FORD NUCLEAR REACTOR ACCIDENT ANALYSIS AND PUBLIC RADIOLOGICAL CONSEQUENCES OF MAXIMUM CREDIBLE ACCIDENT

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AND

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CREDIBLE ACCIDENT

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A thesis submitted in partial fulfillment
of the requirements for the degree of Master on
Science in Nuclear Engineering of the University
of Michigan.

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Ann Arbor, Michigan, December, 1976.

ABSTRACT

An accident analysis at the 2Mw pool-type light-water-coolant-moderator Ford Nuclear Reactor at the University of Michigan has been carried out. A synopsis of the accidents analyzed follows:

1. Loss of Primary Coolant Accident

   The maximum fuel plate center temperature following a complete loss of primary coolant, leaving the reactor core uncovered, is estimated to be 928.13°F at 223.66 minutes after reactor shutdown. The preshutdown operating conditions were 2 Mw power level for a 10 day period. A relation between loss of coolant maximum fuel plate center temperature and reactor power demonstrates that the maximum reactor power at which the reactor can operate safely before melting occurs under the prestated operating conditions is 2.58 Mw.

2. Reactivity Insertion Accidents

2.1 For a maximum reactivity insertion rate of 0.0277% $\Delta k/k/sec$, corresponding to the maximum shim safety rod withdrawal rate, and the reactor initially at its shutdown state (shutdown reactivity 0.035 $\Delta k/k$), the reactor period reached at the safety system power level trip setpoint of 2.4 Mw is estimated to be 440 msec and the energy release during the excursion is 0.1081 Mw-sec.

2.2 For a step reactivity insertion of 0.012 $\Delta k/k$ corresponding to the maximum experiment worth permitted by Technical Specifications and the reactor operating at a steady-state power level of 2 Mw, the reactor period at trip level is 23.4 msec and the energy release 0.5429 Mw-sec.

Comparison of these results with the 3.2 msec period and 30.7 Mwsec energy released at the Spert-1 tests shows that no reactor fuel damage would result.

3. Primary Coolant System Failure

   No melting is expected since the reactor is always covered with water.

4. Fuel Element Blockage Accident

   Coolant flow channel boiling caused by blockage of flow through an element would cause period fluctuations and subsequent scram of the reactor, thus preventing fuel damage.

5. Fuel Element Cladding Failure

Fission product release had to be taken into consideration. Because of this, a maximum credible accident was postulated in which, due to manufacture defects, a rupture of 10% of the cladding material from one fuel element plate cause the release of 1.59×10^{-2} % of the total core inventory based upon a power history of 2 Mw for 1 year of operation. The fission products released, 100% of the iodines and noble gases Xenon and Krypton, were instantaneously released into the pool water. The residency of these fission products within the pool and containment was assumed to be a function of only their decay constants and experimentally determined leak rates from the pool and containment.

For the above reactor core fission product inventory and leak rates, and some conservative assumptions given within this report, the accumulated external, thyroid, and total doses likely to be received by a population at different distances from the reactor site and for different exposure times following the accident are in general well within the recommended maximum permissible doses recommended by the National Council on Radiation Protection (NCRP).

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Decay Constant for One-Delayed-Neutron-Group Model	$\bar{\lambda}$	0.08123	sec^{-1}
Shutdown Reactivity	ρ_o	0.035	$\Delta K/K$
Aluminum Melting Point	none	1200	^{o}F
Temperature at the Bottom of the Fuel Element	none	200	^{o}F
Step Reactivity Insertion	none	0.012	$\Delta K/K$

(*) Values of λ_i are given in table (3.1.1.1-1). λ_i in table (7.1) represent the decay constants of the radioisotopes listed there.

TABLE OF PHYSICAL AND NUCLEAR CONSTANTS

QUANITY	SYMBOL	VALUE	UNIT
Length of Conduction Path to end Box (MTR-type fuel plate)	a	0.97	ft
Cross Sectional Area of Fuel Assembly	A	0.0275	ft^2
Area of Hole 8" diameter	A_h	0.35	ft^2
Surface Area of Reactor Pool	A_p	200	ft^2
Specific Heat of Aluminum	C_p	0.22	BTU/ 16 °F
Acceleration of Gravity	g	32.06	ft/sec^2
Initial Water Head	h_o	23	ft
Thermal Conductivity of Aluminum	k	115	BTU/hr-ft °F
Prompt Generation Neutron Life Time	ℓ	1.04×10^{-4}	sec
Fuel Assembly Plate Mass	m	7.5	lb
Steady-State Reactor Power	P	2	Mw
Shutdown Power Level	P_s	0.8	watts
Trip Level	P_T	2.4	Mw
Maximum Rate of Positive Reactivity Insertion	r	0.0277	$\% \frac{\Delta k/k}{sec}$
Rate of Negative Reactivity Insertion	r'	0.1667	$\frac{\Delta k/k}{sec}$
Reactor Operating Time	T	10	days
Effective Delayed Neutron Fraction (6-groups U-235)	β	0.755	%
Decay Constant for i^{th} Delayed Neutron Presursor	λ_i	(*)	sec^{-1}

From the analysis of these accidents, a maximum credible accident is postulated. Since in any maximum credible accident, special attention is given to the release of fission porducts and their leak rate from the reactor pool and the containment, experiments were performed to determine these parameters.

CHAPTER 1
INTRODUCTION

The profound concern in nuclear reactor safety is widely reflected in the number of experiments simulating nuclear accidents that have been performed since the development of nuclear energy. An analytical, accurate description of the consequences, of a nuclear reactor accident is essential for the continuous development of such facilities. Such descriptions are the basis for estimating the safety margin required in the design and construction of either a research or a power reactor.

Preliminary and final safety analysis reports must be completed and approved in order for a nuclear power plant to be constructed and operate. PSAR and FSAR reports consider an accident which could take place under the most pessimistic conditions and, according to its radiological hazards, the size of the plant including the exclusion area and the characteristics of the reactor containment, are established. In general, for safety analysis in power plants, this maximum credible accident is postulated as a loss of coolant accident (LOCA).

While the consequences of a LOCA type accident in nuclear power plants are studied, the purpose of this report is to analyze the consequences of different postulated accidents that could occur in the Ford Nuclear Reactor (FNR). A considerable number of experimental tests have been performed in nuclear research reactors of the pool-type, using MTR fuel elements and operating at power levels of the order of the 2 Mw FNR power level. Whenever possible, the information from those experiments will be used either to elaborate the FNR analysis or for comparison purposes.

The possible accidents that can occur in a pool-type reactor can be generally classified as follows:

1. Loss of Coolant Accident
2. Reactivity Insertion Accidents
3. Primary Coolant System Failure
4. Fuel Element Cladding Failure
5. Fuel Element Blockage

QUANTITY	SYMBOL	UNIT
Inverse Reactor Period at Trip Level	ω_T	sec^{-1}
Delayed Neutron Buildup Factor	ξ	none

TABLE OF SYMBOLS FOR VARIABLE QUANTITIES

QUANTITY	SYMBOL	UNIT
Gamma Ray Activity	A_γ	Ci
Energy of gamma rays	$\mathcal{E}_\gamma$	Mev
Energy Generated in Excursion	$\mathcal{E}$	Mw-sec
Containment Leak Rate	L_c	sec^{-1}
Pool Leak Rate	L_p	sec^{-1}
Reactor Power Level	P	w, kw
Power Level at Delayed Critical	P_d	Mw
Maximum Power Level	P_{max}	Mw
Heat Transfer Rate	q	BTU/hr
Counting Rate oil samples	R^{oi}	cps
Counting Rate plain samples	R^{p}	cps
Time at which Delayed Criticality is approached	t_d	sec
Time at which level water is dropped 23 ft	t_{dr}	sec
Time to Reduce Reactivity from Trip Reactivity to Delayed Critical	t_r	sec
Time at which Trip Level is Approached	t_T	sec
Maximum Temperature Center Fuel Plate	T_c	°F
Gamma Ray Flux	ϕ_γ	$Mw/cm^2\ sec$
Effective I-131 Decay Constant for oil samples	λ^{oi}_{eff}	sec^{-1}
Effective ^{131}I Decay Constant for plain samples	λ^{p}_{eff}	sec^{-1}
Reactivity	ρ	$\Delta K/K$
Reactivity at Trip Level	ρ_T	$\Delta K/K$
Reactor Period	τ	sec, msec
Instantaneous Inverse Reactor Period	ω_o	sec^{-1}

CHAPTER 2
LOSS OF COOLANT ACCIDENT

A loss of coolant in the FNR will produce an immediate increase in the radiation levels within the building and abnormal heating of the fuel elements due to fission product energy.

Since an increase in the temperature of the fuel elements could lead to melting of the fuel and clad materials with the subsequent release of fission products, it is worth analyzing the maximun temperature that the aluminum clad could reach due to a LOCA to determine if melting occurs.

Loss of coolant from the reactor pool could be caused by:

1. Cracks in concrete pool walls
2. Beam port rupture
3. Pneumatic tube rupture
4. Rupture of a pipe or component in the primary coolant system

2.1 Fuel Element Meltdown:

It will be assumed that the worst possible leak rate result from the rupture of an 8 inch diameter pipe at the bottom of the reactor pool with subsequent water drainage by gravity. Furthermore, for analysis, it will be assumed that the reactor has been operating at a constant power level of 2 Mw for 10 continuous days.

During reactor operation there is a build-up of fission products which, when the reactor is shutdown, release energy as they decay. This energy is the principal source of heat within the fuel. The heat transfer rate due to fission product energy in a single fuel element containing 140 grams of U-235 has been shown experimentally to be given by the expression [1]

$$q = 14 P \left[t^{-0.2} - (t+T)^{-0.2} \right] \qquad (2.1\text{-}2)$$

where

q = heat transfer rate, BTU/hr.

P = reactor power before shutdown, kw.

t = time after shutdown, seconds.

T = operating time before shutdown, seconds.

In order to clarify the analysis of the maximum temperature rise of the fuel element as a function of the heat produced by the fission products, it is necessary outline the assumptions made [1].

First, it is assumed that the temperature of the fuel rises to a maximum value at a time t after shutdown in which the heat generation within the fuel approaches a steady-state condition. This can be stated by the expression

$$q = \frac{kA}{a} \Delta T_{max} \qquad (2.1\text{-}2)$$

where

k = thermal (conductivity of aluminum, 115 BTU/hr-ft-°F (assumed constant over the range 100 – 160°F).

A = cross sectional area of fuel assembly, 0.0275 ft^2 .

a = length of conduction path to end box, 0.97 ft.

ΔT_{max} = maximum temperature rise of fuel plate, °F

Secondly, it is assumed that heat losses by radiation to surrounding components and convection are small as compared with the conduction loss.

Finally, it is assumed that if no heat were lost from the fuel plate during the decay period then its maximum temperature rise should equal the integrated heat after shutdown divided by the heat capacity of the fuel element. That is, ΔT_{max} will be given by:

$$\Delta T_{max} = \frac{1}{m C_p} \int_0^t q \, dt \qquad (2.1\text{-}3)$$

where

C_p = specific heat of aluminum, 0.22 BTU/lb F at 100°F.

m = mass of fuel plates per assembly, 7.5 lbs.

Equate (2.1-1) and (2.1-2) and solve for

Substitute the value of a, k and A.

$$\Delta T_{max} = 4.3\,P\,[\,t^{-0.2} - (t+T)^{-0.2}\,] \tag{2.1-4}$$

Substituting the value of given in (2.1-1) into (2.1-3) and performing the integration yields:

$$\Delta T_{max} = \frac{14\,P}{0.22 \times 7.5 \times 3600 \times 0.8}\,[\,t^{0.8} - (t+T)^{0.8} + T^{0.8}\,]$$

$$= 2.95 \times 10^{-3}\,P\,[\,t^{0.8} - (t+T)^{0.8} + T^{0.8}\,],\ {}^{\circ}F \tag{2.1-5}$$

Equation (2.1-5) must be corrected because in experiments it has been observed that 78% of all the heat generated is conducted away to the surrounding mass of metal before the fuel reaches its maximum temperature. A second correction must be made since the experiments in reference [1] were carried out with a 15 fuel element core as compared to the 35 element FNR core. Hence, the correction factor in equation (2.1-5) is $0.22 \times (15/35) = 0.0943$ and the corrected equation becomes

$$\Delta T_{max} = 2.78 \times 10^{-4}\,P\,[\,t^{0.8} - (t+T)^{0.8} + T^{0.8}\,],\ {}^{\circ}F \tag{2.1-6}$$

Before going further, the minimum time t after shut-down which will be used in equations (2.1-4) and (2.1-6) will be calculated. This can be accomplished by knowing that the pool level alarm would result in a shim safety rod insertion (automatic rundown) and the operator would initiate a manual scram of the reactor during the first 90 seconds . Then, the minimum value for the shutdown time is equal to the time required to drop the water level 23 ft through the 8" diameter hole at the bottom of the pool minus 90 seconds. The time tdr for the water level to be dropped 23 ft can be evaluated from the Tomicelli's equation[2], that is:

$$tdr = \left[\frac{2h_o}{g}\right]^{1/2} \times \frac{A_p}{A_h} \quad , \qquad \text{seconds}$$

where

h_o = initial water head, 23 ft.

A_p = surface area of the pool, 200 ft ^{2}.

A_h = area of the 8" diameter hold, 0.35 ft ^{2}.

g = acceleration of gravity, 32.06 ft/sec.^{2}

Substitution of these values yields tdr = 684.47 sec.

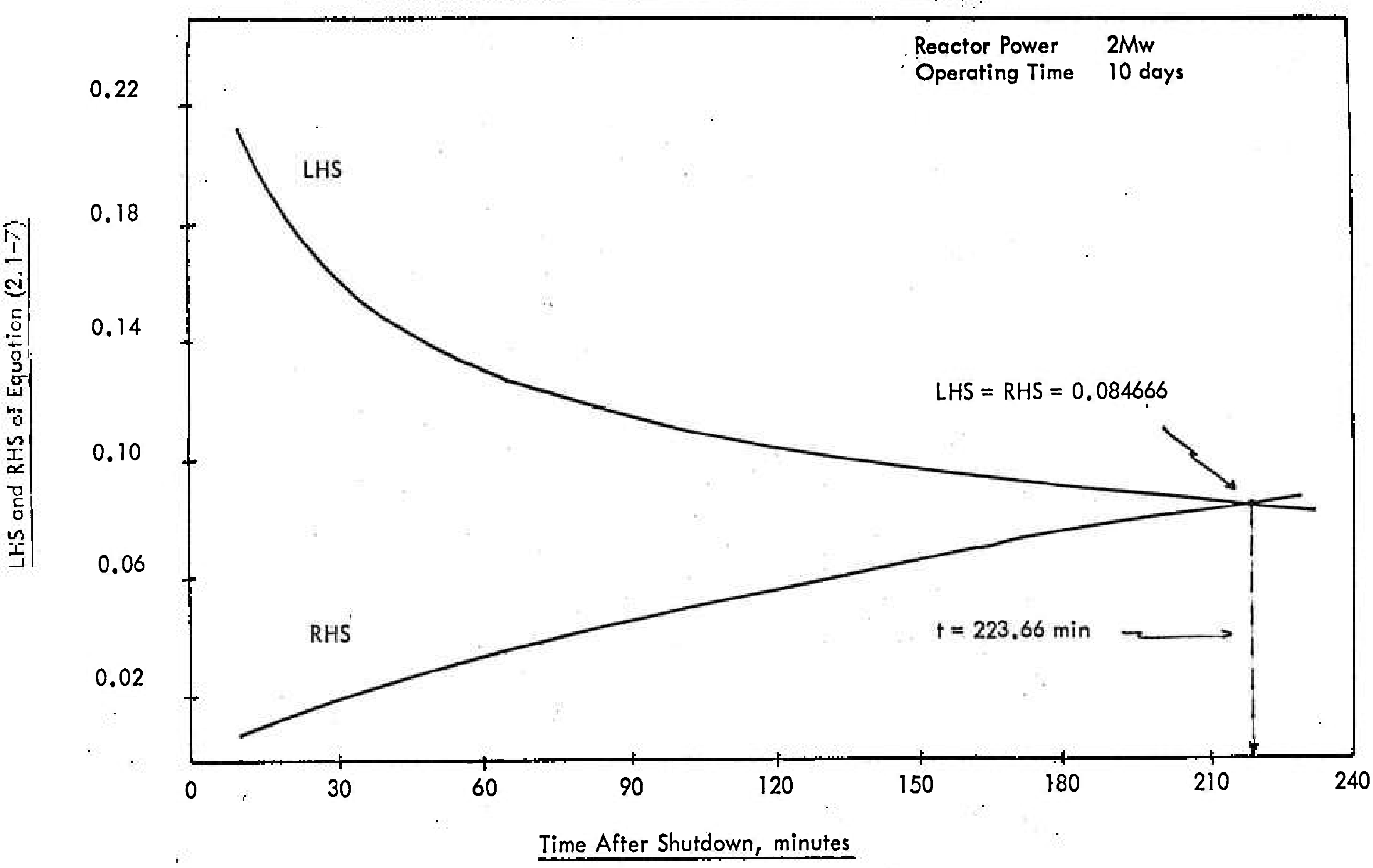
Then, the minimum time t after shutdown to be used in equations (2.1-4) and (2.1-6) is

$$\begin{aligned} t &= 684.47 - 90 \\ &= 594.47 \text{ min} \\ &\approx 600 \text{ sec} \end{aligned}$$

Equating equations (2.1-4) and (2.1-6) yields

Figure (2.1-1)

LHS and RHS of Equation (2.1-7) vs Time After Shutdown



$$4.3\,P\,[t^{-0.2} - (t+T)^{-0.2}] = 2.78\times10^{-4}\,P\,[t^{0.8} - (t+T)^{0.8} + T^{0.8}]$$

or

$$[t^{-0.2} - (t+T)^{-0.2}] = 6.47\times10^{-5}\,[t^{0.8} - (t+T)^{0.8} + T^{0.8}] \qquad (2.1\text{-}7)$$

By plotting the left hand side (LHS) and the right hand side (RHS) of equation (2.1-7) as a function of the time after shutdown and using an operating time of 10 days, the time at which the maximum temperature rise is approached will be given by the intersecting point, that is, when LHS=RHS. Such a plot is given in Fig. (2.1-1) and it is observed that LHS=RHS=0.084666 when t = 223.66 min. Using this value of t in either equation (2.1-4) or (2.1-6) and a power level of 2×10^{3} kw the maximum temperature rise is 728.13° F. Assuming that the temperature at the bottom of the core is 200° F, then the maximum temperature at the center of the fuel plate is $T_c = \Delta T_{max} + 200°F = 928.13\ °F$.

A graphical representation of Tc vs Power Level is given in Fig. (2.1-2). From this figure it is observed that the maximum safe operating power level before melting occurs (1200° F) following a LOCA is 2.58 Mw.

2.2. Gamma Dose Rate from Uncovered Core:

The lack of shielding water would produce a high-intensity radiation field over the uncovered core. The amount of gamma activity in the core can be calculated from the equation [3]

$$A_\gamma = 0.7\,P\,[t^{-0.2} - (t+T)^{-0.2}] \qquad (2.2\text{-}1)$$

where

A_γ = gamma activity, curies.

P = power level, watts

t = 600 sec.

The gamma activity after a shutdown time of 600 sec., an operating time of 10 days and for a power level of 2 Mw is $A_\gamma = 2.9 \times 10^6$ Ci . A crude idea of the value of the exposure rate for a person located just above the core, standing on the reactor bridge can be obtained. In doing so, it will be assumed that the reactor is a point source which emits radiation isotropically and that there is no attenuation between the uncovered core and the bridge.

