SESSION 12

AIR TREATMENT SYSTEMS AND ACCIDENT CONTROL

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Co-Chairmen: R. D. Porco
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VALUE-IMPACT ASSESSMENT FOR RESOLUTION OF GENERIC SAFETY ISSUE 143 - AVAILABILITY OF HVAC AND CHILLED WATER SYSTEMS

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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

(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.

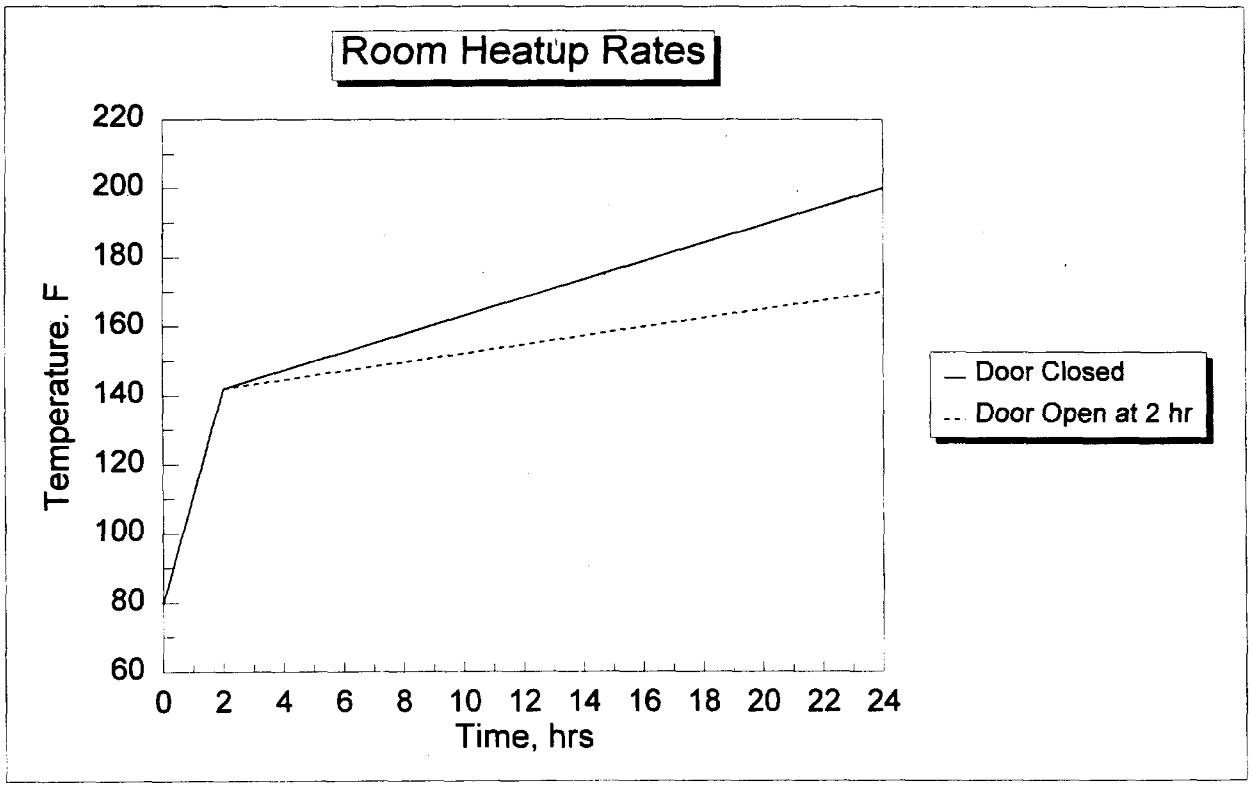

Figure 1 Illustration of transient thermal calculations: low pressure injection pump room heatup rate following loss of room cooling, including the effects of opening the LPI pump room door 2 hr. after the loss of room cooling.

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 - External Events	 4.1E-06 8.2E-06	 5.4E-07 8.1E-06	 1.3E-06 8.2E-06	 1.2E-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 - External Events	 1.7E+00 1.6E+00	 9.9E-02 3.9E+00	 7.2E-01 1.6E+00	 1.7E-01 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)*

F lution 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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(2) U.S. Nuclear Regulatory Commission (NRC). Severe Accident Risks: As Assessment for Five U.S. Nuclear Power Plants. NUREG-1150. Washington D.C. 1989.

(3) U.S. Nuclear Regulatory Commission (NRC). Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission. NUREG/BR-0058, Revision 1. Washington D.C. 1984.

(4) Heaberlin, S. W. et al. Handbook for Value-Impact Assessment. NUREG/CR-3568. Pacific Northwest Laboratory, Richland, Washington. 1983.

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).

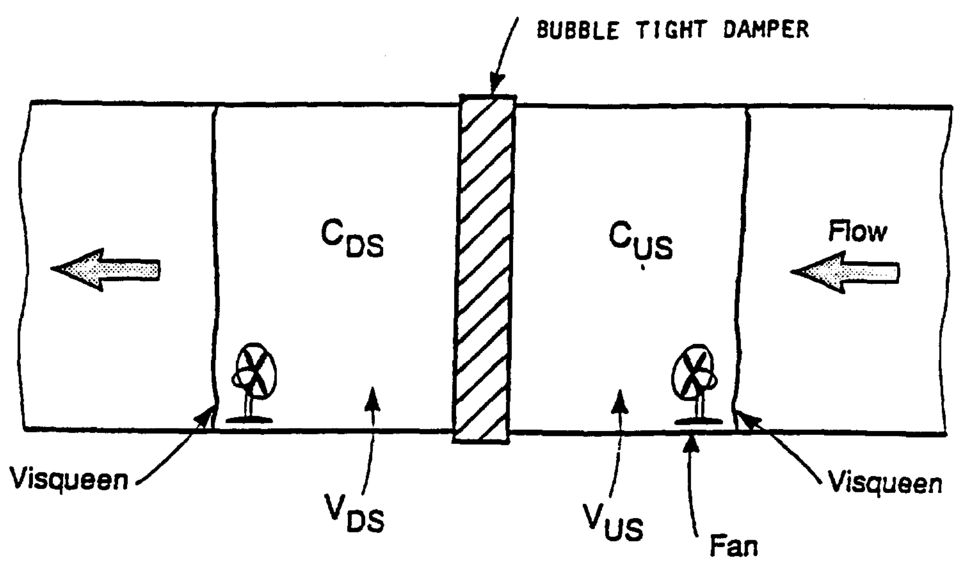

Figure 1. Bubble Tight Damper Test Using Tracer Gas

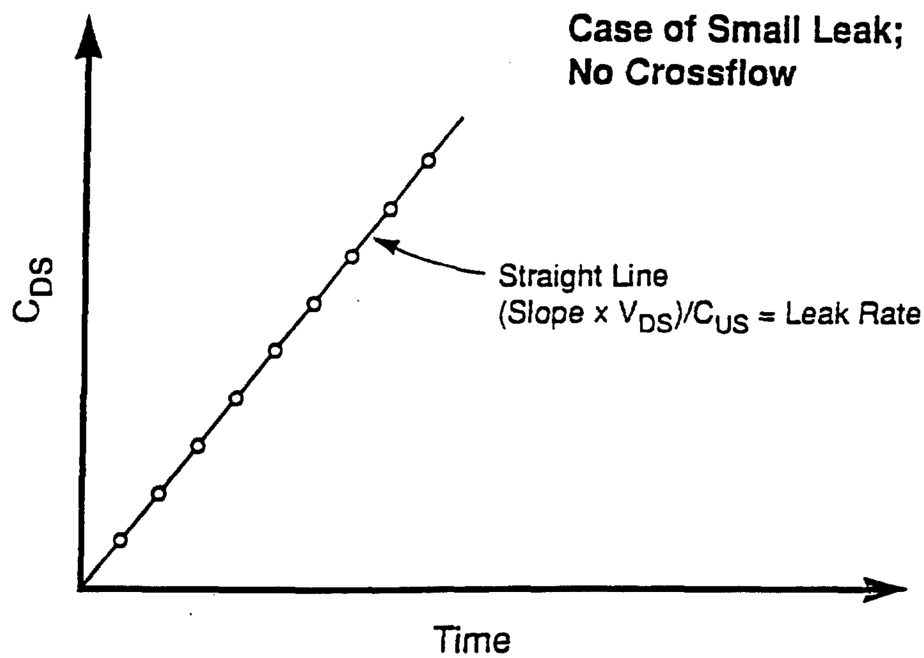

Figure 2. Tracer Concentration versus Time for Damper Test

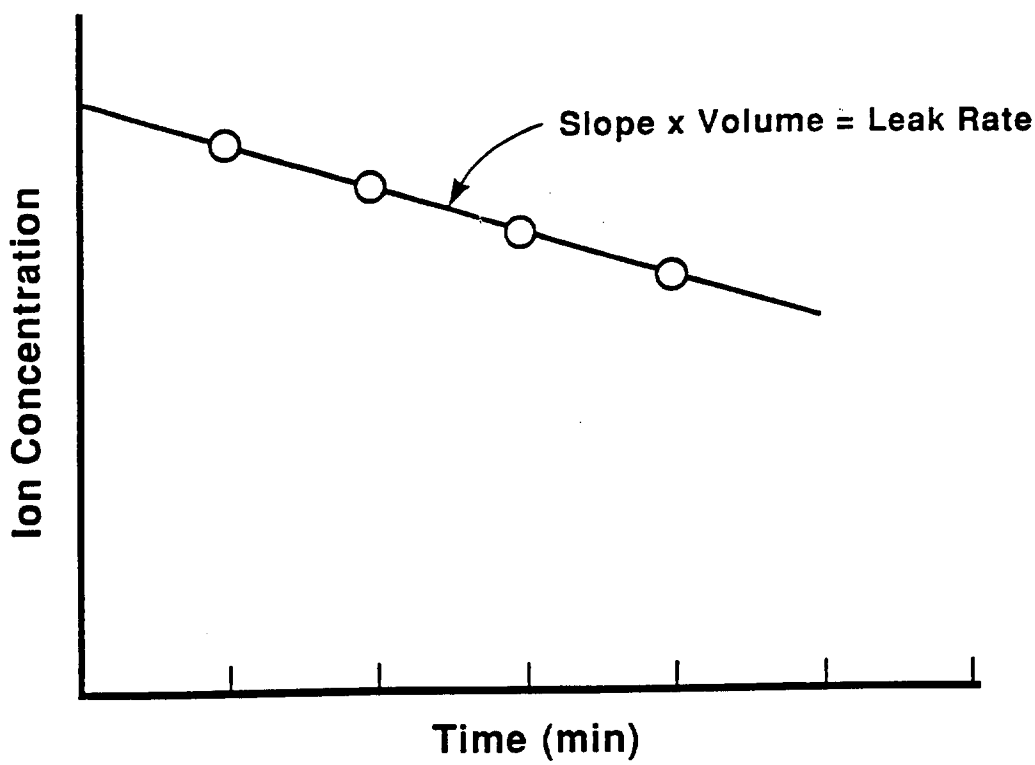

Figure 3. Tracer Decay for High Leakage Through Damper

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).

$$C = 0.12468 \times t - 0.26389$$

$$r^2 = 0.997$$

$$C_{av} = 81 \text{ ppb}$$

Leak Rate = 0.124 x 18 / 81 = 0.03 $\pm$ 0.01 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.

$$C = 30.3 \exp [-0.7429 t]$$

$$r^2 = 0.993$$

$$I = 44.6 \text{ ACH}$$

Leak Rate = I x (DS Volume) / 60 = 21 $\pm$ 4 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

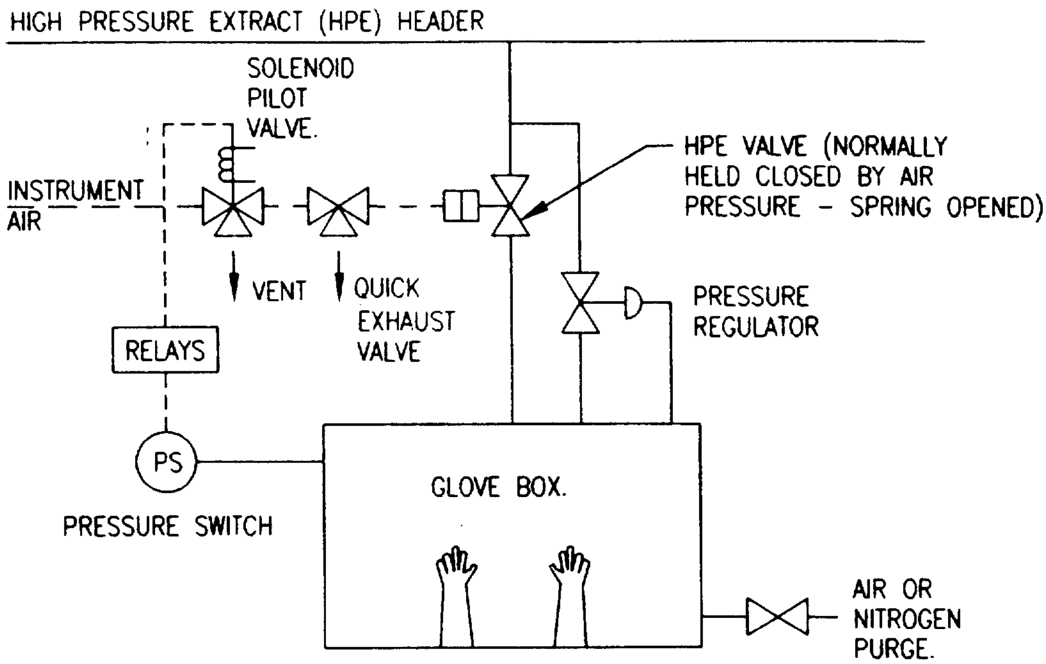


Figure 1 Simplified HPE system

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).

