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# Effectiveness of Engineered Safety Feature (ESF) Systems in Retaining Fission Products

Background Information

Prepared by J. Mishima, D. E. Blahnik, M. A. Halverson, A. K. Postma, F. R. Zaloudek

**Pacific Northwest Laboratory**
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**Prepared for**
**U.S. Nuclear Regulatory Commission**

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# Effectiveness of Engineered Safety Feature (ESF) Systems in Retaining Fission Products

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## ABSTRACT

The Pacific Northwest Laboratory has compiled and reviewed base line data on the effectiveness of Engineered Safety Feature (ESF) systems in the retention of fission products and particulate material resulting from a nuclear reactor accident. This work is part of an NRC project to provide the best estimates of the consequences of severe reactor accidents.

The resulting report describes the ESF systems (containment spray, secondary containment filter, containment recirculating filter, pressure suppression pool, ice condenser, and main steam line isolation valve leakage control systems). Also described are the anticipated atmospheres in which the ESFs must operate, the experimental studies of ESF system effectiveness, and the models currently available for assessing the performance of the various ESF systems. The information gaps identified as a result of this review have resulted in recommendations for additional work in the areas of: 1) performance data and models of containment chiller/coolers; 2) continued development and experimental verification of the ice condenser model; 3) continued development of the pressure suppression pool model; and 4) continued investigations of the behavior of filtration devices.

## SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) is investigating methods for more realistic assessments of the consequences of severe accidents at nuclear power plants. The Pacific Northwest Laboratory is supporting NRC by investigating the capability of Engineered Safety Feature (ESF) systems to retain fission products and core structural material gases/aerosols in a range of accident conditions with emphasis on severe core-melt accidents.

The ESF systems of interest are the containment spray, secondary containment filter, containment recirculating filter, containment recirculating air cooler, pressure suppression pool, ice condenser, and main steam line isolation valve leakage control systems. Data in various areas are required to evaluate the performance of these ESF systems. Base line data on the atmospheres anticipated for a range of reactor accident conditions (gas composition, temperature, pressure, particulate material generation, etc.) were compiled from probabilistic risk assessments, Safety Analysis Reports, and experimental studies. Also compiled was information on the experimental evaluations of ESF performance under accident conditions and on models currently available for predicting ESF performance.

Review of this compiled information showed gaps in the information required to predict ESF system performance. Additional efforts are recommended in the areas of:

- performance data for containment recirculating air cooler systems
- development and experimental verification of the ice condenser performance model
- development and experimental verification of the pressure suppression pool model
- performance data of filtration devices under severe accident conditions.

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## ABBREVIATIONS AND ACRONYMS

AMMD - Activity Mass Median Diameter
ANL - Argonne National Laboratory

B & W - Babcock and Wilcox
BWR - boiling water reactor

CE - Combustion Engineering (Inc.)
CHRS - containment heat removal system
CSE - Containment Systems Experiment
CSIS - containment spray injection system
CSRS - containment recirculating spray system
CSS - containment spray system

DOP - dioctylphthalate

ECC - emergency core cooling (system)
ECI - emergency core injection (system)
EPRI - Electric Power Research Institute
ESF - engineered safety feature

FP - fission product
FSAR - Final Safety Analysis Report
FVC - Filtered Vent Containment

GE - General Electric (Company)

HEPA - high efficiency particulate air (filter)
HX - heat exchanger

IDCOR - Industry Degraded Core Rulemaking (program)

KfK - Kernforschungszentrum Karlsruhe

LOCA - loss-of-coolant accident
LPR - low pressure recirculation (system)
LWR - light water reactor

MSIV - main steam (line) isolation valve
MSIVLC - main steam (line) isolation valve leakage control (system)

NRU - National Research Universal (reactor of Canada)
NSSS - nuclear steam supply system

ORNL - Oak Ridge National Laboratory

PELEC - Philadelphia Electric Company
PNL - Pacific Northwest Laboratory
PRA - probabilistic risk assessment
PSAR - Preliminary Safety Analysis Report
psfd - pounds per square foot, differential
psia/psid/psig - pounds per square inch, absolute/differential/gauge
PWR - pressurized water reactor

RHRS - residual heat removal system
RSS - Reactor Safety Study (WASH-1400) - (USNRC 1975)
RSSMAP - Reactor Safety Study Methodology Applications Program

SAR/PSAR/FSAR - Safety Analysis Report/Preliminary Safety Analysis Report/Final Safety Analysis Report
SNL - Sandia National Laboratory
scfh/scfm - standard cubic feet per hour/standard cubic feet per minute
SGS - Submerged Gravel Scrubber
SGTS - standby gas treatment system

UKAEA - United Kingdom Atomic Energy Authority

W - Westinghouse
WG - water gauge

## 1.0 INTRODUCTION

Engineered Safety Features (ESFs) are systems installed in U.S. commercial power reactors to mitigate the consequences of postulated severe accidents. In general, these systems fall into the general categories of: 1) containment, 2) emergency core cooling systems, 3) control room habitability systems, and 4) fission product removal and control systems. This report relates to those systems specifically provided for the control and removal of fission products and those that can provide fission product retention even though installed for other purposes (e.g., pressure suppression systems).

The document presents information that was compiled as part of an early, background effort of the Engineered Safety Feature Effectiveness Project sponsored by the U.S. Nuclear Regulatory Commission (NRC). The objective of this project is the prediction of ESF system performance, in terms of fission product retention, while emphasizing postulated sequences for severe core-melt accidents. Information compiled includes descriptions of the ESF systems (Section 3.0), the environment that could be encountered under core-melt conditions (Section 4.0), and models and experiments currently being used to predict effectiveness (Sections 5.0 and 6.0). Review of the compiled information showed gaps in the information required to predict ESF system performance. These gaps and recommended areas of work are listed in the conclusions and recommendations portion of this report, Section 2.0.

The Pacific Northwest Laboratory (PNL) performed this work as part of the ongoing effort to provide an understanding of phenomena and verified/validated analytical methods to permit best estimates (rather than conservative, nonmechanistic assessments) of reactor accident consequences.

## 2.0 CONCLUSIONS AND RECOMMENDATIONS

This report discusses the information available in the current literature on postaccident containment atmospheres, engineered safety features, the performance of ESFs, under these conditions, and models available to predict ESF performance. Overall, there appears to be considerable information on the various devices and on reactor containment atmospheres after an accident.

The coverage of accident containment atmospheres was not uniform. The information on gas composition, temperatures, and pressure was calculated for Safety Analysis Reports and probabilistic risk assessments. The estimates will improve as better information on the behavior of ESF systems becomes available.

The data on fission product generation covers the greatest expanse of time and, therefore, the greatest range of interests. Much of the information developed on the fission product release during the initial stages of fuel degradation still appears to be valid. Ongoing experimental efforts should provide better definition of the iodine/cesium forms during the various stages of fuel degradation and of these and other fission product releases during the phases when the fuel is molten in the pressure vessel and when the molten fuel contacts the concrete basemat. Definition of the particulate material generated during these latter two periods is being developed by current experimental efforts.

Much less data is available for ESF systems. Containment spray system performance has been included in some models (e.g., MAEROS, NAUA) based on earlier experimental work. However, the behavior of containment spray systems under currently anticipated conditions of high particulate mass concentrations needs to be assessed. Models have been developed for the ice compartments of ice condenser systems and pressure suppression pools. Ongoing experimental efforts should validate the suppression pool model, but a comparable validation effort is required for the ice compartment model.

Models are not available for containment recirculating cooler systems, the main steam line isolation valve leakage control system, and filter systems. A promising avenue of development for a cooler/chiller model might be an adaptation of the existing ice compartment model. The ongoing TRAP-MELT verification experiments may provide insight into the behavior of the MSIVLC system behavior and provide a basis for modeling its performance. Although a large amount of data is available for filters of various types, little pertains to the performance and failure modes under conditions predicted for severe accidents.

In summary, the information gaps identified as a result of reviewing the information have resulted in recommendations for additional efforts in the following areas:

- performance data (loading curves as a function of conditions, cooling efficiencies as a function of loading, etc.) and performance models for containment chiller/coolers

- continued development and experimental verification of the ice condenser performance model

- continued development of the pressure suppression pool model

- loading curves and failure modes for filtration devices under post-accident conditions, efficiency versus loading curves, and conditions for filtration devices used in LWRs.

## 3.0 PRESENT ENGINEERED SAFETY FEATURE SYSTEMS

ESFs are systems installed in nuclear power reactors to mitigate the potential consequences of postulated severe accidents. Seven of these ESF systems that provide the potential to remove fission products (FPs) removal are discussed in this section. They are:

- Containment sprays
- Secondary containment filters
- Containment recirculating filters
- Containment recirculating air coolers (chiller/coolers)
- Pressure suppression pools
- Ice compartments of ice condenser containment systems
- Main steam (line) isolation valve leakage control (MSIVLC) systems.

Except for the filters and MSIVLC systems, these systems are primarily intended to suppress containment vessel internal pressure and temperature increases that result from loss of coolant accidents (LOCAs). However, the other systems can also remove fission products and particulate material (some fission products may be particulate material). A concept which has recently received attention, Filtered Vent Containment (FVC), will be described but not discussed.

### 3.1 REACTOR CHARACTERISTICS

There are (at this writing) 74 licensed reactors and an additional 78 reactors in various stages of construction or licensing in the United States. A total of 25 boiling water reactor (BWRs) are licensed with an additional 26 planned. All BWR nuclear steam supply systems (NSSS) except La Crosse are provided by one manufacturer. The remainder of the reactors are pressurized water reactors (PWRs) whose NSSS are supplied by three manufacturers (see Table 3.1). An alternative means of categorizing power reactors is by their containment characteristics (see Table 3.2) The ESFs found in each reactor (as determined from the information available to this study) are tabulated according to NSSS manufacturers in Tables 3.3, 3.4, 3.5 and 3.6.

TABLE 3.1. U.S. Power Reactors Listed by Nuclear Steam System Manufacturer

| Manufacturer | Reactor Type | Licensed Reactors | Proposed or Under Construction | Totals |
|---|---|---|---|---|
| Westinghouse (W) | PWR | 31 | 32 | 63 |
| General Electric (GE) | BWR | 25 | 26 | 51 |
| Babcock & Wilcox (B&W) | PWR | 9 | 6 | 15 |
| Combustion Engineering (CE) | PWR | 9 | 14 | 23 |
| TOTAL | | 74 | 78 | 152 |

TABLE 3.2. U.S. Power Reactors Listing Expanded by Containment Type

| Manufacturer | Containment Type | Licensed Reactors | Proposed or Under Construction | Totals |
|---|---|---|---|---|
| Westinghouse (W) | Dry | 26 | 27 | 53 |
| | Ice Condenser | 5 | 5 | 10 |
| General Electric (GE) | Mark I | 23 | 2 | 25 |
| | Mark II | 1 | 9 | 10 |
| | Mark III | 1 | 15 | 16 |
| Babcock & Wilcox (B&W) | Dry | 9 | 6 | 15 |
| Combustion Engineering (CE) | Dry | 9 | 14 | 23 |
| TOTAL | | 74 | 78 | 152 |

### 3.2 ESF SYSTEM DESCRIPTIONS

The following general descriptions indicate how the seven ESF systems are employed in different facilities. Therefore, the descriptions are general and simplified so that the information is applicable to many designs. The Filtered Vent Containment concept is also described.

#### 3.2.1 Containment Spray System (Pasedag, Blond and Jankowski 1981)

A system found in most of the U.S. LWRs is the containment spray system (CSS). The primary function of the containment spray system is to limit the peak pressure of the containment internal atmosphere during the blowdown following a LOCA. The system also removes fission products and particulate materials from the containment atmosphere by absorption and/or the particle capture mechanisms of impaction, interception, and diffusion. A simplified schematic of a containment spray system is shown in Figure 3.1. The spray system usually consists of two to six ring headers placed at the top of the containment vessel. Each header has a large number of nozzles oriented to be able to uniformly spray most of the upper compartment containment volume. In addition to the spray headers, the system includes pumps and necessary water storage tanks, heat exchangers, pipes, and valves. In some systems, chemical additive (e.g., hydrazine, sodium hydroxide) injection systems are also used to enhance the removal of elemental iodine. The spray nozzles typically have a 3/8-inch orifice and a water flow rate of about 15 gpm with a pressure drop of 40 psid. The nozzles produce a mean drop size in the range between 230 to 1100 micrometers (μm) at the rated system conditions. The spray water drains into the containment lower compartment and into sumps (into a suppression pool for a BWR) and is recycled through pumps and heat exchangers, back to the spray

TABLE 3.3. Westinghouse Pressurized Water Reactors

| Name | Containment Type | Containment Sprays | Recirculating Filters | Recirculating Coolers | Suppression Pools | Ice Condensers |
|---|---|---|---|---|---|---|
| Beaver Valley 1,2 | dry | X | | | | |
| Braidwood 1,2 | dry | X | X | X | | |
| Byron 1,2 | dry | X | X | X | | |
| Callaway 1 | dry | X | X | X | | |
| Carroll County 1,2 | dry | | -------------No information------------- | | | |
| Commanche peak 1,2 | dry | X | | | | |
| Diablo Canyon 1,2 | dry | X | X | X | | |
| Farley 1,2 | dry | X | | X | | |
| Ginna | dry | X | X | X | | |
| Haddam Neck | dry | | -------------No information------------- | | | |
| Harris 1,2 | dry | X | | X | | |
| Indian Point 2,3 | dry | X | X | X | | |
| Jamesport 1,2 | dry | | -------------No information------------- | | | |
| Kewaunee | dry | X | | X | | |
| Marble Hill 1,2 | dry | | -------------No information------------- | | | |
| Millstone 3 | dry | X | | X | | |
| North Anna 1,2 | dry | X | | | | |
| Point Beach 1,2 | dry | X | X | X | | |
| Prairie Island 1,2 | dry | X | | X | | |
| Robinson 2 | dry | X | | X | | |
| Salem 1,2 | dry | X | | X | | |
| Seabrook 1,2 | dry | X | | | | |
| San Onofre 1 | dry | X | | X | | |
| South Texas 1,2 | dry | X | | X | | |
| Summer 1 | dry | X | | X | | |
| Surry 1,2 | dry | X | | | | |
| Trojan | dry | X | | | | |
| Turkey Point 3,4 | dry | X | X | X | | |
| Vogtle 1,2 | dry | X | | X | | |
| Wolf Creek 1 | dry | X | X | X | | |
| Yankee Rowe | dry | | | X | | |
| Zion 1,2 | dry | X | | X | | |
| Catawaba 1,2 | ice condenser | X | | | | X |
| Cook 1,2 | ice condenser | X | | | | X |
| McGuire 1,2 | ice condenser | X | | | | X |
| Sequoyah 1,2 | ice condenser | X | | | | X |
| Watts Bar 1,2 | ice condenser | X | | | | X |

TABLE 3.4. General Electric Boiling Water Reactors

| Name | Containment Type | Containment Sprays | Recirculating Filters | Recirculating Coolers | Suppression Pools | Main Steam Isolation Valve Control System |
|---|---|---|---|---|---|---|
| Arnold | Mark I | | | | | XX |
| Big Rock Point | Mark I | | | | | X |
| Brown's Ferry 2,3 | Mark I | X | | | | X |
| Brunswick 1,2 | Mark I | X | | | | X |
| Cooper | Mark I | X | | | | X |
| Dresden 1,2,3 | Mark I | ? | | | | X |
| Fermi 2 | Mark I | X | | | | XX |
| Fitzpatrick | Mark I | | | | | XX |
| Hatch | Mark I | X | | | | XX |
| Hope Creek 1 | Mark I | ------------- | No information | ------------- | X | X |
| Humboldt Bay | Mark I | | | | | X |
| Millstone Point 1 | Mark I | | | | | X |
| Monticello | Mark I | X | | | | X |
| Nine Mile Point 1 | Mark I | | | | | X |
| Oyster Creek | Mark I | X | | | | X |
| Peach Bottom 2,3 | Mark I | X | | | | X |
| Pilgrim | Mark I | X | | | | X |
| Quad Cities 1,2 | Mark I | | | | | X |
| Vermont Yankee | Mark I | X | | | | X |
| LaSalle 1,2 | Mark II | X | | | | XX |
| Limerick 1,2 | Mark II | X | | | | XX |
| Nine Mile Point 2 | Mark II | ? | | | | XX |
| Shoreham | Mark II | X | | | | XX |
| Susquehanna 1,2 | Mark II | X | | | | XX |
| Washington 2 | Mark II | X | | | | XX |
| Zimmer 1 | Mark II | X | | | | XX |
| Black Fox 1,2 | Mark III | ------------- | No information | ------------- | X | X |
| Clinton 1,2 | Mark III | X | | | | XX |
| Grand Gulf 1,2 | Mark III | X | | | | XX |
| Montague 1,2 | Mark III | ------------- | No information | ------------- | X | X |
| Perry 1,2 | Mark III | X | | | | XX |
| Phillips 1,2 | Mark III | ------------- | No information | ------------- | X | X |
| River Bend 1,2 | Mark III | | | | | XX |
| Skagit 1,2 | Mark III | ------------- | No information | ------------- | X | X |

TABLE 3.5. Babcock and Wilcox Pressurized Water Reactors

| Name | Containment Type | Containment Sprays | Recirculating Filters | Recirculating Coolers | Suppression Pools | Ice Condensers |
|---|---|---|---|---|---|---|
| Arkansas 1 | dry | X | | X | | |
| Bellaforte 1,2 | dry | X | | X | | |
| Crystal River 3 | dry | X | | X | | |
| Davis-Besse 1 | dry | X | | X | | |
| Midland 1,2 | dry | X | | X | | |
| North Anna 3 | dry | X | | X | | |
| Oconee 1,2,3 | dry | X | | X | | |
| Rancho Seco | dry | X | X | X | | |
| Three Mile Island 1,2 | dry | X | | X | | |
| Washington 1 | dry | -------------No information------------- | | | | |

TABLE 3.6. Combustion Engineering Pressurized Water Reactors

| Name | Containment Type | Containment Sprays | Recirculating Filters | Recirculating Coolers | Suppression Pools | Ice Condensers |
|---|---|---|---|---|---|---|
| Arkansas 2 | dry | X | | | | |
| Calvert Cliffs 1,2 | dry | X | | X | | |
| Cherokee 1,2,3 | dry | X | | | | |
| Forked River 1 | dry | -------------No information-------------- | | | | |
| Fort Calhoun | dry | X | X | X | | |
| Maine Yankee | dry | X | | | | |
| Millstone 2 | dry | X | | X | | |
| Palisades | dry | X | | X | | |
| Palo Verde 1,2,3 | dry | -------------No information-------------- | | | | |
| Perkins 1,2,3 | dry | -------------No information-------------- | | | | |
| St. Lucie 1,2 | dry | -------------No information-------------- | | | | |
| San Onofre 2,3 | dry | X | | X | | |
| WNP 3 | dry | -------------No information-------------- | | | | |
| Waterford 3 | dry | X | | X | | |

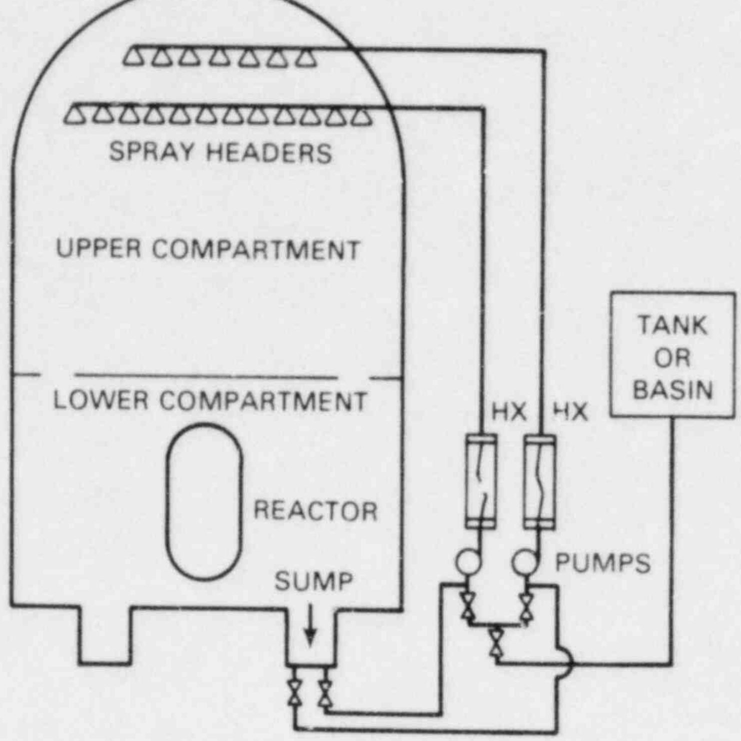

FIGURE 3.1. Containment Spray System

headers. The spray system can be started manually or automatically. Automatic systems usually start when a certain pressure is exceeded in containment. The initial source of water is usually from a water storage tank or basin and, after the tank or basin is emptied, water is recycled from the reactor sump or pressure suppression pool. Usually the water is pumped from the sump or pool through residual heat removal heat exchangers to remove the heat before reuse.

### 3.2.2 Secondary Containment Filter System

These filter systems are employed to clean contaminated air that leaks from the primary containment into the secondary containment. Names given to these systems include the Auxiliary Building and Standby Gas Treatment System (SGTS), the Emergency Gas Treatment System, Penetration Room Exhaust System, and Secondary Containment Air Cleaning System. Probably all of the LWRs have at least one secondary containment filter system with two filter trains. An example of a secondary filter system is shown in Figure 3.2 (ASME 1980). Air is collected by ducts from regions where a potential exists for leaks from the primary containment. The air is moved by a fan through the filter train and out exhaust stacks. The filter system removes the particulate material (including any fission products in this form) before the air is released to the environment. Some containments consist of two walls separated by an air annulus. The air in

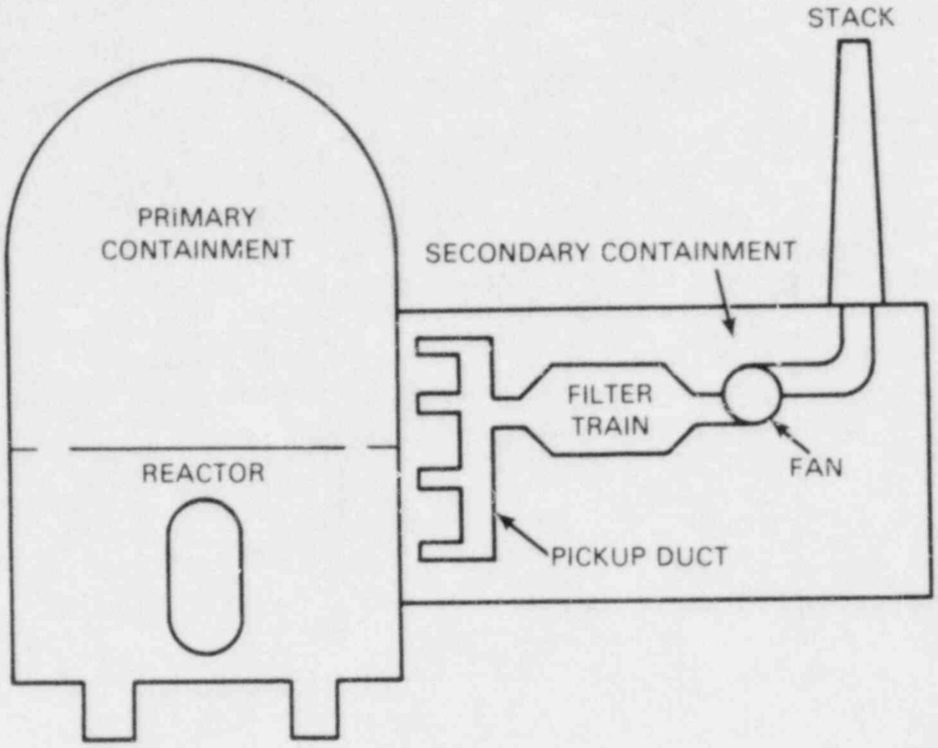


FIGURE 3.2. Secondary Containment Filter System

this annulus is passed through a similar filter system and at least partially exhausted to the environment. Filter systems which clean and recirculate the gases within the primary containment vessel are described in Section 3.2.3.

Typically, the filter system consists of a train of equipment that is mounted in series within the exhaust duct as shown in Figure 3.3. The contaminated gases flow through the filter train, which has a moisture separator, heater, prefilter, high efficiency particulate air (HEPA) filter, charcoal adsorber trap, and another HEPA filter. The gases are exhausted from the containment building exhaust stack via a fan. Air flow rates through a filter train are usually in the 1,000- to 10,000-cfm range at 5 inches to 10 inches water gauge (WG). Generally two parallel trains are used so that backup capability exists. The moisture separator (or demister) removes entrained water droplets and moisture from the air stream when the stream is supersaturated. The removed water flows into a drain at the bottom of the separator. The electric heater unit which follows elevates the temperature of the air to reduce the relative humidity to 70% or less. Higher air humidity could reduce the effectiveness of the downstream HEPA and charcoal adsorber filters.

