Chapter 6 Safety Analysis

6.1 Introduction

6.1.1 Chapter Content

This chapter presents an introduction to the safety analysis. As mentioned previously, safety analysis is required to assist the construction of event trees and to assist the determination of event consequences. In a more general sense, safety analysis is required for the design of safety systems and for the evaluation of the performance of the safety systems once designed. [HOL90] provides a rather nice overview of the role played by safety analysis and is included as appendix 8. Figure 6.1 provides a superb overview of the interaction between the analysis disciplines. Herein we take a look at the estimation of event consequences.

6.1.2 Learning Outcomes

The overall objectives for this chapter are as follows:

Objective 6.1	The student should be able to estimate fission product inventories, source dispersion and dose rates.					
Condition	Open book written examination.					
Standard	100% on methodology, 75% on technical accuracy.					
Related concept(s)						
Classification	Knowledge	Comprehension	Application	Analysis	Synthesis	Evaluation
Weight	a	a	a			

Objective 6.2	The student should be able to calculate power histories and thermalhydraulic response using supplied computer codes.					
Condition	Workshop project exercises.					
Standard	100% on methodology, 75% on technical accuracy.					
Related concept(s)						
Classification	Knowledge	Comprehension	Application	Analysis	Synthesis	Evaluation
Weight	a	a	a			

6.1.3 The Chapter Layout

The exploration proceeds systematically from the core to the dose uptake. Because of the complexity of detailed calculations, only a cursory examination of the subject can be given.

6.2 Reactor Physics Codes

It is beyond the scope of this course to delve into detailed reactor physics calculations. However, some calculations are needed to assist the analysis. To this end, a simple point kinetics model has been coded (PKSIM.C) and is supplied on disk. This code is sufficient to provide estimates of the reactor flux (and hence power) response to a given reactivity insertion.

Decay heat removal is essential in preventing fuel overheating and possible fission product release. Todreas and Kazimi [TOD90] give the approximate decay power as

$$\frac{P}{P_0} = 0.066 \left[(\tau - \tau_s)^{-0.2} - \tau^{-0.2} \right] \quad (1)$$

where P is the decay power, P_0 is the nominal reactor power, τ is the time since reactor startup and τ_s is the time of reactor shutdown.

6.3 Thermalhydraulic Codes

It is also beyond the scope of this course to delve into thermalhydraulic calculations. The event scenarios of most interest to safety analysis (two-phase flow under low flow conditions) are, unfortunately, the hardest to model. A homogeneous equilibrium model (HEM) based code (PDOT.C) is supplied on disk but the user will find that the code is only as good as the correlations and inherent assumptions, which are invalid for all but the least severe accidents. For the purposes of this course, event details will have to be assumed or estimated based on engineering judgement or simple back-of-the-envelop calculations.

6.4 Source Calculations

Chapter 3 discussed how limits on reactor accident consequence (expressed in units of deaths / year) were historically translated into dose limits. This implies limits on radioactive releases since releases lead to dose uptake. Releases are, in turn, a function of the accident events. The sequence is

event ---> release ---> dose ---> deaths

Limits on one implies limits on the others. We have focussed on events - setting frequency criteria, defining design basis events, defining event trees and fault trees to evaluate the event probability and, with the aid of analysis tools, estimate consequences.

To complete the calculation, we need to estimate the releases for the events (source calculation) and to relate the releases to dose uptake.

The calculation of radioactive releases inside containment depend on the fission product inventory and the fuel and heat transport system failure details. The calculation is well beyond the scope of this course

and recourse to bounding calculations will have to be made, if necessary, by assuming that a percentage of the fission product inventory escapes foı a given accident scenario.

6.4.1 Fission Product Inventory

The fission product inventory in an operating reactor can be estimated as follows [taken directly form LAR83, pp587-588]. Suppose that the reactor has been operating at a power of P MW. If the recoverable energy per fission is taken to bc 200 Mev, the total number of fissions occurring per second is

$$\text{Fission rate} = P\ \text{MW} \times \frac{10^6 \text{joule}}{\text{MW-sec}} \times \frac{\text{fissions}}{200\text{Mev}} \times \frac{\text{Mev}}{1.60\text{x}10^{-13}\text{joule}} \qquad (2)$$

$$= 3.13 \times 10^{16}\ P\ \text{fissions/sec.}$$

If the cumulative yield of the i^{th} fission product (the yield of the fission product itself plus the yields of all its short lived precursors) is γ_i atoms per fission, then the rate of production of this nuclide is

$$\text{rate of production} = 3.13\text{x}10^{16}\ P\ \gamma_i\ \text{atoms/sec.} \qquad (3)$$