First, it is necessary to evaluate the gamma energy emission by means of the equation [4]

$$\gamma\text{-energy emission} = 3.7 \times 10^{10} \times 2.9 \times 10^{6}\, E_\gamma, \ \frac{\text{Mev}}{\text{sec}} \qquad (2.2\text{-}2)$$

In order to apply equation (2.2-2) the energy of the gamma rays E_γ has to be known. In an experiment performed at the Livermore Pool Type Reactor (LPTR) for determining the radiation levels at different points within and outside of the reactor containment following drainage of the light-water-moderator coolant, it was determined by absorption techniques and pulse height analysis that the effective energy of the gamma radiation was about 150 kev [3]. Hence, by using this value in equation (2.2-2), the gamma energy emission is

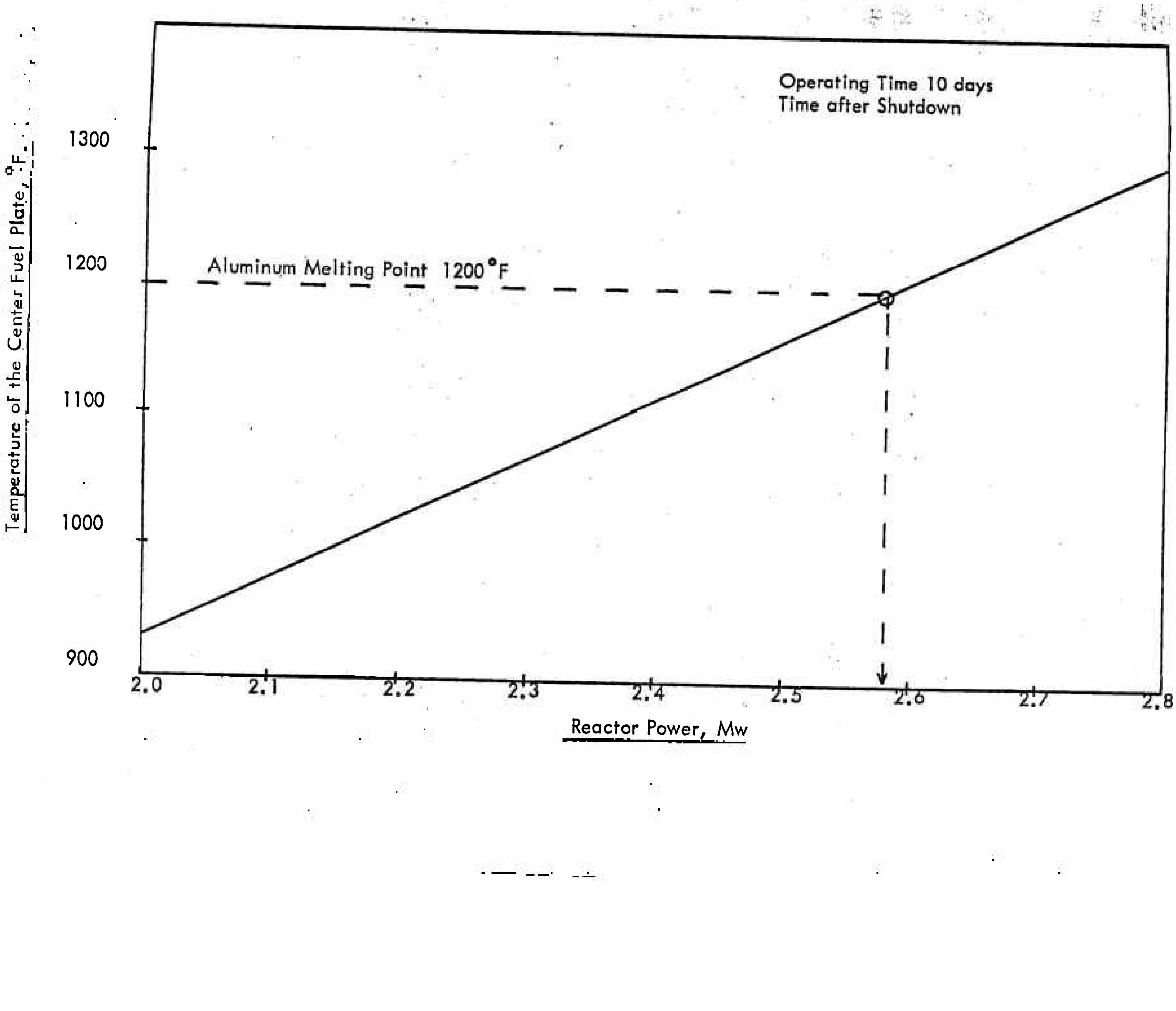
$$\gamma\text{-energy emission} = 1.6095 \times 10^{16} \ \frac{\text{Mev}}{\text{sec}}$$

The gamma-ray flux, ϕ_γ, at a distance, 12-22 ft = 670.56 cm (at the bridge), away from the point source is

$$\phi_\gamma = \frac{1.6095 \times 10^{16}}{4\pi \times (670.56)^2} = 2.8484 \times 10^{9} \ \frac{\text{Mev}}{\text{cm}^2\,\text{sec}}$$

From Fig. 9.2 in reference [4] the gamma-ray flux necessary to produce a exposure rate of 1 mr/hr for photons of energy 0.15 Mev is 6.5×10^2 Mev/cm^{2}sec, hence the exposure rate at 670.56 cm above the reactor core is 4380 R/hr.

Maximum Temperature at the Center of the Fuel Plate vs. Reactor Power



CHAPTER 3
ACCIDENTS DUE TO REACTIVITY INSERTIONS

A reactivity accident takes place whenever reactivity is added to a reactor in an uncontrolled manner and at a rate sufficient to initiate a safety system power level trip.

There are two possible ways in which changes of reactivity can occur in a reactor, those are

1. Insertion of reactivity at startup of operation.
2. Rapid insertion of reactivity (step) as a consequence of experimental facility failure, malfunction of some component, or misoperation of the reactor while at power.

3.1 Insertion of Reactivity at Startup:

This incident has been defined as one in which all of the shim safety rods are withdrawn from the core at their maximum rate of travel.

It will be assumed that all of the control and safety instrumentation fails with the exception of the level safety trips. Under these circumstances and unless the reactor is self-shutdown by an intrinsic mechanism such as the temperature or void coefficient, the power level is increased in an exponential manner until the level safety trip produces a reactor scram. The FNR is scrammed when the level exceeds 120% of the maximum operating power level (2 Mw) and/or for periods less than 5 seconds [5] .

It will be also assumed that after the level of 2.4 Mw is reached the shim safety rods will continue out during 50 MSEC and then fall into the core in 450 MSEC. These values are Technical Specification limits which are periodically verified.

An approximate mathematical analysis of startup incidents has already been studied by Hurwitz [6] and Akcasu [7] . The assumptions that were made originally by Hurwitz will be outlined and the formulas obtained by Akcasu (in regions 1 and 2 below) will be applied in order to better understand the subsequent analysis.

First, it is assumed that the reactivity is far below critical at a value of shutdown reactivity of ρ_o $\Delta k/k$ and that an external source of S_o neutrons maintain the power level at the minimum value that could exist in the reactor, that is, at the shutdown power, Ps. With these initial conditions the reactor is at steady-state prior to time t=o. At time t=o there is a linear reactivity insertion of the form

$$\rho(t) = -\rho_o + rt \qquad (3.1\text{-}1)$$

where

$\rho(t)$ = reactivity at any time $(t > o)$, $\Delta k/k$

ρ_o = shutdown reactivity, $\Delta k/k$

r = rate of reactivity insertion $(d\rho/dt)$, $\frac{\Delta k/k}{sec}$

t = time after insertion of reactivity, sec.

The history of the transient power occuring for $t>o$ has been divided in the following three intervals:

1. In the first interval $o \le t \le t_d$, where t_d is the time at which delayed criticality is approached, the reactivity increases from $\rho = -\rho_o$ to $\rho = o$ and a build-up of the power level from the shutdown value, Ps, to delayed power, Pd, takes place.
2. In the interval $t_d \le t \le t_T$, where t_T is the time

at which trip level receives the signal to shut the reactor down, the reactivity increases above delayed critical at a uniform rate (equation (3.1-1)) while the power level increases from Pd to trip power, P_T. Since this increase in power is rapid, the reactor period becomes short. Most of the power increase during the transient takes place in this interval.

3. In the third interval $t > t_T$ the safety channels have been tripped and the scram begins to reduce reactivity. Most of the energy generated takes place in this interval. Hence, to evaluate the total energy released it is necessary to know the way in which the reactivity reduction varies with time. In most applications, the safety system is designed so that the time required to return the reactor to a subcritical state is short compared to the period which has been attained at t_T. Under this condition the total energy generation will not exceed in order of magnitude the product of the trip level and the time required for the reactivity to be reduced to delayed critical. If the maximum reactivity is above prompt critical, the total energy generation will be more nearly equal to the product of the maximum power and the time required for the reactivity to be reduced to prompt critical, since the decrease in power will begin when the reactivity has fallen below prompt criticality because of the small concentration of delayed neutron emitters.

3.1.1 Approximate Mathematical Models:

The behavior of the transient power level in the first and second intervals as treated in reference (7) follows.

3.1.1.1 First Interval $0 \le t \le t_d$:

Because the reactivities are far below prompt critical, the prompt-jump approximation and the one-delayed-neutron-group model with the group parameters

$\beta^{(1)} = \beta$ = effective delayed neutron fraction, 0.00755, for U-235

and

$$\lambda^{(1)} = \bar{\lambda} \equiv \frac{\beta}{\sum_{i=1}^{6} \beta_i/\lambda_i} = 0.08123 \ sec^{-1}$$

has been used (the values of λ_i and β_i are given in table (3.1.1.1-1).

Table (3.1.1.1-1)

Delayed Neutron Fractions, β_i, and Decay Constants, λ_i, for U-235 and Six-Delayed Neutron Groups.

Delayed Neutron Group	Delayed Neutron Fraction β_i, %	Delayed Neutron Decay Constant λ_i, sec	β_i/λ_i sec
1	0.02567	0.1250	0.0205
2	0.16610	0.0315	0.0527
3	0.21910	0.1537	0.0139
4	0.24085	0.4560	0.0053
5	0.08456	1.6120	0.000525
6	0.02492	13.8629	0.000018
	$\beta = \sum_{i=1}^{6} \beta_i = 0.755$		$\sum_{i=1}^{6} \beta_i/\lambda_i = 0.09294$

A practical condition for the validity of the prompt-jump approximation is that

$$(1-\rho/\beta) \gg \left[\frac{\bar{\lambda} l}{\beta}\right]^{1/2}$$

where

l = prompt neutron generation life-time, 1.04×10^{-4} sec.

For the ramp reactivity insertion as given by equation (3.1-1), the above condition becomes

$$\beta+\rho_0-rt \gg (\beta\bar{\lambda} l)^{1/2} \qquad (3.1.1.1\text{-}1)$$

It will be shown later that this condition is in fact satisfied.

The evaluation of the power level at delayed critical, Pd, can be obtained by solving the differential equation (7)

$$\frac{dP}{dt} = P\left\{\frac{r-\rho_0\bar{\lambda}+\bar{\lambda} rt}{\beta(1+\rho_0/\beta-rt/\beta)}\right\} + \frac{l\bar{\lambda}}{\beta}\frac{S_0}{(1+\rho_0/\beta-rt/\beta)} \qquad (3.1.1.1\text{-}2)$$

subject to the initial condition P(o) = Ps and Ps given by

$$P_s = \frac{l S_0}{\rho_0} \qquad (3.1.1.1\text{-}3)$$

The solution of (3.1.1.1-2) at delayed critical, $t_d = \rho_o/r$, is

$$P_d = \left[\exp\left(-\frac{\bar{\lambda}\rho_o}{r}\right)\right]\left(1+\frac{\rho_o}{\beta}\right)^{\bar{\lambda}\beta/r}\left\{P_o + \frac{l\bar{\lambda}}{\beta} \times \frac{S_o}{[1+\rho_o/\beta]} \times \int_0^{\rho_o/r} dt'\,[\exp(\bar{\lambda}t')] \times \left(1-\frac{rt'}{\beta+\rho_o}\right)^{(\bar{\lambda}\beta/r)-1}\right\} + \frac{l\,S_o}{\beta} \qquad (3.1.1.1\text{-}4)$$

By defining the delayed neutron build-up factor, ξ , as

$$\xi \equiv \frac{P_d\,\beta}{S_o\,l} \qquad (3.1.1.1\text{-}5)$$

and solving (3.1.1.1-4) in terms of ξ after the change of variable $x = \bar{\lambda}t'$ the result is

$$\xi = 1 + \left[\frac{(1+\rho_o/\beta)}{\rho_o/\beta}\right]^{\bar{\lambda}\beta/r} e^{-\bar{\lambda}\rho_o/r} + \int_0^{\bar{\lambda}\rho_o/r} dx \left\{ e^{x} \left[1+\frac{\rho_o}{\beta}-\left(\frac{xr}{\bar{\lambda}\beta}\right)\right]^{(\bar{\lambda}\beta/r)-1} e^{-\bar{\lambda}\rho_o/r}\right\} \qquad (3.1.1.1\text{-}6)$$

Since the shutdown reactivity Po for the FNR is 0.035 and for $\beta = 0.00755$, unity can be neglected as compared to the ratio $\rho o/\beta$ in the above equation. Further, for slow rates of reactivity addition such that $r \ll \bar{\lambda}\rho_0$ (it will be shown later that it is the case in this problem) the second term of the above equation can be neglected and by replacing the upper limit of integration by infinity this equation reduces to

$$\xi = \int_0^\infty dx \left\{ e^{-x} \left(1 + \frac{x r}{\bar{\lambda}\beta} \right)^{\bar{\lambda}\beta/r} \right\}$$

It has been shown that for large values of $\bar{\lambda}\beta/r$ this equation can be approximated as

$$\xi \simeq \frac{2}{3} + \left(\frac{1}{2} \frac{\bar{\lambda}}{r} \beta \pi \right)^{1/2} \qquad (3.1.1.1\text{-}7)$$

By combining equations (3.1.1.1-3) and (3.1.1.1-5) and solving for Pd, the power level at delayed critical is

$$P_d = \frac{\rho_0}{\beta} \xi P_s \qquad (3.1.1.1\text{-}8)$$

3.1.1.2 Second Interval $t_d \le t \le t_T$:

For slow insertions of reactivity the power level in this region can be obtained from the equation (7)

$$P(t) = P_0 F(\rho) \exp\left\{ \frac{1}{r} \int_0^\rho \omega_0(\rho')\, d\rho' \right\} \qquad (3.1.1.2\text{-}1)$$

where

$$P_0 = \frac{\rho_0}{\beta} \xi P_s , \qquad \text{watts}$$

$$\omega_0(\rho) = \text{inverse reactor period, sec}^{-1}$$

$$F(\rho) \equiv \left\{ \left[\ell + \sum_{i=1}^{6} (\beta_i/\lambda_i) \right] \Big/ \left[\ell + \sum_{i=1}^{6} \frac{\beta_i \lambda_i}{(\lambda_i + \omega_o)^2} \right] \right\}^{1/2} \quad (3.1.1.2\text{-}2)$$

The integral in the exponent of equation (3.1.1.2-1) represents the integrated inverse reactor period from delayed critical up to a reactivity, ρ, and it can be written as

$$\int_0^{\rho} \omega_o(\rho')d\rho' = \int_0^{\omega_o} d\omega \left[\omega \frac{d\rho}{d\omega} \right]$$
$$= \omega_o \rho - \int_0^{\omega_o} \rho(\omega)\, d\omega \quad (3.1.1.2\text{-}3)$$

where

ω_o = instantaneous inverse reactor period, sec^{-1}.

If the reactivity ρ is expressed explicitly by the inhour equation (7), namely

$$\rho(\omega_o) = \omega_o \left\{ \ell + \sum_{i=1}^{6} \frac{\beta_i}{\lambda_i + \omega_o} \right\} \quad (3.1.1.2\text{-}4)$$

then equation (3.1.1.2-3) becomes

$$\int_0^{\rho} \omega(\rho')d\rho' = \frac{\omega_o^2 \ell}{2} + \sum_{i=1}^{6} \beta_i \lambda_i \left\{ \ln\left(1 + \frac{\omega_o}{\lambda_i}\right) - \frac{\omega_o}{\lambda_i + \omega_o} \right\} \quad (3.1.1.2\text{-}5)$$

Substitution of Po, (3.1.1.2-2) and (3.2.2.2-5) into (3.1.1.2-1) yields

$$P(t) = \frac{\rho_0}{\beta} \xi P_s \left\{ \left[l + \sum_{i=1}^{6} (\beta_i/\lambda_i) \right] / \left[l + \sum_{i=1}^{6} \frac{\beta_i \lambda_i}{(\lambda_i + \omega_0)^2} \right] \right\}^{1/2} \times$$

$$\times \frac{\omega_0^2 l}{2} + \sum_{i=1}^{6} \beta_i \lambda_i \left\{ \ln\left(1 + \frac{\omega_0}{\lambda_i}\right) - \frac{\omega_0}{\lambda_i + \omega_0} \right\} \qquad (3.1.1.2\text{-}6)$$

3.1.2 Power Levels and Reactor Period Calculations:

With the background given in the preceding section the power level for the first and second intervals and the reactor period at trip level can be evaluated.

The FNR has 3-shim safety rods, 1% Boron in stainless steel, with an active length of 24 inch (9). The total worth of the rods is 0.075 $\Delta k/k$ and they travel with a normal speed of 24"/9 min. The average rate of reactivity insertion is 0.0139% $\Delta k/k/sec$. Assuming that the greatest worth is at 12" the maximum rate of reactivity insertion, r, due to the rods is 0.0277% $\Delta k/k/sec$.

3.1.2.1 Interval $0 \leq t \leq t_d$:

From equation (3.1.1.1-7) the value of ξ is

$$\xi = \frac{2}{3} + \left(\frac{1}{2} \, \frac{0.08123}{0.0277\,\%} \times 0.755\,\% \times \pi \right)^{1/2} = 2.53$$

From equation (3.1.1.1-8) the power level at delayed critical is

$$P_d = \frac{2.53 \times 0.035 \times 0.8}{0.00755} = 9.39 \text{ watts} \qquad \text{(Footnote)}$$

set ρ (td)=0 in equation (3.1.1) and solve for td, the time at which delayed criticality is reached

$$t_d = \frac{\rho_0}{r} = \frac{3.5\,\%\ \Delta k/k}{0.0277\,\%\ \frac{\Delta k/k}{sec}} = 126.35 \text{ sec}$$

Notice that the condition (3.1.1.1-1), for the validity

Footnote: The value of the shutdown power Ps = 0.8 watts was obtained in the "Control Rod Worth Calibration and shutdown Power at the FNR" experiment performed in the course NE-445, Fall 1976.

of the prompt-jump approximation is in fact satisfied for any time in this interval, i.e., for t=o it gives 3.58>> 0.00025 and for t=td gives 0.0075>> 0.00025. Also notice that the value of the rate of reactivity addition satisfies $r \ll \bar{\lambda} \rho_0$, i.e., 0.0277% $\frac{\Delta k/k}{sec} \ll$ 0.29 $\frac{\Delta k/k}{sec}$.

3.1.2.2 Interval $t_d \le t \le t_r$:

The power level can be obtained from equation (3.1.1.2-1) with the values of F (p) and $\int_0^{\rho} \omega_0(\rho')\, d\rho'$, equations (3.1.1.2-1) and (3.1.1.2-5) respectively, as a function of the reactivity given by different values of the instantaneous inverse period through the inhour equation (3.1.1.2-4) and using the data in table (3.1.1.1-1). The values of ρ (w), F (p) and $\int_0^{\rho} \omega_0(\rho')\, d\rho'$ are included in tables (3.1.2.2-1) and (3.1.2.2-3) and plotted in figures (3.1.2.2-1) through (3.1.2.2-3) respectively.

Table (3.1.2.2-1)

Solution of Inhour Equation for U-235 and 6-Delayed Neutron Groups.