The prefilter between the heaters and the HEPA filter primarily removes large particles. The particle collection efficiency of the prefilter is about 85% for standard dust and higher for particles above several micrometers. The

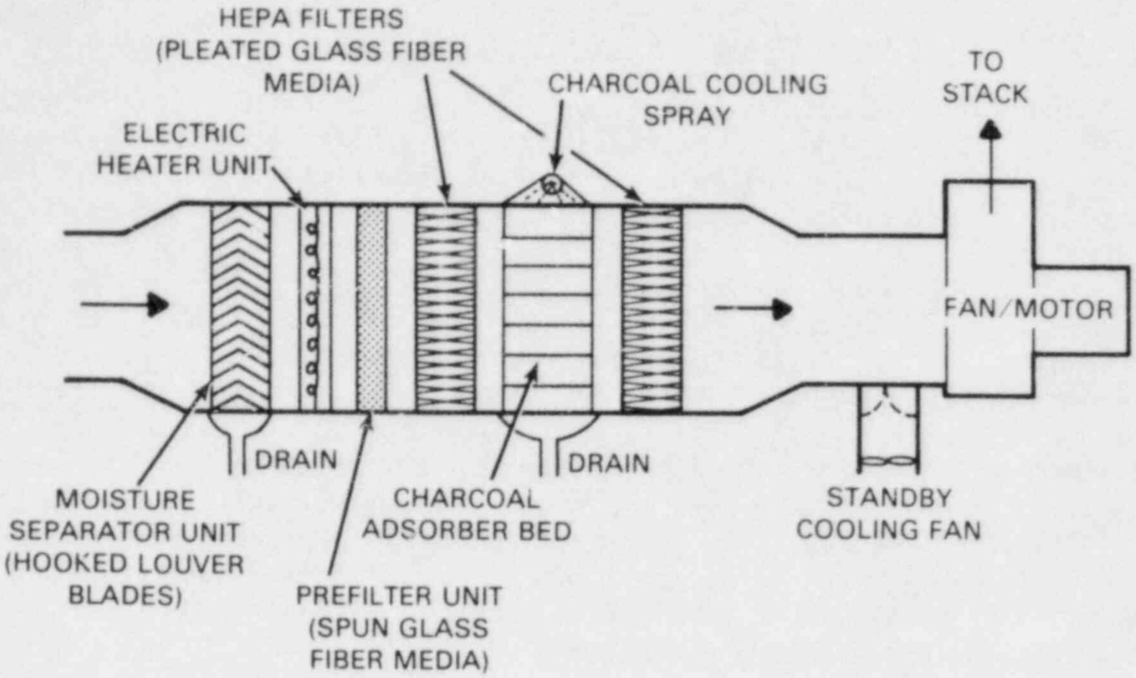


FIGURE 3.3. Typical Filtration Train

smaller particles are removed by the HEPA filters, which have a stated particle collection efficiency of ~99.97% for 0.3-μm DOP droplets. (This is the particle collection efficiency for the filter and does not necessarily represent the collection efficiency for the installed unit.) HEPA filters are water, fire, and radiation resistant, and are capable of functioning at temperatures up to 500°F at gamma doses up to $10^6$ rad. Iodine vapors and particulate materials that pass through the HEPA filter are trapped in the charcoal adsorber unit. The adsorber is made up of potassium iodide-impregnated charcoal encased in flat bed trays. The removal efficiency is up to 99.9% of the methyl iodide and elemental iodine in air with a relative humidity of 70% at 175°F. Because of decay heat generated by the highly radioactive materials collected on the charcoal, there is concern that the charcoal could be heated to temperatures exceeding its ignition temperature of 644°F. A temperature sensor and cooling spray have been installed in some units to cool the charcoal and suppress fires. A drain at the bottom of the charcoal bed removes the water generated by the spray.

The downstream HEPA filter collects particulates released from the charcoal adsorber unit. The fan moves the filtered air to the containment building stack where the air is released to the environment. A standby cooling fan is available in case the main fan fails. The standby fan will maintain a small flow of air through the filter train to help keep the filters cool.

### 3.2.3 Containment Recirculating Filter System

Some of the PWRs use air recirculating filters within primary containment. Figure 3.4 illustrates where recirculating filter units are usually located in nuclear facilities. The filter systems range from filter train units that include all the components shown previously in Figure 3.3 for secondary systems to units with just a single HEPA filter and fan. Some of the units are meant for use after fuel handling accidents, some for post-LOCA use, and some are intended for both purposes. The units designed for post-LOCA use remove fission products and particulate material from the containment atmosphere. Some units recirculate only part of the air, exhausting the rest to the plant stack. In one facility, the containment recirculating air cooler system includes a HEPA and charcoal filter. Because of the many variables in the design basis of the recirculating air filter system, each plant must be evaluated separately for system effectiveness. The primary containment recirculating filter systems are effective only in minor LOCAs because of the limited amount of material (probably less than 1 kg of particulate material per 2-ft by 2-ft HEPA filter) that can be collected before the filters plug.

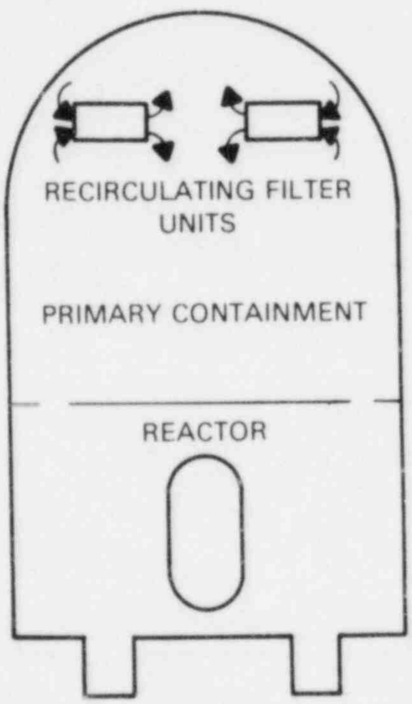


FIGURE 3.4. Containment Recirculating Filter System

### 3.2.4 Containment Recirculating Air Cooler System

Many of the PWR plants have recirculating air coolers which cool the primary containment volume during normal operations and after a LOCA. During normal operations, the system cools and dehumidifies the containment atmosphere to achieve the operating environment required by the mechanical, electrical and structural components. After a LOCA, the coolers work in conjunction with the containment spray system to cool the air and hold the temperature and pressure of the containment atmosphere within safe limits. Usually the recirculating coolers provide long-term cooling while the spray system provides short-term cooling. Figure 3.5 is a schematic of a recirculating cooler and the approximate location of the coolers within the plant.

The containment recirculating coolers are large fan/coil units. The unit shown has a housing where air is drawn through the four sides and passed through a set of damper louvers, a battery of finned-tube water-cooled coil assemblies, a drive fan, and an exhaust system. The exhaust may have appropriate duct work to help balance the circulation of the air within containment. The back-draft dampers are spring operated to protect against a transient-induced reversed flow and the possible adverse affects of such a reversed flow. The containment air is recycled past the coils, which absorb the heat into the coolant water.

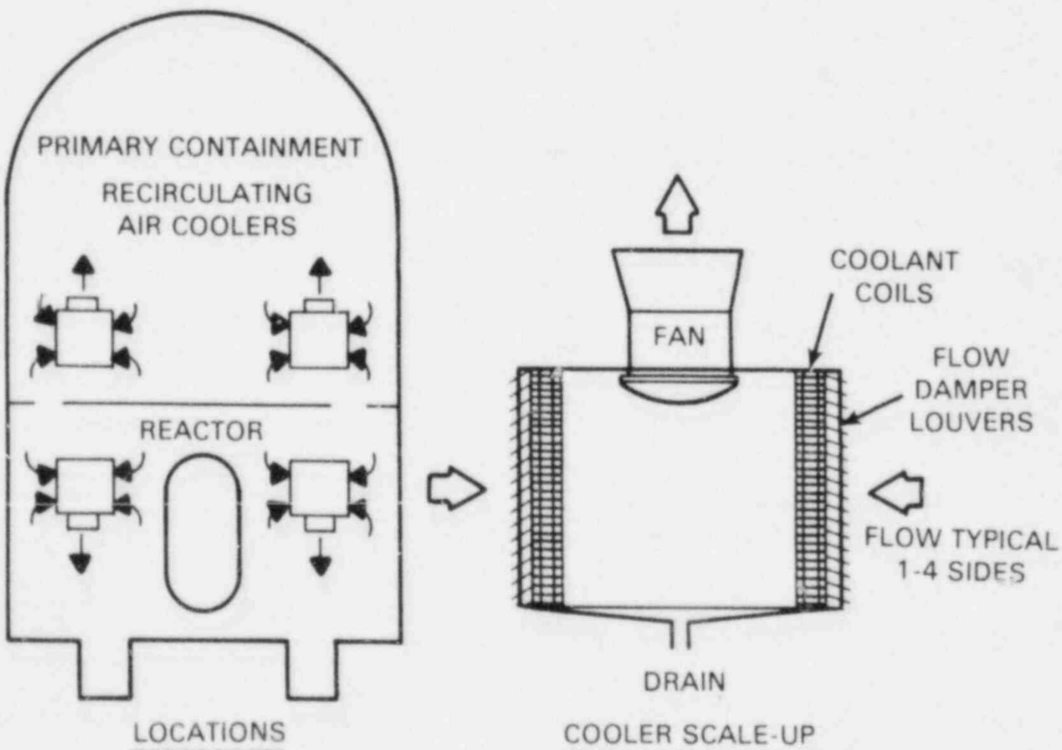


FIGURE 3.5. Containment Recirculating Air Cooler System and Typical Cooler Unit

The coolant water can be either from an open plant service water system or a recirculating closed loop with a heat exchanger where the heat is dumped. After a LOCA, the units can cool at an increased rate (up to an order of magnitude or more) over the rate required for normal operations. Usually this is accomplished by a combination of reducing the fan speed and increasing the coil coolant flow.

The recirculating cooler design varies, depending upon the location, containment volume served, and cooling requirements. After a LOCA, the coolers must have the cooling capacity and air flow needed to condense the containment steam and reduce the air temperature. The steam condensate is collected at the bottom of the coil on the housing floor and is drained into the appropriate storage tank, basin or sump.

The containment recirculating coolers are not normally credited for their ability to remove fission products from the containment atmosphere. However, their ability to remove fission product iodine was demonstrated in the Plutonium Recycle Test Reactor (PRTR) incident (Perkins et al. 1965). Some of the fission products are bound to collect on the coil, damper, and housing surfaces, and some will be washed by condensate into the drain. The amount of fission product collected in the recirculation cooler system is unknown and thus is an area for future research.

The role of containment coolers in particulate removal from postaccident environments has not been assessed. The containment coolers in the five facilities covered in Section 3.0 consist of fans and tubes or coil coolers of various sizes. Some of the coolers were designed to be used under post-accident conditions and are part of the engineered safety features of the facility. Other coolers are not designed for use as an ESF but are designed to withstand the design-basis accident conditions. While no credit is taken for heat removal after an accident by these coolers, they could be available for both heat and particulate material removal. The following paragraphs consider each reference facility.

Grand Gulf (Hatch, Cybulskas and Wooten 1981). The Grand Gulf containment cooling system is designed to maintain average containment conditions of 80°F and 50% relative humidity during normal plant operations. The system is not considered an ESF but could function as one if it survives the accident. The total cooling capacity of the system is $2.88 \times 10^6$ Btu/hr. The fan capacity is 25,000 cfm. A containment ventilation and purge system also operates during normal operation. This system consists of fans, heating coils, prefilters, HEPA filters, and charcoal filters. The system can be used in postaccident environments to clean up the containment atmosphere. The fan capacity of this unit is 10,000 cfm.

Zion (Com. Ed. 1981). The Zion containment fan cooler system is designed to filter, cool, and dehumidify the containment environment during both normal and postaccident conditions. During normal operations, each cooler can remove $3.15 \times 10^6$ Btu/hr from an air flow of 85,000 cfm. During accident operation, each cooler is designed to remove $81 \times 10^6$ Btu/hr from an air flow of 35,000 cfm. Since there are 5 coolers in the containment, the total heat removal capacity under accident conditions is $405 \times 10^6$ Btu/hr. The main mode of heat transfer during accident operation is condensation, which could lead to reasonable particulate removal. The Zion cooler system also has prefilters, HEPA filters, and moisture separators associated with each cooling unit. These devices would also tend to reduce the overall aerosol concentration but may also lead to failure if they become plugged.

Oconee (Kolb et al. 1981). The Oconee reactor building cooling system is similar to the Zion system in that it is designed to operate in both normal and postaccident environments. There are three coolers at Oconee with a rated capacity of $80 \times 10^6$ Btu/hr each under postaccident conditions. Two units are used for normal operating conditions, and all three are used for accident conditions. Each cooler is operated at 54,000 cfm under postaccident conditions. The total heat removal capacity of the system is about $240 \times 10^6$ Btu/hr. This large amount of heat transfer could also result in the removal of large quantities of particulate materials. Filtration devices were not mentioned in the description of this unit.

Sequoyah (Carlson et al. 1981). Sequoyah has a containment air cooling system for normal operating conditions. Four cooling units are installed in both the upper and lower compartments. The lower compartment cooling system will handle $6 \times 10^6$ Btu/hr, while the upper compartment units have one tenth this capacity ($6 \times 10^5$ Btu/hr). An additional $3 \times 10^6$ Btu/hr cooling is available from the air cooling units for the control rod drive mechanism. The maximum amount of cooling capacity is less than $10 \times 10^6$ Btu/hr and is therefore much less significant than the units at Zion or Oconee. Sequoyah also has a system of air return fans in the containment. There are no filters or coolers involved, but air is transported by duct where some deposition of particulate material could occur.

Peach Bottom (PELEC 1970). Peach Bottom has a primary containment cooling and ventilation system. There are seven fan-coil units inside the drywell. These units are considered to be an ESF system. Each unit appears to provide $40 \times 10^6$ Btu/hr, which gives a total heat removal capacity of $280 \times 10^6$ Btu/hr. There are also cooling coils used for normal containment ventilation, but specifications for these units were not found. An auxiliary building is also served by several ventilation systems which have prefilters and HEPA filters.

### 3.2.5 Pressure Suppression Pools (BWR)

The pressure suppression pool is designed to reduce the primary containment pressure following a loss-of-coolant accident by condensing the steam and reducing the temperature generated by the event. In addition, the passage of the materials (gases, vapors, and particulate materials) through the water in the pool results in the removal of certain fission products.

The three basic designs (Mark I, II and III) are illustrated in Figures 3.6, 3.7 and 3.8 [These illustrations were adapted from Oslick (1976) and GE (1980)]. The following discussion pertains specifically to the Mark III design, although much of the information is applicable, in a general sense, to all three designs. The information was extracted from GESSAR II, 238 Nuclear Island, Volume II (GE 1981).

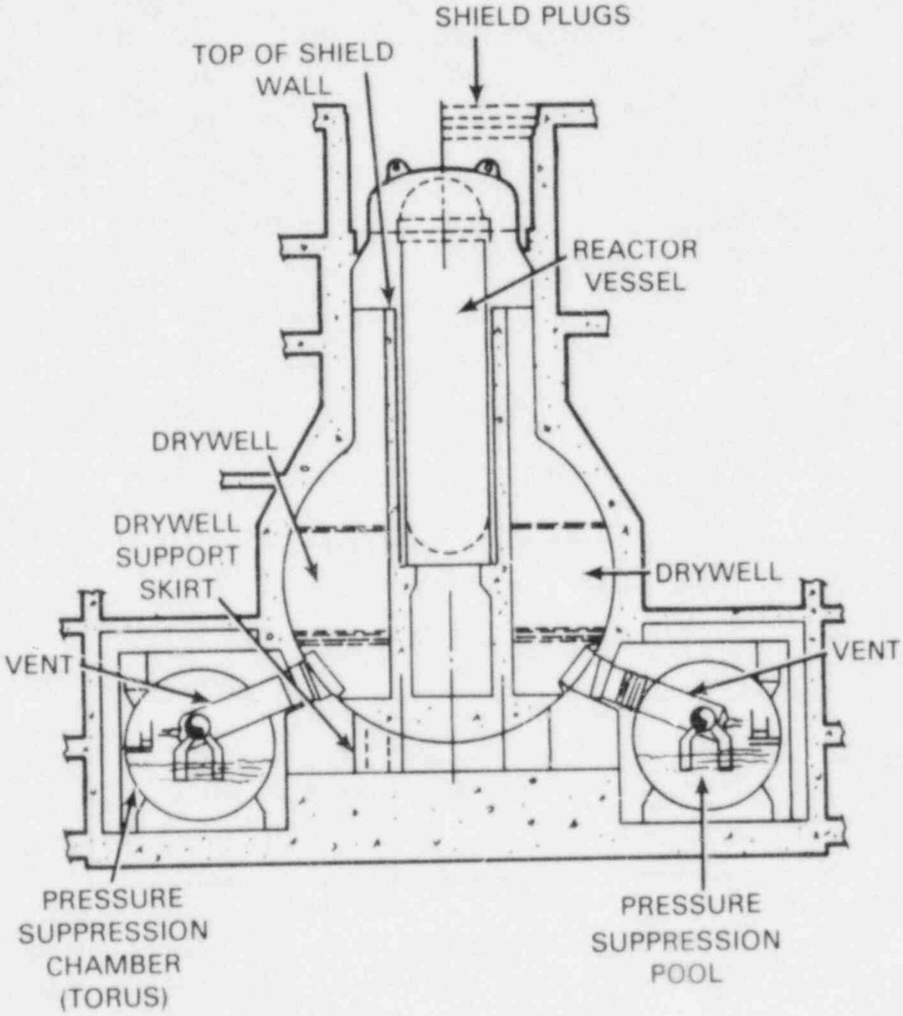


FIGURE 3.6. BWR Mark I Containment System (Oslick 1976)

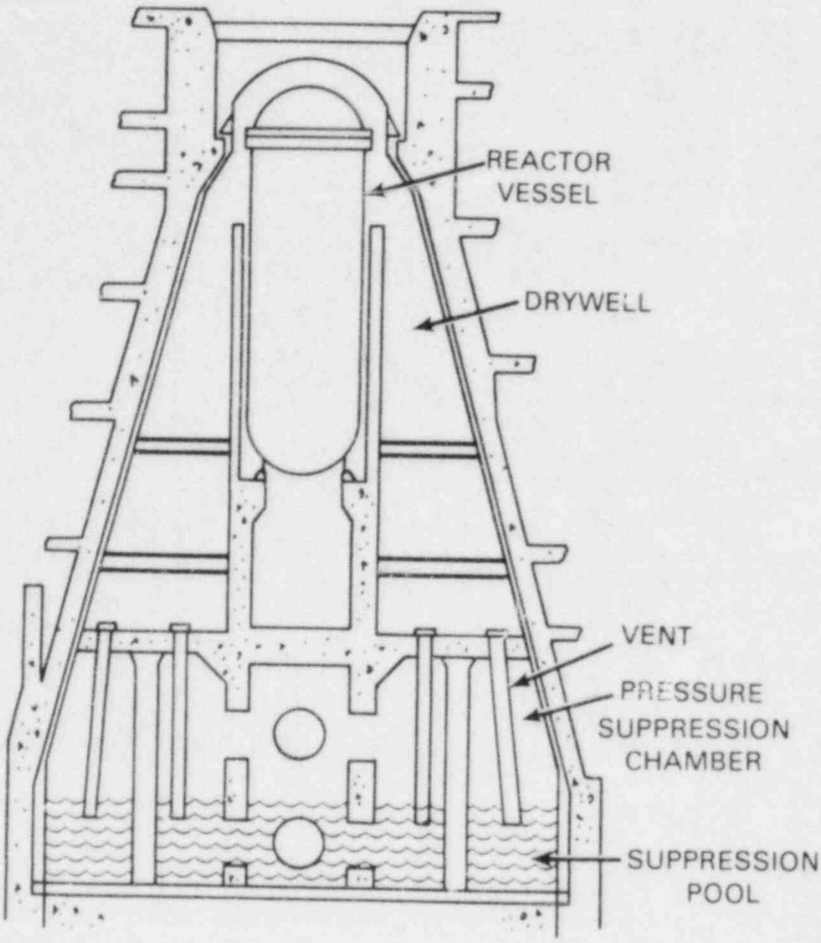


FIGURE 3.7. BWR Mark II Containment System (Oslick 1976)

The suppression pool is an open-top, stainless steel-lined, concrete structure that connects the containment (wetwell) and drywell regions. The pool is annular and filled with demineralized water to maintain a seal between these areas (all three designs). The pool surface area is about 480 $ft^2$ in the drywell and 5900 $ft^2$ in containment. The depth of the water in the pool is nominally ~20 ft. There are 120 2.3-ft-diameter horizontal vents stacked in three rows spaced uniformly around the perimeter; sloping exit tubes into a torus are used in the Mark I and X-quenchers in the Mark II. The depths of the rows are 7 ft, 11.5 ft, and 16 ft below the nominal water level.

In the event of a LOCA, the flash vaporization of the coolant pressurizes the drywell and vents the airborne material through the suppression pool. The pool water condenses the steam and scrubs airborne particulate material and vapors. Residual particulate material and vapors plus any noncondensible gases are vented to the free volume in the containment vessel until pressure equilibrium

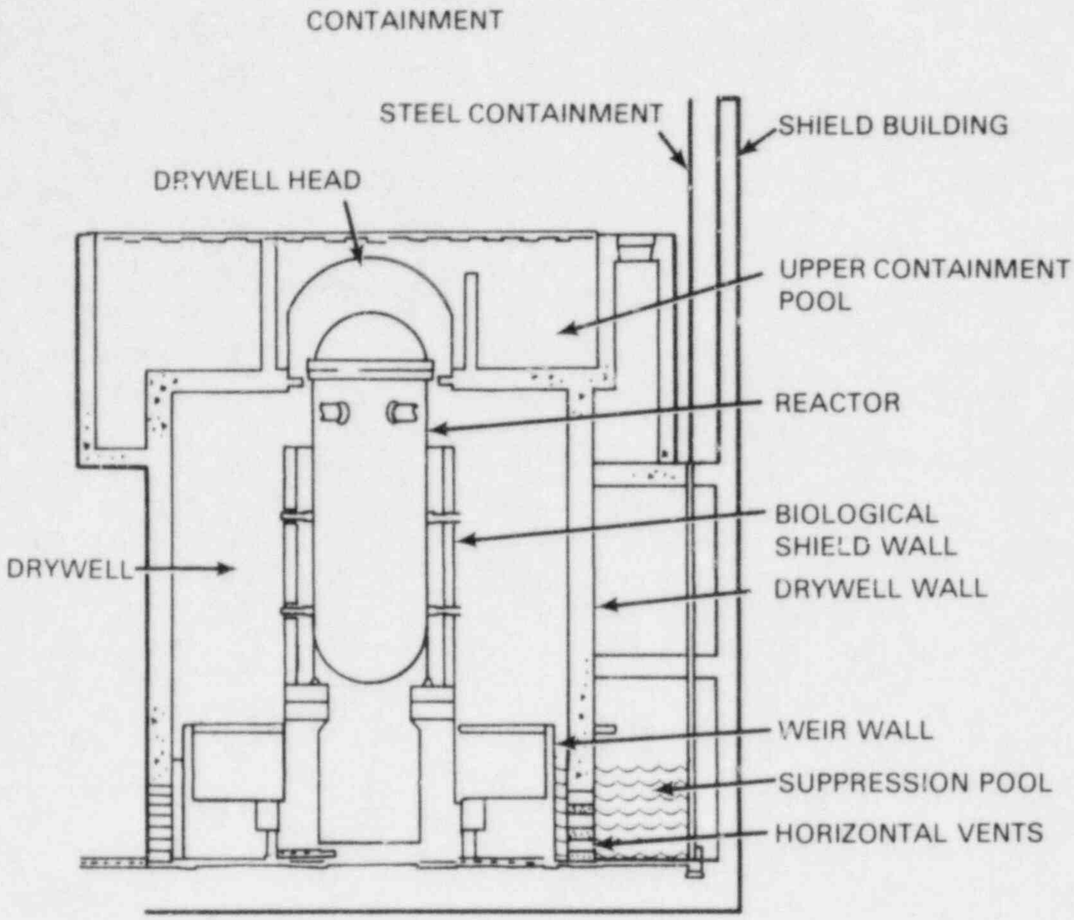


FIGURE 3.8. BWR Mark III Containment System (GE 1980)

is established between the two compartments. Condensed water helps maintain the water level of the pool, and make-up water is available from the upper containment pool.

