TECHNICAL REPORT
W-6139-4 10-31-02

INTERNATIONAL MACCS USERS GROUP

Proceedings of the Fourth International MACCS Users Group Meeting

Principality of Monaco
September 6, 2002

Energy Sciences & Technology Department
Brookhaven National Laboratory, Brookhaven Science Associates
Upton, New York 11973-5000

BROOKHAVEN NATIONAL LABORATORY

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Contract No. DE-AC02-98CH10886

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency, contractor, or subcontractor thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency, contractor, or subcontractor thereof.

Proceedings of the Fourth International MACCS Users Group Meeting

Principality of Monaco
September 6, 2002

V. Mubayi, IMUG Coordinator

C. Conrad, Editor

December 2002

Energy Sciences and Technology Department
Brookhaven National Laboratory
Upton, NY 11973

October 2002

Prepared for
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555
Contract No. DE-AC02-98CH10886
JCN W-6139

ABSTRACT

This report contains technical papers presented at the Fourth International MACCS Users Group (IMUG) Meeting held September 6, 2002 at the Columbus Monaco Hotel in the Principality of Monaco. Six technical presentations were made at the meeting. The papers included proposed enhancements to the MACCS/MACCS2 codes to incorporate uncertainty analyses within a Windows platform for the code, application of MACCS2 to research reactor accidents, and applications of the COSYMA and OSCAAR probabilistic consequence codes to reactor and non-reactor facility accidents.

TABLE OF CONTENTS

Page

Abstract iii

Preface vii

Minutes 1

Meeting Agenda 3

Application of the COSYMA Code for Comparative Analysis of a Certain Accidental Release of Radioactivity
Petr Pecha and Emily Pechova, Czech Republic 5

Future Plans for MACCS
Nathan Bixler, USA 17

Linking Radionuclides Release and Source Terms to the MACCS2 Code, Coupled with the Results of PSA Levels 1, 2, and 3, and Interfaced with a New Algorithm of Nuclear Consequence Measurement
Bill Cho, USA 27

Uncertainty Analysis System Using MACCS2
Susumu Ryufuku, Mikihiro Oheda, Masato Terakado, Naohiro Kurosawa, and Susumu Mitake, Japan 47

OSCAAR Development and Applications
Toshimitsu Homma, Japan 57

Research Reactor Accidents: Analysis and Impacts
Ioannis Kollas, Greece 67

List of Participants 79

PREFACE

The International MACCS Users Group (IMUG) is a forum for the exchange of information and research experience among users of the MACCS consequence code. Membership in IMUG is open to all organizations that use the MACCS and MACCS2 codes and agree to share their research findings with other members on a non-proprietary basis. IMUG also works to provide user support as needed in obtaining and executing the code and interpreting its results, in facilitating exchange between MACCS code users located in various countries, and sharing experience between users of different consequence codes. The U.S. Nuclear Regulatory Commission (NRC) is the permanent chair of IMUG. Brookhaven National Laboratory (BNL) coordinates IMUG activities and organizes IMUG meetings.

The First IMUG Meeting was held at BNL, Upton, NY, USA November 8-10, 1994 and the Proceedings were published in BNL Technical Report W-6139, C. Conrad, Editor, November 1995. The Second IMUG Meeting was held jointly with the Third COSYMA Users Group (CUG) Meeting in Portorož, Slovenia September 16-19, 1996 as part of the larger Third Regional Meeting on Nuclear Energy in Central Europe sponsored by the Slovenian Nuclear Society and the European Nuclear Agency. The Proceedings of the joint Third CUG-Second IMUG Meeting were published in two volumes. Volume 1 contains the papers delivered in the CUG session and Volume 2 is devoted to the papers given in the IMUG session. The Third IMUG Meeting was held at BNL, Upton, NY, USA November 4-5, 1998. The Proceedings of the Third IMUG meeting were published in BNL Technical Report W-6139-3, 8/27/99.