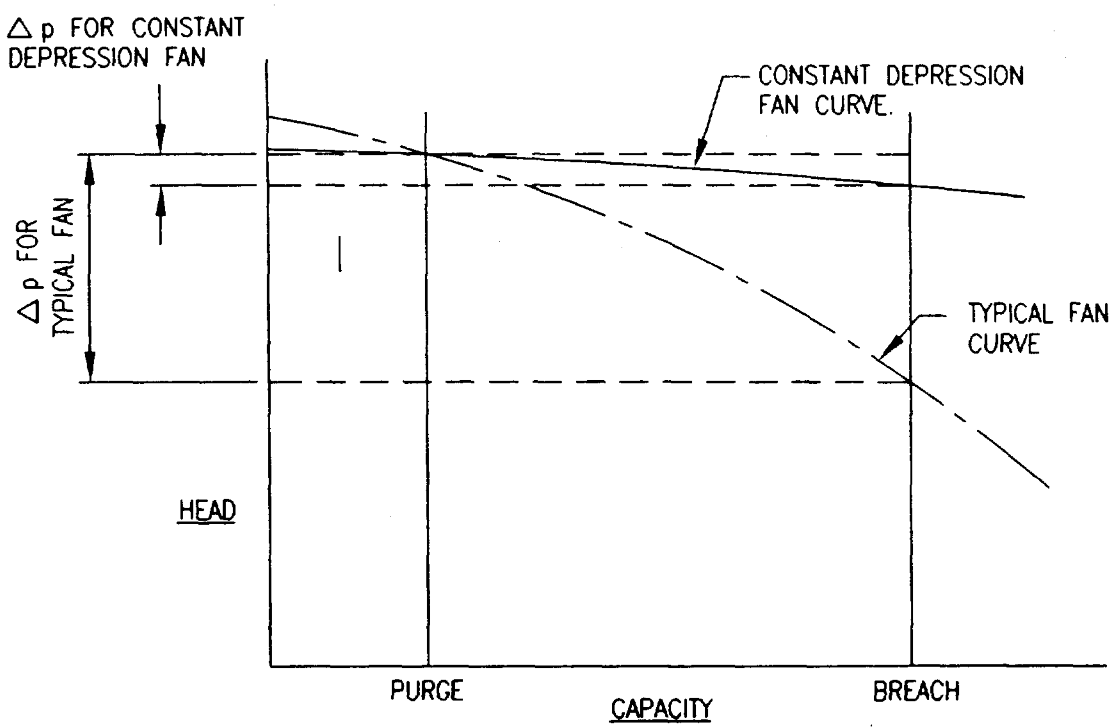


Figure 2 Characteristic of a constant depression fan

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).

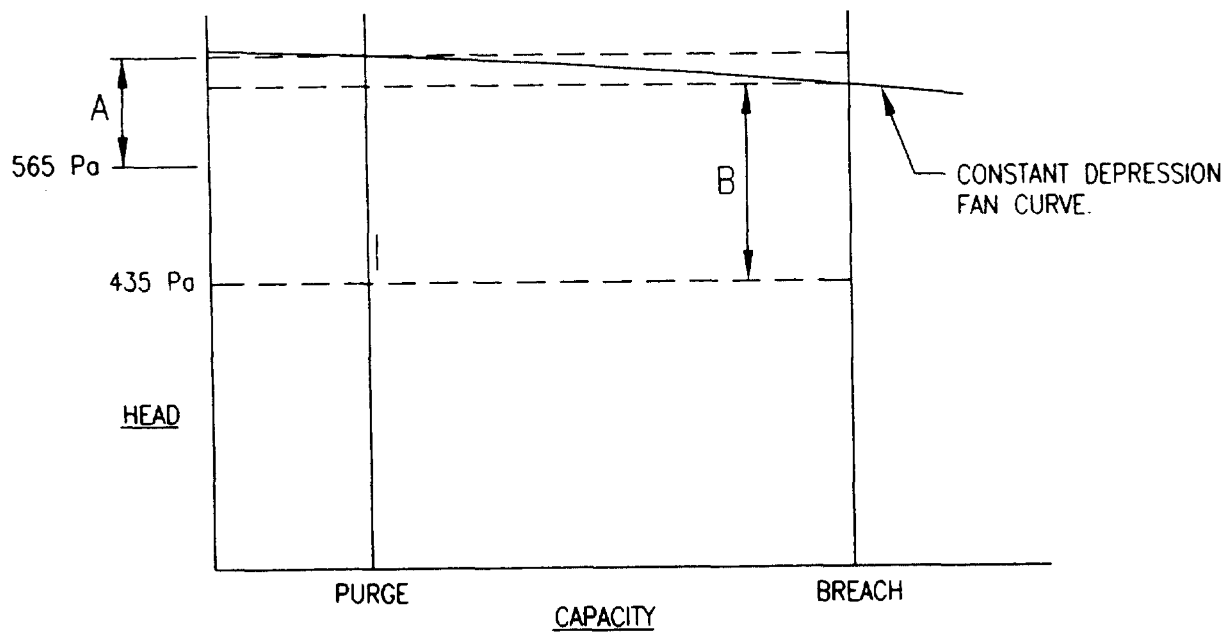


Figure 3 Fan/system characteristics
to achieve header depression tolerance

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

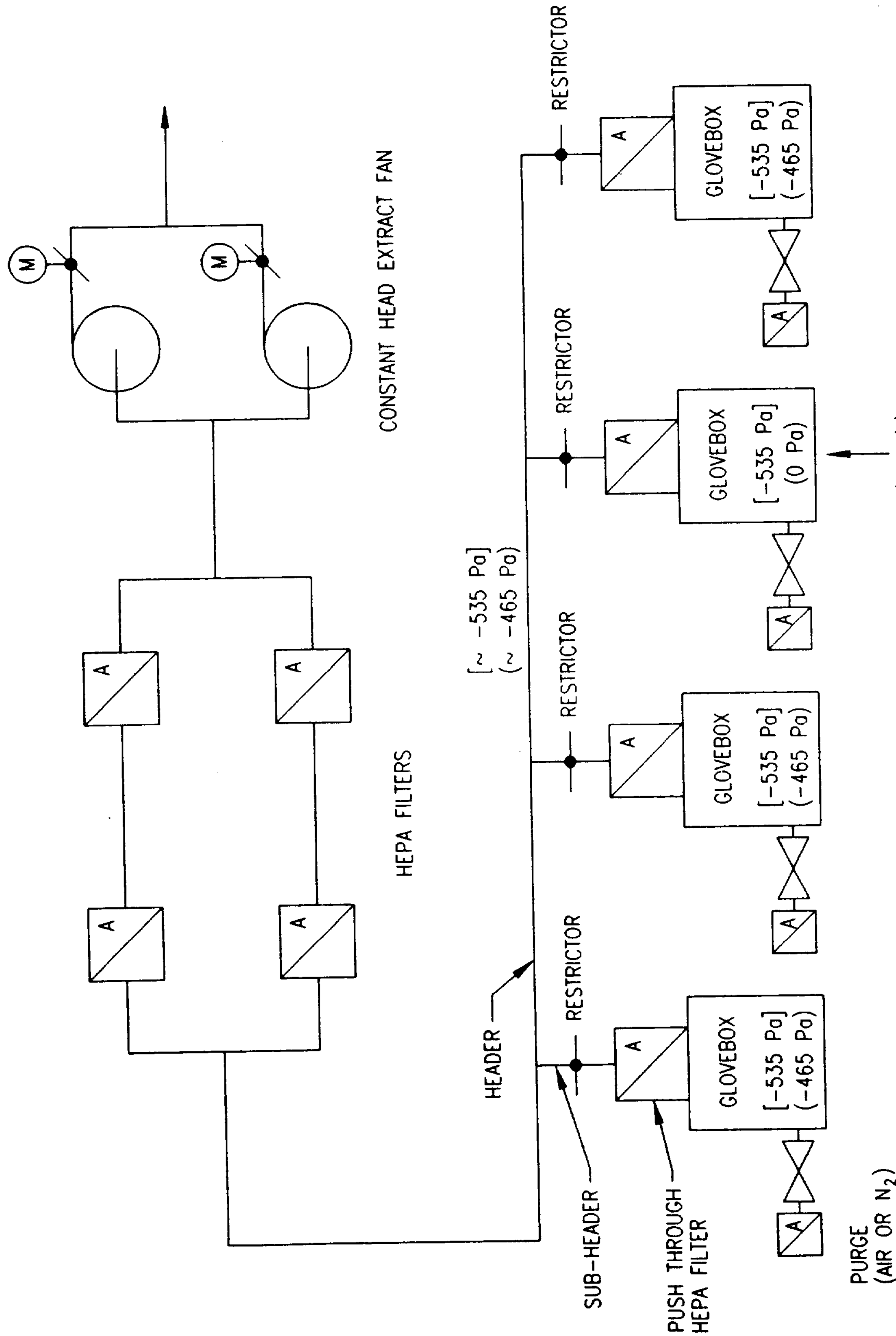
The extract fans must be fitted with a fan autochange facility. To initiate the autochange two parameters to detect fan failure should be used to avoid spurious functioning.

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

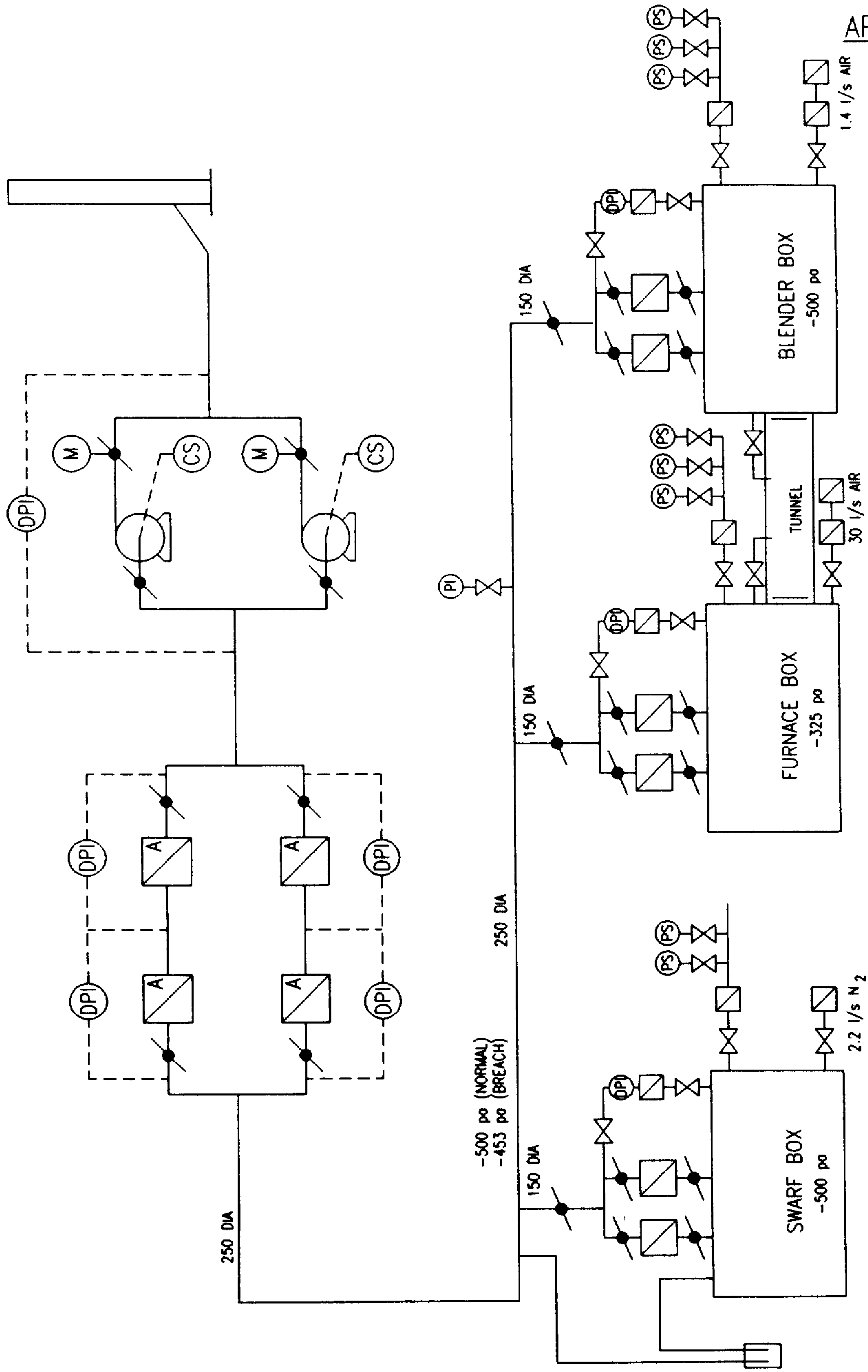
There is one system in operation which serves three gloveboxes, see Appendix B. One glovebox processes swarf under a nitrogen atmosphere with a purge rate of 2 l/s (5 cfm), another contains furnaces which is air purged at a rate of 30 l/s (63 cfm), the third a blender with an air purge of 1.4 l/s (3 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).

