SESSION IV

CONTAINMENT OF ACCIDENTAL RELEASES

Tuesday, August 8, 1978
CHAIRMAN: J. T. Collins

DEMONSTRATION OF AN EMERGENCY CONTAINMENT SYSTEM

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Abstract

A system called an Emergency Containment System (ECS) to be used for tertiary containment of tritium was reported at the 13th Air Cleaning Conference. This system was part of the Tritium Effluent Control Laboratory then under construction at Mound Facility. A series of experiments has recently been conducted to evaluate the performance of an ECS in capturing tritium accidentally released into an operating laboratory.

The ECS is an automatically actuated laboratory air detritiation system utilizing a catalytic oxidation reactor and presaturated oxide adsorption/exchange columns. In the event of an accidental release of tritium into the laboratory, the ECS is automatically activated, and quick-acting pneumatic dampers divert the laboratory air supply and exhaust through the ECS until room concentrations are returned to safe operating levels.

The experiments involved the release of elemental tritium into a 560 m^3 laboratory. Concentrations, which initially were in excess of 5000 $\mu Ci/m^3$, were reduced to less than 5 $\mu Ci/m^3$ in about two hours. During the experiments, data were obtained on the buildup of tritium oxide in the laboratory air, and swipes were taken on several surfaces to determine tritium deposition.

The results of the experiments have shown that a tertiary containment of tritium is feasible. In the event of a catastrophic accident, the ECS is capable of preventing the release of a large quantity of tritium to the environment.

Introduction

Mound Facility began a Tritium Effluent Control Technology Project in January 1972. A goal of this project was to develop and demonstrate technology and equipment to maintain tritium emissions to the atmosphere below 10% of the Radiation Concentration Guide (RCG) levels. Kershner et al. (1) described the "Tritium Effluent Control Laboratory" at the 13th AEC Air Cleaning Conference. A major part of this program was to develop and demonstrate a tritium containment system capable of preventing tritium from reaching the biosphere in the event tritium is released into a typical laboratory.

*Mound Facility is operated by Monsanto Research Corporation for the U. S. Department of Energy under Contract No. EY-76-C-04-0053.

Since 1973 accidents at major nuclear installations in the United States have resulted in excess of one-half (2-4) million curies of tritium reaching the environment. In each case it was determined that the impact on the general public and the environment did not exceed appropriate guidelines (2-4). But the increased emphasis on radioactive emissions and the tightening of regulations indicate that similar incidents in the future may be both politically and legally unacceptable.

An indication of the extent to which standards can be raised is evident in the U. S. EPA Interim Primary Drinking Water Regulations (5). The EPA reduced the acceptable level in drinking water from 3 μCi/liter to 0.02 μCi/liter or a reduction of 150 times.

The EPA has also established total quantity limits on specific radionuclides associated with the uranium fuel cycle (6). In the Final Environmental Statement for 40 CFR 190, the EPA states "Similarly, as knowledge becomes available concerning the capability of technology to limit environmental releases of tritium and carbon-14, the appropriate levels of these radionuclides will be carefully considered by the Agency."

The national effort to develop fusion as an alternative energy supply will result in megacurie quantities of tritium being handled. The Tritium Systems Test Assembly at Los Alamos Scientific Laboratory and the Tokamak Fusion Test Reactor at Princeton (7-8) have incorporated large-scale tritium containment systems. Mound Facility has completed conceptual design of a similar emergency containment system, as described in this paper, for its major tritium facility.

System Design

The Emergency Containment System (ECS) is an automatically actuated, room air detritiation system utilizing a catalytic oxidation reactor and presaturated oxide adsorption/exchange columns. In the event of an accidental release of tritium to the laboratory, the ECS is automatically activated and the quick-acting pneumatic dampers divert the room air supply and exhaust through the ECS until the tritium concentration in the room air is returned to a safe operating level. The air stream is heated to a temperature of 175°C before it enters the catalyst to meet the design goal of 99.9% oxidation of tritium in air at inlet concentrations of 0.5 ppm. The air is cooled to near 20°C before entering the adsorption columns. The oxidized tritium is captured and contained on the adsorption columns in the oxide form. The ECS is designed to provide the oxidation and adsorption capacities for a single pass decontamination factor of 1000:1 in a 0.66 m^3/s air flow containing as high as 1 Ci/m^3 tritium and 0.5 ppm natural hydrogen background. The adsorber section of the ECS consists of two stainless steel vessels containing 1730 kg of Alcoa alumina H151 adsorbent, saturated at 100% relative humidity. These vessels were sized to provide 10 hr of operation with an inlet activity of 1 Ci/m^3 before a defined breakthrough of 100 $\mu Ci/m^3$ occurs. A Spencer turbine controls gas flow for the ECS.

Experiment Design

The efficiency of the ECS to capture tritium released into an operating area was evaluated by releasing three elemental tritium samples into the laboratory. The automatic start-up of the ECS was controlled by a 20-liter ionization chamber monitoring the laboratory. The exhaust from the area was also monitored by a 20-liter ionization chamber. Bubbler-type tritium monitors containing ethylene glycol (9) sampled the laboratory exhaust, the area adjacent to the exterior doors, and the supply to the room coming from the ECS adsorber beds. These monitors provided data on the quantity of tritium oxide at the various sampling locations. Fifteen 100-cm^2 areas at various locations throughout the laboratory were marked off to determine tritium deposition on laboratory surfaces. Tritium deposition was determined after the tritium release by wiping the areas with dry Metricel filter paper. The wipes were counted by liquid scintillation counting.

After each experiment the adsorber bed used was regenerated by purging it with an air stream heated to 100°C. An ethylene glycol bubbler monitor sampled the effluent air stream to determine the quantity of tritium captured on the bed.

First Experiment

In the first experiment 0.87 Ci of elemental tritium was released with 20 cm^3 of hydrogen. The container was flushed with a nitrogen purge to ensure that all the tritium and hydrogen were released. The 20 cm^3 of hydrogen increased the hydrogen concentration to approximately 0.03 ppm above the natural abundance of 0.5 ppm. The hydrogen release simulated an accident where the average room concentration would be 0.1 Ci/m^3 or 10% of the design criteria of 1 Ci/m^3.

Figure 1 is a plot of the tritium concentration in the laboratory resulting from the first experiment. It required 70 min for the laboratory tritium concentration to return to less than 5 $\mu Ci/m^3$. The ECS was automatically activated approximately 10 sec after the sample was released. The air stream was at the design temperature of 175°C after 10 min. Figure 2 shows the effluent levels during the experiment. During the first 2 min of the experiment 0.08 Ci was released to the environment. This release is attributed to tritium that escaped before the dampers closed. The 2 min required for the monitoring system to respond to the tritium release in the first 10 s is attributed to the travel time from the dampers to the monitoring probe which sampled the room exhaust. A large filter-bank between the dampers and the probe is effectively a holding tank which increases the time of travel.

In addition to the 0.08 Ci which escaped before the dampers closed, 0.28 Ci was released which is attributed to leakage through the dampers. The dampers used are fast-acting, conventional ventilation dampers with an estimated leakage rate of 10% per hour. The total quantity lost to the exhaust system was 0.36 Ci or 41% of the quantity released. These data show the importance of fast-acting low-leakage dampers located far enough downstream to accommodate the time required to activate the system.

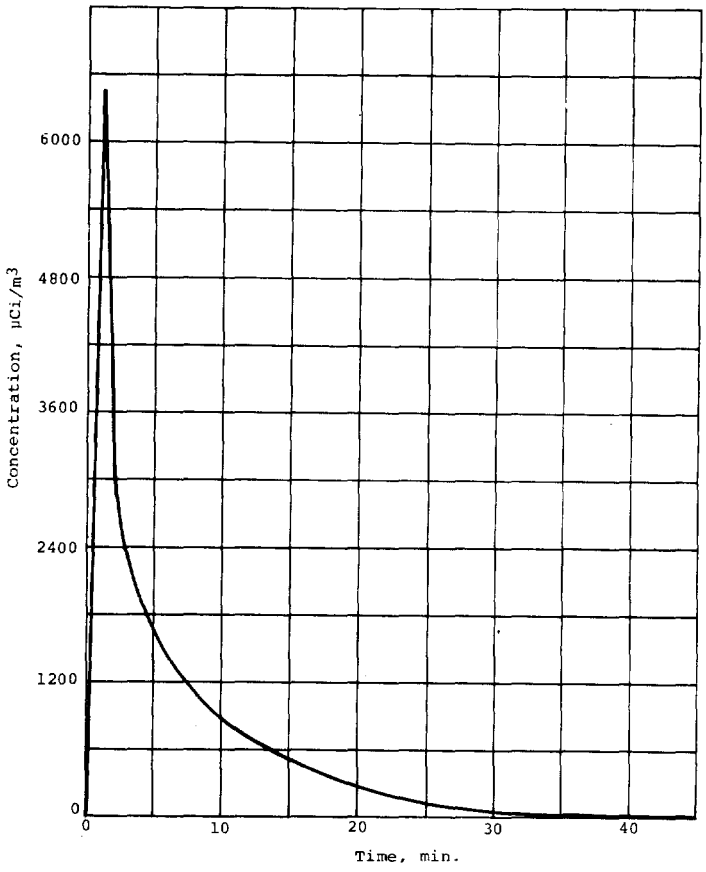

FIGURE 1

ECS EXPERIMENT 1; LABORATORY CONCENTRATION AS A FUNCTION OF TIME

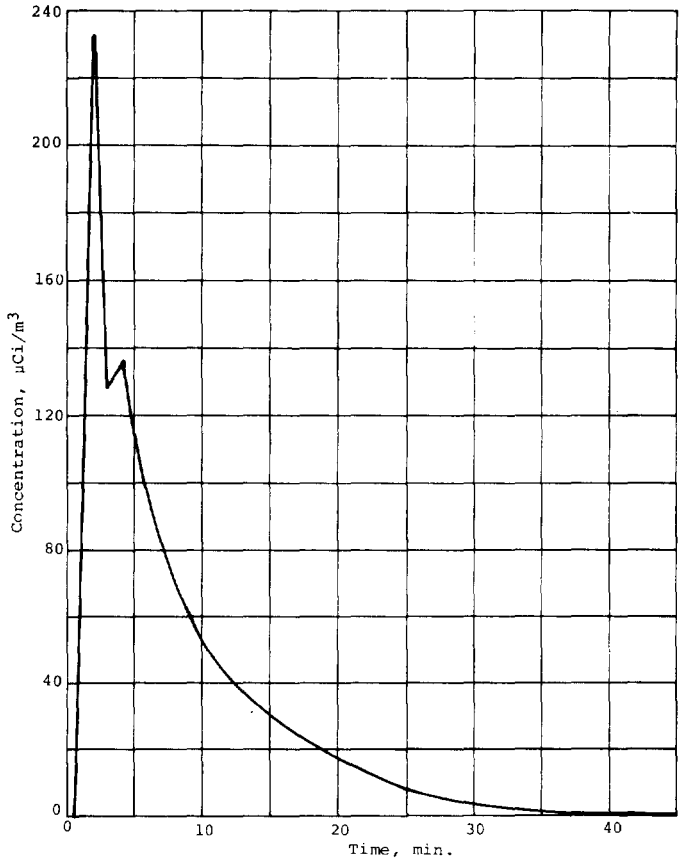

FIGURE 2

ECS EXPERIMENT 1; EXHAUST SYSTEM CONCENTRATION AS A FUNCTION OF TIME

The tritium oxide concentration in the laboratory increased during the experiment to 2.9 $\mu Ci/m^3$ or 60% of the Radiation Concentration Guide. The tritium oxide concentration measured just outside the laboratory doorways was less than 7×10^{-3} $\mu Ci/m^3$. This indicates that the doors, which have refrigeration gasketing and metal door sweeps, were effective in preventing tritium leakage. Sampling the exhaust leaving the adsorber bed indicated that 0.006 Ci of tritium oxide was returned to the laboratory by the ECS.

Examination of individual samples taken from the adsorber bed exhaust showed the data did not follow the pattern predicted by a breakthrough phenomenon, i.e., concentration increase as a function of time. Instead, the data showed that the outlet concentration was dependent on the inlet concentration, indicating that the tritium oxide recycled into the laboratory was not the result of breakthrough but channeling through the bed. After the experiment, the bed was regenerated by first flowing 100°C dry air through the bed to remove the tritium oxide and then passing saturated air at 20°C through the bed. At this temperature, approximately 95% of the tritium should be removed by the drying step. Sampling of the bed exhaust during regeneration indicated that 0.37 Ci or 42% of the quantity released had been captured on the bed. For this experiment 88% of the original 0.87 Ci release could be accounted for. Tritium deposition on laboratory surfaces is shown in Table 1.

Table 1. Tritium deposition on Laboratory surfaces (Experiment 1).

Surface	Tritium Deposited (dis/min/100 cm^2)
Iron	Nondetectable
Sheet Metal	Nondetectable
Plexiglas	Nondetectable
Glass	Nondetectable
Bench Top	Nondetectable
Glass	Nondetectable
Desk	9
Bench Top	Nondetectable
Sheet Metal	Nondetectable
Glass	5
Wood	Nondetectable
Tile	2
Tile	Nondetectable
Desk Top	Nondetectable
Tile	Nondetectable

Second Experiment

In the second experiment 0.86 Ci of elemental tritium was released followed by five liters of hydrogen (diluted with nitrogen to maintain a hydrogen concentration below 50% of the lower explosive limit). The five liters of hydrogen released into the 560 m^3 room raised the hydrogen concentration to approximately 10 ppm, equivalent to a tritium concentration of ~30 Ci/m^3 or 30 times design criteria. To determine the impact of not heating the air stream to 175°C the preheater was shut off. The only heat supplied to the system was the heat of compression of the turbine blower. The air stream reached a temperature of 80°C from the heat of compression. Figures 3 and 4 show the laboratory concentration and the effluent concentration during Experiment 2. The ECS was automatically activated approximately 20 s after the release.

During the first 3 min 0.15 Ci was released. An additional 0.39 Ci was released during the remaining time. The total tritium lost to the exhaust system was 0.54 Ci or 61% of the quantity released. Tritium oxide concentration in the laboratory averaged 3.5 $\mu Ci/m^3$. The total tritium oxide recycled into the laboratory was 0.007 Ci. The concentration data again appeared to be a function of inlet concentration and not that associated with breakthrough. The total quantity of tritium oxide released through the exhaust system was 0.024 Ci or approximately three times that observed being recycled. This experiment required 120 min to reduce the tritium concentration to less than 5 $\mu Ci/m^3$ as compared with the 60-70 min required for the previous experiment.

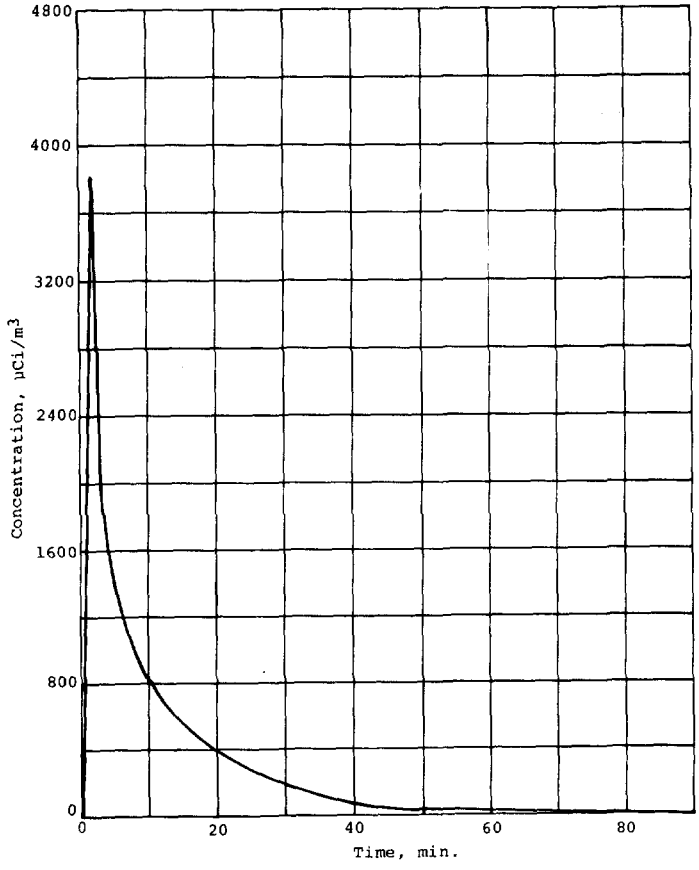

FIGURE 3

ECS EXPERIMENT 2; LABORATORY CONCENTRATION AS A FUNCTION OF TIME.