The activity at time, t, of that fission product while in the core of the reactor is

$$\alpha_i = 3.13\text{x}10^{16}\ P\ \gamma_i\left(1-e^{-\lambda_i t}\right)\text{disintegrations/sec.} \qquad (4)$$

Expressed in curies, this is

$$\alpha_i = \frac{3.13\text{x}10^{16}\ P\ \gamma_i}{3.7\text{x}10^{10}}\left(1-e^{-\lambda_i t}\right)\text{disintegrations/sec.} \qquad (5)$$

$$= 8.46 \times 10^5\ P\ \gamma_i\left(1-e^{-\lambda_i t}\right)\text{Ci}$$

If the activity saturates in the time, t, that is, if $\lambda_i >> 1$, equation 5 reduces to

$$\alpha_i = 8.46 \times 10^5\ P\ \gamma_i\ \text{Ci} \qquad (6)$$

Table 6.1 gives the inventories of the most important noble gases and iodine fission products computed for a typical 1000 MWe (PWR) plant at the end of a fuel cycle.

The amount of a fission product available for release to the atmosphere can be estimated by

$$C_0 = 8.46 \times 10^5\ F_p F_h P\ \gamma_i\left(1-e^{-\lambda_i t}\right)\text{Ci}, \qquad (7)$$

where F_p is the fraction of the radio nuclide released from the fuel into the reactor containment and F_b is the fraction of this that remains airborne and capable of escaping from the building.

6.4.2 Dispersion Calculations

Consider a volume containing a reactor in some accident mode, where the concentration of a particular nuclide is C Bg/m^3. Assume a leak rate of V m^3/ s into the atmosphere at a height h meters as shown in figure 6.2. The release rate Q is given by:

$$Q = C\ \text{Bq/m}^3\ .\ V\ \text{m}^3\text{/s} = C\ V\ \text{Bq/s} \qquad (8)$$

This release is dispersed into the surrounding area via the release plume. For a given weather condition

with wind velocity u and other data, a concentration of radioactivity at some distance and direction from the source can be calculated from:

$$\chi = \left(\frac{2}{\pi}\right)^{1/2} \frac{Q}{\sigma_z \bar{u}} \frac{f}{\theta x} e^{-h^2/2\sigma_z^2} \qquad (9)$$

where χ is the sector-averaged long-term concentration in Bq/m^3 a distance x meters from the source, and will be uniform through the sector
Q is the release rate in Bq/s from a source h meters in height
σ_z is the vertical diffusion coefficient in meters
θ is the angle subtended by the sector [radians]
f is the fraction of time the wind blows into the sector

$\bar{u}$ is the mean wind velocity in m/s.

6.5 Dose Calculations

The local radiation leads to an external dose due to the ambient radiation level and an internal dose due to inhaled species. By way of example, iodine production at McMaster for use in medical applications generates Ar^{41}, which is an external source, and I^{125}, which is an internal source. The β from the iodine is too soft to be much of an external risk since clothing and the outer dead layer of skin provides shielding. Inhalation, however, leads to direct exposure to tissue. This is aggravated since iodine is readily absorbed by the body and concentrates in the thyroid. External dose, then, is a function of the time exposed to an ambient radiation level. Internal dose is a function of the radiation uptake and the residence time in the body.

For the external case using Ar^{41} as an example, 1 Bq/m^3 gives a dose of 2.3×10^{-10} Sv/hr. Thus:

$$\text{Dose}_{\text{external}}\ [\text{Sv/yr}] = \left(Q\ \frac{\text{Bq}}{\text{s}}\right)\left(\chi/Q\ \frac{\text{s}}{\text{m}^3}\right)\left(2.3\times10^{-10}\frac{\text{Sv/hr}}{\text{Bq/m}^3}\right)\left(24\text{x}365\ \frac{\text{hr}}{\text{year}}\right) \qquad (11)$$

A typical dilution factor (χ/Q) is 4.5×10^{-6} s/m^3 at 200 meters from a release at a height of 20 meters, which gives a dose of 9.1×10^{-12} Q Sv/yr. A dose limit of, say, 5 mSv/yr gives a release limit of .005 / $9.1\times10^{-12} = 5.5\times10^{8}$ Bq/s. Equation 8 can then be used to infer a limit on the permitted building concentration.