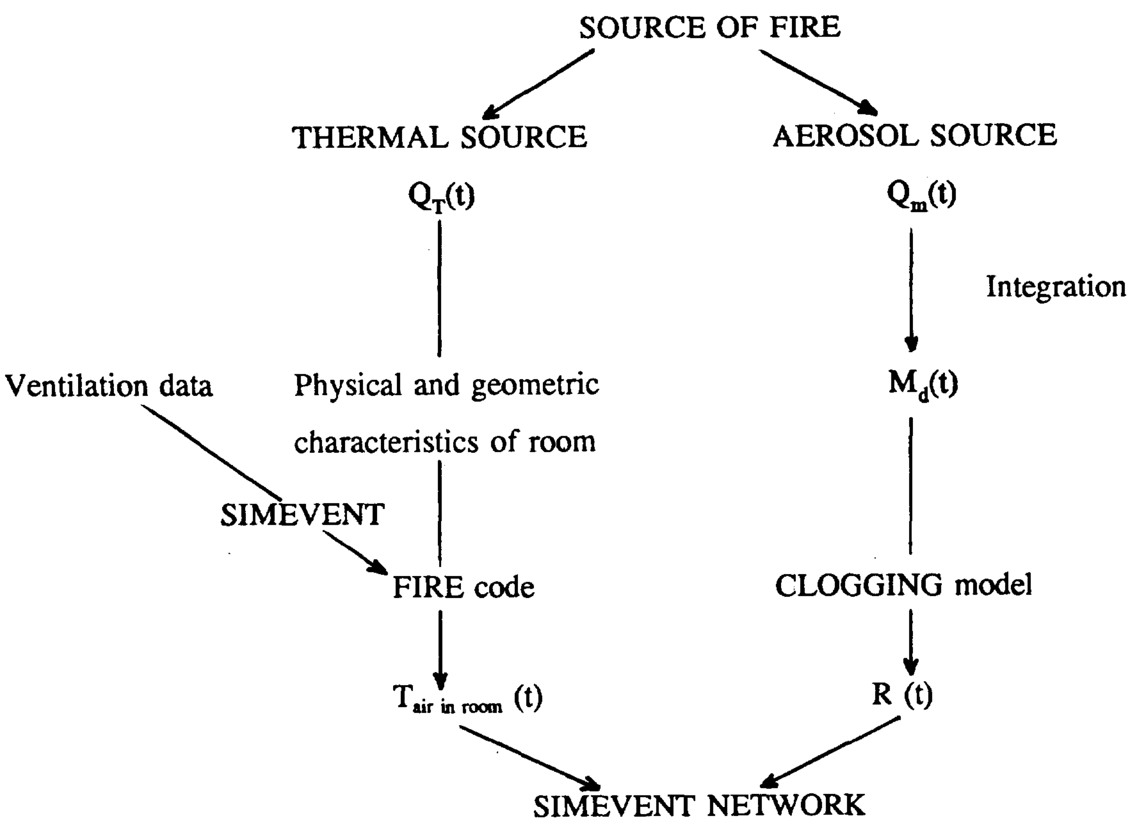
An extension to this code has been proposed to integrate the risk of fire and mass transfer (gases and aerosols) in the ventilation networks.

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.

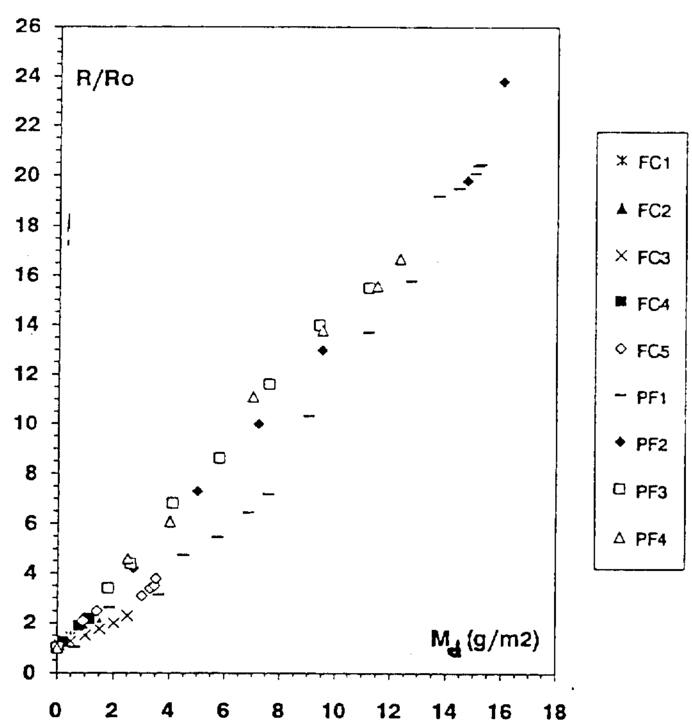


Figure 1 - Variation of R/R_0 as a function of M_d for PMMA fires

<u>Note</u>
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.

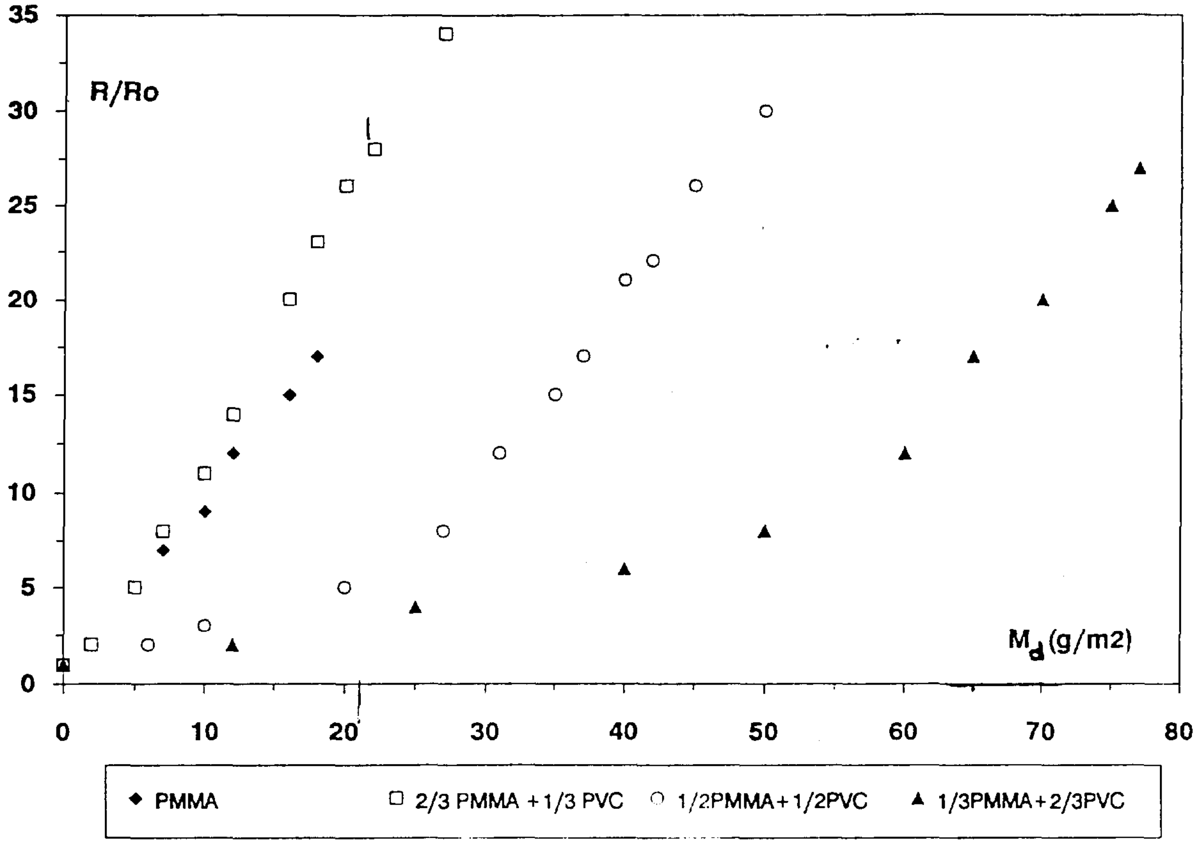


Figure 2 - Effect of the composition of the burned mixture on the variations of R/R_o

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/m^2)
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.

branch

C_e C_s

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 rate	- duct	- gas
- physical properties	- filter	- aerosol
	- register	
	- room	

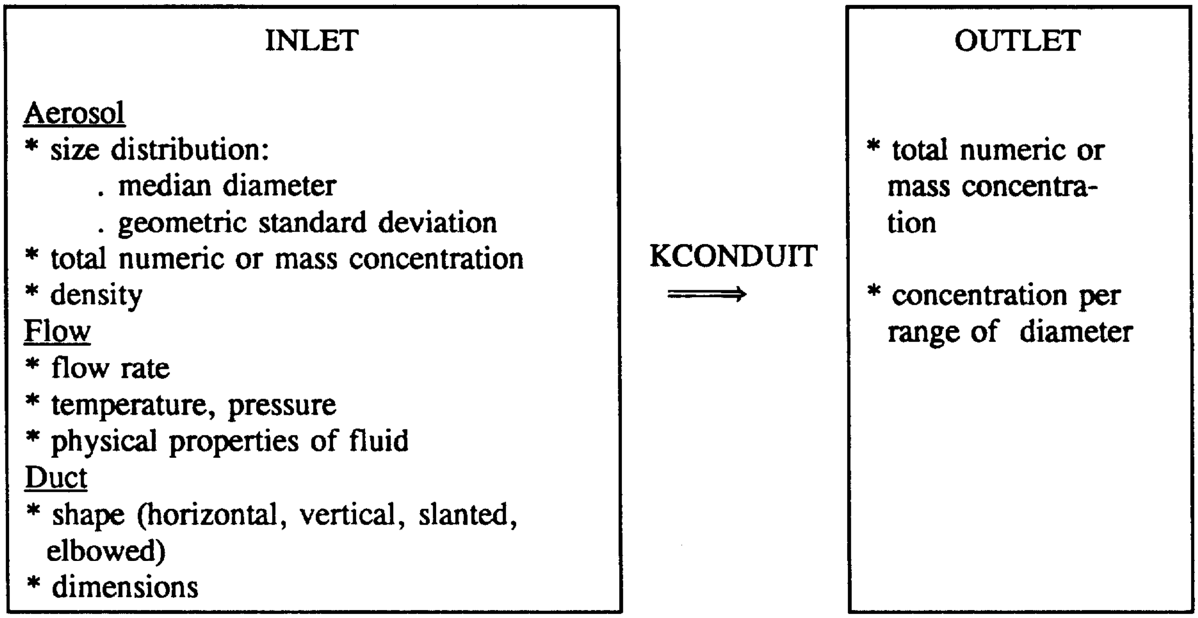
The deposit function can be entered either by the user or calculated by a specific model of the element forming the branch.

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.

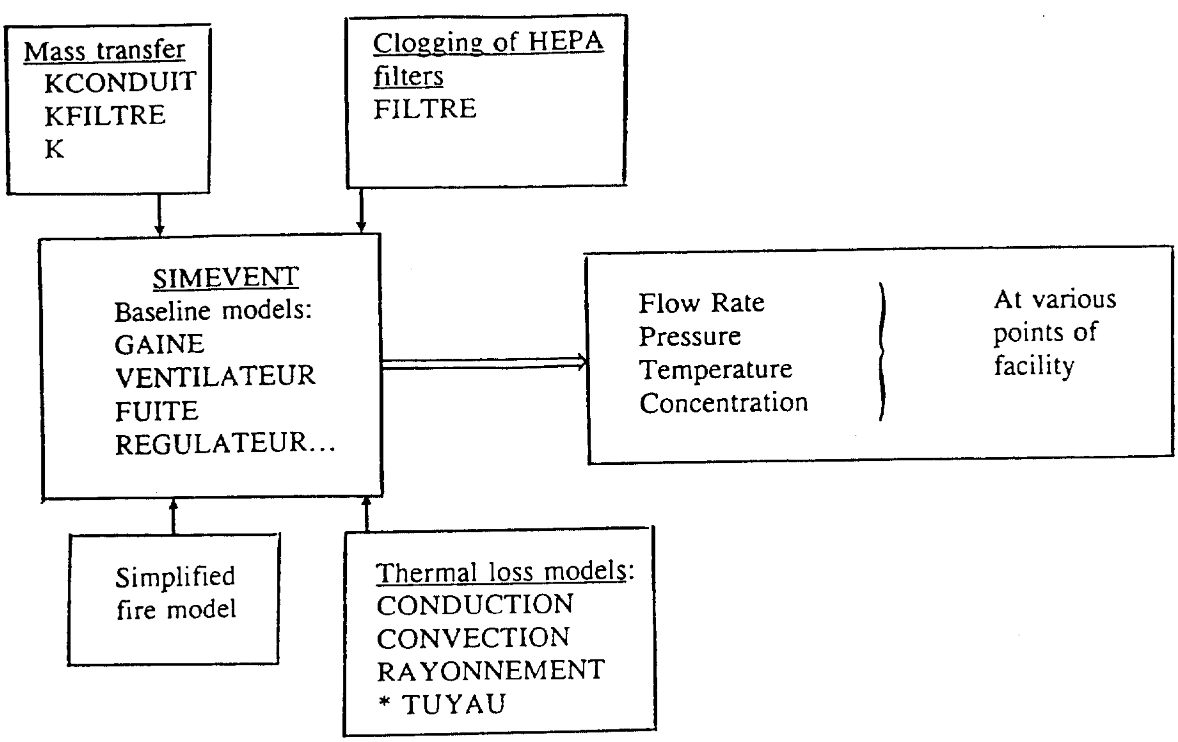
For example purposes, the input data and results provided by the KCONDUIT function are summarized below:



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

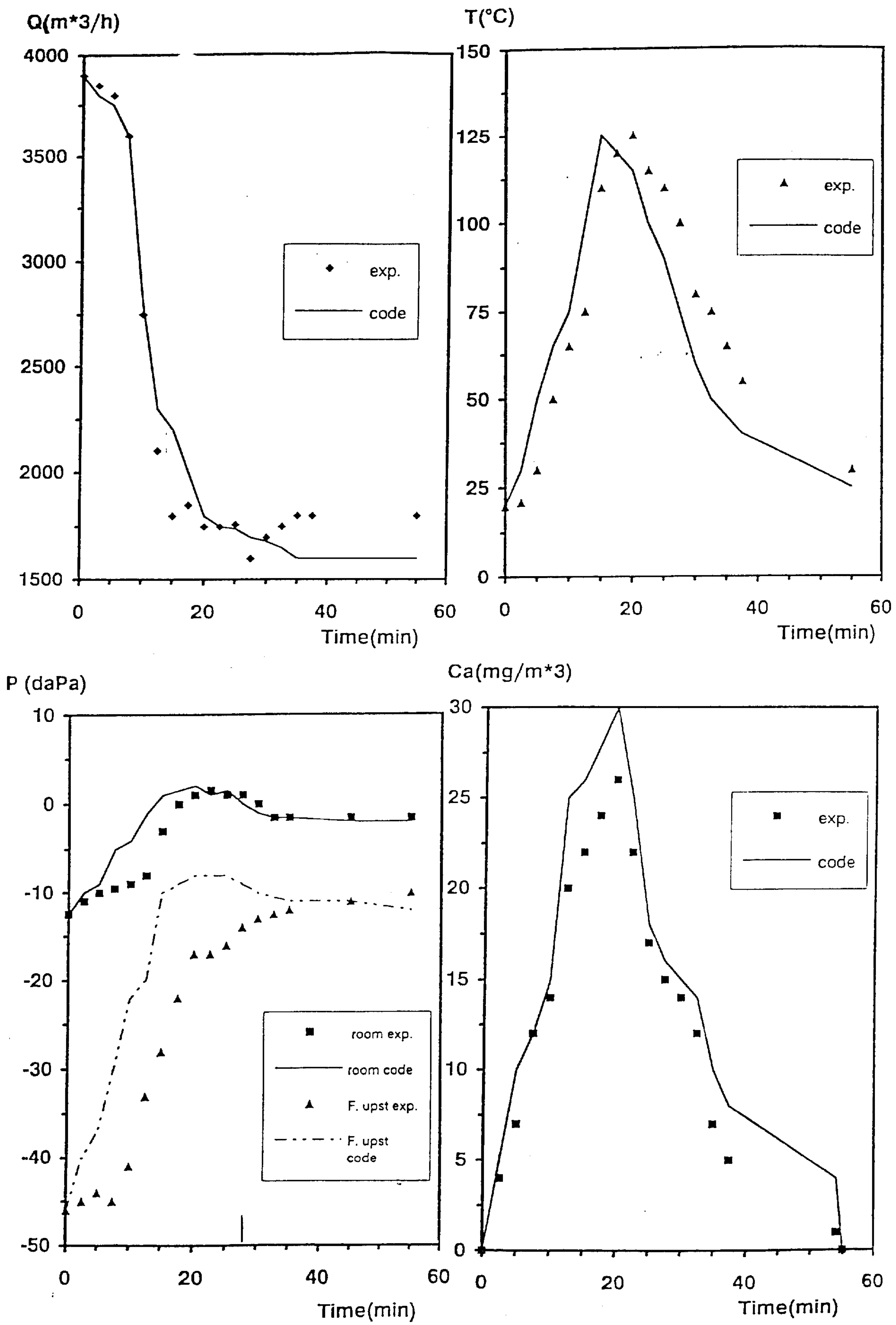
The evaluation of radioactive contamination confinement capacity requires knowledge of the parameters (in particular, temperature, pressure, pollutant concentration) defining the overall state of the ventilation network when a fire breaks out at a given point in the network.

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
exp.
code
0
20
40
60
Time(min)
T(°C)
150
125
100
75
50
25
0
exp.
code
0
20
40
60
Time(min)
P (daPa)
10
0
-10
-20
-30
-40
-50
room exp.
room code
F. upst exp.
F. upst code
0
20
40
60
Time(min)
Ca(mg/m*3)
30
25
20
15
10
5
0
exp.
code
0
20
40
60
Time(min)

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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

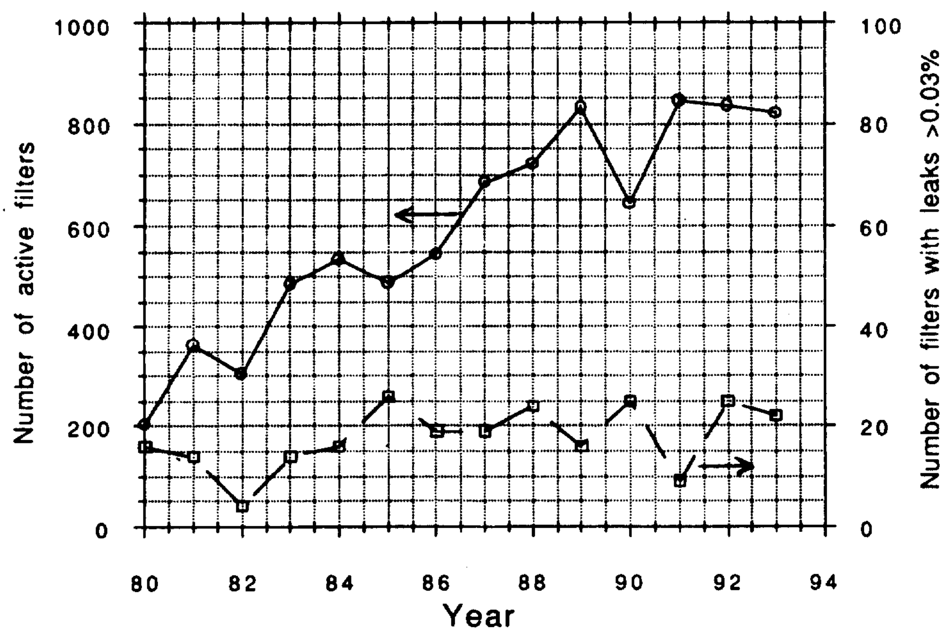

Figure 1 Graph of total number of active filters and filters with uncorrectable leaks

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% respéctively. The potential contributions to the filter leaks from the 1980 and 1989 earthquakes are included in the data in Figure 2.

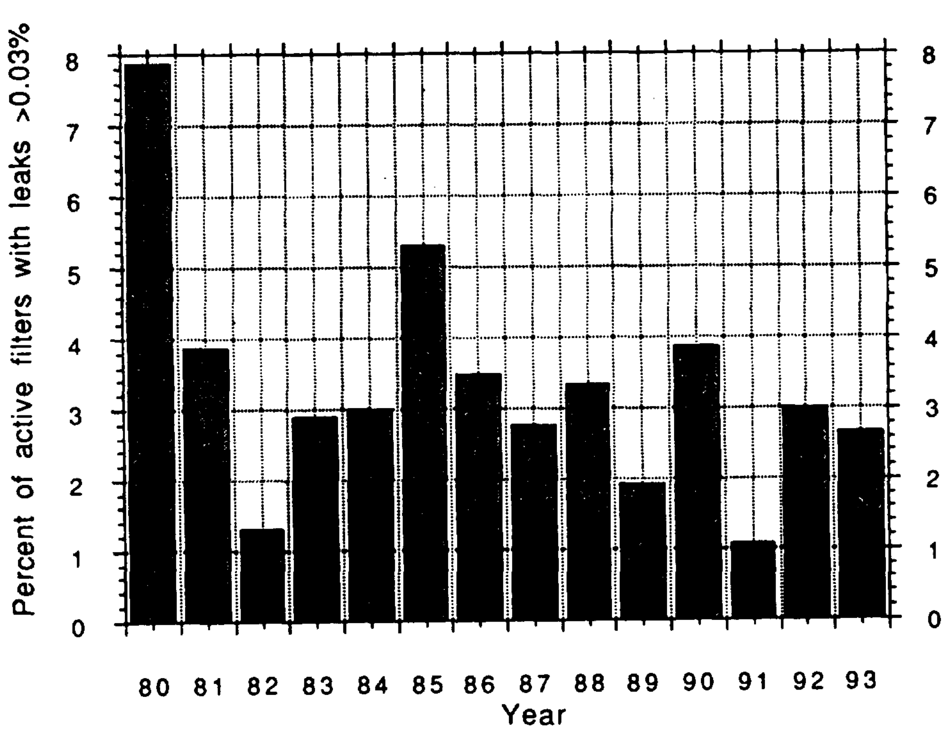

Figure 2 Percent of active filters that have leaks greater than 0.03%

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.

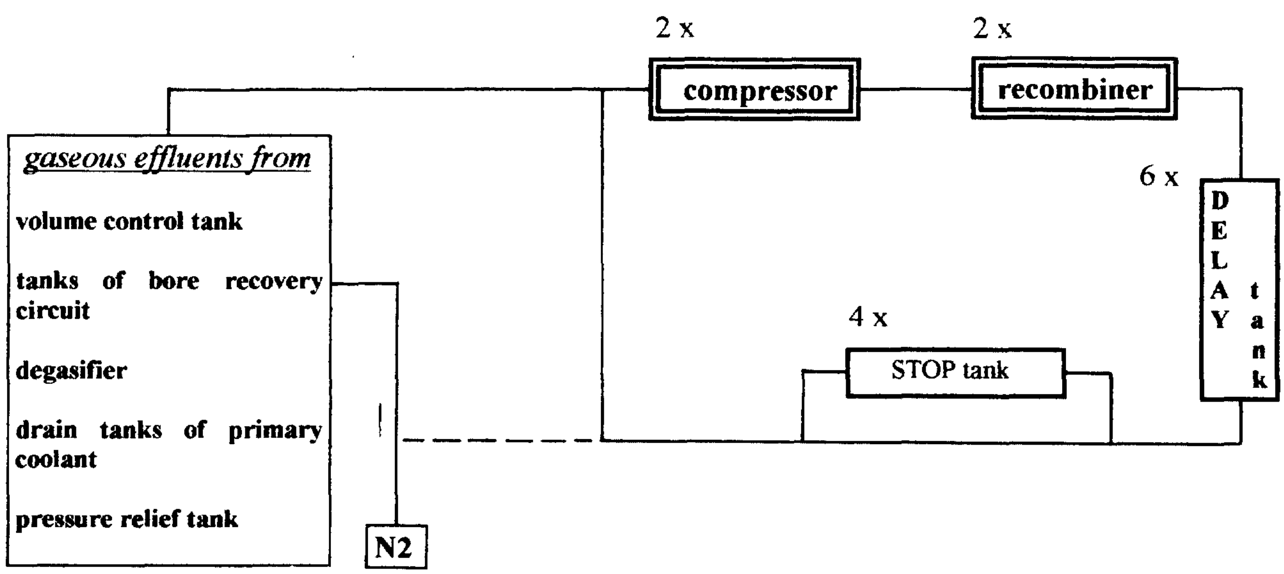

Figure 1 Block diagram of gas recycling loop of DOEL-4.

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.

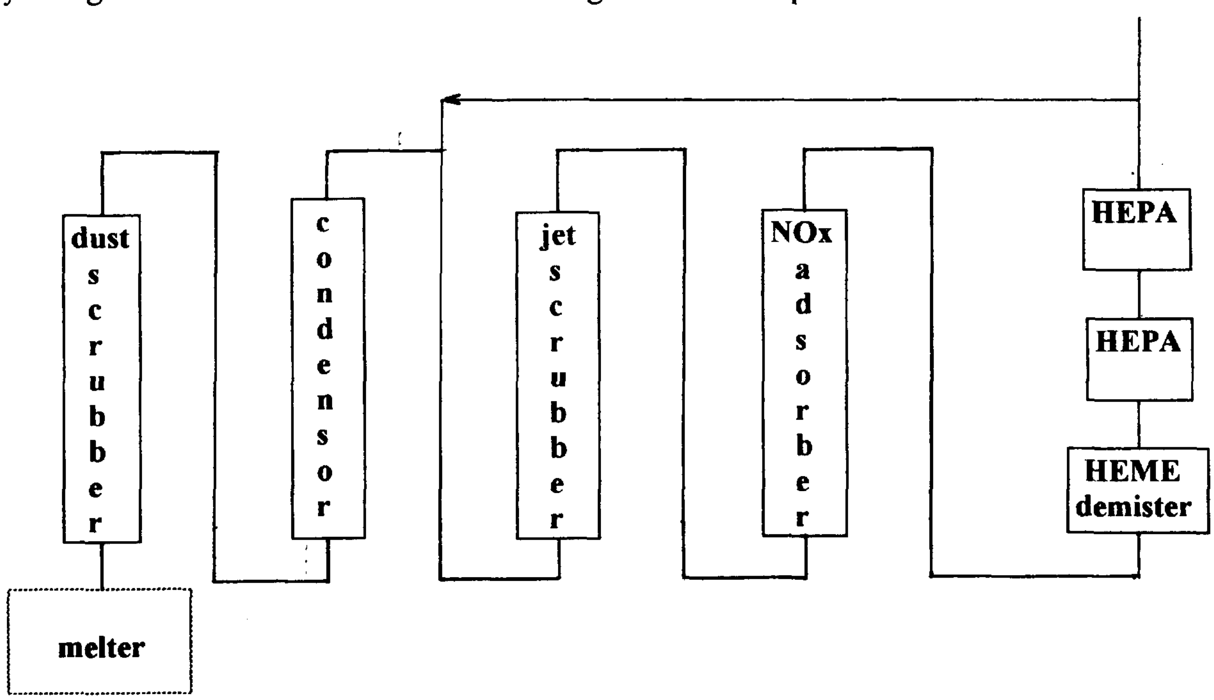

Figure 2 Block diagram of the PAMELA off-gas treatment.

