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A Compilation of Documented Computer Codes Applicable to Environmental Assessment of Radioactivity Releases

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Environmental Sciences Division
Publication No. 983

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National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy $5.00; Microfiche $3.00

ORNL/TM-5830
Distribution Categories UC-11, UC-41, UC-79p

Contract No. W-7405-eng-26

A COMPILATION OF DOCUMENTED COMPUTER CODES APPLICABLE TO ENVIRONMENTAL ASSESSMENT OF RADIOACTIVITY RELEASES

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Date Published - April 1977

ENVIRONMENTAL SCIENCES DIVISION
Publication No. 983

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ACKNOWLEDGMENTS

The authors wish to gratefully thank Elizabeth Etnier and Herbert Pomerance for their assistance in obtaining the documentation that made this compilation possible.

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ABSTRACT

HOFFMAN, F. O., C. W. MILLER, D. L. SHAEFFER, C. T. GARTEN, JR., R. W. SHOR, and J. T. ENSMINGER. 1977. A compilation of documented computer codes applicable to environmental assessment of radioactivity releases. ORNL/TM-5830. Oak Ridge National Laboratory, Oak Ridge, Tennessee. pp. 84

The objective of this paper is to present a compilation of computer codes for the assessment of accidental or routine releases of radioactivity to the environment from nuclear power facilities. The capabilities of 83 computer codes in the areas of environmental transport and radiation dosimetry are summarized in tabular form. This preliminary analysis clearly indicates that the initial efforts in assessment methodology development have concentrated on atmospheric dispersion, external dosimetry, and internal dosimetry via inhalation. The incorporation of terrestrial and aquatic food chain pathways has been a more recent development and reflects the current requirements of environmental legislation and the needs of regulatory agencies. The characteristics of the conceptual models employed by these codes are reviewed. The appendixes include abstracts of the codes and indexes by author, key words, publication description, and title.

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INTRODUCTION

At Oak Ridge National Laboratory (ORNL) a project is currently underway to evaluate models which may be used for assessing accidental and routine releases of radionuclides from the Liquid Metal Fast Breeder Reactor (LMFBR). The objective of this project is to recommend the most suitable methods for predicting the environmental transport of radionuclides and the subsequent doses to man. During the initial phase of this research, computer codes have been identified which are directly applicable to the assessment of radionuclides released to the environment from nuclear power facilities. These codes are compiled and reviewed in this paper.

Codes were identified through the use of *Nuclear Science Abstracts* and the resources of the Ecological Sciences Information Center at ORNL. Computer codes were particularly sought in the subject areas of aquatic and atmospheric dispersion, deposition and resuspension, aquatic and terrestrial food chain transport, and the calculation of doses resulting from internal and external modes of exposure. Only those computer codes referenced in the open literature were included in this study. In some cases, the author was consulted directly because certain aspects of the code were not documented.

Previous compilations of computer codes have been made by Winton[1-3] and Strenge et al.[4] Winton's compilations, however, are primarily concerned with nuclear accident analysis codes. Only a small fraction of the more than 200 codes included in his surveys deal with the environmental transport and dosimetry of radionuclides. The review by Strenge et al.[4] includes a detailed discussion of 23 environmental transport and dosimetric computer codes, as well as a brief description of the program and special notes on mathematical techniques, limitations, and programming considerations. A discussion of near- and far-range atmospheric pathway models also appears as an appendix to their report.

In the present study, 83 computer codes are identified which may be implemented for assessment of radiological consequences of routine and/or accidental discharges from nuclear power facilities. Information about characteristics of these codes and the type of documentation available at this time is represented in tabular form. This information is intended to be a summary of the capabilities of these codes. More detailed comments about each code have been incorporated into a computerized data base which is available upon request through the Ecological Sciences Information Center at ORNL. This data base includes an abstract of the referenced document, a list of keywords, and, where necessary, additional clarifying comments by the abstractor. Appendixes A–E show the contents of the data base as of January 1977.

A DESCRIPTION OF THE TABLE

The codes compiled as a result of this study are listed in Table 1. This table includes information about the specific areas of environmental transport and dosimetry for which the codes were developed, as well as information concerning their use for assessment of accidental or routine releases. In this report, routine releases are considered to be releases of a chronic nature resulting from normal plant operation. Accidental releases are considered to be relatively short, intermittent releases resulting from abnormal operating conditions and producing higher than normal amounts of radioactivity. Although individual models within each code may be potentially applicable to both types of releases, the classification used in the table is based upon the capabilities of the entire code in which these models appear. Table 1 also reports the computer language of the code, the computer for which it was originally developed, and the degree of documentation available to the authors at the time of this writing.

The extent of documentation of the codes contained in this compilation varies considerably. In order to portray this variation, three broad categories of documentation have been arbitrarily defined. The terms chosen to represent these categories are: (1) extensive documentation, (2) partial documentation, and (3) sparse documentation. "Extensive documentation" indicates that a description of the mathematical models, a program listing of the computer code, and a description of the input data needed to run the program are available. "Partial documentation" indicates that a description of the mathematical models is available, but that information on the input needed to run the code, and, or the listing of the code is missing. "Sparse documentation" indicates that only a description of the objectives or the conceptual models of the code is available. In some cases, sparse documentation refers only to the availability of an abstract. The term "complete documentation" has been purposefully avoided because even the most extensively documented codes frequently require consultation with the originator of a code before implementation can be realized. The table also shows those codes that contain a documented sample problem.

Blank areas in the table occur if the column designation is not applicable to the code of reference or if the information could not be obtained from the documentation at hand.

A SUMMARY OF THE TABLE

Figure 1 depicts the frequency of occurrence within the environmental transport and dosimetric categories of the computer codes compiled. The figure indicates that the assessment of atmospheric dispersion, external dosimetry and internal dosimetry via inhalation have predominated in the development of radiological assessment computer

Table 1. A listing of the computer codes and their characteristics

Codes	Environmental Transport										Human Dietary and Behavioral Factors[a]	External Dosimetry				Internal Dosimetry		Computerization				Reference
	Atmospheric	Wet Deposition	Dry Depostion	Resuspension	Surface Water	Ground Water	Sedimentation	Irrigation	Terrestrial Foods	Aquatic Foods		Air Exposure	Ground Exposure	Shoreline Exposure	Water Exposure	Inhalation	Ingestion	Langauge[b]	Computer[c]	Documentation[d]	Sample Problem[g]	
ACRA	A[e]											A				A				P	✓	Stallman and Kam, 1973[1]
ACRA-II	A	A	A									A				A		F	360/ 91	P	✓	RSIC, 1974[2]
ADPIC	A R[e]				A R													L	7600	P		Lange, 1973[3]
AERIN	-															A	A	F IV		E	✓	Voillequé, 1970; 1968[4,5]
AIRDOS	R	R	R						R		R	R	R	R	R	R	R	F IV	360/ 91	E		Moore, 1975[6]
AIREM	R	R	R									R				R		F IV	370	E	✓	Martin et al, 1974[7]
AISITE II	A R											A R				A R		F II	7094	P	✓	Blaine and Bramblett, 1964[8]
AQUAMOD					A R		A R			A R								F IV	360/ 91	S		Booth, 1975[9]
ARADS	R																	F IV	1401/ 7040	P		Plato et al, 1967; 1969[10,11]
ARCON	A		A																	P		Tveten, 1974[12]
Armstrong and Gloyna					A R													F 63	1604	P		Armstrong and Gloyna, 1968[13]
ARRRG					R					R	R			R	R		R	B	1108	E	✓	Soldat et al, 1974[14]
ATM	R	R	R	R														F IV	360/ 91	E	✓	Culkowski and Patterson, 1976[15]
CARDOCC					R					R	R						R	F IV	360	P		Watts, 1976[16]
CEDRIC																A R	A[f] R	F IV		P		Clarke, 1972[17]

Table 1. (continued)

Codes	Environmental Transport										Human Dietary and Behavioral Factors(a)	External Dosimetry				Internal Dosimetry		Computerization				Reference
	Atmospheric	Wet Deposition	Dry Depostion	Resuspension	Surface Water	Ground Water	Sedimentation	Irrigation	Terrestrial Foods	Aquatic Foods		Air Exposure	Ground Exposure	Shoreline Exposure	Water Exposure	Inhalation	Ingestion	Langauge(b)	Computer(c)	Documentation(d)	Sample Problem(g)	
Charak	A	A																F	1604	P		Charak, 1967[18]
CLOUD	A											A				A		F IV	6600	E	✓	Bonzon and Rivard, 1970[19]
COLHEAT					A R															S		Daniels et al. 1970[20]
COMRADEX I, II, III	A											A				A		F		E	✓	Willis et al. 1970[21] Specht et al, 1975[22] Otter and Conners, 1975[23]
CURIE/DOSE THUNDERHEAD	A											A				A		F		P	✓	Kenfield et al, 1965[24]
DACRIN	A R															A R			74	E	✓	Houston et al, 1974[25] Strenge, 1975[26]
DIFOUT	A R		A R																3600	E	✓	Luna and Church, 1969[27]
DOSE B												A R				A R			6600	S	✓	Ellison and Dunham, 1967[28]
DPRWCR					A R													F IV	11/45	P		Ahlstrom and Foote, 1976[29]
DPRWGW						A R												F IV	11/45	P		Ahlstrom and Foote, 1976[29]
Duguid and Reeves						A R												F IV	360/91	E	✓	Reeves and Duguid, 1975[30] Duguid and Reeves. 1976[31]
EERIE	A		A									A				A				S		MacDonald, 1971[32]
EGAD	R											R						F IV	360/65	E	✓	Cooper. 1972[33]
Engelmann and Davis	A R	A R																	7090	P		Engelmann and Davis. 1968[34]

Table 1. (continued)

Codes	Environmental Transport										Human Dietary and Behavioral Factors[a]	External Dosimetry				Internal Dosimetry		Computerization				Reference
	Atmospheric	Wet Deposition	Dry Depostion	Resuspension	Surface Water	Ground Water	Sedimentation	Irrigation	Terrestrial Foods	Aquatic Foods		Air Exposure	Ground Exposure	Shoreline Exposure	Water Exposure	Inhalation	Ingestion	Langauge[b]	Computer[c]	Documentation[d]	Sample Problem[g]	
ERGAM												A							360/ 195	S		Cooper, 1976[35]
ESDORA	A R	A R	A R									A R				A R		F IV	7090	S	✓	Cruz, 1973[36]
EXDOSE	A		A									A						F		E	✓	Hendrickson, 1968[37]
EXGAM												A						F IV	3600	P	✓	Steyn and Kim, 1966[38]
EXREM I, II, III												A R	A R	A R	A R			F IV	360/ 91	E	✓	Turner et al, 1968[39] Turner, 1969[40] Trubey and Kaye, 1973[41] Killough and McKay, 1976[42]
FETRA					A R													F IV	7600	P		Onishi et al, 1976[43]
FOOD		R	R					R	R		R						R	B	1108	P		Baker et al, 1976[44]
GADOSE/ DOSET	A	A	A								A	A				A		F	7040	S	✓	Lee et al, 1966[45]
GDOS	R											R								S	✓	Thykier-Nielsen, 1974[46] Hedemann. 1974[47]
Gogolak	R											R							6600	S		Gogolak, 1973[48]
GRONK	R	R	R						R		R	R				R	R	B	1108	E	✓	Soldat et al, 1974[14]
Heffter et al	A R	A R	A R																	P		Heffter et al, 1975[49]
HERMES	R	R	R		R	R	R	R	R	R	R	R	R	R	R	R	R	F V	1108	E		Fletcher and Dotson, 1971[50] Soldat. 1971[51]
INDOS I, II, III																A R	A R	F IV	10	E	✓	Killough and Rohwer, 1974[52]
INDOS2	A R	A R	A R													A R				S		Thykier-Nielsen, 1974[46]

Table 1. (continued)

Codes	Environmental Transport: Atmospheric	Wet Deposition	Dry Depostion	Resuspension	Surface Water	Ground Water	Sedimentation	Irrigation	Terrestrial Foods	Aquatic Foods	Human Dietary and Behavioral Factors(a)	External Dosimetry: Air Exposure	Ground Exposure	Shoreline Exposure	Water Exposure	Internal Dosimetry: Inhalation	Ingestion	Computerization: Langauge(b)	Computer(c)	Documentation(d)	Sample Problem(x)	Reference
INHEC	A R											A R				A R		F IV	370	E	✓	Mackay and Ely, 1974[53]
INREM																A R	A R	F IV	360/91	E	✓	Turner et al. 1968[39] Killough et al. 1975[54]
ISOLA II	R											R						F IV		P	✓	Hübschmann and Nagel. 1975[55]
KRONIC	R											R							1108	E	✓	Strenge and Watson, 1973[56]
Lanes	A R	A R	A R													A R		F II	1620	E	✓	Lanes. 1964[57]
MESODIF	A R																	F IV	360/75	E	✓	Start and Wendall, 1974[58]
METEO-1	A R	A R	A R																	P		Veverka et al, 1975[59]
Mills et al	A R		A R	A R																F	✓	Mills et al, 1974[60]
MO142	A											A				A		F		P	✓	Eckert. 1964[61]
MUNDO	A		A								A	A				A		F		P		Heller et al. 1967[62]
Nowicki	R	R	R									R				R				P	✓	Nowicki, 1975[63]
NUBE	A	A	A									A				A				S		Alonso, 1965[64]
NURSE-I	A		A									A	A			A		F 3600	3600	P		Couchman et al, 1964[65]
NUS																A R	A R	F	7090	P		Kim, 1963[66]
PLUME	A											A				A		F IV		P	✓	Binford et al. 1967[67]

Table 1. (continued)

Codes	Environmental Transport: Atmospheric	Wet Deposition	Dry Deposition	Resuspension	Surface Water	Ground Water	Sedimentation	Irrigation	Terrestrial Foods	Aquatic Foods	Human Dietary and Behavioral Factors(a)	External Dosimetry: Air Exposure	Ground Exposure	Shoreline Exposure	Water Exposure	Internal Dosimetry: Inhalation	Ingestion	Computerization: Langauge(b)	Computer(c)	Documentation(d)	Sample Problem(g)	Reference
PUDEQ																A(f)				P		Houston and Heid, 1975[68]
RACER	A											A							1108	P	✓	Strenge et al, 1971[69]
RADOS	A											A						F IV	360/ 65	E	✓	Cooper, 1967[70]
RADS	A																	F IV	1401/ 7040	P		Plato et al, 1967;1969[10,11]
Reeves et al	R		R																	P	✓	Reeves et al, 1972[71]
RELISH	A																	F IV	360/ 75	E	✓	Binford et al, 1970[72]
RISC	A	A	A													A			7090	S		Anno et al, 1963[73]
RSAC	A R	A R	A R									A R	A R			A R		SM F IV	7040/ 44	E E	✓	Coates and Horton, 1966[74] Richardson, 1968[75]
RSAC-2	A R	A R	A R									A R	A R			A R		F IV	360/ 75	S		Wenzel, 1974[76]
RUBY												R				R		F IV	370/ 145 or 158	E	✓	Boone et al, 1975[77]
Sato et al	R											R						06N	2206	S		Sato et al, 1968[78]
SEP	A R																	F IV	360/ 75	P		Sagendorf, 1974[79]
SERATRA					A R													F IV	7600	P	✓	Onishi et al, 1976[43]
Shih and Gloyna					A R													F 63	1604	P		Shih and Gloyna, 1967[80]
STAREL	R																	F IV	360/ 75	E	✓	Binford et al, 1970[72]

Table 1. (continued)

Codes	Environmental Transport										Human Dietary and Behavioral Factors[a]	External Dosimetry				Internal Dosimetry		Computerization				Reference
	Atmospheric	Wet Deposition	Dry Depostion	Resuspension	Surface Water	Ground Water	Sedimentation	Irrigation	Terrestrial Foods	Aquatic Foods		Air Exposure	Ground Exposure	Shoreline Exposure	Water Exposure	Inhalation	Ingestion	Langauge[b]	Computer[c]	Documentation[d]	Sample Problem[g]	
STRAP II	A																			P		Bell and Houghton, 1969[81]
SUBDOSA	A											A							74	E	✓	Strenge et al, 1975[82]
TERMOD									A R		A R							F IV	360/ 91	P		Booth et al. 1972[83] Booth and Kaye, 1971[84]
Travis	A		A	A														F IV		E	✓	Travis. 1975[85]
VADOSCA	R		R		R		R	R	R	R	R	R	R	R	R	R	R	F	635	P		Bramati et al. 1973[86]
WEERIE	A R		A R									A R				A R		F IV	370/ 165	P		Clarke. 1973[87]
Wong																A[f] R	A R	F		E	✓	Wong, 1967[88]
WRED	A											A							360/ 65	P		Cooper. 1969[89]
YIELDS																A	A			E		Chester, 1974[90]

[a]Includes such factors as breathing rate, rate of food intake, occupancy rate, etc.

[b]B = BASIC, F = FORTRAN, L = LRLTRAN, SM = SYMBOLIC, N = NARC.

[c]Refers to the model numbers of the following computers: CDC 1604, 3600, 6600, 7600, CYBER 74; GE 635; PDP-10, 11/45 UNIVAC 1108; IBM360, 360/91, S360/65, 360/65, 7040, 1401/7040, 1620, 7090, 370/165, 360/195, 370/145, 370/158; NEAC-2206.

[d]E = Extensive, P = Partial, S = Sparse.

[e]A = Accidental release, R = Routine release

[f]Includes intake via wounds.

