore Silos, exemplifies a "state of the art" nuclear air cleaning system. Constructed in 1953, Plant 1 was the "Sampling Plant" for the FEMP site and the receiving point for incoming ores and residues to be processed for the production of uranium metal. The contents were removed except for small amounts of residue. The objective of the removal action is to mitigate the potential for release of contaminants or potential hazards (radiological and safety) presented by the Plant One Ore Silos until total remediation of the OU-3 area is performed.

All D&D work activity is controlled to prevent the release of contamination. The work areas are isolated with physical barriers and a ventilated containment system. The containment for the silo structures consists of scaffolding and polyethylene fabric sheeting (area containment). The containment material is flame-retardant and corrosion resistant in compliance with DOE Order 6430.1A.

* Fernald Environmental Management Corporation with the U.S. Department of Energy under Contract No. DE-AC05-920R21972

Ventilation air is filtered through portable air cleaning devices equipped with pre-filters and High Efficiency Particulate Air (HEPA) filters and monitored before discharge to the atmosphere.

Contamination control is provided by local ventilation at work stations, and area containments of the silo structures and the size reduction building. Routine radiation and surface contamination surveys are performed on all work levels of the containment structures. Access into the area of containment and size reduction building are controlled by radiological technicians. Control point step off pads are provided at the access locations for personnel and material monitoring. The ventilation system is designed for a minimum of seven air changes per hour.

This paper will illustrate the design of the ventilation in compliance with ASME codes AG-1, N509, N510, and DOE order 6430.1A. The materials of construction and design of the ductwork, stack, portable nuclear air cleaning units, and the type of air sampler used will be addressed. Also, this paper will describe the phase approach to dismantlement and ventilation that resulted in reduced costs and waste minimization.

INTRODUCTION

Plant 1 was the "Sampling Plant" for the FEMP site and the receiving point for incoming ores and residues to be processed, and for FEMP's own waste for further processing. The Plant 1 Ore Silos were constructed in 1953 for the purpose of sampling and blending ore concentrates to feed the refinery (Plant 2/3) after sampling was completed. This system proved to be inefficient and was terminated. In approximately 1955, the silos were temporarily used as overflow storage for the cold metal oxides stream which was a by-product of ore processing. The silos have not been in use since late 1962. The contents were removed except for small amounts of residue.

The Plant 1 Ore Silos include the two groups of silos in an area directly south of Building 1A (see figure 1), consisting of the six reinforced concrete silos to the east and the eight glazed tile silos to the west. Four of the glazed tile silos are 44 feet tall and the remaining four are 10 feet tall. The six reinforced concrete silos are 10 feet tall. The eight tile silos sit on separate superstructures which are approximately 38 feet tall and are connected by a mezzanine. The estimated height of residual material in each of the eight tile silo cones ranges from 1-4 feet. The residual material that was located in the concrete silos was minimal.

On February 6, 1991, a spill was observed on the ground under the northwest tile silo during routine inspections by plant personnel. It is hypothesized that heavy rain on the previous day wetted the residues to the point of flow from the silo. Inspections indicated that residues had also accumulated on the lower platform under both western tile silos and the northwest small tile silo (see figure 2). The residue release caused the start of what is now known as the Plant 1 Ore Silos Removal Action. The objective of the removal action is to remove the source and any potential hazards

(radiological and safety) presented by the Plant 1 Ore Silos until they are addressed under final site remediation.

Ventilation System

All D&D activities are controlled to prevent the spread of contamination. The work areas are isolated with physical barriers and a ventilated containment system. Ventilation air is filtered through pre-filters and High Efficiency Particulate Air (HEPA) filters before discharge to the atmosphere. The DOE Order 6430.1A "General Design Criteria"[B.1] was used as the governing design guide. The DOE Order 6430.1A provided guidance into what standards were to be used in obtaining our final design. A listing of the standards and guides can be found in Appendix A.

Our first step in the design process was to determine types of radionuclide contaminants and the level of radioactivity given off by the contaminants in the areas inside and surrounding the Plant 1 Ore Silos. Samples of the types of radionuclide contaminants and their respective radioactivity can be reviewed in the Table 1-1 below.

Radionuclide	Concentration in Sludge or Debris (pCi/g)
TH-230	140000
RA-226	210
TH-232	2500
RA-228	29
TH-228	660
U-238	1100
U-234	1000
U-235	49
U-236	<2.1
Np-237	13

Table 1-1

The radionuclide contaminants in combination with the deteriorating structure of the Plant 1 Ore Silos dictated that a ventilation system be employed to not only provide dust control during D&D operations, but most importantly to mitigate the emissions of radionuclide dust particulates to the environment. The above contamination concentration values listed in Table 1-1 range from low to high contamination levels. Based on this information the entire ventilation system for the D&D of the concrete and tile silos was designed in compliance with the ERDA 76-21 "Nuclear Air Cleaning Handbook"[C.1] level 4 construction standards. The ventilation

system design accommodated the phased sequence of the D&D operation. There are 5 phases of ventilation to support the tile and concrete silos D&D (Figure 3). The phasing of the ventilation system optimized the original concept of ventilating the entire Plant One Ore Silo area, which was not feasible.

A scaffolding/containment system was erected (Figures 4 and 5) to totally enclose the working areas. The silo containment sheeting is a reinforced woven vinyl polyethylene sheeting. Also, a polyethylene vinyl sheeted aluminum framed building, known as the Size Reduction Building (SRB), was also erected. The SRB was used for size reduction of large equipment and structural steel. Size reduction of various silo materials was necessary in order to put the contaminated debris into storage containers. At various size reduction work stations, local ventilation of torch cutting operations are employed. Local ventilation is provided by use of a hand held portable HEPA vacuum with flexible duct and attachable hood.

Within the silo containment and the SRB, backdraft dampers were used to provide make-up air to the structures during ventilation exhaust operations. All backdraft dampers, ductwork, volume control dampers, air cleaning devices, duct supports and accessories were made of stainless steel. The stainless steel material that was used for the ductwork system and accessories was chosen due to its ease of decontamination, and the potential for reuse on future D&D projects. In Figures 6 and 7, the exhaust volume flow rates of the various containments sizes and the SRB can be observed. The minimum number of air changes required by the DOE design guidelines is typically 6 air changes per hour for personnel occupied areas. The minimum numbers of air changes for both the tile and concrete silo containments is 8. The SRB is designed to maintain a minimum of 11 air changes per hour.

To maintain the above stated number of air changes required state of the art, modular, mobile HEPA filtered air cleaning devices (ACDs) that are in compliance with standard ASME N509[A.1]. Each ACD needed to be equipped with prefilters as apart of its assembly including the industry standard bagin/bagout housing and be in place DOP testable. The three ACDs providing exhaust ventilation to the concrete and tile silos were fabricated by Charcoal Service Corporation. Each ACD provides a nominal 4000 CFM exhaust. The system total nominal exhaust volume flow rate is 12,000 CFM. The ACDs supporting the SRB containment structure were fabricated by Bartlett Nuclear. Each of these ACDs provides a nominal 1,000 CFM exhaust. The system total nominal exhaust volume flow rate is 5,000 CFM.

Radiological Control Data

Due to the effectiveness of our ventilation system on this project, routine contamination surveys performed on the perimeter of the D&D project area revealed that removable surface contamination have not exceeded 20 dpm/100 cm^2 (alpha), which is significantly below the DOE regulatory limit. All containment areas are posted as High Contamination Areas.

Perimeter air monitoring is performed on the project boundary to confirm the integrity of the containment structures and to ensure the adequacy of radiological controls. Average airborne concentrations have not exceeded 2% of the Derived Airborne Concentration (DAC) at the project perimeter. A representative portion of personnel working in the area are equipped with Personal Air Sampling (PAS) devices to monitor exposure to airborne radioactivity. Elevated airborne concentrations have been measured within the SRB and the silo containment areas, but have been well below 4 DAC-hours (per man per week) when considering a protection factor of 50 for full-face air-purifying respirators. The air samplers used on both the exhaust stacks for the SRB and silo containment structures collect particulate samples on a glass fiber filter which is continuously monitored by a Ludlum Model #177 G-M probes. Since the beginning of the D&D operations on this project, this sampler has not indicated that there is any notable contaminant emissions from the ventilation exhausts.