Inverse Period ω_0, sec^{-1}	Reactivity ρ, % $\Delta k/k$	$\omega_0 \sum_{i=1}^{6} \beta_i/(\lambda_i+\omega_0)$
10^{-3}	9.000×10^{-3}	9.000×10^{-5}
10^{-2}	7.020×10^{-2}	7.010×10^{-4}
10^{-1}	0.282	2.810×10^{-3}
1	0.579	5.690×10^{-3}
5	0.734	6.820×10^{-3}
10	0.808	7.040×10^{-3}
10^{2}	1.800	7.580×10^{-3}
10^{3}	11.100	7.300×10^{-3}

Reactivity vs. Inverse Reactor Period for ^{235}U (6-Delayed Neutron Groups)

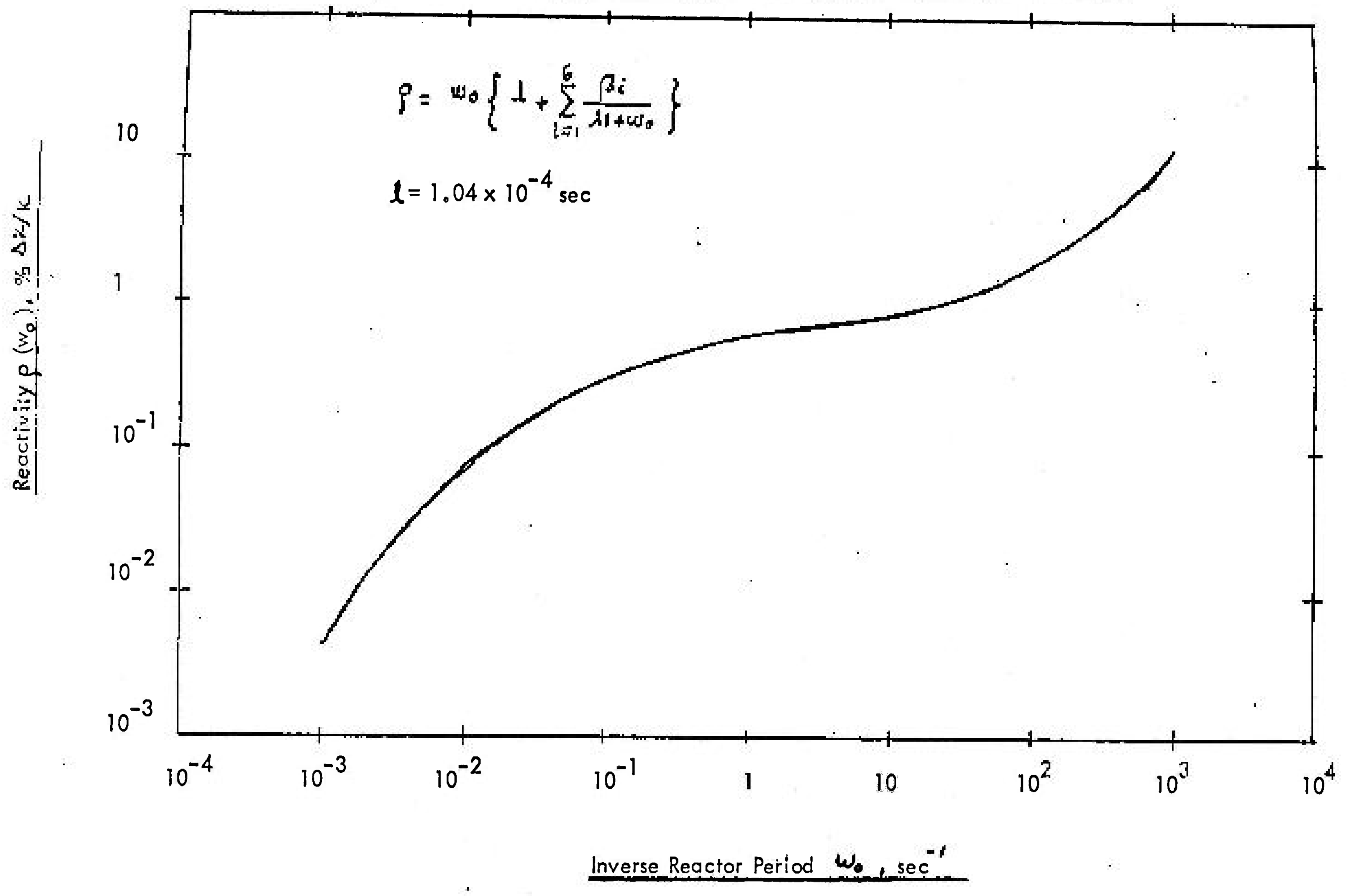


Table (3.1.2.2-2)

Values of F (ρ) for U-235 and 6 - Delayed Neutron Groups

Reactivity ρ, % $\Delta k/k$	$\sum_{i=1}^{6} (\beta_i \lambda_i)/(\lambda_i + \omega_0)^2$	F (ρ)
9.000×10^{-3}	0.0866	1.0358
7.020×10^{-2}	0.0545	1.3052
0.282	0.0124	2.7297
0.579	0.0010	9.1137
0.734	8.26×10^{-5}	22.3
0.808	2.39×10^{-5}	27.0
1.8	1.1×10^{-6}	29.8
11.1	1.13×10^{-8}	29.9

F (ρ) vs Reactivity for ^{235}U and 6-Delayed Neutron Groups

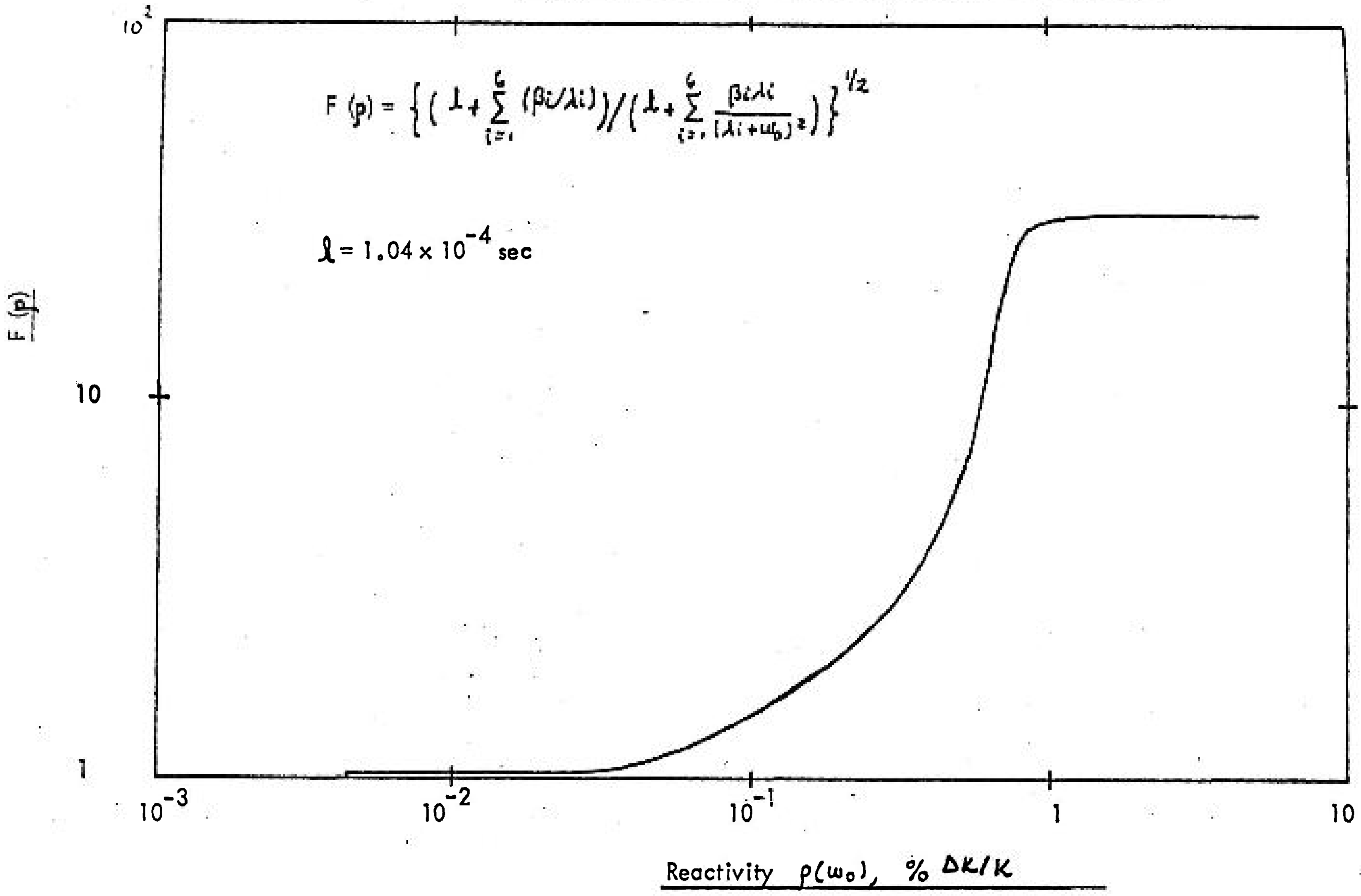


Table (3.1.2.2-3)

Values of $\int_0^{\rho} \omega_0(\rho')\,d\rho'$ for U-235 and 6-Delayed Neutron Groups

Reactivity ρ, % $\Delta k/k$	Σ^* sec^{-1}	$\omega_0^2 \ell/2$ sec^{-1}	$\int_0^{\rho} \omega_0(\rho')\,d\rho'$ sec^{-1}
9.000×10^{-3}	4.43×10^{-8}	5.2×10^{-11}	4.435×10^{-8}
7.020×10^{-2}	3.2×10^{-6}	5.2×10^{-9}	3.205×10^{-6}
0.282	9.68×10^{-5}	5.2×10^{-7}	9.732×10^{-5}
0.579	1.18×10^{-3}	5.2×10^{-5}	1.232×10^{-3}
0.734	3.67×10^{-3}	1.3×10^{-3}	4.970×10^{-3}
0.808	5.22×10^{-3}	5.2×10^{-3}	1.042×10^{-2}
1.8	1.13×10^{-2}	5.2×10^{-1}	5.313×10^{-1}
11.1	1.78×10^{-2}	5.2×10	5.202×10

$$^* \quad \Sigma = \sum_{i=1}^{6} \beta_i \lambda_i \left\{ \ln\left(1 + \frac{\omega_0}{\lambda_i}\right) - \frac{\omega_0}{\lambda_i + \omega_0} \right\}$$

The transient power level values, equation (3.1.1.2-1), using interpolated values of F (p) and $\int_0^{\rho} \omega_0(\rho')\,d\rho'$ from their respective figures are included in table (3.1.2.2-4) and plotted in Fig. (3.1.2.2-4).

Figure (3.1.2.2-3)

Reactivity vs. $\int_0^{\rho} \omega_0(\rho')\,d\rho'$ for ^{235}U and 6-Delayed Neutron Groups

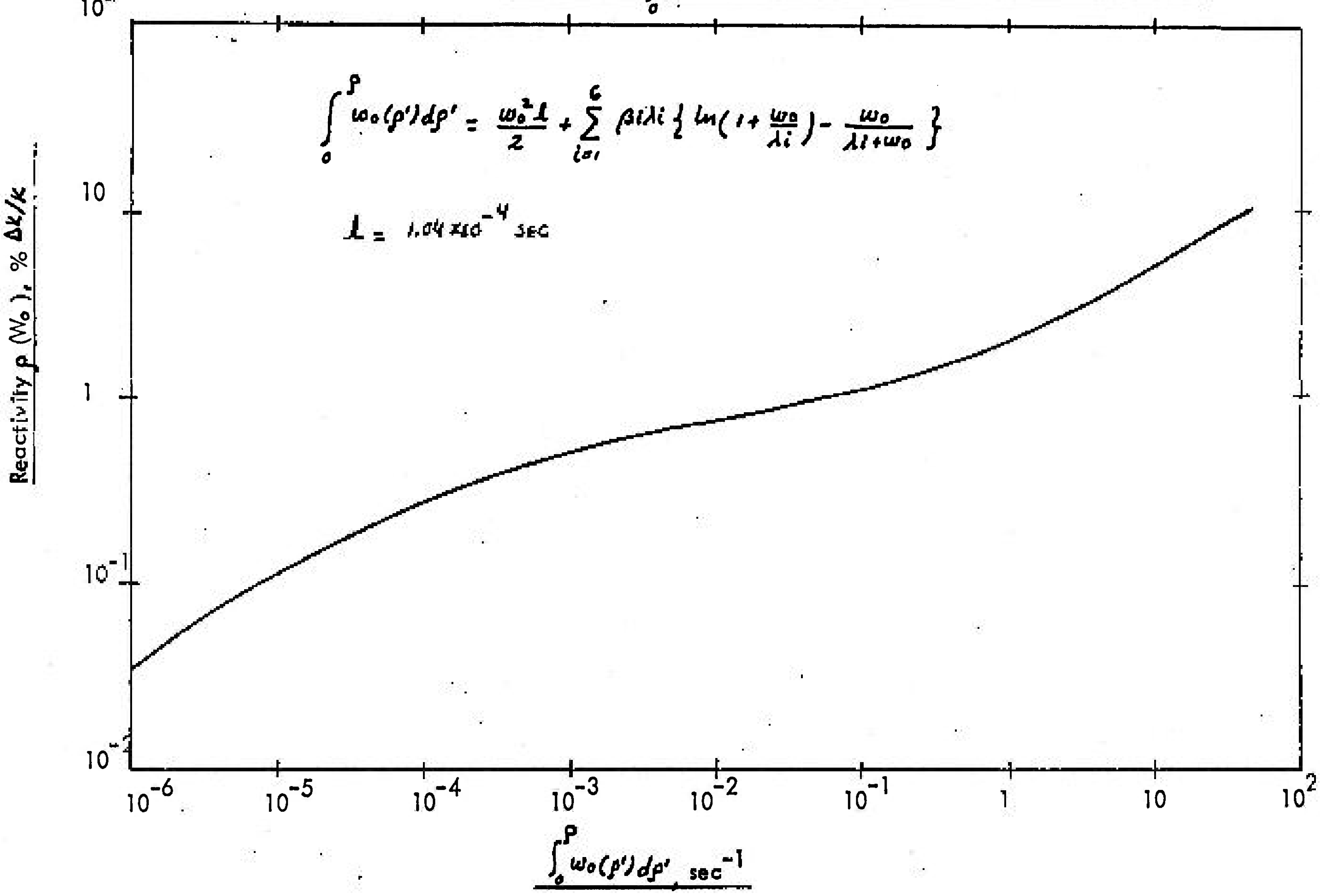


Table (3.1.2.2-4)

Values of Power Level as a Function of Excess Reactivity

Reactivity ρ, % $\Delta K/K$	F (ρ)	$\int_0^{\rho} \omega_b(\rho')\,d\rho'$ sec^{-1}	P (t) watts
9.000×10^{-3}	1.0358	4.44×10^{-8}	9.73
7.02×10^{-2}	1.3052	3.21×10^{-6}	1.24×10
0.282	2.7297	9.79×10^{-5}	3.65×10
0.579	9.1137	1.28×10^{-3}	8.69×10^{3}
0.650	15	2.5×10^{-3}	1.17×10^{6}
0.700	18	3.0×10^{-3}	8.54×10^{6}
0.720	21	3.5×10^{-3}	6.058×10^{7}
0.734	22.3	4.97×10^{-3}	1.297×10^{10}
0.808	7	1.56×10^{-2}	7.286×10^{26}

Fig. (3.1.2.2-4) shows that when the trip level is approached the excess reactivity is about $\rho_T = 0.0067 \Delta k/k$. Knowing the value of ρ_T, it is possible to evaluate the reactor period τ at t_T by taking the natural logarithm of equation (3.1.1.2-1), that is

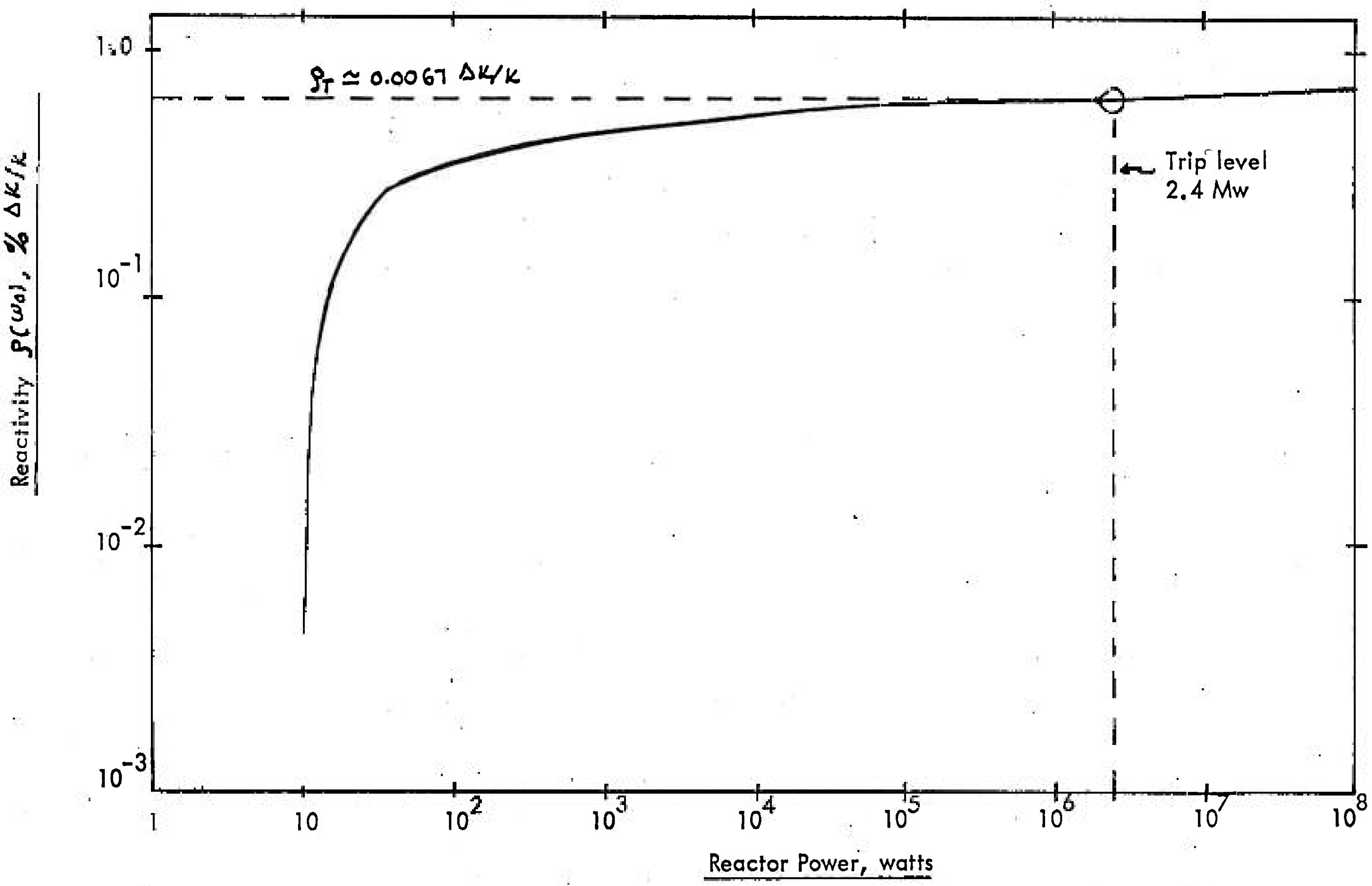
$$\int_0^{\rho_T} \omega(\rho')\,d\rho' = r \ln \frac{P_T}{P_0\,F(\rho_T)} \qquad (3.1.2.2\text{-}1)$$

The reciprocal of (3.1.2.2-1) is the reactor period at t_T.

A comparison of the values of $\omega_b^2 \ell/2$ and the Σ given in table (3.1.2.2-3) for a trip reactivity of 0.0067 $\Delta k/k$ indicates that the later is about one order of magnitude greater than the former. Hence, equation (3.1.1.2-5) can be approximated by writing

Figure (3.1.2.2-4)

Reactivity vs Power Level



$$\int_0^{\rho_T} \omega(\rho')\,d\rho' \simeq \frac{\omega_T^2 \ell}{2} + 10\,\frac{\omega_T^2 \ell}{2}$$

$$\simeq 5\,\omega_T^2 \ell$$

Substitution of this value into equation (3.1.2.2-1) yields:

$$5\omega_T^2 \ell \simeq r \ln \frac{P_T}{P_0 F(\rho_T)}$$

and solving for the instantaneous inverse reactor period ω_T

$$\omega_T \simeq \left(\frac{r}{5\ell}\right)^{1/2} \left\{ \ln \frac{P_T}{P_0 F(\rho_T)} \right\}^{1/2}$$

$$= \left(\frac{0.000277}{5 \times 1.04 \times 10^{-4}}\right)^{1/2} \left(\ln \frac{2.4 \times 10^6}{9.39 \times 16} \right)^{1/2}$$

$$= 2.27 \text{ sec}^{-1}$$

The reactor period at trip level is

$$\tau = \frac{1}{\omega_T} = 440 \text{ msec}$$

The time at which the signal for initiating the scram is received can be obtained from equation (3.1-1) with $\rho(t_T) = \rho_T = 0.0067\,\Delta k/k$, $\rho_0 = 0.035\,\Delta k/k$ and $r = 0.0277\%\,\Delta k/k$, this value is

$$t_T = \frac{0.0067 + 0.035}{0.000277} = 150.54 \text{ sec}$$

Notice that during the elapsed time between delayed critical and t_T, 24.19 sec, the power increases by a factor of 10^6 while in the first interval (126.35 sec) it increases a single order of magnitude.