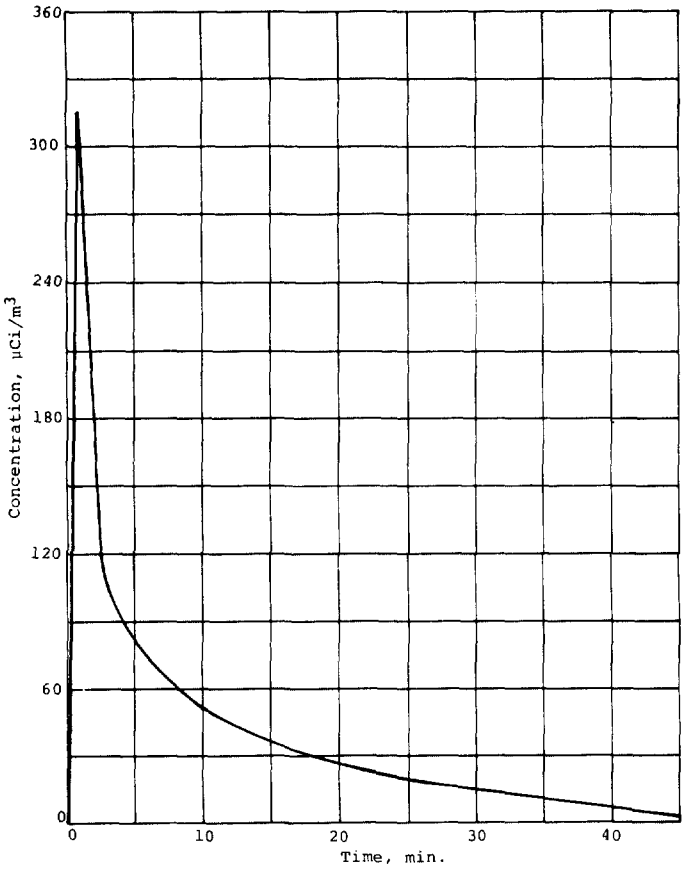

FIGURE 4

ECS EXPERIMENT 2; EXHAUST SYSTEM CONCENTRATION AS A FUNCTION OF TIME.

Table 2 shows the tritium deposition on laboratory surfaces for the second experiment. The deposition on the liquid nitrogen dewar during the second experiment indicates a significant increase in surface contamination on wet or cold surfaces. This is consistent with earlier predictions by Maroni (10). The higher surface contamination after this experiment can be explained by the fact that high airborne concentrations persisted for longer times than in the earlier experiment. Regeneration of the adsorber bed removed 0.61 Ci or 69% of the release. The bed used for this experiment had been contaminated as a result of prior work including Experiment 1. The apparent inventory greater than the release is attributed to this prior work.

Third Experiment

A third experiment was performed to simulate a system in which low leakage dampers would be located at a sufficient distance downstream to prevent any tritium from passing the dampers before they closed. In this experiment 0.88 Ci of elemental tritium was released with no additional hydrogen. Prior to the release, the laboratory supply and exhaust were shut down and weather balloons were inflated inside the exhaust and supply ducts to provide a tight seal. The ECS was manually activated before releasing the tritium to prevent any tritium loss during the time required for the monitoring systems to activate the ECS.

Table 2. Tritium deposition on laboratory surfaces (Experiment 2).

Surface	Tritium Deposited (dis/min/100 cm^2)
Iron	48
Sheet Metal	26
Plexiglas	55
Glass	35
Bench Top	33
Glass	19
Desk	21
Bench Top	84
Sheet Metal	21
Glass	31
Wood	36
Tile	22
Tile	36
Desk Top	21
Tile	
Liquid Nitrogen Dewar	3388
Ice on Liquid Nitrogen Dewar	26704

Figure 5 shows the laboratory concentration of tritium during the third experiment. The tritium concentration was reduced to less than 5 $\mu Ci/m^3$ in 120 min. The exhaust monitoring system level increased to 2 $\mu Ci/m^3$ after 10 min and returned to background in another 10 min. There was no measurable air flow at the sampling location and, therefore, the total quantity released was insignificant. The average tritium oxide concentration in the laboratory was 0.52 $\mu Ci/m^3$ during the experiment. Sampling of the exterior areas showed no detectable concentrations of tritium. The bubbler system monitoring the adsorber bed indicated that a total of 0.005 Ci was recirculated into the laboratory. Table 3 shows the surface contamination levels for the 15 wipe areas. Regeneration of the adsorber bed accounted for 0.59 Ci or 66% of the quantity released.

Data Analysis

The concentration data obtained during the three ECS experiments were used to determine the efficiency and decontamination factor for the ECS. If a well-mixed atmosphere is assumed, the fraction of tritium remaining in the room can be expressed by Equation 1 and the fraction released to the environment can be expressed by Equation 2.

$$T = e^{-(\gamma_L + \varepsilon\gamma_R)t} \quad (1)$$

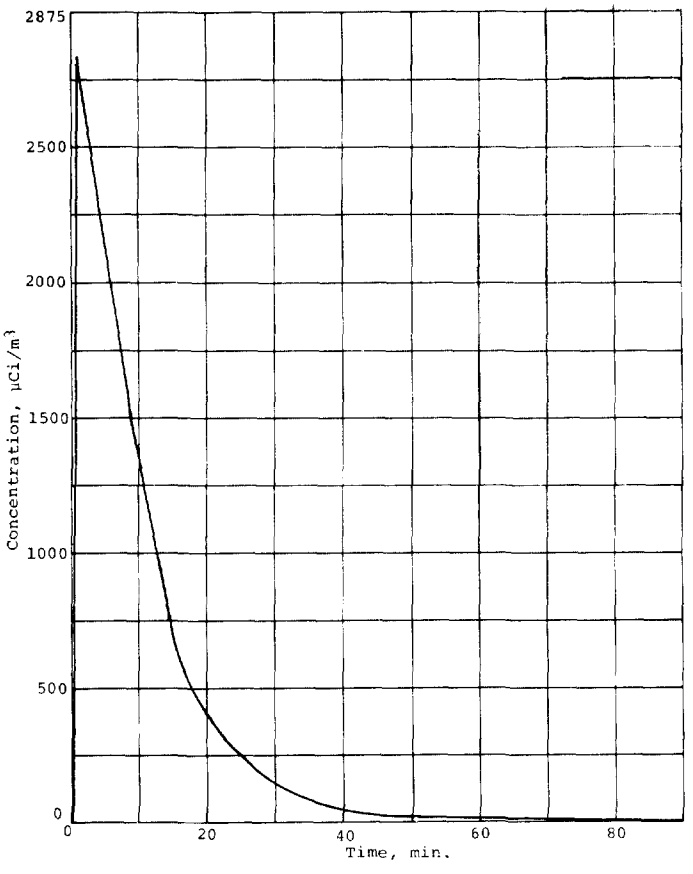

FIGURE 5

ECS EXPERIMENT 3; LABORATORY CONCENTRATION AS A FUNCTION OF TIME.

Table 3. Tritium deposition on laboratory surfaces (Experiment 3).

Surface	Tritium Deposited (dis/min/100 cm^2)
Iron	72
Sheet Metal	104
Plexiglas	95
Glass	111
Bench Top	85
Glass	137
Desk	91
Bench Top	97
Sheet Metal	96
Glass	81
Wood	79
Tile	137
Tile	96
Desk Top	103
Tile	105

$$T' = \frac{\gamma_L}{\gamma_R \varepsilon \ \gamma_L} [1-e^{-(\gamma_R \varepsilon + \gamma_L) t}] \quad (2)$$

where T = the fraction of the tritium initially present that remains in the room after time t.

T' = the fraction of the tritium initially present that is leaked to the environment in time t.

t = time, hr.

γ_R = removal time constant, hr^{-1} = 4.2.

γ_L = leak time constant, hr^{-1}. This is not known directly, but may be determined from the values of T and T'.

ε = system tritium removal efficiency. This too must be calculated from T and T'.

The decontamination factor (DF) is defined as the reciprocal of the fraction of tritium initially present which remains uncontained after time t.

$$DF = \frac{1}{T + T'} \quad (3)$$

After 25 min during Experiment 1, the entire area serviced by the ECS was at a uniform concentration of 110 $\mu Ci/m^3$ as determined by the fact that two air monitors at different parts of the laboratory showed the same concentration. The total quantity of tritium in the 560-m^3 area at that time was 0.06 Ci. The concentration then decreased to 0.009 of the original or 1 $\mu Ci/m^3$ in 0.75 hr. The quantity released to the effluent during this time period was 0.01 Ci or 0.17% of the initial quantity of tritium in the area.

Substituting these values into Equations (1) and (2), one finds that the calculated efficiency of the ECS is 1.4. An efficiency greater than one is presumably the result of incomplete mixing, i.e., the tritium concentration in the laboratory was <u>not</u> uniform.

The decontamination factor for Experiment 1 was calculated to be 6.

A uniform concentration of 38 $\mu Ci/m^3$ was assumed after 60 min during Experiment 2. It required 1 hr more to decrease the concentration to 1 $\mu Ci/m^3$. During this period 0.003 Ci of the 0.021 Ci in the room at the beginning of the period was released to the ventilation system. The calculated efficiency of the ECS system was 0.7. The calculated efficiency of the catalysts at 80°C is 0.9 (11). The decontamination factor during Experiment 2 was determined to be 6.

The equilibrium concentration during the third ECS experiment occurred in 25 min and was 266 $\mu Ci/m^3$. After 1.6 hr the concentration was 1 $\mu Ci/m^3$ or 0.003 of the original concentration. No measurable quantity of tritium was released to the ventilation system. The efficiency of the system was determined to be 0.9. The decontamination factor during the 1.6 hr was 333. The decontamination factor for

the entire 2-hr experiment was 1800. The predicted decontamination factor using the model was 1900. Note here that the decontamination factor increases with time. In Experiments 1 and 2, because of leakage, the decontamination factor approaches a limiting value after long clean-up times.

The third ECS experiment has shown that the ECS will function as designed if low leakage dampers are installed.

Summary

These experiments have shown that the ECS is capable of capturing tritium released into a laboratory before it reaches the biosphere. The ECS is capable of removing tritium at levels that approach natural abundance hydrogen concentration. The importance of low-leakage dampers and their locations were also demonstrated. Experiment 2 showed the ECS will function with only the heat of compression from the blower but with a longer cleanup time. Excessive surface contamination was not observed except on a low-temperature surface. The mathematical model developed for design criteria was shown to be capable of predicting experimental results.

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STUDY OF THE CONTAINMENT SYSTEM OF THE PLANNED SNR-2 FAST BREEDER REACTOR

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Abstract

Research is presented concerning the behaviour of four possible containment concepts for the planned Fast Breeder Reactor SNR-2 under the conditions of a HCDA. Detailed descriptions and calculations are given dealing with the acceptable leakages and the filter loads reached. The transport of the aerosol-type as well as the gaseous activity through the different compartments into the environment is calculated and from this the resulting accident doses.
To cover the uncertainties of the accident course a wide range of accident parameters is examined.

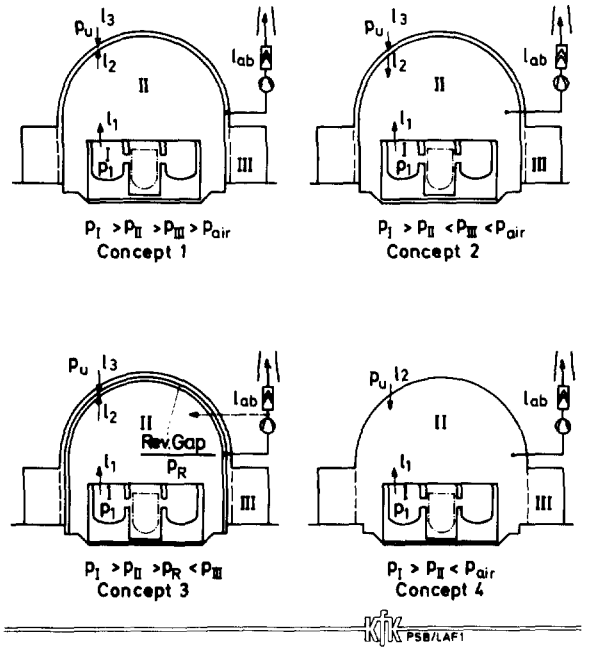
1. Introduction

Research was carried out to find a containment concept for the planned 1300 MW Fast Breeder Reactor SNR-2. As part of a preliminary study four possible concepts for a more detailed quantitative examination were selected. This paper describes part of the research dealing with the demands made on the containment system after a HCDA (Bethe-Tait-accident). As this accident very probably has the greatest hazard potential, it was considered to be of prime importance in designing the containment system. The primary enclosure will be designed in such a way that a disruption is impossible. Criteria are the requirements concerning the acceptable leakage of the containment enclosures, the ventilation and filter system, accessibility to containment areas after an accident, and radiological effects on the environment. It is impossible to consider these items independent from each other because the necessity to respect the radiological limits in the surrounding environment, for example, has a direct effect upon the required ability of the containment to restrain the activity. In order to do this, first the release factor of each radionuclide was estimated conservatively and then the transfer of the gaseous and the aerosol-type activity through the different compartments was calculated, and by use of these results the effect on the environment.

The calculations concerning the behaviour of the aerosols were made by the code Pardiseko IIIb, whose predecessor was reported about in [1, 2, 3].

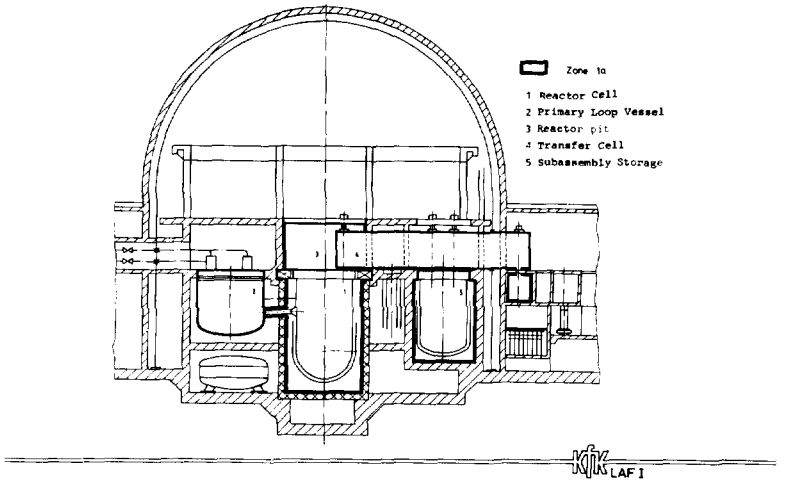
2. Containment concepts

The difference between the four containment concepts (fig. 1) lie only in the



Containment Concept SNR-2

Fig. 1



Reactor Building SNR-2

Fig. 2

outer containments. The concepts 1 and 2 are in principle developed from those of the KWU-LWRs and the KNK-II Fast Breeder Reactor which have already stood the test. The outer containment consists of a double containment. This has an outer enclosure consisting of reinforced concrete protecting against external impacts (e.g. airplance crash, sabotage etc.) and an inner enclosure made of steel which is relatively leakproof and is intended to hold the activity back during an accident. In the case of concept 1 the underpressure is kept in the ring gap (space III) and in the case of concept 2 in the outer containment (space II) which determines the path of the activity. From there the gas is discharged through filters into the environment. Concept 3 basically represents the concept of the SNR-300 which had to be examined again for purposes of comparison since its realization was proved to be possible during the licencing procedure for the SNR-300. The underpressure is kept in the ring gap by repumping the gas in the outer containment by a special blower (reventing system). After a period of several days the maximum pressure of the outer containment is reached caused by the several thermodynamic effects and the gas leakage from the environment into the gap. Then the gas has to be discharged from the outer containment through filters into the environment (exventing). Therefore there is a time with zero release after the accident /5/.