CONCLUSION

The design, fabrication, and installation of this ventilation system does meet all the requirements of ASME N509. All in-place DOP testing of the ACDs and pressure decay testing of ductwork and stacks comply with ASME N510[A.2]. All the stainless steel material and equipment that was used on this project will be decontaminated to the extent necessary for future re-use on other D&D ventilation projects. This exemplifies the cost effectiveness and waste minimization of our design when applied to other Fernald site D&D projects. When applicable, exemptions form DOE Order 6430.1A should be considered and reviewed at site DOE level to allow for deviation from the order. This will keep project costs down by introducing and implementing new technologies or ideas that are not currently addressed in the order but would satisfy environmental, safety, and health requirements.

Parsons Environmental Services and FERMCO are committed to ensuring that all D&D ventilation system designs comply with DOE Order 5400.5 "Radiation Protection of the Public and the Environment"[B.2].

APPENDIX A

A. American Society of Mechanical Engineers (ASME):
   1. ASME N509-89 — Nuclear Power Plant Air Cleaning Units & Components.
   2. ASME N510-89 — Testing of Nuclear Air Treatment Systems.
   3. ASME/ANSI AG-1-88 — Code on Nuclear Air & Gas Treatment.

B. Department of Energy (DOE):
   1. DOE Order 6430.1A — General Design Criteria.
   2. DOE Order 5400.5 — Radiation Protection of the Public and the Environment.