3.1.2.3 Interval $t > t_T$:

If no reactivity compensation other than that due to the shim rods is added to the system and if it is assumed that after the trip level the power continues increasing at a constant period of 440 msec until the safety mechanism begins to actuate (50 msec), the maximum power approached during the excursion is (10)

$$P_m = P_T\, e^{t/\tau}$$
$$= 2.4\, e^{50/400}$$
$$= 2.69 \text{ Mw}$$

The time response of the FNR scram system and the shim rods is 24"/450 msec. The negative rate of reactivity insertion, r', is

$$r' = \frac{0.075\ \Delta k/k}{0.45 \text{ sec}} = 0.1667\ \frac{\Delta k/k}{\text{sec}}$$

The time required to reduce reactivity from $\rho_T = 0.00677\ \Delta k/k$ to delayed critical $\rho = 0$ is, from (3.1-1),

$$t_r = \frac{0.0067}{0.1667} = 0.0402 \text{ sec.}$$

Since $t_r < \tau$ then according to the arguments given in section 3.1 the energy generation, $\mathcal{E}$, in the third interval is given as

$$\mathcal{E} = P_m t_r = 2.69 \text{ Mw} \times 0.0402 \text{ sec}$$

$$= 0.1081 \text{ Mw-sec} .$$

The values for the period and energy release are significantly different from the typical ones that have been measured during the study of nuclear power excursions such as the ones performed in the Borax (11) and Spert-1 (12) destructive tests. Typical period values are of the order of a few milliseconds ($<$ 10 msec) and are obtained by inserting reactivities of several orders of β instantaneously. In one of the tests at Spert-1 it was observed that periods as small as 4.6 msec released a total energy of 19 Mw-sec and the maximum fuel temperatures approached the limit of the aluminum melting point. The step insertion in this test was 0.0204 $\Delta k/k$. In the shortest period test (3.2 msec) the melting of the core was estimated to be about 8% in the fuel plates and a total of 35% of the core. The energy release was about 30 Mw-sec. Hence, the reactivity insertion dealt with in this analysis (0.0277% $\Delta k/k/sec$) is rather safe and conservative because no feed-back effects to compensate reactivity have been considered.

3.2 Step Reactivity Insertion:

The case in which a step reactivity insertion occurs is in general less realistic than the ramp insertion of reactivity because of the limitations inherent to the shim rod withdrawl mechanism. However, as an illustration, the consequences due to a rapid insertion of reactivity will be considered.

It will be assumed that the reactor is operating at a steady-state power level of 2 Mw and that it is critical prior to time t=o. At t=o a sudden insertion of reactivity of 0.012 $\Delta k/k$ occurs. This value corresponds to the maximum experimental worth allowed by Technical Specifications. Further, it will be assumed that when the reactor power reaches the trip level of 2.4 Mw, the safety mechanism is tripped and the shim rods remain out for 50 msec after which they fall down into the core at a speed of 24"/ 450 msec.

The reactor period in the case of step reactivity insertions is customarily expressed as

$$\tau \simeq \frac{l}{\rho - \beta}$$

where all the symbols have already been defined and the reactivity is $\rho = 0.012\ \Delta k/k$. Then, the numerical value for τ becomes

$$\tau = \frac{1.04 \times 10^{-4}}{0.012 - 0.00755} = 0.0234 \text{ sec}$$

The power level up to trip level $P_T = 2.4$ M/w can be written as

$$2.4 = 2\, e^{t_T/0.0234}$$

and solving for t_T

$$t_T = 0.0234 \ln \frac{2.4}{2} = 4.26 \text{ msec}$$

Since $\rho > \beta$ the energy released can be obtained as the product of the maximum power level times the time, t_{pc} , required to reduce the reactivity to prompt critical $(\rho = \beta)$. The maximum power level, in analogy to the case of ramp insertion, is

$$P_{max} = 2.4\, e^{0.05/0.0234} = 20.33 \text{ Mw}$$

The time t_{pc} is

$$t_{pc} = \frac{\rho_{pc} - \rho_T}{-r'} = \frac{0.012 - 0.00755}{0.1667}$$

$$t_{pc} = 0.0267 \text{ sec}$$

and the energy release is

$$\mathcal{E} = P_{max}\, t_{pc}$$
$$= 20.33 \times 26.7$$
$$= 0.5429 \text{ Mw-sec}.$$

Comparison of these results with the SPERT-1 3.2 msec period test indicates that even with this step insertion, no fuel damage will result.

CHAPTER 4

PRIMARY COOLING SYSTEM FAILURE

If a failure of the primary coolant pump occurs or if the primary flow rate is decreased for any reason, a reactor scram will be demanded by a loss of flow signal to the safety system. In this event the neutron density in the core will decrease rapidly and eventually will decay with a period of 80 seconds.

Since in this accident the core will always be covered with water the expected maximum fuel plate temperatures will not approach the values obtained in Chapter 2 considering the same operating conditions.

The analysis of this accident will be only concerned with the power history of the reactor after the scram function commences to actuate. It will be assumed that the reactor is critical and operating at a steady-state power level of 2Mw prior to t = o. A sudden insertion of negative reactivity takes place at t = o and the final reactivity is the shutdown reactivity, that is, -0.035 $\Delta k/k$. Because of the short time behavior of the insertion of reactivity, the problem can be treated by using the prompt-drop approximation.

This approximation states that the drop in power from an initial steady-state value of P_1 to a final level of P_2 following a negative insertion of reactivity ρ_2 is given by (13)

$$\frac{P_2}{P_1} = \frac{\beta - \rho_1}{\beta - \rho_2} \qquad (4.1)$$

where

ρ_1 = critical reactivity prior to t = o, zero.

P_1 = 2 Mw

ρ_2 = -0.035

Solving for P_2 it becomes

$$P_2 = 2\left\{\frac{0.00755}{0.00755 + 0.035}\right\} = 0.3549 \text{ Mw.}$$

After this drop the power goes on a stable period corresponding to the first root of the inhour equation (3.1.1.2-4), that is, it is the inverse of the decay constant of the longest lived delayed neutron precursor ($\lambda = 0.0125 \ sec^{-1}$) . Hence, the shortest stable (negative) reactor period possible after the drop in power is 80 sec and the transient power behavior after the drop will be expressed by

$$P(t) = P_{dr} \, e^{-t/\tau} \qquad (4.2)$$

where

$P_{dr} = P_2$ = power level after drop, 0.3549 Mw.

If it is assumed that the reactor is shutdown when the power is reduced from P_{dr} = 3.549 x 10^5 watts to P_s = 0.8 watts, then the minimum shutdown time is from equation (4.2) and with $P(t) = P_s$

$$t_s = -\tau \ln \frac{P_s}{P_{dr}} = 938.88 \text{ sec} = 15.65 \text{ min}$$

Because of the emission of gamma-rays from the fission products this time is in general prolonged.

CHAPTER 5

FUEL ELEMENT CLADDING FAILURE

Fuel element cladding failure might result from corrosion, erosion, mechanical defects, and local melting due to excessive temperatures. Since the first barrier to the release of fission products is the fuel element cladding, its failure would cause the leaching of the fission products from the fuel region due to the action of the primary coolant.

In Chaper 2 it was shown that no melting of the fuel element cladding would occur following a LOCA type accident. Whatever the cause of rupture is, fission products would be released to the pool water, if the core were still covered with water, and then to the reactor containment.

In this accident it will be assumed that the cladding fails due to defects in manufacture. The conditions under which the accident takes place will be treated in Chapter 7 while the public radiological consequences will be the subject of Chapter 9.

CHAPTER 6

FUEL ELEMENT BLOCKAGE

Fuel element blockage incidents have occurred in many operating reactors as a consequence of accidental introduction of foreign materials into the reactor core, such as neoprene, plastics and soft materials in general, which block the flow of water through the cooling channels or which coat their surfaces thus inhibiting heat transfer.

The amount of melted fuel due to such blockages has been found to be small, i.e., 12.4 g of U-235, contained in 134 g of alloy were lost at the Engineering Test Reactor (ETR) at Idaho Falls in a fuel element flow blockage accident (14). The release of activity in that accident was estimated to be about 6.4 curies to the atmosphere. 42 curies were released to the leaching pond as liquid waste. However, these results would be excessively conservative if applied to the FNR case since the ETR accident occurred while operating at a power level of 90 Mw.

In the case of research reactors, the Oak Ridge Research Reactor (ORR) fuel melting incident (15) demonstrated that detectable fluctuations began at power levels of 9 Mw and burnout at about 12 Mw. The fuel element involved in this incident was almost completely blocked at the top. Hence, if a similar blockage occurred at the FNR while operating at 2 Mw, it would not be likely to cause detectable fluctuations or melting.

FNR operating experience has shown that small period fluctuations due to the insertion of experiments with reactivity worths of the order 0.001 $\Delta K/K$ and which are done slowly and carefully have produced the effect of fuel element blockage, i.e. decrease in neutron moderation as would happen if boiling occurred, and subsequent period scrams.

The FNR control rod is modelled as a hollow, aluminum void which occupies a volume of half an element. In fact the control rod is a water filled aluminum shell which occupies a much smaller volume. The reasoning that follows is therefore quite conservative. Control rod worth is 0.003 $\Delta K/K$. It can be reasoned that since the control rod model is worth 0.003 $\Delta K/K$ and occupies 1/2 of a fuel element, a reactivity worth of -0.001 $\Delta K/K$ results from a void of 1/6 of an element. Thus, a rapidly produced void caused by fuel element blockage boiling that encompasses 1/6 of a fuel element or 1/210 of a 35 element core would cause a reactor period less than 5 seconds and safely scram the reactor before core damage occurred.

CHAPTER 7

MAXIMUM CREDIBLE ACCIDENT WITH PUBLIC RADIOLOGICAL CONSEQUENCES

It has been shown that none of the accidents analyzed results in a public radiological hazard, except for the case of a fuel element cladding failure. Such a failure is a potential radiological hazard because of the radionuclides that could be released from the reactor pool and containment thus affecting the public outside the reactor building. The degree of the hazard will obviously depend upon the size of the rupture and in the retention of radioactive isotopes by the pool-water and the reactor containment. Hence, it will be postulated that the maximum credible accident that could occur at the FNR is a rupture of the cladding due to manufacturing defects.

It will be assumed that the reactor core inventory at the time of the accident is the equivalent of a 2 Mw reactor that has been operating for a one year period and that the rupture takes place in a 35 element core, with 18 plates per element. As the worst situation, 10% of one plate ruptures. The total fraction of the core released in this incident is 1.59×10^{-2}%. After fission product released only I, Xe and Kr are volatile and leak from the pool and containment. Once in the pool-water they will be released to the reactor containment at a rate determined by the I-131 pool-leak rate and from the containment to the atmosphere at the containment-leak rate. The values of these leak rates have been experimentally determined and the bases and results of the experiments are the subject of Chapter 8.

A computer program following American Nuclear Society (ANS) and U.S. Nuclear Regulatory Commission (NRC) guidelines has been developed (16). The program calculates the concentration of radioactive effluents emanating from a nuclear reactor and public doses as a consequence of a reactor core meltdown accident and as a function of the distance from the reactor site, wind direction, wind velocity, and atmospheric conditions. The program calculates the cumulative internal and external body dose caused by exposure to any selected by-product isotopes that represent a potential radiological hazard.

The results of the program and their analysis for the reactor core inventory given in table (7.1) will be treated in Chapter 9.

Table (7.1)

FNR Core Inventory After 1 Year of Operation at 2 Mw (17)
and for 100% of Iodines, Xenon and Kryton
and for a Total Fraction of the Core of 1.59×10^{-2}%

Radioisotope	λ sec^{-1}	Activity megaCi
Kr	Stable	
85	2.14×10^{-9}	3.18×10^{-8}
85 M	4.38×10^{-5}	2.544×10^{-6}
87	1.48×10^{-4}	4.77×10^{-6}
88	6.88×10^{-5}	7.314×10^{-6}
Xe	Stable	
133 M	3.47×10^{-6}	3.18×10^{-7}
133	1.52×10^{-6}	1.7172×10^{-5}
135	2.0892×10^{-5}	1.0812×10^{-5}
I	Stable	
131	1.0026×10^{-6}	7.95×10^{-6}
132	8.369×10^{-5}	1.2084×10^{-5}
133	9.166×10^{-6}	1.749×10^{-5}
134	2.22×10^{-4}	2.0034×10^{-5}
135	2.873×10^{-5}	1.749×10^{-5}

CHAPTER 8

EXPERIMENTAL DETERMINATION OF THE POOL-LEAK RATE AND THE CONTAINMENT-LEAK RATE AT THE FNR

8.1 Pool-Leak Rate:

The experiments performed in determining the leak rate of gaseous fission products from the reactor pool were based on the following assumptions:

1. In the event of fuel element fission product release, the leak rate of radio-iodine, I-131, from the pool will represent the leak rate of all fission products. This assumption is based on the fact that I-131 is the fission product that represents the most serious radiological hazard when it is introduced by any means into the body.
2. The effective decay constant, λ_{eff}, for I-131 is given by

$$\lambda_{eff} \equiv \lambda + L_p \qquad (8.1\text{-}1)$$

where

λ = I-131 decay constant, 0.0863 day^{-1}.

L_p = I-131 pool leak rate, day^{-1}.

The evaluation of λ_{eff} can be accomplished by taking periodical measurements of the counting rate R_i (cps) for a I-131 sample and by plotting the counting rate R_i relative to the initial counting rate R_0 vs. time for each i^{th} time. The curve obtained is an exponential decay curve with a decay constant λ_{eff}. Knowing λ_{eff}, the evaluation of the pool-leak rate, L_p, becomes immediate from (8.1-1) as

$$L_p = \lambda_{eff} - \lambda \qquad (8.1\text{-}2)$$

8.1.1 Experimental Procedure and Results

Since the experiments were not performed directly in the FNR pool, it was required to simulate as close as possible the experimental conditions to those of the FNR. A sketch of the experimental equipment installed in one of the facility exhaust hoods is shown in Fig. (8.1.1-1).

Two aluminum cans containing one gallon of pool-water were used. They had a layer of plastic-paint (Pyroxcote) on their interior walls which simulated the protective layer of ceramic tile on the FNR tank walls. The cans were put on a hot plate which maintained the temperature of the pool-water between 102 - 108°F during the time of the experiments (13 days without interruption). The cans were equipped with thermometers and above each of them a source of make-up pool-water was installed. A constant drip rate of water compensated for the decrease in water level due to evaporation. Should the water level increase above the one gallon limit a hole was drilled on each of the cans just above that limit and the exhaust of water was through a plastic tube ending in a glass bottle. The two cans were identical except that the surface of one was covered with 40 $m\ell$ of vegetable oil. A reduction of the leak rate by absorption of I-131 in the oil can was expected.

Two 1 $m\ell$ samples of I-131 as sodium iodide in NaOH with initial specific activities of 182 $\mu Ci/m\ell$ and 218 $\mu Ci/m\ell$ were added to the plain and oil cans respectively. The specific activities in each can were 0.048 $\frac{\mu Ci}{m\ell}$ and 0.0576 $\frac{\mu Ci}{m\ell}$ respectively.

Three 1 $m\ell$ samples from each can were taken and counted for times ranging between 400 and 500 seconds to have an overall statistical error of 2%. The counts were recorded by using the ND-4420 multichannel analyzer at the FNR. The areas under the 364 kev photo-peak for I-131 were recorded for each sample and corrected by the decay time between the instant in which they were taken out from the cans and the time at which they were counted.

This procedure was followed during 13 days taking samples almost daily. The experimental counting rates R_i are given in table (8.1.1-1). They do not include the statistical deviations since those were as small as 2%. The values of R_i are averages of the 3 samples for each of the conditions plain and oil. In that table, the relative counting rates R_i/R_o and the relative theoretical values for $\lambda = 0.0863$ day^{-1} are also included. The values of R_i as given in table (8.1.1-1) were fitted to an exponential curve using the method of least squares. The fitted values of Ri and R_i/R_o are given in table (8.1.1-2).

Figure (8.1.1-1)

Equipment for Determining ^{131}I Pool-Leak Rate

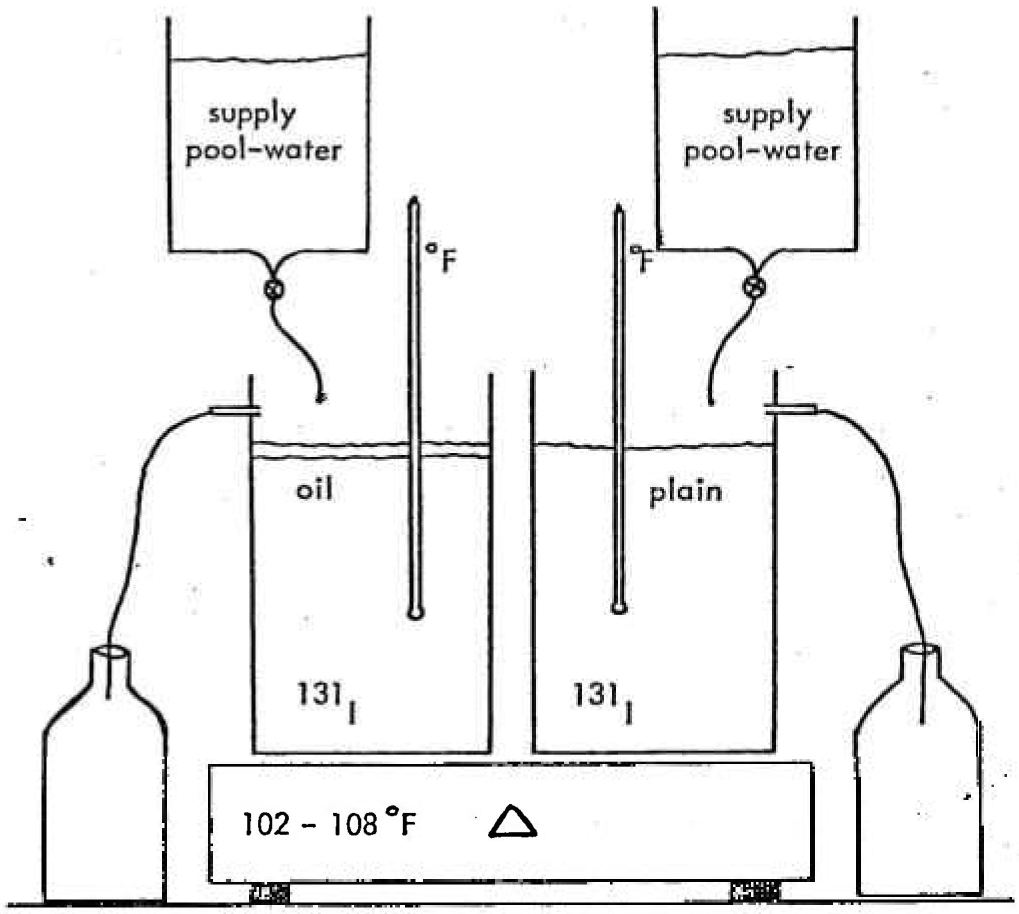


Table (8.1.1-1)

Counting Rates and Relative Experimental and Theoretical Counting Rates

Time i^{th} day	Counting Rate R_i, cps		Relative Counting Rate R_i/R_o		
	Plain	Oil	Plain	Oil	Theoretical
0	7.46	7.71	1.00	1.00	1.00
1	6.58	6.40	0.88	0.83	0.92
2	5.51	5.40	0.74	0.70	0.84
3	5.87	5.37	0.79	0.69	0.77
4	5.16	4.76	0.69	0.62	0.71
5	4.13	4.06	0.55	0.53	0.65
7	3.80	3.41	0.51	0.44	0.55
10	2.66	2.45	0.36	0.32	0.42
12	1.81	1.84	0.24	0.24	0.36
13	1.70	1.82	0.23	0.24	0.33

Table (8.1.1-2)

Exponential Fitted Values of Experimental Counting Rates and Relative Counting Rates

Time i^{th} day	Counting Rate R_i, cps		Relative Counting Rate R_i/R_o	
	Plain	Oil	Plain	Oil
0	7.581	7.261	1.0	1.0
1	6.781	6.505	0.895	0.896
2	6.065	5.827	0.800	0.803
3	5.425	5.221	0.716	0.719
4	4.853	4.667	0.640	0.644
5	4.341	4.191	0.573	0.577
6	3.882	3.754	0.512	0.517
7	3.473	3.364	0.458	0.463
8	3.106	3.013	0.409	0.415
9	2.778	2.699	0.366	0.372
10	2.485	2.418	0.328	0.333
11	2.223	2.167	0.293	0.298
12	1.988	1.941	0.262	0.267
13	1.778	1.740	0.235	0.239

The relative counting rate values from table (8.1.1-2) and the theoretical ones from table (8.1.1-1) are plotted as a function of time in Fig. (8.1.1-2). These figures also include the experimental relative values.