The Fourth IMUG Meeting was held at the Columbus Monaco Hotel in the Principality of Monaco on September 6, 2002 and the papers presented at that meeting are included in these Proceedings.

Nine participants from six countries attended the Fourth IMUG Meeting. Six technical papers were presented at the meeting.

The meeting began with an introduction to IMUG and an outline of future plans for MACCS code development by the U.S. NRC Program Manager, the ex officio Chair of IMUG. This was followed by an introduction to the meeting and agenda by the IMUG coordinator. The technical papers presented at the meeting included development of a Windows platform for MACCS/MACCS2 and incorporation of uncertainty analyses in the code, application of MACCS2 to research reactor accidents, and applications of the COSYMA and OSCAAR probabilistic consequence codes to reactor and non-reactor facility accidents.

Following the technical session, a round-table discussion was held on the proposed enhancements to the MACCS/MACCS2 code and the role IMUG members could play in testing the enhanced code.

MINUTES

The Fourth International MACCS Users Group (IMUG) meeting was held at the Columbus Monaco Hotel in the Principality of Monaco on September 6, 2002. Nine participants from six countries attended the meeting. Following the technical papers, a round-table discussion took place among the participants on the proposed enhancements to the MACCS/MACCS2 code. The discussion was focused on the development of a more user-friendly PC Windows platform for the code and, in particular, on the incorporation of uncertainty distributions for several key variables in the code that impact the results. The following is a summary of the major issues that were discussed at the meeting:

1. Uncertainty analysis – How many samples should be run and what is the basis? This issue was discussed at length. Little or no justification was provided for any specific choice on the number of samples needed to achieve a certain level of confidence in the results. This issue may need to be explored further in order to provide guidance to the user. All of the participants use regression analysis to determine the importance of uncertain variables on results. The methods used are all linear.

2. The Japanese IMUG members belonging to the Visible Information Center, Inc. have made significant progress in developing a PC MACCS2 Windows version incorporating uncertainty analysis. Sandia National Laboratories in the US is also developing such a tool under US NRC sponsorship. Both the US and Japan demonstrated a Windows version of MACCS2 at the meeting. The Japanese have incorporated scatter plots as a tool for evaluating results. This option should be considered for the US development as well.

3. Meteorological sampling - The issue of the choice of technique for weather sampling was discussed at the meeting. Japan Atomic Energy Research Institute (JAERI) is using their own version of weather bin sampling, as described in a paper presented by T. Homma et al. at the meeting that is included in the proceedings. Analyses performed by the IMUG participant from Greece use cyclic sampling. All other countries are using bin sampling.

4. Several participants in the IMUG meeting volunteered to beta test the US Windows version of MACCS2 when it is available as expected in 2003.

Because of the expected developments discussed above in the Windows versions of MACCS2, there was a consensus among the participants that it would be useful to hold the next meeting in the Washington, DC area in conjunction with the October 2004 Nuclear Safety Research Conference.

MEETING AGENDA

The meeting began with a welcome to the participants and an outline of the NRC plans for future development and enhancement of the MACCS/MACCS2 code by Jocelyn Mitchell, the NRC Program Manager for MACCS. The agenda for the technical session was presented by Vinod Mubayi of BNL.

The first paper by Petr Pecha and Emily Pechova of the Institute of Information Theory and Automation and EnergoProjekt, Prague in the Czech Republic, was on the application of the COSYMA code and a locally developed consequence code, HAVAR, to a large break loss of coolant accident involving a release of radioactivity to the environment. The paper presented a comparison of the results of the two codes on various consequence measures as a function of distance from the point of release. Since the authors could not attend the meeting, the paper was presented by Vinod Mubayi, the meeting coordinator.

The second paper on Future Plans for MACCS was by Nathan Bixler of Sandia National Laboratories. This paper outlined the development of a Windows-based version called WinMACCS and discussed the correction of known bugs in the current version of MACCS2 and the incorporation of uncertainty distributions to determine sensitivity of output to uncertain input parameters.