[g]✓ = yes

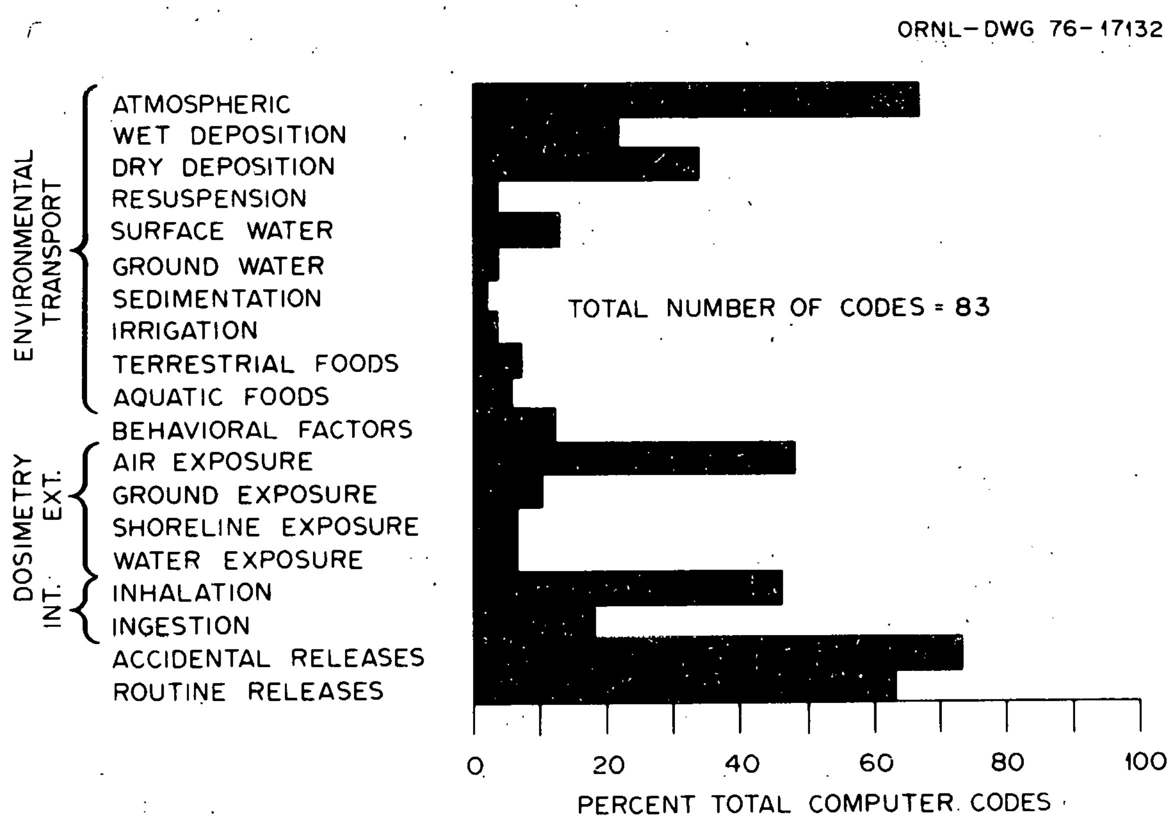

Fig. 1. The frequency of occurrence within the environmental transport and dosimetric categories of the computer codes compiled.

codes. This emphasis probably reflects the need for such codes in safety evaluation work associated with Preliminary Safety Analysis Reports (PSAR's) in which assessment of radiological consequences of accidental releases is of primary importance.

The incorporation of other pathways of exposure into computer codes for assessing radiological safety has been a fairly recent development and reflects the need for satisfying the current requirements of environmental legislation and the needs of regulatory agencies. Previously, calculations of potential exposures resulting from food chain transport and the subsequent ingestion of foods were performed for a few nuclides (e.g., ^{131}I, ^{90}Sr, ^{137}Cs). Computations by hand were usually satisfactory for these cases. This situation changed with the inclusion of a large number of nuclides in the source term and the need to estimate doses resulting from multiple exposure pathways. Nevertheless, fewer than 10% of the codes encountered have the capability of estimating aquatic and terrestrial transport processes.

COMMENTS ON CODE CHARACTERISTICS

Table 1 shows that some codes are applicable to a variety of problems whereas others are very specific in purpose and scope. However, neither the table nor the figure provide any information on the conceptual models used in each code. Although 83 codes have been identified, it will be noted that many of these codes share a similar mathematical approach for calculation of environmental transport and dosimetry of radionuclide releases.

Atmospheric Transport

Nearly all of the codes dealing with atmospheric transport are based on the Gaussian plume dispersion model. Most of the older codes use the Gaussian model formulated by Sutton. The newer ones are based on the formulation of Pasquill.[5] Some codes have modified these basic models to account for ground deposition and depletion of the plume by one or more processes and to account for the presence of an upper bound on the atmospheric diffusion layer. However, most of these simple Gaussian models assume straight-line, one-dimensional air flow and do not consider effects of spatial and temporal meteorological variations. A few codes use more complicated models for calculating dispersion on a regional scale (10 to 100 miles) where these effects become important. MESODIF, for example, employs a two-dimensional puff advection, Gaussian dispersion model, and ADPIC is based on a three-dimensional particle-in-cell trajectory model.

Aquatic Transport

The few aquatic transport models which are surveyed are also of varying degrees of sophistication. For example, some codes such as ARRRG, CARDOCC, HERMES, and VADOSCA calculate radionuclide concentration in water by assuming a simple algebraic relationship between effluent discharge concentration, a mixing factor, and average turnover rate of receiving water at the point of interest. More sophisticated treatments are represented by codes in which the models are based on the solution to a transport equation. Solution of a one-dimensional transport equation for radionuclides introduced into aquatic systems is computed by the code, TRNSPRT, and a code developed by "Armstrong and Gloyna." Two-dimensional transport equations are solved by the finite-element Galerkin model used in FETRA and the finite-difference model used in SERATRA. These codes have been specifically developed for estimating aquatic transport of radionuclides and sediments. ADPIC, DPRWGW, and DPRWCR are in the three-dimensional category. The "particle-in-cell" approach utilized in ADPIC was initially developed for atmospheric transport problems, but the code can, with some limitations, also be adapted to surface water transport. The DPRW codes incorporate generalized transport models using the "discrete-parcel-random-walk" approach.

Food-Chain Transport

The few codes developed for estimation of terrestrial and aquatic food-chain transport all employ a systems approach in which empirically derived transfer coefficients are applied to calculate radionuclide concentrations along various pathways. Transport from water to aquatic foods is generally calculated through use of a single empirical transfer coefficient called a "bioaccumulation factor," whereas calculation of transport from air to terrestrial foods employs a multiple series of transfer coefficients to account for such phenomena as deposition, vegetation retention, animal grazing habits, etc. Most of these codes have been specifically intended for assessment of routine releases in that the models assume steady-state or equilibrium conditions. The models incorporated in AQUAMOD and TERMOD, however, are time-dependent and can be used potentially for both accidental and routine releases. Unlike other codes, AQUAMOD includes more detail in the description of radionuclide transport in the acquatic ecosystem, but much of this detail cannot be used because of insufficient empirical data. It is also of interest to note that of the nine codes which can be used to calculate food chain transport, four of the codes (FOOD, GRONK, ARRRG, and CARDOCC) have been developed from the initial models incorporated in HERMES.

External Dosimetry

Doses resulting from external modes of exposure are calculated by approximately half of the codes included in this survey. The primary mode of external exposure considered by these codes is exposure to contaminated air. Only about 10% of the codes listed in Table 1 are concerned with the external dose received from ground, shoreline, or water contamination. The calculation of air exposure is performed assuming a spatial distribution of radionuclides determined by an infinite or semi-infinite uniform cloud model or a finite Gaussian plume model.[5] The semi-infinite or infinite cloud model is assumed without exception for the calculation of the exposure to beta radiation, but nearly half of the codes employ the finite cloud model for calculation of the exposure to gamma radiation. The finite cloud model is used because the semi-infinite and infinite cloud models tend to underestimate the dose received from gamma radiation at distances close to the point of release from elevated sources. Regardless of the type of radiation considered, the codes which calculate external dose from exposure to contaminated water assume an infinite or semi-infinite medium having a uniform distribution of radionuclides. Codes which calculate ground and/or shoreline exposure usually represent ground and shoreline as infinite planes of negligible thickness having a uniform distribution of radionuclides.[6] Only VADOSCA assumes a finite thickness for shoreline contamination.

Although whole-body dose is calculated by all of the codes which consider external dosimetry, only a few codes such as EXREM III, GRONK and SUBDOSA consider the effect of dose attenuation with depth of tissue penetration by radiation. GRONK, however, is an example of a code which employs dose-conversion factors to calculate external dose after concentrations of radionuclides in an environmental medium have been determined. These factors, which consider the effective absorbed energy, the assumed spatial distribution, and the tissue penetration of radiation emitted from radionuclides, are calculated prior to input in the code.

Internal Dosimetry

Most of the models used for calculation of internal doses are those recommended by the International Commission on Radiological Protection (ICRP).[7] Some codes provide only a listing of values for a single parameter which converts intake rate of a radionuclide into dose. These dose conversion factors must be calculated prior to input. Other codes include the detailed parameters of the ICRP models, such as mass of the organ and fractional uptake, retention and effective absorbed energy of the radionuclides. INREM is the only code for which calculation of age-dependent internal doses is stated as a specific objective. The internal dosimetry models of ICRP Publications 10 (Ref. 8) and 10A (Ref. 9) have been incorporated by INDOS I, II, III, and CEDRIC. The ICRP Task Group Lung Model[10] represents the highest degree of sophistication for calculating doses resulting from inhalation of radionuclides. The only codes which have attempted to implement this model to date are AERIN and DACRIN. Neither of these codes, however, considers the additional contribution to total dose resulting from the complete production of daughter nuclides in nuclide decay schemes, except insofar as this contribution has been factored into effective absorbed energy parameters, which are supplied as input.

COMMENTS ON MODEL UNCERTAINTIES

Among the models incorporated in the computer codes compiled in this survey, the greatest uncertainty is usually associated with those models developed to estimate environmental transport. This is the area in which model validation efforts should concentrate. Much field work still remains to be done to determine specific conditions under which models estimating environmental transport of radionuclides are suitable and to determine potential errors associated with their use. A degree of uncertainty is also inherent with internal dosimetry models, because of natural variation among individuals in a population. Model validation, in this case, is impractical because of the great difficulty encountered in monitoring internal dose to humans. To be effective for general radiological protection purposes, internal dose calculations must be based upon parameters whose values are considered to be adequately representative of individual members of the general population.

COMMENTS CONCERNING COMPUTER CODE TRANSFERABILITY

Some researchers might decide from information given in Table 1 that a particular code has potential application to their needs. Considerable difficulty may be involved in transferring a code from one computer to another. Some codes are written in a language unique to one computer system (e.g., BASIC, FORTRAN V, and LRLTRN) and may require a major rewrite before transfer to another computer is possible. Higher levels of FORTRAN (e.g., FORTRAN IV) also have some fundamental incompatibilities with lower levels of FORTRAN (e.g., FORTRAN II). Consequently, codes written in higher level FORTRAN will probably not compile on a lower level compiler. Conversely, FORTRAN II has a few features which are fundamentally incompatible with FORTRAN IV. Therefore, if a compiler for the appropriate level of FORTRAN is unavailable, a code may not be machine transferable unless a major rewrite is undertaken.

Additionally, problems might arise due to differences between compilers of different vendors for the same source language (e.g. FORTRAN IV). In fact, incompatibilities sometimes exist between two different compilers developed by one particular vendor (e.g., the F-10 and F-40 compilers for the PDP-10 computer). Problems can be expected also with codes that contain macros (i.e., assembly language routines). These codes are machine dependent. Macros can even cause incompatibilities between like computers located at different installations. A further problem is the fact that even a code identified as having "extensive documentation" may still have an incomplete data base and may be so complex in structure and devoid of comments as to render any implementation and/or minor modifications difficult. In attempting to transfer a code from one computer to another, other factors which must be considered are requirements for core size, physical storage of data, word size, cost, and execution time. Any one of these factors can render a computer code impractical for implementation on a machine other than that for which it was originally designed.

THE DETERMINATION OF A SUITABLE CODE

From this initial survey of environmental transport and dosimetric computer codes, no single code can be selected as the best possible for a particular purpose. Even those codes employing models which represent the most sophisticated approaches currently available may not be suitable because of limitations in transferability among computers, input data, validation potential or operation costs. These considerations may commend the use of a less sophisticated code whose results give an acceptable degree of accuracy. The recommendation of a suitable code will depend upon the objectives for which the code will be used and a critical evaluation of its models, parameters, and implementation requirements, as well as verification of the adequacy of its predicted results. Such an evaluation is currently being undertaken by the authors of this paper.

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APPENDIXES

A computerized data base containing information on papers describing the codes included in this study is maintained by the Ecological Sciences Information Center at ORNL. In Appendix A bibliographic information and abstracts of the documents as of January 1977 in the data base are presented. In Appendix B through E, author, keyword, publication description, and title indexes to the bibliographic material are presented.

To obtain additional information concerning the data base, or to initiate a search, contact:

J. T. Ensminger
Oak Ridge National Laboratory
Ecological Sciences Information Center
P.O. Box X, Building 2029
Oak Ridge, Tennessee 37830

Telephone: 615-483-8611, extension 3-6524,
or FTS 850-6524

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APPENDIX A. ABSTRACTS OF THE COMPUTER CODES

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<1>
Reeves, M., and J.D. Duguid, Water Movement Through Saturated-Unsaturated Porous Media: A Finite - Element Galerkin Model.

1975, February. ORNL-4927; 236 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

A two-dimensional transient model for flow through saturated-unsaturated porous media has been developed. This model numerically solves the governing partial differential equations, which are highly nonlinear. The model code uses quadrilateral finite elements for the geometrical assembly, bilinear Galerkin interpolation for the spatial integration, and Gaussian elimination for the solution of the resulting matrix equations. In addition to the usual constant-flux and constant-head boundary conditions, the code is capable of applying pressure-dependent boundary conditions at the ground surface. Thus, infiltration into or seepage from this surface may be simulated. Each element may be assigned different material properties that allow the investigation of layered geologic formations. The formulation of the governing equations and the computer implementation are presented. The report is intended for use as a complete user's manual and contains a listing of the computer code (in FORTRAN) along with both input and output data for two example problems. The results of a computer simulation are compared with experimental data obtained by J.D. Hewlett and A.R. Hibbert from an inclined soil slab at Coweeta Hydrologic Laboratory in North Carolina. The computer model gives good results in this simulation. The Galerkin finite-element method is found to be superior to the finite-difference method used by previous investigators. A comparison with a finite-difference model developed by R.A. Freeze is made. By exploiting the flexibility of the finite-element geometrical discretization, the user may easily reduce computer running time by a factor of two. This code solves for the fluid velocity which is to be used as input to another model which treats material transport. The air-water system is treated as a single phase (no air pockets allowed). (auth) (DLS)

The code requires an extensive data base. The geometry and geology of the region under investigation must be specified.

<2>
Alonso, A., "NUBE" - A Digital Code to Evaluate the Hazards of Different Types of Reactor Accidents.

1965. CONF-650407; Part of Proceedings of the International Symposium on Fission Product Release and Transport Under Accident Conditions, held at Oak Ridge, Tennessee, April 5-7, 1965, p. 1080-1090, 1249 p. (Junta de Energia Nuclear, Madrid, Spain)

An analytical method has been developed to assess abnormal reactor behavior and has been coded for digital computation. The method determines the fission product inventory as a function of reactor power, fuel irradiation, and cooling time. It analyzes the fission product behavior within the containment system, when released from the core as a result of an accident. Finally, the method calculates the atmosphere diffusion of the escaping nuclides under a variety of atmospheric conditions. External and inhalation doses to individuals within the radioactive plume may be assessed at different distances downwind from the reactor. The method has been used to evaluate the hazards caused by different types of reactor accidents such as reactivity accidents, loss-of-coolant accidents, reactor fires, and fuel handling accidents. Different types of containment systems are considered in the analysis. The systems analyzed include negative pressure containment, double containment, and high-pressure and pressure suppression containment systems. The effect of engineered safeguards such as building sprays, filters, condensation pools, etc. can be taken into account. Each nuclide can be considered individually, as the cleaning rates, condensation ratios, deposition velocities, etc. are nuclide dependent. The results are given in terms of the amounts of the most important nuclides retained within the containment barriers or by the engineered safeguards. The most relevant health hazards produced outside the containment system are also given for the accidents analyzed. (auth)

An obscure statement is made to the effect that certain parts of the code are conventional and therefore will not be fully developed. It is not clear whether the effect of ground contamination is included in the calculation of dosage.

<3>
Cooper, R.E., EGAD - A Computer Program to Compute Dose Integrals from External Gamma Emitters.

1972, September. DP-1304; 28 p. (Savannah River Laboratory, Aiken, SC 29801)

EGAD is an External Gamma Dose code programmed in FORTRAN IV for the IBM 360/65 that is intended to provide an estimate of whole body dose resulting from radioactive material released to the environs. The material is assumed to be bounded between the ground and an inversion lid on the vertical axis and bounded in the crosswind direction between fictitious sector boundaries or some real physical constraint such as mountains on both sides of a valley. To model average exposure over long time periods, the distribution of material is assumed to be Gaussian about a given release height in the vertical direction with a homogeneous distribution in the crosswind direction. The code provides output in the form of dose integrals that are a function of sigma sub z (the vertical dispersion parameter), release height, inversion lid, and gamma energy. The calculated integrals are independent of the source, radioactive decay, wind speed, and sector width. It is therefore feasible to cover a large variety of possibilities by using a relatively small set of tabular values as a function of sigma sub z and employing an interpolation procedure. The code is expected to be used primarily in the construction of these tabular values to be used in a more comprehensive program of computing radiation dose. The code is currently being used in this manner as a part of the overall man-rem evaluation program at Savannah River and permits gamma dose calculations to be performed at a rate in excess of 20,000 individual cases per second on the IBM 360/65, including the processing time for the meteorological data involved. (auth)

<4>
Soldat, J.K., Modeling of Environmental Pathways and Radiation Doses from Nuclear Facilities.

1971, October. BNWL-SA-3939; 33 p. (Battelle, Pacific Northwest Laboratories, Richland, WA 99352)

<4> CONT.

A computer model was developed which calculates total annual radiation doses and 50-year dose commitments to several categories of persons at population centers and combines these calculated doses into integrated (man-rem) annual and 50-year doses for large populations. The dose model was designed as one portion of an overall computer program developed to delineate the nuclear facilities expected in the year 2000, the radionuclides released to air and water, their diffusion, dispersion, and reconcentration in the environment, and the resulting radiation doses to people. Equations for calculating dose factors were derived from those given by the International Commission on Radiological Protection (ICRP). Effective decay energies for various radionuclides were calculated from the ICRP model, which assumes that all the radionuclide is in the center of a spherical organ with an appropriate effective radius. Where data were lacking, metabolic parameters for the Standard Man were used for other ages as well. The model includes a sub-routine which calculates radionuclide concentrations in a wide variety of foods at time of harvest from concentrations in air, irrigation water, and soil. The latter concentrations are output from previous portions of the overall computer program. The tremendous variety of data required for proper dose assessment is not always available in the literature. Such data has been extrapolated from environmental studies of fallout nuclides and from planned laboratory and field studies of a limited number of nuclides. (auth)

This paper has been updated by BNWL-1754 (1974) and modified in its implementation by USNRC Reg. Guide 1.109 (1976). The approach is that which was incorporated into the HERMES code. The dose program utilizes data generated by or stored in other portions of HERMES, but could be operated independently if such data were available directly.

<5>

Killough, G.G., P.S. Rohwer, and W.D. Turner, INREM - A FORTRAN Code Which Implements ICRP 2 Models of Internal Radiation Dose to Man.

1975, February. ORNL-5003; 140 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

INREM is a FORTRAN IV computer code to estimate the cumulative dose equivalents to body organs of man following a continuous intake, through inhalation or ingestion, of one or more radionuclides. Dosimetry parameters for the organs are dependent on the age of the individual, and the intake rate is a function of time and the individual's age. The method of estimation is based on adaptations of models presented in Publication 2 of the International Commission on Radiological Protection. The equations for the models are derived, and information about the code and its use is provided. The first version of INREM was described in 1968 by Turner, Kaye, and Rohwer (1968). The code has subsequently undergone a number of modifications, with the result that the 1968 report is now obsolete. In view of the continuing usage of INREM, however, the present report is intended to provide up-to-date documentation. (auth) (POH)

Currently in the process of being updated to include the effects of cross irradiation among internal organs.