C. Energy Research and Development Administration (ERDA):
   1. ERDA 76-21-79 — Nuclear Air Cleaning Handbook.

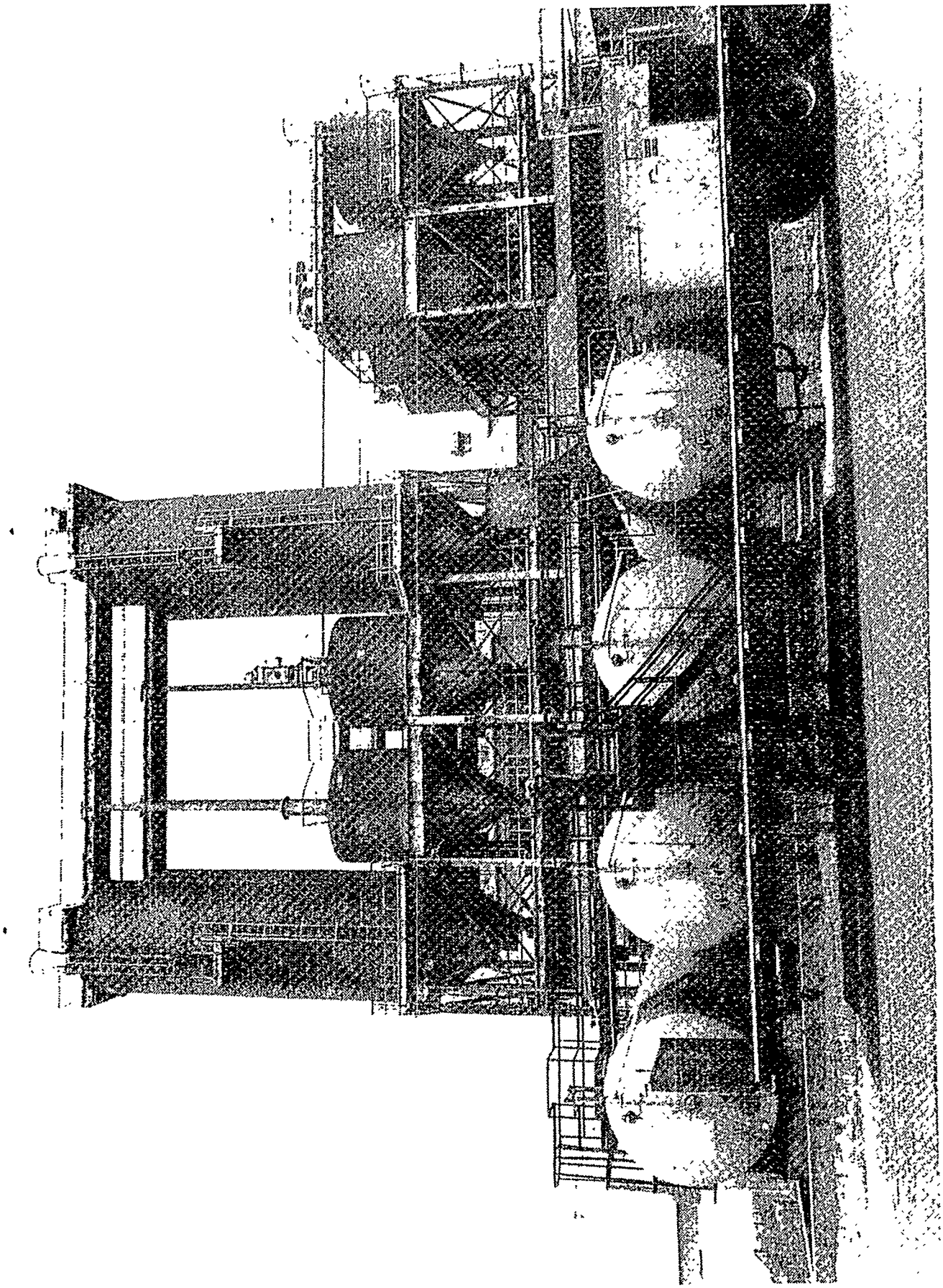


Figure 1 - Overview of Plant 1 Silos Looking North

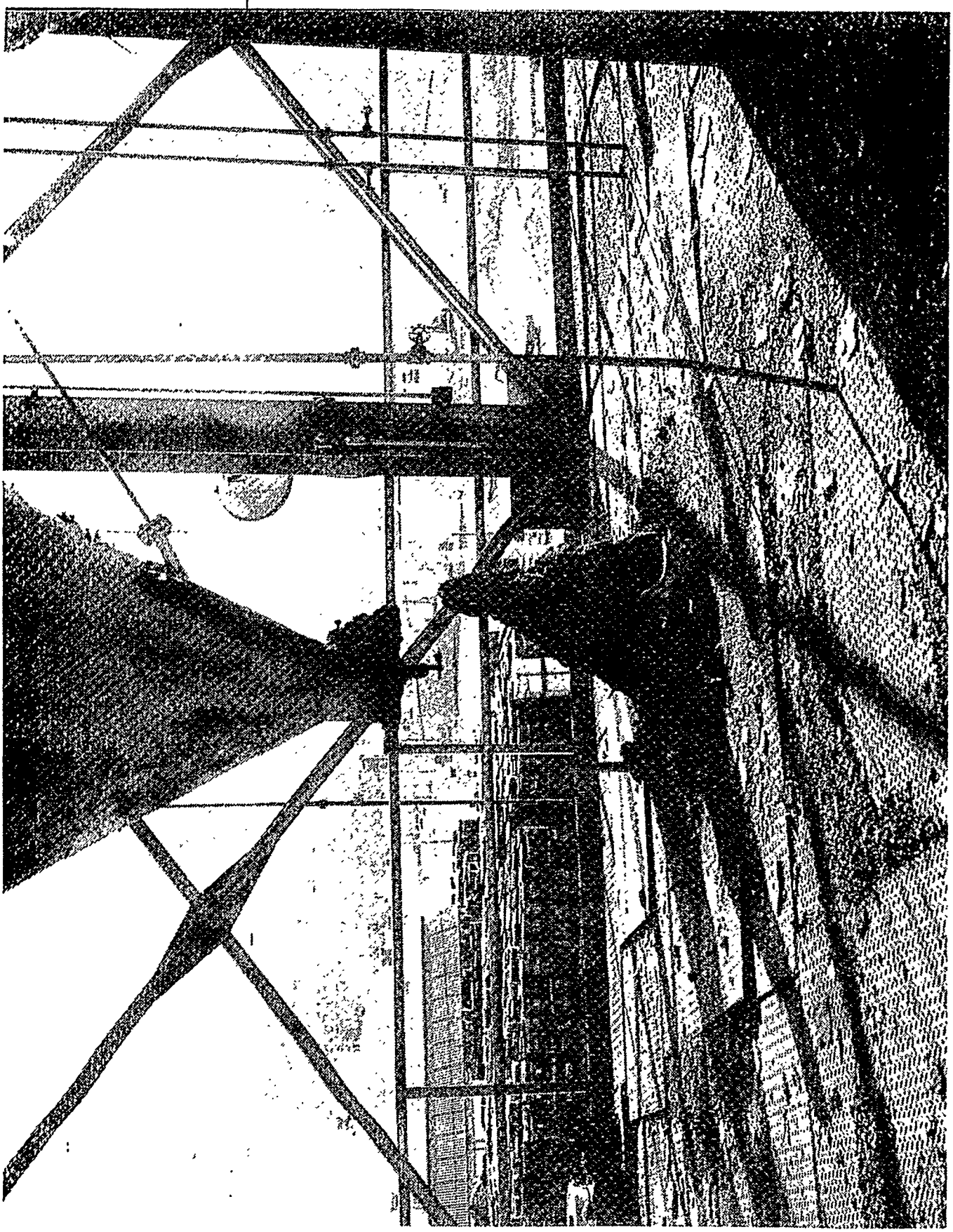


Figure 2 - View Under One of the Tile Silos Showing Accumulated Debris on the Steel Deck