The third paper by Bill Cho of the Illinois Department of Nuclear Safety was on developing an integrated approach to risk assessment for reactor accidents by linking Level 1, Level 2, and Level 3 PSA through the introduction of nuclear consequence measures that would help a plant develop a definable and logistical safety management program. The paper was presented by Vinod Mubayi on behalf of the author who could not attend the meeting.

The fourth paper by Susumu (Steve) Ryufuku and his collaborators at The Visible Information Center, Inc. of Japan, described the development of a PC Windows version of MACCS2 incorporating uncertainty analyses. The paper demonstrated the application of this version of the code to accidental releases from a nuclear fuel cycle facility in Japan.

The fifth paper by Toshimitsu Homma of the Japan Atomic Energy Research Institute dealt with the development and application of OSCAAR, a probabilistic consequence assessment code developed in Japan for use in reactor PSA. The paper described the verification and validation studies of OSCAAR, using data from past accidents at Chernobyl and Hanford, to identify the important variables that contribute the most to the uncertainty of the predicted outputs.

The sixth paper by Ioannis Kollas of the NCSR "Demokritos" Institute of Nuclear Technology and Radiation Protection in Greece focused on research reactor accidents and described the application of MACCS2 to estimate consequences of accidents at the Greek Research Reactor, located within the Athens metropolis. The results indicated that the design basis accident with the current operating schedule would lead to non-trivial but limited consequences; however, no early fatalities or injuries would be expected from any of the accidents.

Application of the COSYMA Code for Comparative Analysis of a Certain Accidental Releases of Radioactivity

P. Pecha - IITA - Institute of Information Theory and Automation, Prague, Czech Rep.
E-mail address: pecha@utia.cas.cz
E. Pechova - ENERGOPROJEKT Prague, Czech Rep., pechova@egp.cz

Introduction

This report mentions the continuous process of comparison of results for important scenarios of accidental radioactive discharges generated by codes HAVAR and COSYMA. The COSYMA code is significant European product developed under sponsorship of EC by several outstanding European institutes and deals with the nuclear accident consequence assessment methods. The code HAVAR is fully localised national code standardised for purposes of calculations in the field of nuclear safety and includes executive regulations of the State Office for Nuclear Safety which are valid and obligatory for the Czech Republic. The code accepts the Czech local conditions from point of view static geographical or demographic gridded data (real spatial distribution of elevation, land use type, roughness of the terrain, population according to age categories on the variable polar computational grid) and dynamic changes of weather conditions (stepwise segmented model including local precipitation), release intensity and food chain parameters. The code offers user-friendly interactive support both for input of data and direct graphical interpretation of results.

The analysis presented here also illustrates some latest results generated by means of system RODOS (Real-time On-line Decision Support system) the customisation of which is just running in IITA.

A little bit of history

Whenever the estimation of radiological consequences of activity releases has been carried out for purposes of safety reports or for other external use some additional procedures for verification of the generated results have been implemented. It usually insisted in a certain partial verification with regard to other commonly accepted computer codes. Even though the COSYMA code is designed predominantly for case of probabilistic calculations, its rich know-how appeared to be worthwhile for the partial deterministic calculations and successive comparative analysis. Then, since 1997 various verification runs enable to tune the local Czech codes and improve their algorithms and reliability.

The first results were published in [1] on the 4-th COSYMA Users Group Meeting in Prague where the application of PC-COSYMA code such a verification tool used in the stage of NPP design has been demonstrated. An audit of dispersion and deposition models of HAVAR system for Batch I and Batch II problems were accomplished and experience has been reflected in the process of local national codes development. The main result of [1] was verification of the local Czech ingestion model ENCONAN (originally developed in early 90's by dr. V. Kliment, NRPI Prague) versus COSYMA food chain model FARMLAND. The general ingestion dynamic model was adopted for average Czech conditions taking into account local consumption habits (dependence on season and age), agricultural production scheme, average agro-climatic conditions and phenological characteristics of the plants, feeding diets of animals, time delays during processing, transport and storage of foodstuffs and feedstuffs etc. Sensitivity analysis with regards to radioactive fallout in a certain Julian day within a year was carried out and good

consent with COSYMA summer ingestion has been found. At the same time the extremely high degree of conservatism introduced by the former static ingestion model was demonstrated.