In the case of concept 4 the underpressure (always compared with the inner containment and the surrounding space) is kept in the outer containment where the gas is directly pumped through filters into the environment. The enclosure is made of reinforced concrete, so a relatively high leakage from the exterior has to be taken into account. The research of this concept allows a comparison of a single outer containment with the double outer containment concepts.

In all containment concepts the inner containment (space I) (Fig. 2) is the boundary of the inertized area and the inner barrier for the activity. It surrounds all areas which contain the primary sodium circuit like the reactor itself, the four primary circuits, the sodium-filled storage for irradiated elements, cleaning systems etc. It consists of the reactor enclosure and the four primary circuit containers and is made of steel. The environment of the containers is filled with air and is not leakproof. For a more detailed description see /4/.

3. The accident taken as a basis for the study

The design basic accident for the SNR-2 is not yet astablished, but in this containment study it was assumed that the design parameters of the containment system were derived from the effects of a Bethe-Tait-excursion accident (HCDA).
The reactor tank and the primary enclosure remain intact following the mechanical loading produced by the excursion energy. Decay heat from the damaged core is dissipated by means of the emergency cooling system and the primary heat removal chain.

The reaction zone contains liquid, solid and gaseous fuel, liquid and gaseous sodium as well as fission products. In this accident it was assumed that only at the beginning active nuclides and gaseous sodium are released to the inner containment following the pressure decrease. Then the released activity is transported by leakages (1_1) into the outer containment. To be conservative the release fractions of fuel and fission products were estimated to be high and those of sodium to be low. Being low in the case of sodium is therefore conservative as a high aerosol concentration causes a faster decay of the aerosol-type activity. Table 1 shows the estimated release fractions used as basic values for the calculations. They lead to an initial aerosol mass concentration in the inner containment of 132,4 g/m^3.

	Release Fraction [%]	Released Activity [Ci]	
Noble gases (Xe, Kr)	100	$1.8 \cdot 10^8$	
Halogens (J)	50	$3.9 \cdot 10^8$	
Volatile fission products (Cs, Te)	50	$7.1 \cdot 10^7$	
Solid fission products (Sr, Y, Zr, Ru, Ba, Ce)	5	$5.7 \cdot 10^7$	
Fuel (U, Pu), Transuranic Elements	5	$4.8 \cdot 10^6$	2500 kg
Sodium	500 kg	$1.9 \cdot 10^4$	

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Table 1 Release from the reactor tank after HCDA

Criterion / Cont. concept	Acceptable leakage* [Vol%/d]	Accessible cont. area	Filter load [g]
1	$l_2 \leq 4$ for $l_1 = 100$	–	≤ 200
2	$l_2 \leq 0.1$ for $l_1 = 100$	area 3	≤ 1000
3	$l_3 \leq 4$ for $0 < l_1 < 50$	area 3	<< 200
4	$l_2 < 0.1$ for $l_1 = 100$	–	≤ 15000

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Table 2 Pros and cons of the 4 containment systems

* Maximum external γ-dose: 5 rem

4. Transport of the activity and the aerosols through the containment

The aerosol mass concentrations in the different compartments were calculated by the code Pardiseko III b. This describes the same aerosol-physical effects as the program Pardiseko III /1/, /2/, /3/, this is the coagulation, sedimentation, diffusion and thermophoresis but it is different in their numerical treatment. Instead of a distribution function the particle size distribution is approximated by a number of monodisperse particle fractions. Thus it was possible to change the integro-differential equation system into a pure differential equation system. By these means the numerical problems of Pardiseko III could be removed and the running time could be reduced by a factor 10. The particle source for the inner containment is taken to be the instantaneous release of the fuel, the fission products and the primary sodium in the quantities given in chapter 3. The particles leaking from the inner containment are the source for the outer containment. To do this the leak rates for each particle class and at each time step are stored on a magnetic tape and read again for the calculation of the second compartment. It should be noted that it is not necessary to make restrictions regarding the particle size distribution for example the restriction on log-normal distributions.
Fig. 3 shows the dependence of the mass concentration in the outer containment on that in the inner containment with given leakage parameters.

The leak rates were varied over a wide range of values according to the following list:

leak rate of the inner containment l_1: 1 to 100 [vol %/d]

leak rate of the outer containment l_2

for concept 1 and 2 : 0.1 to 10 [vol %/d]

for concept 3 : 0.1 to 1 [vol %/d]

for concept 4 : 10 to 100 [vol %/d]

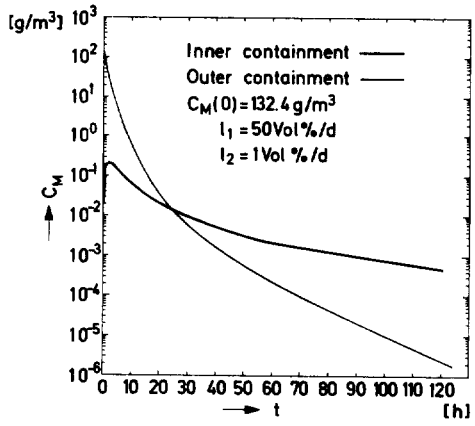
These average leak rates were estimated with thermodynamic considerations and the construction of the containment. In order to be conservative a filter factor was not taken into account during the passage of the aerosol through the leakages. To investigate the dependence of the mass released into the environment and the filter load on the initial concentration a second set of calculations was made using a value of 13.2 g/m^3 as initial mass concentration in the inner containment instead of the reference value of 132.4 g/m^3. As it can be seen in fig. 4 the dependence on the initial release is relatively weak compared with the dependence on the leak rates. The values calculated in the study generally show that a decrease of the initial release affects the filter load only by a factor of 2 to 3 but the effect of an increase of the leak rates is directly proportional to them.

Therefore, it can be deduced that the exact determination of the release from the reactor tank is less important for the aerosol-type activity than the quality of the containment enclosures and of the filter equipment.

The release of the gaseous activity was calculated by means of the estimated pressure course and the leakage parameters.

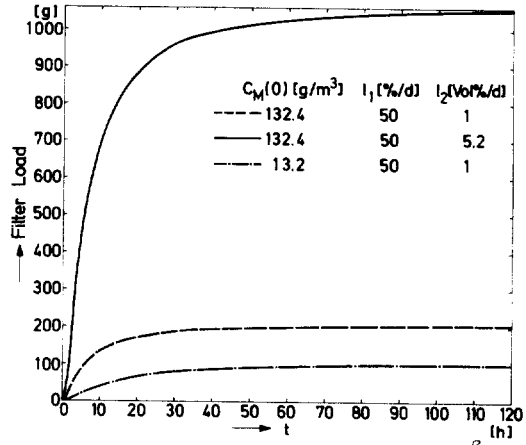
5. Radiological effects on the environment after the accident

The behaviour of the radioactive nuclides during their transport through the containment system can be described by a system of coupled first order differential



Aerosol mass concentration in the inner and the outer containment

Fig. 3



Aerosol Load of the Exhaust Air Filter Equipment

Fig. 4

equations. Its solution gives the time function of the activity load in the different compartments and the effluent of activity from the reactor equipment. For the aerosol-type activity the "plate-out" rates computed in the aerosol-physical calculations are used as input.

The dispersion parameters and the dose rates in the environment are calculated using the equations and statements in [6]. They refer to the points of maximum dose rate and the most unfavourable meteorological situation.

The results of the radiological analysis generally show that the dose for each organe integrated over the whole accident time is determined by the release of the gas type activity. The 2h-dose as well as the 24h-dose, however, are mainly influenced by the release of the aerosol-type activity (mainly Pu). The reason is that the aerosol physical decay processes cause an appreciable reduction of the dose made by the aerosol type activity. The filter efficiency for the aerosols is assumed as 99 % and for the noble gases as 0 %.

In nearly all parameter cases considered the external γ-submersion dose from the exhaust air cloud is the most critical dose.

The calculated accident doses have to be compared with the limit of 5 rem acceptable in view of the radiological protection regulation. Additionally the fact has to be taken into account that the calculated dose should be reasonably lower than the above mentioned limit. Therefore, it is possible to draw conclusions how leakproof the containment enclosures of the considered containment should be (see Fig. 5). The figure shows the 1-rem isodose line and the 5-rem isodose line depending on the leakage parameters of the inner (l_1) and the outer containment (l_2). It can be seen from the figure that containment concept 1 keeps the activity back best because of the two activity barriers. In the case of failure of the inner enclosure(l_1 large) the activity emission is much more limited by the outer enclosures in this concept than in the others. Fig. 6 shows a comparison of the accident dose as a function of the leakage of the inner containment for concept 1, 2 and 4. The concepts 2 and 4 react quite sensitively on an increase of the leakages of the inner containment. Therefore, concept 4 can only be realized if the leakage l_1 is less than some Vol. %/d. In this concept the relatively high inlet leakages badly influence the emission rate and therefore the period of time which the nuclides spend in area 2.

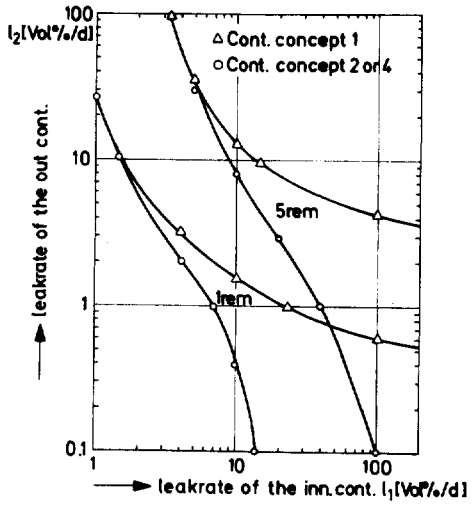
6. Assessment of the containment concepts

It should be emphasized that these assessments only relate to this study and that other problems (finances, engineering) are not considered.

A summarization of the pros and cons of the 4 containment concepts is shown in table 2.

Concepts 1 and 3 have the greatest advantages to keep the activity back, but concept 3 has the disadvantage of doing it by active components (the reventing blowers). The differences of the concepts concerning this point are described in details in chapter 5.

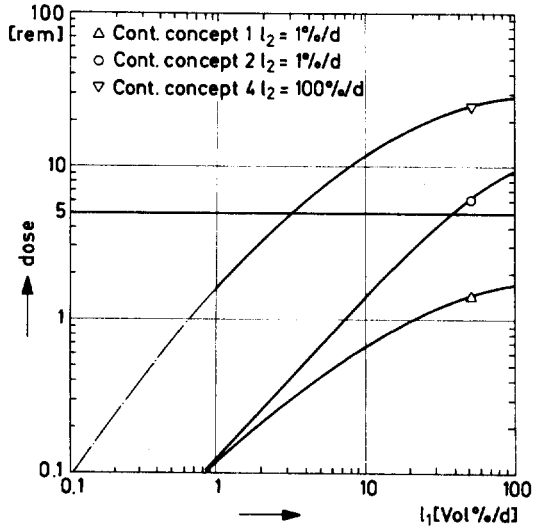
The accessibility to the different containment areas strongly depends on the ventilation system, therefore it is bad for concepts 1 and 4. Since the construction of concepts 1 and 2 is quite similar, it is possible to combine the two concepts to select the ventilation depending on the accident occuring and so to



KfK LAFI-8

Maximum external γ-dose dependant on the leakage parameters

Fig. 5



KfK LAFI

Maximum external γ- dose dependant on the leakage of the inner containment

Fig. 6

avoid the disadvantage of inaccessible areas in many cases.

Regarding the filter loads the disadvantage of concept 4 is obvious because it has a filter load of 15000 g, whereas all other concepts have filter loads less than or equal to 1000 g. This point is strongly correlated to the point of release of aerosol-type activity to the environment. The high filter load in concept 4 is caused by the high inlet leakage of the simple outer containment enclosure.

7. Conclusion

The results of the study show that a combination of the concepts 1 and 2 could be the basic concept:

- an inner enclosure for the inertized area
- an outer steel enclosure relatively leakproof
- a reinforced concrete enclosure surrounding the steel enclosure to protect against external impacts
- a ventilation system allowing the air to be led alternatively according to concept 1 (underpressure in the ring gap) or concept 2 (underpressure in the outer containment).

This concept keeps the activity sufficiently back even if greater leakages of the inner containment exist, and the filter load produced thereby can be disposed of without too much effort.

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FISSION 2120: A PROGRAM FOR ASSESSING THE NEED FOR ENGINEERED SAFETY FEATURE GRADE AIR CLEANING SYSTEMS IN POST ACCIDENT ENVIRONMENTS

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Abstract

A computer program FISSION 2120, has been developed to evaluate the need for various Engineered Safety Feature grade air cleaning systems to mitigate radiation exposures resulting from accidental releases of radioactivity. Those systems which are generally investigated include containment sprays with chemical additives, containment fan coolers with charcoal filters, and negative pressure maintenance systems for double barrier containments with either one-pass filtration or recirculation with filtration. The program can also be used to calculate the radiation doses to control room personnel. This type of analysis is directed towards the various protection aspects of the emergency ventilation system and involves the modeling of the radiological source terms and the atmospheric transport of the radioactive releases. The modeling is enhanced by the inherent capability of the program to accommodate simultaneous release of activity from several sources and to perform a dose evaluation for a wide range of the design characteristics of control room emergency air filtration systems.

Use of the program has resulted in considerable savings in the time required to perform such analyses and in the selection of the most cost-effective Engineered Safety Features.

I. Introduction

The licensing of a nuclear power plant requires an evaluation of the radiological consequences of a spectrum of postulated accidents. The spectrum should include an accident whose consequences are not exceeded by any other accident which is considered credible. A major loss-of-coolant accident (LOCA) is the most limiting credible occurrence which is presently considered. The results of such an evaluation provide a significant contribution to the selection of the design specifications for components and systems from the viewpoint of protecting the public health and safety. As an aid in the performance of safety analyses, the FISSION 2120 computer program has been written to evaluate the radiological consequences of the various postulated accidents. Using this program, one can investigate the need for various Engineered Safety Features (ESF) by analytically evaluating the effects on the plant, its associated personnel, and the public of the simultaneous failure of various components of the facility and some of its redundant safety systems.

The adequacy of the ESFs is measured by their capability to limit the release of radioactive materials into the environment to the extent that the radiation exposures to an individual at the plant

exclusion area boundary (EAB) distance and low population zone (LPZ) will not exceed the guidelines of 10CFR Part 100 [1]. In addition, the program is a useful tool in establishing the design requirements of the control room emergency air filtration system for conformance with the radiological exposure guidelines of General Design Criterion 19 of Appendix A to 10CFR Part 50 [2]. The most economical design, with respect to cost and necessity for periodic testing, may be determined by calculating doses to control room operators following a hypothetical Design Basis Accident as a function of filter efficiency, air handling capacity and other parameters.