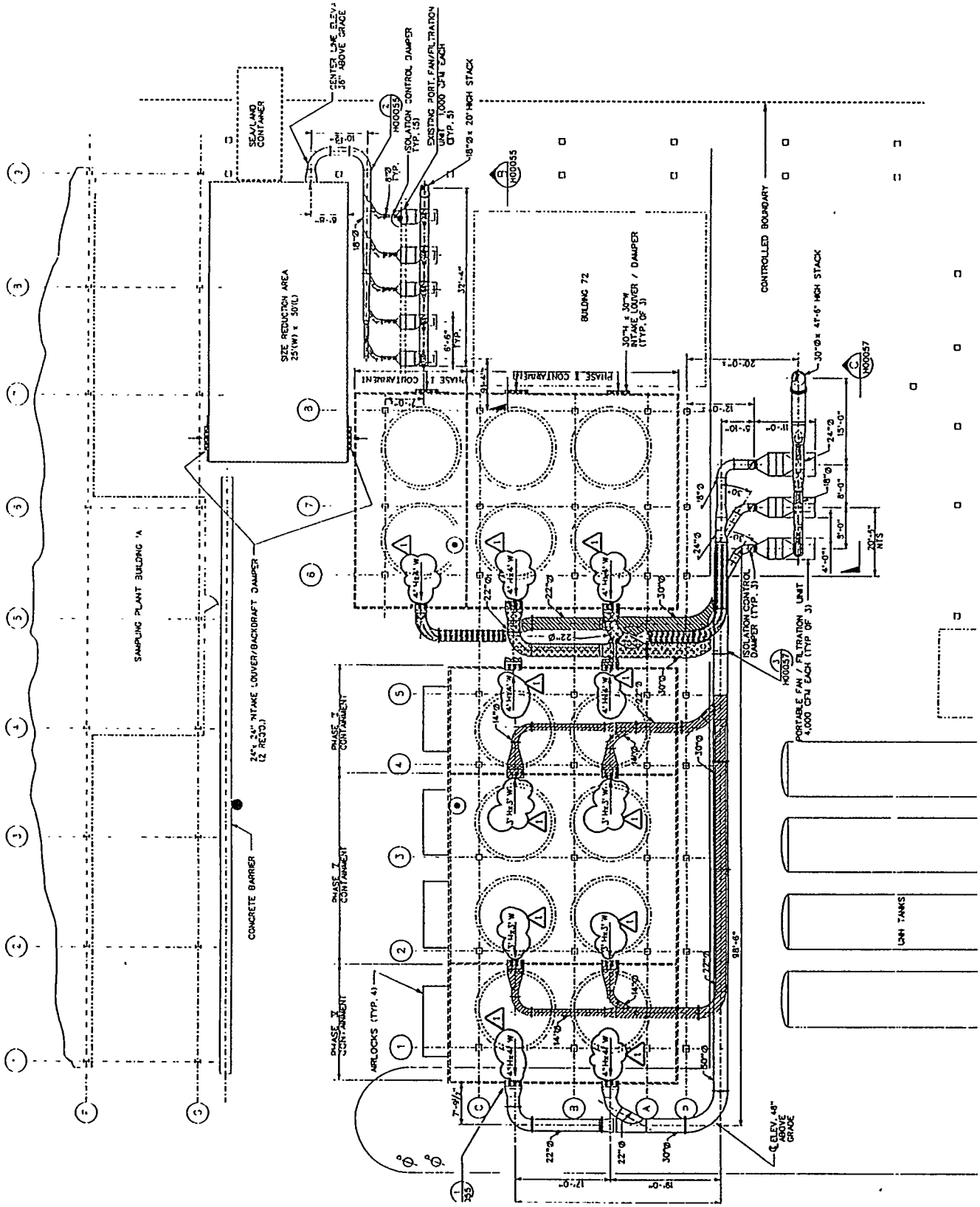


Figure 3 - Plan View of Ventilation Phases

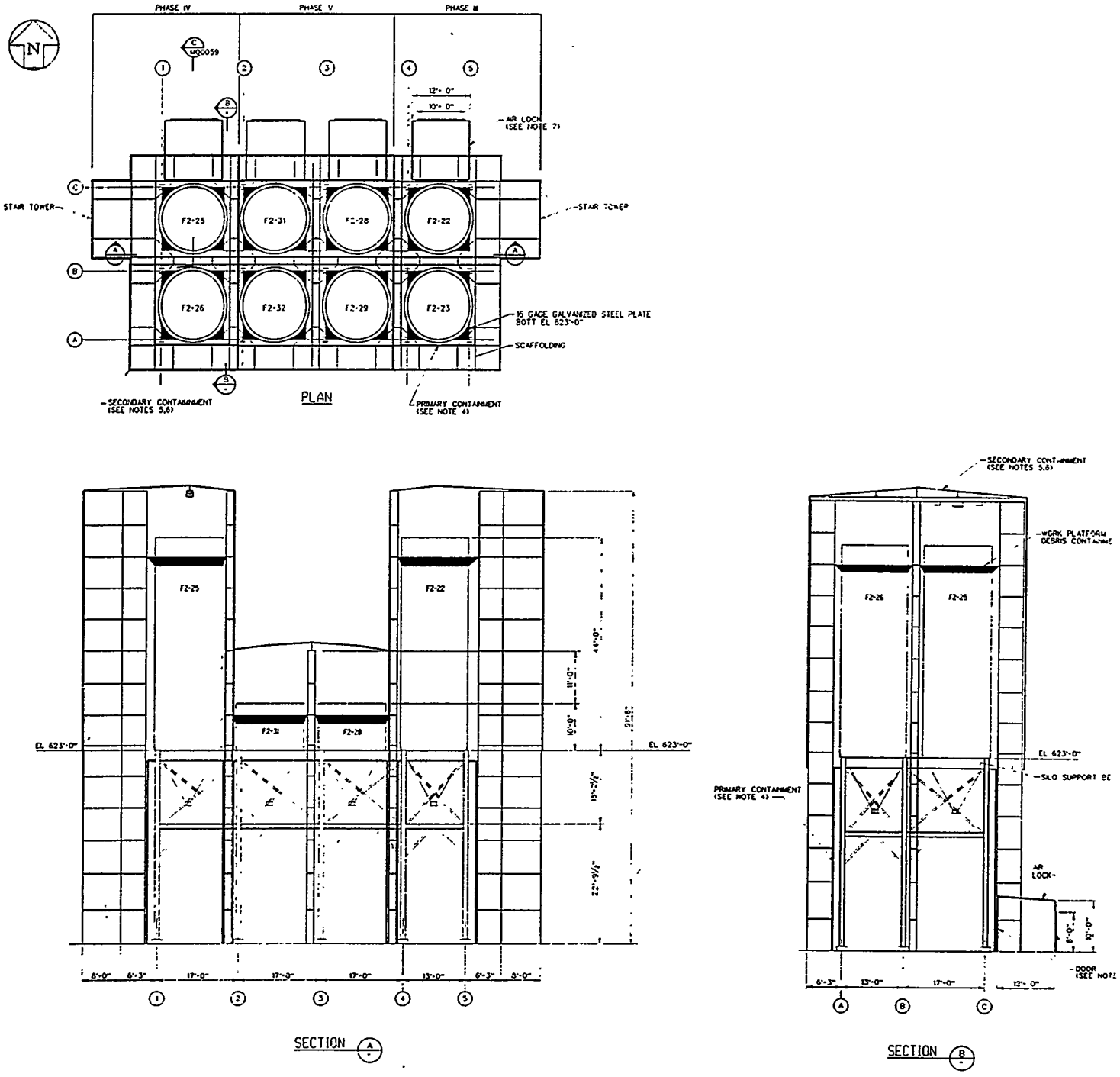


Figure 4 - Tile Silos Scaffolding and Containment

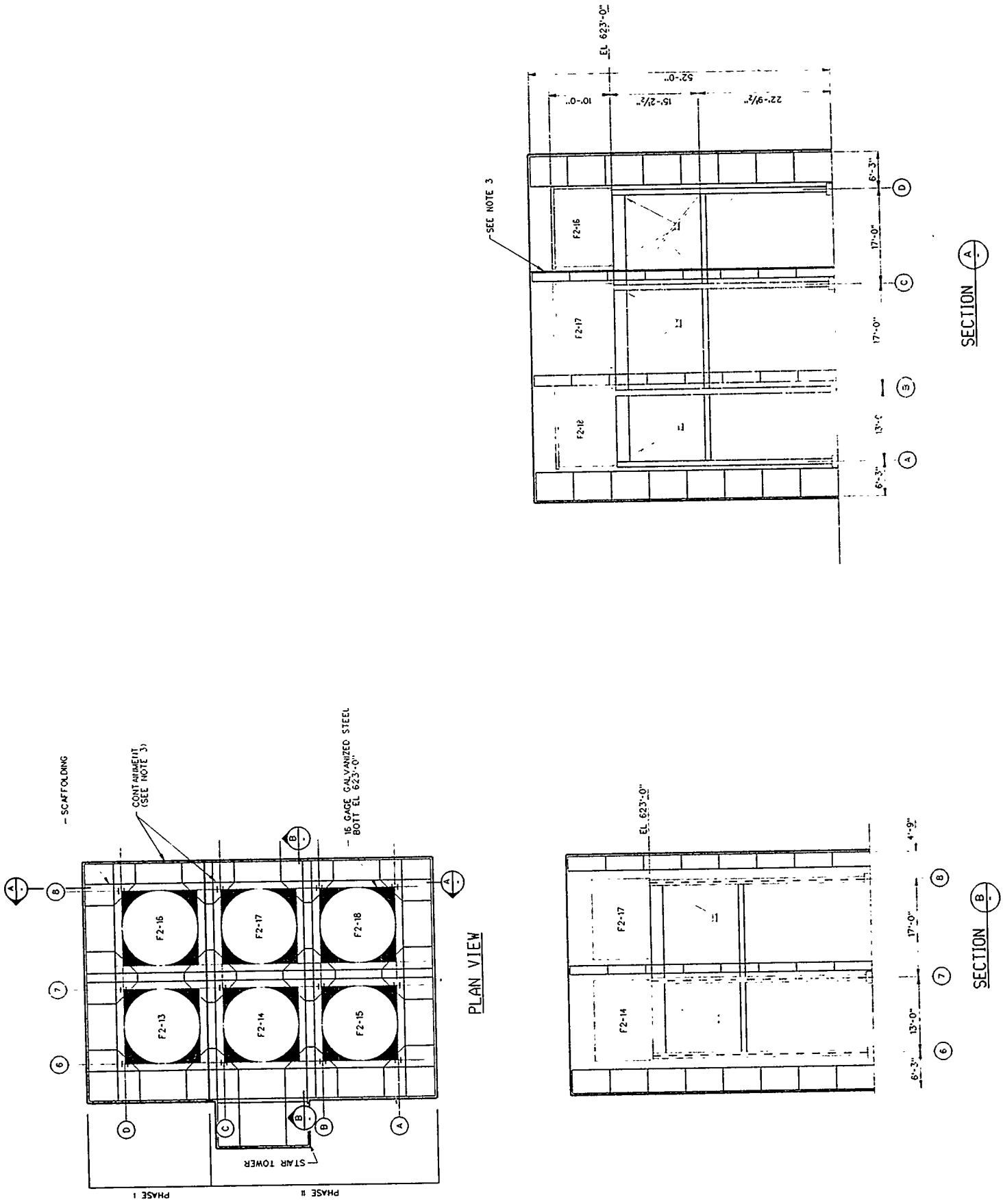


Figure 5 - Concrete Silos Scaffolding and Containment

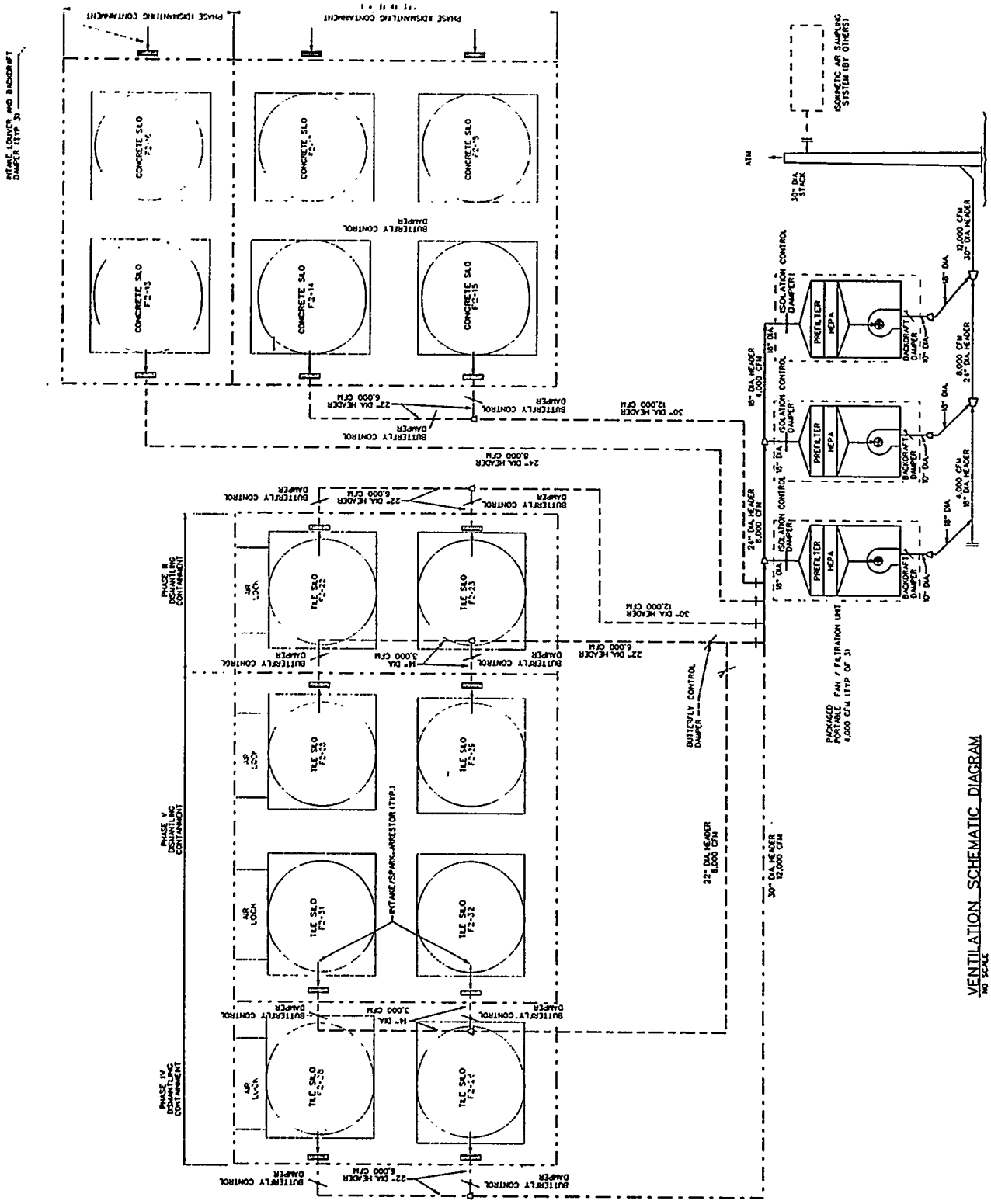


Figure 6 - Ventilation Schematic Diagram for Silos

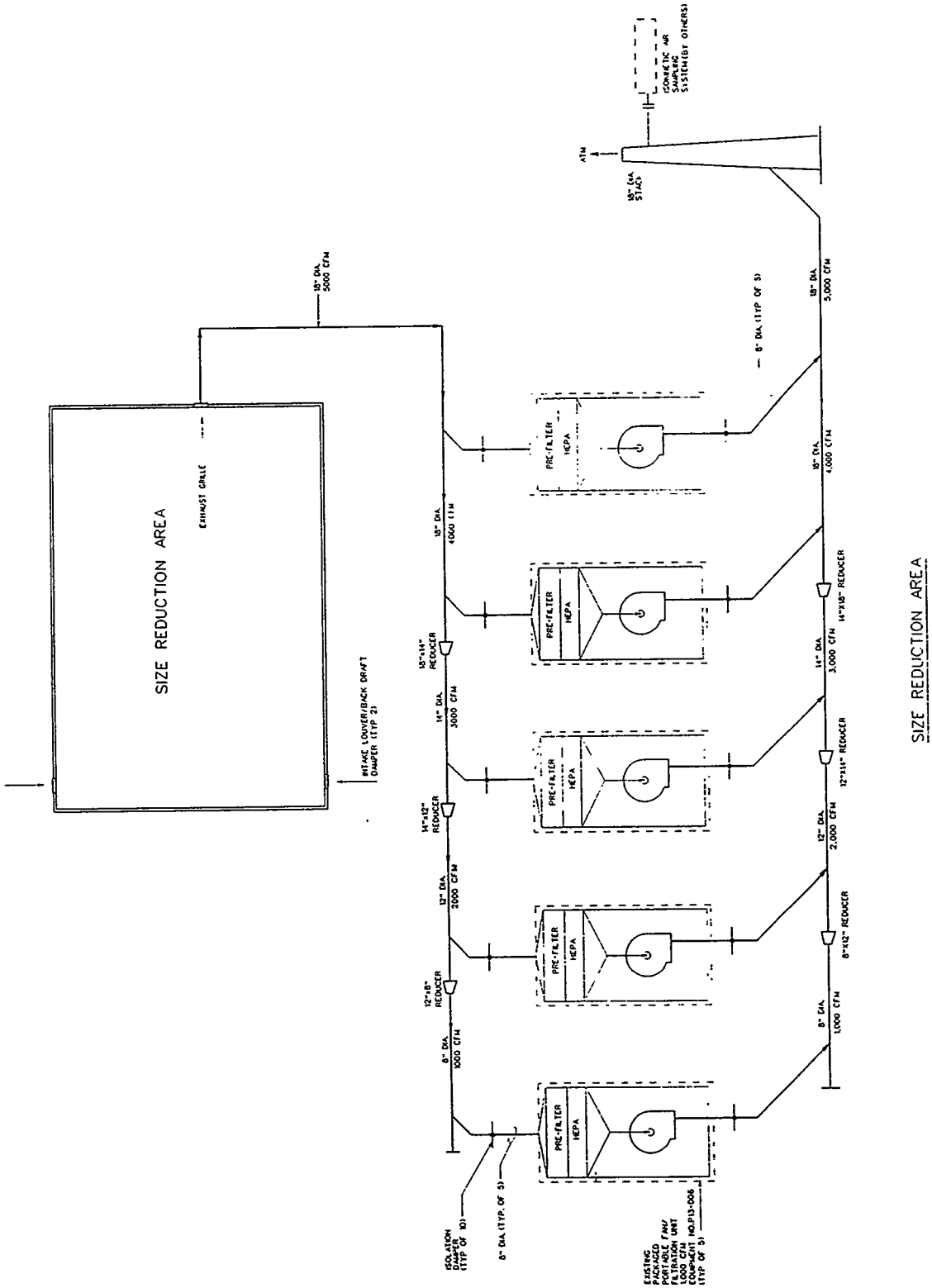


Figure 7 - Ventilation Schematic Diagram for SRB

PANEL DISCUSSION

WRIGHT: I have two comments of a slightly different nature, but related. From my safety and health perspective, I saw that you have a high risk operation, but it is gravity that seems to be your primary risk. You can fall or something can fall on you. Gravitational energy seems to be your predominant risk. The DOE Office of Safety and Health has responsibility for safety and health activities, not programmatic activities, involved in EM operations. We have twelve people to attend to all of these things, so our domain is a lot bigger than the personnel would indicate. Needless to say, I have been directed to look at D&D, to look at HEPA filters, to look at certain SAR issues, etc. We have another working group dealing with hazard categorization. They look at remediation sites, D&D sites, deactivation sites, and look at source terms so that we do not overclassify facilities. You may, with minimal contamination, get the tag "nuclear," and then you do have to worry about nuclear grade equipment and everything, but it may not be appropriate. That is the kind of issue we are trying to deal with. We have to find ways to assure good protection for the worker and the environment without gold plating safety, because nobody wins on that. I would like to say that if you feel we are not protecting the environment, the worker, or the public, let us know. If we are doing things that seem ridiculous and expensive, for which you get no gain, let us know that also, because we cannot solve problems we do not know about. So I would like to give everybody my name and phone number. If something