The HAVAR code is developed for quick estimation of various scenarios of accidental radioactivity releases, for purposes of uncertainty analysis (several options of models are interactively offered to users), sensitivity studies and potential "worst case" studies. In all cases the accent on respect on local specific features (as detailed as possible site specific data, practice of national meteorological service, agricultural production practices and all corresponding dynamic parameters for food chain model etc.) and compliance with the new specific governmental regulations (the Czech Atomic Law) is emphasised. During the code development the new knowledge obtained also from the various comparative benchmarks is gradually absorbed and lead to the implementation of more sophisticated algorithms. It was realised for example on the effects of plume rise, near-standing building wake, washout and extension of diffusion formulas. The broad extensions were implemented into the former ingestion model ENCONAN.

Between COSYMA and HAVAR codes exist a certain differences both in the part of input data and range of solution. It emerged to be necessary to simplify the input data for the particular cases and adjust them according to the existing program limitations. We do believe the adjustment don't reduce the significance of comparison and fulfil the objectives of the process.

Since 1997 some successive activities of COSYMA application were carried out as its verification tool for important results generated for several real scenarios of estimation of radiological impact on population. It consisted in the continuous partial examinations of generated results which is described in the corresponding reports. Let us mention particularly the important scenarios related to the validation tasks no. 1 and no. 2 defined and issued by the Czech Accreditation Board. The submission to the accreditation procedure is obligatory for software used in the Czech Republic in the field of nuclear safety calculations. The results of the code HAVAR supported by the COSYMA calculations are widely commented and demonstrated in [2], part III. The same tasks analysed by the RODOS software is described in [4] and then the good consent with other products can be declared.

Continuous process of comparison

Recent requirements on radiological impact assessment analysis lead to the further code development and its extensions with objective to carry out more detailed sensitivity studies and uncertainty analysis. It calls for more profound comparison procedures which would replace the previous partial verification steps. Extensive estimation of radiological consequences of several types of accidental radioactive discharges from NPP have been carried out recently. It was related both for releases from the primary circuits into containment and postulated releases out of hermetic zone directly to atmosphere (locked rotor accident, control rod ejection accident, steam generator tube rupture, main steam line break accident on PWR, some kinds of loss-off-coolant accidents). On the basis of thermohydraulic analysis of the events and successive conservative solution of activity transport through primary and secondary circuits (effort for maximisation of the released activity) the final strength and dynamics of activity release were estimated. Further reduction into stepwise segmented form has been applied and then the segmented source terms of the releases has been generated and submitted for the successive radiological impact assessment. Among this scenarios for two of them the more detailed and profound verification with COSYMA calculations was accomplished:

- for scenario PIPE : rupture of instrumentation pipe out of hermetic zone for samples extraction from volume compensator

- for scenario MSC : guillotine main steam collector rupture such a representative scenario covering the main steam line break accidents

In both cases we were engaged also in the process of overall activity transport solution and corresponding release source terms generation. The results are in possession of the Czech Energetic Company. Another scenario connected with the Large Break Loss Of Coolant Accident (LB-LOCA) for a certain PWR has been postulated on the basis of available data in common technical references and experience, mainly associated with the design basis accident and severe accidents constructed in the RODOS system for training purposes. Successive overall comparison benchmark has been started and documented in [3] in the middle of 2001 year. The results were extended this year and the extraction of the interesting dependencies is documented here.