II. The FISSION 2120 Radioactivity Releases and Dose Calculation Computer Program

FISSION 2120 was developed to model the release of airborne radionuclides from the containment and to calculate resulting doses to individuals at chosen locations. It was recognized that the program possessed the potential of being used as a design tool, particularly in establishing the specifications of dose mitigating ESFs relative to the atmospheric characteristics of a specific site. As a result, the capability to parametrically vary the ESF design variables was incorporated into the program enabling one to study the dose mitigating effects of various combinations of ESFs.

The parametric aspect is characteristic throughout the program and is especially useful in the following two areas:

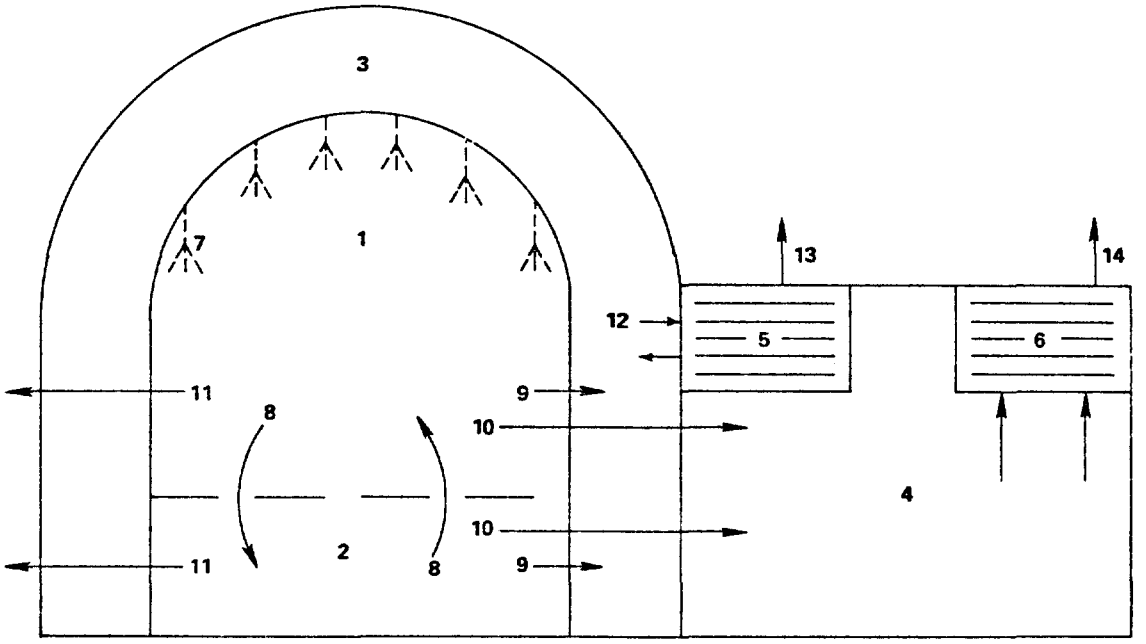
1. containment design characteristics, ESF requirements and site suitability studies;

2. control room emergency air filtration system design and operating modes.

Figure 1 is a diagram of the FISSION 2120 containment leakage model indicating the transport pathways of airborne radionuclides from one compartment to another and to the environment. Also presented in this figure are examples of the ESFs whose design variables can be parametrically studied using the program.

The model shown in Figure 1 is a dual-compartment containment leakage model. This model takes into account the partial coverage of the containment volume by the containment sprays and the mixing that takes place between the sprayed and unsprayed regions. One would normally use this model when taking credit for the iodine scrubbing action of certain chemicals added to the spray solution. In the case of pressurized water reactors, the volume of the containment below the operating deck contains compartments which are essentially excluded from the direct action of the sprays. Consequently, iodine scrubbing either does not take place or occurs at a much reduced rate in those situations where an air exchange exists between the sprayed and the obstructed lower zone.

FIGURE 1
CONTAINMENT LEAKAGE MODEL



1. CONTAINMENT SPRAYED VOLUME
2. CONTAINMENT UNSPRAYED VOLUME
3. SHIELD BUILDING ANNULUS
4. NEGATIVE PRESSURE AREA OF THE REACTOR AUXILIARY BUILDING
5. SHIELD BUILDING VENTILATION SYSTEM
6. CONTROLLED VENTILATION AREA SYSTEM
7. CONTAINMENT SPRAYS
8. AIR EXCHANGE BETWEEN REGIONS 1 AND 2
9. LEAKAGE INTO THE SHIELD BUILDING ANNULUS
10. LEAKAGE INTO THE NEGATIVE PRESSURE AREA OF THE REACTOR AUXILIARY BUILDING
11. BYPASS LEAKAGE
12. SHIELD BUILDING VENTILATION SYSTEM RECIRCULATION
13. SHIELD BUILDING VENTILATION SYSTEM RELEASE
14. CONTROLLED VENTILATION AREA SYSTEM RELEASE

Activity Transport and Dose Calculation Equations

The transport of activity from the containment to the environment and the control room is represented by a set of equations which describe the conservation of activity within the various compartments.

The change in activity with respect to time in the sprayed and unsprayed regions of the containment, respectively, is given by:

$$\frac{dA_s}{dt} = -\left(\lambda + R_s + L_1 + \frac{E}{V_s}\right)A_s + \frac{E}{V_u}A_u \qquad (1)$$

$$\frac{dA_u}{dt} = -\left(\lambda + R_u + L_1 + \frac{E}{V_u}\right)A_u + \frac{E}{V_s}A_s \qquad (2)$$

where:

A_s = activity in sprayed region of containment (Ci)

A_u = activity in unsprayed region of containment (Ci)

t = time (sec)

λ = radioactive decay constant (sec^{-1})

R_s, R_u = first order removal coefficient of the ESFs inside the sprayed and unsprayed regions of the containment (sec^{-1})

L_1 = primary containment leak rate coefficient (sec^{-1})

E = air exchange rate between sprayed and unsprayed region (m^3/sec)

V_s, V_u = volumes of sprayed and unsprayed regions (m^3).

For a secondary enclosure (e.g. the shield building annulus), the time rate of change of activity is expressed as:

$$\frac{dA_2}{dt} = L_1(A_s + A_u) - (\lambda + L_2 + R_2)A_2 \qquad (3)$$

where:

A_2 = activity in secondary containment (Ci)

R_2 = first order removal coefficient of the shield building annulus recirculation system (sec^{-1})

L_2 = secondary containment vent rate coefficient (sec^{-1}).

The activity, Q_2 (Ci), which is released from the secondary enclosure to the environs is given by:

$$Q_2(t) = L_2 \int_0^t A_2(t')\ dt' \qquad (4)$$

The calculation of this quantity permits the calculation of the offsite integrated concentration, or exposure, given by:

$$\psi_2 = Q_2 \left(\frac{X}{Q}\right)_2 \qquad (5)$$

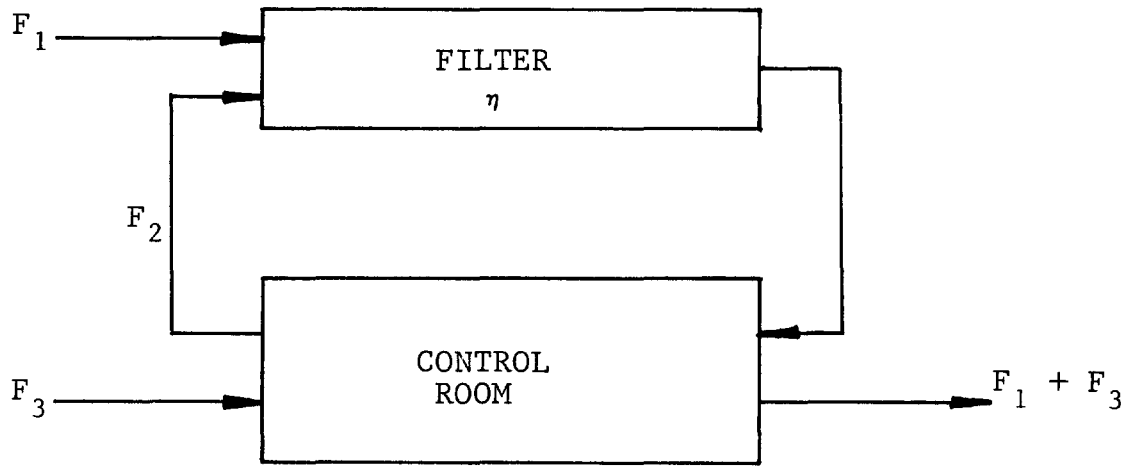
where:

ψ_2 = offsite exposure ($Ci\text{-}sec/m^3$)

$\left(\frac{X}{Q}\right)_2$ = offsite atmospheric dispersion factor (sec/m^3).

Figure 2 shows the model which is used in the dose analysis of a control room emergency air filtration system [3]. The model is general in nature and can represent a control room in either the pressurized or totally isolated mode.

FIGURE 2
CONTROL ROOM EMERGENCY FILTRATION SYSTEM



F_1 = rate of filtered outside air intake

F_2 = rate of filtered air recirculation

F_3 = rate of unfiltered outside air infiltration

η = filter efficiency

The activity in the control room is a function of the activity released from the containment, the atmospheric dispersion that takes place between the source and the point of intake, the intake rate of air into the control room, and the degree of air filtration provided.

The change of activity in the control room is given by the following equation:

$$\frac{dA_3}{dt} = L_2A_2\left[\left(\frac{X}{Q}\right)_1 F_1(1-\eta) + \left(\frac{X}{Q}\right)_3 F_3\right] - \left[\lambda + \frac{F_1 + (1-\eta)F_2 + F_3}{V}\right]A_3 \quad (6)$$

where:

A_3 = activity in the control room emergency HVAC envelope (Ci)

L_2A_2 = the rate of release of activity into the environment from the secondary containment (Ci/sec)

F_1 = rate of filtered outside air intake (m^3/sec)

F_2 = rate of filtered air recirculation (m^3/sec)

F_3 = rate of unfiltered outside air infiltration (m^3/sec)

V = volume of control room emergency HVAC envelope (m^3)

λ = radioactive decay constant (sec^{-1})

η = efficiency of the control room emergency filtration system charcoal adsorber

$\left(\frac{X}{Q}\right)_1, \left(\frac{X}{Q}\right)_3$ = atmospheric dispersion factors representative, respectively, of dilution between the activity release point and main control room air intake and point of unfiltered inleakage (sec/m^3).

The integrated activity concentration in the control room atmosphere, ψ_3 ($Ci\text{-}sec/m^3$), is given by:

$$\psi_3 = \frac{1}{V} \int_0^t A_3(t')\, dt' \qquad (7)$$

The exposure is used to calculate the skin, thyroid and whole body doses, in rem, to offsite receptors and control room personnel according to the following equations:

$$\begin{aligned} \text{Skin Dose} &= SDCF \cdot \psi \cdot K \\ \text{Thyroid Dose} &= B \cdot TDCF \cdot \psi \cdot K \\ \text{Whole Body Dose} &= WDCF \cdot \psi \cdot K \end{aligned} \qquad (8)$$

where:

SDCF = skin dose conversion factor ($rem/sec/Ci/m^3$)

TDCF = thyroid dose conversion factor (rem/Ci-inhaled)

WDCF = whole body dose conversion factor ($rem/sec/Ci/m^3$)

B = breathing rate (m^3/sec)

K = a modifying factor which accounts for the effects of in-transit decay, depletion and effects of finite geometry, as applicable.

The analytical solutions to the above equations are incorporated into the FISSION 2120 program. Input of problem dependent parameters such as filter efficiency and X/Q permits the calculation of offsite doses for up to eight time intervals following the start of the accident.

The FISSION 2120 program is written in FORTRAN IV, requires 18 K words of core storage space and, depending on the extent of the problem, runs about 18 seconds on a Burroughs B7760 computer.

III. Containment Design Characteristics and Plant Requirements in Site Suitability Studies

Offsite doses resulting from a large LOCA are the major factors which are considered in any containment design. 10CFR Part 100 specifies the dose limits as 300 rem to the thyroid from iodine exposure and 25 rem immersion dose to the whole body. However, at the construction permit stage of licensing, the Nuclear Regulatory Commission (NRC) guideline values are 150 and 20 rem, respectively [4,5]. These exposures are calculated for an individual located at any point on an exclusion area boundary (EAB) for two hours immediately following the onset of the postulated fission product release, and on the outer boundary of a low population zone (LPZ) during the entire period of the radioactive cloud passage.

In general, the thyroid dose received at the EAB in the first two-hour period following the accident has been observed to be the controlling factor when using the activity release fractions specified by the NRC [4,5]. An examination of the containment leakage model and the dose calculation equations reveals that the thyroid dose is a function of several factors, which in turn are related to the containment, the type of ESFs and the site. These are:

1. containment design leak rate;
2. overall iodine decontamination factor (IDF);
3. atmospheric dispersion factor, X/Q.

In these calculations the offsite thyroid dose is directly proportional to the containment leak rate; with leak rates ranging from 0.1% to 0.5% of the containment volume per day currently being used.

The IDF as used here is defined as the ratio of the integrated releases of a plant without dose reduction features to those of the same plant including such features and, hence, is a function of the type, the size and the effectiveness of the ESFs provided for iodine removal. It is in the development of the IDFs that the FISSION 2120 program proves to be an effective practical design tool. The dose reduction features may consist of internal containment sprays with additives, filter removal systems, external filter system used in conjunction with a double barrier containment or, as is usually the case, a combination of these features. Table I lists the IDFs for various combinations of these ESFs. In the development of these values, the "base" case (with the IDF equal to 1) has been taken as a plant with a single barrier containment, with a design leak rate of 0.5% per day, and without any ESFs.

The atmospheric dispersion factor, X/Q, is a measure of the dilution of the released fission products at a distance from the point of emission. It is dependent on site meteorology and distance and, in general, its value decreases as the distance increases. The offsite doses are directly proportional to the X/Q value.