Besides acting as a suppression pool and scrubber, the pool provides: 1) a heat sink for the reactor core isolation system during hot standby operation until decay heat can be piped directly to the residual heat removal system; 2) a heat sink for venting the nuclear safety relief valves; 3) a source of water for the emergency core cooling systems; and 4) a heat sink during normal operations.

### 3.2.6 Ice Condenser Containment Systems (PWR)

Ice condenser containment systems are used for pressure suppression in the event of a loss-of-coolant accident in several PWRs (Donald C. Cook, Sequoyah, McGuire, Watts Bar, and Catawba). The system is illustrated in Figure 3.9, and the ice compartment is shown in Figure 3.10 (TVA reference).

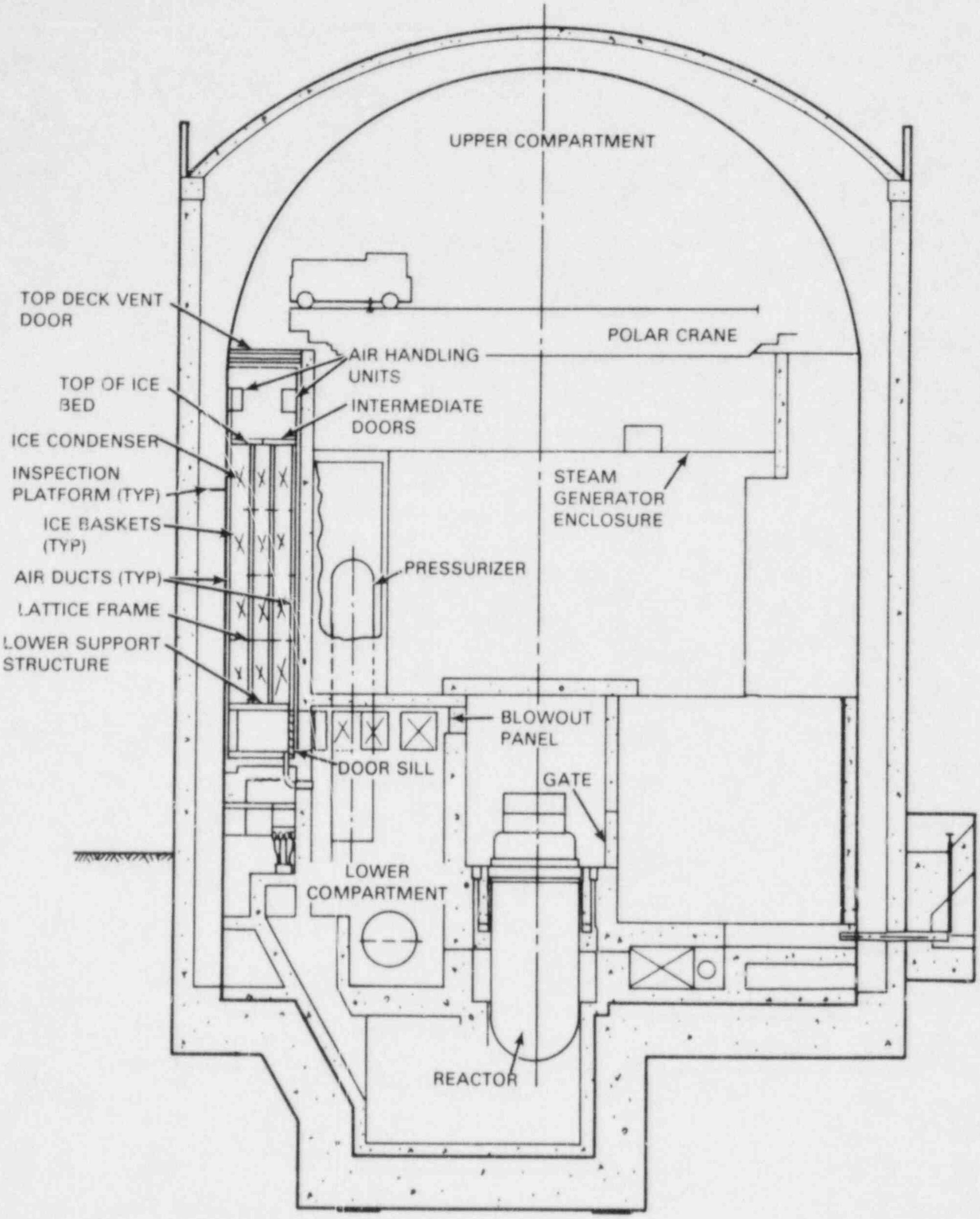


FIGURE 3.9. Ice Condenser System in PWR Containment

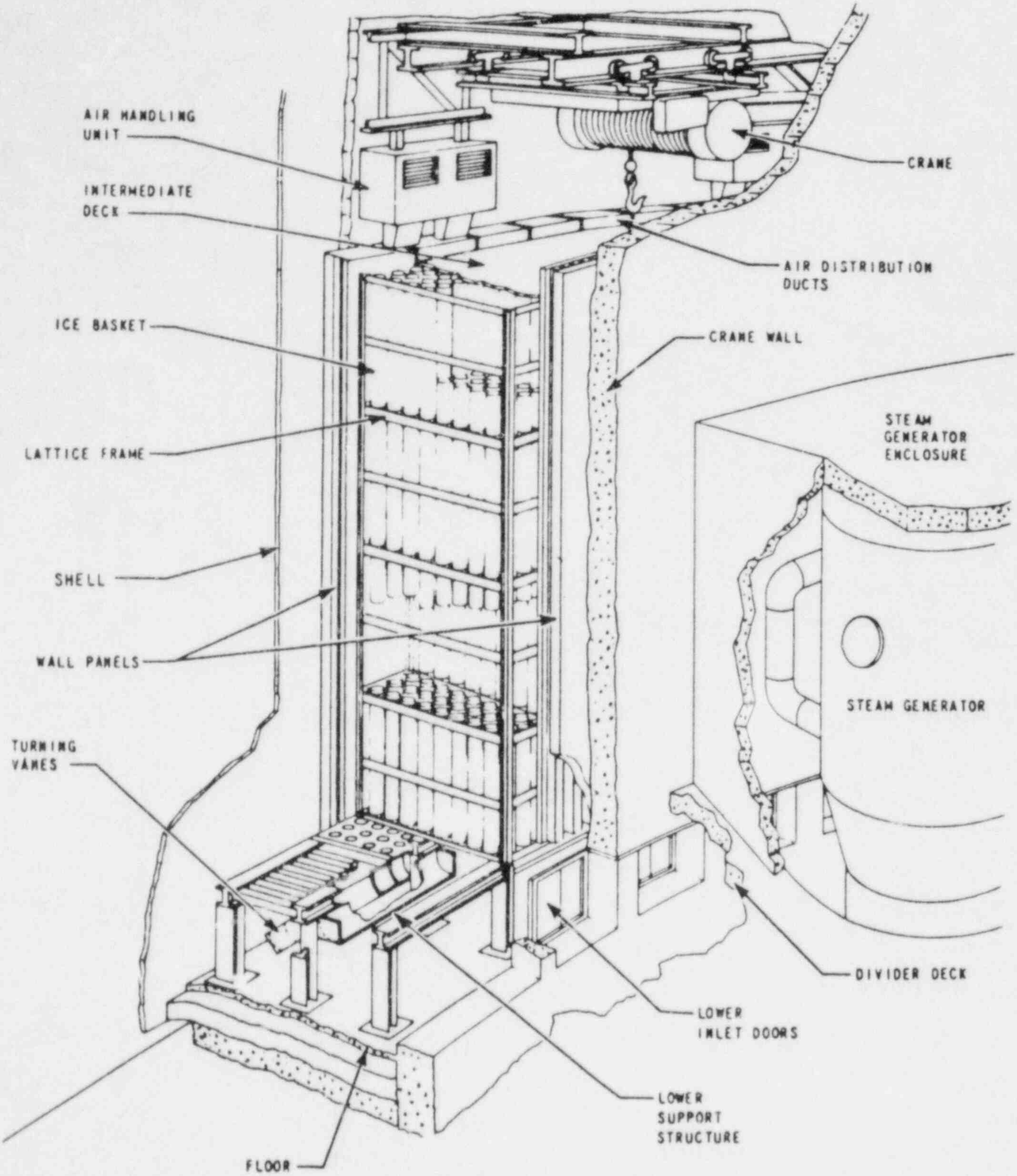


FIGURE 3.10. Ice Compartment

The ice condenser system is designed to suppress the pressure rise anticipated from the vaporization of coolant in the case of a LOCA event. Steam generated by the event is channeled from the lower compartment up through the ice compartment into the upper containment compartment. The ice provides a passive heat sink to readily condense the steam and reduce the containment temperature and pressure. The ice condenser system coincidentally removes vapors and particulate material from the gases. The ice, spiked with sodium tetraborate ($Na_2B_4O_7$), has a high capacity for absorbing elemental iodine. After the pressures in the lower and upper containment compartments reach equilibrium, the gases are recirculated through the system by two 40,000-cfm fans.

The ice compartment, 13-ft wide by approximately 50-ft high, is a partial annulus, shaped in the form of a "C", covering about 300 degrees of arc around the edge of the containment vessel. Both the steel containment and concrete crane walls that bound the ice compartment are lined with thermal insulation and cooling ducts. The ice is maintained at about 15°F by thirty wall-mounted air handling units that circulate glycol solution through the ducts. The refrigeration units are located outside of the containment.

The ice is in the form of flakes approximately 2 in. long X 2 in. wide X 1/8 in. thick. The flakes are contained by 2000 perforated metal baskets that are 1 ft in diameter X 12 ft high. Four baskets are stacked to form a continuous 48-ft-high column on 13-in. to 16-in. centers. The baskets are supported vertically by the ice compartment floor and horizontally by steel lattice frames on 6-ft centers. The compartment contains 2.45 to 3 million pounds of ice.

At the entrance to the ice compartment are 24 sets of double doors that can be opened completely by about a 1-psfd pressure in the lower compartment. The air/steam mixture passes through the lower inlet doors, is directed up by turning vanes, and passes through flow straighteners that distribute the gas uniformly across the compartment. The flow passes up through the ice basket array to condense the steam and vapors while particles are removed by deposition on structural surfaces, ice and water. The flow continues into the upper compartment of the containment through doors at the top of the compartment and at the top of the crane area.

After equilibrium is attained between the lower and upper compartments, the air in the upper compartment is recycled to the lower compartment through ducts from the dome and 10 dead-ended (pocketed) spaces by two fan systems with independent duct systems, dampers, controls and power supplies.

### 3.2.7 Main Steam (Line) Isolation Valve Leakage Control System (BWRs)

The MSIVLC system controls and minimizes the release of fission products by directing the leakage from the closed main steam line isolation valve (MSIV) to the standby gas treatment system (SGTS) described in Section 3.2.2. Figure 3.11 (adapted from SPC Soyland reference) is a simplified schematic of the system.

The schematic does not show the numerous controls, valves, auxiliary lines, etc. that are required to assure that the system can respond to all possible events and meet the design criteria. Basically, two bleed lines remove leakage from the MSIVs for each main steam line. One bleed line is located between the fast closing inboard and outboard MSIV outside the primary containment. The other bleed line is located downstream of the fast-closing outboard MSIV and the slower closing downstream shutoff valve. Each bleed line accommodates up to about 100 scfh of steam flow.

In the event of a LOCA, both the inboard and outboard MSIVs close. If leakage occurs, the bleed lines for the inboard and outboard MSIVLC systems can be initiated by an operator. Initially, the steam is vented through the depressurization branch where it is discharged into a building zone served by the SGTS. After the steam lines are depressurized, the depressurization line valves are closed and the steam is exhausted directly into the SGTS via the blower fan line valves.

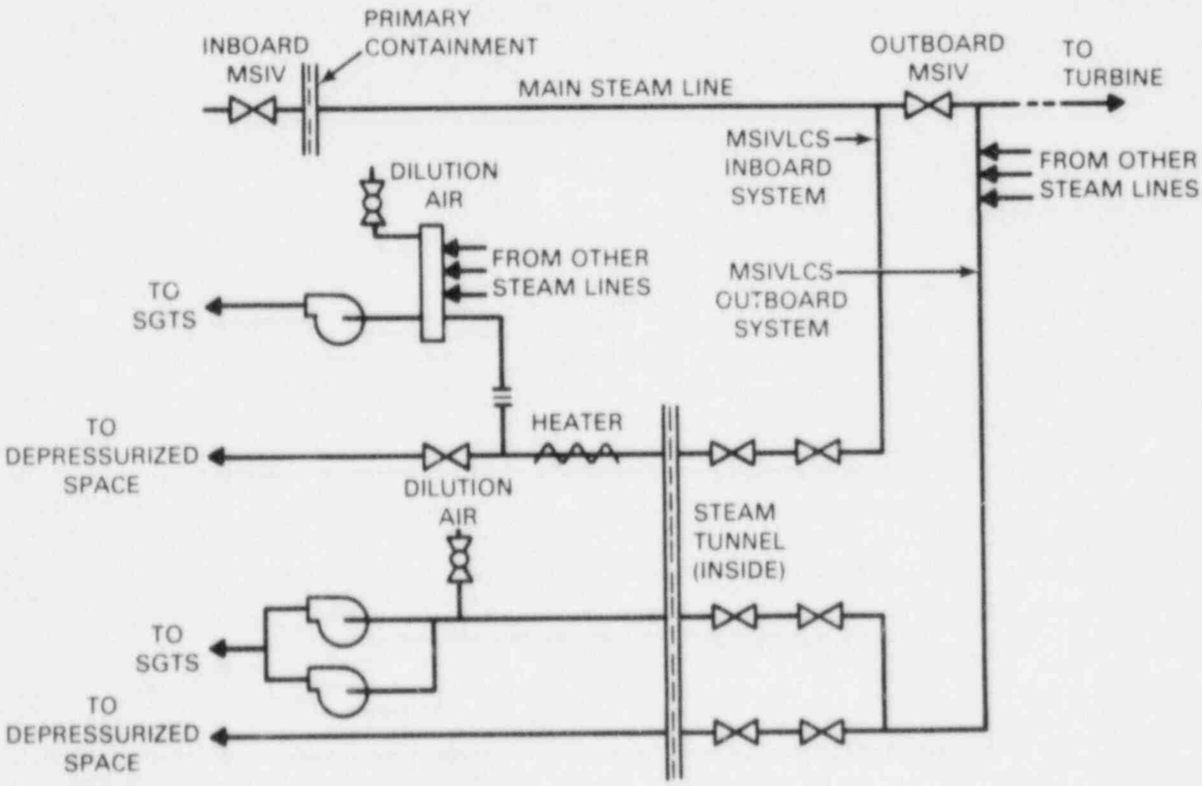


FIGURE 3.11. Main Steam Line Isolation Valve Leakage Control System

Before the effluent from the inboard MSIVLC system reaches the manifold, the effluent is passed through a heater to evaporate any condensate. At the manifold steam from other steam lines and dilution air (at a 1:5 ratio) are blended and passed via a fan to the SGTS before release to the environment.

The outboard MSIVLC system effluents are joined with steam from other sources at a bleed header and are moved through a pair of fans directly to the SGTS.

### 3.2.8 Filtered Vent Containment

A concept that is not incorporated in current designs as an ESF, Filtered Vent Containment, has received considerable attention and is described in this section.

Filtered Vent Containment (FVC) systems are used to prevent nuclear reactor containments from being overpressurized during postulated severe accidents. A large variety of FVC designs have been proposed (unfiltered vent designs have even been proposed for BWRs where the suppression pools can be used to remove the fission products and particulates before release). Although vents are designed in many ways, the most common system proposed is to allow pressure within containment to build up to near the structural design value. A pressure relief valve would then open to allow excess pressure to be released through a filter system and then to the atmosphere. After the pressure within the containment falls below the setpoint, the relief valve would reclose. Some systems provide lower pressure relief setpoints, and some may collect the vented material in a large tank or compartment instead of releasing it to the atmosphere. Some examples are described in the following paragraphs.

A FVC concept developed for use in Liquid Metal Fast Breeder reactors but applicable to Light Water Reactors (LWRs) is the Submerged Gravel Scrubber (SGS) (Hilliard, McCormack and Postma 1981). The basic concept is shown schematically in Figure 3.12. The SGS takes advantage of the passiveness and high loading of a water pool combined with the high particulate collection efficiency of a sand and gravel bed. Gas laden with particulate material is discharged beneath the gravel, where it subsequently flows upward through the bed. The apparent density of the two-phase mixture in the gravel region is less than that of the water pool outside the gravel bed, and liquid flows upward at a significant rate. Particle removal efficiencies are said to be better than efficiencies for either a dry gravel bed or a simple water pool. Gases leaving the submerged bed may be further cleaned by incorporating a high-efficiency fiber demister.

An alternative approach is shown in Figure 3.13 (Hoegberg et al. 1981). It consists of a crushed rock condenser (approximately 30,000 $m^3$) followed by a filter consisting of a 100- to 200-$m^3$ water pool. The steam and gases enter the upper compartment of the tunnel and then flow downward through the gravel

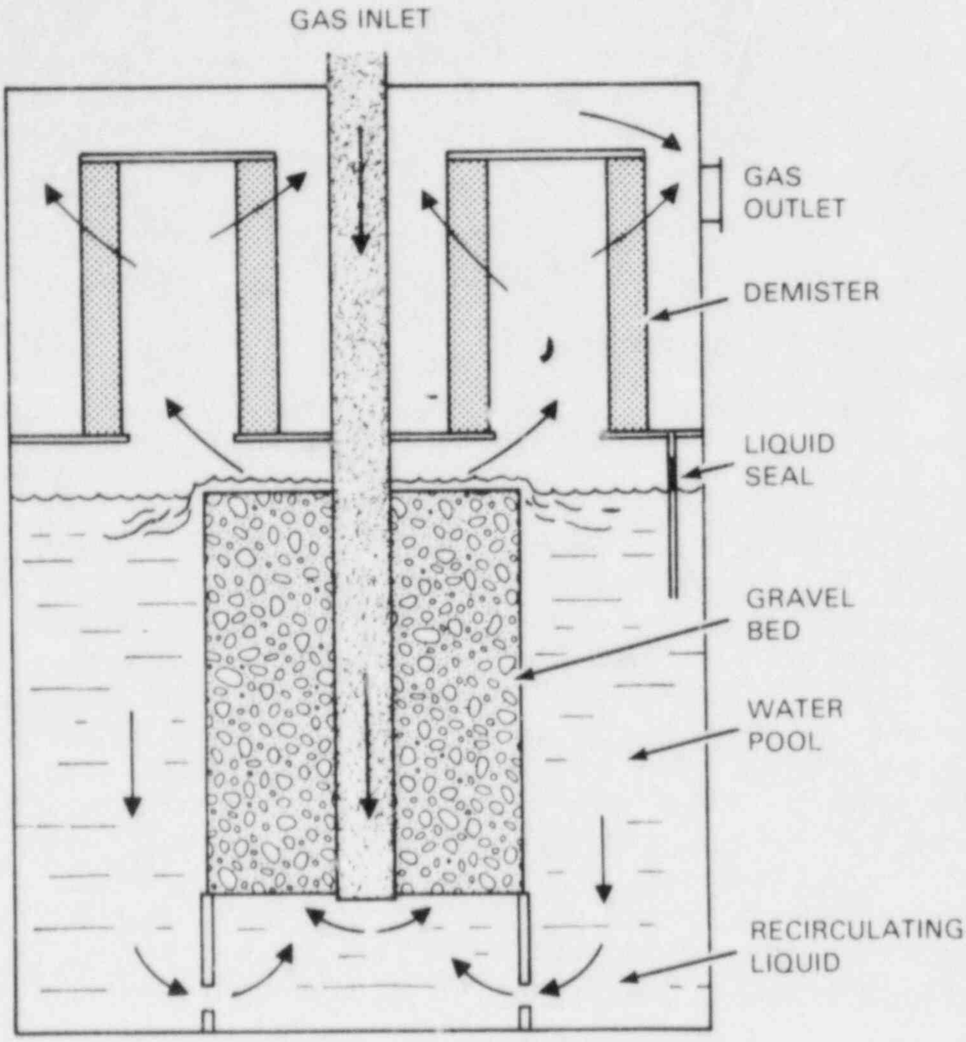


FIGURE 3.12. Submerged Gravel Scrubber (Hilliard, McCormack and Postma 1981)

bed. The steam condenses and the gas is delayed and filtered before exiting to the water filter from the bottom of the crushed rock bed. The total volume of the condenser can be increased or decreased for varying loadings, and the capacity of the unit can be adjusted by varying the number and length of the tunnels used.

Other concepts include providing a heat exchanger for the submerged bed, using a submerged bed after the suppression pool in a BWR, providing zeolite-charcoal filters after steam condensation, and providing noble gas hold-up capability. Venturi scrubbers and sand bed filters have also been considered.

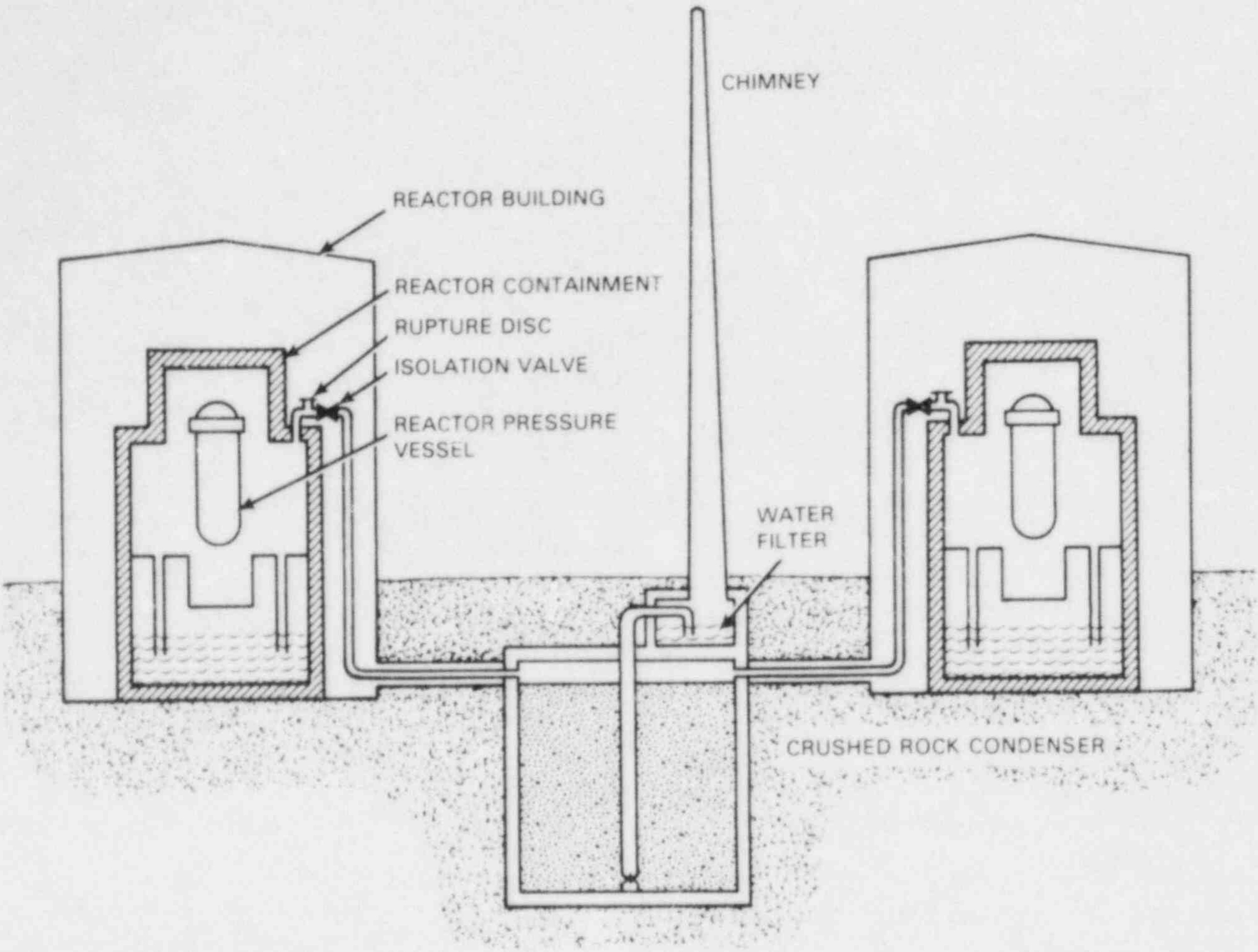


FIGURE 3.13. FILTRA--Filtered Pressure Relief of a Reactor Containment (Hoegberg et al. 1981)

Some competing risks that may occur with a properly operating FVC are

- Actuation of the vent may produce unnecessary releases in cases where the containment internal pressure exceeds the pressure set-point but would not have failed containment.