Comments on LB-LOCA accident scenario and its breef description

A large break loss-of-coolant-accident (LB-LOCA) results from a pipe rupture of the reactor cooling system of the primary circuit of reactor. The large break is defined as a rupture with a total cross-section area equal to or greater than 0.1 m^2. The limiting LB is postulated for double-ended cold leg guillotine process in a section of the main coolant piping . The sequence of events for LB-LOCA starts with depressurization of the reactor cooling system which results to decrease of saturation temperature and increase in containment pressure. An emergency safeguards function signal is generated and initiates the countermeasures for limitation of consequences of the accident such as reactor trip and borated water injection. The successive fuel rod cooling is provided by blowdown of the primary system inventory, injection of liquid primarily from the accumulators refilling the vessel and core and supplemented by pumped safety injection.

Rapid depressurization of the primary system accompanied by high mass flow rates through the broken loop could lead to nearly complete coolant inventory loss and core uncovery. The fuel cladding temperature rise to a maximum as core heat transfer degrades. The cladding temperature rise is terminated as flow reverses through the core. Continued operation of the emergency core cooling system pumps supplies water during long-term cooling and core temperatures are reduced to long-term steady state levels associated with dissipation of residual heat generation. Let us assumed that emergency countermeasures (including the operator emergency actions) are adjusted in order to ensure the acceptance criteria for the LOCA. Large break LOCA evaluation model, which generates the source term for calculations of the radiological consequences of the LB-LOCA, is referred to be based on the advanced thermal-hydraulic computer codes which are consistent with approach used for several European blocks.

Construction of source term for successive assessment of radiological consequences of the specified LB-LOCA

The aim of this report is not thorough description of solution of thermal-hydraulic transient processes and activity propagation. Let us mention only, that the balance of activity in the reactor core for initial reactor power and burnup is solved and the thermal-hydraulic analysis of the containment environment during the transients enables modelling of full-pressure approach. It takes into account the location of inner structures, inner sources of mass and energy due to outflow from the break, effect of the containment spray system, heat transfer on the walls and removal processes affecting the activity balance. The successive balance of activity in the containment environment is computed and no fuel melting is expected. Transport of activity from dehermetised fuel rod claddings ($\approx$ 1% of cladding failures is assumed due to transients in early stage of the accident) into the containment considers the deposition of the activity on the inner containment surfaces, washing of radionuclides by sprays and other activity removal

mechanisms from the containment atmosphere. Under the further assumption that the reactor core remains amenable to cooling during and after the break the main source of the final activity release into free atmosphere is based on the peak pressure technical specification containment leak rate for the first 24 hour.

The intensity of the nuclide releases has rather complicated time dependency. For purposes of the Gaussian plume segmented calculations the total time of release has been split to 4 segments, the release characteristics are assumed to be constant within each segment. Considerable role plays time integral of activity (integrated within each time segment) which represents stepwise function between particular segments. For purposes of comparison procedure the further simplification of input data has been applied in order to bring the values as closest as possible with regard to the limitations of the respective codes (and , simultaneously, to reduce the amount of necessary calculations corresponding to the objective of the validation procedure).

All activity released originally during about 21 000 sec with real release dynamic segmented into 4 stepwise segments mentioned above has been reduced to one segment of 4 800 sec duration. A set of 15 nuclides has been selected for calculation which represents more than 92% contribution to the total doses. Corresponding source term was generated (total release in Bq):

Table 1:

KR85M	**3.5300E+11**	**KR88**	**8.0000E+11**
RB88	**1.4400E+11**	**I131**	**1.2800E+11**
TE132	**2.8900E+08**	**I132**	**1.7100E+11**
I133	**2.5100E+11**	**XE133**	**4.0900E+12**
I134	**2.3700E+11**	**CS134**	**3.7000E+10**
I135	**2.5100E+11**	**XE135**	**1.0700E+12**
CS136	**1.6000E+10**	**CS137**	**1.8500E+10**
CS138	**1.8100E+11**		