TABLE I

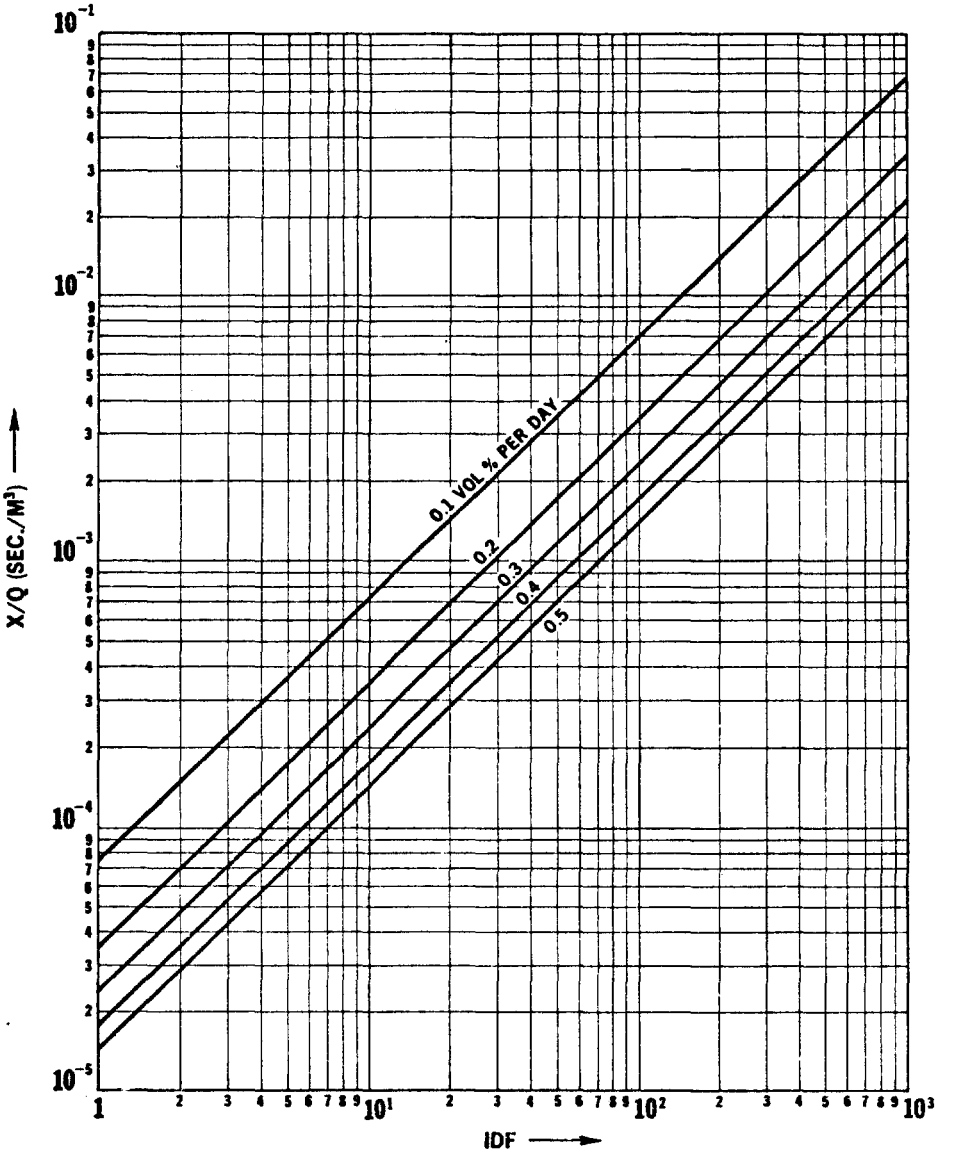
Iodine Removal Effectiveness of Various Engineered Safety Features

Engineered Safety Features	Iodine Decontamination Factor
1. Single barrier containment without iodine spray removal system.	1.0
2. Single barrier containment with iodine spray removal system.	7.0
3. Double barrier containment with external once-through filtration system with 95% efficiency charcoal adsorber.	21.0
4. Double barrier containment with external once-through filtration system with 99% efficiency charcoal adsorbers.	100.0
5. Double barrier containment with iodine spray removal system and external once-through filtration system with 95% efficiency charcoal adsorbers.	158.0
6. Double barrier containment with iodine spray removal system and external once-through filtration system with 99% efficiency charcoal adsorbers.	658.0

Figure 3 is a plot of the IDFs versus X/Q values for a family of 150 rem thyroid isodose curves evaluated for containment leak rates varied over the range of 0.1% to 0.5% per day. Ideally, the development of such a graph should be undertaken as soon as reliable onsite meteorological data are available and should precede, or, at lease coincide with, the preliminary design phases of the nuclear plant. Knowledge of site meteorological characteristics could, through the use of a graph similar to Figure 3, help determine the distance to the EAB, the LPZ and the corresponding plant ESFs required to ensure that the LOCA doses will be below the limits established by the NRC. Using such an approach, economic studies can be performed whereby the cost of installing progressively more expensive ESFs would be measured against that of acquiring additional land.

The results presented in Figure 3 can also provide the basis for setting the technical specifications for the containment leak rate. Examination of Figure 3 will show that choosing a design basis containment leak rate of 0.1% per day will result in either the installation of a minimum number of ESFs and/or a shorter EAB. For example, the choice of the smallest leak rate of 0.1% per day at a site with an EAB X/Q equal to 5×10^{-4} sec/m^3 would require a combination of Engineered Safety Features with an IDF of about 7. Table I indicates that this can be achieved by a single barrier containment with an iodine spray removal system. However, one should be aware that the choice of such a small leak rate may introduce problems in engineering, construction, plant operation and maintenance.

FIGURE 3
X/Q AS A FUNCTION OF IDF FOR A 150 REM THYROID DOSE LEVEL



IV. Emergency Air Filtration Design Requirements For Control Room Habitability

The protection requirements of control room personnel against radiation exposure under accident conditions are specified in General Design Criterion 19 (GDC 19) of Appendix A to 10CFR 50 [2]. As stated in GDC 19, "a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents." Consequently, the evaluation of the performance requirements of a protection system for a control room is based on the same LOCA source term which is used in the assessment of compliance with 10CFR Part 100 dose criteria. Therefore, the control room activity and dose model is a natural extension of the containment leakage model. The evaluation of compliance with GDC 19 is generally directed toward the emergency ventilation system and involves modeling of the radiological source term, the atmospheric transport of the radionuclides released, along with the various protection aspects of the control room ventilation system.

The FISSION 2120 control room model is similar to that of Murphy and Campe [3] and Byoun and Conway [6]. The major difference is that this model does not assume the existence of equilibrium conditions. The calculation of doses is based on the exact solutions of the mass balance equations. Consequently, the dose mitigating effect of slow build-up of activity in an isolated control room followed by a rapid purge, is automatically accounted for without having to resort to the calculation of a purge factor [3].

The control room part of the program is very versatile due to the fact that virtually all the necessary input data can be treated parametrically. For example, a control room may be designed to operate in either a pressurized or an isolated mode under accident conditions. However, in some cases, pressurized control rooms actually go into an isolated mode initially, followed by a time lag, which may be significant, before attaining proper pressurization. This case can be analyzed to provide information relative to the selection of a manual or automatic system and operational flexibility within the control room. An associated aspect is the control room emergency system filters. The filter efficiency can be treated parametrically and be turned "ON/OFF" over specified time intervals to account for time lags required to get the emergency system operational or to attain the specified pressurization. Similarly, the filtered air intake rate for the pressurized mode, the unfiltered inleakage and the emergency filtration system recirculation capacity can be treated as parametric variables.

The program can accommodate specific atmospheric dispersion factors (X/Q) for both the filtered air intake and unfiltered inleakage, as well as X/Qs for simultaneous multi-point releases. This is done by assigning a pair of X/Qs to each source of activity release; one for filtered air intake and one for unfiltered inleakage into the control room. Points of release which can be considered include filtered releases from the Standby Gas Treatment System, unfiltered

containment bypass leakage as well as releases from the Emergency Core Cooling System equipment rooms kept at negative pressure. The X/Qs may be evaluated according to the models of Murphy and Campe[3]. By considering various combinations of release to intake X/Qs one can optimize the location of control room emergency outside air intakes with respect to minimizing potential personnel doses.

Another aspect of design which can be investigated by this program is the choice between automatic and manual isolation of a control room following a LOCA. This choice is particularly significant because the NRC is currently permitting large reduction in the dispersion factors for well separated dual air intakes, and manual or automatic isolation [7]. These reduction factors apply to the determination of the five percentile X/Q for dual inlets located on Seismic Category I structures, for which the current NRC practice is as follows:

1. For well separated dual fresh air intakes the X/Q is obtained from Eq. 6 in Ref. (3) for the intake closest to the point of release and then reduced by a factor of 2.

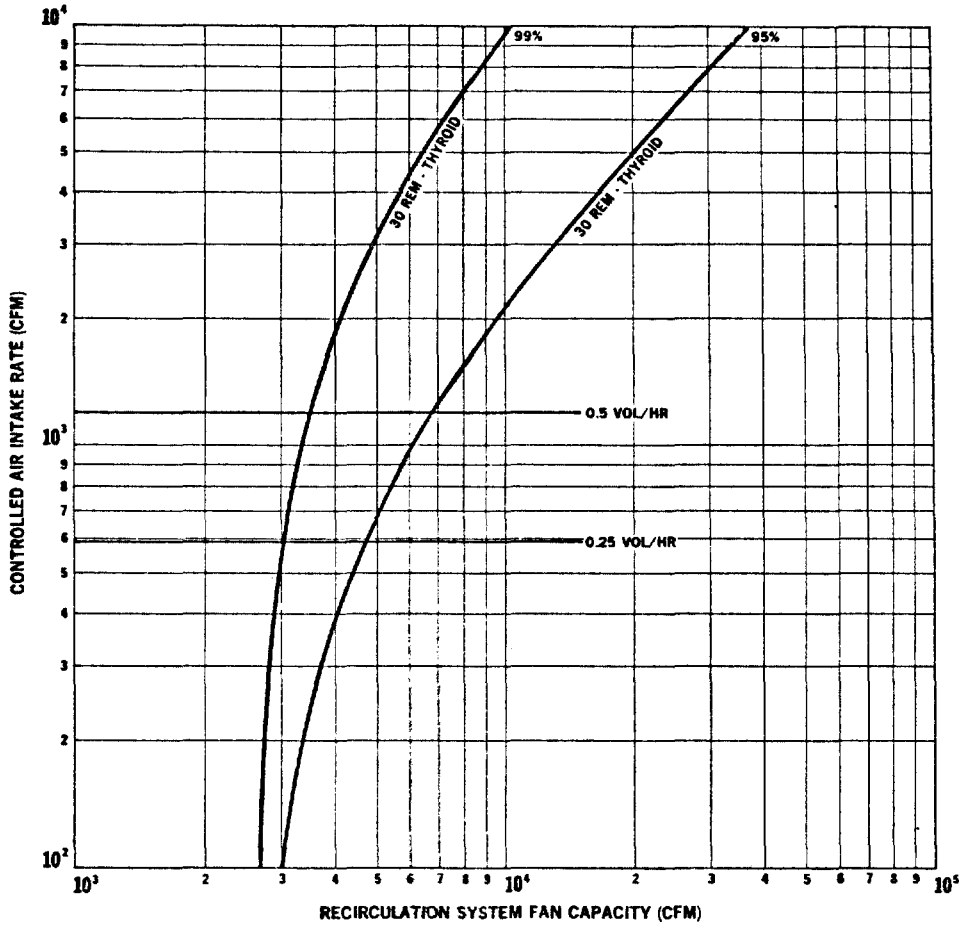
2. If active manual control over source of makeup air has been provided by installing radiation monitors at the inlets and by training control room operators to switch to the cleaner inlets, the X/Q is obtained from Eq. 6 in Ref. (3) for the farthest air inlet and then reduced by a factor of 4.

3. If active automatic control over source of makeup air has been provided by means of radiation monitors and control logic, the X/Q for the farthest air inlet is reduced by a factor of 10.

The reductions in the values of the X/Qs lead to corresponding reductions in calculated doses to control room operators.

The results of the control room dose evaluation, generally presented graphically, define the ventilation system requirements in the form of an "envelope". In this manner the designer can readily select system parameters with assurance that the resulting design would be acceptable from a radiological point of view and also be aware of the testing requirements being imposed. Figure 4 is a graphical representation of the curves that form the "envelopes" for acceptable combinations of the major design parameters, (filter efficiency, makeup air intake rate and filtered air recirculation rate) of a pressurized control room emergency air filtration system. Regions of Figure 4 lying to the right of and below the isodose curves indicate values of air makeup rate and recirculation system fan capacity at which the radiological protection criteria of GDC 19 would not be exceeded. The 0.5 volume/hour and 0.25 volume/hour horizontal lines indicate regions of different requirements for verifying control room leak tightness, as specified in Standard Review Plan 6.4, [8] and provide an added constraint on the sizing of the emergency air filtration system.

FIGURE 4
30 DAY 30 REM THYROID ISODOSE CURVE



V. Conclusion

Use of the FISSION 2120 program has resulted in considerable savings in the time and cost required to perform accident analyses, and in the selection of the most cost effective Engineered Safety Features. This program has proven to be a useful tool for the radiological assessment engineer in advising the designer, the safety and licensing engineer and the environmental engineer through the various stages of the plant design.

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DISCUSSION

MOELLER: Does your program provide a quantitative estimate of the source term required to yield the 150 rem thyroid dose limit?

MICHLEWICZ: The program can be used to calculate the source term. However, the source term would depend on containment design and site conditions.

COLLINS: In considering cost did you factor in operating and maintenance cost or only capital cost?

MICHLEWICZ: We considered both costs. For example, in the design of a control room ventilation system, the costs of periodic testing of that system are considered in the selection of the system parameters.

IMPACT OF SOPHISTICATED FOG SPRAY MODELS ON ACCIDENT ANALYSES

by

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Abstract

In the analysis of postulated accidents involving a primary coolant pipe break and subsequent release of fission products, the detailed mechanisms of plateout, washout by fog sprays and filtering are evaluated as a means of removing fission products from the atmosphere inside the reactor building. The N Reactor confinement system release dose to the public in a postulated accident is reduced by washing the confinement atmosphere with fog sprays. This allows a low pressure release of confinement atmosphere containing fission products through filters and out an elevated stack.

The current accident analysis required revision of the CORRAL code and other codes such as CONTEMPT to properly model the N Reactor confinement into a system of multiple fog-sprayed compartments. In revising these codes, more sophisticated models for the fog sprays and iodine plateout were incorporated to remove some of the conservatism of steam condensing rate, fission product washout and iodine plateout than used in previous studies.

The CORRAL code, which was used to describe the transport and deposition of airborne fission products in LWR containment systems for the Rasmussen Study, was revised to describe fog spray removal of molecular iodine (I_2) and particulates in multiple compartments for sprays having individual characteristics of on-off times, flow rates, fall heights, and drop sizes in changing containment atmospheres. During postulated accidents, the code determined the fission product removal rates internally rather than from input decontamination factors. A discussion is given below of how the calculated plateout and washout rates vary with time throughout the analysis. The results of the accident analyses indicated that more credit could be given to fission product washout and plateout. An important finding was that the release of fission products to the atmosphere and adsorption of fission products on the filters was significantly lower than previous studies had indicated.

I. Introduction/Summary

The CORRAL code was chosen to be used in the accident analyses for N Reactor. This decision was made because the capabilities of the code were suitable for modeling the complex confinement system of N Reactor. The N Reactor confinement system has a low pressure design which will relieve pressure from major pipe accidents during the initial pressure transient.

The confinement building is categorized as four compartment volumes which are interconnected by ventilation flow paths. Each compartment of the confinement building has its unique volume, wall and floor area and temperatures, vapor pressures and pressure transients during the course of the accident. In three of the compartments, 11 unique fog sprays are described. Intercompartmental flow rates and flow rates through a filtered release and building leakage are specified as a function of time. It was therefore clear in considering the complexity of the N Reactor confinement system during postulated accidents that a sophisticated model of the confinement system was needed to properly analyze the postulated accidents.

CORRAL is a complex computer model (developed for the Rasmussen Reactor Safety Study (WASH-1400)) of light water reactor containment space that describes the transport, deposition and leakage of airborne fission products during postulated accidents. CORRAL handles the airborne fission products in four basic models based on their physical and chemical properties. These groups are particulates, elemental iodine, methyl iodine and noble gases. Natural deposition of particulates and I_2 as well as spray washout are modeled to apply to each individual (of N total) compartments' unique features and existing thermodynamic states. The compartments are tied together via cross flows. In the compartments the code is used to account for the mass balance of each fission product group. The resulting 3N time dependent differential mass balances are solved exactly over a specified time interval. Complete inventories of these fission products are computed for each compartment and reprinted as fractions: remaining airborne, deposited and/or washed out, present on filters and lost to the environment by ground or stack level leaks.

The CORRAL code was successfully applied to the fuel ejection accident described in this paper, and other accidents considered in the N Reactor safety analysis report. The code accounted for the distribution, depletion and release to the environs of the above four fission product groups as a function of time. The release fractions of these fission product groups were used as the basis to calculate the release of fission products and the accident doses.