For internal uptake of I^{125}, 1 Bq leads to 2×10^{-6} Sv. This is due to the integrated dose received, which is a function of the dose level (Bq/time) and the residence time. The uptake rate is just the inhalation (volume) rate which is roughly 3.8 m^3/ day for an infant. The dose is thus:

$$\text{Dose}_{\text{internal}}\ [\text{Sv}] = \left(Q\ \frac{\text{Bq}}{\text{s}}\right)\left(\chi/Q\ \frac{\text{s}}{\text{m}^3}\right)\left(3.8\ \frac{\text{m}^3}{\text{day}}\ \text{x}\ 365\frac{\text{days}}{\text{year}}\right)\left(2\text{x}10^{-6}\ \frac{\text{Sv/yr}}{\text{Bq}}\right) \qquad (12)$$

= release rate x dilution factor x volume rate x dose/uptake
= local concentration x volume rate x dose/uptake
= uptake x dose/uptake

Using the same dilution factor of $(\chi/Q) = 4.5\times10^{-6}$ s/m^3, the internal dose is 1.248×10^{-8} Q Sv. Thus a dose limit of 10μSv is received for a release of

$$Q\ [Bq/s] = \frac{10x10^{-6}Sv}{1.248x10^{-8}Sv/(Bq/s)} = 801\ Bq/s \qquad (13)$$

Details can be found in [HP-5-94-0] and [CSA-N288.1].

6.6 Exercises

1. Calculate the saturation activity of I^{131} in a reactor operating at 1000 MWE (3200 MWth) and the activity released in elemental form into the reactor containment in a meltdown assuming that 25% of the iodine is released from the fuel.

2. Consider the case of a research reactor operating at 2 Mwth.
   a. Calculate the saturation activity of I^{131} and the activity released in elemental form into the reactor containment in a meltdown assuming that 25% of the iodine is released from the fuel.
   b. How does the risk of fission product I^{131} induced dose compare to that of the commercial production of I^{125} produced at the same site. Assume I^{125} has the same biological impact as I^{131}. Also assume that 50 Ci of I^{125} is in store at any given time.

Table 1.1 Typical core inventory of selected volatile fission products in a 1000 Mwe PWR at the end of a fuel cycle. Source [LAM83, table 11.4, pg 588].

Typical core inventory of selected volatile fission products in a 1000 MWe PWR at the end of a fuel cycle

Nuclide*	Half-life†	Fission yield‡	Curies ($\times 10^8$§)
^{85m}Kr	4.4 h	0.0133	0.24
^{85}Kr	10.76 y	0.00285	0.0056
^{87}Kr	76 m	0.0237	0.47
^{88}Kr	2.79 h	0.0364	0.68
^{133}Xe	5.27 d	0.0677	1.7
^{135}Xe	9.2 h	0.0672	.34
^{131}I	8.04 d	0.0277	.85
^{132}I	2.28 h	0.0413	1.2
^{133}I	20.8 h	0.0676	1.7
^{134}I	52.3 m	0.0718	1.9
^{135}I	6.7 h	0.0639	1.5

*Superscript *m* refers to a nuclide in an isomeric state (see Section 2.8).

†m = minutes, h = hours, d = days, y = years.

‡Cumulative yields in atoms per fission; equal to yield of nuclide plus cumulative yield of precursor. From M. E. Meek and B. F. Rider, "Compilation of Fission Product Yields," General Electric Company report NEDO-12154, 1972.

§From "Reactor Safety Study" WASH 1400, 1975.