<6>

Chester, R.O., Biological Dose and Radiological Activity from Nuclear Reactor or Nuclear Weapon Fission Products.

1974, December. ORNL-4996; 89 p. (Oak Ridge National laboratory, Oak Ridge, TN 37830)

This report describes the use of a computer code, YIELDS, that performs rapid, accurate calculation of dose and activity from the fission products in a nuclear reactor core or from a nuclear weapon. For example, if a dispersal model is assumed, calculations can be made of contaminated area, dose from inhaled or ingested fission products to each part of a biological system, or dose from external fission products, from a dispersed reactor core, or nuclear weapon fallout. Individual beta and gamma ray energies for each isotope are included, permitting detailed calculation of biological dose or other functions of these energies vs time. The calculation method facilitates the handling of complicated radiological decay chains coupled with multisection biological systems. The option of sorting the output by isotopes or chemical species is available. The model may be modified to account for routine releases. The code is basically an internal dosimetry calculation based on inhalation. Ingestion apparently can be taken into account by modifying the appropriate input parameters. The radionuclide inventory can be specificed as input. (auth) (POH)

This documentation of the YIELDS code is confusing. The reader should utilize care in reviewing this manuscript. The term "Ingestion" often appears to be misused. The code assumes that an atmospheric dispersion model is already available. The documentation is rather opaque; consequently the implementation of this code would be difficult without the assistance of the author.

<7>

Turner, W.D., The EXREM II Computer Code for Estimating External Doses to Populations from Construction of a Sea Level Canal with Nuclear Explosives.

1969, July 21. CTC-8; 91 p. (Union Carbide Corporation, P.O. Box P, Oak Ridge, TN 37830)

EXREM II is a computer code to estimate the dose equivalent rate and the total dose equivalent from both beta and gamma radiation resulting from submersion in contaminated water, submersion in contaminated air, and exposure to a contaminated surface. There can be more than one environmental release, and exposure can begin at any time after the first release. EXREM II considers contributions from environmental releases and from nuclide chains. For a particular problem the user may choose to calculate either the dose rates, or the total doses, or both for any of the three modes of exposure. A separate solution array is printed for each mode of exposures. (auth)

This code has been updated by EXREM III, EXREM II has evolved from EXREM by considering nuclide decay schemes.

<8>

Stallmann, F.W., and F.B.K. Kam, ACRA - A Computer Program for the Estimation of Radiation Doses Caused by a Hypothetical Reactor Accident.

1973, April. ORNL-TM-4082; 82 p. (Oak Ridge National Laboratory, Oak Ridge, TN)

<8> CONT.

The radioactive material released to the atmosphere on account of a hypothetical reactor accident is transported away by the wind and is dispersed to a 3-dimensional normal distribution. The fission product activity is described throughout the space domain as a function of time. Presently the code calculates two types of exposure from the passing cloud of finite size: internal exposure from inhalation of the radionuclides in the air, and external dose from gamma radiation. The cloud depletion terms by washout and dry deposition have not been incorporated. However, this option can be easily implemented. Up to 100 receptor locations and 20 time intervals can be handled in one run. The program accepts information for up to 300 isotopes. Computer time is about 2 seconds per receptor and time interval. This time has to be doubled if inversion layers are present. (auth)

Mathematical analysis based on ORNL-4086. Density distribution of the disposed cloud is assumed to be Gaussian. Only linear nuclear decay sequences are handeled. This code has been updated by ACRA II in which the effects of plume depletion resulting from wet and dry deposition are considered.

<9>
Killough, G.G., and P.S. Rohwer, INDOS-Conversational Computer Codes to Implement ICRP-10-10A Models for Estimation of Internal Radiation Dose to Man.

1974, March. ORNL-4916; 93 p. (Oak Ridge National Laboratory, Oak Ridge, TN)

The INDOS codes are special purpose conversational programs written to function in a time-sharing environment. They have been implemented on the ORNL PDP-10. These codes use ICRP-10-10A models to estimate the radiation dose to an organ of the body of reference man from accidental or routine releases. They compute dose and dose rate to an organ as functions of time. INDOS 1 computes dose rate and dose resulting from deposition of a specified quantity of a radionuclide in the organ. The user must specify all parameters related to radionuclide and organ, his choice of mathematical model (exponential or power function), and intake information as these items are requested by the code. The term 'intake' in messages output by INDOS 1 refers to quantity of material deposited in the organ, not total quantity introduced into the body. The output of INDOS 1 is a tabulation of time (days), dose rates (rems/day), and doses (rems.) INDOS 2 computes dose rate and dose resulting from deposition of a specified quantity of any one of several radionuclides in the organ, and the organ is the one which is regarded as critical for the radionuclide selected. Once the user designates his choice of radionuclide, INDOS 2 secures parameter values associated with the material and its critical organ from a direct-access file stored in the user's disk area. This stored data base is one feature which distinguishes the capabilities of INDOS 2 from those of INDOS 1. A second distinction is the meaning of the term 'intake.' For INDOS 2, this term refers to the quantity of material inhaled or ingested into the body (the user chooses the route of intake). INDOS 2 converts the intake into a proportionate deposition in the critical organ for the calculations. The output of INDOS 2 is capable of an optional typewritten graph of dose vs. time. Computationally, INDOS 3 is identical to INDOS 2, and these codes access the same direct-access files. The codes differ only in the form of output. INDOS 3 provides punched-card output which can serve as input to other codes such as graphical programs. Explanatory comments are included only at points where INDOS 3 differs substantially from INDOS 2. (FOH)

<10>
Sagendorf, J.F., A Program for Evaluating Atmospheric Dispersion from a Nuclear Power Station.

1974, May. NOAA TM ERL ARL-42; 12 p. (National Oceanic and Atmospheric Administration, Air Resources Laboratory, Idaho Falls, ID)

A computer code (SEP for Site Evaluation Program) is described. The program uses a joint frequency distribution of winds and stability classes to evaluate the atmospheric dispersion potential near a nuclear power station. The code includes models for short-term and long-term effluent releases. A description of the input parameters is included. The program is written in Fortran IV and uses 250 K memory on an IBM 360/75 computer. Plume rise may be calculated using formulas from Briggs, or it may be ignored. (CWM)

Only the ratio effluent concentration/source strength is calculated, not concentrations or doses. Building wake effects are included but not cloud deposition and depletion.

<11>
Martin, J.A., Jr., G.B. Nelson, and P.A. Cuny, AIREM Program Manual: A Computer Code for Calculating Doses, Population Doses, and Ground Depositions Due to Atmospheric Emissions of Radionuclides.

1974, May. EPA-520/1-74-004; 127 p. (U.S. Environmental Protection Agency, Office of Radiation Programs, Field Operations Division, Washington, DC 20460)

A computer code useful for the calculation of doses to the general population due to atmospheric emissions of radionuclides is presented and discussed. The code is written in Fortran IV, requires 188k storage, and runs in about 20 seconds on an IBM 370 system. A standard sector-averaged gaussian-diffusion equation is solved repeatedly for each radionuclide, wind sector, stability class and downwind distance. Radionuclide contributions to doses of up to four critical organs are summed and printed by sector and downwind distance. Population doses (person-rem) are also calculated. The code accounts for the following physical processes; cloud diffusion, ground and inversion-lid reflections, radionuclide decay by time of flight, first daughter-product buildup, ground deposition of particulates and halogens (independently), cloud depletion, in-plant holdup and decontamination factors, and sector-to-sector contributions to external gamma dose. The code is dose model independent in the sense that dose conversion factors, provided as input data, are used for calculations of dose that are proportional to radionuclide concentrations in the cloud, and dose conversion tables obtained from a model that considers the finite extent of the overhead cloud, also required as input data, are used for calculations of whole body dose due to external gamma emitters in the cloud. A set of dose tables obtained using one finite cloud model (EGAD) are provided in this manual. Up

<11> CONT.

to 20 radionuclides can be considered in a single problem. The program has been written to be used to evaluate routine releases, but the authors state that it can be used, with limitations, for accidental releases, especially those of several hours duration. Activity density on the ground is calculated and printed out, but no doses due to ground deposition are calculated. (Auth)(CWM)

<12>

Dunguid , J.D., and M. Reeves, Material Transport through Porous Media: A Finite - Element Galerkin Model.

1976, March. ORNL-4928; 210 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

A two-dimensional transient model for flow of a dissolved constituent through porous media has been developed. Mechanisms for advective transport, hydrodynamic dispersion, chemical adsorption, and radioactive decay are included in the mathematical formulation. Implementations of quadrilateral finite elements, bilinear spatial interpolation, and Gaussian elimination are used in the numerical formulation. The programming language FORTRAN IV is used exclusively in the computer implementation. A listing of the program is included. This material-trapsport model is completely compatible with the moisture-transport model (Reeves and Duguid, 1975) for predicting advective Darcy velocities for porous media which may be partly unsaturated. In addition to a description of the mathematical formulation, the numerical treatment and the computer implementation results of two computer simulations are included in this document. One is a comparison with a well-known analytical treatment (Lapidus and Amundson, 1952) and is intended as a partial validation. The other simulation, a seepage-pond problem, is a more realistic demonstration of the capabilities of the computer model. Complete listings of input and output are given in the appendices so that this simulation may be used for check-out purposes. This report, thus, is intended to be a comprehensive description of the material-transport computer model. Simplifying assumptions are made in the treatment of the chemical reactions of dissolved constitutents. Reacting species in the soil solution are independent of each other. Local equilibrium is assumed for reversible processes. Only one kind of reversible reaction is permitted, a reaction for which the rate of change is proportional to the total quantity of the constitutent presenting both the solid and liquid phase. (Auth)(DTS)

<13>

Fletcher, J.F., and W.L. Dotson, HERMES - A Digital Computer Code for Estimating Regional Radiological Effects from the Nuclear Power Industry.

1971, December. HEDL-TME-71-168; 759 p. (Hanford Engineering Development Laboratory, Richland, WA)

The HERMES model is designed to calculate radionuclide release and radiation dose occurring within a study area in a given year. While cumulative radionuclide release as such is not considered, the accumulated buildup and deposition of key radionuclides is "back-calculated" for a five-to-ten year period for use in dose calculations. The initial development of the HERMES model was based on a study of the region comprising the Upper Mississippi and Lower Missouri River basins, excluding the Chicago metropolitan area. To estimate the effects on dose in the study area caused by air transport of radionuclides from adjacent regions, an "air envelope" some 200 miles in width was included around the study area. Considerations in the study included the operation of 400,000 MW generating capacity (nuclear and fossil) at 185 separate sites, and reprocessing plants at 9 sites. The air transport code (ARTRAN) utilizes monthly climatological data applied to standard mathematical models for diffusion, deposition, and transport. The water transport code (WTRAN) calculates diffusion from emission points, concentrations in sediments and surface water at each location requested, and ground water transport. The code (DOSE) considers exposure to the whole body and six specific organs in (1) direct submersion in air or water (2) inhalation (3) external exposure from surface contamination (4) ingestion of food and water over numerous food-chain pathways. Life habits and dietary differences are also considered in the code (TPM). (POH)

Extensive modification of the HERMES model has occurred, and is being prepared for publication. The current model does not consider near-site conditions, however this publication is the primary documentation for many parameters that are used in later codes which are designed to calculate near-site doses. The HERMES model is large and complex in order to provide comprehensive estimates of population dose over regions of the size indicated.

<14>

Trubey, D.K., and S.V. Kaye, The EXREM III Computer Code for Estimating External Radiation Doses to Populations from Environmental Releases.

1973, December. ORNL-TM-4322; 82 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

EXREM III is a computer code to estimate the dose equivalent rate and the total dose equivalent from beta, positron, electron, and gamma radiation resulting from submersion in contaminated water, submersion in contaminated air, and exposure to a contaminated surface. There can be more than one environmental release, and exposure can begin at any time after the first release. EXREM III considers contributions from environmental releases and from nuclide decay chains. For a particular problem the user may choose to calculate either the dose rates, or the total doses, or both for any of the three modes of exposure. A separate solution array is printed for each mode of exposure. EXREM III is a revised version of EXREM II which was available earlier. The principal revisions include treatment of positron and electron radiations, selection of nuclear data from a data base, variable dimensioning of large data arrays, and free field input. The code is available from the Radiation Shielding Information Center. (Auth)

<15>

Baker, D.A., G.R. Hoenes, and J.K. Soldat, Food - An Interactive Code to Calculate Internal Radiation Doses from Contaminated Food Products.

1976, February. BNWL-SA-5523; 5 p. (Battelle Pacific Northwest Laboratories, Richland, WA 99352)

The computer code Food calculates internal does due to ingestion. Fourteen types of

<15> CONT.
produce, grains and animal products are considered. Vegetation is contaminated by air or irrigation water via direct deposition on leaves or uptake from soil through root systems. Doses may be calculated for total body plus six internal organs for 19 radionuclides. Food is compatible for usage along with ARRRG. The term "irrigation" means sprinkler irrigation. The translocation of externally deposited radionuclides to the edible parts of plants is assumed independent of the radionuclide. Milk transfer factors for elements lacking in radionuclide data were taken to be one-half those in report UCRL-50163. The assumption is made that plants obtain all their carbon from airborne CO2 and animals obtain all their carbon through ingestion of plants. Because the transfer of carbon from water to air or soil is unknown, food conservatively assumes that plants obtain all their carbon from the irrigation water. Thus carbon concentrations in plants could be high by an order of magnitude. (DLS)

<16>
Watts, J.R., Modeling of Radiation Doses from Chronic Aqueous Releases.

1976, July. DP-MS-75-126; 13 p. (Savannah River Laboratory, Aiken, SC 29801)

A general model and corresponding computer code, called Cardocc, were developed to calculate personnel dose estimates from chronic releases via aqueous pathways. Cardocc uses the internal dose portion of the ARRRG code, increases its generality and scope, and simplifies the input requirements. Potential internal dose pathways are consumption of water, fish, crustacean, and mollusk. Dose prediction from consumption of fish, crustacean, or mollusk is based on the calculated radionulcide content of the water and applicable bioaccumulation factor. 70-year dose commitments are calculated for whole body, bone, lower large intestine of the gastrointesinal tract, and six internal organs. In addition, the code identifies the largest dose contributor and the dose percentages for each organ-radionuclide combination in the source term. The 1974 radionuclide release data from the Savannah River Plant were used to evaluate the dose models. The dose predicted from the model was compared to the dose calculated from radiometric analysis of water and fish samples. The whole body dose from water consumption was 0.45 mrem calculated from monitoring data and 0.61 mrem predicted from the model. Tritium contributed 99% of this dose. The whole body dose from fish consumption was 0.20 mrem calculated from monitoring data and 0.14 mrem from the model. Cesium 134,137 was the principal contributor to the 70-year whole body dose from fish consumption. (auth)(DLS)

<17>
Gogolak, C.V., Comparison of Measured and Calculated Radiation Exposure from a Boiling Water Reactor Plume.

1973, September. HASL-277; 25 p. (Health and Safety Laboratory, United States Atomic Energy Commission, New York, NY 10014)

Field measurements of exposure rates for noble gases were compared with model predictions to determine the sensitivity to approximations for build-up of gamma radiation, plume rise and cloud size. The model used was based on May and Stuart (1970), i.e., a crosswind averaged Gaussian plume. The vertical dispersion coefficient is that due to Watson and Gammertsfelder. Holland's formula with Moses' correction for stack diameter is used to calculate plume rise. Growth and decay of daughter radionuclides is considered. Deposition and inhalation are not taken into account. For the buildup factor, Capo's formula is used for gamma energies greater than 0.5 mev, and Berger's formula is used for gamma energies less than 0.5 mev. Exposure rate is averaged over 4 atmospheric stability classes, 16 wind directions and 6 wind speed ranges weighted by their frequency of occurrence. Gamma ray measurements were made for two months near a boiling water reactor with ionization chambers and thermoluminescence dosimeters. Measured and calculated exposure rates agree within a factor of 2. The model predictions were compared with those obtained with EGAD, the Gammerstfelder model and calculations made as part of the safety analysis for the plant. The linear buildup factor approximation is a sensitive function of meteorology and gamma ray energy. Calculations of dose rates for the different models differed by more than 100% at large distances from the stack under unstable conditions. Models treating off-sector exposures separately match the data more accurately than those that do not. (DLS)

<18>
Radiation Shielding Information Center, RISC Computer Code Calculations, ACRA-II Kernal Integration Code Estimation of Radiation Doses Caused by a Hypothetical Reactor Accident.

1974, July. CCC-213; 75 p. (Oak Ridge National Laboratory, Oak Ridge, TN)

The purpose of ACRA-II is to expand the capability of ACRA-I to include the effects of fallout, washout, and ground contamination. The built-in containment model of ACRA-I was eliminated and the emission of nuclides to the atmosphere is described by a step function of time. This decision to separate the containment model calculation from the meteorological part of the calculations was prompted by the fact that the containment models vary depending on the radionuclides, the leakage pathways, and the filter mechanisms involved. The program computes external dose, internal dose, and ground contamination as a function of location and points in time. The three dimensional normal distribution is used for dispersion, and the cloud concentration accounts for washout by rain, fallout, and radioactive decay. Fallout is described by Chamberlain's model. (auth)(DLS)

<19>
Moore, R.E., AIRDOS - A Computer Code for Estimating Population and Individual Doses Resulting from Atmospheric Releases of Radionuclides from Nuclear Facilities.

1975, January. ORNL-TM-4687; 80 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

AIRDOS, a Fortran IV computer code, was written to estimate population and individual doses resulting from the continuous simultaneous atmospheric release of as many as 36 radionuclides from a nuclear facility. This report presents details of the code and complete instructions for its use. Five pathways to man are considered: (1) inhalation of air containing radionuclide gases or particulates, (2) immersion in

<19>

<19> CONT.

contaminated air, (3) exposure to surfaces contaminated by radioactive fallout, (4) ingestion of food produced on contaminated ground surfaces, and (5) immersion in contaminated water, as by swimming. Dose and dose commitments are estimated for each pathway and the following eleven reference organs: whole body, GI tract, bone, thyroid, lungs, muscle, kidneys, liver, spleen, testes, and ovaries. The environmental model in AIRDOS consists of a 20 x 20 square grid with the nuclear facility located at the center. The size of each grid is specified in the input data. Human population, numbers of beef and dairy cattle, and identification as to whether an area is predominately used for production of vegetable crops or is a water area are specified for each of the 400 grids. Population doses are summarized in output tables in every possible manner--by nuclides, pathways, and organs. The highest individual dose received in the area and its location are printed in the output. AIRDOS is partially validated against data available in the literature. (auth)(DLS)

<20>

Turner, W.D., S.V. Kaye, and P.S. Rohwer, EXREM and INREM Computer Codes for Estimating Radiation Doses to Populations from Construction of a Sea-Level Canal with Nuclear Explosive.