is coming up with which headquarters can be of assistance, please let me know. I am Tom Wright, Department of Energy, EM 23/Quince Orchard, Washington, DC 20585, and my phone number is (301) 427-1629. Forget hierarchy, forget protocol, this is a case where it is easier to get forgiveness than permission. Call and get the communications started and we will look at it and if there is something we can do we will address the issue. We want to work together on this and we will coordinate with your program office for your individual operations.

WILLIAMS: I appreciate your comments. Perhaps we can put together a "lessons learned" brief summary for this program so we can start working together.

HAYES: It seems from the presentation this morning that you all have been very proactive in terms of getting the public informed about what is going on, the potential impact, etc. With respect to the sharing of risk, you indicated that you have to have DOE and EPA buy off on the particular approach you are taking. Where does the public fit in with respect to this sharing of risk? Have you included them prior to going to DOE, after going to DOE, or after going to EPA? Where do they fit in this decision making process?

EDWARDS: All interactions are approved through the Fermco management. We are now using what are known as DEC teams, design engineering and construction teams, where construction is involved very early. We even have DOE representatives on the DEC teams. By the time a plan is written, DOE is already familiar with it and we have a buy in. Then, we generally have a public comment period. Our DOE folks have been a lot more active with the EPA. They regularly scheduled meetings, and I think they do a lot of informal discussion of what will be in the written plan. A lot of things are happening at the same time, but you want to have your plan pretty much in its final form before you have the public comment on it. The last thing you want to do is have them comment on it and make a lot of changes. But you do have to allow time to incorporate their comments.