II. Airborne Fission Product Depletion Mechanisms and Their Models in a LWR Containment System

The rates of processes controlling the depletion of airborne fission products depends on the physical and chemical properties of the individual species. These properties allow grouping all important products into four major categories:

- elemental iodine (as a vapor)
- methyl (organic) iodide
- noble gases
- aerosol particles

Detailed description of the models and mechanisms are described in Reference (1) of the Rasmussen Report. Summaries of the mechanisms and models follow.

II.1 Noble Gases

Although means for collecting noble gases have been proposed, none have yet been implemented in LWR containment systems, so no removal terms were added to the differential equations for the material balances of noble gases.

II.2 Methyl (organic) Iodide

The airborne alkyl halides that can form following a postulated accident are relatively unreactive. Special treated activated charcoal filters, thiosulphate sprays, and special reactive paints can remove these species at rates faster than the very slow water hydrolysis reaction. None of these were present in systems analyzed for N Reactor, so methyl iodide was treated as a noble gas in the present CORRAL calculations. CORRAL is easily amenable to organic halide depletion mechanisms if desired.

II.3 Elemental Iodine

Airborne elemental iodine (I_2) can exist in both vapor and particulate forms. The quantity in the vapor phase usually predominates and is discussed below. Particualate iodine becomes part of the composition of the solid aerosol particles and is treated as such.

Two mechanisms exist for the depletion of elemental iodine: one is referred to as natural deposition or "plateout" and the other is spray washout. Natural deposition can occur on nearly all exposed surfaces below a certain temperature (250°C). The rate of deposition is limited by diffussion through a gas boundary layer along the surface of the containment system. A preferred model for boundary layer resistance involves natural convection that is driven by the temperature difference between containment gas phase and walls. The natural deposition removal constant $\lambda(hr^{-1})$ for a given compartment is thus

$$\lambda_{ND} = \frac{k_g A}{V} = \frac{A}{V} \frac{D_g}{L} a \, (Gr \; Sc)^b \qquad (1)$$

where k_g = gas phase mass transfer coefficient

V, A = compartment volume and surface area

D_g = iodine diffusivity in the gas phase

L = vertical height along convective boundary

Gr = Grashov number for heat transfer

Sc = Schmidt number for mass transfer

a, b = constants ($a = 0.13$, $b = 1/3$ for Laminar $Gr < 1.5 \times 10^8$ and $a = 0.59$, $b = 1/4$ for turbulent $Gr > 10^{10}$)

Spray removal of I_2 can be accomplished much faster than natural deposition. A conservative model for spray removal employs a stagnant gas film around the drop and a well mixed drop interior. The resulting spray lambda is

$$\lambda_s = \frac{FH}{V}\left\{1-\exp\left\{\frac{-6\ k_g\ t_e}{d\left(H + \frac{k_g}{k_e}\right)}\right\}\right\} \qquad (2)$$

where F = spray flow rate

H = equilibrium partition coefficient

t_e = drop terminal falling velocity

d = drop diameter

k_g = gas phase mass transfer coefficient

$$= \frac{D_g}{d}\left(2 + 0.6\ Re^{0.6}\ Sc^{0.33}\right)$$

Re = drop Reynolds number

k_l = liquid phase mass transfer coefficient

$= 2\pi^2\ D_g/3d$

Limitations on this spray drop model are discussed by Postma and Pasedag (2). The equilibrium partition coefficient, H, has the following applicable values (2):

- for neutral to pH = 5 (Boric acid), H = 200
- for caustic, pH = 9, H = 5000
- for basic sodium thiosulphate, $H = 10^5$

Equation (2) holds until iodine concentrations are reduced to about 1% of the initial airborne amount. After this, the iodine removal rate is independent of the spray volume rate and a pseudo-equilibrium sets in where H becomes dependent on time and a positive dH/dt leads to a slower rate of scrubbing. Each spray solution has its own pseudoequilibrium pattern. CORRAL is presently programmed for three of these: pH = 5-7, pH = 9, and no spray.

II.4 Particulates

Airborne particulates also have two removal mechanisms: natural settling on all up-facing horizontal surfaces and spray removal. Natural settling from a well mixed compartment has the removal coefficient

$$\lambda_p = U_t A_h / V \qquad (3)$$

where U_t is the terminal particle velocity, A_h = horizontal area component of all up-facing surfaces in compartment of volume V. CORRAL is programmed to determine a U_t given an initial particle size at the moment the particulates are airborne; then CORRAL linearly interpolates between that particle size and another input size at some later time. Based on CSE data (1)* the usual inputs were: initial size = 15×10^{-6} meter, final size = 5×10^{-6} meter beginning four hours later.

Spray removal of particulates is modeled using non-interacting drops that sweep out particles in their falling paths with efficiency E. This model gives

$$\lambda_{sp} = \frac{3h\ F\ E}{2\ dV} \qquad (4)$$

where h = spray fall height. The efficiency E monotonically decreases with an increasing value in the integrated dimensionless spray volume Ft/V. The CSE experiments determined an initial E = 0.06 which became 0.0015 for $Ft/V > 0.02$. This spray aging is programmed into CORRAL for each possible spray group.

III. Quantitative Comparison of Mechanisms in a Single Compartment

To show the relative effects of the various depletion mechanisms using the models discussed above, a single compartment was chosen for this case study. This compartment has the following typical PWR parameters: 10^6 ft^3 volume, 50 ft height, 2×10^4 effective floor area, 10^6 ft^2 total surface area. This compartment has one spray type: 0.1 cm diam drops or 0.0465 cm, 50 ft fall height, 8×10^5 ft^3/hr liquid flow rate,

* CSE: Containment Systems Experiments - Several references to work done here are listed in Reference (1).

and spray liquid is water, pH = 5-7 (H= 200). The compartment conditions following the spike release of airborne fission products are: p = 1 atm absolute; T = 180°F initially changing to 75° after one hour; atmosphere saturated with steam at the compartment pressure and temperature.

III.1 Particulate Behavior

Figure 1 is a plot of $\log_{10}$ of the fraction of the initial particulate aerosol cloud that remains airborne in the compartment versus time. No leaks or radioactive decay were assumed. Each of the six curves represents a different removal in the basic depletion equation

$$\frac{d \ \ln C/C_o}{dt} = -\lambda \qquad (3)$$

where C/C_o is the fraction of particulates remaining airborne.

Curve A represents C/C_o for the single removal mechanism of natural settling. Here the particles were 15 x 10^{-6} meters initially changing with time to 5 x 10^{-6} meters after four hours. Here $\lambda \approx 1.5$ hr^{-1}. The nonlinearity of $\log_{10}$ C/C_o reflects the influence of this changing particle size and changing compartment conditions on λ.

Curves B-D show how an internal compartment filter could compare to natural settling. These curves represent 100% efficient filtering of particles through filters at flow volumes of one, two and three compartment volumes per hour (or 17,000, 33,000 and 50,000 ACFM).

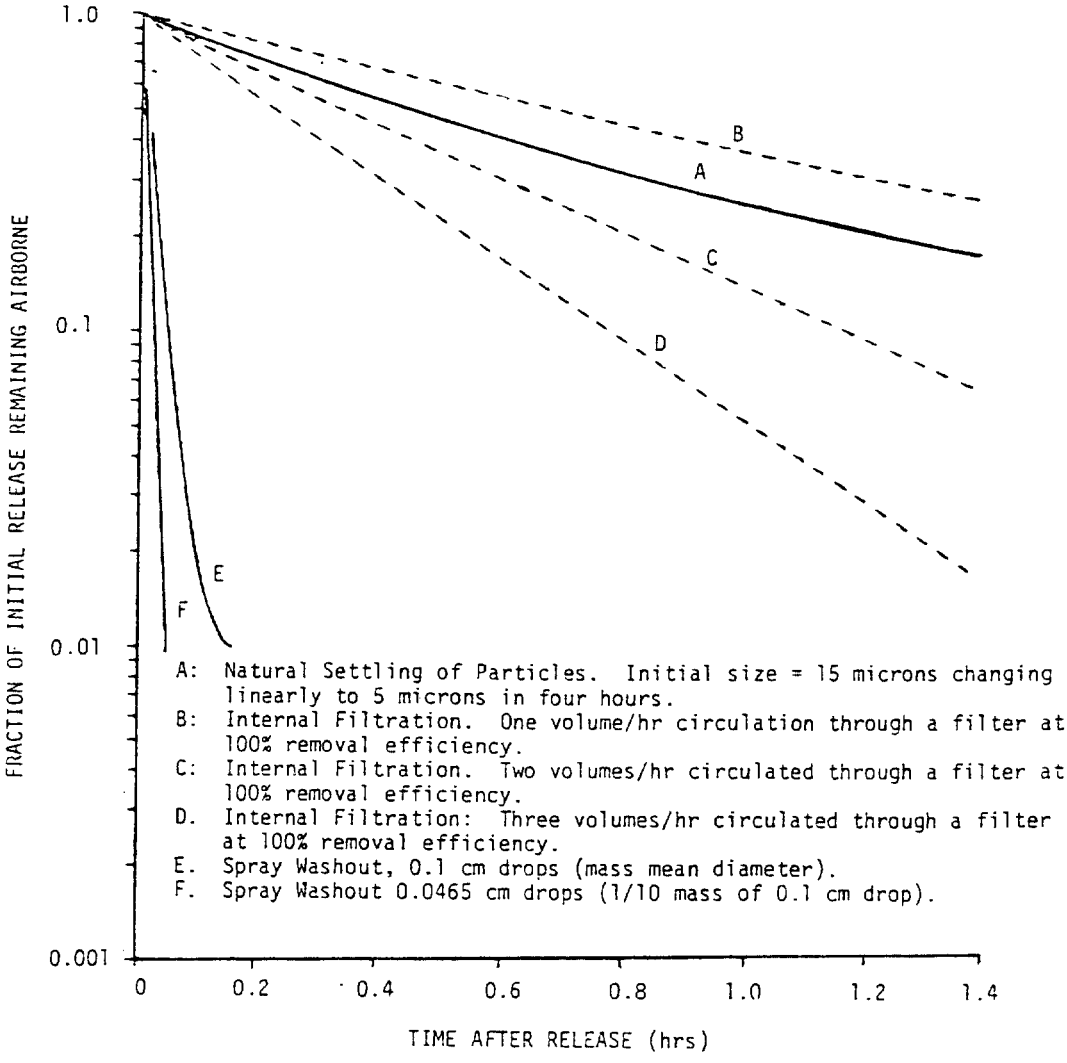
Curves E and F compare two drop sizes, 0.1 cm and 0.0465 cm (1/10 mass of 0.1 cm drop). Here the removal λ's are about 40 and 70 hr^{-1}. The curves terminate at C/C_o = 0.01 reflecting an observed zero washout rate at that point in the CSE experiments. Equation (4) shows the direct influence of drop size on λ and this is apparent on the observed λ's. The nonlinearity of the curves (F does not show any due to only one computed C/Co) is due to the decreasing scrubbing efficiency in Figure 4 as sprays continue as well as the changing compartment conditions. These conditions would primarily affect the terminal drop velocity through the gas viscosity changes with compartment temperature.

For combined mechanisms in the compartment, one needs to simply add the various λ's. If these are constant over a time interval, t_1 to t_2, the differential equation solution is

$$C_{t_2}/C_{t_1} = \exp \ (-\Sigma\lambda)(t_2 - t_1) \qquad (5)$$

However, if one needs to consider multiple mechanisms that change with time, the use of CORRAL for even a single compartment becomes advantageous for the accuracy delivered.

Single Compartment Behavior of Particles Showing the Effects of Three Depletion Mechanisms - Natural Settling, Spray Washout and Internal Filtration



A: Natural Settling of Particles. Initial size = 15 microns changing linearly to 5 microns in four hours.
B: Internal Filtration. One volume/hr circulation through a filter at 100% removal efficiency.
C: Internal Filtration. Two volumes/hr circulated through a filter at 100% removal efficiency.
D. Internal Filtration: Three volumes/hr circulated through a filter at 100% removal efficiency.
E. Spray Washout, 0.1 cm drops (mass mean diameter).
F. Spray Washout 0.0465 cm drops (1/10 mass of 0.1 cm drop).

Figure 1

III.2 Elemental Iodine Behavior

Figure 2 is a plot similar to Figure 1 but for I_2 behavior in a single compartment. Curves A and B are $\log_{10}C/C_o$ versus time for natural deposition. The driving force for depletion is the bulk gas wall temperature difference and is modeled by equation (1). Here the typical initial temperature differences were 60^oF and 30^oF respectively. These differences changed linearly to 0.1^oF at one hour and remained there. Since this λ is proportional to the temperature difference to the 1/3 or 1/4 power, the difference between curves A and B is small. The average λ was generally of the order of one hr^{-1} for the first hour and gradually became less after that. Curve C shows that internal filtration at 1.2 hr^{-1} can equal typical natural deposition processes.

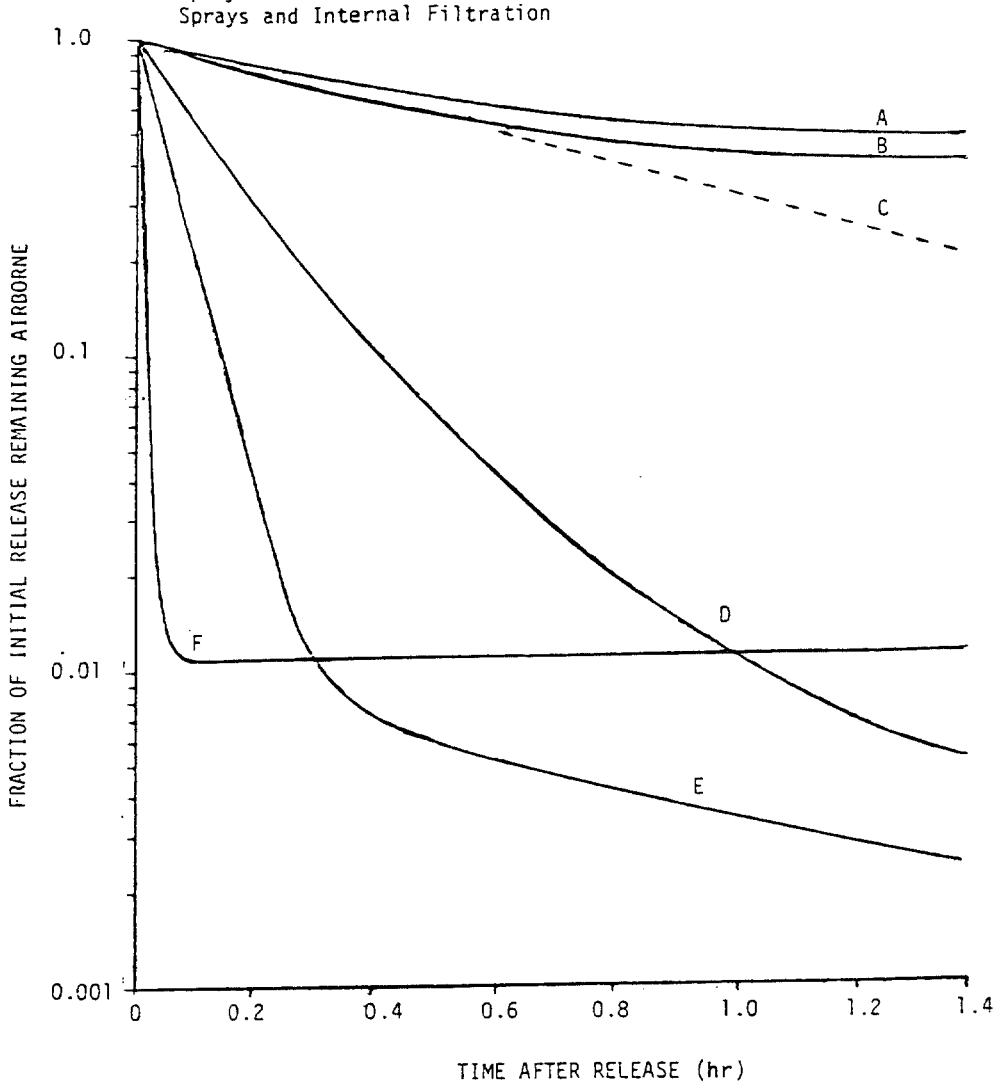
Curves D and E compare the pH 5-7 (H = 200) spray removal mechanism using equation (2) for two drop sizes: 0.1 cm and .0465 cm. Here the drop diameter affects the λ in the exponent (Equation (2)) and the λ's are about 5 and 18 respectively over the first hour. For $C/C_o > 0.01$ the nonlinearity in curves D and E is due soley to changing compartment conditions. For $C/C_o < .01$, the pseudoequilibrium conditions hold and this removal mechanism in these cases resulted in a fairly smooth transition at the cutoff point ($C/C_o = .01$)

Curve F is like D except caustic (pH = 9) replaced the pH 5-7 spray. Here the resulting $\lambda = 125\ hr^{-1}$ until the cutoff $C/C_o = .01$. At that point the removal halts for 100 minutes and then the pseudoequilibrium condition begins to remove I_2 slowly. The net effect is that the neutral and caustic sprays produce identical C/C_o conditions at one hour via much different paths. This can greatly affect net fission product losses, depending on when the leakage occurs to the environment.