- During certain accidents, venting can cause a reduction of the containment noncondensible gas inventory, resulting in a loss of net positive suction head (NPSH) for residual heat removal pumps.

- During the same accidents, an inadvertent operation of containment sprays could lead to the development of a severe vacuum in containment.

- Venting over a long period of time can deplete the water inventory in the containment sump.

## 3.3 PRESENT USE OF ESF SYSTEMS IN U.S. LWRs

The incorporation of various ESF systems into the design of power plants varies with the type of reactor and the NSSS. The data extracted from interpretation of the information contained in PSAR and FSAR summary reports on the utilization of various ESF systems is shown graphically in Figure 3.14. These values are to be considered estimates; a survey of utility companies would be required for more precise values.

The graph shows that nearly all the plants incorporate some sort of secondary filter and containment spray systems. Primary containment recirculation coolers (chiller/coolers) are used extensively in PWRs as both an ESF system and a normal operating system while such units are used in BWRs for normal operations only. Suppression pools and MSIVLC systems are only used in BWRs. Some PWRs by all three NSSS vendors have used containment recirculating filter systems. Ice condensers are only found in the PWR Westinghouse plants.

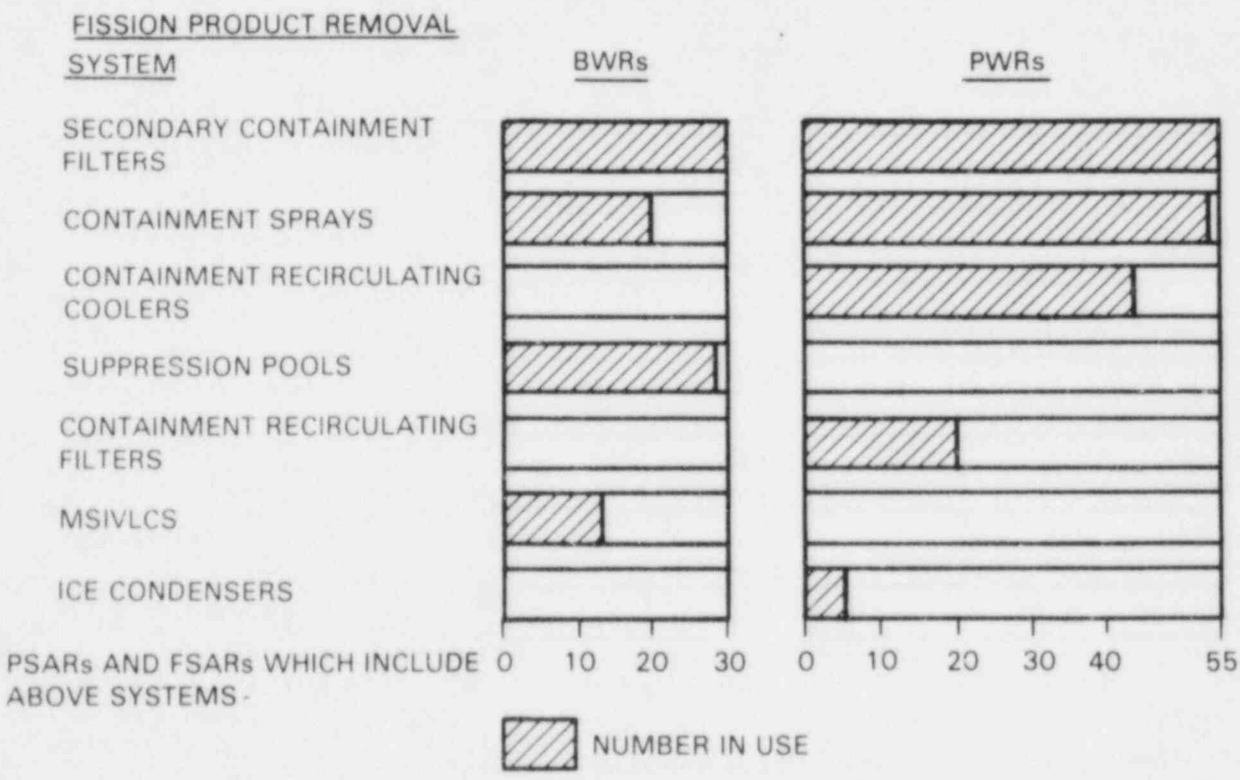


FIGURE 3.14. Estimated Current Utilization of ESF Systems

# 4.0 ACCIDENT ENVIRONMENTS

One of the major considerations in predicting the effectiveness of a particular ESF is the definition of the environment or conditions that could conceivably challenge the system. This report section presents information developed concerning such conditions, including information related to temperature, pressure, aerosol concentration, and particle size. The information is based on either past assessments of postulated accident sequences, ranging from the design basis to severe accidents, or the results of experimental investigations conducted by others.

## 4.1 SEQUENCE-BASED ASSESSMENTS

In general, sequence-based assessments can be described as either consequence- or risk-based investigations. The latter include consideration of sequence likelihood as well as consequences. Consequences are usually expressed in terms of the extent of radionuclide release from containment (source terms) or the resulting radiological effects. In terms of sequences the probabilistic risk assessments tend to be more comprehensive, while assessments based solely on consequences tend to focus on specific kinds of scenarios. In either case, several quantities need to be defined: the material released directly from the reactor core, subsequently released materials that are the result of interactions between molten and structural materials, and finally the extent of the transport and retention of these released materials. Such an analysis of material behavior, by following material transport from the core region to the final release to the environment, can be extremely valuable in predicting conditions that could affect ESF system performance.

Nine probabilistic risk assessments (the last nine entries in Table 4.1) were considered in the development of information for this background report. The Reactor Safety Study covering the Surry plants (the first entry in Table 4.1) could be considered a precursor of later probabilistic risk assessments (Ritzman et al. 1975). Information available from the risk assessments is considered in the following sections. Where possible, information from the risk assessments is combined with information from recent consequence-based investigations to provide "... a description of the best technical information currently available for estimating the release of radioactive material during postulated severe accidents in commercial light water reactor nuclear power plants, (USNRC 1981).

Several of the reactor plants discussed in the following sections are the subjects of ongoing investigations. These studies will undoubtedly provide new insights into the definition of atmospheres that might challenge ESF systems.

TABLE 4.1. Reactors Covered by Probabilistic Risk Assessments

| Reactor | Type | Program | Year |
|---|---|---|---|
| Surry[a] | PWR | Reactor Safety Study (RSS) | 1975 |
| Peach Bottom[a] | BWR | Reactor Safety Study (RSS) | 1981 |
| | | RSSMAP[b] | 1981 |
| Sequoyah[a] | PWR | RSSMAP | 1981 |
| Oconee | PWR | RSSMAP | 1982 |
| Calvert Cliffs | PWR | RSSMAP | 1982 |
| Grand Gulf[a] | BWR | RSSMAP | 1981 |
| Oconee | PWR | RSSMAP | 1981 |
| Limerick | BWR | Utility Sponsored | 1982 |
| Indian Point | PWR | Utility Sponsored | 1982 |
| Zion | BWR | Utility Sponsored | - |

(a) Plants also covered by recently completed (or in-progress), detailed and consequence-based assessments--see text.
(b) Reactor Safety Study Methodology Applications Program (RSSMAP).

One of the studies,[c] essentially an extension to NUREG-0772 (USNRC 1981), represents the identification and formulation of a systematic, mechanistic approach to estimating source terms for selected postulated sequences for PWR ice condenser containments and large, dry containments (Surry and Sequoyah plants) as well as for BWR Mark I and Mark III plants (Peach Bottom and Grand Gulf). In a similar but independent effort, the nuclear industry is sponsoring the Industry Degraded Core Rulemaking program (IDCOR). The purpose of the IDCOR program is to develop a comprehensive, integrated, technically sound position to aid in determining whether changes in regulations are needed to reflect degraded-core and core-melt accidents.

#### 4.1.1 Surry

The Surry #1 and #2 power plants were included in one of the earlier comprehensive reviews of reactor safety, the Reactor Safety Study (Ritzman et al. 1975).

(c) J. A. Gieseke, P. Cybulskis, R. S. Denning, M. R. Kuhhman, and K. W. Lee. July 1983. Radionuclide Release Under Specific LWR Accident Conditions. BMI-2104, Vol. I, DRAFT, Battelle Columbus Laboratories, Columbus, Ohio 43201.

Surry #1 and #2 are Westinghouse subatmospheric pressure containment PWRs that began operations in the early 1970s.

The procedures and computer codes used in the Reactor Safety Study (RSS) were used as a basis for the MARCH code. The MARCH code calculates the thermal-hydraulic behavior of an LWR during a meltdown accident including a LOCA and a reactor transient (Bieniarz, Prassinos and Engi 1982). Although the MARCH code was not applied per se for this analysis, the results in the RSS are comparable to other safety and risk assessments using this code. The pressure containment response for nine combinations of events are given in section VIII of the RSS (Ritzman et al. 1975).

The first situation studied was the design basis accident with and without low-pressure recirculation (LPR) failure. With no failure, the pressure in the containment vessel rapidly reaches 54 psia, then falls to normal atmospheric pressure. With LPR failure, the containment pressure starts to climb again and ultimately reaches 22 psia at the time of pressure vessel melt-through.

With failure of the containment spray injection system (CSIS), the situation is similar, although the highest pressures are about 60 psia. With failure of both CSIS and LPR, the containment pressure again increases up to 30 psia after reduction of the initial pressure spike.

When the containment spray recirculation system (CSRS) fails, the pressure fails containment after reaching approximately 120 psia. When the containment heat removal system (CHRS) fails, the situation is similar to the loss of the CSRS. When both CHRS and CSRS fail, the initial blowdown results in a pressure of 55 psia in the containment vessel.

For CSIS and CSRS failure, the initial pressure surge does not decrease, and pressure in the containment vessel increases until failure of the containment vessel at approximately 120 psia. For the loss of all three of the systems (CSRS, CSIS, and LPR), the result would be about the same.

For loss of electric power, the pressure within the containment increases rapidly to about 80 psia and then decreases to approximately 60 psia as the core melts through the basemat.

Failure of the emergency core injection (ECI) system was calculated to result in a pressure decrease to 20 psia and a subsequent increase to 30 psia. Loss of both the ECI and the CSIS results in a much slower pressure decrease to 20 psia.

For ECI, CSIS, and CSRS failure, the pressure initially rises to 80 psia and then decreases to 40 psia at the time of pressure vessel melt-through. Loss of the ECI and CHRS was calculated to result in a pressure decrease to 20 psia and a subsequent rise to containment failure at approximately 120 psia. For failure of both the ECI and CSRS, the pressure was calculated to rise to 60 psia at the time of pressure vessel melt-through.

The Surry facility was also studied in NUREG-0772 (USNRC 1981), a report to provide the best technical information available at that time for estimating the release of radioactive materials during postulated severe accidents in commercial LWR nuclear power plants. Seven accident sequences were considered for this consequence-based assessment. Peak atmospheric temperatures and pressures, peak aerosol concentrations, and the total aerosol released were covered for each sequence. Table 4.2 (taken from NUREG-0772) tabulates the results. An excerpt from NUREG-0772 follows.

> "For most sequences in which containment cooling is not functioning, the containment pressure would be elevated and the temperature would be saturated at the steam partial pressure during the period of fission product release. In some cases, the quantity of steam in the atmosphere would be high enough to suppress hydrogen flammability. In other sequences, however, a hydrogen deflagration event would be possible at least during some phase of the accident.
>
> In the Reactor Safety Study, some sequences were identified in which containment failure was predicted to precede core meltdown. For example, in the $S_2$ sequence (a small diameter pipe break LOCA), loss of ability to remove heat from the containment atmosphere would lead to a steady increase in containment pressure which would eventually result in containment failure. Depressurization of the containment building would lead to cavitation and failure of the emergency core cooling pumps with subsequent fuel uncovery and meltdown. The condition in the containment at the time of fission product release would depend upon the mode of containment failure. For a localized failure in containment, the pressure could be elevated, ranging from atmospheric pressure to containment failure pressure, depending on the size of the leak. The atmosphere would primarily be composed of steam and hydrogen at a temperature approximately equal to saturation at the steam partial pressure. The amount of air would be depleted due to release from containment. If the failure mode of the containment were massive, air circulation into the containment would be expected. The pressure in the containment would be near atmospheric. Although temperature could be very hot near the point of the break in the primary system, sharp temperature gradients would exist in the containment atmosphere determined by the circulation patterns of incoming air. Since there is a great uncertainty in the mode of containment failure, the latter assumption is usually made in analyzing this type of sequence. However, it should be recognized that the size of the breach in containment can affect not

TABLE 4.2. Containment Parameters, Dry (Surry)

| Containment Parameter | Terminated LOCA | TMI Type | Terminated AD | Severe $S_2D$ | Severe TMLB | Severe TMLB | Severe $S_2D$ |
|---|---|---|---|---|---|---|---|
| Spray Operation? | Yes | No | Yes | Yes | No | NO | No |
| Recirc. Filter Operation | No | Yes | No | No | No | No | Yes |
| Hydrogen Burn | No | Yes | No | No | No | No | No |
| Steam Explosion | No | No | No | No | No | No | No |
| Time of Containment Failure, hr | None | None | None | None | 4.7 | None | None |
| Peak Atmosphere Temperature, °C | 120[b] | 40[b] | 120[b] | [illegible] | 160 | 190 | 140 |
| Peak Atmosphere Pressure, MPa Absolute | 0.376[b] | 0.14[b] | 0.376[b] | 0.26 | 0.68 | 0.68 | 0.21 |
| Aerosol Mass Released, kg | <1[b] | <1[b] | 400[b] | 1110 | 1110 | 1110 | 1110 |
| Peak Aerosol Concentration, $g/m^3$ | $<1 \times 10^{-4}$[a] | $<1 \times 10^{-4}$[b] | 3.1[b] | 9.6 | 12.2 | 11.3 | 10.9 |
| Iodine Release, Fracture of Core Inventory[a] | $<2 \times 10^{-3}$ | $<6 \times 10^{-4}$ | 0.50[b] | 1.0 | 1.0 | 1.0 | 1.0 |
| Iodine Form: | | | | | | | |
| Fraction CsI[b] | 0.99-1.0 | 0.99-1.0 | 0.99-1.0 | 0.99-1.0 | 0.99-1.0 | 0.99-1.0 | 0.99-1.0 |
| Fraction $I_2$[b] | 0-0.01 | 0-0.01 | 0-0.01 | 0-0.01 | 0-0.01 | 0-0.01 | 0-0.01 |

(a) Iodine released as aerosol or gas to the containment atmosphere.
(b) Numerical value for this parameter was assumed for purposes of evaluating ESFs only.

only the conditions in the containment but also the time available for deposition. The result of the analyses can therefore be quite sensitive to the assumed failure mode."

### 4.1.2 Peach Bottom

Both Peach Bottom #2 and #3 were included in the RSS (Ritzman et al. 1975). Both are GE BWR 4/Mark I models which began operations in the early 1970s. Containment response analyses were performed for 34 sequences, all variations of the design-basis accident. The first sequence considered failure of the emergency core cooling system (ECC). The blowdown spike was calculated to be about 60 psia; this initial spike is followed by a pressure drop and gradual rise to the same level. The pressure rises sharply thereafter to containment failure, which was postulated to occur at about 180 psia. The response of the system to a loss of the ECC after it begins operations is similar to the above.

The sequences with the loss of long-term cooling showed a gradual rise from the initial pressure of about 40 psia to the failure pressure of the containment, 180 psia. This type of sequence takes about 24 hours to reach containment failure, as opposed to the 3 to 4 hours required in the case of loss of the ECC.

Peach Bottom was also covered in the consequence-based assessment presented in NUREG-0772 (USNRC 1981). The results from that study are shown in Table 4.3 taken from that report. The report states,

> "The three types of BWR containment design are similar in concept. Some differences in the design can influence the conditions in the containment volume during accident sequences, however. The responses of the Mark I and Mark II design would be nearly the same for most accidents. The current intent of the NRC is to have all Mark I and II plants inerted. Up until the time of failure of containment, an oxidizing atmosphere would not, therefore, exist in the drywell. It should be recognized, however, that in some sequences, such as TW (transient event tree with failure to remove residual core heat), containment failure would precede core meltdown and that air ingress cannot be precluded. One major difference in accident behavior between Mark I and Mark II designs relates to the location of the suppression pools. For the Mark II design, the molten core would penetrate into the suppression pool in the Stage 2 time period. The Mark III design is in many respects more similar to the PWR ice condenser design than to the other BWR designs. In this design, the vapor space above the suppression pool is actually an outer containment volume.

TABLE 4.3. Containment Parameters, Pressure Suppression Pool (Peach Bottom)

| Containment Parameters | Mark I Pool AE | Mark I Pool TC | Mark I Pool TW | Mark III Pool TQUV |
|---|---|---|---|---|
| Spray Operation (PWR) | -- | -- | -- | -- |
| Ice Available (PWR) | -- | -- | -- | -- |
| Pool Subcooled (BWR) | Yes | No | No | Yes |
| Hydrogen Burn | No | No | No | No |
| Time of Containment Failure, h | 0.8 | 1.5 | 55.3 | 6.7 |
| Steam Explosion | No | No | No | No |
| Peak Atmosphere Temperature, °C | 417 | 592 | 262 | 440 |
| Peak Atmosphere Pressure, MPa Absolute | 1.21 | 1.21[b] | 1.21[b] | 0.31 |
| Aerosol Mass Released, kg | 1110 | 1110 | 1110 | 1110 |
| Peak Aerosol Concentration, $g/m^3$ | 40.9 | 55 | 10 | 12.9 |
| Iodine Release, Fraction of Core Inventory[a] | 1.0 | 1.0 | 1.0 | 1.0 |
| Iodine Form: | | | | |
| Fraction CsI | 0.99-1.0[b] | 0.99-1.0[b] | 0.99-1.0[b] | 0.99-1.0[b] |
| Fraction $I_2$ | 0-0.01[b] | 0-0.01[b] | 0-0.01[b] | 0-0.01[b] |
| Leak Path for Mark I | Annulus | Annulus | Direct | -- |

(a) Iodine released as aerosol or gas to the containment atmosphere.
(b) Numerical value for this parameter was assumed for purposes of evaluating ESFs only.

While the drywell is intact, the suppression pool can be an effective scrubber of fission products as well as a condenser of steam for each of the designs. Gases released from the vessel during an accident would either be discharged directly into the suppression pool or, having been released to the drywell, would flow through vent pipes into the pool. Even if the water in the suppression pool is saturated, the pool may still have some effectiveness in removing fission products from the gas stream prior to release to the wetwell vapor space.

Once the containment boundary fails, the subsequent pathway of release of fission products to the environment and the amount of retention in the pathway would be sensitive to the location of failure. For the Mark I design, failure could occur either in the drywell or the torus. Failure in the drywell would lead to bypassing of the suppression pool for the remainder of the accident. Whether or not failure in the torus region would also prevent further scrubbing by the pool would depend upon the type of failure. Following failure in the drywell or torus, fission products either travel up the narrow annular space surrounding the drywell before release to the operating floor of the secondary containment, or are released to the lower compartment of the secondary containment building or directly to the environment, depending upon the location and mode of failure.

Because of the small volumes of the BWR design, hydrogen generation presents a considerable problem as a non-condensible gas which is predicted to eventually lead to failure of the containment by overpressurization. Since the Mark I and II designs will probably be operated in an inert mode in the future, hydrogen deflagration would in general not be possible for these designs. Since the containment volume is small and the suppression pool will in general be expected to suppress the release of steam to the outer containment volume, flammable conditions could be expected by the end of Stage 1 for most accident sequences in the Mark III design."

### 4.1.3 Sequoyah

The Sequoyah #1 power plant was the first facility studied in the Reactor Safety Study Application Program (RSSMAP) (Carlson et al. 1981). Sequoyah #1 is a Westinghouse ice condenser PWR that began operations in October of 1980.

MARCH code analyses were performed for many sequences, but graphical results were not presented. The only analytical results presented were in tables of times to core melt, pressure vessel failure, and the initiation of concrete melting. Since none of these times are of primary concern in our study, they are not considered here.

A generic ice condenser facility was included as a part of NUREG-0772 (USNRC 1981). The results from that study are presented in Table 4.4. Its performance under severe accident conditions is described in NUREG-0772 as, "Since the ice-condenser is a passive safety feature, it would be expected to function in most meltdown sequences. The ice would not only be effective in condensing steam but would act as a filter for fission products. After some period of time, however, the ice would be completely consumed. The timing of core meltdown, relative to the availability of ice, is therefore critical in determining the amount of decontamination available. The volume of the containment is comparatively small so that flammable conditions can be readily achieved in the upper compartment. Whether or not flammable conditions would occur in the lower compartment depends on whether or not recirculating fans are operating.

### 4.1.4 Oconee

The Oconee #3 power station was the second plant studied in the RSSMAP (Kolb et al. 1981). Oconee #3 is a Babcock and Wilcox PWR with a large, dry containment vessel. The plant began operations in April 1974.

MARCH analysis were performed for many sequences, but only results from the core melt sequences were shown. These core melt sequences included four LOCAs, the interfacing system accident, and four transient-induced sequences. Detailed MARCH analyses results are covered for six scenarios -- TMLQD, $S_3D$, V, TMLU, $T_1(B_3)MLUOO'$, and TMLOOO' (the accident scenario nomenclature is explained below). A total of 11 plots generated by the MARCH code were shown in the PRA. The data of interest to this report are the total and partial $H_2$ pressure for the containment and the containment temperature.

For the TMLQD sequence (a transient-induced event tree with failure of the Power Conversion System, Emergency Feedwater System, recovery of the Power Conversion System, reclosure of Pressurizer Safety or Relief Valve, and the Emergency Coolant Injection System), the total pressure in the containment shows a spike of 45 psia at the time of head failure, followed by a gradual rise to 45 psia. Discussion in the text of the report indicates that the containment pressure would peak at 75 to 80 psia if fragmentation of the core and rapid boiling were assumed. This pressure is below the estimated containment failure pressure of 133 ± 20 psia. The partial hydrogen pressure reaches approximately 5 psia shortly after head failure and stays approximately constant for the rest of the calculation with a maximum burn pressure of about 180 psia. The containment temperature shows a spike of 240°F at the time of head failure, followed by a period of constant temperature at about 150°F.

TABLE 4.4. Containment Parameters, Ice Condenser (Sequoyah)

| Containment Parameters | Ice Condenser TMLB' | Ice Condenser $S_2HF$ | Ice Condenser AD |
|---|---|---|---|
| Spray Operation (PWR) | No | up to 1.9 h | up to 1.1 h |
| Ice Available (PWR) | Yes | No | Yes |
| Pool Subcooled (BWR) | -- | -- | -- |
| Hydrogen Burn | No | No | At 1.1 h |
| Time of Containment Failure, h | 4.0 | 3.15 | 1.1 |
| Steam Explosion | No | No | No |
| Peak Atmosphere Temperature, °C | 137 | 212 | 253 |
| Peak Atmosphere Pressure, MPa Absolute | 0.29 | 0.29 | 0.29 |
| Aerosol Mass Released, kg | 1110 | 1110 | 1110 |
| Peak Aerosol Concentration, $g/m^3$ | 15.2 | 19.5 | 11.6 |
| Iodine Release, Fraction of Core Inventory[a] | 1.0 | 1.0 | 1.0 |
| Iodine Form: | | | |
| Fraction CsI | 0.99-1.0[b] | 0.99-1.0[b] | 0.99-1.0[b] |
| Fraction $I_2$ | 0-0.01[b] | 0-0.01[b] | 0-0.01[b] |
| Leak Path for Mark I | -- | -- | -- |

(a) Iodine released as aerosol or gas to the containment atmosphere.
(b) Numerical value for this parameter was assumed for purposes of evaluating ESFs only.

Results for the Oconee $S_3D$ sequence (a small break LOCA, D less than 4 inches, with failure of the Emergency Coolant Injection System), were not available in graphic form. Discussion in the text indicates that the pressure in containment is likely to remain below failure pressure unless hydrogen burning occurs.

The V sequence (interfacing system LOCA) results were also not presented in the form of plots. Since this sequence postulates the bypassing of all ESF in the reactor and results in release to a building that is postulated to fail under these conditions, the sequence is not of great interest to this study.

The TMLU sequence (a transient event tree with failure of the Power Conversion System, Emergency Feedwater System, recovery of the Power Conversion System, High Head Auxiliary Feedwater System, and High Pressure Injection System) was represented by a plot of the containment pressure with and without hydrogen burning. A pressure spike of about 160 psia was predicted for hydrogen burning, dropping to 110 psia and followed by a gradual rise to 130 psia. These values are very close to the anticipated containment failure pressure. No additional data were presented.