1968, September 16. K-1752; 193 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

EXREM is a computer code to estimate the dose equivalent rate and the total dose equivalent from both beta and gamma radiation resulting from submersion in contaminated water, submersion in contaminated air, and exposure to a contaminated surface. There can be more than one environmental release, and exposure can begin at any time after the first release. INREM is a computer code to estimate the cumulative dose equivalent to body organs resulting from a continuous intake. The organ parameters are dependent on the age of the individual, and the intake is a function of post-detonation time and the individual's age. Although the codes were specifically written to estimate doses to populations from construction of a sea-level canal, they may be useful in other dose-estimation studies involving releases of radioactive materials to the environment. (auth)

This is the first documentation of these codes which have been revised and updated via EXREM II and EXREM III and INREM II

<21>

Engelmann, R.J., and W.E. Davis, Low-Level Isentropic Trajectories and the MIDAS Computer Program for the Montgomery Stream Function.

1968, April. BNWL-441; 31 p. (Battelle, Pacific Northwest Laboratories, Richland, WA)

The optimum trajectory for releases of materials at low elevations in the atmosphere is constructed with isentropic charts. Isentropic analysis reveals vertical motions of the material and predicts where the material will appear in the air near the ground. This paper presents an example of such a trajectory, provides a thorough error analysis of isentropic data, and gives the details of a digital analog simulation computer program for obtaining isentropic data. These trajectories can be calculated using teletyped weather data, and precipitation scavenging may be included if desired. The solutions are found using the Modified Integral Digital Analog Simulator (MIDAS) technique. (Auth) (CWM)

This program calculates long distance three-dimensional trajectories for particles or gases in the air of any type, not just radioactive ones. Radioactive decay corrections are not included in the code.

<22>

Bell, G.D., and J. Houghton, Risk Evaluation for Stack Releases.

1969. AHSB (S) R 173; 18 p. (United Kingdom Atomic Energy Authority, Risley, England)

A method of calculating the risk to the population distribution around a reactor site is presented for both ground level and elevated releases. The reactor is assumed to satisfy a control curve of the Farmer type for accidental releases of radioactivity, and the dose-risk relationship is assumed to be a simple power law. Results are presented in which iodine is imagined to be released, on a number of typical sites. The resulting frequency distributions of thyroid cancer have been calculated and are given in the form of frequency-casualty curves (f-N curves). The program 'STRAP II' used in these calculations is described.

The program depends on having input the number of rem per curie air concentration for a given radionuclide and the number of casualties of a given type per rem for that radionuclide. (CWM)

<23>

Cooper, R.E., WRED, A Siting Code to Estimate Dose Probability Distribution from Measured Meteorology Data.

1969. Health Physics, 16, 735-738. (Savannah River Laboratory, E. I. duPont de Nemours and Co. Aiken, SC 29801)

WRED is a reactor siting code designed to estimate probability distributions of whole body gamma and thyroid doses using measured meteorological parameters as input data. Distribtuion analyses are performed as a function of downwind distance, activity release height, and wind direction. This statistical approach for gamma dose estimates, employing thousands of sets of meteorology data, is made economically feasible by using two-dimensional tables of space distribution-attenuation integrals and interpolation techniques. Measured meteorological parameters are converted to atmospheric dispersion parameters according to techniques developed at Brookhaven National Laboratory. Space distribution-attenuation integrals are computed by techniques developed at the Savannah River Laboratory. When utilizing the full capacity of the code, approximately 2000 sets of data (involving 16,000 each gamma and thyroid dose calculations from raw meteorological data) are processed per minute on the IBM System 360/65. Concurrently, meteorological statistics are compiled, and all data are summarized for output. Provision is made for including compiled statistics from previous data processing to update the distribution analyses for each job. Code options allow minimum data input (wind speed, direction, temperature) or full input including measured standard deviations of the horizontal and vertical wind directions. (Auth)

<23> CONT.
This code includes the dosimetric calculations computed by RADOS for gamma doses.

<24>
Clarke, R.H., A Multi-Compartment Model for Studying the Dose-Intake Relationship for Radioactive Isotopes.

1972, April. RD/B/N 2291; 38 p. (Central Electricity Generating Board, Berkeley Nuclear Laboratories, United Kingdom)

A model (CEDRIC) has been developed in which the various organs of the body are represented by single or multiple compartments, coupled to provide a simplified metabolism. The method of entry of radioactive material may be by inhalation, ingestion, through contaminated skin or a wound, using standard models for the respiratory and gastrointestinal tracts. The distribution of a parent nuclide and all its daughter products throughout the compartments is followed as a function of time by the solution of the set of first order differential equations which represent transport within the body. Either acute or chronic exposures are treated and facilities exist for varying metabolic parameters with time. Special attention has been given to dealing with multi-exponential retention functions, both for individual organs and for the whole body, and methods of treating power function retention data are given. Examples of the use of the compartment model under these conditions have been included and the results compared with published results given by the International Commission on Radiological Protection ICRP 10 (1968) and by Voilleque (1971), who has published details of a simplified compartment model which he developed (Voilleque, 1970). The program, which has been designated CEDRIC (Computer Evaluation of Doses from Radioactive Isotopes by Compartments) is written in Fortran IV. Typical running times vary from a few seconds to approaching 60 seconds for a complex case. A complete user's guide to the program has been provided by Clarke and Beynon (1972) in which running details are given including input/output specifications. (Auth) (FDH)

The compartment model CEDRIC is a FORTRAN IV which improves upon the SAURON (1970) code which was based upton the metabolic retention data and lung models adopted by ICRP II (1959). These models have been developed as part of the WEERIE program.

<25>
Cruz, F.D., "Esdora" Computer Program for an Estimate of the Radiation Dose in the Evaluation of Risks for Nuclear Power Plants and Radioactive Facilities.

1971, March. SS/01/71; ORNL-tr-2628; 4 p. (Junta De Energia Nuclear, Spain)

The program ES.DO.RA. (EStimacion de DOsis RAdiologicas (estimate of radiation doses)) estimates the damage produced by the continuous or instantaneous escape of fission products into the atmosphere. It was programmed in FORTRAN IV for the IBM 7090 computer and is divided into 4 subroutines: ESD 1. Generates an inventory of fission products according to the operating process (sic., burnup?) of the fuel of the facility. ESD 2. Determines the quantity of fission products emitted to the atmosphere through the stack or the containment building, taking into account the technical safety measures employed. ESD 3. Calculates the integrated doses of the radioactive cloud, as well as the ground-level contamination due to fallout of the fission products, with consideration of meteorological conditions. ESD 4. Estimates the external beta and gamma doses by submersion in the radioactive cloud as well as the internal doses received by the various organs of the human body by inhalation. This report includes the mathematical models used, the simplifying assumptions adopted, the data used, lists and flowsheets of the specific subroutines, as well as the necessary instructions for their use. The program was developed to evaluate the risks resulting from an explosive type of accident in a nuclear facility, but in special cases it can be used with routine releases. (Auth) (CWM)

<26>
Eckert, R.J., A Fortran Program for the Calculation of the Potential Hazards from an Accidental Fission-Product Release - M0142.

1964, July. WAPD-TM-348; 22 p. (Bettis Atomic Power Laboratory, Pittsburgh, PA)

A FORTRAN program (M0142) has been written to perform a detailed analysis of the space and time-dependent radiation doses resulting from a potential fission-product release incident from a nuclear reactor. The program permits the calculation of the hazards from any radioactive source of known concentration and isotopic distribution. The nuclides which form the radioactive cloud are specified as program input. For each nuclide, an initial concentration, release rate, radioactive dacay constant, and average gamma and beta emission energy must be supplied. In addition, the exposure period, the distance from the source to the receptor point, and the appropriate meteorological conditions must be specified. For each nuclide in the calculation, the program computes the external gamma and beta dose and the inhalation dose to several critical organs. The total in each category is obtained by a straightforward summation of the individual nuclide doses. Atmospheric dispersion is calculated using Sutton's equation and the resulting air concentration is corrected for decay before the dose is calculated. The program allows for a time-dependent release rate to be used if desired. (Auth) (CWM)

<27>
Steyn, J.J., and Y.S. Kim, Users manual for Code EXGAM: A code to Calculate Gamma Photon Dose from an Airborne Radioactive Release.

1966, November. NUS-329; >25 p. (NUS Corporation, 1730 M Street, N.W., Washington, DC 20036)

A FORTRAN IV code - EXGAM - for use on a CDC-3600 digital computer has been written to predict the integrated photon dose at a receptor exposed to an airborne release of radioactivity. The code assumes the source spatial distribution to be Gaussian. Input options allow the user to choose source decay mode, cloud thickness, cloud wind speed, cloud stability class, cloud drift direction, air attenuation and build-up coefficients and source photon energies. The code user's requirements and FORTRAN listing are given, along with a sample input and output. The code has been optimized with respect to computer execution time. (Auth)

This code was designed to be used as part of the space nuclear propulsion program, not for releases from nuclear power facilities.

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<28>
Charak, I., Atmospheric Dispersion of Fission Products Including the Effects of Building Dilution and Radioactive Decay.

1967, May. Nuclear Applications, 3, 283-286. (Argonne National Laboratory, Argonne, IL 60439)

Sutton's diffusiton equation, which is commonly used for the computation of atmospheric dispersion of fission products, is modified to account for the effect of dilution of the fission products due to mixing of the contaminated air inside the reactor building with air prior to exhaust. The effect of radioactive decay is also considered. The resulting integral equation is solved for a specific example using a Romberg integration subroutine, and the application of the equation to a washout calculation is described. The solution is done by a Fortran program on a CDC-160A computer. (Auth) (CWM)

As written, this code is designed for accidental rather than routine reactor releases.

<29>
Lanes, S.J., A Procedure to Determine the Relative Extent of Reactor Fission Product Releases Under Various Meterological Conditions.

1964, July. HASL-148; TID-4500; 45 p. (Health and Safety Laboratory, New York Operations Office, New York, NY)

Several versions of Sutton's equations expressing the atmospheric concentrations, doses, etc. due to a fission product release have been programmed for solution by an IBM 1620 computer, as an aid in evaluating the effects of postulated reactor accidents. In addition, each of these equations has been solved over a wide range of meteorological parameters, and the results graphed, providing an illustration of the effects of changes in one parameter or a combination of them on the solutions. A procedure for the application of these programs and graphs to the review and evaluation of reactor safety problems is presented. The code handles both instantaneous and continuous point sources. Cloud depletion by both wet and dry processes are included using Chamberlain's procedure. The only dose calculated is the inhalation dose to the thyroid. (Auth) (CWM)

This early report should be read with care as there is some question on the use of the terms rainout and dose, and on the derivation of the thyroid dose equation.

<30>
Lange, R., ADPIC: A Three-Dimensional Computer Code for the Study of Pollutant Dispersal and Deposition Under Complex Conditions.

1973, October. UCRL-51462; TID-4500; 60 p. (Lawrence Livermore Laboratory, Livermore, CA)

Atmospheric Diffusion Particle-In-Cell (ADPIC) is a three-dimensional, Cartesian, particle-diffusion code, capable of calculating the time-dependent dispersion of inert or radioactive air pollutants under many conditions, including stratified shear flow, calms, topography, and wet and dry deposition, for individual or multiple instantaneous or continuous sources and space- and time-variable diffusion parameters. The code solves the three-dimensional diffusion-advection equation by the pseudo-velocity technique for a given mass-consistent advection field. The method is based on the particle-in-cell technique with the pollutant concentration represented by Lagrangian particles in an Eulerian grid mesh. ADPIC has been verified for a number of selected advection-diffusion problems for which analytic solutions are available, and has been found to give results to within plus or minus 5% of the analytic solutions. ADPIC can be used for both routine and accidental atmospheric releases. The program requires most of the core of a CDL 7600 to run. It may be scaled to handle different size problems, 20 m to 20 km horizontal cell size. The output gives cloud center or surface air concentrations and cumulative surface deposition. The ADPIC code should also be applicable to water dispersion studies. (Auth) (CWM)

<31>
Hubschmann, W., and D. Nagel, ISOLA II - A Fortran IV Code for the Calculation of the Long-Term Alpha- and Gamma-Dose Distributions in the Vicinity of Nuclear Installations.

1976, December. KFK 2210; 14 p. (Gesellschaft fur Kernforschung m.b.H., Karlsruhe, Germany)

The computer code ISOLA is used to calculate the annual radiation doses caused by alpha- and beta-active off-gases in the environment of the Karlsruhe Nuclear Research Center. In the revised version, ISOLA II, the double Gaussian distribution model is strictly observed. As a consequence, the contribution of activity from neighbour sectors is taken into account. Up to 15 emitters may be coped with simultaneously. The emission rates are considered to be constant during the given time interval. Optionally, either the isodoses chart of a specified area (for instance a square 20 by 20 km) or a list of doses calculated at up to 2000 locations (for instance the living areas) in the environment may be set up. Input and output are shown for a specific case. (Auth)

<32>
Binford, F.T., J. Barish, and Kam, F.B.K., Estimation of Radiation Doses Following a Reactor Accident.

1967. ORNL-4086; 29 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

Based upon the "Gaussian Plume" formula, a model has been developed which permits calculation of the downwind radiation doses to be expected following a reactor accident. Although the calculation presented here is concerned only with the thyroid dose due to iodines and the whole body dose due to iodines and noble gases, it can easily be adapted to handle other radioisotopes. the basic assumptions include instantaneous release of a given quantity of radioactive material into a reactor building followed by emission at a constant (volume) rate from a point source. Radioactive decay both within the building and during passage downwind are accounted for as are the growth and decay of daughter products and the effects of physical separation such as filtration. A Fortran IV computer program "PLUME" has been written for the purpose of solving the equations developed. The code calculates dose due to inhalation and from submersion in the gamma emitting cloud. (Auth) (DLS)

<33>
Houston, J.R., D.L. Strenge, and E.C. Watson,

<33> CONT.
DACRIN - A Computer Program for Calculating Organ Dose from Acute or Chronic Radionuclide Inhalation.

1974, December. BNWL-B-389; UC-41 (Battelle, Pacific Northwest Laboratory, Richland, WA 99352)

The computer program, DACRIN, permits rapid and consistent estimates of the effective radiation dose to the human respiratory tract and other organs resulting from the inhalation of radioactive aerosols. The program is an outgrowth of the development of a mathematical model for the organ dose following the basic precepts of the ICRP Task Group on Lung Dynamics, and a a simple exponential model for retention by an organ of interest. Mathematical models describing atmospheric dispersion have been included for the purpose of evaluating doses resulting from either accidental or chronic atmospheric releases of radionuclides. The program will calculate the effective radiation dose to any of 18 organs and tissues from inhalation of any one or combination of radionuclides considered by the ICRP. A maximum of 10 organs may be selected for each case. Organ doses from inhalation can be calculated by specifying either the quantity of a radionuclide inhaled or the quantity released to the atmosphere. In the latter case, the duration of release, the release height, wind speed, atmospheric dispersion parameters and downwind distance at which the dose is to be calculated must also be specified. As many as 10 distances may be specified for each use. Output of the code consists of the effective radiation dose to the selected organs at selected time intervals, for each radionuclide inhaled as indicated by the input. (FOH)

This edition of DACRIN has been updated by a supplementary document (BNWL-B-389 SUPP, 1975) to include dose to the GI tract via inhalation. Information pertaining to Pasquill Stability Categories are given for distances out to 100,000 m, but the atmospheric dispersion model does not include the effects of wet or dry deposition on plume depletion. Sutton and Hanford atmospheric dispersion information is also stored in the data base.

<34>
Strenge, D.L., DACRIN - A Computer Program for Calculating Organ Dose from Acute or Chronic Radionuclide Inhalation: Modification for Gastrointestinal Tract Dose.

1975, February. BNWL-B-389 SUPP; 105 p. (Battelle, Pacific Northwest Laboratories, Richland, WA 99352)

The computer program DACRIN uses the lung model proposed by the International Commission on Radiological Protection (ICRP) Task Group on Lung Dynamics to calculate the effective dose to the respiratory tract and other organs following either acute or chronic inhalation of radionuclides. The program has now been extended to calculate the dose to the four gastrointestinal tract compartments; stomach (S), small intestine (SI), upper large intestine (ULI) and lower large intestine (LLI). The ICRP G.I. tract model is used to describe movement of material within the G.I. tract compartments with input to the G.I. model described by the lung model of DACRIN. (Auth) (FOH)

This is an expanded version of BNWL-B-389 (1974). It incorporates the ICRP II GI tract dose assessment model.

<35>
Strenge, D.L., E.C. Watson, and J.R. Houston, SUBDOSA - A Computer Program for Calculating External Doses from Accidental Atmospheric Releases of Radionuclides.

1975, June. BNWL-B-351; 169 p. (Battelle, Pacific Northwest Laboratories, Richland, WA 99352)

A computer program, SUBDOSA, has been developed for calculating external gamma and beta doses to individuals from the accidental release of radionuclides to the atmosphere. Characteristics of SUBDOSA are: (1) Doses from both gamma and beta radiation are calculated as a function of depth in tissue, summed and reported as skin, eye, gonadal and total body dose. (2) Doses are calculated for releases within each of several release time intervals. Up to six time intervals can be allowed and separate nuclide inventories and atmospheric dispersion conditions are considered for each time interval. (3) Radioactive decay is considered during the release and/or transit using a chain decay scheme with branching to account for transitions to and from isomeric states. (4) The dose from gamma radiation is calculated using a numerical integration technique to account for the finite size of the plume. (5) The program computes and lists the normalized air concentrations at ground level as a function of distance from the point of release. The organs of reference most frequently considered in external dose calculations are the skin, lens of the eye and total body. Corresponding tissue depths for these organs used in SUBDOSA are 0.007, 0.1 and 5 cm respectively. The male gonads are less often considered in dose calculations but when such calculations are made with SUBDOSA, the tissue depth is 1 cm. (Auth) (FOH)

SUBDOSA does not consider the dose resulting from radionuclides deposited in the body and its organs via inhalation.

<36>
Otter, J.M., and P.A. Conners, Description of the COMRADEX-III Code.