HAYES: And this meeting, Doris, is something that normally is put in the newspaper?

EDWARDS: Yes it is put in the newspaper and the documents are available for review in public information centers.

MCCULLOUGH: We also have a very organized public out at the Frenald. They have their own group called Fresh Organization, and then there is the Citizens Advisory Committee. The public does review all these documents in advance. There was a lot of talk with the public before a final decision was made on how building 7 was going to come down. I would say that the public is involved because they are fairly well organized. And DOE has given the public another avenue of bringing up their concerns about the site and getting responses; this is called the ENVOY program.

WORTHINGTON: I have a question regarding your operations. You had several slides that dealt with aerosol generation. Did you have any unique problems? For instance, did you have to spray down containments and did it pose any unique problems with respect to the HEPA filters? Were there other activities or lessons learned that you might want to share with us regarding such problems and the ways that you resolved them?

PALMER: The biggest problem with HEPA filters occurred in the size reduction building when we cleaned the surfaces by hosing down the inside of the building. Moisture was a problem that caused us to change prefilters. Big problems occur also during acetylene torch cutting operations. A HEPA filter is not a dust holding device. There were times during the project when we were cutting steel beams and we would have to shut down after just a few hours operation to change all our prefilters because they would be plugged up. So HEPA filters and prefilters were not always user friendly, even though they were protecting the public. I would say, on average, we got a two week life out of our prefilters for the size reduction building, and about four to six weeks for the silo containment structure. We have been fortunate, we have only had to replace the HEPA filters once on the size reduction building. Because we use a 90% ASHRAE filter on the 4000, we have not seen a significant rise in differential pressure across the HEPA filters.

MISHIMA: I think you have a wonderful opportunity because you are the first group of people doing a large scale project-by-project dismantlement or decommissioning, and you have made a fairly good characterization of what is present. Has any attempt been made to correlate what is on the ground with what gets into the air? It has a significant impact on the level of concern you would have during activity. If you could get some idea about the fraction of radioactivity and any amount of material made airborne, and in what form, when you are knocking down your tile or concrete silos, or torch cutting inside your buildings, it would be very useful.

TSCHAENN: We do continuous surveys of the work area and the perimeter of the work area and we have not observed contamination problems outside the containment areas. There is a lot of uranium-bearing surface activity from past operations. We have not seen any increase above baseline levels at the perimeter air monitoring stations. We have not characterized the particle size of the material but assume a particle size of 1 μm. We have not done any particle size studies and we have used the most limiting DAC for this project, a total inventory of 425 mci, 405 from thorium-230. The material is a by-product of the uranium extraction

process in oxide form.

MISHIMA: I think you are looking at this from a health physics perspective, i.e., to protect the public and workers. What is important is to know what comes off while you are doing the job so you can determine what level of protection is necessary for future jobs. For instance, are respirators needed when there are very coarse particles that do not go anywhere? When you know ahead of time that there will be nothing to cause problems, or in Randy's case, to clog prefilters, it might be more advisable to use high loading capability dust collecting cyclones as opposed to prefilters, which are atmospheric dust air cleaning devices.

SEEL: I have two questions, the first of which I have had some informal discussions with Randy about, but I will now pose formally. Has FERMCO or DOE considered publishing guidelines for the use of portable ventilation equipment similar to those used by commercial utilities such as Regulatory Guide 1.140 or the new 10CFR20 guidelines for engineered controls? Second, what consideration is being given in these D&D activities to the reduction of waste by sorting of contaminated vs clean wastes, volume reduction activities, or incineration?

PALMER: We have developed internal procedures for operating portable devices and we are in the process of developing site-specific requirements for vacuum cleaners, HEPA filters, and things of that nature that we are buying. Getting the information to all the different sites represents a problem, but it is nothing we could not overcome. We have developed a few guidelines for people to use based on this project. They are probably going to be like specifications. The other thing that I am glad to see is that DOE is starting to talk about portable devices. That is where we are looking for guidance. The April 1994 revision to the RAD manual is also very helpful in giving us guidance on what is expected of us in the way of testing and HEPA filter selection when using portable equipment. These are the best answers I have got for your questions.

MCCULLOUGH: Regarding the second part of your question, you may remember the slide of the whole plant where everything on the process side is usually considered to be contaminated. We have been working hard on this, because in our hearts we think there is stuff there that is not contaminated, and if we can prove it, it does not have to go to Nevada, it can go to a local landfill. This is a hard nut to crack.

TSCHAENN: We have a very active waste minimization program and we are currently segregating contaminated from non-contaminated materials. It usually involves office trash and other materials which have a low potential for volume or indepth contamination. We have tried to prove that there is no radioactive material added to other materials that are targeted for treatment, storage, or disposal as well. We have implemented a very rigid free-release criteria for materials leaving our site. A technician has to fill out several forms and get approvals from radiological control management to release an item. In addition, when there is a potential for hazardous constituents to be involved, other criteria for release are kicked in for treatment, storage, and disposal options. There are a lot of special circumstances that impact our ability to segregate items for free-release. Does that answer your question?

SEEL: Yes it does. I was pointing out that Fermco is the first remediation project of this nature in the DOE. Has there been emphasis at other DOE sites that they will very shortly

be experiencing the same situation that Fermco is in now, and that they need to utilize the lessons learned, from you and from the commercial sector, to fully implement the same kinds of waste minimization? I am fairly certain they are not currently the top concern at DOE sites.

MCCULLOUGH: We had been involved with some user groups from the DOE community and I believe they interface with the commercial world as well.

WILLIAMS: I would like to add to that. Last October Fermco sponsored a waste recycling workshop at the Greater Cincinnati Airport where the waste minimization program was described by Jerry Motl, the manager of the recycling program at the site. Available DOE guidance states that there are no real free release criteria. What we mean by free release is being able to sell our scrap metal to any scrap metal recycling facility. We had a scrap metal pile on the site that we had to get rid of. Through a category exclusion, we sent it to SEG at Oak Ridge, where they smelted it and made it into waste containers. Now, we are reusing the same metal from our site as waste containers. There was some question whether all the metal that was smelted was contaminated and that's still being looked into. Nevertheless, the waste metal we sent out was recycled and shipped back to us. Unlike the NRC Regulatory Guide 1.86, there is no established free-release criteria for DOE sites.

TSCHAENN: I can add that we use R. G. 1.86 for our free release contamination limits although we take exception to the 100 d/min average. We use a 300 d/min maximum total activity for transuranics, thorium-230, and radium-226. The DOE criterion for free release currently has these limits reserved. We use the most limiting regulatory driver with the exception of the 100 d/min value. Our free release is also very restrictive for common materials brought on site. We go through very rigid free release criteria to get basic materials, such as heavy equipment, off the site.

MCCULLOUGH: Perhaps we should communicate what we have learned, so far, to other DOE sites, but keep in mind that we are only about half way through the project. We hope to learn more before we are done.

DAVIS: As the first into this activity, what you are doing is very interesting. I have a question on your HVAC system. You are apparently going to use it for each one of your projects as it is very portable, from what I could view in the slides. What did it cost to put the system together and what are the future plans for using the system?

WILLIAMS: The costs of the 3000 and 4000 CFM air cleaning units plus the 5000 CFM units came to a total of $81,700. The cost of the associated duct work and installation is still being estimated because the project is only 60% complete. We are still in the process of estimating how many labor hours are involved in tear-down, and setting back up again. The duct work material, in 1991 dollars, was $2.50/lb for stainless steel.

PALMER: I do not know if that equipment has been designated for any other projects yet, but I think it has not, because this project is not over yet, it should be completed in September. Later this year we will know what other projects can used it. It is also likely that these eight systems could be segregated and used on several projects, not just one. We do plan to use every one of them again. They are nice equipment, and we are happy with them.