Because of a certain limitation of the codes other input data has been simplified and adjusted as close as possible in order to use the code capability optimally. The negligible differences remained, but those are not expected to cause remarkable influence on comparison. Uniform terrain approach has been adopted and the basic input definitions adopted as:

source geometry :	**single point**
atm. stability category :	**F** (alternatively **D** with rain)
wind speed at 10m :	**1.0 m/s** (alt. **5m/s** for **D** cat.)
no. of windrose dir.:	**16**
no. of radial dist. :	**35** (from 0.666 to 100 km from the source)
terrain orography:	**flat**
terrain roughness:	**urban type**
dispersion σ_z , σ_y :	**exponential KFK-Júlich**
no. of time segments:	**1 segment**, total activity is released during 4800 s with constant intensity
release height:	**45 m**
thermal power of release :	**0 MW**
vertical velocity of release :	**0 m/s**
near-standing build.:	**50 m** (width), **44 m** (height)
precipitation:	**no** (alternatively **5mm/h** in the whole region for **D** cat.)
mixing height for F cat.:	**200 m** (**560 m** for **D** cat.)
release duration :	**4 800 s**
critical age group:	**adults**

Besides that, the other first class priority values for calculation had to be adjusted. As for dispersion, σ_z and σ_y according to KfK / Jülich empirical exponential formulas for urban rough terrain has been used for all codes involved (COSYMA, HAVAR, RODOS). A certain reduction into the box approximation has been applied far from the source (> 10 km). Similarly, for dry

deposition velocities v_g the following values (spatially constant) have been adjusted in each code (correspond to the grass land use type everywhere):

$v_g = 0.015$ m/s ..... dry deposition velocity for elemental iodine form
$v_g = 0.0015$ m/s ..... dry deposition velocity for aerosols
$v_g = 0.00015$ m/s ..... dry deposition velocity for organically bound iodine

Washout coefficient Λ (s^{-1}) represents the effect of precipitation scavenging and the rate of wet deposition on the ground. It depends on physical-chemical form *f* of airborne nuclides (aerosol, elemental iodine, organically bound) and precipitation intensity I (mm/h). COSYMA and RODOS use such a default the power law $\Lambda(f,I) = a(f) * I^{b(f)}$. HAVAR code uses liner expression $\Lambda(f,I) = const(f) * I$. Thanks to flexibility of the code HAVAR its source code could be changed for the comparison purposes and adjusted according to COSYMA defaults. Good agreement is illustrated on Fig. 2a. Nevertheless, the necessity to renew some obsolete data is evident (e.g. [5]-aerosol washing in relation to particle sizes)

For precise comparison procedure other criteria have to be fulfilled. From this point of view we must have on mind that the individual Gaussian dispersion model itself are not quite same (volume source model in PC COSYMA, segmented plume for HAVAR). A certain differences could appear in other semi-empirical formulas used for description of particular effects of plume rise, near-standing building wake, vertical wind speed profile, spatial discrimination of the polar grid etc. Some details were discussed in [1]. At this stage we have not adjusted the effect into the precise correspondence, partly because other additional changes in the source code would be necessary in a particular code.

Generation of results using COSYMA and HAVAR codes for simplified LB-LOCA

All results relate to the total activity releases according to Table 1. The basic driving variables for the successive calculation of radiation doses are presented and compared:

1. ***Near-ground activity concentrations of radionuclides in air and its time integrated values***
2. ***Activity deposited on the ground for each radionuclide***

Both variables are directly used for the doses calculation when (roughly speaking) multiplying them by corresponding dose conversion factors. The following pathways of irradiation are taken into account:

<u>*External irradiation*</u>**: cloudshine, groundshine**

<u>*Internal irradiation*</u>**: inhalation, inhalation of resuspended nuclides, ingestion**

To comply with necessary safety criteria, various kinds of doses are generated, mainly:

3. ***Effective dose for early phase of accident – during period 2 days (without ingestion)***
4. ***Effective dose for early phase of accident – during period 7 days (without ingestion)***
5. ***Effective dose during period of 1 year including the committed doses from internal activity intake***
6. ***Committed effective dose for late phase of accident – during period of 50 years (with ingestion)***

Some partial results related to the stability category F (Pasquill-Giffort notation) for the main driving variables of near-ground activity concentration in the air and deposition on the ground have been selected here in Fig. 1, 2, 3. The behaviour of aerosol (Cs137) and elemental (I131) physical-chemical forms of nuclides are illustrated on figures 1 and 2. The only exception regarding to stability category is displayed on Fig. 2 a, where category D is assumed with overall precipitation 5 mm/hour and wind velocity 5 m/s at meteorological measuring height 10m.

Fig.1: Time integrated near ground activity concentration in air
near-standing building 50 x 44 m, direction 1, flat terrain,
uniform landuse (grass) + roughness (0.20m), model KFK - urban

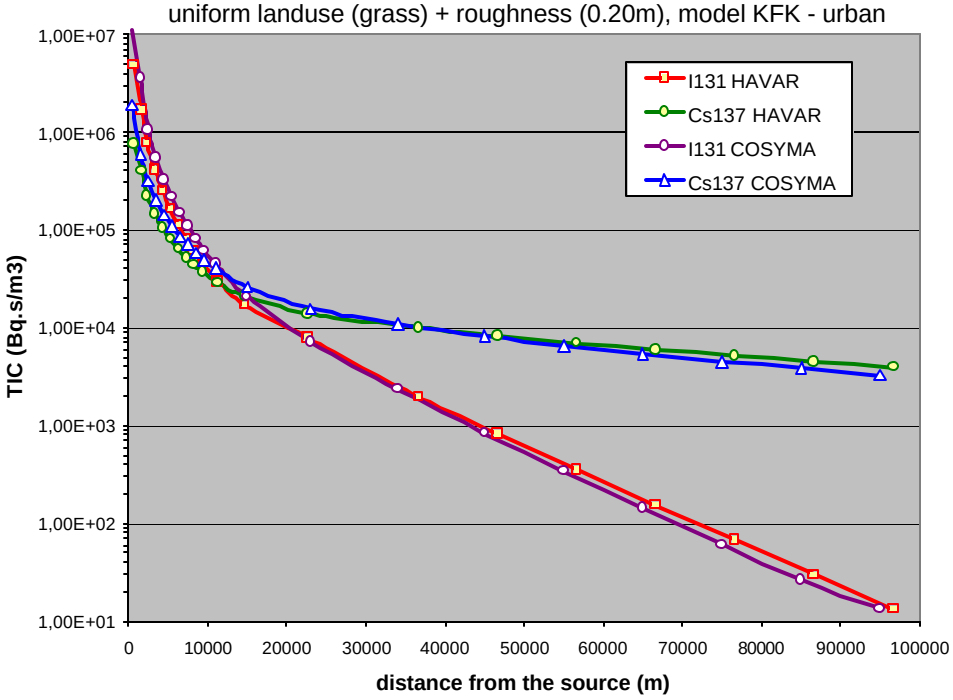


Fig.2: Activity concentration deposited on the ground
near-standing building 50 x 44 m, direction 1, flat terrain,
uniform landuse (grass) + roughness (0.20m); model KFK - urban