1975, June. TI-001-130-053; 22 p. (Atomics International Division, Rockwell International, Canoga Park, CA 91304)

The COMRADEX-III (COntainment and Meteorology of environmental RADiation EXposure) Code is described briefly, input instructions are given, and test cases and Fortran coding are listed. Changes from the previous version (COMRADEX-II) include variable weather modeling for external doses, a direct site meteorology option, and an improved direct dose formulation. The code was developed for the calculation of radiological doses from hypothetical power reactor accidents. It permits the user to analyze as many as four levels of containment with time varying leakage and cleanup rates in each level. Filtration in each containment region may be by isotopic class (noble gas, solids, or halogens). Shielding may be introduced between the various containment shells. The environmental doses calculated include the direct gamma dose from the containment building, the internal doses for up to 12 body organs (thyroid, bone, lung, etc.) due to inhaling the airborne radioactivity, and the external gamma and beta doses from the cloud. Leakage and cleanup rates, and initial inventory are input data. A direct numerical technique, designed especially for

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quasi-linear differential equations, is used to solve the time dependent isotope population equations. Meterology models may be that of Pasquill or site meteorology. Factors are given in this document for computing internal doses with the ICRP-2 model. The cloud gamma dose is obtained by integrating over an anisotropic Gaussian cloud for Pasquill meteorology or from a semi-infinite cloud for direct site meterology. The external beta dose calculation assumes a semi-infinite radioactive cloud. (Auth) (CWM)

<37>
Kenfield, G.P., W.R. Lahs, W.B. Sayer, R.M. MacAdams, and N.A. Harris, Curie-Dose-Thunderhead A Digital Computer Program for External and Internal Radiation Dose Calculations.

1965, June. NAA-SR-8884; 39 p. (Atomics International, a Division of North American Aviation, Inc.)

The CURIE-DOSE-THUNDERHEAD program is a combination of three basic computer programs employed in succession. The first, CURIE, calculates the fission product buildup and decay as a function of reactor operating time, operating power, and decay time. The second, DOSE, uses the fission product inventory obtained from CURIE to calculate the dose to 14 internal body organs due to inhalation of fission products released to the atmosphere. The third, THUNDERHEAD, also uses fission product activity data from CURIE to calculate the external cloud gamma exposure from the released fission products. Cloud diffusion is calculated from Sutton's equation or the Convair modification. In calculating the internal dose it is assumed 62.5 percent of the inhaled isotopes are ultimately swallowed and pass through the entire G.I. tract. An accidental release situation is considered. (Auth) (CWM)

<38>
Soldat, J.K., N.M. Robinson, and D.A. Baker, Models and Computer Codes for Evaluating Environmental Radiation Doses.

1974, February. BNWL-1754; 160 p. (Battelle, Pacific Northwest Laboratories, Richland, WA 99352)

Since HERMES is considered too complex and unsuitable for the evaluation of the radiological consequences of point-source releases, a simplified model for calculation of radiation doses from radioactive effluents was developed and programmed into a conversational language, providing the fast turnaround time required. The new model is divided into four independent parts, each written as a separate program: (1) ARRRG: calculates individual and population doses from liquid effluents (2) CRITR: calculates internal radiation doses to four common classes of aquatic organisms and to organisms which consume them (3) FOOD: calculates doses from consumption of food crops and animal products produced on irrigated farms (4) GRONK: calculates doses from gaseous effluents to individuals, and to the total population within 50 miles. It includes an option for building wake effects but does not calculate plume depletion. The model can be used to calculate radiation doses to the total body and selected organs of individuals and population groups, and to organisms other than man. It includes all air and liquid exposure pathways thought to be significant and for which a reasonable amount of supporting data is available. Internal doses to man are based on a 1-year radionuclide intake, assuming no prior accumulation in the body. The radionuclide content of ingested food is assumed to be at equilibrium with the environment. This paper discusses the models in detail and describes the programs ARRRG, CRITR and GRONK; the program FOOD is still being developed and is not included in this report. The question-and-answer format of these programs allows then to be used by nonprogrammers. Although the programs, which are in the BASIC language, were originally intended specifically for nuclear reactors, they are applicable to any nuclear facility which releases radioactive effluents to air or water. (FOH)

This is the document upon which many of the models recommended by the US NRC in Reg. Guide 1.109 is based. The Dose Conversion Factors are for the calculation of annual doses, not dose commitments. The data base for much of the material discussed has been taken directly from the original HERMES code.

<39>
Luna, R.E., and H.W. Church, DIFOUT, Model for Computation of Aerosol Transport and Diffusion in the Atmosphere.

1969, January. SC-RR-68-555; 47 p. (Sandia Laboratories, Albuquerque, NM)

In conjunction with Roller Coaster, a joint United States-United Kingdom project conducted at the Nevada Test Site in 1963, a computer-programmed model for prediction of the atmospheric transport and diffusion of aerosols was developed. The program was a melding of a U.S. code describing gravitational fallout of a log-normal site distribution of particles in a vertically varying wind field and a U.K. code describing the dispersion of a particle cloud by atmospheric turbulence. In the program, the aerosol cloud is approximated by a series of cylindrical layers in the vertical direction characterized by a source strength, lateral diameter and thickness, and a distribution of aerosol mass with particle diameter. In turn, each layer is represented by several superposed line source elements of equisize particles to conform to the aerosol mass-diameter distribution for the layer. Each element descends at its gravitational fall rate and is transported by its mean vector wind while expanding by the action of turbulent diffusion. Deposition and/or airborne dosage is calculated at ground level on a specified circular grid of ranges and azimuths whose origin is at the initial position of the cloud. Details of the development of the unique line source equations used in the model are given, indicating the analytical method used to account for linearly changing transport direction over the length of the source. In addition, the expressions used to calculate turbulent spread, deposition, particle fall rates and transport vectors are discussed in terms of appropriate experiment and Gaussian theory. Various input and output options are described which permit great flexibility in program use, either in prediction of the dispersal from a given cloud or in the determination of the input necessary to produce a given result. Detailed instructions are given for preparing the input data for the program. A sample problem is used to illustrate the usefulness of the program in modeling a fairly arbitrary selection of inputs. Both the input preparation and output interpretation for the sample problem are

<39> CONT.
shown in some detail. The program may be used for both continuous and instantaneous releases. (Auth) (CWM)

<40>
Tveten, U., Areas Affected by Ground Deposition of Cs 137 and Description of the Computer Code ARCON.

1975. IAEA-SM-188/35; Part of Proceedings of a Symposium on Siting of Nuclear Facilities held in Vienna, Austria December 9-13, 1974. p. 497-515 (Institute for Atomenergi, Kjeller, Norway)

The paper deals with the combined probability analysis and parametric study of Cs 137 deposition after a postulated nuclear reactor accident. The study is performed for a 3000 MW(th) reactor at a site typical of the Oslo Fjord area. No efforts have been made to determine definite accident sequences or amounts of release. It is simply postulated that an accident has taken place that will lead to release of a large amount of radioactivity to the atmosphere during a relatively short time period: on the order of one hour. The size of the area affected by a certain release of radioactive cesium depends upon the meteorological conditions during the release time period. Local weather data have been used to determine the probability of a certain magnitude of the area. The results are given in the form of conditional, cumulative probabilities. Conditional because they give the probability of a certain magnitude of the consequence if the release has taken place and the wind is blowing in the direction for which the weather data are valid. Cumulative because, e.g., the 5% probability means that the weather conditions may give these or worse consequences with a 5% probability or, expressed differently, they will give less severe consequences with a 95% probability. Since the calculations do not apply to a specific accident sequence, several pertinent release characteristics cannot be assigned a definite value. The most important are: amount of Cs 137 releases; initial heat at release point; heat generated during plume transport due to decay of radioactive materials; and deposition velocity. THe last item is only partially a release characteristic. Besides depending upon particle size and chemical conditions, the deposition velocity depends upon characteristics of the land area over which deposition takes place. A parametric study is performed by varying each of these characteristics over a certain range while keeping the others fixed at what it is reasonable to assume are typical values under extreme accident conditions. A computer code ARCON has been developed for this analysis, adaptable to a large range of calculations connected to atmospheric releases during accident conditions as well as normal operation, e.g. calculation of reference levels for milk consumption, health effects on specified populations, and the area within any kind of concentration limit or dose limit (except external gamma dose). The program may be modified for longer release times, but not easily. While Cs 137 is the only isotope treated in this paper, other isotopes may be treated in a similar manner. (Auth) (CWM)

<41>
Hendrickson, M.M., EXDOSE, A Computer Program for Calculating the External Gamma Dose from Airborne Fission Products.

1968, September. BNWL-811; 113+ p. (Battelle, Pacific Northwest Laboratories, Richland, WA)

The EXDOSE program calculates the external gamma radiation dose to the total body from a half-infinite cloud of fission products. It is designed primarily for estimating accident consequences. The code calculates a fission product inventory according to specified input parameters using an adaptation of the program RIBD. Any fraction of these fission products may then be released to the atmosphere at a desired rate. The subsequent downwind dispersion and resulting dose to individuals is then calculated. The dispersion of the plume is based on the bivariate normal mode and Simpson and Fuquay's equation or Sutton's equation. Cloud depletion by ground deposition is an available option. Correction for radioactive decay and daughter product growth is included. (Auth) (CWM)

The Simpson-Fuquay stability constants are based on experiments done at Hanford and they should be examined carefully for their applicability to other areas.

<42>
Bocne, J.S., B.L. Parks, and B.G. Kniazewycz, RUBY, A Program for the Calculation of Activity Releases and Potential Doses from a Pressurized Water Reactor Plant.

1975, May. Not Given (Carolina Power and Light Company, Raleigh, NC)

The RUBY program is designed for the calculation of radiation releases and doses resulting from the operation of a large pressurized water reactor. The approach used in RUBY is similar to an analog simulation of a real system. Each major component or volume in the plant containing radioactive material is represented by a subroutine which keeps track of the production, transfer, decay, and removal of radioactivity in that volume. During the course of the analysis, activity is transferred from subroutine to subroutine in the program as it would be transferred from place to place in the plant. The necessary rates of production, transfer, cleanup, leakage, and release are read in as input to the program. Subroutines are also included which calculate the off-site radiation exposures at various distances, for individual nuclides and sums of nuclides. The program contains a library of physical data for the fifty-five nuclides of most interest in licensing calculations, and other nuclides can be added or substituted. These calculations can be performed for any or all time periods during the lifetime of the plant. Both external and thyroid internal doses can be calculated on an individual and a population basis. Values for atmospheric dilution at desired downwind distances and dose conversion factors for the thyroid doses are input by the user. (Auth) (CWM)

One nebulus feature of RUBY is its scrap-paper ability. Complete printout of all available information results in approximately 175 pages of output per period. (sic)

<43>
Culkowski, W.M., and M.R. Patterson, A Comprehensive Atmospheric Transport and Diffusion Model.

1976, April. ORNL/NSF/EATC-17; 117 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

A comprehensive version of the Atmospheric Transport Model (ATM) is described that

<43> CONT.
includes the effect of aerodynamic roughness on dispersion constants, clarifies the roles of the terminal velocity and deposition velocity, incorporates a tilting plume for heavy particulates, and includes an episodic calculation of exposure maxima. This model also limits the maximum value of the dispersion constants in order to retain the emitted material in the planetary boundary later. The structure of the program has been modularized in order to clarify the flow of calculation and allow more flexibility. Values for atmospheric concentration as well as both wetfall and dryfall deposition are calculated. The model is applied to the vicinity of three power plants, and correlations between model predictions and observed values are presented. The dispersion calculation is based on the Gaussian plume model, corrected by the above considerations. The model allows the use of point sources, line sources, area sources, and windblown resuspension sources. (Auth)(CWM)

This program does not include corrections for radioactive decay or daughter product formation so care must be exercised if it is to be used for atmospheric transport of radionuclides. (CWM)

<44>
MacDonald, H.F., Assessment of Environmental Hazards Following Nuclear Reactor Accidents.

1971. IAEA-SM-148/38; CONF-710705; Part of Proceedings of the International Symposium on Rapid Methods for Measuring Radioactivity in the Environment, held in Munich, Germany, July 5-9, 1971, p. 43-54 (Central Electricity Generating Board, Berkeley Nuclear Laboratories, Berkeley, Glos., United Kingdon)

Theoretical models have been developed describing the build-up of fission products in irradiated nuclear fuel and the behaviour of these in a reactor and subsequently in atmosphere under definable accident conditions. These models have been incorporated into a computer program EERIE (Environmental Evaluation of Radioactive Isotopic Effluents) which enables inhalation doses to body organs, cloud beta and gamma doses, and the quantity of radioactivity deposited on the ground in the area surrounding the site of a release of activity to be evaluated. The data obtained in this study are critically examined in order to provide guidance on the best methods of obtaining adequate monitoring results with sufficient speed to institute effective precautionary measures. The comprehensive nature of the calculations, which enables the complete history of the isotopic content of the damaged fuel to be followed continuously from the time of its release from the fuel through its dispersal in the atmosphere, makes it possible to identify those parameters which are important in determining sampling and monitoring techniques. In addition it makes it possible to evaluate the usefulness of different analytical methods at various stages during an incident. It is concluded that the assessment of inhalation hazards from measurements of isotopes deposited on the ground is considered difficult and unreliable, while measurements of exposed persons would impose too great a time delay. Finally in the extreme of high or low levels of released activity, where the major hazard results from external radiation and contamination of food chains respectively, simple exposure rate measurements are considered adequate for the initial determination of emergency control actions. (Auth)(FOH)

<45>
Mills, M.T., R.C. Dahlman, and J.S. Olson, Ground Level Air Concentrations of Dust Particles Downwind from a Tailings Area During a Typical Windstorm.

1974, September. ORNL-TM-4375; 61 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

This paper describes an atmospheric transport model for suspension by wind of toxic particulates that have been deposited upon the ground surface. Dimensional arguments are given relating suspension and saltation rates. Sample calculations of ground level air concentrations are carried out for a square tailings area containing uranium mill dust. A list of the computer code and sample output from this code are included. Ground level air concentrations downwind from the source are calculated with a Gaussian plume equation modified for plume depletion due to dry deposition. (Auth)(CWM)

<46>
Strenge, D.L., M.M. Hendrickson, and E.C. Watson, RACER - A Computer Program for Calculating Potential External Dose from Airborne Fission Products Following Postulated Reactor Accidents.

1971, June. BNWL-B-69; 67 p. (Battelle, Pacific Northwest Laboratories, Richland, WA 99352)

RACER, a computer code, permits rapid and consistent estimates of whole body dose resulting from exposure to a plume of airborne radionuclides accidentally released from a nuclear reactor. An attempt has been made to make the program flexible enough so that new models, as they are developed, can be incorporated without extensive reprograming. The mathematical models used in RACER include: (1) Source inventory-choice of radionuclides, decay schemes (photon energies), quantity (initial activity); (2) Aerosol behavior within containment spaces. (3) Leakage rates from containment barriers (multiple containment effects). (4) Atmospheric dispersion. (5) Cloud size effect on external dose estimates. (6) Build-up factors. (7) Whole body dose from a cloud. Each of the preceding stages is incorporated into the model and program; however, there are several possible starting places for the calculation depending on the information one has available. (FDH)

Cloud depletion calculations due to ground deposition have not been included because of the large amount of computer space required. However, a depletion factor as a function of distance may be included through a punched card input. A Gaussian plume atmospheric dispersion model is used because it is the most widely accepted method of dispersion assessment.

<47>
Booth, R.S., and Rohwer, P.S., Methodology for Prediction of Dose to Man from Environmental Release of Radioactivity.

1975. Presented at the Fourth National Symposium on Radioecology held at Oregon State University, Corvallis, Oregon May 12-14, 1975, 25 p., unpublished (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

A systems analysis approach has been taken in the development of a computer code called

<47> CONT.

AQUAMOD in order to simulate radionuclide movement through aquatic food chain pathways. The major objective of this model is to estimate the time-dependent intake by man of radionuclides released into an aquatic ecosystem. A secondary objective is to estimate the environmental buildup of long-lived radionuclides if released over extended time periods. These objectives are achieved within the extent and accuracy of existing data. However, the model is structured so that in the event of data limitations the predicted radionuclide intakes by man are more likely to be overestimated than underestimated. Transfer coefficients of AQUAMOD are expressed in terms of basic environmental and ecological data so that the model can be realistically applied to a variety of aquatic systems. The data required to implement the model for a particular system are limited to that which is easily measurable and/or often available, and the more important model parameters are identified. (DLS)

AQUAMOD exists as a code but the working condition of that code is unknown. Also, there is no documentation on the code itself.

<48>

Cooper, R.E., RADOS, A Code to Estimate Gamma Dose from a Cloud of Radioactive Gases..

1967, June. DP-1098; 23 p. (Savannah River Laboratory, Aiken, SC 29801)

RADOS is a computer code that represents the finite spatial distribution of airborne source material as an infinite number of line sources. The code provides a means of rapidly calculating whole body gamma dose from a finite cloud of radioactive material. The simplifying assumptions used to minimize computation time are: (1) The material release is instantaneous and results in a radioactive cloud of unit thickness in the X or downwind direction. (2) There is no change in the size and shape of the cloud during passage over the effective range of the receptor. (3) The gamma buildup factors can be expressed analytically with sufficient accuracy. These assumptions limit the applicability of RADOS as follows: the material release should occur over a relatively short time; the receptor distance should be more than 600 meters downwind from the release point; and the gamma energies should be in the specified range. A 4 group, 12 isotope, 1 receptor-point problem requries approximately 1.75 seconds on the IBM System/360-65. This is approximately 8 percent of the running time for the more detailed CLOUD code. Dispersion is esentially based on the Gaussian assumption. It is used for accidental releases from reactors. (Auth) (CWM)

<49>

Willis, C.A., G.A. Spangler, and W.A. Rhoades, A New Technique for Reactor Siting Dose Calculations.

1970. Health Physics, 19, 47-54 (McDonnell Douglas Astronautics Company, Santa Monica, CA)

The COMRADEX code was developed to calculate radiation doses from accidents postulated for advanced reactors with sophisticated engineered safeguards. The use of a direct numerical technique, designed especially for quasilinear differential equations, to solve the leakage and decay chains, permits the inclusion of several special features. These include the capability for considering up to four levels of containment, each with an independent time-varying leak rate. Provision is also included for filtration, particle coagulation and fallout, and special cleanup systems. Shielding may be introduced between the various containment shells. A full fission product inventory (plus special isotopes) is used with input release fractions. The meteorology model is that of Pasquill and Gifford. Internal doses are determined with the International Commission on Radiological Protection (ICRP) model. The cloud gamma dose is obtained by integrating over the anisotropic Gaussian cloud; this calculation is simplified by assuming a single air attenuation kernel for all gammas. (Auth)

<50>

Strenge, D.L., and E.C. Watson, A Computer Program for Calculating Annual Average External Doses From Chronic Atmospheric Releases of Radionuclides.

1973, June. BNWL-B-264; 132 p. (Battelle, Pacific Northwest Laboratories, Richland, WA 99352)

In KRONIC the atmospheric dispersion effects are estimated using data on joint frequency of occurrence of wind speed, wind direction and stability for a particular site. Each sector is considered to be a plume for which the center line ground level dose is calculated for specified downwind distances. The result of the calculation is a table of annual dose rates as a function of direction and distance from the release point. Both photons and beta particles can give significant contributions to the external total body dose. The beta dose contribution is easily calculated using a semi-infinite cloud model. This model may be used because the range of beta particles in air is short compared to the dimensions of plumes considered. The gamma dose calculation is more complicated because of the relatively long range of photons in air. To properly determine the gamma contribution it is necessary to perform a space integration over the plume volume. The integration technique used in the reactor accident analysis computer program RACER is employed in KRONIC except that here the plume width is determined by sector boundaries rather than by a Gaussian concentration gradient. The gamma dose is calculated as a tissue dose at the body surface, at 1 cm depth and at 5 cm depth. The beta dose and gamma doses are reported separately, however, the beta dose and gamma surface dose may be added and reported as skin dose if desired. (FOH)

Mathematical models and a computer program are described for calculation of annual average beta and gamma doses resulting from chronic release of radionuclides to the atmosphere.