WILLIAMS: The equipment was designed for the most contaminated areas on the site and it is very expensive compared to some of the other operations we had. Building 7 on the slide we showed was the tallest structure on the site. It is in the process of going through the D&D process. We are removing the transite panels. Then we are going to pull down the structural steel. We do not have radiological concerns in that building, our major concern was asbestos abatement. So we utilized cheaper asbestos abatement units that did not require a leak rating of 0.1% of volume capacity as specified in the Nuclear Air Cleaning Handbook. The units were not all-welded construction, they used pop-rivetted housings. The equipment that you saw will be utilized again. Determining where, is something that we work out with Kevin Tschaenn's group when they go in and do their initial characterizations of the buildings that we have to D&D. I hope it is not the same at other sites, but when this plant was shut down in 1989, they just went in and threw the switch. There is process material still in the piping and in some of the reactor furnaces. They did not finish processing all the material, they simply stopped operations as they were flowing, so, in some cases, we are not sure what we are dealing with. That is why we have to go through a very slow step-by-step analysis to ensure that we protect the environment and personnel doing the D&D operations.

ISZKIEWICZ: I have two brief questions, what type of air makeup system did you use for your various containment areas and did you use any internal air cleaning devices in areas of high dust or fume generation as a protection for your prefilters on the main system?

EDWARDS: For phases one and two, we used gravimetric dampers for the makeup air. For phases three, four and five, and the tile silo, the original design used containment sheeting on the outside and interior of the scaffold for the upper levels, giving us makeup air around the entire perimeter. Phases one and two were so successful that we changed three, four and five, and are bringing the sheeting all the way to the ground. We believe that will give us better containment of debris when the brittle tile falls. And we will go back to using gravimetric dampers at worker elevation, so the workers will get the benefit of the makeup air. As for your second question, we are changing the method used to demolish the tile silo. At one point, it was a very complicated process, but we decided to take advantage of gravity and just throw it all down the middle using some chutes. We are going to put some additional point source HEPA air filtration devices right at the base of the chutes.

PATEL: Most of my experience is on nuclear power, not on DOE types of operation, I only have a little bit on that. I have three questions. Two are simple and one may be complicated. First, do you have iodine filtration like power plants do? I have seen prefilters and HEPA filters but no charcoal adsorber.

PALMER: No, we do not have any charcoal beds in any of these devices.

PATEL: You mentioned that you are only 50 % finished on your project schedule right now but you have a facility for breaking up the equipment, I think I got lost on that. My second question is, where do you send the scrap after you process all the small pieces of concrete or metal? Where do you send it for waste storage? What waste storage facility do you use?

EDWARDS: The way the plan is written right now, we are using a separate size reduction facility, so we do not impede the workers in containment. The concrete and tile will go in white metal boxes designated to be shipped to DOE's Nevada Test Site. The structural steel is being cut

into lengths of about nineteen feet to fit the size of the trailers we have and it will go to another D&D facility where it will be cleaned to make it eligible for free release. Right now, they are struggling with free release issues. The steel will be set aside until the issues are resolved.

PATEL: Will it be underground or above ground storage?

EDWARDS: It will be pad storage above ground.

PATEL: How long will it be stored on the site?

EDWARDS: It is not known. A change that I think will take us a step farther is to leave all the foundations in. The original removal order was to take the foundation out with the steel, i.e., complete demolition of the whole area. CRU-3 has gotten an Interim Record of Decision (IROD) approved. We are deferring decisions on foundations until it is determined whether materials such as concrete, which may not have a lot of activity, but some surface contamination, will stay on the site. It is not known right now but it is a big issue.

PATEL: The third question that I have is, you mentioned certain codes and standards you follow and they include DOE 6310.

EDWARDS: DOE 6430.1A contains the general design criteria.

PATEL: You are also supposed to meet EPA regulations, plus N-509 and N-510. I believe you said you have some questions about whether you should use nuclear grade materials. Basically, R. G. 1.52 and 1.140 were used at nuclear power plants before we went to N-509 and N-510. If the AG-1 code is approved by NRC, it will replace N-509 and N-510. My question is, do you have exemptions from DOE that you do not have to comply with N-509 and N-510? What are the exemptions you have on N-509 and N-510 compliance?

WILLIAMS: We have no exemptions for meeting the requirements of N-509 in the design criteria or of N-510 in the testing, because we have met them. We do not want to have to comply with them but we need documentation to justify deviating from the standards. The standards required spending some $80,000 for air cleaning devices that may not have been needed. For the full site cleanup we are looking at $20 billion over the next 15-20 years. That is how long they think remediation will take. Therefore, we want to reduce expenses as much as possible. These requirements in the standards you cited are for NRC regulated nuclear power plants, but they do not really apply to non-reactor nuclear facilities. Because there is no other guidance available, and we have to be legally responsible for our designs, these standards are the only thing we can use.

PATEL: So, you meet material requirements, you meet testing requirements, you meet design requirements, but you do not qualify the equipment and you do not require seismic qualification. Are these the exceptions you take?

EDWARDS: We took an exception in building where we were doing asbestos abatement. We looked for guidance on how to build the containment and what equipment to use in the asbestos standard, because that was the type of work we were doing. The first activity, what

we call safe shutdown, will be to get the residual production material out, and that will call for one kind of equipment. After all the radioactive materials have been removed, we will get to a new stage, which will be more like a normal construction phase, where there will be hazards. When you are dealing with asbestos, for example, you should look at different standards for the guidance.

PATEL: So, you totally do not meet N-509 and N-510, but you are meeting the intent of N-509 and N-510?

WILLIAMS: We totally meet the applicable regulations.

EDWARDS: This project met all regulation; we took an exception on another project based on the type of work we were doing.

PATEL: I still get lost, when you meet the regulatory requirements of N-509 and N-510 literally, you meet the nuclear standard.

WILLIAMS: Exactly right. But we do not want to have to, we do not want to be required to meet standards that may not be necessary in D&D work. And that is where we are at at this stage.

PATEL: Do you send your HEPA filters to a DOE filter test facility before you use them?

PALMER: Yes.

PATEL: Do you use commercial equipment which is cheaper than nuclear grade?

PALMER: Yes, even in the commercial equipment used for the asbestos abatement projects in plant 7, we used nuclear grade HEPA filters, and UL approved prefilters. That is part of the problem. We made those devices operate at 1,000 CFM instead of 2,000 CFM because we couldn't find a supplier that would certify a 2,000 CFM nuclear grade HEPA filter. That is why I brought up the last issue, although it was associated with a different project. The devices used for asbestos abatement have two speeds, we operated them at the lower speed.

WILLIAMS: I would like to add that all the duct work was sized to maintain a transport velocity of 2,500 fpm and qualified for use in a reactor. For the low levels of radioactivity that we are dealing with in the D&D operational mode, we do not want to have unnecessary expense.

WILLIAMS: To summarize, we would like to reflect on whether we are going overboard on what we are doing. As I remarked, we may have gone too far, and it might be in everybody's best interest to consider the use of more appropriate criteria.

DUVALL: The D&D process that you described with the associated ventilation systems and monitoring controls seem to be designed to address removal of particulate material such as uranium and thorium. I suspect that radon is emitted, also, and I wonder if it is a matter of concern?

TSCHAENN: At this particular project we had some radon activity, but it was at levels that

were not above NESHAPS standards. Our largest radon generators are at silos one and two. I believe them to be the largest radon generators in the world.

DUVALL: Subpart H of NESHAPS addresses sources of radionuclides other than radon. However, subpart Q does address radon emissions but the standards associated with subpart Q particularly address piles, where there is a flux standard. It would not be an appropriate standard for this type of operation. If you are considering compliance under subpart Q for this particular project, how would you demonstrate it?

TSCHAENN: I really cannot answer your question, but it was considered by our compliance group. We do not have a representative here, but your question can be directed to Kip Klee, at our plant.

DUVALL: Would it be a major concern for activities where there is uranium?

TSCHAENN: We have thoron and radon concerns at our site and they have been characterized. The measurements collected at the Ore Silos indicate that we have been in compliance with subparts H & Q.