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 cleaning 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

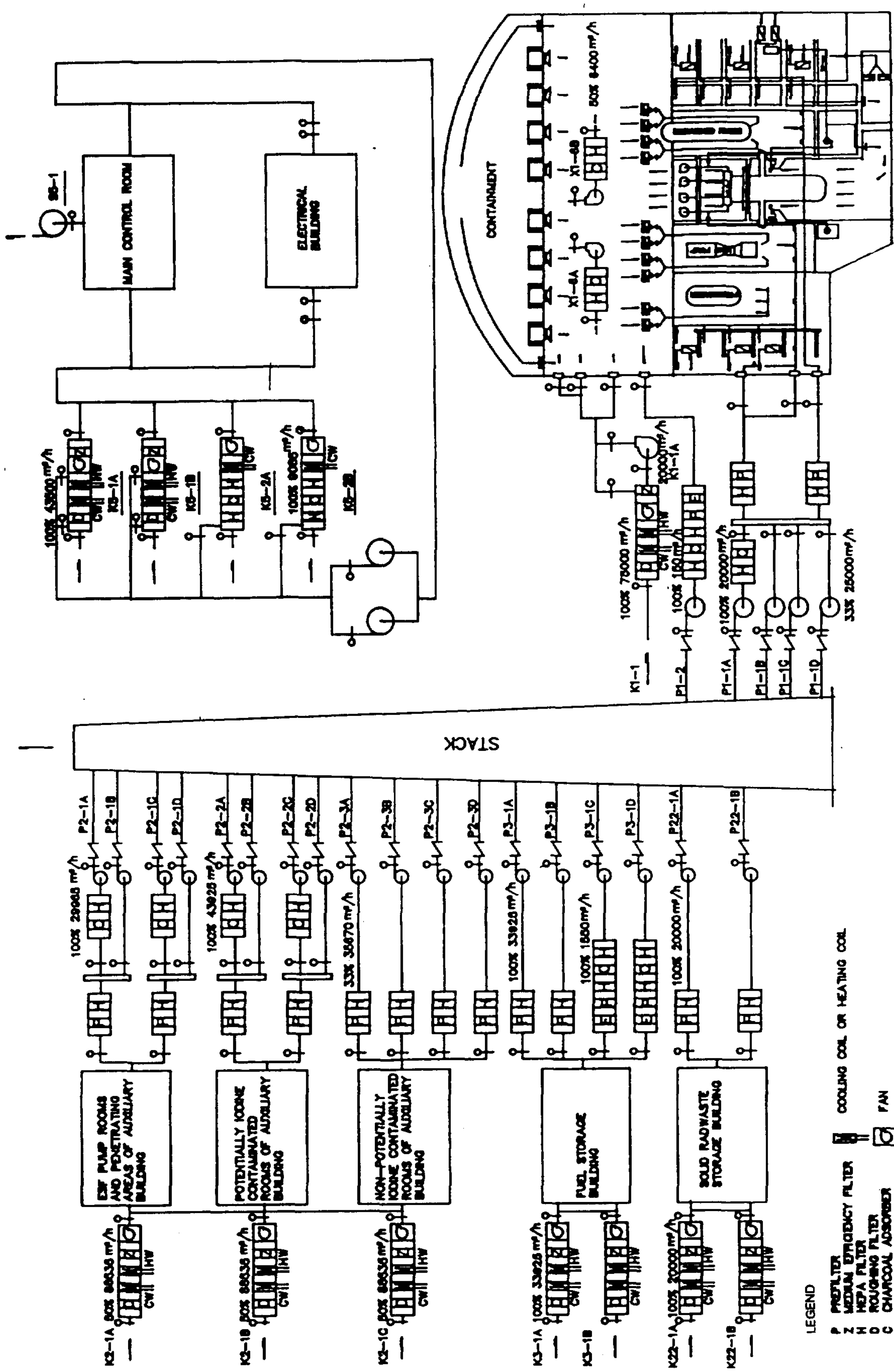

FIGURE 1 VENTILATION SYSTEMS SCHEMATIC DIAGRAM

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 filter 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 control 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

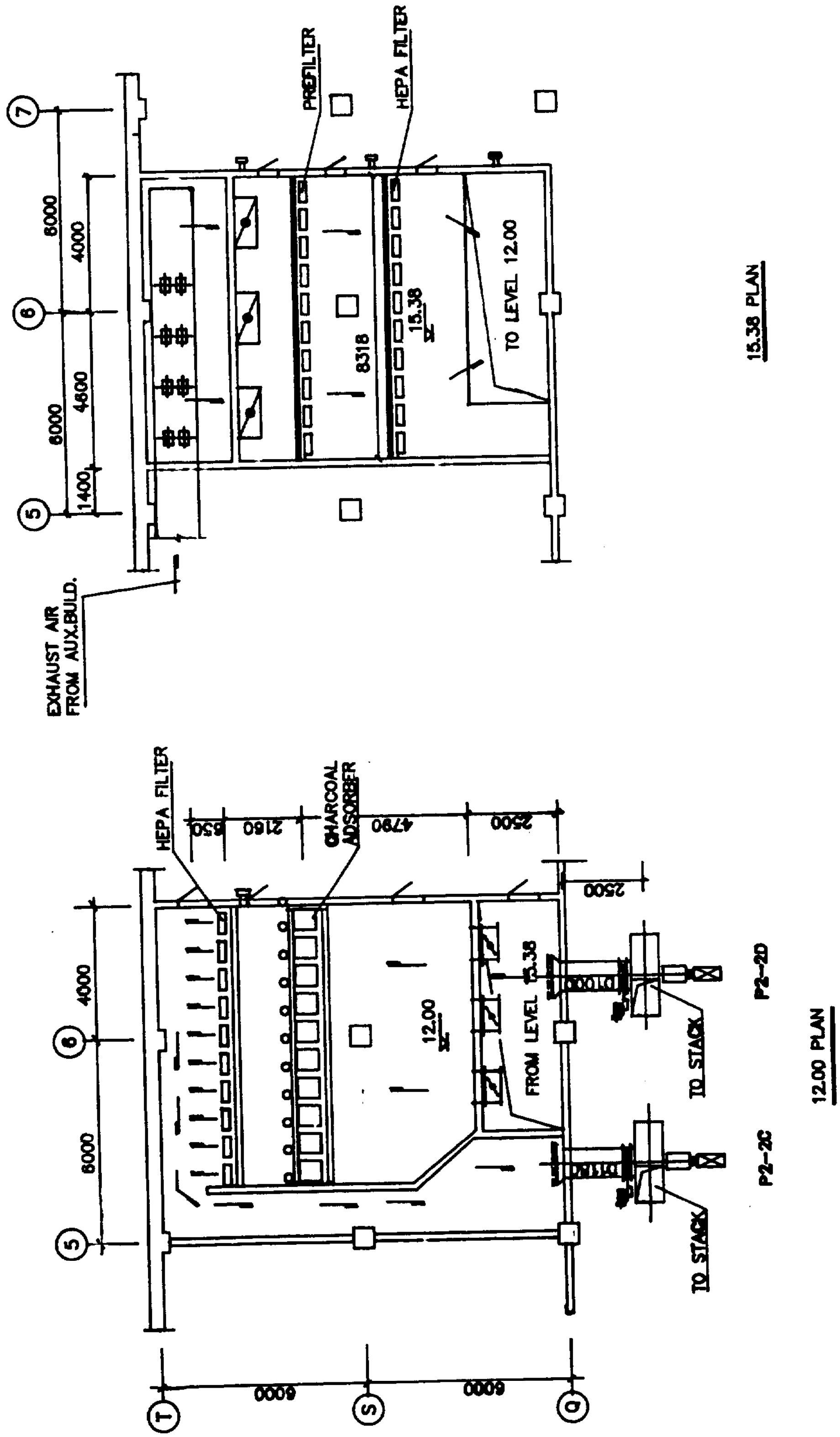

FIGURE 2 AUXILIARY BUILDING AIR CLEANING UNIT LAYOUT P2-2C(D)

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 adsorber 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 $\eta=70\sim 80\%$ 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.

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).

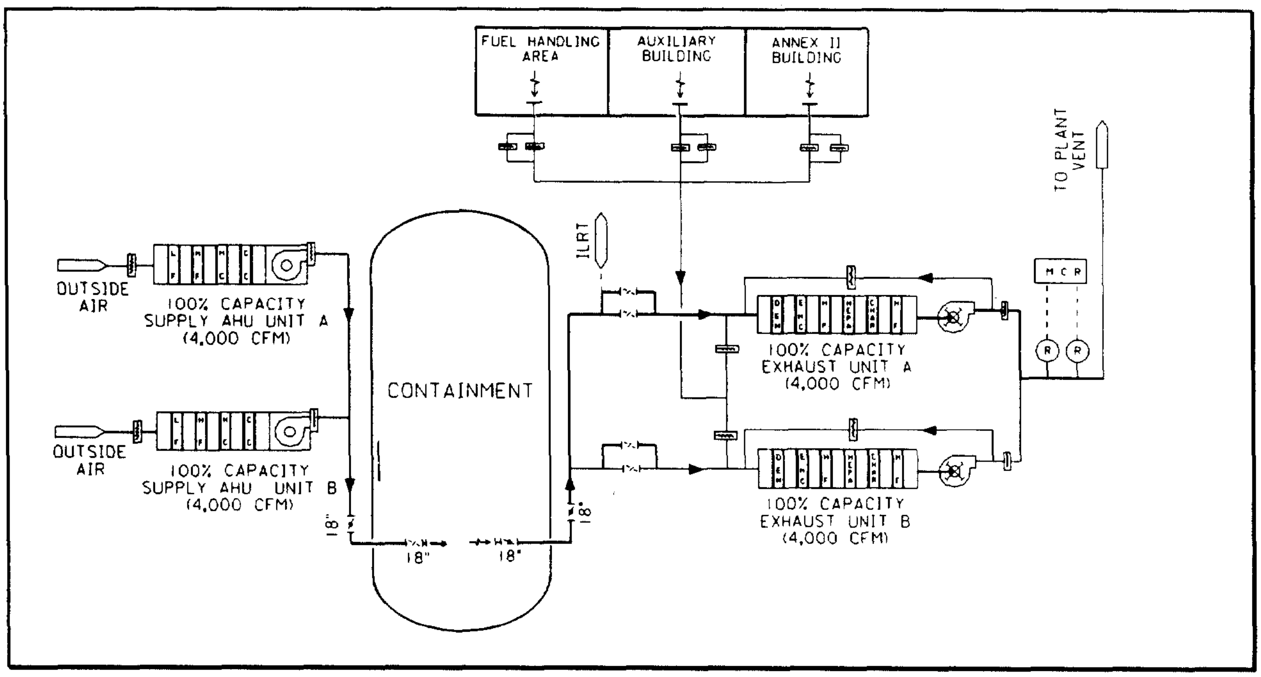

Figure 1 - AP600 Containment Air Filtration System (VFS)

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.

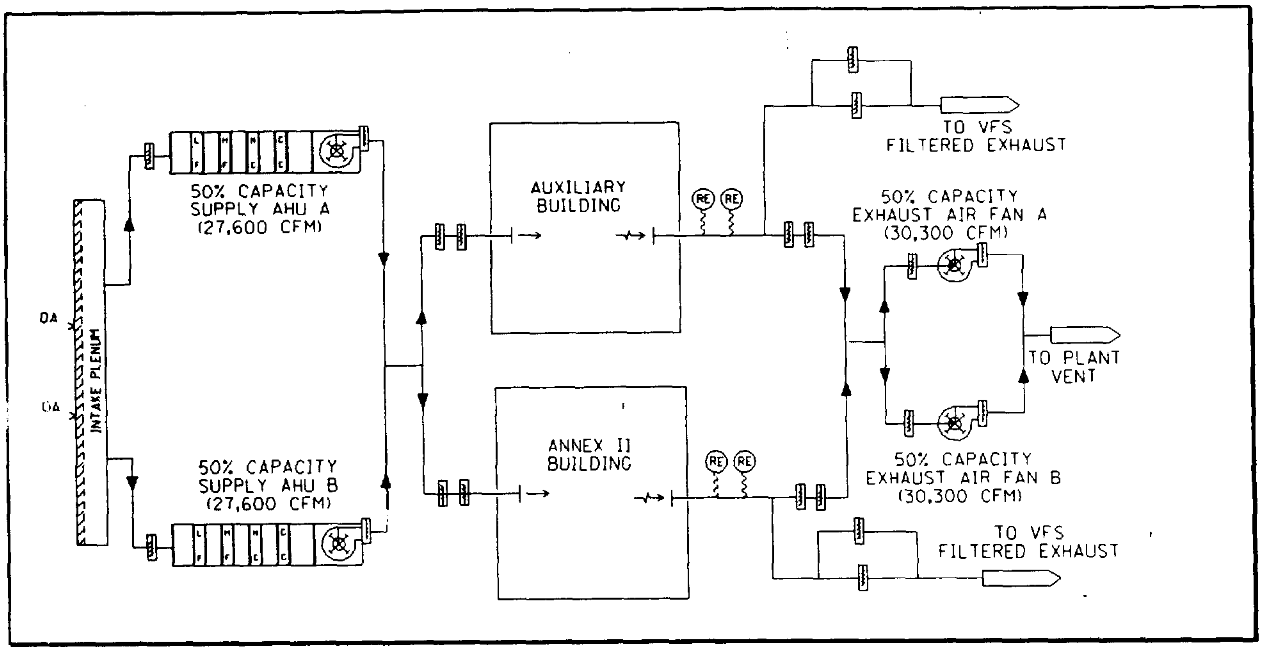

Figure 2 - Auxiliary Building HVAC Subsystem

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.