The Oconee $T_1(B_3)$MLUOO' (a transient event tree with failure of the Emergency Power System, Power Conversion System, Emergency Feedwater System, recovery of the Power Conversion System, High Head Auxiliary Feedwater System, High Pressure Injection System, Air Recirculation and Cooling System, and Containment Spray Injection System) is a transient with a complete loss of electric power. Since Oconee is a large containment PWR, loss of the Air Recirculation and Cooler and Containment Spray Injection System disables all the ESFs in the facility. Thus this sequence is also of little interest to this study. A pressure spike of 120 psia occurs at the time of head failure, followed by a gradual rise to approximately 130 psia. The calculations did not go beyond this point, which is the nominal failure pressure for containment. Hydrogen partial pressures are generally low enough that the mixtures are generally not flammable.

The TMLOO' sequence (a transient event tree with failure of the Power Conversion System, Emergency Feedwater System. Recovery of the Power Conversion System, High Head Auxiliary Feedwater System, Air Recirculation and Cooling System, and Containment Spray Injection System) was represented by a single plot showing a gradual rise of pressure to containment failure at about 133 psia. Again, this sequence is of little interest to our study since it does not allow the functioning of any ESF.

### 4.1.5 Calvert Cliffs

The Calvert Cliffs #2 power plant was the third facility studied in the RSSMAP (Hatch et al. 1982). Calvert Cliffs is a Combustion Engineering PWR (large, dry containment) which began operations in August 1976.

The first sequence examined was TML (a transient event tree with failure of the Power Conversion System, Auxiliary Feedwater System or recovery of the Power Conversion System). The containment pressure peaks at 160 psia, which is well beyond the containment failure level. The steady-state pressure is about 30 psia. If hydrogen burning occurs, the burn pressure can reach 160 to 170 psia. The mole fraction of steam, hydrogen, and oxygen as a function of time were also plotted. Steam mole fractions range from 0 to 0.85 while the hydrogen mole fractions range from 0 to about 0.2. The oxygen mole fraction is normally about 0.2 but does drop to about 0.04 when the steam mole fraction is at its highest. The containment temperature ranges from about 100°F to 350°F (at the time of head failure) with a long-term temperature of about 175°F. Wall temperatures in the containment are at about the same levels.

A second series of calculations was performed for the same sequence with a hydrogen burn occurring at the time of head failure. In this case, the pressure spike is 140 psia, and the compartment temperature spike is about 2000°F with a long-term temperature of about 200°F. Wall temperatures reach 300°F with a long-term temperature of about 200°F.

The TMLOO' sequence (a transient event tree with a failure of the Power Conversion System, Auxiliary Feedwater System or recovery of the Power Conversion System, Containment Air Recirculation and Cooling System, and Containment Spray Injection System). This sequence is similar to TML and is of little interest to this study because no containment safeguards are involved. In this case, the pressure spike reaches 120 psia with a fairly long-term (several hours) pressure of 80 psia. Containment temperature reaches 500°F and then drops off following head failure, which occurs at about the same time as containment failure.

Data on the TMQ sequence (a transient event tree with failure of the Power Conversion System, and reclosure of Pressurizer Safety/Relief valve) was limited to information on the primary system characteristics and is of little value to this study.

All LOCA sequence data again dealt with characteristics of the primary system and are not discussed here.

### 4.1.6 Grand Gulf

The Grand Gulf #1 power plant was the fourth facility studied in the RSSMAP (Hatch, Cybulskis, and Wooton 1981). Grand Gulf #1, which is a General Electric BWR 6 with a Mark III containment, began operations in June 1982. Because Grand Gulf is a BWR with both a drywell and a wetwell, both areas are considered in our discussions.

The first sequence analyzed was TQW (transient event tree with failure of the Power Conversion System and the Residual Heat Removal System after the transient). The drywell pressure rises to about 65 psia and then continues gradually up to a peak of 90 psia. The wetwell or containment pressure plots follow approximately the same outline. Since normal failure pressure is about 45 psia, failure is certain to occur. These same plots show what fraction of the pressure is due to hydrogen, other noncondensible gases, and steam. From these plots, mole fractions of these materials can be estimated. The hydrogen mole fractions ranges from 0 to about 0.22; other noncondensible gases (presumably oxygen and nitrogen) range from 0.08 to about 1.0; and the steam mole fraction ranges from 0 to about 0.85. All the aforementioned values are for the drywell. For the wetwell, hydrogen ranges from 0 to about 0.2, noncondensible gases from 0.08 to 1.0, and steam from 0 to about 0.5. Wetwell temperature rises gradually to 250°F, while drywell temperatures rise to spikes of 500°F and 900°F with long plateaus at both temperatures.

The next sequence analyzed was TPQI (a transient event tree with failure of a safety/relief valve, Power Conversion System, Residual Heat Removal System after LOCA, including transient-induced LOCAs). The only relevant data showed containment pressure peaking at 55 psia, at which time failure occurred.

The next series of sequences involved coolant makeup failure accidents. The sequence TPQE (a transient event tree with failure of a safety/relief valve, Power Conversion System, and the Emergency Core Cooling System) shows drywell pressure gradually increasing to 65 psia, while wetwell pressures follows about the same course. The estimated mole fraction ranges for hydrogen, steam, and noncondensible gases for the drywell are 0 to 0.15, 0 to 0.2, and 0.5 to 1.0, respectively. Corresponding values for the wetwell are 0 to 0.16, 0 to 0.2, and 0.5 to 1.0. Hydrogen burn pressures are about 250 psia for the wetwell and 50 psia for the drywell. Drywell temperatures peak at about 1000°F with an extended period of time above 800°F. Wetwell temperatures stay fairly constant at about 120°F.

The next sequence covered was TQUV (a transient event tree with failure of the Power Conversion System, the High Pressure Core Spray and Reactor Core Isolation Cooling System, and the Low and High Pressure ECC Systems to provide core flow). In this sequence, drywell pressures were calculated to reach 60 psia, at which time the calculations were stopped. Wetwell pressure follows the same pattern. Steam, hydrogen, and noncondensible gas mole fractions range from 0 to 0.33, 0 to 0.16, and 0.5 to 1.0, respectively, in the wetwell. The ranges of drywell mole fractions are 0 to 0.75, 0 to 0.16, and 0.12 to 1.0, respectively. Hydrogen burn pressures are high enough to fail the containment. The drywell temperature peaks at 1000°F and stays above 800°F. The wetwell temperature remains below 150°F.

The final sequence covered was the AE sequence (a large LOCA with failure of the ECC System). The drywell pressure is calculated to peak at 42 psia and then continues up indefinitely, as does the wetwell pressure. The calculated ranges for the drywell gas mole fractions are 0 to 0.16 for hydrogen, 0 to 0.9 for steam, and 0 to 1.0 for the noncondensible gases. The wetwell ranges are 0 to 0.3 for hydrogen, 0 to 0.15 for steam, and 0.4 to 1.0 for the noncondensible gases. Drywell temperatures peak at 1100°F and remain at 900°F for extended periods. Wetwell temperatures reach about 170°F.

### 4.1.7 Limerick

Limerick Generating Station was studied by the Philadelphia Electric Company, the General Electric Company, and Science Applications, Inc. (PELEC 1982). Limerick #1 and #2 are General Electric BWR 4/Mark II BWRs scheduled to begin operation over the next two years.

Limerick was examined using the INCOR code package (a computer code package for predicting pressure-temperature response while determining time to core uncovery, core melt, pressure vessel melt-through, and molten core interactions with concrete; PELEC 1982) rather than MARCH. Four sequences were examined and the temperature and pressure in each sequence presented.

The first sequence covered was TQUV (a transient event tree with failure of Power Conversion System, the High Pressure Core Spray and Reactor Core Isolation Cooling System, the Low and High Pressure ECC Systems to provide core flow). In this sequence the drywell pressure flattens to a plateau at 20 psia, then increases to 90 psia. The wetwell pressure follows a similar curve, although the pressure never gets above 60 psia. The wetwell temperature rises gradually to 165°F.

The next sequence covered was TW (a transient event tree with failure of the Power Conversion System). In this sequence, the drywell pressure rises to a peak of 160 psia and then decays to about 30 psia. The wetwell pressure rises to 165 psia at which time containment fails. The wetwell temperature rises to about 350°F at the time of containment failure.

The next two sequences covered were anticipated transients without scram. Both cases assumed failure of the coolant inventory makeup. Case #1 assumed containment failure after core melt, while case #2 assumed containment failure before core melt. In the first case, the drywell pressure rises to 25 psia, levels off, then rises to 90 psia. In case #2, the drywell pressure peaks at 120 psia, at which time containment fails. The wetwell pressure for case #2 reaches 160 psia just before failure occurs. The wetwell temperature reaches 370°F and then decreases to about 240°F.

### 4.1.8 Indian Point

Indian Point Generating Station, consisting of two Westinghouse PWRs, was studied by the owning utilities (PASNY 1982). Indian Point #2 became operational in October 1971, while Indian Point #3 went on the line in December 1975.

The Indian Point study is one of the most complete studies examined. The sequences are divided into six classes. Class I considered large LOCAs with loss of coolant injection. Class II considered large LOCAs with loss of coolant injection at the time of coolant recirculation. Classes III and IV considered small LOCAs with some variability in the timing of the loss of coolant. Classes V and VI considered transient-initiated events with failure of all primary cooling sources. Class V sequences involved loss of secondary or containment cooling capability as well. Class VI sequences have functional containment safeguards.

In each class, the best-estimate scenario was analyzed along with multiple variations on the scenario. A total of 50 cases were examined in this study. MARCH outputs were given for many of the estimates.

For class I, the best-estimate scenario showed a peak pressure of 61 psia, which is well below the 141 psia postulated for containment failure. The pressure stays fairly constant at 25 psia. This study utilizes a calculated "flow temperature" to tell whether or not a hydrogen burn is possible. Many graphs were presented to show when this calculated value is near the critical flow temperature of 710°F. In this best-estimate scenario, hydrogen burning is not considered probable.

In examining the variations of case #1, there is little apparent difference in the result. The containment pressure reaches about 140 to 150 psia in several sequences but the long-term pressure remains about 25 psia. Burning of hydrogen is possible under certain conditions with a 150-psia pressure spike resulting from the worst burn.

For class II sequences, the numerical values of the pressure are about the same as for class I. Burning of hydrogen is possible.

For class III sequences, the best-estimate peak pressure is about 50 psia. Other cases reach peak pressures of 130 to 150 psia if hydrogen burning is allowed.

For class IV sequences, the best-estimate sequence gives a peak pressure of about 50 psia. The variations of the sequence give peak pressures of 75, 100, 60, 110, and 130 psia (the 130 psia is for the worst-case hydrogen burn). The class V best-estimate pressure peak is about 75 psia. The other cases give peak pressures of 80, 120, 160, and 130 psia. Results from the class VI sequences were found to give less severe pressures than from the class V sequences. Graphs from these calculations were not presented.

### 4.1.9 Zion

The Zion PRA (Com. Ed. 1981) covered the Zion facility, which is a large, dry Westinghouse PWR. This study is formatted along the same lines as the Indian Point report. Five classes of accidents were investigated with numerous variations on each scenario. The accident sequences covered are AD, AHF, $S_2D$, $S_2HF$, and TMLB. The report presents some 300 pages of graphs generated by the MARCH code.

For the AD sequence (a large-break LOCA with failure of the ECC System), pressures of 75 psia occur rapidly along with temperatures of up to 500°F without hydrogen burning. The mole fraction of $H_2$ reaches 0.15 if burning is not postulated. The adiabatic burn pressure goes over 200 psia. If hydrogen burning occurs at the time of vessel failure, a pressure of 150 psia and temperatures of 1100°F are postulated. If hydrogen burning occurs at the time of maximum containment temperature, the local temperature can briefly reach 3000°F.

Results of calculations for the AHF sequences (a large break LOCA with failure of the ECCS Recirculation System and Containment Spray System) indicate that the vessel fails at about 1500 sec with pressures up to 75 psia and temperatures up to 400°F. The mole fraction of $H_2$ reaches 0.17. If hydrogen burning occurs at the time of maximum containment temperature, containment temperature reaches 3145°F.

For the $S_2D$ sequence (a small break LOCA with failure of the ECCS System), pressures reach 75 psia with temperatures of 400°F at 5000 sec. For hydrogen burns at the time of vessel failure, the pressure reaches 140 psia and the temperature 1200°F. For hydrogen burns at the time of maximum containment temperature, the pressure reaches 150 psia and local temperature reaches 3100°F briefly.

Calculations for the $S_2HF$ sequence indicate that the pressure reaches 60 psia and the temperature reaches 400°F in 4000 sec. If a hydrogen burn occurs at the time of vessel failure, the pressure reaches 125 psia and the temperature reaches 1150°F. For hydrogen burns at maximum containment temperature, the temperature reaches 2880°F.

For the TMLB sequence (a transient event tree with failure of the Power Conversion System, Auxiliary Feedwater System, and electric power for 1 to 3 hours), the pressure reaches 130 to 140 psia (at the time at which the calculation was terminated) and the temperature reaches 450°F. In the cases where hydrogen burns occur at the time of vessel failure, the pressure peaks at 120 psia and the temperature reaches 800°F. In the cases where hydrogen burns occur after dispersion of the hydrogen, the pressure reaches 175 psia and the temperature peaks at 1300°F. In the cases where the hydrogen burns during the period of maximum containment temperature, the temperature reaches 3000°F and the pressure reaches 140 psia.

## 4.2 DATA FROM EXPERIMENTAL STUDIES

Included here are discussions of fission product release, aerosols, and the influence of natural processes on fission product removal.

### 4.2.1 Fission Product Release

One of the current methods for calculating the release of fission products as a consequence of core-melt accident conditions is described in NUREG-0772 (USNRC 1981) and is based upon experimental work performed prior to 1981 (Lorenz, Collins and Manning 1978; Lorenz, Collins and Malinauskas 1980a and 1980b; Parker et al. 1967; Parker, Martin and Creek 1963; Albrecht, Matschoss and Wild 1979a and 1979b). Prior to the release of any material, the initial barrier (the fuel cladding) must fail. Chung (1981) described the progression of fuel damage in phases as a function of temperature:

| | |
|---|---|
| (greater than 700°C) | - ballooning of Zircaloy cladding |
| (750-1070°C) | - rupture of Zircaloy cladding |
| | - oxidation of metal components/hydrogen generation |
| | - embrittlement by oxidation |
| (1400-1900°C) | - reaction between solid $UO_2$ and metallic Zircaloy |
| (greater than 1900°C) | - dissolution of $UO_2$ in Zircaloy-$ZrO_2$ eutectic |
| | - breach of $ZrO_2$ shell by $UO_2$ |
| | - flow-down of liquified fuel and Zircaloy |
| (greater than 1980°C) | - melting of remaining Zircaloy |
| (approx 2700°C) | - melting of remaining solid $ZrO_2$ |
| (approx 2820°C) | - melting of remaining $UO_2$ |

The effects of the other materials present (e.g. control rods), although not well understood, are under investigation.

The release of fission products during the various phases just described was investigated by Lorenz et al. (1978a and 1978b, 1979, 1980). From 0.25% to 25% of the total inventory of stable, long-half-life fission gas can be released by the initial rupturing of the fuel cladding. An additional 1 to 1.5% of the fission gas inventory, shallowly embedded in the fuel and cladding, will be released shortly thereafter. Smaller quantities of cesium and iodine will be released (for PWRs, approximately 0.02% of the stable cesium and 0.04% of the stable iodine). Cesium and iodine in the gap between the fuel and cladding diffuse out slowly during the next phase. Up to 20% of the noble gases, cesium and iodine (from high burn-up fuel) could be released from the grain boundary. The rate of release of the remaining noble gases, cesium and iodine doubles every 100°C during the diffusion from the $UO_2$ phase. At 2000°C, the rate is approximately 10%/min. Because of the uncertainties of the effects of other materials, the release of fission products from molten fuel is not well characterized.

Iodine is one of the fission products that has a significant effect upon the risk from core-melt accidents. Recent calculations indicate that the most probable form released is cesium iodide (CsI) with the remainder of the cesium as CsOH (USNRC 1981). Cesium iodide is less volatile than elemental iodine and is soluble in water, which would result in a reduced release during core-melt accidents. Tellurium that decays to iodine is also of concern. However, tellurium combines with the Zircaloy cladding to form compounds that show a strong tendency to plate-out on surfaces (Genco et al. 1969; Lorenz, Collins and Manning 1978; Allison 1965; Allison and Rae 1967).

The behavior of fission products released into containment was studied (Hilliard and Postma 1980 and 1981) for two sizes of vessels. The larger was approximately a 1/5-linear-scale model of a typical PWR containment vessel. Uranium dioxide spiked with $I_2$, $CH_3I$, and cesium was injected as a fuel simulant. Other fission products (e.g., Te, Ba, Ru, Xe, etc) were used in some tests. Typically, an atmosphere (250°F and 50 psia) containing steam was established and maintained during each experiment. In a series of tests to determine the effect of only natural and passive removal mechanisms, all of the fission products were retained in the release apparatus, removed by the surfaces in the system, or removed in leak paths from the vessel. Although efforts were made to release all the material, averages of 28% of the iodine and 67% of the cesium were retained in the release apparatus and injection line. Upon release into steamy atmospheres, the simulant appeared to act as condensation nuclei and formed fog droplets. The $I_2$ was rapidly absorbed into the drops and the initial removal half-times ranged from 9 to 24 min. Removal half-times for the cesium ranged from 8.5 to 50 min. After 75 minutes, the half-times slowed to 660 to 800 min for $I_2$ and 66 to 470 min for cesium. Uranium removal half-times were similar to those for cesium. Of the iodine released to the containment vessel, 50% was bound to the paint and 50% was in the condensate. Approximately 15% of the cesium was found in the paint and 85% in the condensate.

Several tests of reactors to destruction have provided information on the release of fission products. The results of the tests performed in the presence of water are summarized in Table 4.5. The results indicate the

TABLE 4.5. Summary of Reactor Tests with Water

| Reactor | Energy Released (MW·s) | Atmospheric Release | | Products | Reference |
|---|---|---|---|---|---|
| | | Noble Gas | Iodine | | |
| BORAX-1 | 135 | (a) | (a) | (a) | Dietrich 1957 |
| SPERT-1 | 31 | 7% | <.01% | <.79% | Miller et al. 1964 |
| | 165 | .06% | <.01% | | |
| | 615 | .06% | <.01% | | Grund 1964 |
| SNAPTRAN 3 | 45 | <4% | 0 | 0 | Cordes et al. 1965 |

(a) Not available. All fuel could be accounted for within a 350-ft radius.

effectiveness of natural processes in attenuating the release of fission products during core melt accidents.

Although accidents are not experimental studies, studies of the fission product releases from serious reactor accidents that have occurred provide some of the most realistic information available. The information on serious reactor accidents that have resulted in severe core damage are summarized in Table 4.6. The significant observation from these reports is that only a small fraction of the fission products available appear to reach containment during degraded-core accidents.

### 4.2.2 Aerosols

If a core melt occurs, particles may be generated by the condensation of vaporized fission products and other core materials.

Experimental results (Parker et al. 1967; Baurmash, Johnson and Koontz 1973; Parker, Creek and Sutteen 1979) indicate that the aerodynamic mass median diameter (AMMD) of the oxides of uranium are approximately equal to the 2.5 root of the released aerosol concentration at low vapor concentrations (less than 60 $g/m^3$). In the presence of condensing steam, the agglomerated particles have a chain-like structure which contracts into compact clusters because of the surface tension of the water condensing on them. Malinauskas et al. (1980) concluded that the most probable size of particles released at temperatures from 1800°C to 2700°C was less than 0.5 μm. It appears that low-volatility materials form the larger particles while the higher volatility materials are released as smaller particles.

Morewitz et al. (1979) found the fall-out of $UO_2$ particles at high concentrations (200 $g/m^3$) occurred in two distinct stages. Most of the mass was

## TABLE 4.6. Releases from Damaged Reactors

| Reactor | $MW_t$ | MWd | Release | | | Comments | Reference |
|---|---|---|---|---|---|---|---|
| | | | Noble Gases | Iodine | Other Fission Products | | |
| Windscale-1 (England) | 250 | 4000 | $3.4\times10^5$ Ci in atmosphere | $2\times10^4$ Ci in atmosphere (12%) | 1600 Ci Te<br>600 Ci Cs<br>80 Ci $^{89}Sr$<br>9 Ci $^{90}Sr$<br>in atmosphere | Gas cooled; No containment | Dunster et al. 1958<br>Clarke 1974 |
| SL-1 (Idaho Falls) | 3 | 932 | $10^4$ Ci in atmosphere | 20 Ci in atmosphere | 0.1 Ci Sr<br>0.5 Ci Cs<br>on ground | Water cooled; No containment | Horan and Gammil 1963<br>Islitzer 1962<br>IDO 1962 |
| Three Mile Island-2 (Middleton, PA) | 2720 | $2.42\times10^5$ | $10\times10^6$ Ci in atmosphere | 17 Ci in atmosphere | Not detected in atmosphere | Commercial PWR | Stratton et al. 1969<br>Pickard Lowe and Garrick 1979<br>Pelletier and Thomas 1980 |
| Crystall River-3 (Crystal River, Florida) | 2452 | - | 1000 Ci in containment | 70 Ci in containment water; 2 Ci in air | -- | Commercial PWR | EPRI 1980 |
| Plutonium Recycle Test Reactor (Hanford) | 70 | - | 50% containment | 205 Ci in containment water; 7 Ci in air | -- | Heavy water moderated and cooled; Predefected fuel rod | Perkins et al. 1965 |
| Westinghouse Test Reactor (Waltz Mills, PA) | -- | One Element | <800 Ci in atmosphere | 0 Ci in atmosphere | 10,000 Ci in containment water | One element | Catlin 1960 |
| High Temperature Reactor Experiment | .12 | .001 | -- | 34 Ci in atmosphere (14%) | 0.1 Ci Sr<br>400 Ci gross in atmosphere | Gas cooled; No containment | GE 1959 |
| Oak Ridge Research Reactor | 24 | 1.66 | -- | .15-.2 Ci in atmosphere | ~1000 Ci in primary system | | Sims and Taber 1964 |
| Materials Testing Reactor (Idaho Falls) | 40 | 491 | -- | -- | 15 times normal in primary system | | Dykes et al. 1965 |
| Engineering Test Reactor (Idaho Falls) | 90 | -- | -- | -- | 6.4 Ci in atmosphere<br>42 Ci in leach pond | | Keller 1962 |
| NRU (Chalk River) | 200 | -- | -- | -- | "Large" amount release to water & containment | Natural uranium; heavy water moderated and cooled | Nuclear Safety 1960 |
| NRX (Chalk River) | 30 | -- | -- | -- | $10^4$ Ci in containment water | Natural uranium; Heavy water cooled | Thompson 1964 |

deposited in the initial stage, which lasted 10 seconds. The airborne mass was more persistent in the second stage. The average projected diameter of the particles airborne during the early stage was approximately 40 μm. Nelson and Beyak (1980) found an AMMD of 39 μm for $UO_2$ particles airborne at an average concentration of 65 g/m$^3$.

Following a pressure vessel melt-through, the molten materials (core and structural materials) react aggressively with the concrete basemat. Upon contact, hydrates and carbonates thermally decompose to steam and carbon dioxide. The concrete is rapidly eroded, producing substantial amounts of noncondensible gases (Powers et al. 1978; Powers and Arellano 1982). These gases sparge through the melt, forming particulate material with a typical mean diameter of about 1 μm. Two mechanisms are involved - vaporization and subsequent condensation of volatile species, and mechanical agitation of the molten material.

Two recent articles presented at the Eleventh Water Reactor Safety Meeting provide some additional information on the interaction of the molten core with the basemat. Chu (1983) presented initial data from experiments in a large melt facility (several hundred kilograms). One experiment used a charge of 230 kg, a melt temperature of 2600°C, and gravity flow; extrapolation of the experimental results indicate that 41 kg of particulate material was made airborne. The size distribution was trimodal in the early stages and unimodal (~1 μm) in the later stages.

Tarbell and Brockmann (1983) reported on experiments using high-pressure ejection (15.2 MPa) of the molten core material. The material ejected does not behave as a coherent mass after ejection. The distribution of material made airborne from the interaction of the molten material and concrete basemat was multimodal, with diameter modes at 0.5, 5.0 and greater than 10 μm. The finer particles appeared to be agglomerates of particles from 0.05 to 0.1 μm in diameter, while the large particles appeared to be due to the mechanical breakup of the ejected molten core material.

### 4.2.3 Influence of Natural Processes on Fission Product and Particle Removal

Natural processes occurring in the reactor core, adjacent areas, and in the containment system can have a significant environmental impact that may challenge the ESF systems. Some of these mechanisms are: 1) plate-out on various surfaces; 2) dissolution in water; and 3) agglomeration, gravitational settling, and deposition of particles.