Again, as with sprayed particles, combined mechanisms can be handled by just adding component λ's together. In the case of combined natural deposition and sprays, it can be argued that spraying negates the other. This would seem to be so if the sprays destroyed the temperature difference driving force as well as convective flow patterns. In the comparison here, any natural deposition can be neglected in terms of the overwhelming spray washout.

The use of CORRAL for I_2 washout can easily give more accurate answers than using average λ's over large time periods. This is particularly important if the containment system is breached and the airborne contents (if under positive pressure) can directly puff to the environment. Average λ's could give airborne concentrations that are more than 50% in error if the λ is averaged over one or more hours.

Single Compartment Behavior of Molecular Iodine (I_2) Showing Effects of Three Depletion Mechanisms - Natural Deposition, Spray Washout with Neutral or Boric Acid and with Caustic Sprays and Internal Filtration



Legend:

A. Natural Deposition: Driving force temperature difference initially 30% falling linearly to 0.1°F on one hour.
B. Natural Deposition. Driving force initially 60°F falling linearly to 0.1°F in one hour.
C. Internal Filtration. One volume/hour circulated through a filter at 100% removal efficiency.
D. Spray Washout with neutral or boric acid sprays, 0.1 cm drops (mass mean diameter).
E. Spray Washout with neutral or boric acid sprays, 0.0465 cm drops.
F. Spray Washout with caustic pH = 0.5, 0.1 cm drops.

Figure 2

IV. Multicompartment Application in a Complex Postulated Accident

The CORRAL code was used as part of the accident analysis of N Reactor. One of the postulated accidents is a fuel ejection accident.

N Reactor differs significantly from commercial reactors in that it's core is composed of metallic fuel elements which lie in 1003 horizontal pressure tubes inside a graphite moderator which measures 39'-5" by 33'-0" by 33'-4 1/2". Water is circulated through the pressure tubes at high pressure which cools the fuel elements.

The N Reactor confinement Model used by the CORRAL code is shown in Figure 3 having the reactor core surrounded by the 105-N building front and rear compartments. An adjacent 109-N building pipe space and steam generator cells are connected to the 105-N building rear compartment by atmospheric flow paths. A flow path through filters and out an elevated stack is shown leading from the 105-N building front compartment.

The fuel ejection accident assumes the front nozzle of a single pressure tube fails and fuel is discharged from the pressure tube into 105-N building front by the primary coolant back pressure. A single fuel element is assumed to lodge in piping or structure where it is uncooled by fog sprays or contact with the floor.

The confinement building is pressurized as shown in Figure 4, to a peak pressure of 0.6 psig by steam flashing to hot water from the nozzle discharge. The sequence of events of the confinement are tabulated in Table 1. The rate of condensation by the fog sprays increase as steam is dispersed throughout the building until the pressure falls to 3 inches W.C. after 20 minutes and flow is established through filters and out a 230 ft. stack.

Building leakage also occurs during the pressurization and filter release stages of the accident. The leakage rate is assumed to be 200 cfm at a corresponding pressure of 0.1 inch W.C. The leakage rate was assumed to be proportional to the square root of the confinement atmospheric pressure and proportional to the volume of each compartment.

The total release during the filtered release stage is equal to the pool displacements from the fog sprays and nozzle spills. The building leakage rate is 5 percent of the total release rate and is also assumed proportional to the compartment volumes.

Fission products are released from the uncooled fuel element beginning 292 seconds after the accident and are completely released after 468 seconds.

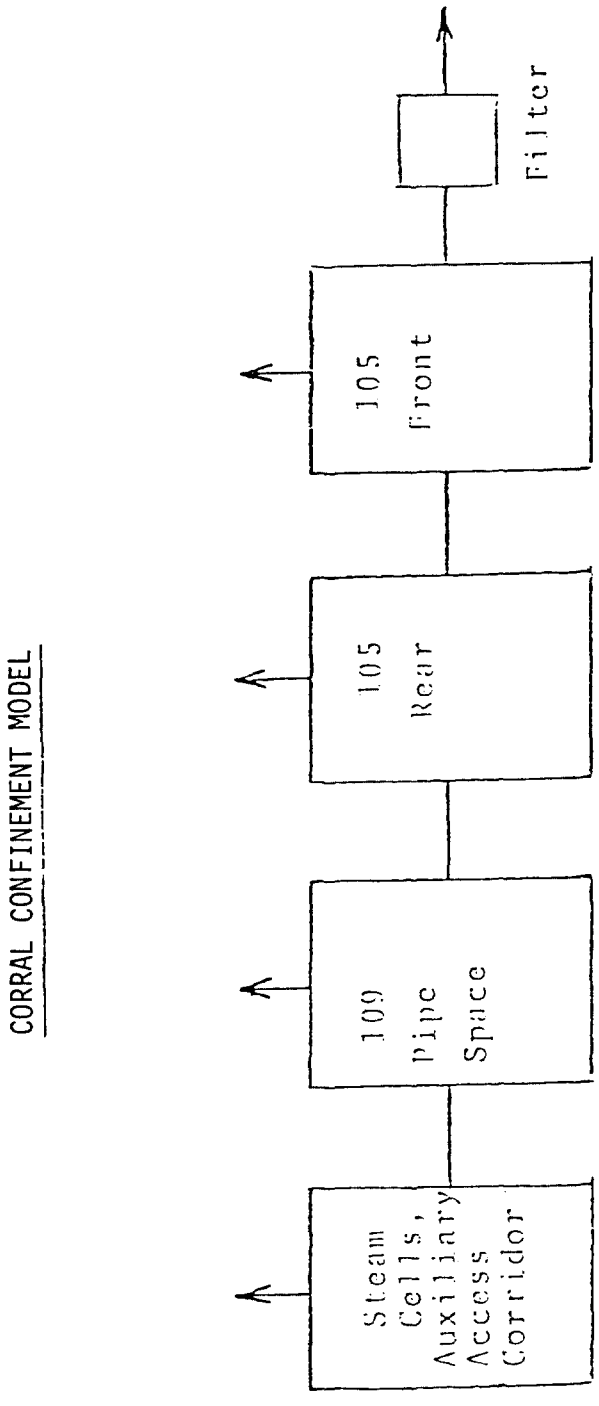

CORRAL CONFINEMENT MODEL
Steam Cells, Auxiliary Access Corridor
109 Pipe Space
105 Rear
105 Front
Filter

Figure 3

FUEL EJECTION ACCIDENT CONFINEMENT PRESSURE VERSUS TIME

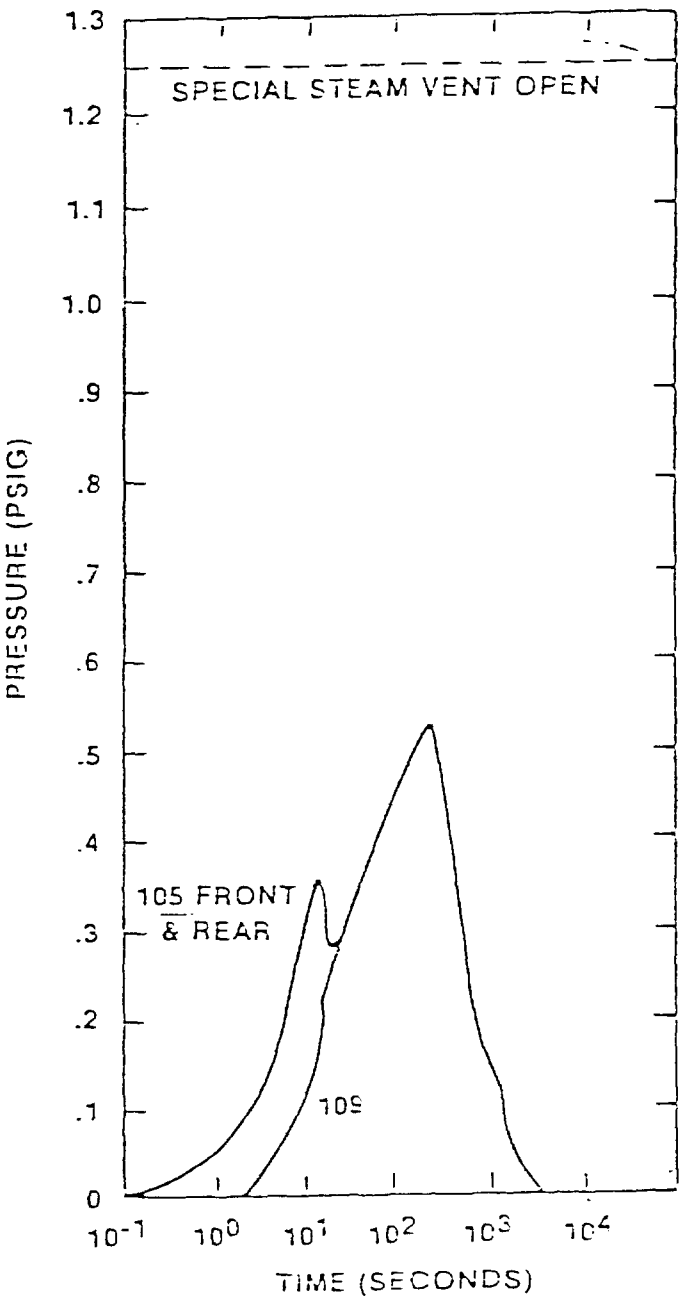


Figure 4

TABLE I

SEQUENCE OF EVENTS
FUEL EJECTION ACCIDENT

Time (Sec)	Event
0	Front nozzle fails and fuel is ejected.
0.5	Flow monitor actuates low flow trip.
0.7	Insertion of control rods begins.
2.2	Control rods 90 percent inserted.
2.5	105-N confinement trip at set pressure of 2 inches w.g.
3.5	105-N Zone I ventilation exhaust valves closes to 80 degrees
4.5	105-N Zone I ventilation supply valves closes to 80 degrees.
7.0	109-N confinement trip at set pressure of 2 inches w.g.
12.0	109-N ventilation exhaust and supply valves close.
15.0	Fog sprays in 105-N come on at 12 inches w.g.
56.7	Fog sprays in 109-N come on at 10 inches w.g.
280.0	Peak confinement pressure of 0.54 psig reached.
292.0	Ejected fuel begins to fail and release fission products to confiner atmosphere.
468.0	Fuel failure ends and no more fission products are released to confiner atmosphere.
1100.0	105-N Zone I exhaust valves open at 3 inches w.g. and filtered flow is established at 1740 cfm with a leak rate of 87 cfm.

The fog sprays consist of two nozzle types in the 105-N building and a third type in the 109-N building. Table 2 gives a breakdown of flows, fall heights and droplet sizes which represent the fog spray system for the accident analysis.

The compartment volumes, wall areas and floor areas are shown in Table 3. The compartment air temperatures are assumed to be ambient and the differential temperature between the compartment air and walls is 0.1°F for times greater than 2 hours after the accident.

The washout and plateout of elemental and particulate iodine is shown in Figure 5 as a function of time for unit amounts of elemental and particulate iodine released to the confinement atmosphere.

The fraction of noble gas and methyl iodine contained in the confinement atmosphere versus time is shown in Figure 6. Similarly, the fractions of elemental iodine and particulates are shown in Figures 7 and 8 respectively.

The removal of elemental iodine and particulate iodine by the filters is 95 percent and 99.95 percent respectively. No removal of noble gas or methyl iodine by the filter was assumed. The fraction of noble gas and methyl iodine, elemental iodine and particulate iodine, released to the atmosphere from filtered release are shown in Figures 9 through 11 respectively.

Similarly, the release by building leakage to the atmosphere is shown for noble gas and methyl iodine, elemental iodine and particulates in Figures 12 through 14 respectively.

It is evident in Figures 5 and 6 that the fog sprays and plateout of N Reactor are effective in removing iodine from the confinement atmosphere. The elemental iodine decreases to about 2 percent of the original value in about 1 hour. The removal of particulates is almost instantaneous. These removal mechanisms for iodine and particulates show a significant decrease in the building release fractions compared to the noble gas release. A similar comparison is seen for the filtered release where the filter removals are also factored into the results.

The results of the releases to the atmosphere versus time were used to calculate accident doses for individual isotope releases to the confinement atmosphere. Appropriate depletion by radioactive decay during holdup on the confinement building was taken into account to arrive at release rates of isotopes as a function of time which were used to calculate doses.