The exponential fitting curves for the plain and oil conditions are given by the equations:

$$R^{p}(t) = 7.581\, e^{-0.115\, t} \qquad (8.1.1\text{-}1)$$

and

$$R^{oi}(t) = 7.2607\, e^{-0.1099\, t} \qquad (8.1.1\text{-}2)$$

where

$R^{p}(t)$ = counting rate for plain samples, cps.

$R^{p}(o)$ = Ro = 7.581 cps

λ^{p}_{eff} = effective decay constant for plain samples, 0.1115 day^{-1}

$R^{oi}(t)$ = counting rate for oil samples, cps.

$R^{oi}(o)$ = R^{oi}_{o} = 7.2607 cps

λ^{oi}_{eff} = effective decay constant for oil samples, 0.1099 day^{-1}

t = time, days

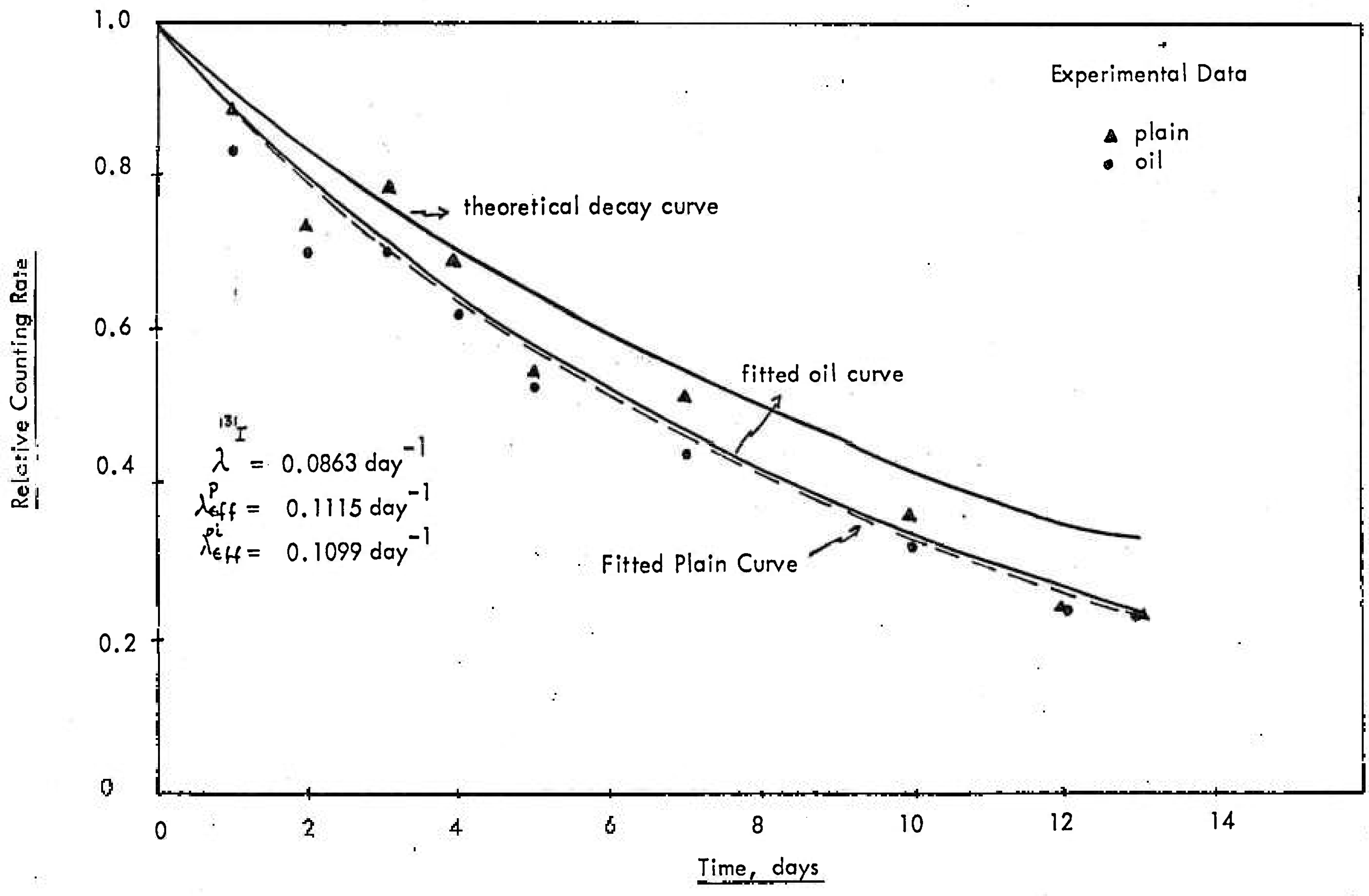
Equations (8.1.1-1) and (8.1.1-2) fit with the experimental data within 0.983 and 0.993 respectively. Using equation (8.1-2) and the values of λ_{eff} for each condition, the respective pool-leak rates are:

plain samples: L^{p}_{p} =0.1115-0.086 = 0.0252 day^{-1}

and

Figure (8.1.1-2)

Relative Counting Rate vs Time



$$\text{oil samples:} \quad L_p^{oil} = 0.1099 - 0.0863 = 0.0236 \text{ day}^{-1}$$

Once the experiments were concluded, samples of 1ml of oil from a remaining total of 25ml were counted and it was found that the retention of I-131 by the oil was only about 1.08%. which is $\cong 10^{-2}$, the ratio L_p^p / L_p^{oil}. The plastic layers in each can were also counted and a total of 10% I-131 in each one was found. In order to have conservative results the above values for L_p^p and L_p^{oil} will not be corrected by this 10%.

8.2 Reactor Containment - Leak Rate:

A sketch of the Ford Nuclear Reactor Building (FNR) and the Phoenix Memorial Laboratory Building (PML) is shown in Fig. (8.2-1).

Initially the Building Supply Damper (BSD), the Building Ventilation Exhaust Damper (BVED) and the Beam Port Floor Exhaust Damper (BPFED) were shut and the doors which communicate both buildings on each of the three floors were sealed with masking tape. Three Vacuum gauges (G) were installed inside the FNR (one per floor) with reference legs connected to the PML through the small holes located at the bottom of the doors as shown in the figure. In the room 3103, on the third floor of the FNR, an extraction hood was completely covered with an aluminum sheet. The sheet had a 6"x6" square cut at its center. A linear flow meter (LFM) of cross sectional area 6"x2" was put at the center of the square, producing an effective air flow opening of 4"x6" or 24 in^2.

Initially the vacuum gauges read a slight vacuum. The 3103 hood damper was cracked open gradually. The exhaust drawn out through the 3103 exhaust damper by PML stack 2 (S2) exhaust fan produced a measurable vacuum in the FNR. The air flow necessary to produce the measurable vacuum was considered the containment air leak rate, Lc. The linear flow measured was 60ft/min. This flow was through the free area of 24 $in^2 = 1/6\,ft^2$, thus the volumetric air flow was 10 ft^3/min. For an estimated value of the FNR volume of $6.2741 \times 10^9\ cm^3 = 2.2154 \times 10^5\ ft^3$, the resultant containment leak rate is

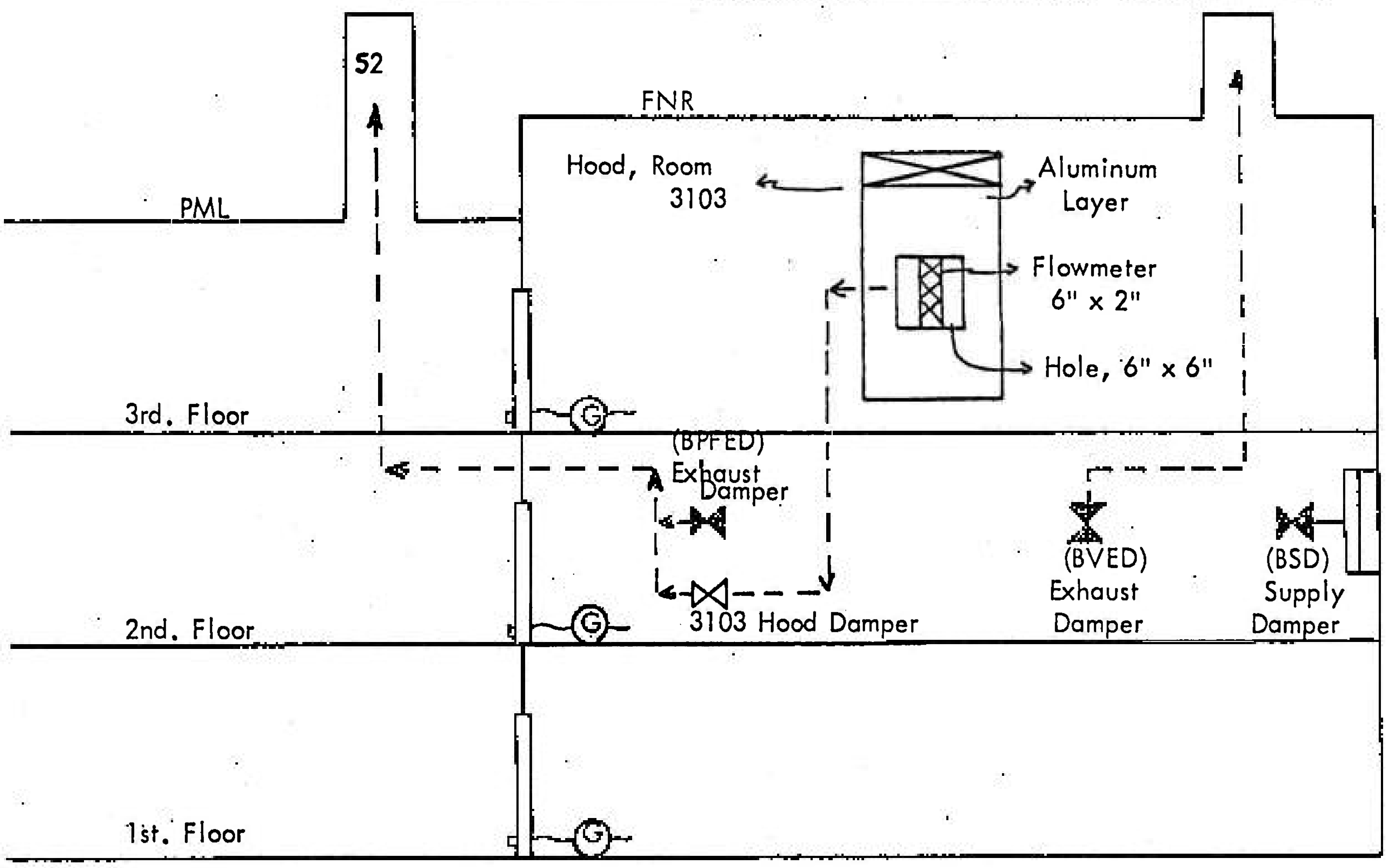
$$L_c = \frac{10\ ft^3/min}{2.2154 \times 10^5\ ft^3}$$

$$= 4.5138 \times 10^{-5}\ min^{-1}$$

$$= 0.065\ day^{-1}$$

Figure (8.2-1)

Arrangement Used in Determining the Leak Rate from Reactor Containment



CHAPTER 9

PUBLIC RADIOLOGICAL CONSEQUENCES OF MAXIMUM CREDIBLE ACCIDENT

The results of the computer program for the reactor core inventory in table (7.1) are given in tables (9.1) through (9.3). These results are the accumulated external, internal (thyroid) and total dose a person would receive as a function of the distance from the reactor site and for the exposure times as indicated. Fig. (9.1) is a graphical representation of the total accumulated doses for the values in table (9.3). Besides the data in table (7.1), the other data used in the program is given in Appendix II and a list of the program is in Appendix I.

According to table (9.3) it is seen that the maximum value of total accumulated dose is the one corresponding to an exposure time of 30 days and at a distance of 250 ft from the FNR site. A realistic as possible analysis of the results requires the knowledge of the location of the different population centers around the FNR and the kind of activities performed in such places.

9.1 FNR Site (5):

The FNR is located on the North Campus (NC) of the University of Michigan, Ann Arbor, Michigan. It is 1.5 miles away from the center of Ann Arbor. The NC development plan incorporates the following resolutions:

1. Only laboratory and research buildings will be constructed within 500 ft of the reactor.
2. No housing or other building containing housing facilities will be erected within 1500 feet of the FNR.

The location of some buildings on the NC and their distance to the FNR are given in table (9.1-1). The major population center near Ann Arbor is the City of Ypsilanti which is 7 miles down river.

9.2 Analysis:

Based on the data given in section 9.1 the analysis of the results will be made by assuming that the exposure time will be constant within certain distances from the FNR, this dependence is given in table (9.2-1).

Inspection of table (9.2-1) shows that in the last two cases the values of the total dose delivered are below the MPD values within a minimum factor of one order of magnitude for the case of Ann Arbor downtown and an order of magnitude of 10 less than MPD for Ypsilanti. For distances within 500 and 1500 ft of the FNR the values of total dose are greater than the values of MPD by a factor of approximately 10. Since this difference is not significantely high and because of the assumptions that have been made in the calculation of the doses, namely:

1. All the released fission products appear in the building instantly,
2. The building and its contents afford no shielding,
3. There is no shielding of any kind between the receptor and the building,
4. No retention of radioactive gases by filters occurs,
5. There is no deposition on the walls or structure of the containment,
6. The breathing rate is constant and the population does not move during the exposure time,
7. All the population center are at the same altitude,
8. The release is at ground level; it is considered that this maximum credible accident represents a gross upper limit and no serious public radiological consequences are expected.

FORD NUCLEAR REACTOR ACCIDENT ANALYSIS.
ANS 15.7
POTENTIAL RADIOLOGICAL CONSEQUENCES OF A
REACTOR CORE MELTDOWN.

POLLUTANT RELEASED FROM THE GROUND LEVEL.

ARRAY VALUES: EXTERNAL DOSE (REM)

DISTANCE FROM REACTOR (FEET)	TIME AFTER ACCIDENT 2.0HOURS	6.0HOURS	15.0HOURS	24.0HOURS	48.0HOURS	72.0HOURS	10DAYS	30DAYS
250.	.2464E-05	.1372E-04	.1096E-03	.1473E-03	.1846E-03	.2231E-03	.4248E-03	.6902E-03
500.	.7391E-06	.4115E-05	.3288E-04	.4419E-04	.5530E-04	.6676E-04	.1270E-03	.2065E-03
750.	.3520E-06	.1960E-05	.1566E-04	.2104E-04	.2636E-04	.3184E-04	.6045E-04	.9806E-04
1000.	.1848E-06	.1029E-05	.7746E-05	.1057E-04	.1339E-04	.1628E-04	.3157E-04	.5172E-04
1250.	.1267E-06	.7055E-06	.5139E-05	.7078E-05	.8989E-05	.1096E-04	.2125E-04	.3478E-04
1500.	.9239E-07	.5144E-06	.4111E-05	.5524E-05	.6921E-05	.8362E-05	.1587E-04	.2573E-04
1750.	.7039E-07	.3920E-06	.3132E-05	.4209E-05	.5275E-05	.6375E-05	.1207E-04	.1954E-04
2000.	.5835E-07	.3249E-06	.2596E-05	.3489E-05	.4367E-05	.5273E-05	.9968E-05	.1613E-04
2250.	.4927E-07	.2744E-06	.2192E-05	.2946E-05	.3686E-05	.4449E-05	.8368E-05	.1350E-04
2500.	.3696E-07	.2058E-06	.1644E-05	.2210E-05	.2768E-05	.3345E-05	.6339E-05	.1027E-04
2750.	.3101E-07	.1727E-06	.1380E-05	.1854E-05	.2324E-05	.2808E-05	.5330E-05	.8643E-05
3000.	.2640E-07	.1470E-06	.1200E-05	.1604E-05	.2006E-05	.2421E-05	.4579E-05	.7413E-05
4000.	.1732E-07	.9646E-07	.8380E-06	.1103E-05	.1365E-05	.1635E-05	.3027E-05	.4852E-05
5000.	.1232E-07	.6859E-07	.6304E-06	.8189E-06	.1006E-05	.1198E-05	.2185E-05	.3478E-05
6000.	.9239E-08	.5144E-07	.4728E-06	.6142E-06	.7524E-06	.8950E-06	.1622E-05	.2572E-05
7000.	.7199E-08	.4009E-07	.3604E-06	.4706E-06	.5779E-06	.6886E-06	.1248E-05	.1978E-05
8000.	.5543E-08	.3087E-07	.2837E-06	.3685E-06	.4515E-06	.5370E-06	.9712E-06	.1538E-05
9000.	.4738E-08	.2638E-07	.2425E-06	.3150E-06	.3861E-06	.4595E-06	.8293E-06	.1311E-05
10000.	.3960E-08	.2205E-07	.2026E-06	.2632E-06	.3230E-06	.3847E-06	.6966E-06	.1104E-05
15000.	.2240E-08	.1247E-07	.1177E-06	.1520E-06	.1851E-06	.2193E-06	.3902E-06	.6123E-06
20000.	.1459E-08	.8123E-08	.7773E-07	.1001E-06	.1216E-06	.1438E-06	.2541E-06	.3973E-06
25000.	.1109E-08	.6173E-08	.6520E-07	.8217E-07	.9843E-07	.1152E-06	.1980E-06	.3054E-06
30000.	.8399E-09	.4677E-08	.4816E-07	.6101E-07	.7321E-07	.8579E-07	.1476E-06	.2276E-06
35000.	.6386E-09	.3834E-08	.4214E-07	.5268E-07	.6256E-07	.7275E-07	.1225E-06	.1869E-06
40,000.	.5656E-09	.3150E-08	.3651E-07	.4516E-07	.5328E-07	.6164E-07	.1024E-06	.1549E-06

Table (9-1)

External Dose Delivered as a Function of Distance from the Reactor and for Different Exposure Times Follwing Maximum Credible Accident

FORD NUCLEAR REACTOR ACCIDENT ANALYSIS.
ANS 15.7
POTENTIAL RADIOLOGICAL CONSEQUENCES OF A
REACTOR CORE MELTDOWN.

POLLUTANT RELEASED FROM THE GROUND LEVEL.