The iodine, cesium and tellurium form CsI, CsOH and $Cs_2Te$ prior to release; these species are not likely to change when exposed to reducing atmospheres (Forsyth et al. 1976; Cubicciotti and Saneki 1978; Lorenz et al. 1980). These compounds are water soluble and can be collected in any water present. Experimental results indicate that cesium plates out on surfaces at 1000 to 1800°C and tellurium at 80°C to 600°C (Roberts 1963). Castleman (1963) reports that 90% of the iodine released in air at a reduced state due to a steam atmosphere will collect on surfaces below 120°C. All other fission products released from the fuel deposit in the region around the fuel. Removal of airborne materials by the condensation of steam is an effective removal mechanism (Hilliard et al. 1961; Cottrell et al. 1963; Parker 1963).

Deposition of particles generated from damaged fuel rods onto adjacent undamaged fuel rods has been reported to decrease the airborne concentration of such particles by a factor of 100 (Parker and Lorenz 1963).

Levenson and Rahn (1981) summarized some of the most significant known characteristics of released fission products and the subsequent removal by natural processes:

- Highly concentrated aerosols coalesce rapidly; low-density aerosols increase their effective density rapidly in the presence of water serving as condensation nuclei.
- Agglomerated particles formed at high mass airborne concentrations are dense and settle out close to their source.
- Iodine in many forms is chemically and physically reactive and is easily immobilized by reaction with organic coatings inside the pressure vessel.
- The containment system and the enclosed equipment provide a large surface area for fission product plate-out and absorption.
- The moisture in the reactor containment building will cause most of the soluble airborne material to dissolve.
- The presence of a large amount of water and vapor along with the heat capacity of the containment building would be sufficient to immobilize a large fraction of the radioactivity even in the event of a massive reactor building failure.

## 4.3 EXPERIMENTAL STUDIES IN PROGRESS

Table 4.7 summarizes current U.S. and European experimental studies on the release and transport of fission product and aerosol under LWR severe accident conditions. The information on the U.S. programs was in many cases obtained by direct contact (by F.A. Zaloudek) with the principal investigator(s) or contractor representative(s). The information on European programs was mainly

TABLE 4.7. Fission Product Release and Transport Experiments

| Laboratory/Sponsor | Experiment Designation | Subject | Experimenter/ Contact |
|---|---|---|---|
| 1. ORNL/NRC | | FP release--discharged fuel pins (20 cm) | Osborne/Lorenz |
| 2. KfK/PNS | SASCHA | FP release--simulated fuel | Albrecht |
| 3. ORNL/NRC | 10 kg arrays | FP release--trace irradiated pins<br>FP transport--simple structures; emphasis on aerosol source term, composition, and release rate | Parker |
| 4. ANL/EPRI | TREAT | FP release--chemical and physical form<br>FP transport--upper plenum | Vogel/Huercig |
| 5. EG&G/NRC | PBF | Series 1--FP release and preliminary measurements<br>Series 2--FP release, transport and deposition under primary system conditions (pins, 1 m) | Hobbins |
| 6. PNL/NRC | NRU | | Panisko |
| 7. KfK/PNS | BETA | Aerosol and FP release during core/ concrete interaction | Hosemann |
| 8. Sandi/NRC | Core/concrete | Aerosol and FP release during core/ concrete interaction | Powers |
| 9. Sandi/NRC | Separate effects | FP/FP and FP/surface interactions | Elrick |
| 10. ORNL/NRC | TRAP-MELT verification test program | Aerosol transport and deposition in simulated/scaled primary system sections | Tony Wright |
| 11. EPRI | -- | Aerosol transport and deposition in scaled sections of reactor primary system | -- |
| 12. Studsvik | Marviken | Large-scale vapor and aerosol transport in upper plenum, piping, pressurizer, and pressurizer relief tank | Collen |
| 13. ORNL/NRC | NSPP | Aerosol behavior in steam atmospheres ($38\text{-m}^3$ vessel) | Adams |
| 14. BF/GRS | DEMONE | Aerosol behavior in steam atmospheres ($640\text{-m}^3$ vessel) | Schock (KfK) |
| 15. BCL/EPRI | | Removal of aerosols from steam/air bubbles in water pools | Cunnane or Kuhlman (BCL) Oehlberg (EPRI) |
| 16. SANDIA | SCAR | Separate effects | J. Walker |

extracted from published information and, therefore, may not be as current. A brief description of the scope and objectives of many of the programs is given in the following subsection.

### 4.3.1 Fission Product Release and Behavior

Fission Product Release from LWR Fuel (ORNL/NRC). The objectives of the study are to measure the fission product release rates from commercial, irradiated LWR fuel under accident conditions, to correlate the release date with physical and chemical changes in the fuel specimen, and to determine the physical and chemical forms of the released material. The temperatures the fuel is currently subjected to are in the range of 1400 to 2000°C with plans to increase to between 2000 and 2400°C when modifications to the apparatus are completed. The carrier gas is 10 to 80% steam in argon and measurements of krypton, iodine, cesium, and strontium deposition are made. Results of some completed tests series have been published (Lorenz et al. 1980a, 1980b, 1982).

Fission Product Chemistry (ORNL/NRC). The objective of this ongoing program is to determine the chemical mechanisms for iodine transport as a function of temperature and pressure (maximum temperature is 300°C and maximum pressure is 2000 psi).

High Temperature Fission Product Chemistry and Transport (SNL/NRC). The program objective is to investigate the chemistry and interactions that might affect the transport of fission products from the fuel into the containment. The scope of the work includes:

- definition of the thermodynamic and chemical reaction characteristics of particular fission products of interest
- examination of the chemistry and transport of fission products in typical steam and hydrogen environments
- comparison of the observed behavior of the fission products with the behavior predicted by thermodynamic methods.

The experimental apparatus will generate steam up to 1000°C with up to 3 fission products of interest. Measurements include steam flow rate, temperature, pressure, rates of chemical species production, and steam condensation rates. Post-test analysis of coupons and liquid samples are to be performed.

Feasibility of Studying Volatile Fission Products Released from Irradiated LWR Fuel Under Simulated Accident Conditions (ANL/EPRI). The objective of the program is to characterize the volatile fission products present in the gap of an irradiated fuel pin. The experiments are on a laboratory scale with maximum temperatures up to 900°C (temperatures up to 3000°C may be possible with some modifications). The rate of fission product release is measured as a function of temperature and chemical species.

NRU/PNL Radionuclide Release/Transport Study (PNL/NRC). The experiments are to be performed in a test loop installed in the National Research Universal (NRU) reactor of Canada. The objectives of the study are: 1) determine the nature (chemical and physical) of radionuclides released from full-length fuel rods during overheating, 2) determine radionuclide transport through the core and upper plenum structure, and 3) determine the affect of rod length on hydrogen generation, fuel slumping/flow blockage cooling, fuel refreezing, and fission product release. Proposed tests involve pre-irradiated fuel pins melted to supply the radionuclides, and the airborne material carried by gas to test surfaces maintained at 300, 350, 500, and 1000°C.

Fission Product Release Program at the Sascha Facility. The objectives of this program, performed by Kfk-PNS Institut für Radiochemie, is to determine the source term for release of activity, decay heat, and aerosol from irradiated fuel failure of the reactor pressure vessel. From 150 to 250 g of simulated irradiated fuel and stainless steel are melted in a 2-bar steam and water atmosphere. The maximum fuel temperatures range from 1800 to 2800°C. The parameters measured for each test include: initial activity; activity on filters as a function of time; total activity released; fractional activity released to the filters; and relative release as function of time.

PITEAS Research Program (Ross 1982). The objectives of this program recently initiated by the Commissariat a l'Energie Atomique (CEA) are to:

- modify and validate aerosol codes, initially developed for LMFBR, to deal with CsI, CsOH, and particles at high concentrations under PWR accident conditions
- study the chemical behavior of CsI under various thermal and environmental conditions, including intense radiation fields
- investigate the efficiency of conventional filters and sandbed filters for iodine and cesium aerosols
- measure the iodine partition coefficients between air and water, air and walls, and in chemical environments representative of PWR coremelt accidents
- study the loss of iodine removal efficiency of conventional charcoal filters in $CO_2$-steam atmospheres.

LWR Source Term Measurements in TREAT (ANL/EPRI). This program was in the early planning stages at the time this document was being prepared. A proposal was to be submitted. As presently planned, these experiments will include in-reactor testing of fuel under simulated accident conditions to study fission product release under a typical chemical and physical environment.

Power Burst Facility; Severe Fuel Damage Tests (EG&G Idaho/NRC). The program is being performed in two phases. The objectives of phase 1 are: 1) to determine the coolability of severely damaged fuel assemblies, to characterize fuel damage in terms of $UO_2$ dissolution, redistribution and fragmentation; and 2) to measure the magnitude and timing of the hydrogen generation and the release of fission products. Phase 2 is concerned with fission product behavior, debris coolability, melt dynamics, hydrogen generation, decay heat and parametric effects.

### 4.3.2 Aerosols

Core Melt Aerosol and Fission Product Release (ORNL/NRC). This ongoing program consists of three parts: 1) basic aerosol experiments in the Containment Release Installation (CRI) II; 2) aerosol release tests using up to 1 kg of fuel pins induction heated in a split crucible in steam-hydrogen atmospheres; and 3) aerosol tests in the CRI II using airborne materials from up to 10 kg of fuel bundle eutectic melt, induction heated in a split crucuble. The activities in the basic aerosol experiments include pre-component aerosol characterizations of fuel structural and control rod materials including $U_3O_8$, $Fe_2O_3$, $SnO_2$, and metallic Cd, Ag, and Sn. Iodine adsorption on the particles will also be measured. Fuel sample temperatures up to 2600°C are possible. The information generated during the tests will include: fuel and structural vaporization rates; release rates of fission products (I, Cs, Te, Sr, and Ru); physical and chemical characteristics of fission products released; aerosol behavior in containment; and characterization of the meltdown phase (alloys, eutectics, metallics, and fission product partitioning and scaling between large and small core melts.

Trap-Melt Verification Program (ORNL/NRC). The program objectives are to verify the aerosol transport and deposition models used in BCL Trap-Melt code. The experimental program includes aerosol and fission product transport and particle resuspension tests. The aerosol transport tests will investigate primary vessel deposition and transport for simulated core-melt accident conditions by varying the particle residence time, aerosol generation rate, and pipe wall temperature. The fission product transport tests will investigate the primary system transport behavior of volatile fission product species and core material produced under simulated core-melt accident conditions by varying the fission product species, wall temperature, and flow through the test section. The particle resuspension tests will investigate the resuspension of material deposited on the reactor system surfaces under core-melt conditions. Resuspension will be measured as a function of flow velocity, deposit thickness, system humidity, and surface deposition orientation.

Molten Core Containment Program (SNL/NRC). The program is an experimental and analytical effort to identify and quantify the safety-related processes that could occur during the interaction between molten core debris and reactor containment structures. The experimental program involves the interaction of prototypical core materials at realistic temperatures with structures representative of those found in existing and planned LWRs. The objective is to identify more suitable materials for molten core retention than concrete (e.g., magnesia, firebrick, high alumina cement, borax). The experiments are performed in Sandia's "Large Melt Facility." This facility produces super-heated oxides (70% depleted $U_3O_8$ + 30% lanthanum sesquioxide) melts of 100 to 1000 kg at temperatures up to 2800°C.

Nuclear Safety Pilot Plant (NSPP): Aerosol Release and Transport Program (ORNL/NRC). The program objective is to investigate the behavior of LWR accident-generated aerosols in a contained, condensing steam environment. The particles under consideration include $U_3O_8$, cladding, structural materials, $Fe_3O_8$, concrete and control rods. Containment conditions are temperatures up to 150°C and pressures to 60 psi. Current areas of study include the effects of moisture on: aerosols from molten fuel/cladding/control rods; aerosols from molten core structural material; aerosols from molten core/concrete interactions; and co-agglomeration of mixtures of some or all of the preceding.

Beta Project: Melt/Concrete Interaction. This program is being performed by Kfk-PNS Institut für Radiochemie. The program objectives are to study heat transfer and chemical reactions in the metallic and oxide phases at 1700°C, to study the behavior of the melt and define the refreezing behavior, and to study the characteristics of the solidified material. Into crucibles made of siliceous and calcareous concrete, the experiments introduce melts containing one of the following materials:

- 300 kg steel
- 300 kg steel + 300 kg $Al_2O_3$ and $SiO_2$
- 300 kg $Al_2O_3$ + 100 kg steel
- 300 kg $Al_2O_3$ and $SiO_2$ + 100 kg Fe

The temperature distribution, humidity distribution, melt penetration velocity, and the depth of penetration of the melt into the concrete are measured. Melt temperatures between 1500 and 2000°C are under consideration.

Marviken Full-Scale Aerosol Transport Test (ATT) (MARV 1981, 1982). The ATT program is sponsored by a consortium led by Studvik Energetecknik AB, Sweden. Participants in the U.S. include NRC and Electric Power Research Institute (EPRI). The primary objective is to determine the transport of relatively dense (high mass concentration) aerosols through a full-scale PWR system. The transport of volatile fission products will also be investigated. Approximately 800 kg of a simulated "corium" (a mixture of uranium and zirconium oxides with iron) will be heated to 2500°C in a tank representing a LWR pressure vessel. The aerosol will be blown by a steam-air mixture through pipes into a dry and a wet compartment. Samples will be taken at intervals to characterize the material airborne and deposited and liquids. Measurements will be taken of temperatures, pressures, steam quality, and vapor pressure. Fission product simulants such as CsI, CsTe, Sr and Fe will be vaporized using approximately 100 kg at temperatues from 300 to 1500°C. Modifications are planned to simulate BWR configurations.

## 4.4 SUMMARY OF CONTAINMENT ATMOSPHERES FROM CORE MELT ACCIDENTS

Containment temperature can reach 500°F for short periods of time in PWRs with long-term temperatures up to 300°F. If hydrogen burning occurs, temperature spikes up to 3000°F may occur. The wetwell temperature in BWRs rarely exceeds 250°F, but drywell temperatures may reach 800°F for extended periods. Drywell temperatures may reach 1000°F in the event of a hydrogen burn in that region.

Containment peak pressure is limited by the failure pressure of the containment. The assumed failure pressure varies, depending on what type of failure mode is postulated. A higher pressure is required to rupture the containment building than to cause leakage around electrical penetrations. The ultimate yield strength of the containment can be used as an upper bound for the various containment failures. A rule of thumb is to assume that the ultimate yield strength is twice the design pressure. Based upon a maximum design pressure of 60 psia, most containments will fail at 120 to 150 psia and most certainly by 200 psia. Many of the accident calculations surveyed in the literature showed pressures in the 50- to 60-psia range in both the short and long term. Another group of calculated accident pressures clustered around the 100- to 110-psia range, which approaches the assumed failure pressure for most containments.

Hydrogen mole fractions of up to 0.2 have been calculated. The range of values for the steam mole fraction extends to 0.85, and noncondensible gases (including the oxygen) have ranged from 0.2 to 1.0.

The quantity of fission products released during fuel deterioration ("gap-release," diffusion from the pellet-to-gap, and difussion from the $UO_2$ grains) is reasonably well defined. Almost all of the noble gases can be released during these phases with greater than 20% of the iodine and cesium released during the diffusion from the $UO_2$ grain. Because of the uncertainities in composition, release of fission products from molten fuel is less well defined. The best data to date indicate that the most probable form of iodine is in the form of CsI with lesser quantities of other iodine forms.

Much of the fission products released from the fuel is rapidly removed from the air by adjacent surfaces. An average of 28% of the iodine and 67% of the other fission products used in the CSE experiments did not leave the generation apparatus. Much of the cesium and iodine is removed from the atmosphere by natural processes and is found in the organic coatings (paint) and condensate.

The evidence from both reactor destruction tests and accident summaries indicates that very little of the released fission products escapes to the ambient atmosphere.

Large quantities of particulate material can be generated during two phases of the severe accident scenario -- during melting and slumping of the core within the reactor pressure vessel, and after the core melts through the pressure vessel and contacts the basemat. The sequence during core melting and slumping will be of continuing uncovery and heating, resulting in cladding rupture, core slumping, and eventually melting through the bottom of the pressure vessel. During this period, heating is not uniform throughout the core (either radially or longitudinally), so various phases of melting may be present simultaneously in various portions of the core. Fission product release during this sequence is generally accepted to be "gap-release" (release of the fission products accumulated in the free volume of the fuel elements), diffusional release of the pellet-to-gap inventory, diffusion from the $UO_2$ grains, and release from the molten material. The large masses of particulate material are made airborne during the final phase -- release from the molten material.

The second period during which large quantities of particulate material can be generated is when the molten core contacts the basemat. It has not been determined whether the release of the molten material from the pressure vessel is pressurized or under gravity flow. The type of release will affect the quantity and characteristics of the particulate material generated (Tarbell and Brockman 1983; Chu 1983). In either case, contact between the molten core and concrete will generate large quantities of gas and particles. Much of the material will be nonradioactive, but some fission products are released at this time. Calculations of the total quantity of material range as high as 1300 to 2000 kg. The most probable particle size during the core deterioration within the pressure vessel is initially 0.5 $\mu m$. Released directly into the containment, a large fraction falls out within 10 sec. The projected diameter of this material is ~40 $\mu m$.

Under gravity flow conditions, approximately 18% of the mass of corium is made airborne in its interaction with the concrete basemat. The size distribution of the airborne material is trimodal in the early stages and unimodal at about

1 μm in the later stages. Under pressurized flow, the three modes of the size distribution of the material airborne are at 0.5, 5.0 and greater then 10 μm.

The presence of water (e.g. suppression pool, ice condenser, condensate, residual water in the primary coolant piping, water resevior, etc.) can play an important role in the quantity of both fission products and particulate material released to containment. Other natural processes will also have a significant effect under various conditions.

## 5.0 EXPERIMENTAL EVALUATION OF ESF SYSTEMS

This section of the report summarizes information concerning experimental evaluations of the ESF systems. Evaluations related to fission product retention are not available for all the ESF systems under conditions simulating core-melt accident conditions. Some data from completed programs were found for four of the six systems -- Pressure Suppression Pools, Containment Sprays (and Pumping Systems associated with the Recirculation System), Ice Condenser Systems, and Filter Systems. One ongoing project concerned the retention effectiveness of pressure suppression pools.

### 5.1 PRESSURE SUPPRESSION POOLS

BWR containment systems are arranged so that for most postulated accident sequences, the steam and other airborne materials released from the reactor pressure vessel will be vented after passing through a pool filled with water. The steam will be condensed and some of the other airborne materials scrubbed from the gases by passage through the water.

Representative suppression pool decontamination factors (DFs) could not be estimated from a recent review of the technical literature by General Electric (Rastler 1981). The data generated by the review is summarized in Table 5.1. The lower bound decontamination factors estimated by GE are shown in Table 5.2. Review of the information indicates that most of the experiments considered $I_2$ with some information on the scrubbing of other iodine forms ($CH_3I$, HI, HIO, and small insoluble particles). The test conditions represented do not reflect those anticipated for degraded-core/core-melt conditions. The realism of some of the other test conditions is adversely affected by excessive flow rate, shallow pool depths, lack of tests at elevated pool temperatures, or nonrepresentative particle size distributions. Although particulate material was used in one test, no data were generated on the scrubbing of soluble particles such as CsI or CsOH.

The authors felt that the lower bound DFs could be increased by several orders of magnitude if additional experiments could be performed under conditions more representative of a degraded-core incident. Based upon data on the scrubbing of volatile $I_2$ and 0.06-μm NiCr particles summarized by the experiments in references in Table 5.1, it was concluded that a DF of at least 100 or greater could be expected from a subcooled pool during the core/concrete interaction. Single bubble DFs of 100 to 4200 were measured for 0.05- to 10-μm particles (activated $Eu_2O_3$) from air and were found to depend upon bubble residence time and particle size (Marble et al. 1982).

TABLE 5.1. Summary of Pool Scrubbing Tests

| Experimenter/ Date | Pool Size (gal) | Depth Injection (ft) | pH | Temp (°C) | Carrier | Flow Rate | Simulated f.p. | DF | Comments |
|---|---|---|---|---|---|---|---|---|---|
| 1 GE (1959) | 1000 | 1.5 | 7 | Subcooled | Flashing Water and Air | Rapid Blowdown | Xe, Kr<br>$I_2$<br>NaI (Soluble)<br>$ZnS_2$ (Insoluble) (2 μm) | 2-4<br>$10^5$-$10^6$<br>$5x10^5$<br>$5x10^7$ | Facility simulated LOCA Blowdown to Pressure Suppression System |
| 2 Hillary et al. (1966) | 2000 | 2 | 6.8 | 10-60 | Steam, Air, and Steam, Air Mixtures | Steady<br>Steam: 3-9 lb/sec<br>Air: .1-7.0 lb/sec | $I_2$<br>Ni-Cr (0.06 μm) | 14-320<br>15-1680 | Scale Model of a SGHW Reactor |
| 3 Diffey et al. (1965a) | 3 | .17 | - | 50 | Steam/Air | Steam 4 lb/sec-$ft^2$ | $I_2$ (0.6-40 ppm $I_2$)<br>$I_2$ (0.01-0.4 ppm $I_2$) | 10-500<br>10-500 | Includes both large and small scale tests |
| | 150 | 1.7 | - | 50 | Mixtures | Air 3 lb/sec-$ft^2$ | HI<br>$CH_3I$<br>Ni-Cr (0.06 μm) | 10-1000<br>1.5-5<br>50-100 | |
| 4 Dadillion and Geisse, France (1967) | 11,000 | 6-20 | - | Subcooled | $CO_2$ (400°C and 280 psi) | .02-04 lb/sec | $I_2$ | 70-$10^4$ | PIREE Experiment |
| 5 Stanford and Webster ORNL (1972) | 260 | 4 | - | Subcooled | Steam/Air Mixture (Air 0-2 wt %) (125 psig sat. steam) | 22-66 lb/sec-$ft^2$<br>Steady | $I_2$ (0.5-10 ppm) | 70-11000 | Simulated pressure suppression pool |
| 6 Siegwarth and Siegler GE (1971) | 150 | 1-4 | 7-10 | 32-66 | Flashing Water and Air | Rapid Blowdown | $CH_3I$ | 1.1-3.2 | Simulated LOCA in 1/10,000 scale model of BWR MK-I |
| 7 Marviken Sweden (1974) | $10^5$ | 9 | - | 20-40 | Flashing Water and Air | Rapid Blowdown | $CH_3I$ | 1.2-4.9 | Full-scale reactor |
| 8 McGoff and Rodgers MSA (1957) | 300 | 10-20 | - | Subcooled | Flashing Water and Air | Rapid Blowdown | $I_2$<br>$Y_2O_3$<br>Rb Cl | $2x10^5$<br>$3x10^5$<br>$10^7$ | Study of venting in seawater |
| 9 Malinowski et al. W (1971) | 30 | 3-8 | 4-5 | 49 | He | 0.05 lb sec-$ft^2$ | $I_2$(20 mg/liter) | 88-1500 | Glass column 9 in. OD and 8 ft high |
| 10 Hilliard HEDL (1981) | - | 2 | - | Subcooled | $N_2$ | 0.5 $ft^3$/min | $Na_2O$ (4 μm) | 20 | Scrubbing of particles by water |
| 11 Devell et al. (1967) | 530 | 3-12 | 0-12 | Saturated | Superheated Steady (175-300°C) | Steady 0.1-0.4 kg/sec<br>90-360 lb/sec-$ft^2$ | $I_2$(0.1-2 ppm) | 2-200 | Saturated Pool Testing included both laboratory and large-scale tests |
| 12 Strikovitch et al. (1964) | - | - | 5.5-10 | Saturated 120-180 | Steam | Steady<br>0.2-1.0 kg/hr | $I_2$ (2.5-250 ppm) | 10-250 | Small scale laboratory test |
| 13 Marble et al. (1982) | 45 | 34-167 cm | - | 60 | $N_2$, Air 20°C | Single bubbles<br>0.4-1.4 cm | $Eu_2O_3$ (0.05-10 μm) | 100-4000 | Single bubble tests |

TABLE 5.2. Minimum Suppression Pool Decontamination Factors

| $I_2$ | Subcooled Pool | Saturated Pool |
|---|---|---|
| $I_2$ | $10^2$ | 30 |
| CsI | $10^3$ | $10^2$ |
| Particulates | $10^2$ | $10^2$ |

These data were used as a basis for a computer model of the particle scrubbing behavior of a BWR 6/Mark III containment system. The model predicted DFs of 9 X $10^3$ and 6 X $10^{12}$ for discharges of steel/corium and corium/concrete, respectively, through horizontal vents. For release through the X-quencher, the DF predicted for a steel/corium discharge was 4 X $10^4$.