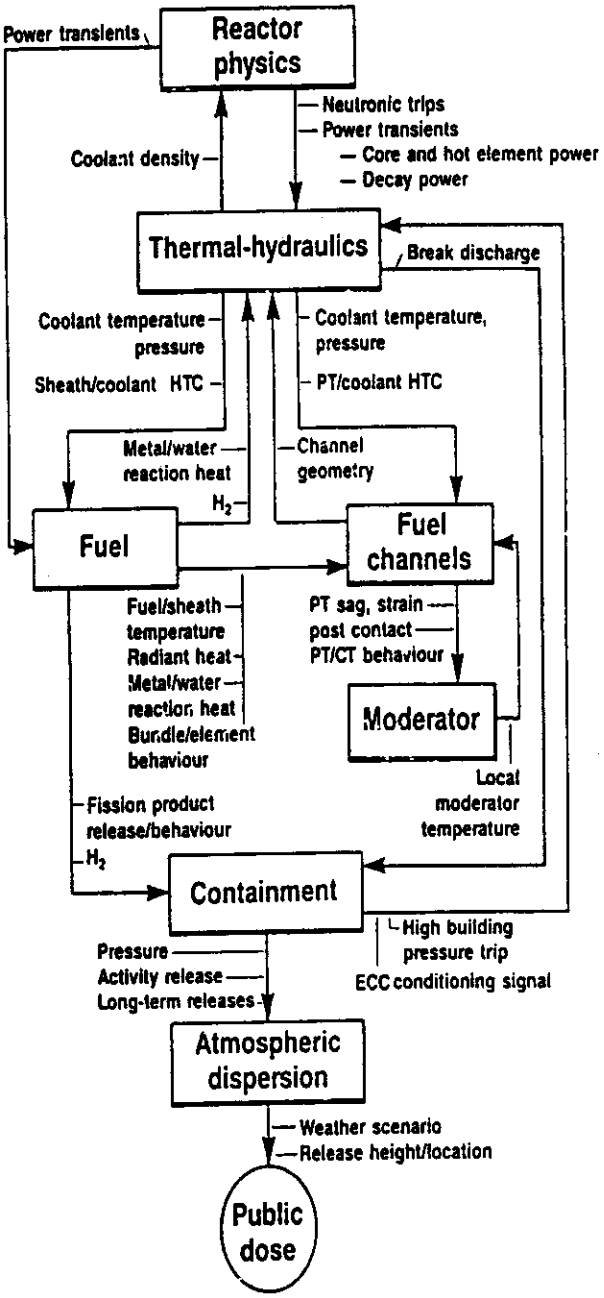

Figure 6.1 Interaction Between Analysis Disciplines - Large Breaks in the PHTS, Single Failure [Source NATH85]

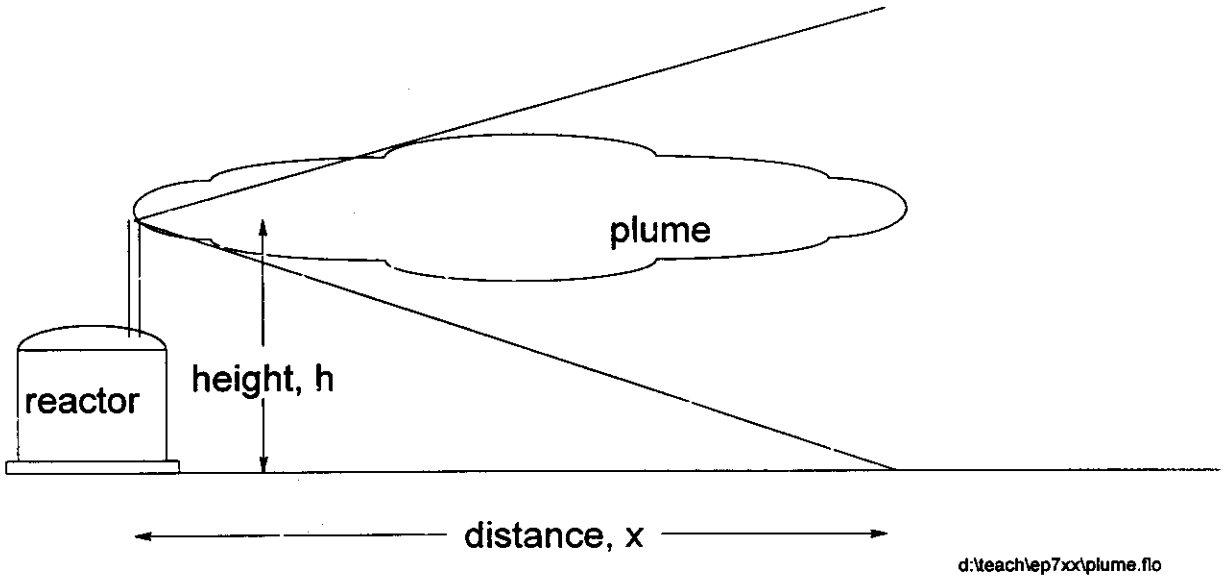

Figure 6.2 Dispersion of effluents