ARRAY VALUES: THYROID DOSE (REM)

DISTANCE FROM REACTOR (FEET)	TIME AFTER ACCIDENT							
	2.0HOURS	6.0HOURS	15.0HOURS	24.0HOURS	48.0HOURS	72.0HOURS	10DAYS	30DAYS
250.	.6088E-03	.5059E-02	.5383E-01	.9054E-01	.1432E+00	.2135E+00	.6922E+00	.1442E+01
500.	.1827E-03	.1518E-02	.1615E-01	.2716E-01	.4283E-01	.6376E-01	.2068E+00	.4312E+00
750.	.8698E-04	.7228E-03	.7690E-02	.1293E-01	.2043E-01	.3045E-01	.9833E-01	.2046E+00
1000.	.4566E-04	.3795E-03	.3837E-02	.6590E-02	.1055E-01	.1585E-01	.5214E-01	.1091E+00
1250.	.3131E-04	.2602E-03	.2559E-02	.4446E-02	.7141E-02	.1074E-01	.3516E-01	.7338E-01
1500.	.2283E-04	.1897E-03	.2019E-02	.3395E-02	.5365E-02	.7997E-02	.2580E-01	.5366E-01
1750.	.1740E-04	.1446E-03	.1538E-02	.2587E-02	.4091E-02	.6100E-02	.1961E-01	.4072E-01
2000.	.1442E-04	.1198E-03	.1275E-02	.2144E-02	.3383E-02	.5037E-02	.1618E-01	.3358E-01
2250.	.1218E-04	.1012E-03	.1077E-02	.1811E-02	.2854E-02	.4248E-02	.1354E-01	.2804E-01
2500.	.9133E-05	.7589E-04	.8075E-03	.1358E-02	.2146E-02	.3199E-02	.1030E-01	.2141E-01
2750.	.7664E-05	.6368E-04	.6776E-03	.1140E-02	.1802E-02	.2687E-02	.8670E-02	.1803E-01
3000.	.6523E-05	.5421E-04	.5874E-03	.9807E-03	.1548E-02	.2306E-02	.7427E-02	.1543E-01
4000.	.4281E-05	.3557E-04	.4069E-03	.6650E-03	.1034E-02	.1528E-02	.4830E-02	.9984E-02
5000.	.3044E-05	.2530E-04	.3039E-03	.4874E-03	.7505E-03	.1102E-02	.3444E-02	.7096E-02
6000.	.2283E-05	.1897E-04	.2279E-03	.3655E-03	.5605E-03	.8210E-03	.2545E-02	.5229E-02
7000.	.1779E-05	.1478E-04	.1742E-03	.2815E-03	.4328E-03	.6351E-03	.1962E-02	.4023E-02
8000.	.1370E-05	.1138E-04	.1367E-03	.2193E-03	.3363E-03	.4526E-03	.1522E-02	.3124E-02
9000.	.1171E-05	.9730E-05	.1169E-03	.1875E-03	.2878E-03	.4219E-03	.1298E-02	.2660E-02
10000.	.9785E-06	.8131E-05	.9767E-04	.1567E-03	.2410E-03	.3536E-03	.1093E-02	.2243E-02
15000.	.5535E-06	.4599E-05	.5655E-04	.8992E-04	.1367E-03	.1992E-03	.6040E-03	.1231E-02
20000.	.3605E-06	.2996E-05	.3728E-04	.5901E-04	.8937E-04	.1299E-03	.3912E-03	.7958E-03
25000.	.2740E-06	.2277E-05	.3091E-04	.4743E-04	.7036E-04	.1010E-03	.2972E-03	.6007E-03
30000.	.2076E-06	.1725E-05	.2290E-04	.3541E-04	.5262E-04	.7560E-04	.2220E-03	.4479E-03
35000.	.1702E-06	.1414E-05	.1989E-04	.3015E-04	.4408E-04	.6270E-04	.1806E-03	.3624E-03
40000.	.1398E-06	.1162E-05	.1714E-04	.2557E-04	.3701E-04	.5229E-04	.1487E-03	.2972E-03

Table (9-2)

Thyroid Dose Delivered as a Function of Distance from the Reactor and for Different Exposure Times Following Maximum Credible Accident

FORD NUCLEAR REACTOR ACCIDENT ANALYSIS.
ANS 15.7
POTENTIAL RADIOLOGICAL CONSEQUENCES OF A
REACTOR CORE MELTDOWN.

POLLUTANT RELEASED FROM THE GROUND LEVEL.

ARRAY VALUES: TOTAL DOSE (REM)

DISTANCE FROM REACTOR (FEET)	TIME AFTER ACCIDENT							
	2.0HOURS	6.0HOURS	15.0HOURS	24.0HOURS	48.0HOURS	72.0HOURS	10DAYS	30DAYS
250.	.6113E-03	.5073E-02	.5394E-01	.9069E-01	.1434E+00	.2137E+00	.6926E+00	.1443E+0
500.	.1834E-03	.1522E-02	.1618E-01	.2721E-01	.4288E-01	.6383E-01	.2070E+00	.4314E+0
750.	.8733E-04	.7247E-03	.7706E-02	.1296E-01	.2046E-01	.3048E-01	.9839E-01	.2047E+0
1000.	.4585E-04	.3805E-03	.3845E-02	.6601E-02	.1057E-01	.1587E-01	.5218E-01	.1091E+0
1250.	.3144E-04	.2609E-03	.2564E-02	.4454E-02	.7150E-02	.1075E-01	.3518E-01	.7341E-0
1500.	.2292E-04	.1902E-03	.2023E-02	.3401E-02	.5372E-02	.8005E-02	.2582E-01	.5369E-0
1750.	.1747E-04	.1449E-03	.1541E-02	.2591E-02	.4096E-02	.6107E-02	.1962E-01	.4074E-0
2000.	.1448E-04	.1202E-03	.1278E-02	.2148E-02	.3387E-02	.5042E-02	.1619E-01	.3360E-0
2250.	.1223E-04	.1015E-03	.1079E-02	.1814E-02	.2858E-02	.4252E-02	.1355E-01	.2805E-0
2500.	.9170E-05	.7610E-04	.8091E-03	.1360E-02	.2149E-02	.3202E-02	.1031E-01	.2142E-0
2750.	.7695E-05	.6386E-04	.6790E-03	.1142E-02	.1804E-02	.2689E-02	.8675E-02	.1804E-0
3000.	.6550E-05	.5435E-04	.5886E-03	.9823E-03	.1550E-02	.2309E-02	.7431E-02	.1544E-0
4000.	.4298E-05	.3567E-04	.4077E-03	.6661E-03	.1036E-02	.1529E-02	.4833E-02	.9989E-0
5000.	.3057E-05	.2537E-04	.3045E-03	.4882E-03	.7515E-03	.1103E-02	.3446E-02	.7100E-0
6000.	.2292E-05	.1902E-04	.2284E-03	.3662E-03	.5612E-03	.8219E-03	.2546E-02	.5232E-0
7000.	.1786E-05	.1482E-04	.1746E-03	.2819E-03	.4334E-03	.6358E-03	.1963E-02	.4025E-0
8000.	.1375E-05	.1141E-04	.1370E-03	.2197E-03	.3367E-03	.4931E-03	.1523E-02	.3125E-0
9000.	.1176E-05	.9756E-05	.1171E-03	.1878E-03	.2882E-03	.4223E-03	.1299E-02	.2661E-0
10000.	.9825E-06	.8153E-05	.9787E-04	.1569E-03	.2413E-03	.3540E-03	.1094E-02	.2244E-0
15000.	.5557E-06	.4612E-05	.5667E-04	.9007E-04	.1369E-03	.1994E-03	.6043E-03	.1232E-0
20000.	.3620E-06	.3004E-05	.3736E-04	.5911E-04	.8949E-04	.1301E-03	.3915E-03	.7962E-0
25000.	.2751E-06	.2283E-05	.3098E-04	.4751E-04	.7046E-04	.1011E-03	.2974E-03	.6010E-0
30000.	.2084E-06	.1729E-05	.2295E-04	.3547E-04	.5269E-04	.7569E-04	.2221E-03	.4482E-0
35000.	.1709E-06	.1418E-05	.1994E-04	.3021E-04	.4415E-04	.6277E-04	.1807E-03	.3626E-0
40000.	.1404E-06	.1165E-05	.1717E-04	.2561E-04	.3706E-04	.5236E-04	.1488E-03	.2974E-0

Table (9-3)

Total Dose Delivered as a Function of Distance from the Reactor and for Different Exposure Times Following Maximum Credible Accident

Figure (9-1)

Total Accumulated Dose vs. Distance from Reactor for Different Exposure Times

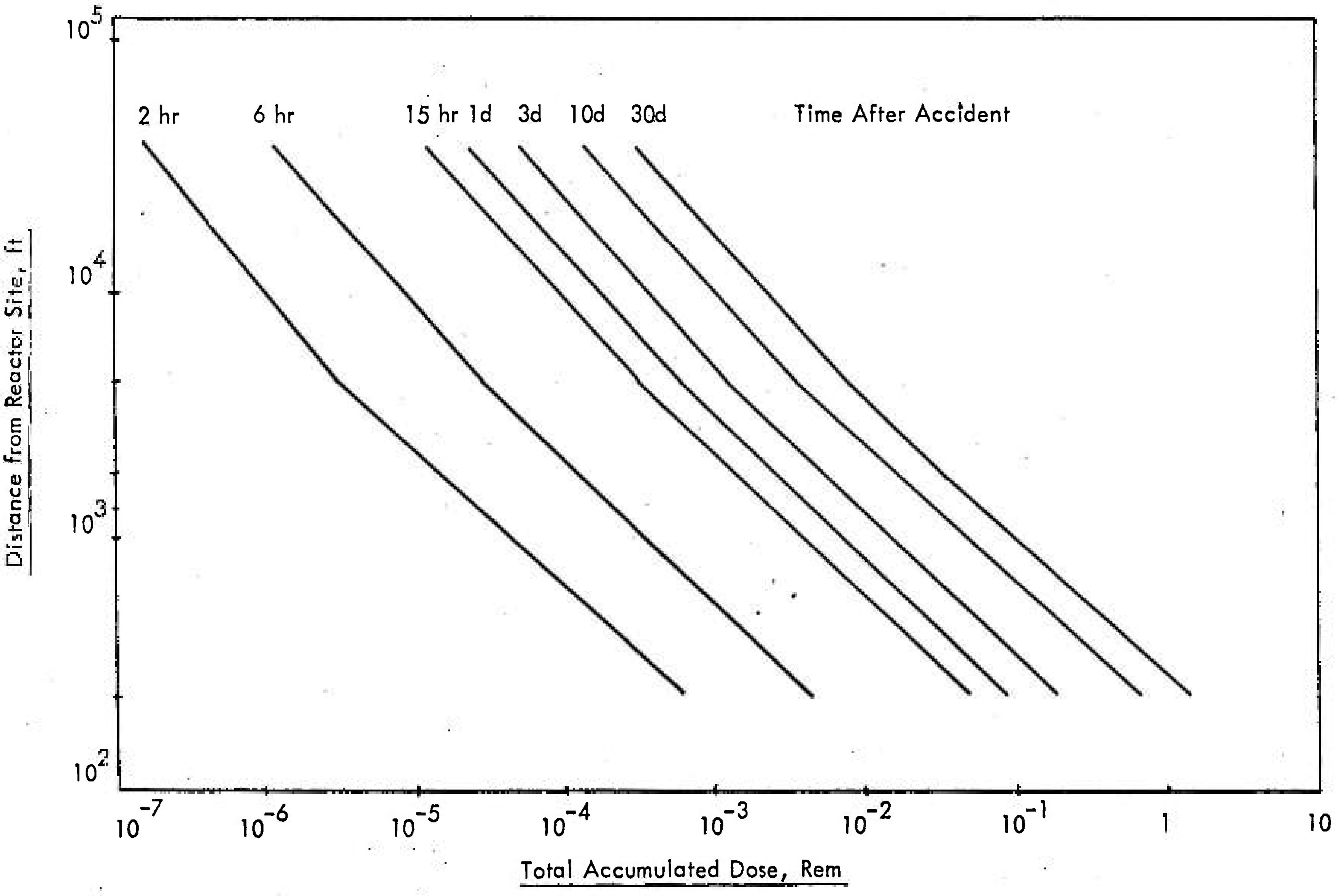


Table (9.1-1)

North Campus Buildings and Their Distance to the FNR

Building	Distance to FNR ft
Mortimer E. Cooley Bldg.	400 South East
Automotive Engineering Lab.	300 North East
Library Storage Bldg. and bindery	600 South East
Printing Shop Bldg.	700 South East
Aeronautical Research Labs.	1200 to 1500 North East
Fluids Engineering Lab.	650 North
Cyclotron Bldg.	1500 North East
Institute of Science and Technology Bldg.	520 South
Northwood Apartments	1500 North
Veterans Administration Hospital	1500 South

Table (9.2-1)

Assumed Exposure Times for Different Average Distances from FNR, with Expected Total Doses from Program and MPD*

Type of Population Center	Average Distance ft	Exposure Time	Total Dose Rem	MPD Rem
Research and Laboratory Bldgs.	500	6 hr	1.524×10^{-3}	1.1806×10^{-4}
Housing and Hospital	1500	15 hr	2.025×10^{-3}	2.9514×10^{-4}
Ann Arbor Center	8000 + (1.512 miles)	24 hr	2.2×10^{-4}	4.72×10^{-4}
Ypsilanti	40000 + (7.56 miles)	30 days	2.97×10^{-4}	1.42×10^{-2}

+ Those values are not averages.

* The value of the Maximum Permissible Dose selected from NCRP (table 6, Report 39) was 0.170 Rem average per year for a population as a whole.

APPENDIX I

LIST OF THE COMPUTER PROGRAM

Some of the features under which the program was run are the following:

1.- The release of radioactive effluents was assumed to occur at ground level.

2.- No particular wind direction was taken into account.

3.- The pollutant release was assumed to be continous.