Radon fluxes have been modeled to be below the 20 pCi/m^2 flux limit in the NESHAPS standard. Thoron is not a regulatory issue to my knowledge, but is an occupational concern.

FARRIS: What have been the heat stress related time limits for workers wearing protective gear, during various seasons of the year?

WILLIAMS: The heat stress related time limit for workers wearing full personal protective equipment is approximately 3 hours, but this varies for different workers.

EDWARDS: Last month we found we were getting heat stress effects starting about 11:00 a.m. Early in the summer we implemented an earlier start for the shift whereby they would be taking their first lunch break at about then, and we had continuing activities after lunch. We have rescheduled so that we are doing more safety meetings and training in the afternoon. When heat is worst, we use 15 min in the work area, 45 min out. Individuals are monitored by pulse rate. We bought an air conditioner for the cool down area, we have got an area zoned for water. Getting an air conditioner ended up being more of a challenge than we thought it would be because of the World Cup. No one had the size we needed. Starting Monday, we will go to a second shift for the demolition of the tile so we can work during the cooler hours. Where lighting and access may be more of a hazard consideration, we will use a limited day shift to support the second shift.

FARRIS: Have you considered using a blow-back type stainless steel filter to protect the HEPA filters from heavy loadings and to concentrate the contaminants? They are used in the chemical process industry in similar situations.

WILLIAMS: No, we have not considered stainless steel filters for protection of HEPA filters used. We have used two prefilters with spark arresters upstream of the HEPA filters to prevent premature dust loading of the HEPA filters. This method has been proven successful for this

application.

LILLYMAN: Are the materials you are dealing with in the silo natural uranium mill tailings? Presumably, this is material from the processing of uranium through solvent extraction. So what you have is mainly thorium and uranium.

TSCHAENN: Exactly. We have seen some radium in this material, but mainly it was thorium-230 that was concentrated in this by-product material. And it is termed K-11 metal oxides, the historical name given to it. It is a by-product of the extraction process.

LILLYMAN: Please define the DAC. What is it related to?

WILLIAMS: Derived Airborne Concentration (DAC) is the amount of airborne radiating particulates that could be considered an inhalation hazard. DAC federal limits are set forth to limit radiation exposures through inhalation by workers. (See 10CFR, part 835.) As stated in my paper, the air within the containment was well below 4 DAC-hours (DOE limit) per person per week.

LILLYMAN: What units is the DAC in?

TSCHAENN: It's in microcuries per cc.

LILLYMAN: You do not use bequels?

TSCHAENN: We are supposed to, we are slowly implementing it.

LILLYMAN: You were citing averages of 0.5 of a DAC. By the ICRP definition, half a DAC is a very high dose rate. We normally work to less than 5% of a DAC, and preferably 1%, the standard working limits in UK facilities, especially plutonium facilities. What is your definition?

TSCHAENN: In answer to your question, we do have a bioassay program which complements our air monitoring program for occupational exposure. The DAC is referenced in DOE 548011, it is the DAC for thorium-230 class W material. It is the most restrictive DAC that we could apply. Material makeup shows 95% of the activity due to thorium-230. We assumed 1 μm AMAD, we were very conservative. Our bioassay program consists of routine fecal sampling.

LILLYMAN: What is your DAC in alpha activity/m^3, of air that you are breathing? I am trying to relate what your material is with, say, plutonium.

TSCHAENN: It is an alpha emitter.

LILLYMAN: I am trying to relate it to plutonium. The next question is, what sort of surface contamination levels are you getting in your tent and in your facility? Because that is a good guide to what material is being disbursed from your operation.

TSCHAENN: Inside the containments themselves, we have seen a maximum of several

thousand D/M of loose contamination. The area was posted as a high contamination area.

LILLYMAN: What were the surface contamination levels on the tent and surfaces?

WILLIAMS: Surface contamination levels did not exceed 20 dpm/100 cm^2 (alpha), during the ventilation system operation.

LILLYMAN: How active are the inside surfaces of your tent going to be when you are finished?

EDWARDS: It will be above releasable limits. We take steps to control generation at the source of the contamination and we enforce housekeeping to maintain the area free of contamination as best we can during the D&D. It is a very difficult task, you are working in a containment and yet you are not. Engineering controls on this particular D&D activity are very difficult to employ. We would have liked to have put the 165-ton crane inside the containment, but could not. Those are some of the things we have had to deal with.

ANON: Have you taken wipe samples of the inside of the tent?

EDWARDS: We rented the scaffolding that is in the containment. The scaffolding we took down at phases one and two is being surveyed right now. It will all go to the D&D disposal facility, but it is believed that most of it will be cleanable. The spray from the washdown is limited to areas where the work was being performed. We were not getting that much spray. The water ran down the side of the cylinder and into the cone, in the center of the structure.

We use water for decontamination and HEPA filter portable vacuums. We enforce good housekeeping and good radiological work practice.

LILLYMAN: Your big danger is in removing activity from inside the vessels under your tent and structure. It does not get you very far in real terms.

EDWARDS: In construction projects, we usually end up buying some part of the vendors material when we rent it, but it is very small compared to the overall cost if we had bought all the scaffolding.

LILLYMAN: Has the use of ventilated suits been considered?

WILLIAMS: Ventilated suits are too cumbersome, and due to elevated platforms, a lack of space, and sharp angles on the scaffolding, they would have been impractical.

LILLYMAN: Concerning this matter of heat stress, as a person from a northern clime, I find even 80° rather stressful at times, and so do our footballers. When working in these conditions, we would normally be using airline suits. Have you considered airline suits? With airline suits, you have nice cold air coming in at a high rate and it is much more comfortable to work. If you ask the boys doing the job, they will prefer an airline suit anytime to just a respirator and a PVC suit.

EDWARDS: One of the slides showed concrete cutting where we had water and the workers were

fully dressed. The workers building the scaffolding were in a suit, but not in a respirator. We had ice vests for them. You are talking about a vortex cooled suit that we use in other buildings. But it was so difficult building the scaffold with 100% tie-off, they are in two lanyards as it is. There are only certain parts of the scaffold they are allowed to step off. They have to have one lanyard hooked on, and they have to reach across, get the next one, and do this monkey tail business all across there. To complicate the whole thing by adding an airline would have made that job extremely difficult. That is why we chose the option of changing the entire work shift. In another building with asbestos work, we have used the vortex suits. They work great, no question about it.

ANON: We use cooling vests and vortex suits. And we also use powered air purifying respirators. The air in the mask gives some relief although it is not quite as nice as an airline hood.

GLISSMEYER: It seems like your DOE office does not give you a lot of guidance on requirements for your ventilation systems. Are you able to use local air cleaning devices in your metal and concrete cutting? Something similar to the wet device that has a demister and some kind of bag filter, or one of these elephant nose exhaust hoses with an electrostatic precipitator on it for your metal cutting? That might cut down your dust loading inside the containment.

PALMER: In the size reduction building, we would have had the room but we just did not think of it. We do use standard industrial vacuum cleaners with HEPA filters to clean up. We use Wet-Vacs to pick up liquids and we use a dry vacuum to pick up dry solids. I do not know if they had enough room with all the scaffolding in the containment to be able to do what you suggest. When we do metal cutting, we only have a problem with oxy-acetylene cutting. When we do concrete cutting, we wet it down. It does not seem to cause much of a problem to the HEPA filters, if you call a 4-6 week maximum prefilter life not much of a problem. We are pretty happy with over four hours prefilter life.

MCCULLOUGH: That is another valuable lesson learned for the future. We thank you very much for your attention. You have given us a lot of good ideas and we hope we have given you some.

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This report contains the papers presented at the 23rd DOE/NRC Nuclear Air Cleaning Conference and the associated discussions. Major topics are: (1) nuclear air cleaning codes, (2) nuclear waste, (3) filters and filtration, (4) effluent stack monitoring, (5) gas processing, (6) adsorption, (7) air treatment systems, (8) source terms and accident analysis, and (9) fuel reprocessing.

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