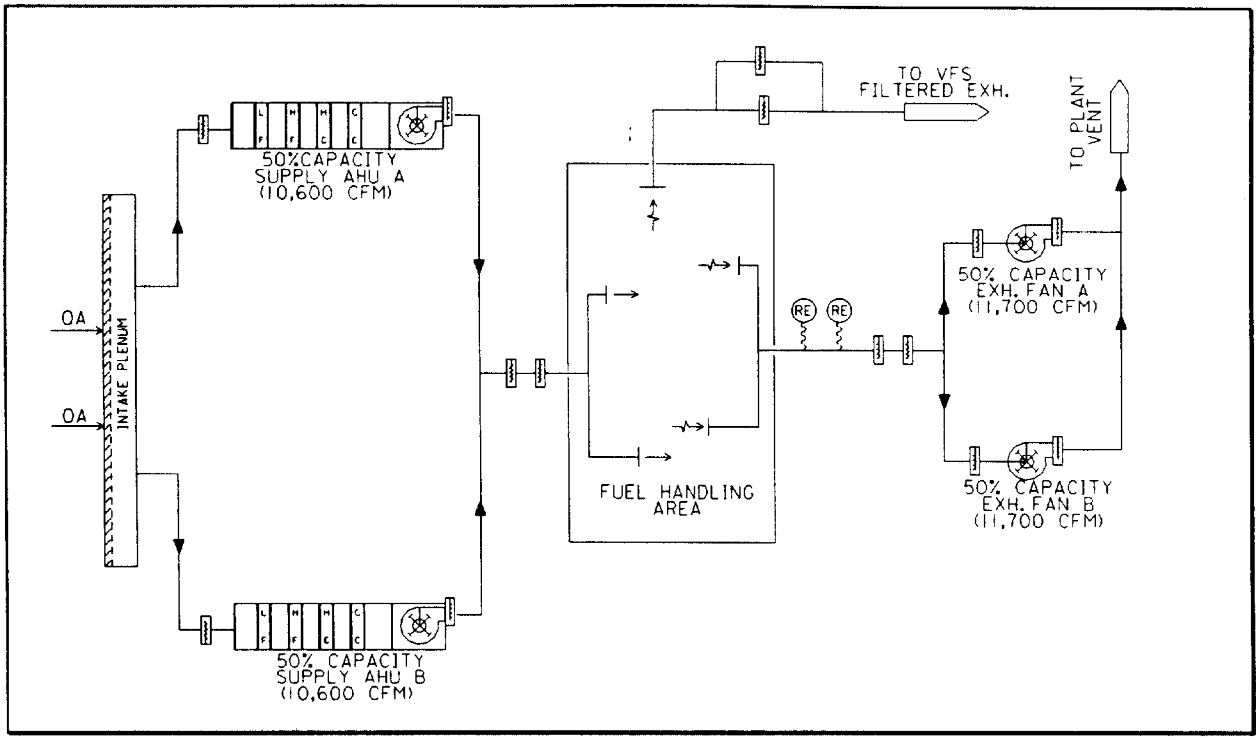

Figure 3 - Fuel Handling Area Subsystem

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:

$$\frac{dA^i(t)}{dt} = R^i - \lambda_r^i A^i(t) \tag{1}$$

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:

$$\lambda_r^i = \lambda_d^i + \lambda_p^i + \frac{F_p}{V_c} + \frac{F_k \varepsilon^i}{V_c} \tag{2}$$

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:

$$C^i(t_2) = \frac{R^i}{\lambda_r^i V_c}\left(1 - e^{-\lambda_r^i(t_2 - t_1)}\right) + C^i(t_1) e^{-\lambda_r^i(t_2 - t_1)} \tag{3}$$

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:

$$R_n^i = l_c \cdot f_f \cdot C_c^i + A_n^i \tag{4}$$

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.

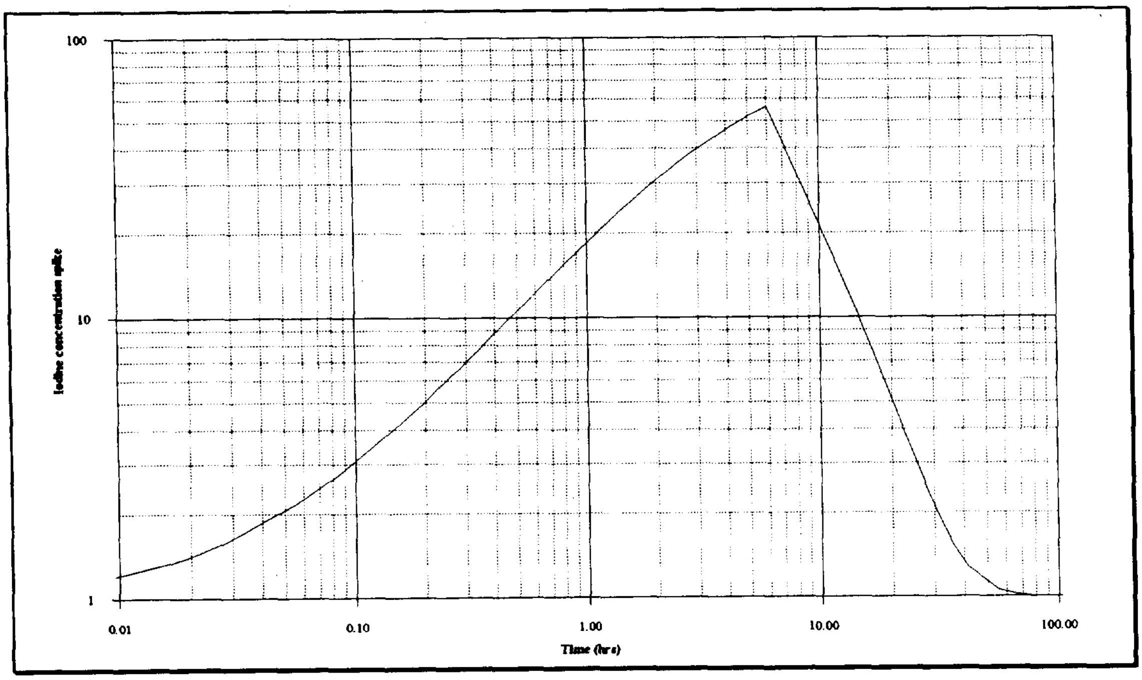

Figure 4 – Iodine Spiking Factor

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.

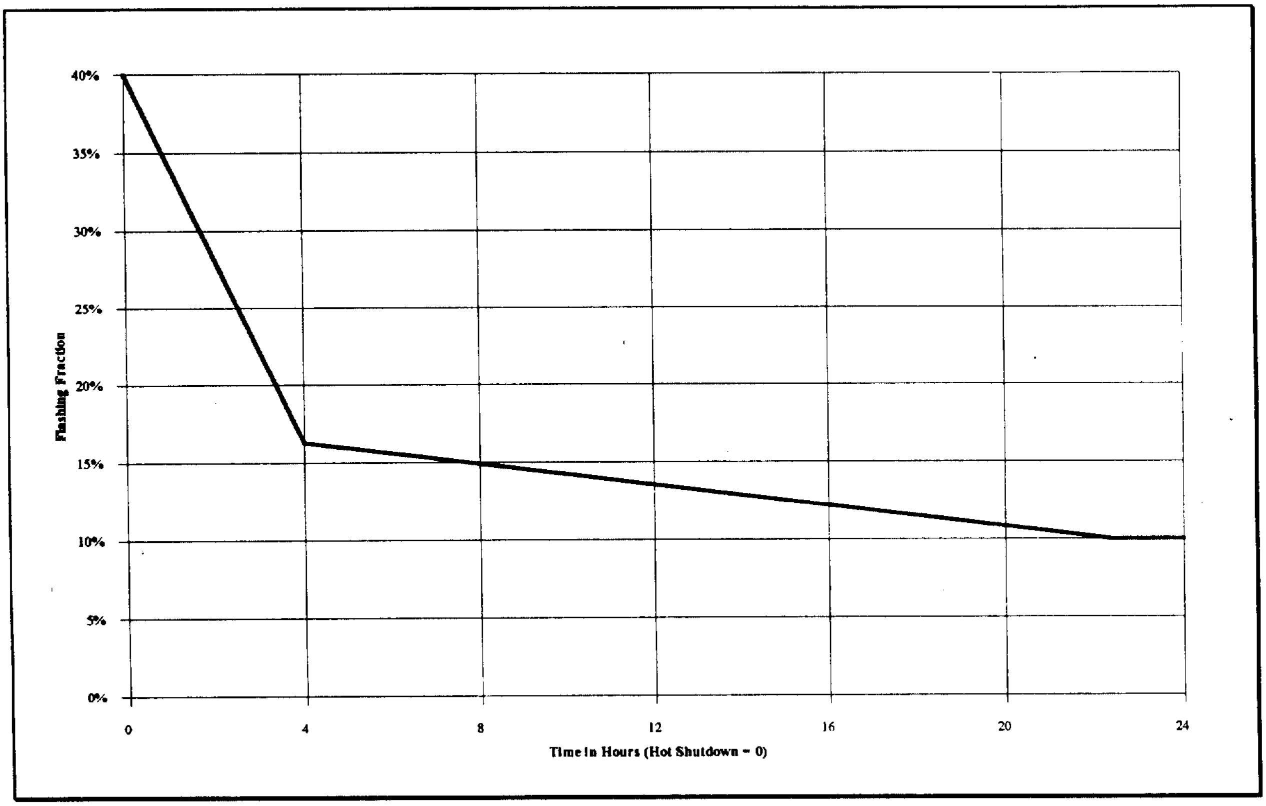

Figure 5 – Flashing Fraction Time History

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 (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	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

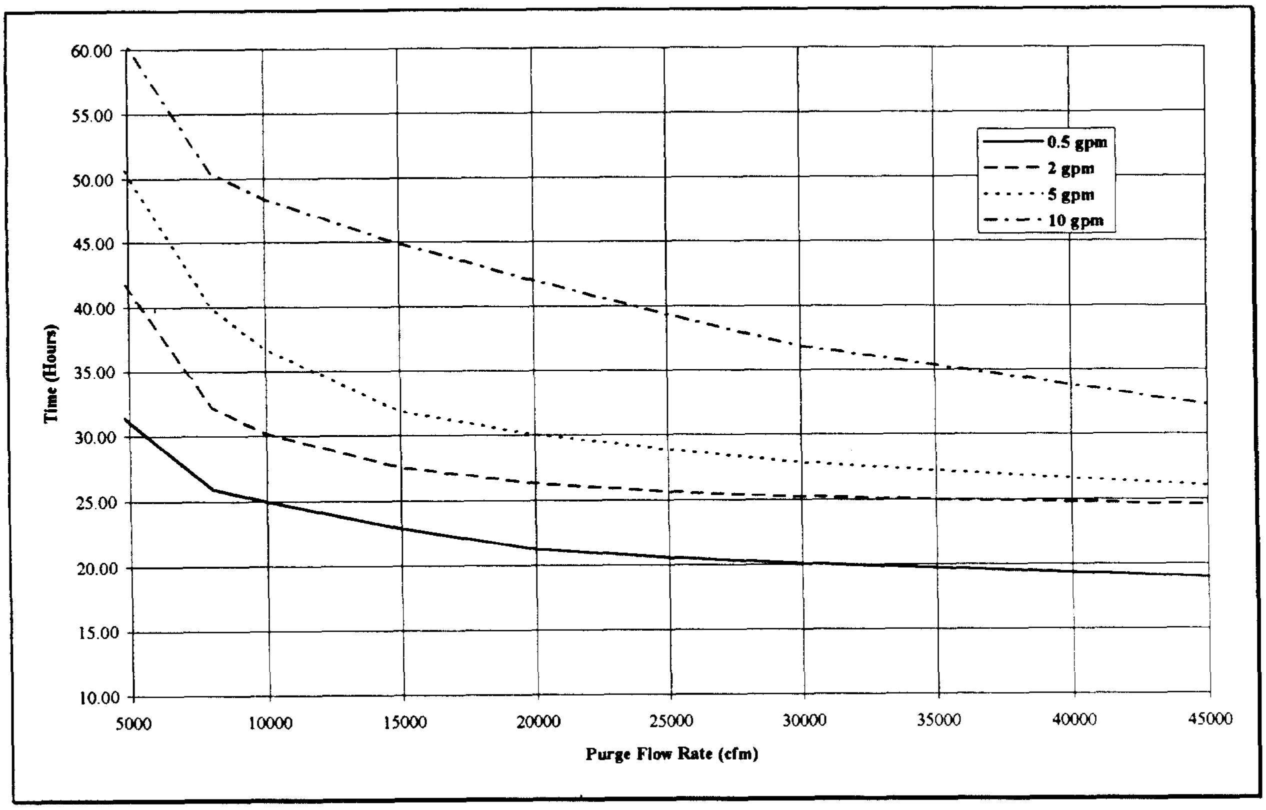

Figure 6 – Time to Reach 0.5 DAC for Various Shut-Down Purge Rates beginning at Hot Shutdown