<51>

Reeves, M., P.G. Fowler, and K.E. Cowser, A Computer Code for Analyzing Routine Atmospheric Releases of Short-Lived Radioactive Nuclides.

1972, October. ORNL-TM-3613; 20 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

A computer code is presented which calculates average annual ground level air concentrations, deposition rates, and ground concentrations for a decaying chain of radioactive nuclides. Members of this chain are assumed to have been emitted from a stack under routine nonaccidental operating conditions. Averages are performed relative

<51>

<51> CONT.
to atmospheric stabilities, wind speeds, and wind directions. (Auth)

<52>
Daniels, D.G., J.C. Sonnichsen, and R.T. Jaske, The Estuarine Version of the COLHEAT Digital Simulation Model.

1970, June. BNWL-1342; 40 p. (Battelle, Pacific Northwest Laboratories, Richland, WA 99352)

COLHEAT is an aquatic dispersion model which in this case was modified to simulate spatial and temporal variations of temperature, chemical, and radionuclide concentration above the salt intrusion in coastal plain estuaries where the river flow is dominant. This report explains the modifications and describes a case study on the Columbia River estuary. (CTG)

<53>
Blaine, R.A., and E.L. Bramblett, AISITE II, A Digital Computer Program for Investigation of Reactor Siting.

1964. NAA-SR-9982; 94 p. (Atomics International, Canoga Park, CA)

AISITE II automatically varies any one of 46 parameters such as reactor power, building leak rate, iodine cleanup rate, halogen filter efficiency, etc. The siting criteria of 10CFR100 are used in AISITE II with computation of the exclusion area, low population boundary zone, and low-population-center distances as functions of any parameter. In addition, dose versus distance curves are obtained for inhalation, direct building, and direct cloud doses. The printed edit consists of the dose versus distance data, fractional contribution by isotope group to the inhalation dose, and critical distances for both a 2 hr and a 30 day release. Graphical displays can be obtained for critical distances as a function of the variable parameter and dose versus distance for any organ or the critical organ, for each value of the parameter. Three isotope release models with up to four levels of containment are provided. (Auth) (CTG)

<54>
Bonzon, L.L., and J.B. Rivard, Computational Method for Calculation of Radiological Dose Resulting from Hypothetical Fission Product Release.

1970, July. SC-RR-70-338; 148 p. (Reactor Studies Division 5222, Sandia Laboratories, Albuquerque, NM)

An improved method for the calculation of radiological doses resulting from the release of fission products is presented, including the derivation of, and the limitations on, the inhalation dose equation. The method is applied in the linked computer codes FISSP and CLOUD to the calculation of external and internal doses resulting from the hypothetical release and dispersal of 326 fission product nuclides and metastable states. The sensitivity of the results to parameter variation is reported, including a study of the influence of unlisted MPC values upon the inhalation doses. (Auth)

<55>
Travis, J.R., A Model for Predicting the Redistribution of Particulated Contaminants from Soil Surfaces.

1975, August. LA-6035-MS; 63 p. (Los Alamos Scientific Laboratory, Los Alamos, NM 87545)

A computerized model was developed to describe the redistribution of wind eroding soil-contaminant mixtures. Potentially mobile particulate contaminants can, in the first approximation, be assumed to be indistinguishable from the wind eroding soil in which they are distributed. A grid network characterizes important soil and surface conditions, and mass conserving control volumes are constructed on each cell. Material is transported through the vertical and top surfaces of a control volume by a modified Bagnold-Chepil horizontal flux formulation and modified Gillette vertical flux formulation, respectively. The vertical emissions, considered as puffs from area sources, create at regular time intervals a contaminant cloud which is proportional to the suspendable ground concentration. These puffs diffuse downwind under time-dependent wind velocity and atmospheric stability conditions, maintaining during the time interval a three-dimensional Gaussian distribution of concentration with cloud volume. Material from each puff is deposited in downwind cells, leading to the possibility of many different flights from these new soureces. The usefulness of this predictive tool is demonstrated by calculations involving mixtures of particulate Pu 238 Pu 02 in highly erodible soils under dust storm conditions. Time-dependent surface concentration and breathing zone exposure isopleths, evolving from a small contaminated area, show the potential hazard from wind eroding toxic materials. Radioactive decay is not considered. The model essentially assumes a one time external deposition. (Auth) (CWM)

<56>
Ellison, T.W., and A.B. Dunham, A CDC 6600 Program for Calculation of Inhalation and External Dose for Establishing Reactor Siting Distance Requirements.

1967, January. KAPL-M-65; 24 p. (General Electric Company, Knolls Atomic Power Laboratory, Schenectady, NY)

The Dose B Program has been written to facilitate the calculation of distance factors for nuclear reactor sites as described in the well known USAEC Document "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844. The calculations are essentially as described in this document except that the dose code allows for arbitrary operating power histories and variable containment leak rates. The program calculates the thyroid and direct gamma doses received by an individual standing at a distance D from the point of fission product release for a time TAU subsequent to the time of release and based upon a given reactor operating history. (Auth)

This document does not contain a complete description of the conceptual model on which DOSE B is based.

<57>
Houston, J.R., and K.R. Heid, PUDEQ: A Computer Code for Calculating Dose Equivalent from Internal Deposition of Plutonium at Hanford.

1975, October. BNWL-B-450; 40 p. (Battelle Pacific Northwest Laboratories, Richland, WA 99352)

<57> CONT.
Presented here are the procedures and mathematical models used in developing PUDEQ, a computer program for computing the dose equivalent to body organs from intake of plutonium. The program was designed specifically to use the data recorded on the Hanford Internal Exposure (HIE) System (1) magnetic tape as input. Insofar as was possible, the recommendations of the Advisory Committee on Dose from Plutonium and other Transuranics was followed. (2) Some deviations were made where errors, omissions, or inconsistencies were found, after consultation with members of the Committee. In the current version of the program only plutonium and its immediate important daughters are considered. The program could, however, be expanded to include other transuranic nuclides. At present, only a few depositions of transuranic nuclides other than plutonium are recorded out of about 450 individuals involved in a total of over 700 plutonium intakes. (Auth)

<58>
Booth, R.S., S.V. Kaye, and P.S. Rohwer, A Systems Analysis Methodology for Predicting Dose to Man from a Radioactively Contaminated Terrestrial Environment.

1971, May. CONF-710501; ORNL 546; Part of Nelson, D.J. (Ed.) Proceedings of the Third National Symposium on Radioecology held in Oak Ridge, Tennessee, May 10-12, 1972, p. 877-893 (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

A systems analysis methodology has been developed to predict intakes by man, estimate dose commitments, and identify "critical" exposure pathways resulting from radioactivity releases to a terrestrial environment. A mathematical model was constructed simulating selected terrestrial pathways by which fallout radioactivity can be transferred ultimately to man. This model is intended for preliminary predictions of radionuclide intakes by man through consumption of milk, beef, and plant parts contaminated directly by fallout as well as by uptake from the soil. Differential equations were written describing the radionuclide transfers, and the parameters required to implement these equations were derived. In verification tests, model predictions of Cs 137 concentrations in environmental compartments agreed within a factor of 2 with experimental data. More extensive verification studies involving other radionuclides are needed. The intake rates and total intakes of seven sample radionuclides were predicted for a "reference" man based on a hypothetical fallout of 1p Ci/cm(E+2) of each radionuclide. These predicted intakes were used as input to an internal dosimetry model which calculated dose commitments for each radionuclide and pathway. The results of these calculations are discussed in relation to individual radionuclide contributions to dose commitment. This generalized food-chain model is sufficiently versatile that it can be applied to many terrestrial environments and to all radionuclides. The first version of the model is expected to undergo future refinements. The predicted intakes are calculated with a computer code called TERMOD. The internal dosimetry calculation is performed with the INREM computer code. (Auth) (DLS)

This is the first of two slightly different reports describing TERMOD.

<59>
Booth, R.S., and S.V. Kaye, A Preliminary Systems Analysis Model of Radioactivity Transfer to Man from Deposition in a Terrestrial Environment.

1971, October. ORNL-TM-3135; 21 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

A mathematical model was constructed simulating selected terrestrial pathways by which fallout radioactivity can be transferred ultimately to man. This model is intended for preliminary predictions of radionuclide intakes by man through consumption of milk, beef, and plant parts contaminated directly by fallout as well as by uptake from the soil. The intake rates and total intakes of 25 radionuclides were predicted for a "standard" man based on a hypothetical fallout of 1 uCi/m(E+2) of each radionuclide. In the first few days after deposition, milk and directly contaminated plant parts provided the greatest intake of radioactivity from this mixture of radionuclides. After about 30 days, the intake rate via beef was 2.5 times more important than that from directly contaminated parts. Finally by 500 days, the intake rate from plant parts contaminated by uptake from the soil was the most important source of radioactivity to man in the contaminated area. This generalized dynamic model is sufficiently versatile that it can be applied to many terrestrial environments and to all radionuclides. This first version of the model is to undergo future refinements. (Auth)

This is one of two slightly different reports describing TERMOD. This report presents some results not contained in the earlier report.

<60>
Heller, F., W. Schikarski, and A. Wickenhauser, MUNDO - A Digital Code for Computing Accident Dose Rates in Reactor Environs.

1967, September. KFK-653; EUR 3684d; EURFNP-461; 33 p. (Institut fur Angewandte Reaktorphysik, Karlsruhe, Germany)

The MUNDO computer code calculates the dose equivalent received at the point P and time T in rem as a function of meteorological and internal physical and engineering properties of the reactor. The code considers fuel release factors, decontamination factors, filter factors, leak functions, multi-containment, ground release, stack release, weather conditions, external irradiation, internal irradiation, and direct irradiation from the reactor building. Approximately 80 nuclides can be included in the calculation. Atmospheric dispersion includes Sutton's diffusion parameters, and plume depletion via deposition and decay are accounted for. Exposures are calculated for inhalation, external exposure to the plume, and external exposure to the ground. Ingestion and wet deposition are not considered. (FOH)

<61>
Bramati, L., T. Marzullo, I. Rosa, and G. Zara, VADOSCA: A Simple Code for the Evaluation of Population Exposure Due to Radioactive Discharges.

1973, Sept.. CONF-7309007-P2; Part of Proceedings of the Third International Congress of International Radiation Protection Association held in Washington, DC, September 9-14, 1973, p. 1072-1077 (Italian National Electric Energy Agency (ENEL))

<61>

<61> CONT.
The code consists of two parts, one for liquid discharges (VADOSCA-LI) and one for gaseous discharges (VADOSCA-GAS). It incorporates the transfer parameters of twenty-four radioisotopes in the case of liquid discharges, and of twenty radioisotopes in the case of gaseous discharges. It allows the evaluation of the concentrations of the various isotopes in all the compartments of the critical paths outlined in the International Commission on Radiological Protection (ICRP) Publication No. 7, and the evaluation of the annual doses for five critical organs (whole body, G.I. tract, thyroid, bones, lung) for the various critical groups of population on the basis of environmental parameters such as the time of residence in a certain area, diet, type of activity, hydrological regimen, irrigation methods, and meteorological conditions. Although extremely simple, the code allows rapid performance of all the evaluations required to define the amount of radioactivity that can be released and the associated exposures. In its present form, the code VADOSCA-LI covers twenty-four radionuclidesincluding some fission products, some activation products, plus an alpha emitter. VADOSCA -LI calculates the doses to four special groups, namely, fishermen, other workmen, local population, farmers. A separate calcualtion is made for the doses originated by drinking water. The doses are evaluated for five critical organs and they are expressed in mrem/year if the discharges are expressed in Ci/yr. In VASDOSCA-GAS, twenty nuclides are considered , including eight noble gases (fission and activation products, various isotopes as particluates, and H3, N13, C14 in the form of vapor. Atmospheric diffusion is evaluated with Pasquill's theory adapted to the meteorological data available. In its present form VADOSCA-GAS does not take into account particular effects such as the cloud depletion due to deposition, down draft and building effect; the particulate deposition rate is assumed constant at 3 x 10(E-2) m/s. (FOH)

This report is a description of the capabilities and applicability of VADOSCA. No reference is given for complete documentation of the code. The code VADOSCA LI is written in FORTRAN; it needs 20 K memories and it takes approximately 15 seconds in a GE 635 computer. VADOSCA-GAS is written in FORTRAN and needs 60 K memories and takes about 60 seconds in a GE 635 computer.

<62>
Coates, R.L., and N.R. Horton, RSAC -- A Radiological Safety Analysis Computer Program.

1966, May. IDO-17151; TID-4500; 208 p. (Phillips Petroleum Company, Atomic Energy Division, Idaho Operations Office)

RSAC is designed to compute the potential radiological doses resulting from either a continuous or instantaneous release of radioactive fission products to the atmosphere. The program has been developed as part of LOFT and the general nuclear safety programs to provide a generalized computer program for a complete parametric investigation of the radiological hazards. The doses considered are cloud gamma, deposition gamma, ingestion, and inhalation. They are based upon the release of up to 200 isotopes, the relative abundance of each isotope being dependent upon the operating history of the nuclear reactor under consideration. Included are the mathematical models, conversion factors, assumptions, flow charts, program listings, and other data considered pertinent to the overall development of this program. The program is written in SYMBOLIC LANGUAGE (MAP) for the IBM 7040 computer and is divided into the following five major ctegories: (1) Curie -- The fission product inventory generated for a given set of operating conditions (2) Cloud Gamma Dose -- The external gamma dose received as a consequence of exposure to airborne radioactive materials (3) Deposition Gamma Dose -- The external gamma dose received as a consequence of exposure to deposited fission products (4) Inhalation Dose -- The internal dose to eight critical organs received as a consequence of inhaling airborne radioactive materials (5) Ingestion Dose -- The internal dose to the thyroid received as a consequence of ingesting iodine contaminated milk. This dose is calculated using the inhalation dose conversion factor and multiplying this factor times a correction factor to account for bioaccumulation and milk consumption. (FOH)

A FORTRAN version of this code has been published in IDO-17261 (1968, June). Sutton's equation is used for atmospheric dispersion. Although the description of the code refers to the prediction of ingestion doses, these exposures are only calculated via a single adjustment factor converting the inhalation dose factor to an ingestion dose factor. The ingestion pathway is only determined for exposure to the thyroid.

<63>
Richardson, L.C., User's Manual for the Fortran Version of RSAC A Radiological Safety Analysis Computer Program.

1968, June. IDO-17261; TID-4500; 136 p. (Phillips Petroleum Company, Atomic Energy Division, Idaho Operations Office)

This report constitutes a user's manual for the FORTRAN version of the RSAC computer code. The mathematical model and equations are described in IDO 17151, RSAC -- A Radiological Saefty Analysis Computer Program, by R.L. Coates and N.R. Norton, dated May 1966. The program computes the potential radiological doses resulting from either a continuous or instantaneous release of radioactive fission products to the atmosphere. The doses considered are cloud gamma, deposition gamma, ingestion, and inhalation. They are based upon the release of up to 450 isotopes, the relative abundance of each isotope being dependent upon the operating history of the nuclear reactor under consideration. The program is divided into the following five major categories. (1) Curie -- The fission product inventory generated for a given set of reactor operating conditions (2) Cloud Gamma Dose -- The external gamma dose received as a consequence of exposure to airborne radioactive materials (3) Deposition Gamma Dose -- The external gamma dose received as a consequence of exposure to deposited fission products (4) Inhalation Dose -- The internal dose to one of nine critical organs received as a consequence of inhaling airborne radioactive materials, and (5) Ingestion Dose -- The internal dose to one of nine critical organs received as a consequence of ingesting radioactive materials. This report constitutes a user's guide for a later version of the code written for the most part in FORTRAN IV for the IBM 7044. With the exception of input options, input and output format changes, the addition of an option for calculation of the horizontal and vertical diffusion coefficients using Pasquill's

<63> CONT.
parameters, and the inclusion of building turbulence effects, this version is the same as the original (machine language) version. The mathematical models and calculations are the same as described in IDO-17151 and are not described in this report. (Auth) (POH)

A user-supplied multipler is needed in order to convert the inhalation dose to an ingestion dose.

<64>
Mackay, T.F., and R.F. Ely, Jr., Computation of Radiological Consequences Using INHEC Computer Program.

1974, February. GAI-TR-101P; 210 p. (Gilbert Associates Incorporated, 525 Lancaster Avenue, Reading, PA 19603)

INHEC is a computer code specifically prepared for parametric investigation of radiological consequences involved in the accidental release of radioactivity from a nuclear power plant. The computer code is designed to calculate off-site doses from time-varying releases of radioactivity to the environment with an option to calculate the resulting activity buildup in a secondary volume (e.g., a nuclear power plant control room). The code has wide applicability since it can be used in the analysis of both single and double containment systems with varying purge and/or leak rates from either containment. also, the code has the flexibility of computing either instantaneous or time-varying activity releases to the innermost level of containment and activity reduction from internal cleanup systems such as containment sprays and filtering units. The meteorology contained in Regulatory Guides 1.3 and 1.4 is available as an internal data table; however, other meteorology may be used as desired. The printed computer output includes the off-site inhalaticn doses, the off-site cloud center and ground level gamma doses, the off -site cloud center and surface body beta doses, and activity inventories in the containment(s) and on filters. The printed computer output resulting from the secondary volume option includes inhalation and surface body beta doses, integrated activity exposure inside the secondary volume, and activity inventories in the secondary volume and on related filters. Also, a gamma source sort of each activity inventory is provided to facilitate shielding calculations. All activities and doses are calculated as a function of time and, where applicable, as a function of distance.

<65>
Plato, P.A., D.F. Menker, and M. Dauer, Computer Model for the Prediction of the Dispersion of Airborne Radioactive Pollutants.

1967. Health Physics 13, 1105-1115 (University of Miami, School of Medicine, Miami, FL)

Presented are two computer programs written for an IBM 1401/7040 digital computer in a FORTRAN IV computer language. The programs are called RADS and ARADS, acronyms for Radiological Atmospheric Dispersion Study and Alternate RADS, respectively. Both programs are designed to investigate the use of various equations formulated to predict the dispersion of radioactive effluents deposited into the atmosphere from a smoke-stack. The release of the effluents may be of either an instantaneous or continuous nature. Predictions of atmospheric concentrations are made for approximately 600 points throughout an observation area surrounding the smoke source, allowing contour lines of equal concentrations to be drawn. The RADS program was designed for one particular effluent release. The program predicts the path of the smoke from the source to one of the boundaries of the observation area. Predictions are then made concerning the dispersion of the smoke. The ARADS program uses average meteorlogical conditions to predict the air concentrations resulting from long-term releases of a smokestack effluent. The output data for both programs contain a scaled map of the observation area showing the location of the predicted concentrations in order to simplify the drawing of the contour lines. While the authenticity of the prediction equations is not certain, a definite relationship between relative air concentrations with respect to the location of the source was established. The programs permit fast, accurate, and voluminous solutions to the complex equations, and provide a tool with which to examine the prediction equations themselves. (Auth)

<66>
Plato, P.A., D.P. Menker, and M. Dauer, Some Examples and Limitations of the RADS and ARADS Computer Programs.