Table 2 Fog Spray Parameters for Fuel Ejection Accident

Compartment	Spray Height (ft)	Drop Diam. (ft)	Spray Flow Rate (lb/sec)
105-N front	74.5	0.043	42.24
105-N front	74.5	0.055	0.0
105-N front	18.0	0.043	309.76
105-N front	18.0	0.055	0.0
105-N rear	8.0	0.043	21.12
105-N rear	8.0	0.055	81.01
105-N rear	74.0	0.043	178.34
105-N rear	74.0	0.055	324.04
105-N rear	20.0	0.043	152.53
105-N rear	20.0	0.055	253.15
109-N pipe space	51.0	0.0665	152.73

Table 3 Compartment Volumes, Wall and Floor Areas of CORRAL Confinement Building Model

Compartment	Volume (ft^3)	Wall Area (ft^2)	Floor Area (ft^2)
105-N front	3.58×10^5	4.2×10^4	8.5×10^3
105-N rear	3.92×10^5	4.7×10^4	7.2×10^3
109-N pipe space	7.22×10^5	7.6×10^4	1.39×10^4
109-N steam gen. cells, aux. cell and access corridor	1.51×10^6	1.39×10^5	2.0×10^4

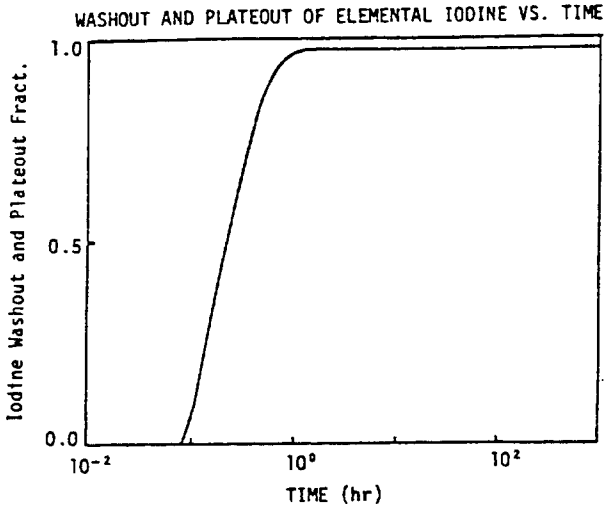


*FIGURE 5

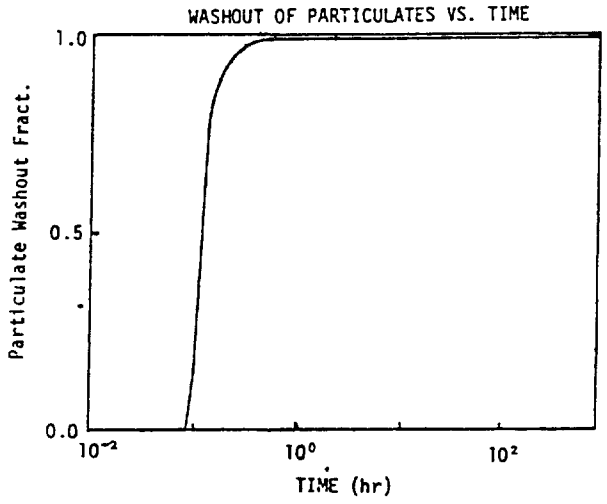


FIGURE 6

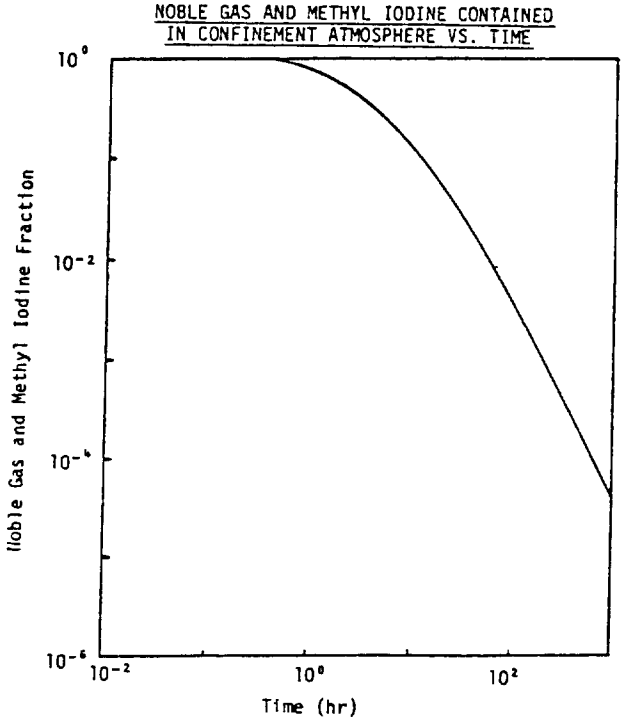


FIGURE 7

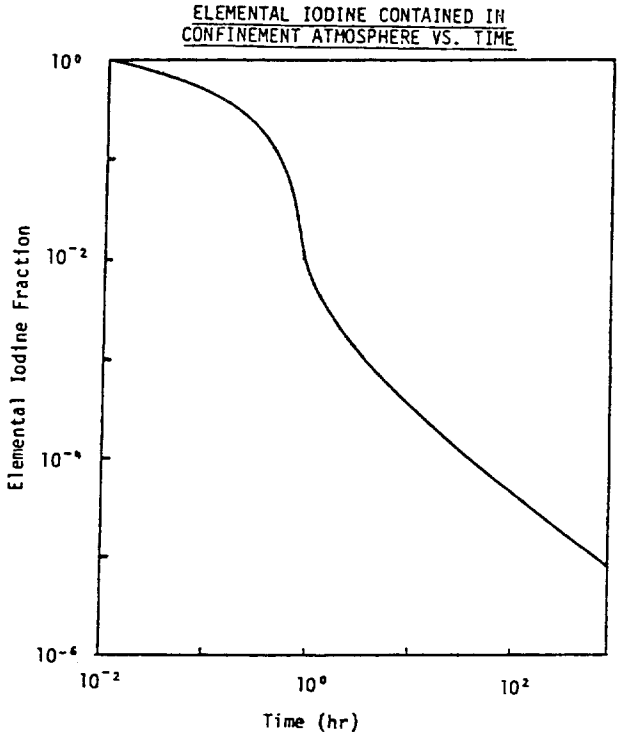


FIGURE 8

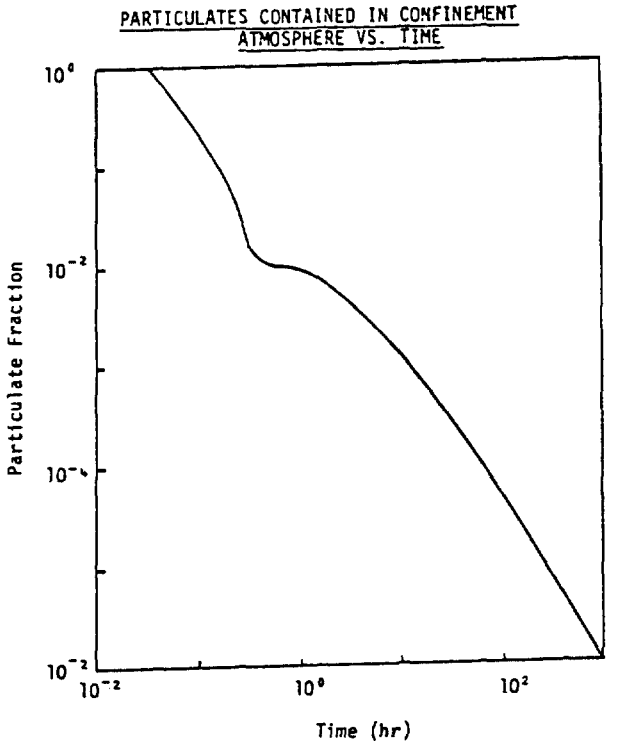


FIGURE 9

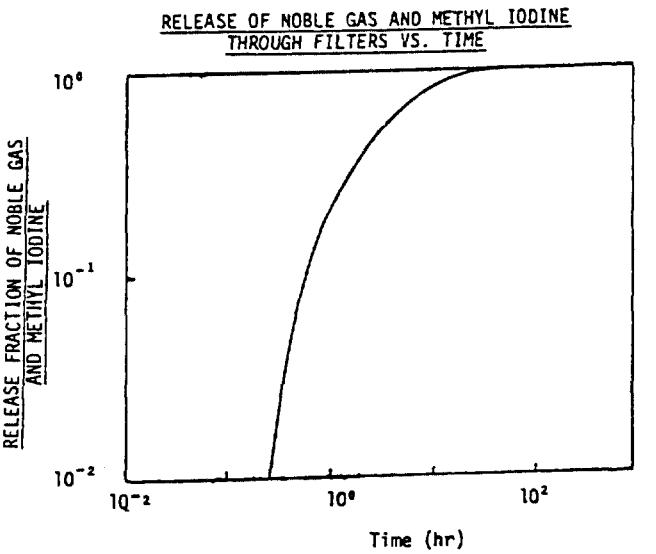


FIGURE 10

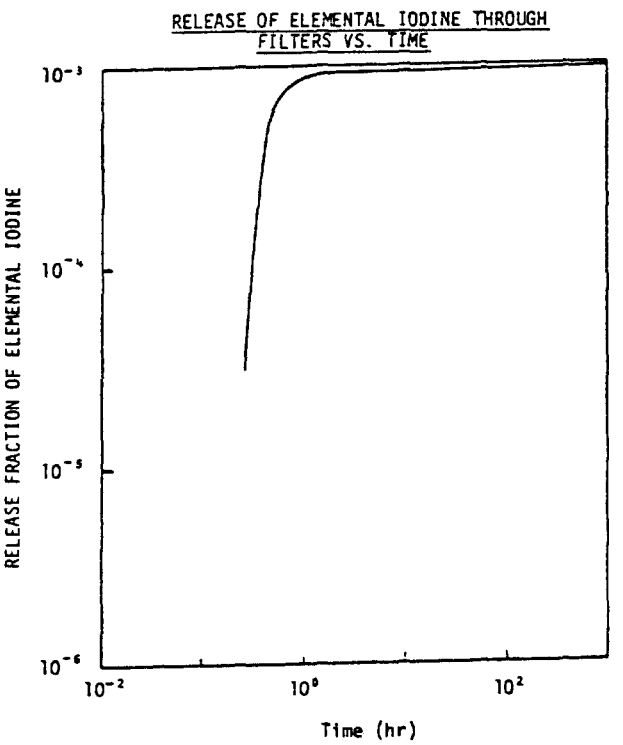


FIGURE 11

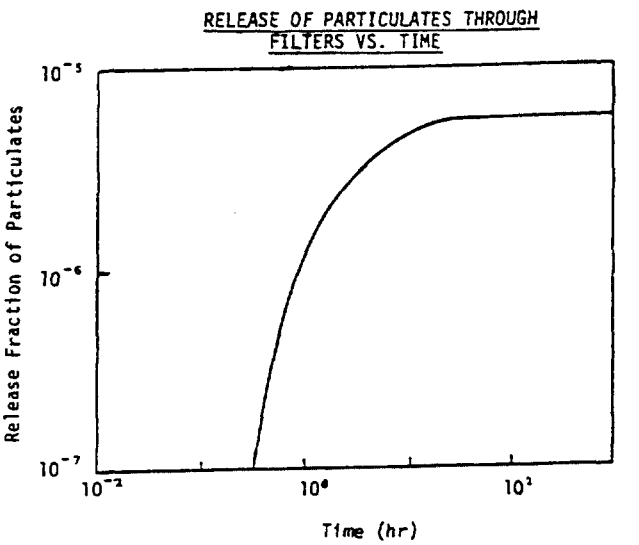


FIGURE 12

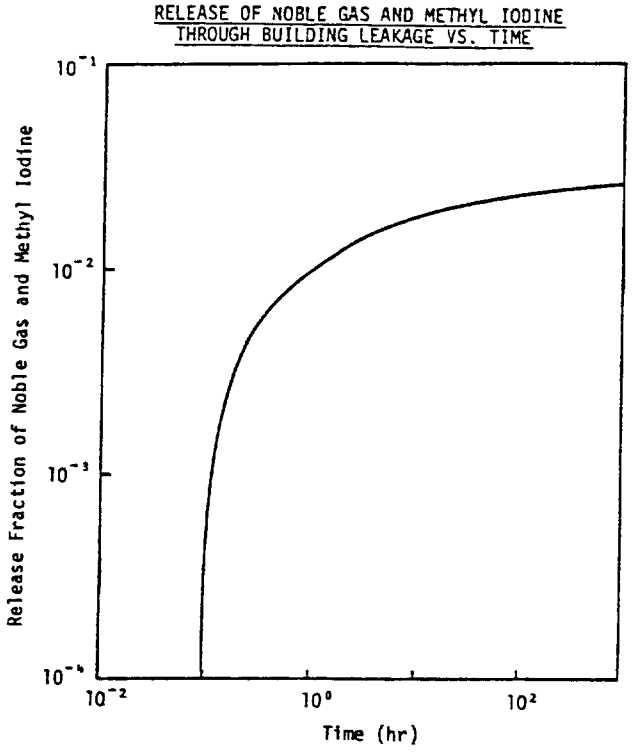


FIGURE 13

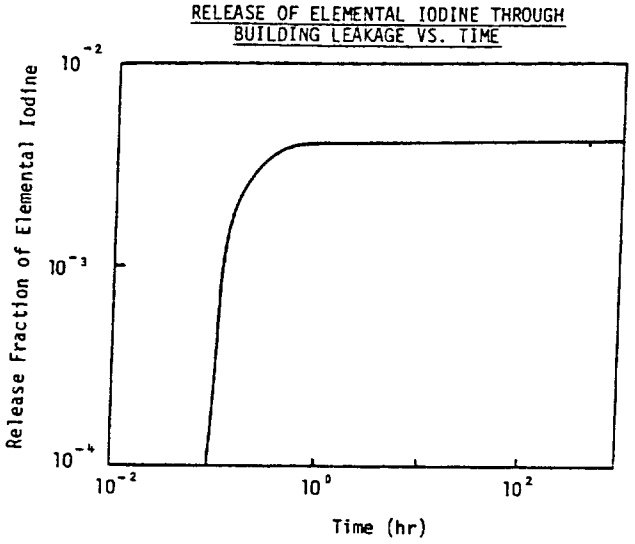


FIGURE 14

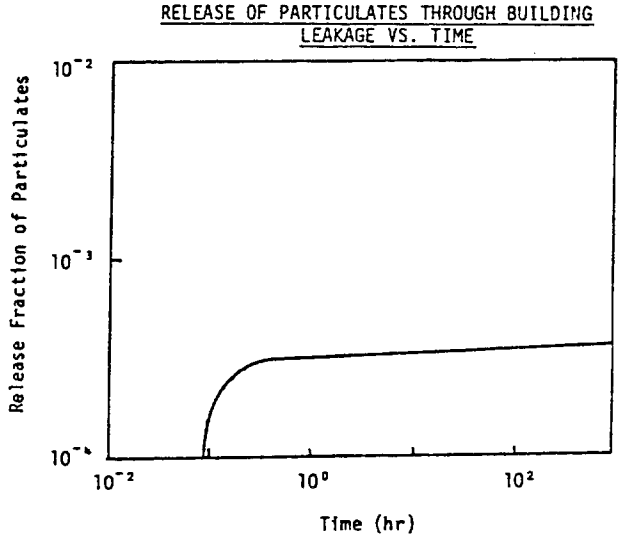


FIGURE 15

References

(1) A.K. Postma, P.C. Owzarski, and D.L. Lessor, "Transport and Deposition of Airborne Fission Products in Containment Systems of Water Cooled Reactors Following Postulated Accidents," Appendix J of Appendix VII, WASH-1400 (Nureg 75/014), U.S. Nuclear Regulator Commission, October 1976.

(2) A.K. Postma and W.F. Pasedag, " A Review of Mathematical Models for Predicting Spray Removal of Fission Products in Reactor Containment Vessels," BNWL-B-268, Battelle-Northwest, Richland, WA (1973).

DISCUSSION

ALVARES: Has there been any experimental verification of the model?

ROBLYER: Yes. The removal models in the CORRAL code are based on the work of A. K. Postma and W. F. Pasedag in WASH-1329 dated June, 1974. Their mathematical models were based on the Confinement Systems Experiment conducted on the Hanford Site.

SHAW: Would you explain the inverse relationship of particle radius for your spray efficiency and what is the average droplet size?

ROBLYER: The removal rate is proportional to the ratio of flow rate and drop diameter and is proportional to the droplet removal efficiency. The droplet diameters from the non-clogging nozzles at Hanford N Reactor range from 0.11 to 0.17 cm which is much larger than those of most commercial reactors.

SCHIKARSKI: The CORRAL code is a very nice tool to handle fission product transport problems. However, in its present state its treatment of the behavior of particulates is poor. It contains only the removal by wash-out and settling but neglects coagulation, diffusion, and thermophoresis. Therefore, it underestimates particulate removal by orders of magnitude. Are you going to improve the code in that respect?

ROBLYER: We have no plans to incorporate these models into the CORRAL code because the dose from particulates is relatively small for our problems. However, I am sure that Dr. Owzarski, who now works for Pacific Northwest Laboratories in Richland, Washington, could incorporate these models into the CORRAL code to suit your individual needs.