4.- The 100% of the reactor core inventory in table (7.1) was available for releasing.

```
C                      FORD NUCLEAR REACTOR ACCIDENT ANALYSIS.
C    ***********************************************************************
C    *                                                                     *
C    *  A PROGRAM TO CALCULATE CUMULATIVE INTERNAL, EXTERNAL AND TOTAL     *
C    *  DOSES AT DIFFERENT DISTANCES, DIRECTIONS AND TIMES DUE TO RADI     *
C    *  -ACTIVITY RELEASES FROM THE FORD NUCLEAR REACTOR UNDER THE FOL     *
C    *  -LOWING ACCIDENT SITUATIONS:                                       *
C    *                                                                     *
C    *    "A POTENTIALLY HAZARDOUS ACCIDENT CONDITION EXISTS AT THE        *
C    *     REACTOR AND RADIOLOGICAL CONSEQUENCES OF THE ACCIDENT WOULD     *
C    *     BE SIGNIFICANT                                                  *
C    *                                                                     *
C    *  BASED UPON REGULATORY GUIDES 1.3 AND 1.4, ASSUMPTIONS USED IN      *
C    *  EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS       *
C    *  OF COOLANT ACCIDENT, AND ANS 15.7, GUIDE FOR RESEARCH REACTOR      *
C    *  SITE EVALUATION.                                                   *
C    *                                                                     *
C    ***********************************************************************
C       VARIABLES:
C
C       ACCDNT    INTEGER VALUED PARAMETER TO DESCRIBE THE TYPE OF CORE
C                 MELTDOWN:
C
C                        1     ONE PERCENT CORE MELTDOWN.
C                        10    TEN PERCENT CORE MELTDOWN.
C                        100   (UNDECLARED) ONE HUNDRED PERCENT CORE MELTDOWN.
C       CHI       RADIOACTIVE POLLUTANT CONCENTRATION, CI/CUBIC METER.
C       DELDOS    INCREMENTAL RADIOACTIVE POLLUTANT DOSE.
C       DINDOS    INCREMENTAL RADIOACTIVE INTERNAL DOSE ( THYROID ).
C       DIST      DISTANCE FROM REACTOR TO DOSE MEASUREMENT POINT, METERS.
C       DOSETP    INTEGER VALUED PARAMETER TO DECIDE TYPE OF DOSE:
C                        1     INTERNAL DOSE.
C                        OTHER VALUES:  EXTERNAL DOSE WITH THE FOLLOWING
C                        MODIFICATIONS:
C                 THE ORIGINAL PROGRAM WAS SET UP TO CALCULATE THE EXTER
C                 -NAL AND THYROID DOSES AT DIFFERENT DISTANCES AND DIRE
C                 -CTIONS FOR FIXED EXPOSURE TIMES. THE EFFECT OF VARIOU
C                 -S EXPOSURE TIMES WAS NOT CONSIDERED. THE PRIGRAM WAS
C                 THEN MODIFIED SO AS TO CALCULATE THE EXTERNAL,THYROID
C                 AND TOTAL DOSES AT DIFFERENT DISTANCES FOR VARIOUS
C                 EXPOSURE TIMES(PRESENTLY EIGHT DIFFERENT EXPOSURE
C                 TIMES 2 HOURS TO 30 DAYS.). THIS WAS ACHIEVED BY INTRO
C                 -DUCING ANOTHER INTEGER VALUE OF 3 FOR INTEGER VARI
C                 -ABLE 'DOSETP'. AS BEFORE DOSETP=1 GIVES THYROID DOSE
C                 -AT DIFFERENT DIRECTIONS AND DOSETP=2 GIVES EXTERNAL
C                 DOSE FOR FIXED EXPOSURE TIMES( 2 HOURS - 30 DAYS).
C       EFFLNT    THE NAMELIST OF THE INPUT VARIABLES.
C       EXPOTM    TIME AFTER ACCIDENT, HRS. EIGHT VALUES: 2 HRS. TO 30 DAYS.
C       HEADER    A SUBROUTINE TO WRITE PROPER OUTPUT HEADING.
C       HGT       REACTOR STACK HEIGHT,METERS.
C       KRAP      INTEGER VALUED PARAMETER USED INTERNALLY FOR SELECTING
C                 PROPER OUTPUT HEADING.
C       LEVEL     INTEGER VALUED PARAMETER TO DESCRIBE RADIOACTIVE RELEASE
C                 LEVEL:
C                        0     GROUND LEVEL RELEASE.
C                        1     STACK RELEASE.
C       N         NON DIMENSIONAL STABILITY PARAMETER FOR WIND CONDITION.
C       NDIR      NUMBER OF WIND DIRECTIONS. ALSO USED FOR NUMBER OF EXPOSUR
C                 TIMES DESIRED.
C       NDIST     NUMBER OF DISTANCES FROM THE REACTOR BUILDING.
```

```
C      POLTAN     INTEGER VALUED INPUT PARAMETER TO DESCRIBE THE TYPE OF
C                 POLLUTANT BEING RELEASED FROM THE REACTOR:
C                          1     IODINE.
C                          2     NOBLE GAS.
C                          3     FISSION PRODUCT SOLIDS.
C      Q          RADIOACTIVE POLLUTANT RELEASE RATE, CI/SEC.
C      QCHI       CHI/Q, SEC/CUBIC METER.
C      QCHITR     INTEGER VALUED INPUT PARAMETER:
C                          1     WRITES Q/CHI VALUES OF THE LAST ISOTOPE
C                                PROCESSED.
C                          0     SKIPS THE WRITING STEP.
C      QNOT       TOTAL RADIOACTIVE POLLUTANT RELEASED INTO THE REACTOR
C                 CONTAINMENT AS A RESULT OF AN ACCIDENT AT T=0, CURIES.
C      SIG(X,Y,Z) ARE STANDARD DEVIATIONS OF THE PLUME OBTAINED FROM
C                   FIGS. 3.10 AND 3.11, METEROLOGY AND ATOMIC ENERGY,
C                   CHAP. 3,1968.
C      SIGZ(1,3,4,5,6,7)ARE THE VERTICAL DIFFUSION PARAMETERS CORRESPOND
C                 -ING TO THE PASQUILL'S CONDITIONS F,C,F,C,D,F RESPEC
C                 -TIVELY.(USED IN SUBROUTINE PEANTS ONLY)
C      T          TIME AFTER ACCIDENT, SEC.
C      TIMDEP     INTEGER VALUED INPUT PARAMETER : DESCRIBES WHETHER THE
C                 POLLUTANT RELEASE FROM THE CONTAINMENT IS TIME DEPENDENT
C                 OR CONTINUOUS:
C                          1     TIME DEPENDENT.
C                          0     TIME INDEPENDENT.
C      TIMINT     INTEGER VALUED INPUT PARAMETER: TAKES EIGHT VALUES (1-8)
C                 CORRESPONDING TO EXPOSURE TIMES(2. HOURS- 30 DAYS).
C                 MUST BE USED WITH WINDIR=0.
C      THYDOS     TOTAL THYROID DOSE FROM ALL DIFFERENT IODINE ISOTOPES USE
C      TRACE      INTEGER VALUED INPUT PARAMETER USED FOR WRITING THE INPUT
C                 PARAMETERS OF EACH ISOTOPE AS IT IS PROCESSED.
C      WINDIR     INTEGER VALUED INPUT PARAMETER, HAS EIGHT VALUES (1-8)
C                 CORRESPONDING TO THE FOLLOWING INTEGER VALUE ASSIGNMENTS:
C                          1     NORTH
C                          2     NORTH-EAST.
C                          3     EAST.
C                          4     SOUTH-EAST.
C                          5     SOUTH.
C                          6     SOUTH-WEST.
C                          7     WEST.
C                          8     NORTH-WEST.
C      WINVEL     REAL VALUED WIND VELOCITY, METERS/SEC.
C       WNVEL(1,3,4,5,6,7) ARE THE WIND VELOCITIES CORRESPONDING TO
C                 PASQUILL'S CONDITIONS F,D,FC,D,F RESPECTIVELY.
C                 (USED IN SUBROUTINE PEANTS ONLY)
C      WINDY      A SUBROUTINE WHICH MAY BE CALLED TO CHECK THAT PROPER VAL
C                 OF WINDIR WAS CHOSEN FOR WIND DIRECTION. ACHIEVED
C                 BY TRACE=1 INPUT.
       INTEGER POLTAN,WINDIR,TIMDEP,DOSETP,TRACE,ACCDNT,DUOLK,TIMINT
       REAL  QEXLP1,QEXLP2,QEXLC1,QEXLC2,Q,N,LAMDA,LEAK,
      1LEAKP,LAMDAE
       COMMON Q,EXPOTM,TIMINT,A,CHI,DELDOS,DOSE,DINDOS,THYDOS,TOTDOS,
      1NDIST,EBETA,EGAMA,LAMDAE,BR,ZSI,LEVEL,HGT,TRACE,QCHI,
      1SIGY,SIGZ1,SIGZ3,SIGZ4,SIGZ5,SIGZ6,SIGZ7,QNOT,LAMDA,
      1WNVEL1,WNVEL3,WNVEL4,WNVEL5,WNVEL6,WNVEL7,LEAK,LEAKP,DUOLK
       DIMENSION CHI(25,10),A(25,1),B(25,1),DOSE(25,10),DELDOS(25,10),
      1DIST(25,10),DMT(25,10),SIGD(25,10),EXPO1(25,10),DENOM2(25,10),
      2DNUM(25,10),DENOM3(25,10),EXPO2(25,10),DENOM(25,10),
      3DINDOS(25,10),THYDOS(25,10),SIGX(25),SIGY(25),SIGZ(25),
      4HGT(25),DENOM1(25),QCHI(25,10),TOTDOS(25,10),T(25),
```

```
     1SIGZ1(25),SIGZ3(25),SIGZ4(25),SIGZ5(25),SIGZ6(25),SIGZ7(25)
      NAMELIST/EFFLNT/ LEVEL, WINDIR, WINVEL,HGT, EXPOTM,TIMINT,N,
     1SIGX,SIGY,SIGZ,QNOT,POLTAN,T,TIMDEP,NDIR,NDIST,EBETA,EGAMA,
     1LAMDA,LEAK,DELT,DOSETP,ZSI,TRACE,ACCDNT,QCHITR,BR,LEAKP,DUOLK
      NAMELIST/PENTS1/SIGZ1,SIGZ3,SIGZ4,SIGZ5,SIGZ6,SIGZ7,
     1WNVEL1,WNVEL3,WNVEL4,WNVEL5,WNVEL6,WNVEL7
C  *   INITIALIZING THE INPUT PARAMETERS.
      DO 340 I=1,25
      DO 345 J=1,10
      DELDOS(I,J)=0
      DINDOS(I,J)=0
      THYDOS(I,J)=0
      TOTDOS(I,J)=0
      QCHI(I,J)=0
      CHI(I,J)=0
  345 DOSE(I,J)=0
  340 CONTINUE
C  *********************************************************************
C  *
C  *  CALCULATE THE DISTANCE FROM THE REACTOR
C  *
C  *********************************************************************
      DO 360 I=1,25
      IF(I .GT. 12) GO TO 183
      B(I,1)=250.*I
      GO TO 360
  183 L=I-1
      IF(I .GT. 19) GO TO 184
      B(I,1)=B(L,1)+1000.
      GO TO 360
  184 L=I-1
      B(I,1)=B(L,1)+5000.
  360 A(I,1)=.3048*B(I,1)
      K=1
      READ(5,PENTS1)
      WRITE(6,PENTS1)
   10 READ(5,EFFLNT, END=500)
C  *********************************************************************
C  *
C  *  CALCULATE THE RADIOACTIVE POLLUTANT RELEASE RATE FROM THE REACTOR
C  *  BUILDING LEAK RATE IS ASSUMED TO BE 0.065 PER DAY.
C  *  QNOT VALUES ARE TAKEN FROM THE ARTICLE " FISSION PRODUCT
C  *  RELEASE " BY PARKER ET AL IN THE TECHNOLOGY OF NUCLEAR REACTOR
C  *  SAFETY, VOL 2,PP 525-618,M.I.T. PRESS.
C  *
C  *********************************************************************
      IF(TRACE .EQ. 0) GO TO 75
      WRITE(6,EFFLNT)
   75 IF(TIMDEP .EQ. 0) GO TO 79
      Q=LEAK*QNOT*EXP(-(LAMDA+LEAK)*DELT)
      GO TO 80
   79 IF(DUOLK .EQ. 0) GO TO 76
      QLCPT=LEAKP*LEAK*QNOT/(3600.*EXPOTM*(LEAK-LEAKP))
      QEXLP1=(1.-EXP(-(LAMDA+LEAKP)*3600.*EXPOTM))
      QEXLP2=1./(LEAKP+LAMDA)
      QEXLC1=(1.-EXP(-(LAMDA+LEAK)*3600.*EXPOTM))
      QEXLC2=1./(LEAK+LAMDA)
      Q=QLCPT*(QEXLP1*QEXLP2-QEXLC1*QEXLC2)
      GO TO 80
   76 QC=QNOT*LEAK/(3600.*EXPOTM*(LEAK+LAMDA))
```

```
      Q=QC*(1.-EXP(-(LEAK+LAMDA)*36CC.*EXPOTM))
      IF(ACCDNT .EQ. 10) Q=Q/10.
      IF(ACCDNT .EQ. 1) Q=Q/1C0.
   80 LAMDAE=LAMCA+5.81E-8
C     *********************************************************************
C     *
C     *   CALCULATE DOSES AS A FUNCTICN OF TIME AFTER THE ACCIDENT AND DIS
C     *   -TANCE FROM THE REACTOR.
C     *
C     *********************************************************************
      IF(WINDIR .EQ. 0) GO TC 36
      J=WINDIR
      GO TO 37
   36 J=TIMINT
   37 IF(TRACE .EQ. 0) GO TO 106
      IF(POLTAN .EQ. 1) GO TO 88
      IF(POLTAN .EQ. 2) GO TO 89
      WRITE(6,95)
   95 FORMAT('1THE POLLUTANT IS FISSICN PRODUCT SOLIDS.')
      GO TO 105
   89 WRITE(6,99)
   99 FORMAT ('1THE PCLLUTANT IS NOBLE GAS.')
  105 WRITE(6,100) Q,EXPOTM
  100  FORMAT('Q=',E8.3,',',3X,'EXPOSURE TIME GAP=',F4.0,2X,
     1'HCURS.')
      GO TO 106
   88 WRITE(6,101)
  101 FORMAT('THE POLLUTANT IS ICDINE')
      GO TO 105
  106 IF( LEVEL .EQ. 0 .AND. TIMCEP .EQ. 0) GO TO 111
      IF(LEVEL .EQ. 0 .AND. TIMCEP .EQ. 1) GO TO 112
      IF( LEVEL .EQ. 1 .ANC. TIMDEP .EQ. 0) GO TO 113
      IF (LEVEL .EQ. 1 .ANC. TIMDEP .EQ. 1) GO TO 114
      WRITE (6,73)
   73 FORMAT('2YOU GOOFED IN DATA ENTRY')
      GO TO 505
  111 IF(EXPOTM .GT. 2.) GO TO 116
C     *********************************************************************
C     *
C     *  CALCULATES DOSES FCR GROUND LEVEL, TIME INDEPENDENT RELEASE
C     *  EXPOSURE TIME LESS THAN 2. HOURS.
C     *
C     *********************************************************************
      DO 2 I=1,NDIST
      CHI(I,J)=Q/(3.14*WINVEL*SIGY(I)*SIGZ(I))
      QCHI(I,J)=CHI(I,J)/Q
      DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.*EXPOTM
      DOSE(I,J)=DELDOS(I,J)+DOSE(I,J)
C     *********************************************************************
C     *
C     *  CALCULATES INTERNAL THYRCID CCSE FRCM IODINE. FORMULA FOR DINDO
C     *  ARE OBTAINED FRCM INTRODUCTICN TO NUCLEAR ENGINEERING,
C     *  LAMARSH, PAGE 521.
C     *
C     *********************************************************************
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)
     1*3600.*EXPOTM
      THYDOS(I,J)=THYDOS(I,J)+DINDCS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DINDOS(I,J)+DELDOS(I,J)
    2 CCNTINUE
```

```
      IF(TRACE .EQ. 0) GO TO 301
      CALL WINDY(WINDIR,WINVEL,LEVEL,HGT,Q)
      GO TO 301
C  ****************************************************************************
C  *
C  *   CALCULATES DOSES FOR GROUND LEVEL, TIME INDEPENDENT RELEASE
C  *   EXPOSURE TIME GREATER THAN 2. HOURS.
C  *
C  ****************************************************************************
  116 IF(EXPOTM .GT. 8.) GO TO 302
      DO 3 I=1,NDIST
      CHI(I,J)=Q*2.032/(SIGZ(I)*WINVEL*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
      DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.*EXPOTM
      DOSE(I,J)=DOSE(I,J)+DELDOS(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)
     1*3600.*EXPOTM
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DINDOS(I,J)+DELDOS(I,J)
    3 CONTINUE
      GO TO 303
  302 CALL PEANTS
  303 IF(TRACE .EQ. 0) GO TO 301
      CALL WINDY(WINDIR,WINVEL,LEVEL,HGT,Q)
      GO TO 301
C  ****************************************************************************
C  *                                                                          *
C  *   CALCULATES DOSES FOR GROUND LEVEL ,TIME DEPENDENT RELEASE FROM         *
C  *   THE CONTAINMENT.                                                       *
C  *                                                                          *
C  ****************************************************************************
  112 DO 4 I=1,NDIST
      DMT(I,J)=(A(I,1)-WINVEL*T(I))**2
      SIGD(I,J)=2.*SIGX(I)**2*(WINVEL*T(I))**(2-N)*A(I,1)**(N-2)
      EXPO1(I,J)=EXP(-(DMT(I,J)/SIGD(I,J)))
      DENOM1(I)=(3.14**1.5*1.417*SIGX(I)*SIGY(I)*SIGZ(I))
      DENOM2(I,J)=(WINVEL*T(I))**(1.5*(2-N))*A(I,1)**(1.5*(N-2))
      DENOM(I,J)=DENOM1(I)*DENOM2(I,J)
      CHI(I,J)=Q*EXPO1(I,J)/DENOM(I,J)
      QCHI(I,J)=CHI(I,J)/Q
      DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.*EXPOTM
      DOSE(I,J)=DELDOS(I,J)+DOSE(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)
     1*3600.*EXPOTM
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DINDOS(I,J)+DELDOS(I,J)
    4 CONTINUE
      IF(TRACE .EQ. 0) GO TO 301
      CALL WINDY(WINDIR,WINVEL,LEVEL,HGT,Q)
      GO TO 301
  113 IF(EXPOTM .GT. 2.)  GO TO 117
C  ****************************************************************************
C  *                                                                          *
C  *   CALCULATES DOSES FOR TIME INDEPENDENT RELEASE FROM THE STACK,          *
C  *   EXPOSURE TIME LESS THAN TWO HOURS.                                     *
C  *                                                                          *
C  ****************************************************************************
      DO 5 I=1,NDIST
      CHI(I,J)=Q*EXP(-((HGT(I)/SIGZ(I))**2/2))/(3.14*WINVEL*SIGY(I)*SIGZ(
      QCHI(I,J)=CHI(I,J)/Q
```

```
      DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.*EXPOTM
      DOSE(I,J)=DOSE(I,J)+DELDOS(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)
     1*3600.*EXPOTM
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DINDOS(I,J)+DELDOS(I,J)
    5 CONTINUE
      IF(TRACE .EQ. 0) GO TO 301
      CALL WINDY(WINDIR,WINVEL,LEVEL,HGT,Q)
      GO TO 301
C   ******************************************************************
C   *                                                                *
C   *  CALCULATES DOSES FOR TIME INDEPENDENT RELEASE FROM THE STACK, *
C   *  EXPOSURE TIME GREATER THAN TWO HOURS.                         *
C   *                                                                *
C   ******************************************************************
  117 IF(EXPOTM .GT. 8.) GO TO 217
      DO 6 I=1,NDIST
      CHI(I,J)=Q*2.032*EXP((-(HGT(I)/SIGZ(I))**2/2))/(WINVEL*SIGZ(I)*A(I,
      QCHI(I,J)=CHI(I,J)/Q
      DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.*EXPOTM
      DOSE(I,J)=DOSE(I,J)+DELDOS(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)
     1*3600.*EXPOTM
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DINDOS(I,J)+DELDOS(I,J)
    6 CONTINUE
      GO TO 318
  217 CALL PEANTS
  318 IF(TRACE .EQ. 0) GO TO 301
      CALL WINDY(WINDIR,WINVEL,LEVEL,HGT,Q)
      GO TO 301
C   ******************************************************************
C   *  CALCULATES DOSES FOR TIME DEPENDENT RELEASE FROM THE STACK.   *
C   *                                                                *
C   *                                                                *
C   ******************************************************************
  114 DO 7 I=1,NDIST
      DNUM(I,J)=((A(I,1)-WINVEL*T(I))/SIGX(I))**2+(HGT(I)/SIGZ(I))**2
      DENOM3(I,J)=(WINVEL*T(I))**(1.5*(2-N))*(A(I,1)**(1.5*(N-2)))
      EXPO2(I,J)=EXP(-DNUM(I,J)/(2.*DENOM3(I,J)))
      CHI(I,J)=Q/(3.14**1.5*1.47*SIGX(I)*SIGY(I)*SIGZ(I)*DENOM3(I,J))
     1*EXPO2(I,J)
      QCHI(I,J)=CHI(I,J)/Q
      DELDOS(I,J)=(.228*EBETA+.253*EGAMA)
      DELDOS(I,J)=DELDOS(I,J)*CHI(I,J)*3600.*EXPOTM
      DOSE(I,J)=DELDOS(I,J)+DOSE(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)
     1*3600.*EXPOTM
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DELDOS(I,J)+DINDOS(I,J)
    7 CONTINUE
      IF(TRACE .EQ. 0) GO TO 301
      CALL WINDY(WINDIR,WINVEL,LEVEL,HGT,Q)
      GO TO 301
  301 K=K+1
      GO TO 10
  500 IF(QCHITR .EQ. 1) GO TO 171
      GO TO 511
  171 WRITE(6,175)
```

```
175   FORMAT('CDISTANCE',40X,'CHI/Q VALUES')
      DO 172 I=1,NDIST
172   WRITE(6,173) B(I,1),(QCHI(I,J),J=1,NDIR)
173   FORMAT(3X,F6.0,4X,8(E9.4,2X))
509   WRITE(6,510)
510   FORMAT('1')
      WRITE(6,234)
234   FORMAT('ODISTANCE',40X,'CHI VALUES')
      DO 235 I=1,NDIST
235   WRITE(6,173) B(I,1),(CHI(I,J),J=1,NDIR)
      WRITE(6,510)
511    KRAP=0
      IF(DOSETP .EQ. 1) GO TO 56
      IF(DOSETP .EQ. 3) GO TO 520
      CALL HEADER(POLTAN,EXPOTM,DOSETP,ACCDNT,LEVEL,KRAP)
      DO 499 I=1,NDIST
499   WRITE(6,501)B(I,1),(DOSE(I,J),J=1,NDIR)
501   FORMAT(3X,F6.0,4X,8(E9.4,2X))
      GO TO 505
56    CALL HEADER(POLTAN,EXPOTM,DOSETP,ACCDNT,LEVEL,KRAP)
      DO  503 I=1,NDIST
503   WRITE(6,502) B(I,1),(THYDOS(I,J),J=1,NDIR)
502   FORMAT(3X,F6.0,4X,8(E9.4,2X))
      GO TO 505
520   CALL HEADER(POLTAN,EXPOTM,DOSETP,ACCDNT,LEVEL,KRAP)
      DO 522 I=1,NDIST
522   WRITE(6,502) B(I,1),(DOSE(I,J),J=1,NDIR)
      WRITE(6,506)
      KRAP=1
      CALL HEADER(POLTAN,EXPOTM,DOSETP,ACCDNT,LEVEL,KRAP)
      DO 523 I=1,NDIST
523   WRITE(6,502) B(I,1),(THYDOS(I,J),J=1,NDIR)
      WRITE(6,506)
      KRAP=KRAP+1
      CALL HEADER(POLTAN,EXPOTM,DOSETP,ACCDNT,LEVEL,KRAP)
      DO 521 I=1,NDIST
521   WRITE(6,622) B(I,1),(TOTDOS(I,J),J=1,NDIR)
622   FORMAT(3X,F6.0,4X,8(E9.4,2X))
505   WRITE(6,506)
506   FORMAT('1')
      STOP
      END
C     ******************************************************************
C     *                                                                *
C     *  THE SUBROUTINE 'HEADER' IS CALLED TO WRITE THE RELEVENT HEADINGS *
C     *  BEFORE WRITING THE OUTPUT DOSES.                               *
C     *                                                                *
C     ******************************************************************
      SUBROUTINE HEADER(POLTAN,EXPOTM,DOSETP,ACCDNT,LEVEL,KRAP)
      INTEGER POLTAN,DOSETP,ACCDNT,LEVEL,KRAP
      WRITE(6,130)
130    FORMAT('1')
      WRITE(6,131)
131   FORMAT(30X,'FORD NUCLEAR REACTOR ACCIDENT ANALYSIS.')
      WRITE(6,147)
147   FORMAT(45X,'ANS  15.7')
      IF(ACCDNT .EQ. 1) GO TO 400
      IF(ACCDNT .EQ. 10) GO TO 410
      WRITE(6,411)
411   FORMAT(30X,'POTENTIAL RADIOLOGICAL CONSEQUENCES OF A',/
```

```
      140X,'REACTOR CORE MELTDCWN.')
      GO TO 412
  400 WRITE(6,413)
  413 FORMAT(30X,'POTENTIAL RADIOLOGICAL CONSEQUENCES OF A',/
     140X,'REACTOR CORE MELTDCWN.')
      GO TO 412
  410 WRITE(6,414)
  414 FORMAT(3CX,'POTENTIAL RADICLOGICAL CONSEQUENCES OF A',/
     140X,'REACTOR CORE MELTDCWN.')
  412 WRITE(6,132)
  132 FORMAT('C')
      IF(DOSETP .EQ. 3) GO TO 690
      IF(POLTAN .EQ. 1) GO TO 133
      WRITE(6,135)
  135 FORMAT('ISOTOPE:  NOBLE GAS')
      GO TO 136
  133 WRITE(6,134)
  134 FORMAT('ISOTOPE:  IODINE')
  136 WRITE(6,137) EXPOTM
  137 FORMAT('CTIME AFTER ACCIDENT:',F5.2,3X,'HOURS.')
  138 IF(DOSETP .EQ. 1) GO TO 139
      WRITE(6,140)
  140  FCRMAT('0','ARRAY VALUES:',6X,'EXTERNAL DCSE(REM)')
      GO TO 142
  139 WRITE(6,141)
  141 FORMAT('0','ARRAY VALUES:',6X,'INTERNAL DOSE(REM)')
  142  WRITE(6,143)
  143 FORMAT('CDISTANCE FROM',35X,'WIND DIRECTICNS:')
      WRITE(6,145)
  145 FORMAT('REACTOR(FEET)')
      WRITE(6,146)
  146  FORMAT(18X,'N',9X,'NE',10X,'E',10X,'SE',8X,'S',10X,
     1 'SW',9X,'W',10X,'NW')
      GO TO 697
  690 IF(LEVEL .EC. 0) GO TC 695
      WRITE(6,691)
  691 FCRMAT('POLLUTANT RELEASEC FRCM THE STACK.')
      GO TO 696
  695 WRITE(6,692)
  692 FORMAT('POLLUTANT RELEASEC FRCM THE GROUND LEVEL.')
  696 IF(KRAP .EQ. 0) GO TO 800
      IF(KRAP .EQ. 1) GO TO 803
      WRITE(6,701)
  701 FORMAT('0','ARRAY VALUES:',6X,'TOTAL DOSE (REM)')
      GO TO 802
  800 WRITE(6,801)
  801 FORMAT('0','ARRAY VALUES:',6X,'EXTERNAL DOSE (REM)')
      GO TO 802
  803 WRITE(6,804)
  804 FORMAT('0','ARRAY VALUES:',6X,'THYROID DOSE (REM)')
      GO TO 802
  802 WRITE(6,702)
  702 FORMAT('CDISTANCE FROM',30X,'TIME AFTER ACCIDENT')
      WRITE(6,703)
  703 FORMAT('REACTOR (FEET)')
      WRITE(6,704)
  704 FORMAT(14X,'2.0HOURS',3X,'6.CHOURS',2X,'15.0HOURS',2X,'24.0HOURS',
     12X,'48.0HOURS',2X,'72.0HOLRS',4X,'10DAYS',5X,'30DAYS')
      WRITE(6,333)
  333 FORMAT(14X,'--------',3X,'--------',2X,'---------',2X,'---------',
```

```
     12X,'---------',2X,'---------',4X,'------',5X,'------')
 697 WRITE(6,698)
 698 FORMAT('0')
     RETURN
     END
C    *********************************************************************
C    *                                                                   *
C    *  THE SUBROUTINE 'WINDY' IS BYPASSED IN THE CALCULATION OF THE     *
C    *  DOSE PARAMETERS. ITS PURPOSE IS TO CHECK WHETHER THE PROPER      *
C    *  INTEGER VALUE OF WINDIR WAS CHOSEN FOR A PARTICULAR WIND DIREC   *
C    *  -TION.                                                           *
C    *                                                                   *
C    *********************************************************************
     SUBROUTINE WINDY(WINDIR,WINVEL,LEVEL,HGT,Q)
     INTEGER WINDIR
     IF(WINDIR .EQ. 1) GO TO 118
     IF (WINDIR .EQ. 5) GO TO 202
     IF(WINDIR .EQ. 3) GO TO 206
     IF(WINDIR .EQ. 7) GO TO 210
     IF(WINDIR .EQ. 2) GO TO 214
     IF(WINDIR .EQ. 4) GO TO 218
     IF(WINDIR .EQ. 6) GO TO 222
     IF(LEVEL .EQ. 0) GO TO 226
     WRITE(6,227)
 227 FORMAT('2 THE POLLUTANT IS FROM THE STACK; AND THE WIND IS BLOWING
    1 IN THE  NORTH WESTERLY DIRECTION.')
     CALL RESULT(Q)
     GO TO 350
 226 WRITE(6,228)
 228 FORMAT('2 THE POLLUTANT IS FROM THE GROUND; AND THE WIND IS
    1BLOWING FROM THE NORTH WESTERLY DIRECTION.')
     CALL RESULT(Q)
     GO TO 350
 118 IF(LEVEL .EQ. 0) GO TO 119
     WRITE(6,200)
 200 FORMAT('2THE POLLUTANT RELEASE IS FROM THE STACK; AND THE WIND
    1 IS BLOWING IN THE NORTHERLY DIRECTION.')
     CALL RESULT(Q)
     GO TO 350
 119 WRITE(6,201)
 201 FORMAT('2THE POLLUTANT RELEASE IS FROM GROUND LEVEL; AND THE
    1 WIND IS BLOWING IN THE NORTHERNLY DIRECTION.')
     CALL RESULT(Q)
     GO TO 350
 202 IF(LEVEL .EQ. 0) GO TO 203
     WRITE(6,204)
 204 FORMAT('2THE POLLUTANT RELEASE IS FROM THE STACK; AND THE WIND IS
    1 BLOWING IN THE SOUTHERNLY DIRECTION.')
     CALL RESULT(Q)
     GO TO 350
 203 WRITE(6,205)
 205 FORMAT('2THE POLLUTANT RELEASE IS FROM GROUND LEVEL; AND THE
    1 WIND IS BLOWING IN THE SOUTHERNLY DIRECTION.')
     CALL RESULT(Q)
     GO TO 350
 206 IF(LEVEL .EQ. 0) GO TO 207
     WRITE(6,208)
 208 FORMAT('2THE POLLUTANT RELEASE IS FROM THE STACK; AND THE WIND
    1 IS BLOWING IN THE EASTERNLY DIRECTION.')
     CALL RESULT(Q)
```

```
      GO TO 350
  207 WRITE(6,209)
  209 FORMAT('2THE POLLUTANT RELEASE IS FROM GROUND LEVEL: AND THE
     1 WIND IS BLOWING IN THE EASTERNLY DIRECTICN.')
      CALL RESULT(Q)
      GO TO 350
  210 IF(LEVEL .EQ. 0) GO TO 211
      WRITE(6,212)
  212 FORMAT('2THE POLLUTANT RELEASE IS FROM THE STACK: AND THE WIND IS
     1BLCWING IN THE WESTERNLY CIRECTION')
      CALL RESLLT(Q)
      GO TO 350
  211 WRITE(6,213)
  213 FORMAT('2THE POLLUTANT RELEASE IS FROM THE GROUNC; AND THE WIND IS
     1BLCWING IN THE WESTERNLY CIRECTION')
      CALL RESULT(Q)
      GO TO 350
  214 IF(LEVEL .EQ. 0) GO TO 215
      WRITE(6,216)
  216 FORMAT('2THE POLLUTANT RELEASE IS FROM THE  STACK; AND THE WIND
     1 IS BLOWING IN THE NORTH-EASTERNLY DIRECTICN')
      CALL RESULT(Q)
      GO TO 350
  215 WRITE(6,217)
  217 FORMAT('2THE POLLUTANT RELEASE IS FROM THE GROUND; AND THE WIND
     1BLOWING IN THE NORTH-EASERNLY DIRECTION')
      CALL RESULT(Q)
      GO TO 350
  218 IF(LEVEL .EQ. 0) GO TO 219
      WRITE(6,220)
  220 FORMAT('2THE POLLUTANT RELEASE IS FROM THE STACK: AND THE WIND
     IS BLOWING IN THE SOUTH-EASTERNLY DIRECTICN')
      CALL RESULT(Q)
      GO TO 350
  219 WRITE(6,221)
  221 FORMAT('2THE POLLUTANT RELEASE IS FROM THE GROUNC; AND THE WIND
     1IS BLOWING IN THE SOUTH-EASTERNLY DIRECTICN')
      CALL RESULT(Q)
      GO TO 350
  222 IF(LEVEL .EQ. 0) GO TO 223
      WRITE(6,224)
  224 FORMAT('2THE POLLUTANT RELEASE IS FROM THE STACK: AND THE WIND
     1IS BLOWING IN THE SOUTH-WESTERNLY DIRECTION')
      CALL RESULT(Q)
      GO TO 350
  223 WRITE(6,225)
  225 FORMAT('2THE POLLUTANT RELEASE IS FROM THE GROUND; AND THE WIND
     1 IS BLOWINGIN THE SOUTH-WESTERNLY DIRECTICN')
      CALL RESULT(Q)
      GO TO 350
  350 RETURN
      END
      SUBROUTINE RESULT(Q)
      WRITE(6,260) Q
  260 FORMAT('THE RELEASE RATE IS:',F5.2,2X,'CURIES A SECOND.')
      RETURN
      END
      SUBROUTINE PEANTS
      REAL   QEXLP1,QEXLP2,QEXLC1,QEXLC2,Q,LAMDA,LEAK,
     1LEAKP,LAMDAE
```

```
      INTEGER TIMINT,NDIST,LEVEL,TRACE,DUOLK
      COMMON Q,EXPOTM,TIMINT,A,CHI,DELDOS,DOSE,DINDOS,THYDOS,
     1TOTDOS,NDIST,EBETA,EGAMA,LAMDAE,BR,ZSI,LEVEL,HGT,TRACE,QCHI,
     1SIGY,SIGZ1,SIGZ3,SIGZ4,SIGZ5,SIGZ6,SIGZ7,QNOT,LAMDA,
     1WNVEL1,WNVEL3,WNVEL4,WNVEL5,WNVEL6,WNVEL7,LEAK,LEAKP,DUOLK
      DIMENSION A(25,1),CHI(25,10),DELDOS(25,10),DOSE(25,10),
     1DINDOS(25,10),THYDOS(25,10),TOTDOS(25,10),HGT(25),
     1SIGY(25),SIGZ1(25),SIGZ3(25),SIGZ4(25),SIGZ5(25),SIGZ6(25),
     1SIGZ7(25),QCHI(25,10)
      J=TIMINT
      IF(DUOLK .EQ. 0) GO TO 335
      QLCPT=LEAKP*LEAK*QNOT/(3600.*8.*(LEAK-LEAKP))
      QEXLP1=(1.-EXP(-(LAMDA+LEAKP)*3600.*8.))
      QEXLP2=1./(LEAKP+LAMDA)
      QEXLC1=(1.-EXP(-(LAMDA+LEAK)*3600.*8.))
      QEXLC2=1./(LEAK+LAMDA)
      Q=QLCPT*(QEXLP1*QEXLP2-QEXLC1*QEXLC2)
      GO TO 336
  335 QC=QNOT*LEAK/(3600.*8.*(LEAK+LAMDA))
      Q=QC*(1.-EXP(-(LEAK+LAMDA)*3600.*8.))
  336 DO 305 I=1,NDIST
      IF(LEVEL .EQ.1)  GO TO 311
      CHI(I,J)=Q/(3.14*WNVEL1*SIGY(I)*SIGZ1(I))
      QCHI(I,J)=CHI(I,J)/Q
      GO TO 312
  311 CHI(I,J)=Q*EXP(-((HGT(I)/SIGZ1(I))**2/2))/
     1(3.14*WNVEL1*SIGY(I)*SIGZ1(I))
      QCHI(I,J)=CHI(I,J)/Q
  312 DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.*8.
      DOSE(I,J)=DOSE(I,J)+DELDOS(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)*
     13600.*8.
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DINDOS(I,J)+DELDOS(I,J)
  305 CONTINUE
      DIFF=EXPOTM-8.
      PIFF=DIFF
      IF(DIFF .GT. 16.) DIFF=16.
      HIFF=DIFF+8.
      TIFF=DIFF/2.+8.
      IF(DUOLK .EQ. 0) GO TO 337
      QL1=3600.*DIFF*LEAK
      QL2=3600.*DIFF*LEAKP
      QL3=QL1-QL2
      QLCPT=LEAK*LEAKP*QNOT/QL3
      QEXLP1=EXP(-(LAMDA+LEAKP)*3600.*8.)-EXP(-(LAMDA+LEAKP)
     1*3600.*HIFF)
      QEXLC1=EXP(-(LAMDA+LEAK)*3600.*8.)-EXP(-(LAMDA+LEAK)
     1*3600.*HIFF)
      Q=QLCPT*(QEXLP1*QEXLP2-QEXLC1*QEXLC2)
      QD1=QNOT*LEAK*LEAKP/(LEAK-LEAKP)
      QD2=EXP(-(LAMDA+LEAKP)*3600.*TIFF)
      QD3=EXP(-(LAMDA+LEAK)*3600.*TIFF)
      Q1=QD1*(QD2-QD3)
      GO TO 338
  337 QC=QNOT*LEAK/(3600.*DIFF*(LEAK+LAMDA))
      Q=QC*(EXP(-(LEAK+LAMDA)*3600.*8.)-EXP(-(LEAK+LAMDA)
     1*3600.*HIFF))
      Q1=QNOT*LEAK*EXP(-(LAMDA+LEAK)*3600.*TIFF)
  338 IF(Q .LE. 0) GO TO 350
```

```
  350 Q=Q1
      IF(Q .LE. 0) GO TO 308
  351 DO 306 I=1,NDIST
      IF(LEVEL .EQ. 1) GO TO 313
      CHI(I,J)=Q*2.032/(SIGZ1(I)*WNVEL1*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
      GO TO 314
  313 CHI(I,J)=Q*2.032*EXP((-(HGT(I)/SIGZ1(I))**2/2))/(WNVEL1*SIGZ1(I)
     1*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
  314 DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.*DIFF
      DOSE(I,J)=DOSE(I,J)+DELDOS(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)*
     13600.*DIFF
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DINDOS(I,J)+DELDOS(I,J)
  306 CONTINUE
      IF(PIFF .GT. 16.) GO TO 307
      GO TO 308
  307 DIFF=EXPCTM-24.
      PIFF=DIFF
      IF(DIFF .GT. 72.) DIFF=72.
      HIFF=DIFF+24.
      TIFF=DIFF/2.+24.
      IF(DUOLK .EQ. 0) GO TO 339
      QL1=3600.*DIFF*LEAK
      QL2=3600.*DIFF*LEAKP
      QL3=QL1-QL2
      QLCPT=LEAK*LEAKP*QNOT/QL3
      QEXLP1=EXP(-(LAMDA+LEAKP)*3600.*24.)-EXP(-(LAMDA+LEAKP)
     1*3600.*HIFF)
      QEXLC1=EXP(-(LAMDA+LEAK)*3600.*24.)-EXP(-(LAMDA+LEAK)
     1*3600.*HIFF)
      Q=QLCPT*(QEXLP1*QEXLP2-QEXLC1*QEXLC2)
      QD1=QNOT*LEAK*LEAKP/(LEAK-LEAKP)
      QD2=EXP(-(LAMDA+LEAKP)*3600.*TIFF)
      QD3=EXP(-(LAMDA+LEAK)*3600.*TIFF)
      Q1=QD1*(QD2-QD3)
      GO TO 340
  339 QC=QNOT*LEAK/(3600.*DIFF*(LEAK+LAMDA))
      Q=QC*(EXP(-(LEAK+LAMDA)*3600.*24.)-EXP(-(LEAK+LAMDA)
     1*3600.*HIFF))
      Q1=QNOT*LEAK*EXP(-(LAMDA+LEAK)*3600.*TIFF)
  340 IF(Q .LE. 0) GO TO 352
      GO TO 353
  352 Q=Q1
      IF(Q .LE. 0) GO TO 308
  353 DO 309 I=1,NDIST
      IF(LEVEL .EQ. 1) GO TO 315
      CHI(I,J)=.4*Q*2.032/(SIGZ3(I)*WNVEL3*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
      GO TO 316
  315 CHI(I,J)=.4*Q*2.032*EXP((-(HGT(I)/SIGZ3(I))**2/2))/(WNVEL3*SIGZ3(
     1*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
  316 DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.
     1*DIFF
      DOSE(I,J)=DOSE(I,J)+DELDOS(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)
```

```
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DELDOS(I,J)+DINDOS(I,J)
  309 CONTINUE
      DO 320 I=1,NDIST
      IF(LEVEL .EQ. 1) GO TO 321
      CHI(I,J)=.6*Q*2.032/(SIGZ4(I)*WNVEL4*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
      GO TO 322
  321 CHI(I,J)=.6*Q*2.032*EXP((-(HGT(I)/SIGZ4(I))**2/2))/(WNVEL4*SIGZ4(
     1*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
  322 DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.
     1*DIFF
      DOSE(I,J)=DOSE(I,J)+DELDOS(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)
     1*3600.*DIFF
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DELDOS(I,J)+DINDOS(I,J)
  320 CONTINUE
      IF(PIFF .LE. 72.) GO TO 308
      DIFF=EXPCTM-96.
      HIFF=DIFF+96.
      TIFF=DIFF/2.+96.
      IF(DUOLK .EQ. 0) GO TO 341
      QL1=LEAK*DIFF*3600.
      QL2=LEAKP*DIFF*3600.
      QL3=QL1-QL2
      QLCPT=LEAK*LEAKP*QNOT/QL3
      QEXLP1=EXP(-(LAMDA+LEAKP)*3600.*96.)-EXP(-(LAMDA+LEAKP)
     1*3600.*HIFF)
      QEXLC1=EXP(-(LAMDA+LEAK)*3600.*96.)-EXP(-(LAMDA+LEAK)
     1*3600.*HIFF)
      Q=QLCPT*(QEXLP1*QEXLP2-QEXLC1*QEXLC2)
      QD1=QNOT*LEAK*LEAKP/(LEAK-LEAKP)
      QD2=EXP(-(LAMDA+LEAKP)*3600.*TIFF)
      QD3=EXP(-(LAMDA+LEAK)*3600.*TIFF)
      Q1=QD1*(QD2-QD3)
      GO TO 342
  341 QC=QNOT*LEAK/(3600.*DIFF*(LEAK+LAMDA))
      Q=QC*(EXP(-(LEAK+LAMDA)*3600.*96.)-EXP(-(LEAK+LAMDA)
     1*3600.*HIFF))
      Q1=QNOT*LEAK*EXP(-(LAMDA+LEAK)*3600.*TIFF)
  342 IF(Q .LE. 0) GO TO 354
      GO TO 355
  354 Q=Q1
      IF(Q .LE. 0) GO TO 308
  355 DO 323 I=1,NDIST
      IF(LEVEL .EQ. 1) GO TO 324
      CHI(I,J)=.333*Q*2.032/(SIGZ5(I)*WNVEL5*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
      GO TO 325
  324 CHI(I,J)=.333*Q*2.032*EXP((-(HGT(I)/SIGZ5(I))**2/2))/(WNVEL5*SIGZ5
     1*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
  325 DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.
     1*DIFF
      DOSE(I,J)=DOSE(I,J)+DELDOS(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)
     1*3600.*DIFF
```

```
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DELDOS(I,J)+DINDOS(I,J)
  323 CONTINUE
      DO 326 I=1,NDIST
      IF(LEVEL .EQ. 1) GO TO 327
      CHI(I,J)=.333*Q*2.032/(SIGZ6(I)*WNVEL6*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
      GO TO 328
  327 CHI(I,J)=.333*Q*2.032*EXP((-(HGT(I)/SIGZ6(I))**2/2))/(WNVEL6*SIG
     1*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
  328 DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.
     1*DIFF
      DOSE(I,J)=DOSE(I,J)+DELDOS(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)
     1*3600.*DIFF
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DELDOS(I,J)+DINDOS(I,J)
  326 CONTINUE
      DO 329 I=1,NDIST
      IF(LEVEL .EQ. 1) GO TO 330
      CHI(I,J)=.333*Q*2.032/(SIGZ7(I)*WNVEL7*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
      GO TO 331
  330 CHI(I,J)=.333*Q*2.032*EXP((-(HGT(I)/SIGZ7(I))**2/2))/(WNVEL7*SIGZ
     1*A(I,1))
      QCHI(I,J)=CHI(I,J)/Q
  331 DELDOS(I,J)=(.228*EBETA+.253*EGAMA)*CHI(I,J)*3600.
     1*DIFF
      DOSE(I,J)=DOSE(I,J)+DELDOS(I,J)
      DINDOS(I,J)=136.16*BR*ZSI*CHI(I,J)/(20.*LAMDAE)
     1*3600.*DIFF
      THYDOS(I,J)=THYDOS(I,J)+DINDOS(I,J)
      TOTDOS(I,J)=TOTDOS(I,J)+DELDOS(I,J)+DINDOS(I,J)
  329 CONTINUE
  308 RETURN
      END
OF FILE
```