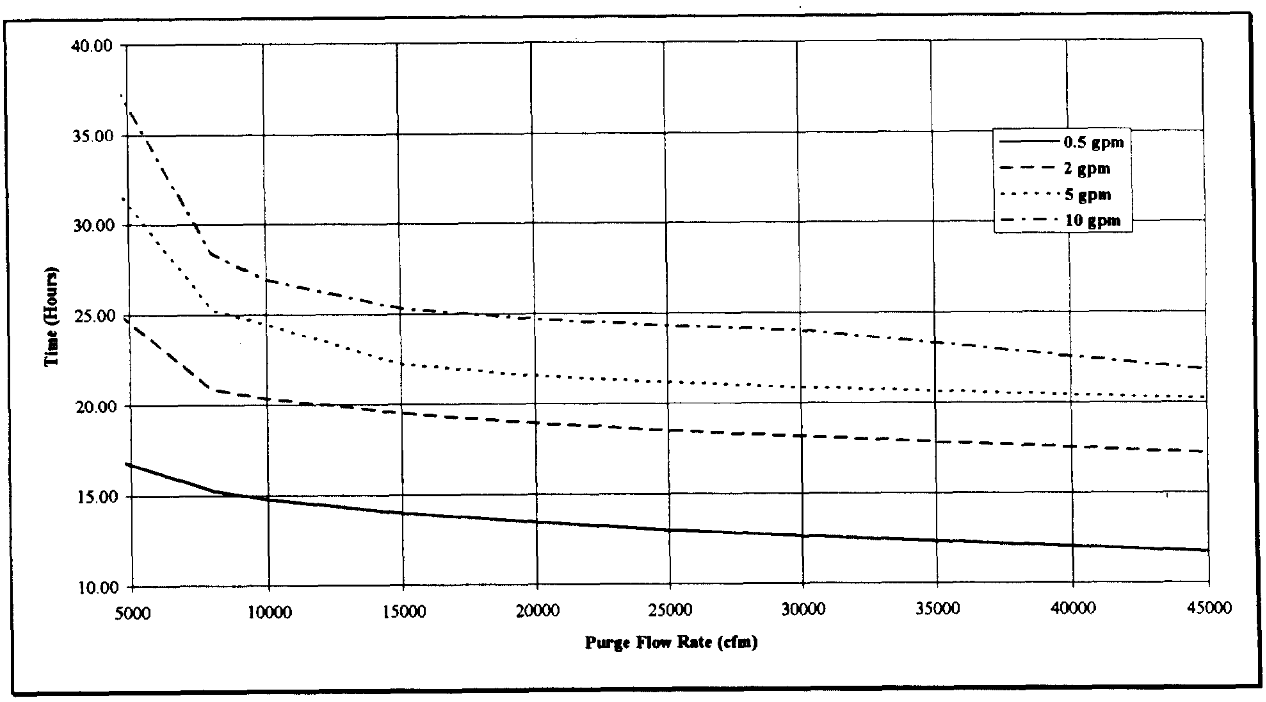

Figure 7 – Time to Reach TEDE Limit for Various Shut-Down Purge Rates beginning at Hot Shutdown

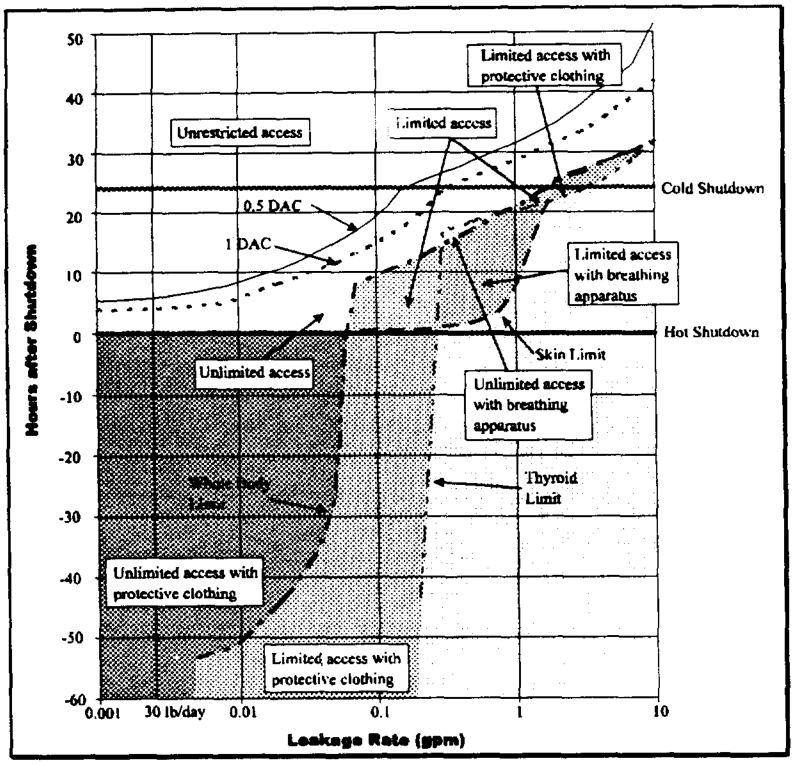

Figure 8 - Time History as a Function of RCS Leak for Post Shutdown Purge of 8,000 cfm

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.

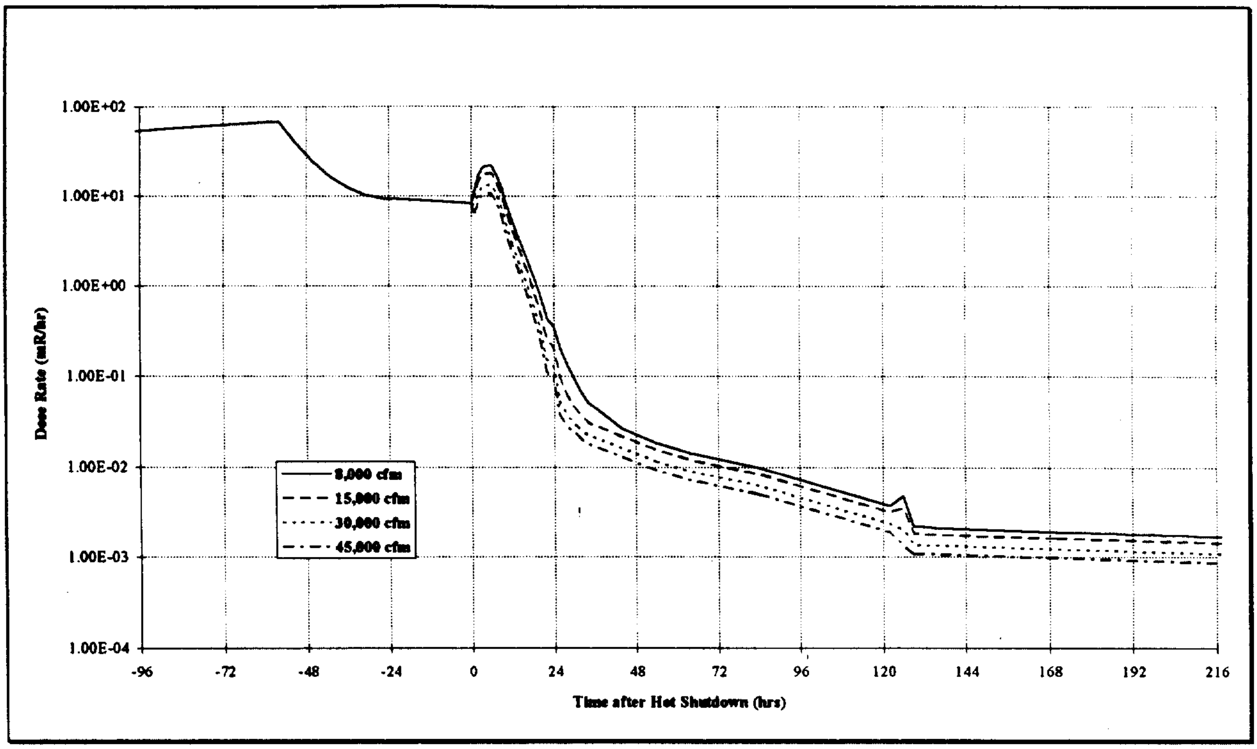

Figure 9 – TEDE Dose Rate as a Function of Time for 0.5 gpm RCS Leakage

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).

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.

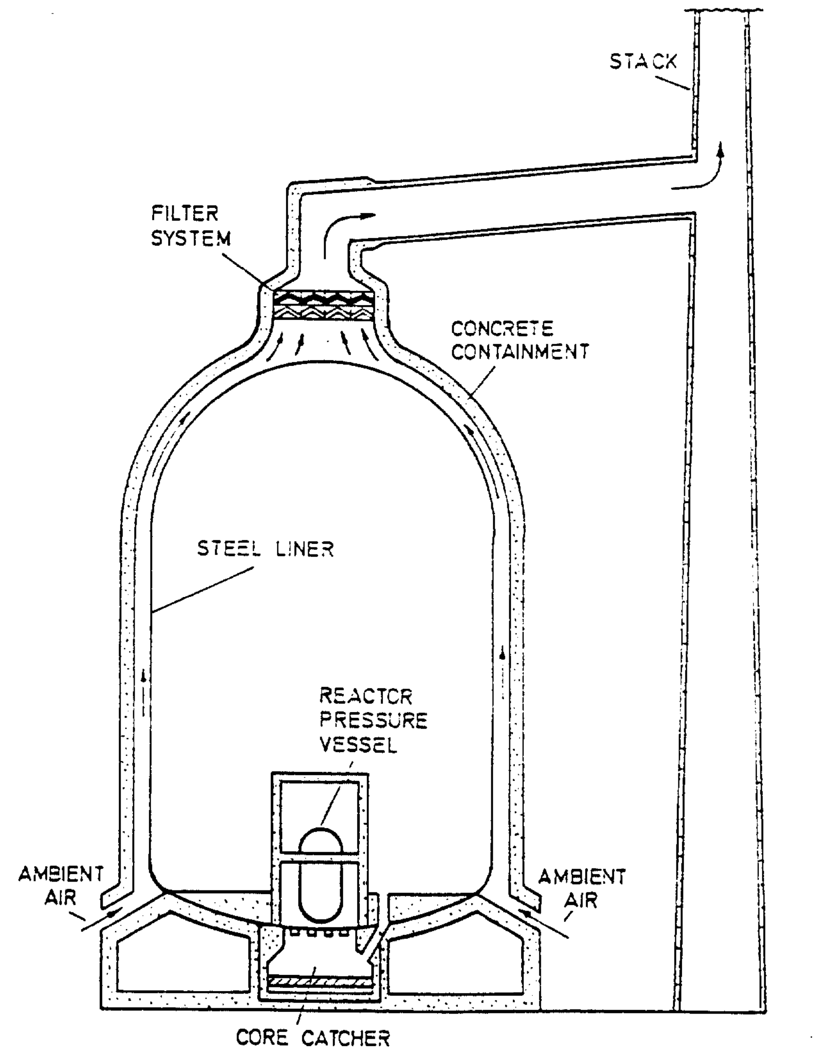

FIG. 1

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.