Removal of Radionuclides by Water Pools Under Severe Accident Conditions (BCL/EPRI). This ongoing program includes scrubbing experiments to allow measurement of the decontamination factor under accident conditions and hydrodynamic experiments to determine the type of gas/water interface, bubble rise velocity, bubble size distribution, and residence time. Three types of tests are planned:

- Single-orifice tests -- 0.391-inch ID BWR X and T quencher and PWR 0.75-inch ID quench tank nozzles.
- Multiple-orifice tests.
- Large-scale injections simulating downcomers and horizontal vents up to 6-inch ID.

Model aerosols will be chosen to simulate soluble CsI and insoluble Te.

## 5.2 CONTAINMENT SPRAYS AND RECIRCULATION PUMPS

The Containment Spray System (CSS) is part of the Residual Heat Removal System (RHRS) and provides containment cooling following a LOCA. The CSS can also provide a fission product and particle removal function. The CSS is supplied with water from the Pressure Suppression Pool, Containment Sump, or Refueling Water Storage Tank, depending upon reactor type and conditions. The ability of the spray system pumps to continue functioning following an incident affects the CSS and is discussed here.

The Containment Spray Experiments (CSE) were designed to test models of iodine washout, but significant removal of cesium and $UO_2$ particles was also demonstrated (Hilliard and Postma 1980, 1981). The experiments are described in

Section 4.2.1. The initial application of spray provided the greatest reduction in iodine concentration, and subsequent applications gave progressively slower washout rates as the concentration in the vessel was reduced. Typical spray attenuation factors of 0.03 were observed for the first 2-hour period and 0.014 after 24 hours. It was concluded that the spray was not as effective as a high-efficiency filtration system for the removal of airborne iodine but greatly exceeded the performance of natural removal processes during the first two hours following the release. Over a 24-hour period, the removal rates of natural processes compared more favorably with the engineered safety features.

The RHRS and CSS are supplied by pumps which take water from either the Pressure Suppression Pool or Containment Sump. Because of the debris that would be generated by the incident and carried to these water supplies, some concern has been expressed about the ability of these pumps to continue to function. The debris could plug the inlet screens or cause accelerated wear of the pump seal.

Creare, Inc. (Kamath et al. 1982) investigated the performance of the RHRS and CSS pumps under debris and air intake conditions for a group of power reactors (Arkansas #2, Calvert Cliffs #1 and #2, Crystal River #3, Kewanee, McGuire #1 and #2, Midland #2, Millstone #2, Oconee #3, Prarie Island #1 and #2, and Salem #1. The analysis was based upon: 1) experimental data on pump performance and net positive suction head and 2) data found in the literature regarding the performance of pumps subject to debris and air ingestion. Several types of debris were considered, and conservative estimates were made of the quantities of each that would pass through the pumps. The debris considered were:

- insulation fibers (0.3% volume concentration)
- aluminum and zinc hydroxide precipitates (0.04% volume concentration)
- paint flakes (0.025% volume concentration)
- concrete dust (0.05% volume concentration).

The total concentration of debris reaching the pumps was estimated to be less than 0.5%. The investigators concluded that the hydraulic performance degradation of the RHRS and CSS pumps would be negligible and that mechanical wear would be too small to seriously impair long-term pump operation. If the shaft seal were to fail, leakage would be less than 0.1% of the pump flow rate.

Burns and Roe, Inc. (Wysocki and Kolbe 1982) investigated the possibility of debris blocking the inlet pump screens enough to degrade pump performance. The reactors examined included Salem #1, Arkansas #2, Main Yankee, Sequoyah #2, and Prairie Island #1. The debris was assumed to be principally insulation generated by pipe whip, pipe impact, and jet impingement. The estimated amount of debris was found to depend on the plant and on the type of break; estimates ranged from 100 to 7200 $ft^3$. The results of the analysis were not sufficiently conclusive to allow generalization about whether screen blockages could cause pump malfunctions. Depending on the reactor and type of break, screen blockage ranged from 0 to 100%. It was concluded that the question of screen blockage must be addressed on a reactor-by-reactor and accident-by-accident basis.

None of the above studies addressed the problems caused by core/concrete interactions following a core-melt event.

## 5.3 ICE CONDENSER SYSTEMS

Malinkowski (1968) performed an experimental study to determine the effect of vapor concentration, ice additives, ice loading, vapor temperature, and the characteristics of iodine concentration on the removal of elemental iodine from the vapor stream entering an ice condenser. Two sizes of apparatus were used: 1) a 1.5-inch diameter by 18-inch glass tube, and 2) a 9-inch ID X 4-ft tube. The reported findings were:

- Alkaline additives enhance the retention of iodine in the ice melt by hydrolysis reactions which convert the iodine to nonvolatile soluble forms (iodide and iodate).
- Greater than 95% removal of iodine was achieved with sodium tetraborate ($Na_2B_4O_7$) in the ice.
- The physical form of ice did not appear to have a strong influence on iodine removal.
- The effect of iodine concentration was small.
- Iodine removal was observed to be a strong function of air in the steam/air mixture.
- The ice condenser was not effective in the removal of methyl iodide.

None of the experiments addressed the retention of other gaseous or particulate materials; nor were the effects of high mass concentrations considered.

## 5.4 FILTER SYSTEMS

A series of tests were performed in the CSE facility to demonstrate the effectiveness of ESF filters (Postma and Johnson 1971; McCormack, Hilliard and

Postma 1971). Five tests were performed with an internal recirculating filter-adsorber loop designed for a flow rate of 100 cfm; the loop consisted of a pre-filter, demister, high-efficiency particulate air filter, and a charcoal bed in series. The filter system was found to remove iodine very effectively compared to the water spray and natural removal mechanisms, especially during the first half hour of the test, because of the filter system's superior capability to remove methyl iodide. Even after 24 hours in a steam-air environment, the filter units were observed to remove methyl iodide at nearly the initial rate for a new unit. Typical attenuation factors for the filter loop was 0.13 for the initial 2-hour period and 0.013 for a 24-hour period.

## 5.5 SUMMARY

Except for pressure suppression pools, experimental programs appear to be needed to provide the data base for the validation of models proposed for evaluating ESF system effectiveness. The effectiveness of two of the ESF systems of interest (MSIVLC and Chiller/Coolers) has not been evaluated. Tests have been performed on four (Pressure Suppression Pool, Containment Sprays, Ice Condensers, and Filter Systems), but the range of conditions used in these evaluations are not adequate to predict the effectiveness of fission product retention under core-melt accident conditions. The ongoing BCL/EPRI experimental study of the effectiveness of suppression pools appears to be adequate.

## 6.0 MODELS FOR PREDICTING ESF PERFORMANCE

This report section briefly reviews current models for predicting the retention of airborne fission products. The models described herein are based on the assumption that the ESF is functional. The degree to which an ESF is functional depends on the assumed accident sequence and the ability of the system to survive in the accident environment.

### 6.1 MODELING APPROACHES

Models for predicting ESF performance have been tailored for two different end uses: 1) the licensing of power plants, and 2) probabilistic risk assessments and assessments of severe accident consequences. For the latter use, models that will provide best-estimate predictions are usually provided as part of the assessment.

#### 6.1.1 Modeling Related to the Licensing of Power Plants

Federal law 10 CFR 100 requires that nuclear reactors be housed in containment buildings that will limit doses to the public, even in the unlikely, postulated accident in which a significant fraction of the core inventory of fission products is released into the atmosphere of the containment building. The calculated leakage of radioactive material from the plant should account for retention by ESF systems, and as noted by Culver (1966), ESF systems were brought into siting calculations in the mid-1960s. Regulatory Guides (1.3 Rev. 2 and 1.4 Rev. 2) were published to provide guidance to applicants on acceptable means for estimating site suitability source terms. These guides allude to ESF systems but state that credit should be given on a plant-specific basis.

The regulatory guides credited only certain of the ESF systems discussed here with a fission product retention function. The systems credited included sprays in the containment buildings of PWRs and filtration systems for various applications in both BWRs and PWRs. Suppression pools and ice beds were not specifically credited, apparently because no licensing applications specifically claimed scrubbing credit for these ESF systems. A recent exception is the General Electric Standard Safety Analysis Report, which included materials on scrubbing by pressure suppression pools (GE 1981, Appendix 15D).

Regulations outlined in 10 CFR 100 required conservative analyses to be made to assure that dose guidelines would not be exceeded by any accident considered to be credible.

In addition to the formulation of conservative models, the physicochemical form of the fission product source term to the containment atmosphere was specified.

Iodine, the most biologically significant radionuclide, and the one identified in 10 CFR 100, was assumed to be present in containment in three forms: 91% $I_2$, 4% $CH_3I$, and 5% particulate. It was assumed that 50% of the core inventory of iodine would be released to the containment atmosphere as a puff release and that half of that (25% of core inventory) would be instantly deposited irreversibly onto surfaces. Models for predicting fission product retention in site suitability licensing analyses have focused on these three iodine forms. Noble gases were assumed to be unaffected by ESF operation, and other fission products, assumed to be present as aerosols, were not specifically accounted for.

### 6.1.2 ESF Models for Probabilistic Risk Assessments (PRAs) or Assessment of Severe Accident Consequences

Evaluations of severe accident consequences or of risk (probability considered in addition to consequences) are most meaningful if fission product transport is analyzed by means of best-estimate models. Such models tend to be mechanistic and consider all relevant physical and chemical processes. Thus such models are often significantly more complex than the conservative models used for site suitability analyses. However, in the first major PRA (WASH-1400), ESF system performance was evaluated with semi-empirical models (Ritzman et al. 1974):

- For spray washout of aerosols, washout was modeled in terms of a single drop collection efficiency, $\epsilon$, whose value was obtained from a correlation of CSE (Postma and Johnson 1971) test results.

- Spray absorption of elemental iodine was modeled with commonly accepted drop uptake models based on the stagnant film theory for liquid phase mass transfer resistance.

- Suppression pool scrubbing was accounted for in a simplistic way: a decontamination factor (DF) of 100 or 1 was applied, depending on whether the pool was sub-cooled or saturated.

Little consideration was given in WASH-1400 to questions of ESF availability under the environmental conditions imposed by the accident. For example, the plugging of filters by aerosols was not specifically accounted for. In more recent evaluations of severe accident consequences, ESF performance has been treated more mechanistically in the hope that truly realistic assessments can be achieved.

## 6.2 MODELS FOR DEPLETION BY CONTAINMENT SPRAYS

In this report section, published models for spray washout will be described in enough detail so that the reader may grasp the technical basis and the governing equations employed in each model.

### 6.2.1 WASH-1400

The CORRAL Code used in WASH-1400 (Ritzman et al. 1975; Owczarski, Postma and Lessor 1974) assessed aerosol washout by considering the spray to be an assemblage of noninteracting drops, each of which exhibited a collection efficiency for suspended particles in a swept volume. The fraction of aerosol particles removed per unit time was expressed as

$$\lambda = \frac{3hF\varepsilon}{2dV} \tag{6.1}$$

where $\lambda$ = fraction of aerosol removed per second
$h$ = spray fall height, m
$F$ = spray flow rate, $m^3/s$
$\varepsilon$ = drop collection efficiency, dimensionless
$v$ = volume of contained gases, $m^3$
$\alpha$ = drop size, m

The terms h, F, d, and v are all parameters of the containment spray system. The collection efficiency, $\varepsilon$, is known to depend strongly on particle size. The value of $\varepsilon$ was obtained from a correlation developed from CSE results (Hilliard et al. 1971), where $\varepsilon$ was found to be relatable to a dimensionless spraying time, Ft/V, where t is the time that sprays operate. The following relationships were used:

| Dimensionless Spraying Time (Ft/V) | Collection Efficiency ($\varepsilon$) |
|---|---|
| 0 - 0.002 | $15.85(Ft/V) + 0.055$ |
| 0.002 - 0.0193 | $0.04125 - (0.08626 + 42.68(Ft/V))^{1/2}/21.34$ |
| 0.00193 and greater | 0.0015 |

These formulations cause $\varepsilon$ to vary from 0.55 to 0.0015 as spray time increases from zero.

For elemental iodine, the removal rate constant was related to spray parameters using

$$\lambda = \frac{FHE}{V} \tag{6.2}$$

where $\lambda$ = fraction of $I_2$ removed per second
$F$ = spray flow rate, $m^3/s$
$H$ = equilibrium partition coefficient for $I_2$ in the drop
$E$ = drop approach to equilibrium
$V$ = volume of contained gases, $m^3$.

The value of E was estimated from a drop exposure model that considered mass transfer-resistance in both the gas and liquid phases (Postma and Pasedag 1974):

$$E = 1 - \exp = \frac{6k_g t_e}{d(H + \frac{k_g}{k_\ell})} \tag{6.3}$$

where

$$k_g = \frac{D_v}{d}(2 + 0.6\ Re^{0.5} Sc^{0.33})$$

$$k_\ell = \frac{2\pi^2 D_\ell}{3d} = \text{liquid phase mass transfer coefficient, m/s}$$

$t_e$ = drop exposure time, s
$D_v$ = $I_2$ diffusivity, $m^2/s$
$d$ = drop diameter, m
$v, \ell$ = subscripts referring to vapor phase and liquid phase, respectively
$Re$ = Reynolds number for falling drop
$Sc$ = Schmidt number for $I_2$ in steam/air gas phases.

The partition coefficient, H, applicable to spray washout, was assigned values that were consistent with CSE spray measurements:

for caustic, pH = 9.5, H = 5000
for boric acid, pH = 5, H = 200
for basic sodium thiosulfate, H = 100,000.

### 6.2.2 SPIRT Code

The SPIRT code (Postma and Pasedag 1974; Postma, Sherry, and Tam 1978) has been used in crediting spray systems in licensing evaluation of site suitability;

its formulation was specifically tailored to yield conservative predictions of the spray removal lambda.

A first parameter computed in SPIRT is the drop size, as influenced by drop-to-drop coalescence. For the calculation, the containment is divided into a number of height increments. The number of coalescences between drops of two different sizes in a height increment dz is given by:

$$n_{ij} = \pi E_{ij} P_i P_j (a_i + a_j)^2 (1 - \frac{U_j}{U_i}) dz \tag{6.4}$$

where $n_{ij}$ = number of coalescences between drop size groups i and j
$E_{ij}$ = collection efficiency for drop sizes i and j
P = drop population per unit volume of gas
a = drop radius
v = fall velocity.

The drop size used for calculating scrubbing efficiency is the size distribution found for the lowest height, the point where the drop size is maximum. Use of this maximum drop size yields conservative results because scrubbing efficiency decreases with increasing drop size.

Elemental iodine washout is computed from the stagnant film model described in Equations (6.2) and (6.3).

Organic iodide absorption was accounted for in SPIRT by models that accounted for chemical reaction within the liquid phase. Reactions within the liquid phase are important for sprays which use sodium thiosulfate additive. For drops, two model options were available to the code user. In the first, the drop was considered to be a rigid sphere; mass transfer resistance in both fluid phases was accounted for. The amount of solute gas absorbed by a single drop as it fell through the containment atmosphere (Postma et al. 1975) is calculated from:

$$Q = 8\pi h^2 C^* D a^2 \sum_{n=1}^{\infty} \frac{kt(k + D\alpha_n^2) - D\alpha_n^2 \,(\exp(-t\, k + D\alpha_n^2) - 1)}{(k + D\alpha_n^2)^2 [a\alpha_n^2 + h(\ell a h - 1)]} \tag{6.5}$$

where: Q = mass of solute absorbed
h = $k_g/HD$
$k_g$ = gas phase mass transfer coefficient
H = equilibrium partition coefficient
C* = surface concentration in liquid phase
D = diffusivity of solute in liquid
a = drop radius
k = first order reaction rate constant
$\alpha_n$ = nth root of $(a\,\alpha)$ cot$(a\,\alpha)$ tah-1 = 0.

This equation may be expected to yield lower-limit estimates of drop absorption because convective mixing within the drop has been neglected.

A second drop absorption model option is one that yields an upper-limit estimate. The governing equation was obtained from a simplification of Equation (6.5) for the limiting case where h and $D/a^2$ become indefinately large. The result is:

$$Q_{max} = \frac{4}{3} \pi a^3 (kt + 1)C^* \tag{6.6}$$

where $Q_{max}$ is the quantity of solute absorbed by a perfectly mixed drop with negligible gas phase mass transfer resistance (Postma et al. 1975).

CSE test results (Postma and Hilliard 1969) illustrated that absorption by wall films was significant compared to drops for organic iodides when reactive sprays were used. Therefore SPIRT included an equation for wall film absorption:

$$q = C^* \sqrt{kD} \ \mathrm{tah}\text{-}1 \left( \frac{\sqrt{k}}{D} \ \delta \right) \tag{6.7}$$

where q = absorption rate per unit area
s = film thickness.

Wall film thickness was estimated using the assumption of a laminar film on a vertical wall:

$$\delta = \left(\frac{3 \nu \Gamma}{g}\right)^{1/3} \tag{6.8}$$

where $\nu$ = kinematic viscosity of liquid
$\Gamma$ = film flow rate per unit length of perimeter
g = acceleration due to gravity.

The use of equations (6.7) and (6.8) for estimating the absorption of methyl iodide by aqueous wall films containing sodium thiosulfate and hydrazine is supported by experiments reported by Postma and Hilliard (Postma and Hilliard 1969) and by Postma (1970).

The washout of aerosol particles is not modeled in SPIRT.

### 6.2.3 MATADOR

The MATADOR code (Baybutt, Raghuram and Ava 1982) treats the washout of elemental iodine in the same way that CORRAL does, i.e., by the use of Equations (6.2) and (6.3).

For aerosol particle washout, Equation (6.1) is used to relate the washout rate to spray parameters and to the single drop collection efficiency. The collection due to two mechanisms, impaction and interception, is accounted for. For impaction, the following formula is used:

$$\varepsilon_{imp} = \left[1 + \frac{0.75\ln(2Stk)}{Stk - 1.214}\right]^{-2} \tag{6.9}$$

The term Stk is the Stokes number, defined by:

$$Stk = \frac{2(V_d - V_{pi})r_i^2 \rho_{pi}}{9\mu R} \tag{6.10}$$

where $V_d$ = settling velocity of drop
$V_{pi}$ = settling velocity of particle
$r_i$ = radius of particle in i-th group
$\rho_{pi}$ = density of particles
$\mu$ = gas viscosity
R = radius of drop.

Interaction efficiency is computed from an equation that applies for viscous flow around spheres:

$$\varepsilon_{int} = \frac{3}{2}\,\frac{(r_i/R)^2}{(1 + r_i/R)^{1/3}} \tag{6.11}$$

As is evident from Equations (6.9) and (6.11) the capture efficiency is computed for each particle size group. Particle size is computed mechanistically, accounting for agglomeration by Brownian motion, gravity settling and fluid turbulence. The particle size spectrum is assumed to be log-normal at all times.

### 6.2.4 NAUA Version Used in Source Term Reassessment Program

The NAUA aerosol code has been recently modified to account for spray washout[a]. The removal rate due to drop scrubbing is computed from:

$$\frac{dn}{dt} = -\varepsilon\pi R^2 N(V_d - V_p)n \tag{6.12}$$

where n = number concentration of aerosol
t = time
N = spray drop number concentration
R = radius of drop.

The single drop collection efficiency is computed using Equations (6.9) and (6.11), the same as used in the MATADOR.

## 6.3 MODELS FOR RETENTION IN ICE COMPARTMENTS

In the past relatively little effort has thus far been devoted to the modeling of fission product retention in the ice compartment of ice condenser containments. Ice condenser plants were not studied in WASH-1400 (Ritzman et al. 1974), so ice bed scrubbing was not analyzed in that study. Licensing evaluations have not as yet accounted for ice bed scrubbing, so no models applicable to site suitability source terms are available.

Ice bed scrubbing was treated parametrically in NUREG-0778 where a user-specified decontamination factor was used in the CORRAL code to analyze fission product transport in ice condenser plants. The same approach is used in MATADOR.

Recent work at PNL (Winegardner, Postma and Jankowski 1983) has resulted in preliminary models for fission product scrubbing within ice compartments. For elemental iodine, the compartment was modeled as a single, well-mixed volume. Retention by absorption in liquid water (formed by the melting of ice and the condensation of steam) and by deposition onto solid surfaces were accounted for. The fractional penetration for $I_2$ was found to be expressible as:

$$p = \frac{1}{1 + \frac{V_d A_d}{G_o} + \frac{LH}{G_o}} \tag{6.13}$$

(a) J. A. Gieseke, P. Cybulskis, R. S. Dennings, M. R. Kuhhman, and K. W. Lee. July 1983. Radionuclide Release Under Specific LWR Accident Conditions. BMI-2104, Vol. I, DRAFT, Battelle Columbus Laboratories, Columbus, Ohio 43201.

where $p$ = fractional penetration of $I_2$
$V_d$ = $I_2$ deposition velocity onto surfaces
$A_d$ = surface area for deposition
$G_o$ = outlet gas flow rate
$L$ = liquid flow rate
$H$ = $I_2$ equilibrium partition coefficient.

Particle scrubbing was analyzed by dividing the compartment into a number of volumes, in each of which the gas phase could be well mixed. For each volume, or mode, the fractional penetration was predicted from

$$p = \frac{1}{1 + \sum_i K_i/G_o} \tag{6.14}$$

where $p$ = fractional penetration per mode
$K_i$ = removal rate constant for i-th mechanism
$G_o$ = gas flow rate exiting from the mode.

A number of mechanisms were found to be applicable to particle trapping. For sedimentation, $K_s$ is the product of settling velocity and surface area

$$K_s = V_s A_s \tag{6.15}$$

where $K_s$ = removal rate constant for sedimentation
$V_s$ = particle settling velocity
$A_s$ = upward-facing surface area.

The ice is contained within baskets formed from perforated steel sheets. The wires, or strips, that make up the baskets represent targets against which particles could impact. The removal rate constant for impaction was expressed as:

$$K_I = V_I A_I \varepsilon_I \tag{6.16}$$

where $K_I$ = removal rate constant for impaction
$V_I$ = fluid velocity approaching the wires
$A_I$ = projected surface area of the wires
$\varepsilon_I$ = impaction efficiency.

Numerical values of $V_I$ were assigned parametrically; $A_I$ was estimated from basket geometry; and $\varepsilon_I$ was predicted by means of correlations developed for cylinders. The sum of efficiencies for impaction and interception were expressed as:

$$\varepsilon_I = \frac{stk^2}{(stk + 0.5)^2} - 0.04 + 2\frac{d_p}{d_c} \tag{6.17}$$

where $d_p$ and $d_c$ are diameters of particle and collector, respectively.

For very small particles, Brownian diffusion can be a significant depletion mechanism. Two fluid flow regimes were visualized: immersed flow around basket strips and flow parallel to a surface. For the first, the removal rate constant can be expressed in terms of a target efficiency:

$$K_{BD} = V_I A_{BD} \varepsilon_{BD} \tag{6.18}$$

where $K_{BC}$ = removal rate constant for diffusion
$V_I$ = gas velocity approaching strips
$A_{BC}$ = projected area for diffusional deposition
$\varepsilon_{BD}$ = diffusional capture efficiency.

$V_I$ was assigned values parametrically, $A_{BD}$ was computed from basket design, and $\varepsilon_{BD}$ was estimated from an equation presented by Pich (1966):

$$\varepsilon_{BD} = \frac{1}{Pe} + 1.727\frac{Re^{1/6}}{Pe^{2/3}} \tag{6.19}$$

where Pe = Peclet number = $d_c V_I / D$
$d_c$ = diameter of collector
$D$ = particle diffusivity
Re = Reynolds number = $\rho d_c V_I / \mu$
$\rho$ = gas density
$\mu$ = gas viscosity.

For the second flow regime, the removal rate constant was written as:

$$K_{BD} = \sum_i k_i A_i \tag{6.20}$$

where $k_i$ = mass transfer coefficient to i-th surface,
$A_i$ = surface area for the i-th surface.

For naturally convected flows, a mass transfer coefficienct can be predicted using a heat transfer/mass transfer analogy (Knudsen and Hilliard 1969; Bird et al. 1960, pp. 644-648):

$$\frac{k_d \ell}{D} = 0.13(Gr,Sc)^{1/3} \tag{6.21}$$

where $k_d$ = mass transfer coefficient, m/s
$\ell$ = length of surface in direction of flow, m
Gr = Grashof number
Sc = Schmidt number = $\mu/\rho D$.

The Grashof number characterizes the flow of nat··rally convected boundary layers. It may by expressed as:

$$Gr = \frac{\ell^3 g}{\nu^2} \frac{\Delta\rho}{\rho} \tag{6.22}$$

where $\ell$ = length of surface in direction of flow, m
g = acceleration due to gravity, $m/s^2$
$\nu$ = kinematic viscosity of gas, $m^2/s$
$\Delta\rho$ = density, difference in fluid-bulk compared to fluid at the surface, $kg/m^3$
$\rho$ = bulk density, $kg/m^3$.