APPENDIX II

DATA USED IN RUNNING THE PROGRAM

The horizontal (σ_y) and vertical (σ_z) dispersion coefficients were obtained from figs. (4.3) and (4.4) in reference (16) respectively, and for the atmospheric conditions given in table (A. II-1).

Table (A. II-1)

Atmospheric Conditions and Wind Velocities for Different Exposure Times After the Accident

Exposure Time	Atmospheric Condition	Wind Velocity m/sec
2-24 hr	Pasquill's F	1
1-4 days	40% Pasquill's D	3
	60% Pasquill's F	2
More than 4 days	33.3% Pasquill's C	3
	33.3% Pasquill's D	3
	33.3% Pasquill's F	2

The values of the pool and containment leak rates were 2.9166×10^{-7} sec^{-1} and 7.523×10^{-7} sec^{-1} respectively, and the breathing rate was 3.47×10^{-4} m^3/sec

```
&PENTS1 SIGZ1=1.8,3.,4.2,6.,7.,8.,9.,9.5,10.,12.,13.,14.,
16.,18.,20.,22.,25.,26.,28.,33.,38.,40.,44.,46.,49.,
WNVEL1=1.,SIGZ3=4.,7.,9.5,13.,16.,18.,20.,22.,23.5,27.,29.,
30.,36.,40.,48.,54.,60.,61.,63.5,85.,100.,110.,130.,150.,
160.,WNVEL3=3.,SIGZ4=1.8,3.,4.2,6.,7.,8.,9.,9.5,10.,12.,13.,
14.,16.,18.,20.,22.,25.,26.,28.,33.,38.,40.,44.,46.,49.,
WNVEL4=2.,SIGZ5=7.,11.,17.,22.,28.,33.,40.,41.,48.,51.,54.,
60.,75.,90.,105.,130.,140.,165.,170.,230.,290.,330.,390.,
425.,500.,WNVEL5=3.,SIGZ6=4.,7.,9.5,13.,16.,18.,20.,22.,
23.5,27.,29.,30.,36.,40.,48.,54.,60.,61.,63.5,85.,100.,
110.,130.,150.,160.,WNVEL6=3.,SIGZ7=1.8,3.,4.2,6.,7.,8.,9.,
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