1969. Health Physics, 16, 383-391 (University of Miami School of Medicine, Miami, FL 33152)

Computer programs dealing with the prediction of the dispersion of airborne radioactive pollutants have already been written. Continued use of these programs has shown the flexibility of these programs, as well as some of their weaknesses. Presented in this paper are examples of both conditions. The RADS program predicts the path of smoke from the source to one of the boundries of the observation area. The ARADS program uses average meteorological conditions to predict the air concentrations resulting from long term releases of a smokestack effluent. Both RADS and ARADS programs have proven to be extremely flexible in their handling of various sizes of observation areas. To further improve the flexibility of RADS, additional work is needed to allow it to differentiate between turning and converging conditions and to handle the condition of wind calms. (Auth) (JTE)

<67>
Wong, P., A Fortran Program to Determine Absorbed Organ Dose Due to Inhalation of Radionuclides.

1967, January. USNRDL-TR-67-7; 106 p. (U.S. Naval Radiological Defense Laboratory, San Francisco, CA 94135)

A FORTRAN computer program has been developed to determine the maximum internal dose received by a given body organ as a result of inhalation of specific radionuclides. The program, based upon equations reported in the literature, determines the effective energy absorbed by the body organ per disintegration of the radionuclide, and computes the absorbed organ dose for various periods of intake for single or continuous release and for various periods of internal exposure. The organ dose includes effects from all descendants of the nuclide originally considered to have entered the body organ. Tabulated results for 133 radionuclides and their contributions to the dose to seven body organs and the whole body are presented. Input to the FORTRAN computer program may be modified to yield results for other radionuclides or body organs. The computer

<67> CONT.
program may also be used to determine (by summing) the total absorbed organ dose for a mixture of fission products if the concentration in air of each major contributor at initial intake time is known. (Auth)

<68>
Clarke, R.H., The WEERIE Program for Assessing the Radiological Consequences of Airborne Effluents from Nuclear Installations..

1973. Health Physics, 25, 267-280 (Central Electricity Generating Board, Berkeley Nuclear Laboratories, Berkeley, Gloucestershire, England)

A mathematical model has been devised for guidance in the safety and siting aspects of nuclear installations under operational or accident conditions. The model begins with the full fission product inventory applicable to the fuel at the time of interest generated for any reactor type and irradiation history. The amount of fuel involved in the incident may vary as a function of time, and either a time decaying or constant fission product inventory may be used to specify the release. The activity leaks from the circuit with allowances for time-dependent plateout, resuspension and filtration of each elemental species. By preserving the full nuclide decay schemes, nuclides leaking into the atmosphere depend not only upon their own release behaviour but also on that of their precursors. The effluent in the atmospheric is dispersed from an effective stack height and allowance is made for building entrainment. Standard meteorological dispersion models are used and the effects of radioactive buildup and decay, ground and inversion reflections and ground deposition are taken into account. WEERIE then evaluates the inhalation and the cloud beta doses, while integration over the volume of the plume leads to estimates of the cloud gamma exposure. WEERIE is written in FORTRAN IV (H) and executes on the IBM 370/165 computer requiring 250 kbytes of fast core storage. Execution times vary with the degree of precision requested in the cloud gamma routine, but typically doses and exposures may be evaluated at the rate of 200 observation points per minute. (Auth)

A Time-dependent code for atmospheric dispersion and dosimetry used extensively in the United Kingdom for environmental assessment is presented.

<69>
Voilleque, P.G., AERIN, A Code for Acute Aerosol Inhalation Exposure Calculations.

1970. Health Physics, 19, 427-432 (Health Services Laboratory, U.S. Atomic Energy Commission, Idaho Falls, ID)

AERIN is a FORTRAN IV program written to simplify the application of equations for computing the organ and tissue burdens and doses resulting from an acute inhalation exposure to a radioactive aerosol. The models which form the basis for the calculations are (1) the deposition and clearance models of the International Commission on Radiological Protection (ICRP) Task Group on Lung Dynamics, (2) the gastrointestinal (GI) tract model of Eve (1966) and (3) a single exponential model for the "critical" organs and tissues. The fundamental data for the respiratory and GI tract models are written into the program. The user can select suitable dosimetric parameters for the respiratory and GI tract tissues, choose the appropriate metabolic and dosimetric quantities to be used in calculations for the "critical" organs and tissues, and specify the number of computations to be made and the time interval between them. The output consists of a complete listing of the assumed parameters as well as the organ and tissue burdens and doses as functions of time post exposure. (Auth)

<70>
Specht, E., C. Martin, J. Otter, and R. Hart, Description of the COMRADEX-II Code.

1975, February. TI-001-130-048; 113 p. (Atomics International Division, Rockwell International, Canoga Park, CA)

The COMRADEX-II (COntainment and Meteorology of environmental RADiation EXposure) code was developed for the calculation of radiological doses from hypothetical power reactor accidents. It permits the user to analyze as many as four levels of containment with time varying leakage and fallout rates in each level. Filtration in each containment region may be by isotopic class (noble gas, solids, or halogens). Shielding may be introduced between the various containment shells. The environmental doses calculated include the direct gamma dose from the containment building, the internal doses to as many as 12 body organs (including the thyroid, bone, lung, etc.) due to inhaling the airborne radioactivity, and the external gamma and beta doses from the cloud. Leakage and fallout rates, and initial inventory must be given as input data. A direct numerical technique, designed especially for quasi-linear differential equations, is used to solve the leakage (as well as containment region fallout) and decay chains. The meteorology model is that of Pasquill. Factors are given in this document for computing internal doses with the ICRP-2 or the TGLD "New Lung" models. The cloud gamma dose is obtained by integrating over an anisotropic Gaussian cloud. The external beta dose calculation assumes a semi -infinite radioactive cloud. This code represents a modification of the earlier COMRADEX code. (Auth) (CWM)

<71>
Onishi, Y., P.A. Johanson, R.G. Baca, and E.L. Hilty, Studies of Columbia River Water Quality, Development of Mathematical Models for Sediment and Radionuclide Transport Analysis.

1976, January. BNWL-B-452; 42 p. (Battelle, Pacific Northwest Laboratories, P.O. Box 999, Richland, WA 99352)

The program undertaken by PNL to study the water quality of the Columbia River consists of two separate segments: (a) Sediment and Radionuclide Transport Program, and (b) Columbia River Temperature Analysis. For the Sediment and Radionuclide Transport Program, quasi-two dimensional (longitudinal and vertical directions) mathematical simulation models have been developed for determining radionuclide inventories, their variations with time, and movements of sediments and individual radionuclides in the freshwater region of the Columbia River below Priest Rapids Dam. These codes are presently being applied to the river reach between Priest Rapids and McNary Dams for the initial sensitivity analysis. In addition, true two-dimensional (longitudinal and lateral directions) models have been formulated and are presently being programmed to provide more detailed information on sediment and

<71> CONT.
radionuclide behavior in the river. For the Temperature Analysis Program, river water temperature data supplied by the U.S. Geological Survey for six ERDA-sponsored temperature recording stations have been analyzed and cataloged on storage devices associated with ERDA's CDC 6600 located at Richland, Washington. These models may be used to study the transport of radionuclides routinely or accidentally released from a nuclear reactor. (auth) (DLS)

Contains a description of two different 2-D transport models, the names of which were obtained via private communication with Y. Onishi. FETRA is the 2-D (lateral and longitudinal) finite element transport model used for estuaries, coastal areas and wide lakes. It is written in FORTRAN IV for a Cyber 7600 computer. SERATRA is the 2-D (vertical and longitiudinal) sediment and radionuclide transport model used for narrow estuaries, rivers, and deep lakes. It uses finite differencing and is written in FORTRAN IV for the PDP 11/45.

<72>
Start, G.E., and L.L. Wendall, Regional Effluent Dispersion Calculations Considering Spatial and Temporal Meteorological Variations.

1974, May. ERL ARL-44; 63 p. (Air Resources Laboratories, National Oceanic and Atmospheric Administration, Idaho Falls, ID)

An objective regional trajectory analysis scheme has been combined with a Gaussian diffusion model to yield a technique called MESODIF (mesoscale diffusion). The trajectory analysis scheme utilized wind data from a network of tower-mounted wind sensors to consider the effects of spatial variabilities of horizontal wind flow near the surface, incorporated time changes in rates of diffusion, and used an upper level lid to vertical mixing. The MESODIF calculations of total integrated concentrations were compared with corresponding conventional calculations, using wind-rose joint frequency statistics (single wind station). Comparisons were made within a region about 100 by 130 km, and for time spans from 6 hr to 1 yr. The diagnostic comparisons of regional dispersion effects from each technique showed significant differences over the range of scales considered. Effluent recirculations and stagnations, related to local wind variabilities about terrain features, were believed to produce localized zones of enhanced exposure to airborne effluents. These zones were not resolved by the conventional wind-rose model. For short or accidental type of emissions, the greatest shortcoming of the single wind-station dispersion model was its failure to identify, when applied within a region of spatially variable winds, the subregion which would be affected. At distances beyond about 25 to 50 km, the wind-rose model calculations were significantly biased to overestimation of annual total integrated concentrations by about an order of magnitude. The inability of the wind rose model to accommodate time changes in stability category during effluent transport to the more distant receptors was the single most influential factor of this bias. Current usage of the wind rose technique for regional dispersion calculations, especially at the longer distances, incorporates some systemic bias in the evaluations. These shortcomings are points of concern and should be reconciled with whatever impact assessment schemes are to be utilized within the mesoscale or regional domain. (Auth)

<73>
Fletcher, J.F., W.L. Dotson, D.E. Peterson, and R.P. Betson, Modelling the Regional Transport of Radionuclides in a Major United States River Basin.

1973. CONF-730503-5; IAEA-SM-172/40; Part of Proceedings of a Symposium on Environmental Behavior of Radionuclides Released in the Nuclear Industry, held in Aix-En-Provence, France, May 14-18, 1973, p. 449-465 (Hanford Engineering Development Laboratory, Richland, WA; Tennessee Valley Authority, Knoxville, TN)

A comprehensive study of the radiological implications of large-scale use of nuclear power generation, addressed to the Tennessee-Cumberland River Basins in the year 2000, is being undertaken. In the course of this study computer modelling techniques have been developed to evaluate the movement of radionuclides, by water and air transport, from points of release at a large number of plant sites and to calculate patterns of radionuclide concentration throughout the region of study. Air transport calculations utilize a modification of the bivariate normal equation, with the vertical parameter sigma(z) characterized as a function of distance from the source, stability and mixing layer depth. Local variations in air flow caused by major terrain features are modelled. Depletion of the airborne radionuclides by wet and dry precipitation processes is accounted for and the concentrations of deposited radionuclides are calculated. The model sums contributions from all sources to provide resultant patterns of concentration of airborne and deposited radionuclides, including contributions from radionuclides in solution and absorbed on sediments. Effects of flow stratification and sediment trapping in reservoirs are considered. The solution of air-deposited radionuclides and their subsequent transport in ground and surface waters are modelled. Calculational routines are provided for simulating radionuclide transport in each stream in the basin and for calculating the resultant concentrations in solution and associated with suspended and deposited sediments. Together, the air and water transport models are designed to calculate regional patterns of radionuclide concentrations in the environment that result from the operation of nuclear facilities in a region. These concentration patterns can then be applied to the evaluation of radiological dose to man. (Auth)

<74>
Cowser, K.E., R.S. Booth, B.R. Fish, W.S. Snyder, J.P. Witherspooon, G.R. Siegel, and W.H. Wilkie, Methods of Estimating Dose to Man from Regional Growth of Nuclear Power.

1973. CONF-730503-5; IAEA-SM-172/41; Part of Proceedings of a Symposium on Environmental Behavior of Radionuclides Released in the Nuclear Industry, held in Aix-En-Provence, France, May 14-18, 1973, p. 467-482 (Oak Ridge National Laboratory, Oak Ridge, TN; Tennessee Valley Authority, Muscle Shoals, AL; Tennessee Valley Authority, Chattanooga, TN)

In co-operation with the United States Atomic Energy Commission (USAEC), the Oak Ridge National Laboratory (ORNL), the Tennessee Valley Authority (TVA), the Atmospheric Turbulence and Diffusion Laboratory-National Oceanographic and Atmospheric Administration

<74> CONT.
(ATDL-NOAA), and the Hanford Engineering Development Laboratory (HEDL) have undertaken a joint study of the potential radiological impact of an expanding nuclear power economy on the Tennessee Valley Region (TVR). The TVR study involves an appraisal of the generation, management, and control of radioactive effluents from nuclear facilities in the region to the year 2000. Consideration is given to nuclear power requirements to the year 2000 and sites for nuclear facilities, to radionuclides that may be released and their transport in air and water and concentration in terrestrial and aquatic systems, and to the estimation and interpretation of the potential dose to man. This is the second region study initiated by the USAEC in the continental United States of America, the first considering the region defined as the Upper Mississippi River Basin (UMRB). A computer model known as HERMES (Hanford Engineering Regional Model for Environmental Studies) was developed by HEDL and applied in the UMRB study. The model permits estimates, on a regional basis, of the potential radiation dose that may be received by individuals and population groups as a result of radioactive materials estimated to be released from reactors and fuel reprocessing plants. Modifications in HERMES are discussed within the context of the uniqueness of the TVR and additions or improvements based on the initial application of HERMES. The paper, one of three companion reports, includes information on the data requirements and the components of HERMES concerned with the estimation of dose to man from internal and external modes of exposure. (Auth)

<75>
Soldat, J.K., D.A. Baker, and J.P. Corley, Applications of a General Computational Model for Composite Environmental Radiation Doses.

1973. IAEA/SM-172/82; BNWL-4511; CONF-730503-5; Part of Proceedings of a Symposium on Environmental Behavior of Radionuclides Released in the Nuclear Industry, held at Aix-en Provence, France, May 14-18, 1973, 19 p. (Battelle, Pacific Northwest Laboratories, Richland, WA)

A mathematical model for calculation on a large general-purpose digital computer of regional radiation doses resulting from large-scale use of nuclear energy was previously developed and reported. This general model has now been sub-divided to permit rapid calculations for the several exposure pathway groupings in an interactive mode using the BASIC computer language. The sub-programs are completely flexible as to the nuclides, body organs, and pathways from which radiation doses are to be calculated. They include at the present time: approximately 150 radionuclides, including transuranics; doses to whole body, skin, bone, lungs, thyroid, and gastrointestinal tract; sub-programs for cloud submersion, inhalation of nuclides other than radioiodines (resuspension of deposited nuclides is not included), ingestion of water and aquatic foodstuffs along with external dose from water and sediments, ingestion of irrigated crops, thyroid dose from inhalation and ingestion of irrigated crops, thyroid dose from inhalation, and ingestion and dose to aquatic biota can also be calculated. Dose factors in the programs for the various media-nuclide-organ combinations have been calculated using ICRP methods. For radionuclides with long effective half-lives, the sub-programs calculate either total dose commitment for a single year's intake or the dose-rate at the end of a specified period of years at constant annual intake. Transfer factors between trophic levels have been taken in most part from summaries published by others, and are updated as newer data becomes available. The major application to date of the compartmented model has been the calculation for the U.S. Atomic Energy Commission of environmental impact statements. Use is also being made of the same model for evaluation of potential radiological impacts associated with the ultimate fate of radioactive wastes for various long-term disposal concepts. (Auth)

<76>
Dotson, W.L., A Regional Air Transport Model for Radiological Dose Studies.

1973. CONF-730401-2; HEDL-SA-514; Part of Realism in Environmental Control, Proceedings of the Nineteenth Annual Technical Meeting and Equipment Exposition of the Institute of Environmental Sciences, held at Anaheim, CA, April 2-5, 1973 (Hanford Engineering Development Laboratory, Richland, WA)

A computer code, HERMES, was developed for calcultions from which a reasonable estimate may be derived of the potential radiation dose and dose commitment to individuals and population groups residing in regions where nuclear power reactors, and ancillary activities such as fuel reprocessing plants and waste storage facilities, are located in the year 2000. The radionuclide air transport module, ARTRAN, of the code is described. The three types of data used include: source to receptor travel distance; effluent source rate and constituent characteristics; and meteorological variables including wind speed and direction, atmospheric stability, mixing depth, and precipitation characteristics. (CH)

<77>
Hedemann, J.P., Comparison of Mathematical Models for Calculation of External Gamma Doses Originating from Releases of Radioactivity to the Atmosphere.

1974, July. RISO-M-1726; 176 p. (Danish Atomic Energy Commission, Riscoe Research Establishment)

A brief description is given of different mathematical models for estimating external gamma doses from a continuous plume of radioactive gases (iodines and noble gases) released to the atmosphere from a nuclear plant, and the uncertainties of such estimates are discussed. Gamma doses from unit releases of fourteen noble gases and iodines were calculated from these models and compared with the doses calculated from the computer program GDOS, which is an implementation of a gamma dose model developed at Riso. Good agreement was found between GDOS and seven other models, one of which was experimentally verified within a downwind distance of a few kilometres. Estimated gamma doses from unit releases of fourteen noble gases and iodines for two weather categories (Pasquil types D and F), and with three different release heights (0.24 and 100 meters) calculated with four of the models (including GDOS), are given tabularly and graphically for values of downwind distances from 0.1 to 5.0 kilometres. (Auth)

<78>
Niehaus, F., Long-Term Aspects of the Environmental Burden from Energy Production: CO2 and H3.

<78> CONT.
1975, February. Ph. D. Thesis; 167 p.
(Technische Hochschule, Aachen, West Germany)

Carbon dioxide and tritium emissions from fossil or nuclear power plants were studied. The analysis for possible effects of alternative strategies to meet future energy demand on the temperature of the atmosphere depends on the mathematical model of the global carbon dioxide cycle which uses nonlinear differential equations. As there have to be substitutions of fossil power plants by nuclear facilities another model was constructed to simulate the water cycle of the northern and southern hemispheres by which it is possible to compute the effects of tritium emissions in the future. (Auth)

Extensive bibliography contains 149 references

<79>
Artemova, N.E., Methods for the Calculation and Prediction of Contamination of Close-to-Ground Air Layers by Stack Emission of Nuclear Power Plants.