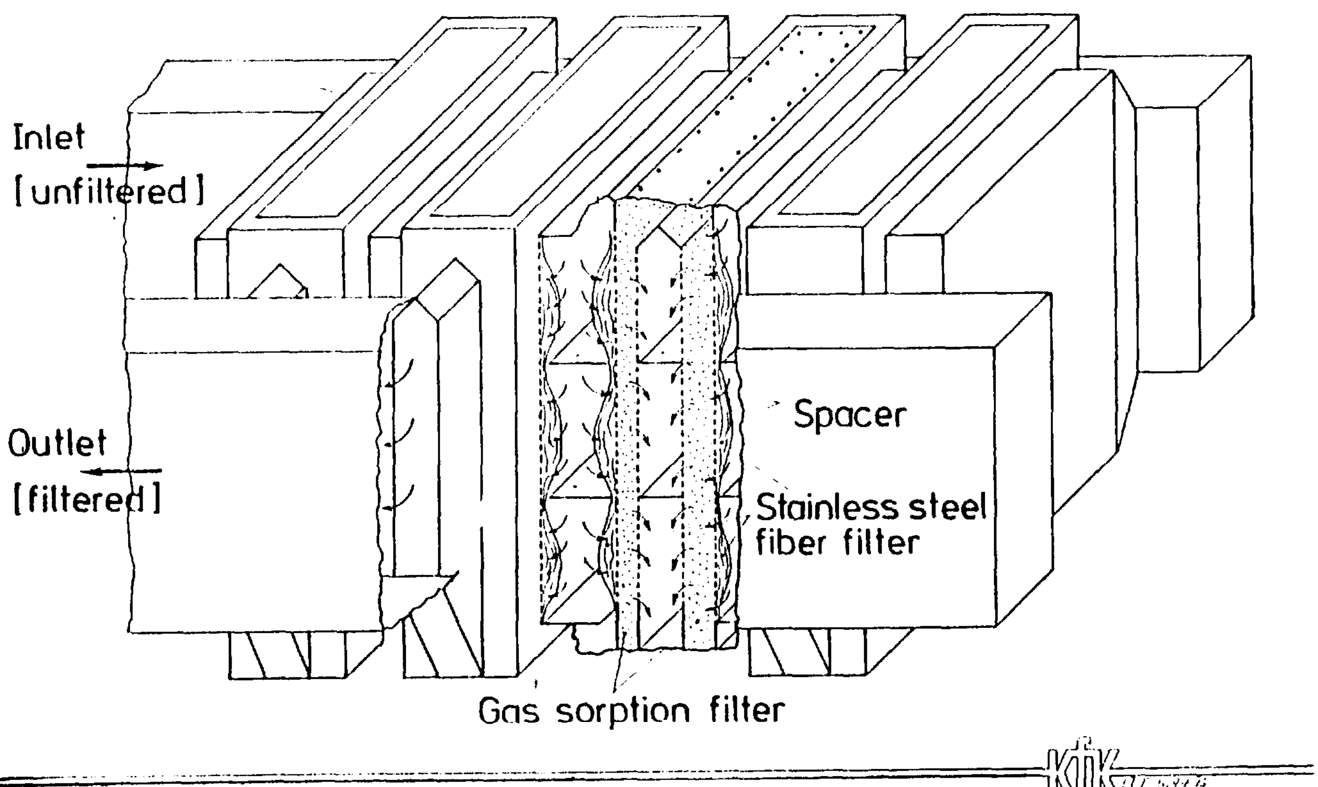

FIG. 2 Low pressure drop filter

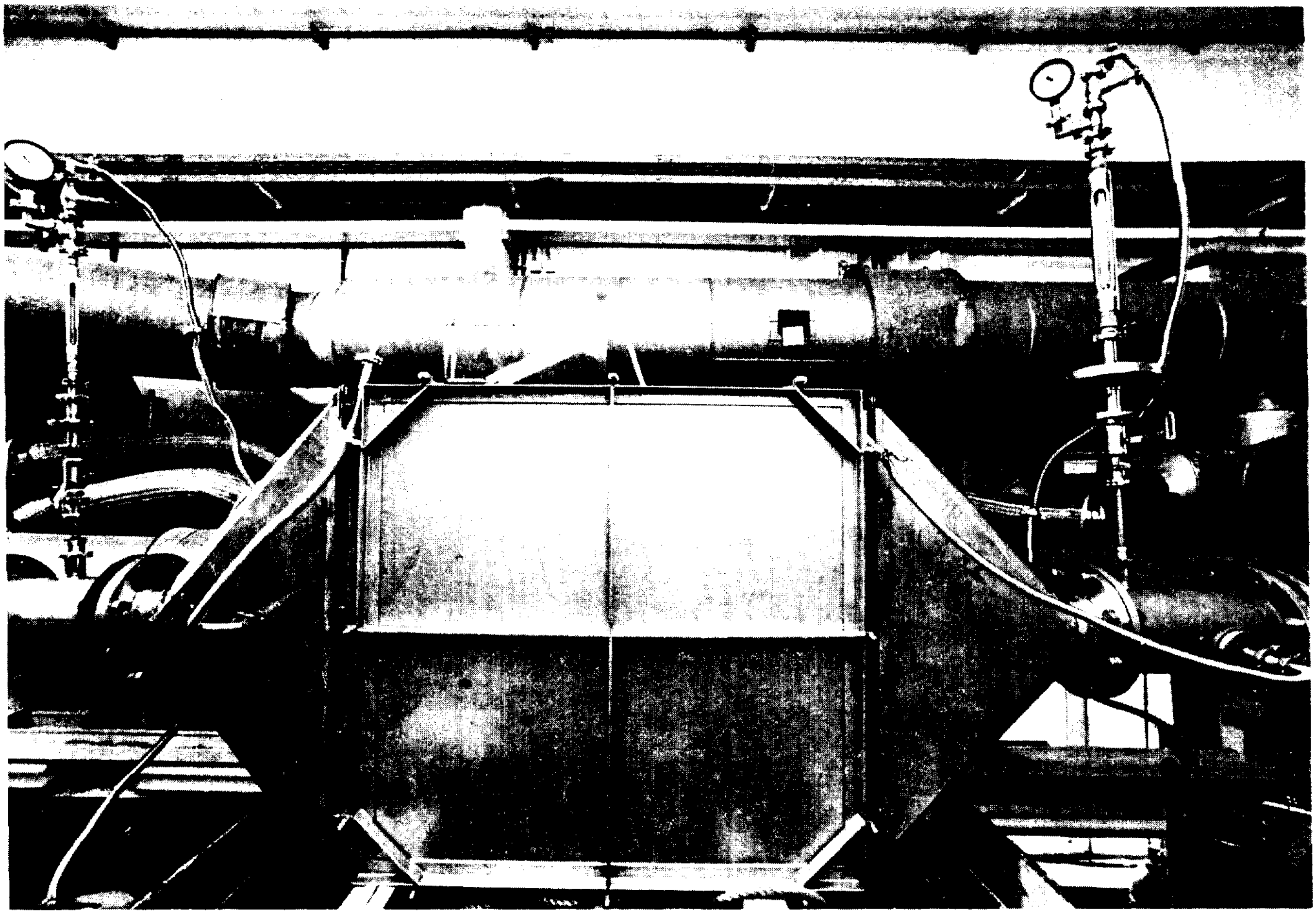

FIG. 3 The installed test filter module in the test rig

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.

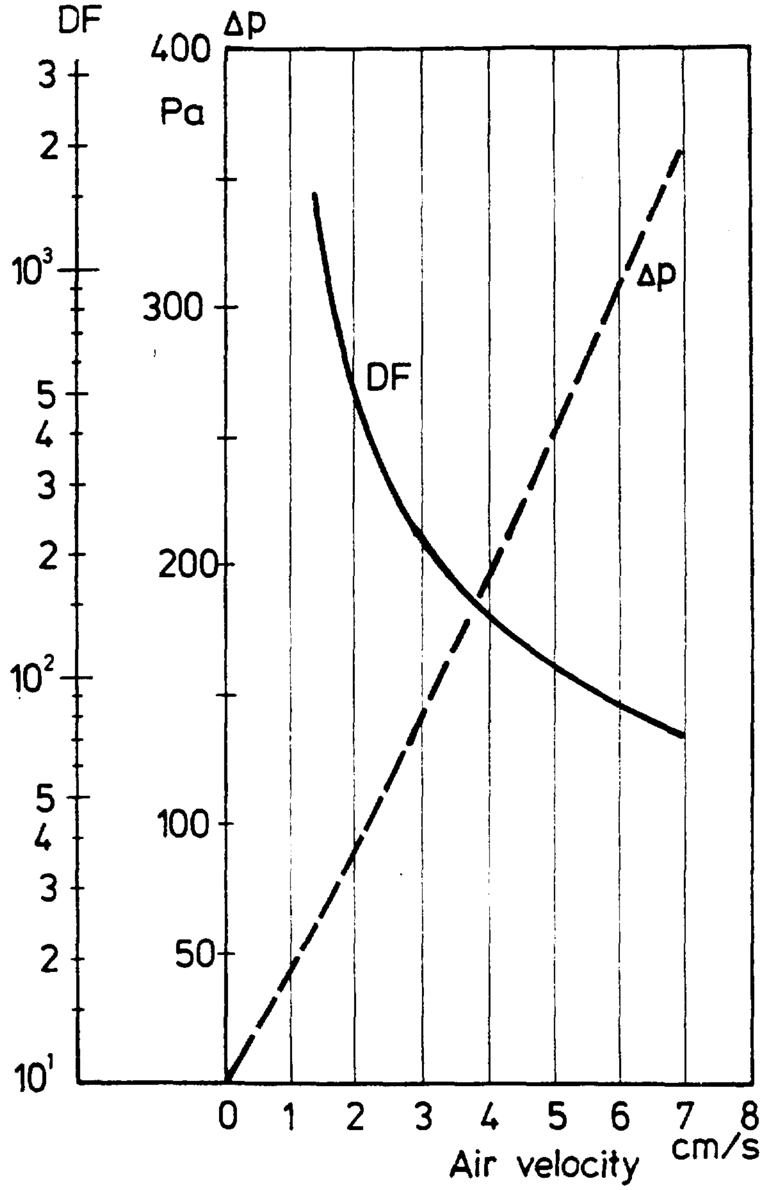

FIG. 4 DF and Δp as functions of the air velocity of a stainless steel fiber fleece filter fiber ∅ 2 um, 600 g/m² sintered

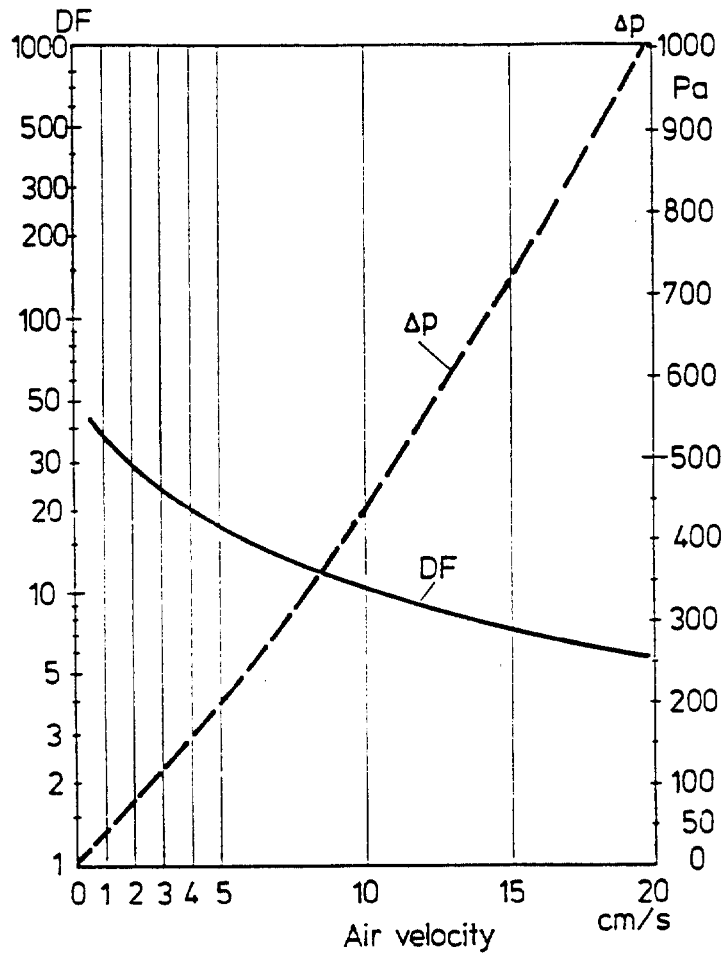

FIG. 5 DF and Δp as functions of the air velocity of a stainless steel fiber fleece filter module fiber ø 2 µm, 300 g/m² sintered

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.

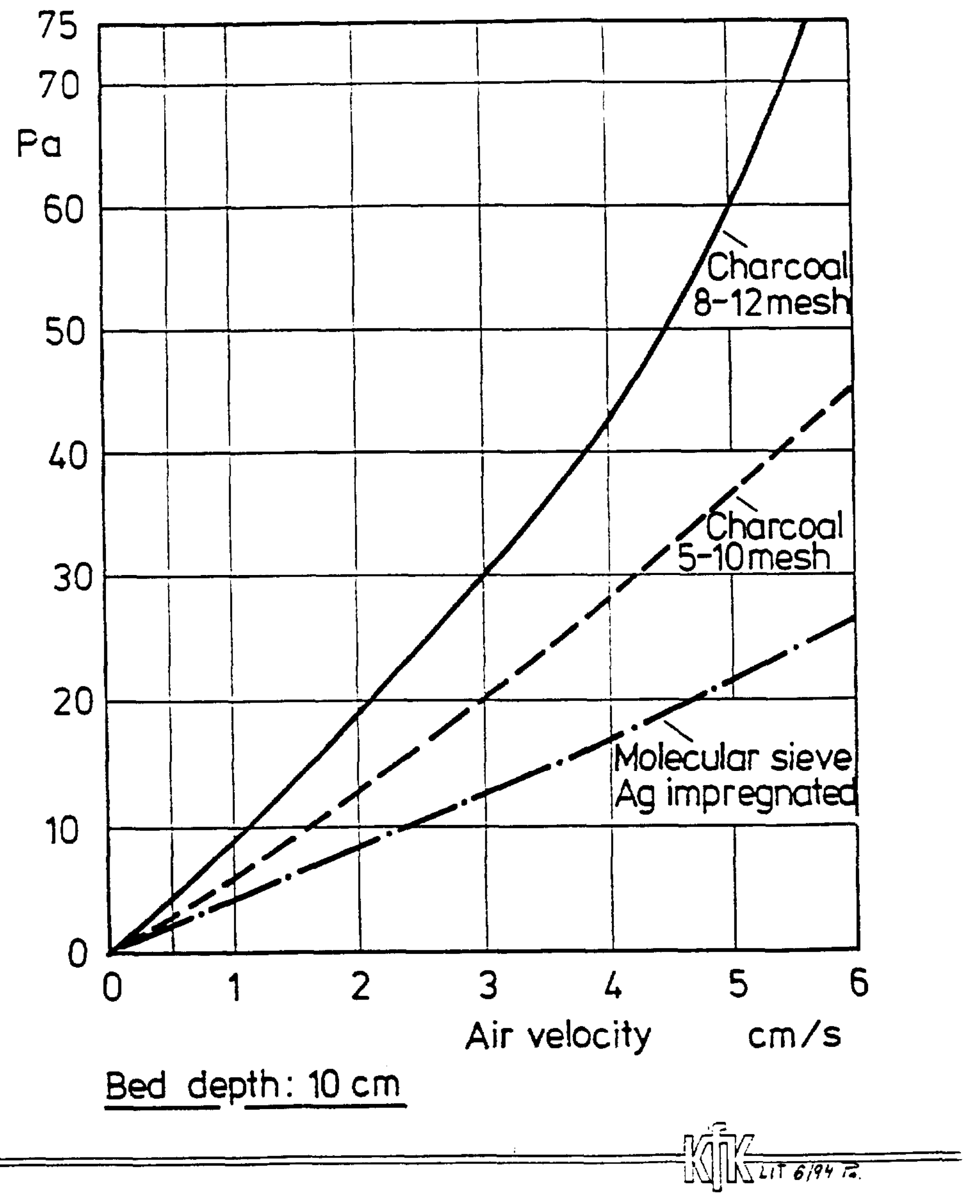

FIG. 6 Pressure drop of Iodine sorption filter materials

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.