An alternative formulation of the Sherwood number may be based on correlation for forced convection along surfaces. Typically, the Sherwood number varies with Sc and Re based on length. A correlating equation for flow along a flat plate (Sherwood et al. 1975) is expressed as:

$$\frac{k_d \ell}{D} = 0.037\ Sc^{1/3}(Re_\ell^{0.8} - 15{,}500) \tag{6.23}$$

The symbols in Equation (6.23) are as previously defined except that $Re_\ell$ uses the length of the surface (in the direction of flow) as the characteristic length.

The total value of $K_{BD}$ is the sum of the contributions expressed in Equations (6.18) and (6.20).

Diffusiophoretic deposition, which occurs as the result of steam condensation, may be formally numerically characterized by

$$K_D = V_D A_D \tag{6.24}$$

where $V_D$ = particle deposition velocity, m/s
$A_D$ = surface area for diffusiophoretic deposition, $m^2$.

Because both the value of $V_D$ and the steam condensation flux are directly proportional to the steam concentration gradient at the surface, it was found (Winegardner, Postma and Jankowski 1983) that the value of $K_D$ could be expressed in terms of steam mole fractions of gases entering and exiting from the ice compartment:

$$K_D = 0.9\ G_o \left[\frac{X_{2o}X_i}{X_{2i}} - X_o\right] \qquad (6.25)$$

where $G_o$ = gas outlet flow rate, $m^3/s$
x = mole fraction gases, refers to water vapor if no subscript and superscripts are defined as follows:
2 refers to air
o refers to outlet conditions
i refers to inlet conditions.

Thermophoretic deposition was found to be relatable to the difference in temperature of gases entering and leaving the ice compartment:

$$K_T = C_1 \frac{G_o}{\alpha} (BT_i - T_o) \qquad (6.26)$$

where $K_T$ = removal rate constant for thermophoresis, $m^3/2$
$C_1$ = particle thermal mobility, $m^2/s{\cdot}K$
$\alpha$ = thermal diffusivity of gas, $m^2/s$
B = a constant
T = gas temperature, °C
i,o = subscripts referring to inlet and outlet conditions, respectively.

A computer code, ICEDEF, was written to predict scrubbing efficiencies as part of the source term reassessment study[a]. The ICEDEF Code calculates scrubbing efficiency as a function of particle size for the mechanisms noted above. Ice surface area and availability are treated as inputs.

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(a) Gieseke, J. A., et al. 1983. Radionuclide Release Under Specific LWR Accident Conditions. BMI-2104, DRAFT, Battelle Columbus Laboratories, Columbus, Ohio.

## 6.4 SUPPRESSION POOL MODELS

Several models have been developed for predicting scrubbing efficiencies in BWR suppression pools. Much of the effort is quite recent and, as will be noted, has not yet been released in open publications.

### 6.4.1 WASH-1400

The CORRAL Code used in WASH-1400 (Ritzman et al. 1974) treated pool scrubbing parametrically. The code user may, through code inputs, select a decontamination factor (DF) for pool scrubbing. Decontamination factors used in WASH-1400 were 100 or 1, depending on whether the pool was subcooled or saturated. These DF values were used for both elemental iodine and for aerosol particles. A DF of unity was always used for noble gases and for organic iodides.

### 6.4.2 MATADOR

The MATADOR code treats pool scrubbing parametrically. The user inputs numerical values of the decontamination factors to be used.

### 6.4.3 Proposed by Diffey et al. (1965)

Diffey et al. (1965) proposed a model for the scrubbing efficiency of elemental iodine. It was assumed that iodine in the gases leaving the pool was in equilibrium with iodine in the pool water. The DF was expressed as:

$$DF = \left[1 - \frac{1 - e^{-x}}{x}\right]^{-1} \tag{6.27}$$

where $x = Qt/VH$

$Q$ = gas flow rate through pool, $m^3/s$

$t$ = time, s

$V$ = volume of pool water, $m^3$

$H$ = equilibrium partition coefficient.

Experimental measurements reported by Diffey et al. seem to support the plausibility of Equation (6.27) but scattered significantly. The scatter is thought to be attributable mainly to uncertainties in the partition coefficient, H.

### 6.4.4 Proposed by Devell et al. (1967)

Devell et al. (1967) carried out experiments with $I_2$ in water at 100°C, and concluded that iodine in gas bubbles did not necessarily reach equilibrium with iodine in the liquid. A proposed factor, $\alpha$, accounts for the degree of saturation:

$$\alpha = \frac{(1 - e^{-\mu}) + \varepsilon}{1 + \varepsilon} \qquad (6.28)$$

where $\alpha$ = degree of saturation of leaving gases,

$$\mu = \frac{k_{LB} A_B H h}{V_B U}$$

$$\varepsilon = \frac{k_{LS} A_s H}{Q}$$

$k_{LB}$ = mass transfer coefficient for rising bubbles
$A_B$ = area of bubbles
$h$ = height of water
$V_B$ = volume of bubbles
$\mu$ = bubble rise velocity
$k_{LS}$ = mass transfer coefficient for water surface
$A_s$ = area of water surface.

Experiments reported by Devell et al. (1967) generally confirmed the applicability of Equation (6.27) when corrected by the factor of Equation (6.28). Factors which controlled the iodine concentration (H) and the pH of pool water had the biggest effect on pool DF, as expected.

### 6.4.5 Proposed by Fuchs (1964)

Fuchs (1964) presents a "bubbling" model which predicts particle capture in single bubbles by three different mechanisms. For single, spherical bubbles whose surfaces are assumed to flow (circulate) the same as is calculated for ideal, nonviscous flow about a sphere, a relatively simple expression can be produced. For inertial deposition, due to centrifugal forces acting on suspended particles near the outer wall of the bubble, the fraction of suspended particles deposited per centimeter of path is given as

$$\alpha_i = \frac{9 V_b T}{2R^2} \qquad (6.29)$$

where $\alpha_i$ = fractional deposition due to inertia,
$V_b$ = bubble rise velocity
$T$ = inertial property c particle
$d$ = particle diameter
$\rho$ = particle density
$\mu$ = gas viscosity
$R$ = bubble radius.

For sedimentation, settling onto the bottom of the bubbles causes depletion characterized by a coefficient of absorption expressed as follows:

$$\alpha_s = \frac{3gT}{4RV_b} \qquad (6.30)$$

where $\alpha_s$ = fractional deposition (due to sedimentation) per centimeter of path
$g$ = acceleration due to gravity.

Depletion of particles by Brownian diffusion was computed by the penetration theory of mass transfer, and the absorption coefficient was expressed as

$$\alpha_D = 1.8 \left[\frac{D}{V_b R^3}\right]^{1/2} \qquad (6.31)$$

where $\alpha_D$ = fractional deposition (due to diffusion) per centimeter of path
$D$ = particle diffusion coefficient.

The total removal rate is assumed to be the sum of that due to these three mechanisms. It may be shown that the aerosol concentration inside a bubble varies exponentially with height. The decontamination factor for the Fuchs model is:

$$DF = \exp[(\alpha_i + \alpha_s + \alpha_D)h] \qquad (6.32)$$

where h = bubble rise distance.

### 6.4.6 SPARC

The SPARC (Suppression Pool Aerosol Removal Code) is being developed at PNL for use in the NRC Source Term Reassessment Program.[a] The following processes have been accounted for in the SPARC scrubbing model:

- convective flows resulting from the condensation or evaporation of steam
- particle growth caused by water vapor sorption by soluble aerosol material

(a) Owczarski, P. C., A. K. Postma, and R. I. Schreck. 1983. Technical Bases and User's Manual for SPARC - A Suppression Pool Aerosol Removal Code. DRAFT, NUREG/CR-3317, PNL-4742, Pacific Northwest Laboratory, Richland, Washington.

- sedimentation resulting from gravitational forces
- inertial deposition resulting from centrifugal forces
- diffusional deposition
- mechanical entrainment of pool liquid by the breaking of bubbles at the surface.

Particle depletion by the condensation of steam is computed under the assumption that particles are swept along with condensing steam. A DF applicable to all particle sizes may be expressed in terms of the steam content of entering gas and that in bubbles in equilibrium with the pool water:

$$DF = \frac{X_o}{X_i} \tag{6.33}$$

where $X_o$ = mole fraction of noncondensibles in inlet gas,
$X_i$ = mole fraction of noncondensibles in a gas bubble after the bubble attains thermal equilibrium in the pool near the inlet location.

Particle growth due to uptake of water by soluble particles is predicted from an equation that relates equilibrium drop diameter to relative humidity (Fletcher 1962):

$$s = \frac{\left[\exp \frac{2\sigma}{n_L KTa}\right]}{1 + \frac{imM_o}{M(\frac{4}{3}\pi a^3 \rho - m)}} \tag{6.34}$$

where $s$ = saturation ratio (the relative humidity)
$\sigma$ = surface tension of solution
$n_L$ = no. of molecules/cm$^3$ of solution (solvent and solute)
$K$ = Boltzmann constant
$T$ = temperature, K
$a$ = radius of drop
$i$ = van't Hoff ionization factor
$\rho$ = density of solution
$M$ = mass of solute in the drop
$M_o$ = molecular weight of solvent
$m$ = molecular weight of solute.

Particle growth is computed in SPARC using the assumptions that s = 0.99 and that the solute is cesium iodide.

Particle depletion due to gravity settling is computed for spherical bubbles from a model which accounts for the upward vector of the steam evaporation flux as well as the downward motion due to gravity. A decontamination factor that results after time t is found to be:

$$DF = \exp \left\{ \frac{3\Delta t}{2D} \left[ V_s - \frac{2}{3} V_v \left( \frac{V_s}{V_v} \right) \right] \left( \frac{V_s}{V_v} \right)^2 \right\} \tag{6.35}$$

where DF = decontamination factor due to sedimentation
$\Delta t$ = residence time of bubble in the pool
D = bubble diameter
$V_s$ = particle settling velocity
$V_v$ = gas bulk flow velocity due to steam evaporation.

Centrifugal deposition in a circular bubble is computed from a model which accounts for the retarding effect of the steam evaporatioh flux at the surface of the bubble. The DF is predicted to be:

$$DF = \exp \left[ \frac{6\Delta t\ V_{cm}}{D} \left( 1 - \frac{V_v}{V_{cm}} \right)^{1/2} \left( \frac{2}{3} + \frac{V_v}{V_{cm}} \right) \right] \tag{6.36}$$

where DF = decontamination factor due to centrifugal force,

$$V_{cm} = \frac{V_b \rho_p d_p C_m}{4D\mu} = \text{maximum centrifugal drift velocity.}$$

Deposition of suspended particles as a result of diffusion is modeled in SPARC by means of the penetration theory for mass transfer (Bird et al. 1960, pp. 636-681). The DF due to this mechanism is expressible as:

$$DF = \exp \left[ \frac{12\theta}{D} \left( \frac{DV_b}{\pi D} \right)^{1/2} \Delta t \right] \tag{6.37}$$

where $\theta$ = a correction factor due to the inward steam flux at the bubble interface.

The correction factor is predicted from the penetration theory:

$$\theta = (1 + \text{erf}\,\phi)^{-1} \exp(-\phi^2) \tag{6.38}$$

where $\phi$ is equal to $\phi_{AB}/\pi$ and where $\phi_{AB}$ is defined as

$$\phi_{AB} = \frac{V_v}{D\left(\frac{DV_b}{\pi D}\right)} \tag{6.39}$$

The overall DF is computed as the product of the DFs calculated for each of the four mechanisms. Residence time for bubbles is computed as the ratio of rise distance divided by swarm rise velocity.

Entrainment of pool liquid by the breaking of bubbles is accounted for in SPARC by limiting the maximum DF to $10^5$.

Also available are optional equations that account for the oblate spheriod shape of larger bubbles. To exercise this option, the user must input the ratio of major to minor axes. For gravity settling, the argument of the exponential term in Equation (6.35), the rate constant or $K_s$, is calculated by:

$$K_s = \frac{3}{2}(V_s - V_v)(R_a)^{2/3}\Delta t/D \text{ and} \tag{6.40}$$

$$DF = \exp^{K_s \Delta t}$$

where $R_a$ = is the ratio of the major axis to the minor axes.

Centrifugal deposition in elliptical bubbles is accounted for in SPARC by a multiplication factor applied to the particle draft velocity, $V_c$, illustrated in Equation (6.36) for spheres:

$$V_c' = y^* V_c \tag{6.41}$$

where $V_c'$ is the average drift velocity in elliptical bubbles. The term $y^*$ is related to the axis ratio by

$$y^* = 4.222\ R_a - 6.232 \text{ for } R_a > 3 \tag{6.42}$$

$$y^* = 0.9444R_a^2 - 1.0776R_a + 1.1332 \text{ for } R_a \leq 3 \tag{6.43}$$

Diffusional deposition in the oblate spheriod is computed from the penetration theory, accounting for the increased surface area and exposure time. The diffusional deposition velocity for spheres is reduced by $R_a^{1/6}$:

$$V_D' = V_D/R_a^{1/6} \tag{6.44}$$

where $V_D'$ = deposition velocity for the oblate spheriod,
$V_D$ = deposition velocity for spheres.

The surface area for both diffusional and centrifugal deposition is computed for the spheriod by:

$$A_s = 2\pi a^2 + \frac{\pi b^2}{e} \ln \left[\frac{1 + e}{1 - e}\right] \tag{6.45}$$

where $A_s$ = surface area of spheriod
$a$ = major axes length
$b$ = minor axis length
$e$ = eccentricity.

### 6.4.7 Unpublished Codes

Both the General Electric Company and the Electric Power Research Institute have developed computer codes to predict particle scrubbing in suppression pools. Although published references are not yet available, informal discussions with people who are familiar with the codes indicate that the codes are mechanistic.

## 6.5 FILTRATION MODELS

Filter/absorber system performance is conservatively credited in design basis accident evaluations related to nuclear plant licensing. Relatively modest removal efficiencies (95% to 99%) are allowed for filtration system that meet ESF requirements. Key design requirements of ESF system are: 1) redundancy of active components, 2) in-place testability, and 3) stringent design standards (USAECO 1973). It should be noted that Regulatory Guide 1.52 for filtration systems (USAEC 1973) was formulated on the basis of hypothetical design basis accidents that did not include the large aerosol masses which would accompany the servere accidents of interest to the present study. Therefore the Regulatory Guide would not be expected to be applicable to core melt accidents.

A large body of literature exists on filters and filtration theory. The literature can be divided into two main classifications -- filtration theory and filter performance. The data on filtration theory is primarily in the area of the so-called "single-fiber filtration" theory. Particles in flowing air are captured on fibers intercepting the flow by one or more mechanisms. The principal mechanisms identified are interception, impaction, diffusion, and electrical effects (charged particles or materials, induced charges, etc.).

The single-fiber efficiency for a mechanism is the fraction of particles which are collected by the fiber from all the particles challenging the filter. The volume of interest is shown in Figure 6.1. The mechanisms of removal for the three mechanical collection modes are illustrated in Figure 6.2. These drawings were adapted from Hinds (1982).

The equation for the single-fiber collection by interception is:

$$E_R = \frac{1}{2Ku} \; 2(1+R) \ln (1+R) - (1+R) + \frac{1}{1+R} \tag{6.46}$$

where Ku = Kuwabara hydrodynamic factor = $\frac{\ln\alpha}{2} - \frac{3}{4} + \alpha - \frac{\alpha^2}{4}$

$\alpha$ = solidity (the volume fraction of the fibers)

R = dimensionless interception parameter, df/dp

$d_p$ = diameter of the particle

$d_f$ = diameter of the fiber

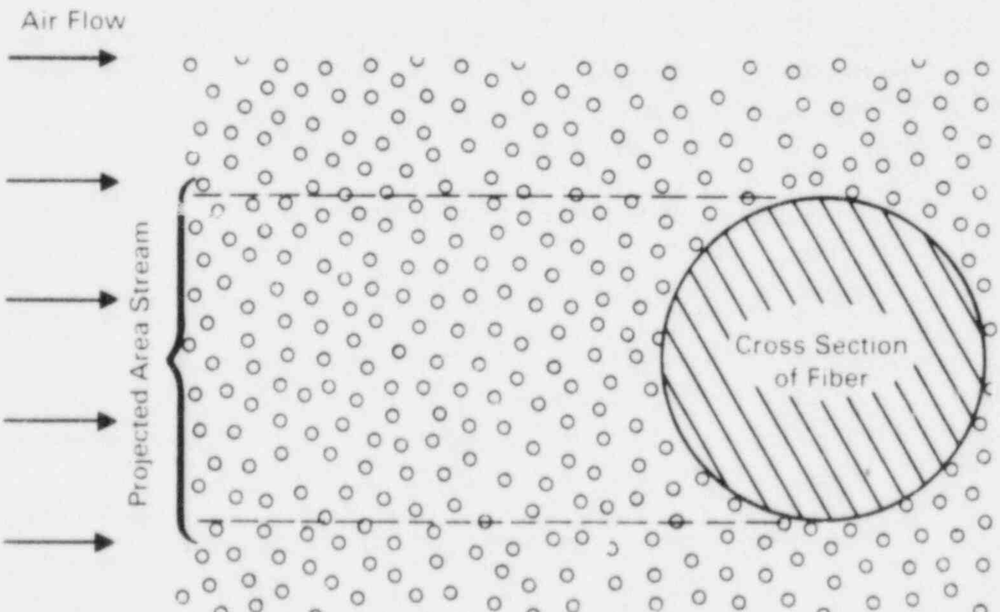


FIGURE 6.1. Single-Fiber Efficiency

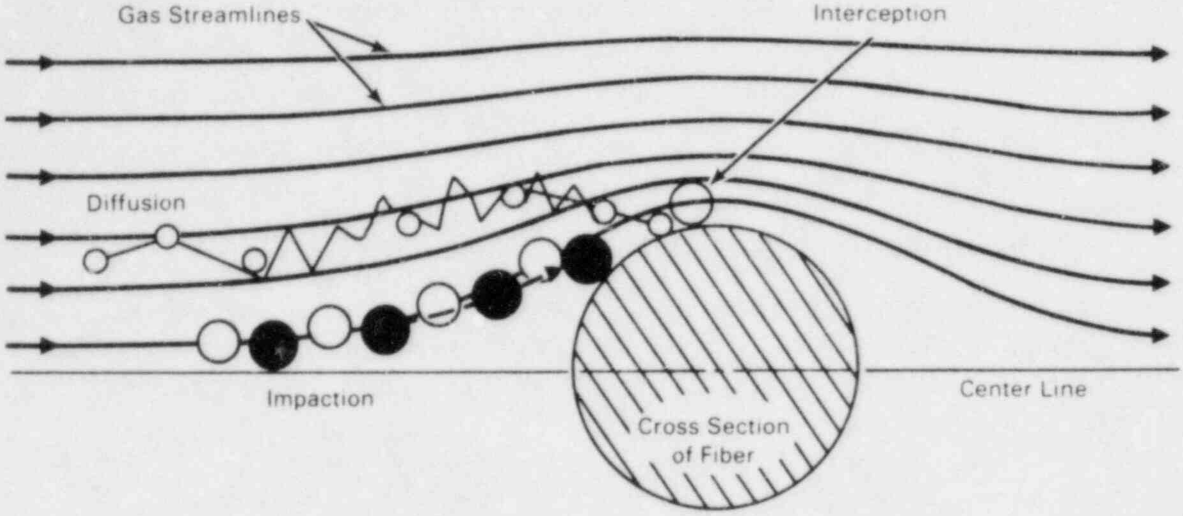


FIGURE 6.2. Mechanical Removal Mechanisms

More complex expressions of the Kuwabara factor are required when df is less than the mean free path ($\lambda$) of the gas (Hinds 1982).

The equation for the single-fiber efficiency for impaction is:

$$E_I = \frac{J\ (StK)}{2\ Ku^2} \qquad (6.47)$$

where J = Particle flux due to diffusion, $(29.6 - 28\alpha^{0.62})\ R^2 - 27.5\ R^{2.8}$ for R less than 0.4

Stk = Stokes number, $\frac{\Gamma\ U_o}{d_f} = \frac{\rho_p\ d_p{}^2 C_c\ U_o}{18 \eta d_f}$

$\Gamma$ = relaxation time for particle
$U_o$ = face velocity at filter
$C_c$ = Cunningham correction (slip) factor, $1 + \frac{\lambda}{dp}\left[2.514 + 0.08 \exp\left(-0.55 \frac{dp}{\lambda}\right)\right]$
$\eta$ = viscosity
$\rho_p$ = density of particle
$d_p$ = diameter of the particle
$d_f$ = diameter of the fiber

The equation for the single fiber efficiency for diffusion is based solely on the Peclet number:

$$E_D = 2\ Pe^{-\frac{2}{3}} \tag{6.48}$$

where $Pe$ = Peclet number, $\frac{d_f\ U_o}{D}$

$D$ = diffusion coefficient, $3\pi\eta\ V_d/C_c$

$V_d$ = relative velocity between particle and gas

and the remaining terms have been previously defined.

The distortion of the stream lines around the fibers is calculated by one of several methods. Two methods [Kuwabara (1959) and Happel (1959)] are based upon unit-cells of fluid within hypothetical boundary surfaces having the shape of a cylinder. Kuwabara assumes decreasing vorticity and fibers transverse to the flow while Happel assumes decreasing drag with fibers both transverse and parallel to flow. Kirsch and Fuchs (1967) and Davies (1973) have used empirical correlations. Spielman and Goren (1968) based their concept on Brinkman flow and determine the average dissipation due to all the cylinders over the entire fluid field by assuming typical cylinders and uniform flow. All these models assume uniformity in the filters, which are known to be nonuniform, and the pressure drop through filters is overestimated (Bergman et al. 1980). Thus, although the models are useful in understanding trends in particle removal, they are not currently capable of quantitatively predicting the behavior of filters.

There is a large body of information on filter performance. There have been seventeen biannual AEC/ERDA/DOE Air Cleaning Conferences with many papers devoted to this subject. The articles primarily cover the behavior of clean HEPA filters subjected to assorted challenges. Other types of filtration devices (e.g., demisters) have also been covered (First and Leith 1976). However, despite the large amount of information, little pertains to the performance and failure modes of filters and systems when challenged by the atmospheres predicted for severe accidents.

Some data are available on the increase in pressure drop through the filters with increasing load (Gregory et al. 1982; Lee 1974) and on the failure of filters subjected to shock overpressure (Andrae et al. 1980; Cuccuru et al. 1982). Finally, SGTS failure modes are discussed in the recent evaluation of a small break LOCA outside BWR containment (Wichner et al. 1983).

The performance of filter systems is modeled in CORRAL (Ritzman et al. 1976) and MATADOR (Baybutt, Raghuram and Ava 1982) in terms of user-specified decontamination factors. Neither the effect of particle size distribution on removal efficiency nor the effect of accumulated particle mass on system operability are specifically accounted for in these models.

Filter/absorber system performance is conservatively credited in design accident evaluations related to nuclear plant licensing. Relatively modest removal efficiencies (93% to 99%) are allowed for filtration systems that meet ESF requirements. Key design requirements for ESF systems are (USAEC 1973): 1) redundancy of active components; 2) in-place testability; and 3) stringent design standards. It should be noted that the regulatory guide for filtration systems was formulated on the basis of the hypothetical design basis accidents that did not include the large particulate mass airborne concentrations that would accompany the severe accidents of interest to this present study. Thus the regulatory guide would not be expected to apply to the core melt accident.

## 6.6 CONTAINMENT COOLERS

Fission product depletion by containment coolers has apparently not been accounted for in models used for accident analysis. Both CORRAL (Ritzman et al. 1974) and MATADOR (Baybutt, Raghuram, and Ava 1982) could account for such removal on the basis of a user-specified removal efficiency, but best-estimate predictions of accident consequences will require the development of a mechanistic model.

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The Pacific Northwest Laboratory has compiled and reviewed base line data on the effectiveness of Engineered Safety Feature (ESF) systems in the retention of fission products and particulate material resulting from a nuclear reactor accident. This work is part of an NRC project to provide the best estimates of the consequences of severe reactor accidents.

The resulting report describes the ESF systems (containment spray, secondary containment filter, containment recirculating filter, pressure suppression pool, ice condenser, and main steam line isolation valve leakage control systems). Also described are the anticipated atmospheres in which the ESFs must operate, the experimental studies of ESF system effectiveness, and the models currently available for assessing the performance of the various ESF systems. The information gaps identified as a result of this review have resulted in recommendations for additional work in the areas of: 1) performance data and models of containment chiller/coolers; 2) continued development and experimental verification of the ice condenser model; 3) continued development of the pressure suppression pool model; and 4) continued investigations of the behavior of filtration devices.


17. KEY WORDS AND DOCUMENT ANALYSIS

ESF
Engineered Safety Features
fission product retention
severe reactor accidents
core-melt accidents
reactor containment atmospheres

17a. DESCRIPTORS

recirculating air cooler
containment spray
recirculating filter
pressure suppression pool
ice condenser
filter system

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