1974, May. Archiv fuer Energiewirtschaft, 28(10), 503-512 (Not given)

Two fast methods are described which are suitable for the practical calculation of air pollution, the Setton-Pasquill method introduced by the British Atomic Energy Commission and a Russian model to calculate the air pollution due to industry. The advantages and disadvantages of both methods are comparatively explained with experimental data. (GE)

<80>
Ahlstrom, S.W., and H.P. Foote, Transport Modeling in the Environment Using the Discrete-Parcel-Random-Walk Approach.

1976, April. Part of Proceedings of the Environmental Protection Agency Conference on Modeling, Cincinnati, OH, April 1976, 5 p. (Battelle, Pacific Northwest Laboratories, Richland, WA)

When formulating a mathematical model for simulating transport processes in the environment, the system of interest can be viewed as a continuum of matter and energy or as a large set of small discrete parcels of mass and energy. The latter approach is used in the formualtion of the Discrete-Parcel-Random-Walk (DPRW) Transport Model. Each parcel has associated with it a set of spatial coordinates as well as a set of discrete quantities of mass and energy. A parcel's movement is assumed to be independent of any other parcel in the system. A Lagrangian scheme is used for computing the parcel advection and a Markov random walk concept is used for simulating the parcel diffusion and dispersion. The DPRW technique is not subject to numerical dispersion and it can be applied to three-dimensional cases with only a linear increase in computation time. A wide variety of complex source/sink terms can be included in the model with relative ease. Examples of the model's application in the areas of oil spill drift forecasting, coastal power plant effluent analysis, and solute transport in groundwater systems are presented. (Auth)

Communication with one of the authors revealed that there are actually two distinct codes, i.e., one for ground water transport (DPRWGW) and one for surface water transport (DPRWCR). The last two letters stand for "coastal reactors," the original application of the code. These codes are written in FORTRAN IV for a PDP11/45 computer. Core requirements are 62K bytes. The codes are overlayed. Data requirements are 20 million words. Data are segmented and stored on disk.

<81>
Shih, C.S., and E.F. Gloyna, Radioactivity Transport in Water - Mathematical Model for the Transport of Radionuclides.

1967, June. EHE-04-6702; CRWR-18; 179 p. (Center for Research in Water Resources, University of Texas at Austin, Austin, TX 78712)

The purpose of this study was to develop a mathematical model which describes the transport of radionuclides injected instantaneously in a stream. Particular emphasis was directed to the influence of sediments on transport. Instantaneous release of dye and continuous release of Sr85, respectively, were used to measure the dispersion and mass transfer coefficients. Aquaria and model river experiments were undertaken to determine various parameters which define the mechanism of sorption and desorption of radionuclides by sediments. Instantaneous injection of Sr85 into the model river provided data for establishing the relationship between the analytical solution and a proposed mathematical model. Fortran programs were designed for the analyses of gamma spectra, dispersion coefficients, and transport equations. Some of the important factors affecting the transport of Sr85 in the model river were studied. The mass transfer coefficient was found to increase with increased velocity. High uptake of Sr85 by sediments may be provided by increased temperature and the presence of an organic pollutant. An expotential relationship was established between the Sherwood's number and the Reynold's number for the model river. (Auth)

This code may be applied to accidental or routine releases. It was written in FORTRAN 63 for a CDC 1604 computer.

<82>
Binford, F.T., T.P. Hamrick, and B.H. Cope, Some Techniques for Estimating the Results of the Emission of Radioactive Effluent from ORNL Stacks.

1970, October. ORNL-TM-3187; 171 p. (Oak Ridge National Laboratory, Oak Ridge, TN 37830)

The basic purpose of this report is to delineate a procedure for determining (a) the maximum possible release rates for each of the five stacks at ORNL under normal operating conditions, and (b) the consequences of the inadvertent release of a relatively large quantity of radioactive material from one of the stacks. For routine releases a Gaussian dispersion model is used with the joint wind rise stability category climatology for ORNL to calculate the annual average stack dilution factor for each stack and for each of 16 wind directions. This is done by the compouter program STAREL. The information is combined with the stack release rates to insure that the annual average ground-level concentration due to all stacks at the controlled area boundry does not exceed the maximum permissable concentration. Holland's formula for plume rise is used to correct the stack height for the initial velocity of the stack effluent and corrections are applied for the

<82> CONT.
presence of an inversion. For accidental releases the release is of a short duration and the meteorological conditions are assumed to be known and to remain constant throughout the release. The program RELISH uses the Gaussian dispersion model to calculate (a) the ground-level stack-dilution factors along the plume centerline, and (b) the coordinates of the isopleths of constant, given stack-dilution factors. Calculations are compared with measurements at 9 sampling points around ORNL. (CWM)

<83>
Wenzel, D.R., A Computer Code to Calculate Doses Resulting from Releases of Radionuclidees to the Atmosphere.

1974, July. Part of Proceedings of the 19th Annual Meeting of the Health Physics Society, held in Houston, Texas, July 7-11, 1974

The Radiological Safety Analysis Computer Code (RSAC) is used to calculate the radiological consequences of the release of radionuclides to the atmosphere. A revised version of the code, RSAC-2, can be used to calculate (1) fission product inventory source terms, (2) lung doses using either the clearance model developed by the ICRP Task Group on Lung Dynamics or the ICRP Committee II model, (3) gastrointestinal tract doses using either the model developed by Dolphin and Eve or the ICRP Committee II model, (4) doses to various internal organs using recommendations of the ICRP Committee II, (5) ground deposition-gamma doses, and (6) cloud-gamma doses using either a "finite" cloud or the "semi-infinite" cloud model. RSAC-2 is a versatile code written in FORTRAN IV for the IBM 360/75 computer. Significant features and typical results of the code are presented. (Auth)

<84>
Sato, K., Y. Ito, I. Iyori, and A. Yamaji, A Program for Computing External Dose Rate from a Radioactive Cloud and Numerical Examples for Nuclear Ships.

1968, March. Senpaku Gijutsu Kenkyujo, 5(2), 55-63 (Not given)

A program for calculating the external dose rate from a radioactive cloud was written in 06 NARC language (analogous to FORTRAN). The basic equation used for the atmospheric dispersion is equivalent to the so-called generalized Gaussian plume formula with the stack and the inversion lid modifications. The numerical integration was performed using the Legendre-Gauss quadrature technique together with the polar coordinate system developed by D.S. Duncan. The time required for the integration was found around forty minutes on the NEAC-2206 computer. The typical results from the program are shown in graphical from putting emphasis on the safety of the persons in the vicinity of a nuclear ship. To help the discussion, the results from the usual method of evaluation based on the assumption of infinitely spread cloud are also presented. Thus, the great safety factors involved in the conventional analysis are easily estimated as functions of downwind distance. For example, the actual finite cloud dose for the meteorological category F is lower than that computed for the infinite cloud by a factor of approximately 70 at 100m downwind distance, and by a factor of approximately 2 even as far from the stack as 10 km. Another important result of the calculation is the remarkable discrepancy between the isodose and the isoconcentration lines at distances close to the stack. It must be remembered, however, that the basic equation for the region was made by extrapolating the available data. For this reason, the results may be considered as a measure for estimating the safety factors involved in the conventional analysis. (Auth)

<85>
Nowicki, K., Estimation of Population Doses from a Nuclear Power Plant During Normal Operation.

1975, July. RISO-M-1808; 102 p. (Danish Atomic Energy Commission, Risoe Research Establishment)

A model is presented for estimation of the potential air submersion and inhalation radiation doses to people located within a distance of 1000 km from a nuclear power plant during normal operation. The model was used to calculate these doses for people living 200 to 1000 km from a hypothetical nuclear power facility sited near the geographical center of Denmark. Two sources of radioactivity considered include the unit release of 15 isotopes of noble gases and iodines and liquid effluent releases from two types of 1000 MWe Light Water Power Reactors (PWR and BWR). Parameter variations were made and analyzed in order to obtain a better understanding of the mechanisms of the model. Atmospheric dispersion was calculated using a Gaussian plume model corrected for dry deposition, wet deposition, and radioactive decay. (Auth) (CWM)

This document is based to a large extent on a number of USAEC Reports and models, such as HERMES.

<86>
Voelz, E., Prediction Methods of Concentration Fields of Radioisotopes and Radiation Doses in the Environment of Nuclear Power Plants with Extraordinary Emission.

1973, April. ATS-TULL-1073; 93 p. (Technische University of Hannover, Germany)

The radiation risks from radioisotopes from nuclear power plant smoke stacks and leaks are estimated. Currently used methods and formulas are listed and discussed. The formulas are evaluated in form of diagrams concerning the direct radiation out of the reactor, the outer beta and gamma radiation from the exhaust air plume, the exposure due to the inhalation of contaminated air, and the radiation from radioactive deposits. (GE)

<87>
Hladky, E., I. Kubik, and J. Moraveck, General Model for Evaluating Transport of Radioactive Products Released to Environment and its Consequences.

Not given. INIS-MF-887; p. 204-209 (Vyskumny Ustav Energeticky, Jaslouske Bohunice, Czechoslovakia)

A mathematical model is described for the evaluation of the transport and behavior of radioactive products released into the environment during the operation of nuclear power plants. It is based on the behavior of the products on the main protective barriers of nuclear power plants. A block diagram is presented of the model and the respective main programs are characterized. (INIS)

<88>
Anno, G.H., P.J. DeBois, T.P. Wilcox, and J.K. Witthaus, RISC: Radiological Inhalation Safety Code for IBM 7090.

1963, June. Transactions of the American Nuclear Society, 6, 106-107 (Aerojet)

Radiological Inhalation Safety Code (RISC) is a digital machine program used to compute internal exposures (according to the methods of Morgan and Ford) to seven body organs or tissues (thyroid, bone, muscle, lungs, gastro-intestinal tract, soft body tissues, and testes), incurred over the first year following short-duration inhalation. Doses to these body organs are based on computations of up to 50 radionuclide sources. These computations include both radioactive and atmospheric source, strength determination, and the dose contributions from any of the 50 radionuclides to any or all of the above-mentioned body organs. RISC may be used to estimate the environmental consequences from reactor operations and accidental power excursions at stationary power plants, vehicle launch or test facilities, and from in-flight vehicles. The basic diffusion theory of O.G. Sutton is utilized in developing specific atmospheric diffusion models with the appropriate initial-volume-source corrections. The initial cloud size and height of rise are input parameters, along with isotope-release fractions, and downwind-dose points. The effects of a capping temperature inversion are determined in the program from a subroutine applicable for low-altitude releases. Also, the reduction of the cloud source material due to washout and dry fallout may be taken into account by exercising a seperate option. The radionuclide source terms for each of 20 specified positions downwind are also computed in a subroutine which considers specific fission and decay times. This part of the program also permits dose normalization so that the output is in terms of Rem/Mw-sec or Rem/Mw. (CWM)

<89>
Lee, E., R.J. Mack, and D.B. Sedgley, GADOSE and DOSET - Programs to Calculate Environmental Consequences of Radioactivity Release.

1966, April. GA-6511; 62 p. (General Atomic, San Diego, CA 92112)

GADOSE is a FORTRAN program which calculates the radioactivity in various reactor plant locations and the doses to the public resulting from instantaneous accidental release of activity into any of the plant spaces. The calculations can be performed for any combination of isotopes and include consideration of decay, buildup, filtration, atmospheric dilution, fallout, and rainout. The companion program DOSET is also available to include the effects of a time dependent accidental fission product release into the various plant spaces. A typical calculation of the activities and doses at five times and five distances from the plant with 134 isotopes requires approximately 4 centihours of IBM 7044 machine time for the basic GADOSE program. The program was written specifically for the High-Temperature Gas-cooled Reactor (HTGR) type of plant. (Auth)

<90>
Couchman, M.L., A.W. DeAgazio, and Y.S. Kim, NURSE-1, A Nuclear Rocket Safety Evaluation Code for the Control Data 3600.

1964, December. NUS-180; 282 p. (Nuclear Utility Services, Inc., Washington, DC)

The NURSE-1 Code evaluates the radiation hazards resulting from the rapid release of fission products from a nucler rocket engine. NURSE-1 is programmed in FORTRAN 3600 for use on a CONTROL DATA 3600. NURSE-1 requires 64K memory. Six tapes are used in addition to those required by the SCOPE Monitor System. A library tape containing nuclear and atmospheric diffusion data is necessary for the operation of the code. The library tape generation routine and the library data are described in Appendix E. The program determines several different doses at positions down-and cross-wind from the point of the excursion. This program considers only a release occurring in the lower atmosphere (on or near the ground). The code has several options which permit selection of the kinds of doses to be calculated. The models and parameters employed are believed to represent the best information available at the time of preparation. It is anticipated that the models and parameters may change as the nuclear rocket program matures and the NURSE-1 code is designed for ready replacement of models or parameters as better information becomes available. (Auth) (CWM)

While this old code was written for the nuclear rocket program, it could be used for any type of near-ground accidental release. (CWM)

<91>
Kim, Y.S., NUS Effective Energy Program.

1963, October. NUS-156; 14 p. (Nuclear Utility Services, Inc., Washington, DC)

The NUS Program for Effective Energy is an IBM-7090 Computer program written for the purpose of obtaining the effective energies absorbed by internal body organs due to disintegration of radioactive fission products inhaled and ingested by a human being. The internal body organs considered in the program are thyroid, bone, gastrointestinal (G.I.) tract, and lung. The results are presented in units of Mev per disintegration of the parent nuclide in a given chain. The program considers a fission product decay chain, examines whether the chain elements have affinity with a particular body organ, and computes effective energies. The average beta and total gamma energies, the radioactive decay constant, and the biological decay constants for each elements of a given chain must be provided as the input to the program. (Auth)

<92>
Heffter, J.L., A.D. Taylor, and G.J. Ferber, A Regional-Continental Scale transport, Diffusion, and Deposition Model.

1975, June. ERL ARL-50; 28 p. (Air Resources Laboratories, Silver Spring, MD)

The Air Resources Laboratories has developed a computerized post-facto trajectory model intended primarily for calculating the transport, diffusion, and deposition of effluents on regional and continental scales. A month, season, or year of trajectories at 6-hourly time intervals may be calculated forward or backward in time from any origin in the Northern Hemisphere for durations up to 10 days. Trajectory computations use winds at any altitude above sea level or winds averaged through any desired layer above average terrain. Computer output includes a listing

<92> CONT.

of individual trajectory and points after selected durations, plotted trajectories for any desired map scale on either Mercator or Polar Stereographic projections, and plotted maps showing percent frequency of trajectory traverses over grid squares. A Gaussian plume model is combined with the trajectory model to calculate long-term (monthly, seasonal, annual) mean ground-level air concentrations and deposition amounts. Both wet and dry deposition are incorporated in the model. Another model calculates short-term (once or twice a day) mean ground-level air concentrations for selected sampling stations. Computer output includes plotted maps showing long-term concentrations and deposition amounts, and tables of short-term concentrations where the contributions from individual plumes are identified. (Auth)

<93>

Dotson, W.L., and J.F. Fletcher, Mathematical Aspects of the Year 2000 Radiological Study.

1975, April. CONF-750413 (Vol. 2); Part of Proceedings of the Conference on Computational Methods in Nuclear Engineering, held at Charleston, SC, April 15-17, 1975, p. 77-92 (Hanford Engineering Development Laboratory, Westinghouse Hanford Company, Richland, WA)

The Year 2000 Radiological Study, a series of regional evaluations of radiological dose from nuclear facility operation, is based on a series of complex, interlinked computer codes designated as the HERMES model. These codes are used to model the release, environmental transport, biological uptake, and resulting radiological dose to regional populations which result from operation of large numbers of nuclear facilities within a region. Characteristics of these computer codes, and their application to specific regional studies, are described. (Auth)

<94>

Armstrong, N.E., and E.F. Gloyna, Radioactivity Transport in Water. Numerical Solutions of Radionuclide Transport Equations and Role of Plants in SR-85 Transport.

1968, January. EHE-12-6703E; CRWR-23; ORO-14; 139 p. (Center for Research in Water Resources, University of Texas at Austin, Austin, TX 78712)

The role of plants in Sr85 transport in aquaria and a model river was examined, and a numerical solution to the dispersion equation with convection was derived and extended to include plant sorption. Using the macrophyte, VALLISNERIA AMERICANS Michanx, it was found that almost instantaneous equilibrium was reached with Sr85 in solution through adsorption followed by a slower and more complete uptake by absorption. A characteristic pattern of uptake and release was observed after instantaneous releases into aquaria and at stations downstream from the point of release in a model river. Under continuous release, uptake tended toward an equilibrium level which was a function of the amount of attached algae present. It was concluded for conditions used in this study that plants played a negligible role in Sr85 transport. The uptake of Sr85 by plants after instantaneous release could be modeled using a sorption-desorption equation. Instantaneous dye and Sr85 releases were made in the model river to determine the dispersion coefficient and tracer cloud velocity with the two station method using only a selected portion of the concentration-time curves. The cut-off point for this portion was the point where one-tenth the peak value was observed. Solutions of the numerical model using constants calculated in this manner gave better fit to the observed data than methods previously used. Constant release data could also be predicted with this model using revised initial conditions. A numerical solution to the equations relating dispersion to plant uptake was also derived but not extensively tested. It was concluded that numerical solutions hold more promise in this area than analytical solutions because of their versatility and ease of computation. (Auth)

This code may be applied to accidental or routine releases. It was written in FORTRAN 63 for a CDC 1604 computer

<95>

Veverka, O., V. Valenta, and K. Vlachovsky, METEO-1 A Programme for Calculations of Atmospheric Dispersion of Activities from Nuclear Power Reactors.

1975. ZJE-166; 30 p. (Skoda Works, Plzen, Czechoslovakia)

This paper gives a description of the programmes METEO-H (reactor-crash outflows) and METEO-N (operational outflow statistics). A complete list of the used fomulae for both cases based on the Pasquill-formulation of the problem is given. The following phenomena are enclosed: 1. Reactor-crash outflow. 2. Operational outflow statistics 3. Normal temperature gradient (decrease with height), inversion (increase with height) 4. Lofting 5. Fumigation. 6. High stack (h is greater than or equal to 2.5h sub b), h = stack height, h sub b = building height. 7. Short stack (h sub b is less than h is less than 2.3h sub b). 8. Leakage from the reactor building. 9. Two terms of decay chains for rare gases, one term for other isotopes (due to practical possibilities, but we are able to include general decay chains 10. Fall out and wash out deposition processes. 11. Statistical fluctuations of the wind direction for the case of reactor-crash outflow. For cases 6, 7, 8 a uniform system of formulae is used in contradiction to the American and German system. A short description of that method is given in this paper. The wind direction and the mean wind speed are assumed to be constant. Supposing an effective containment system the plume rise during the atmospheric transport due to the radiation-induced heat is not included. The generalization of the programmes including this effect will be published later. The output of the programmes are the surface concentrations and the total deposition intensities. The programmes METEO-H and METEO-N are linked to a system of programmes solving the transport of activities on a nuclear power establishment. The link of the METEO-programmes are the programmes DOSIS and INTERCORP (to be published). (Auth) (JTB)

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