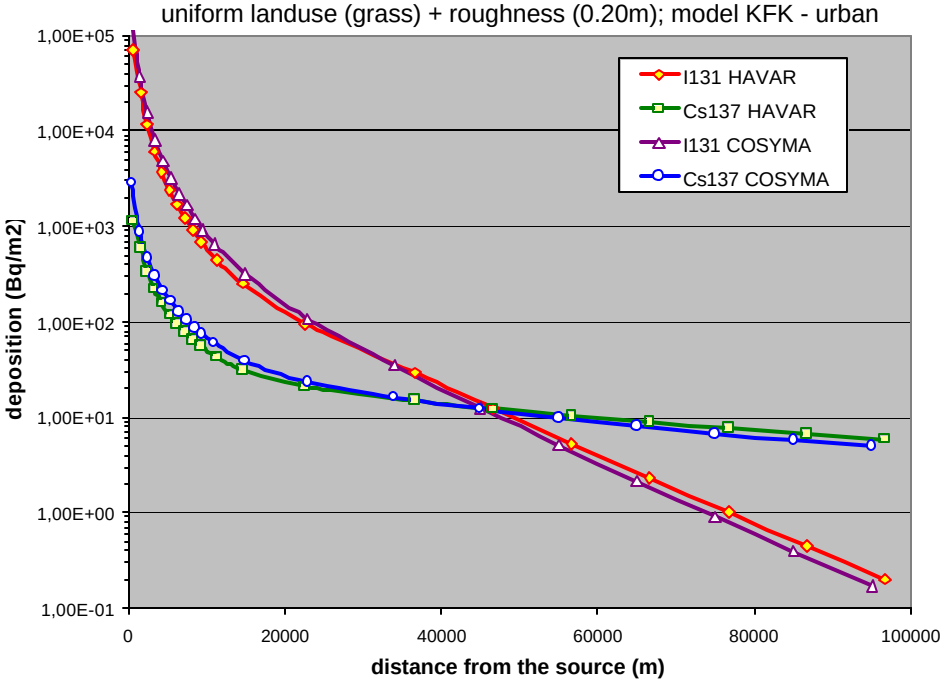


Fig.2a: Activity concentration deposited on the ground

the same as Fig.2; but category D, precipitation rate 5 mm/h, wind speed 5 m/s

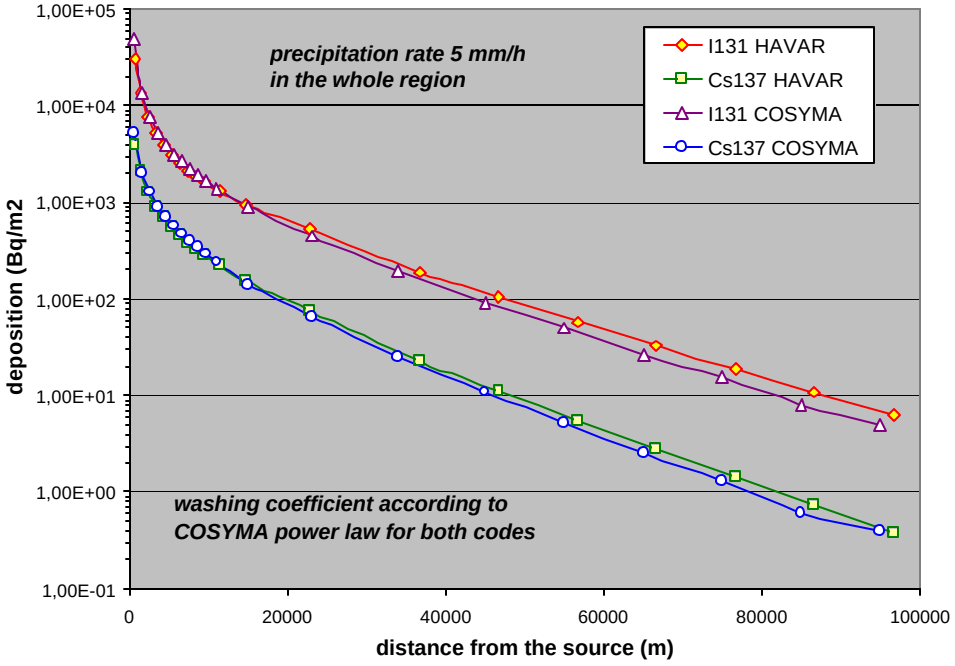
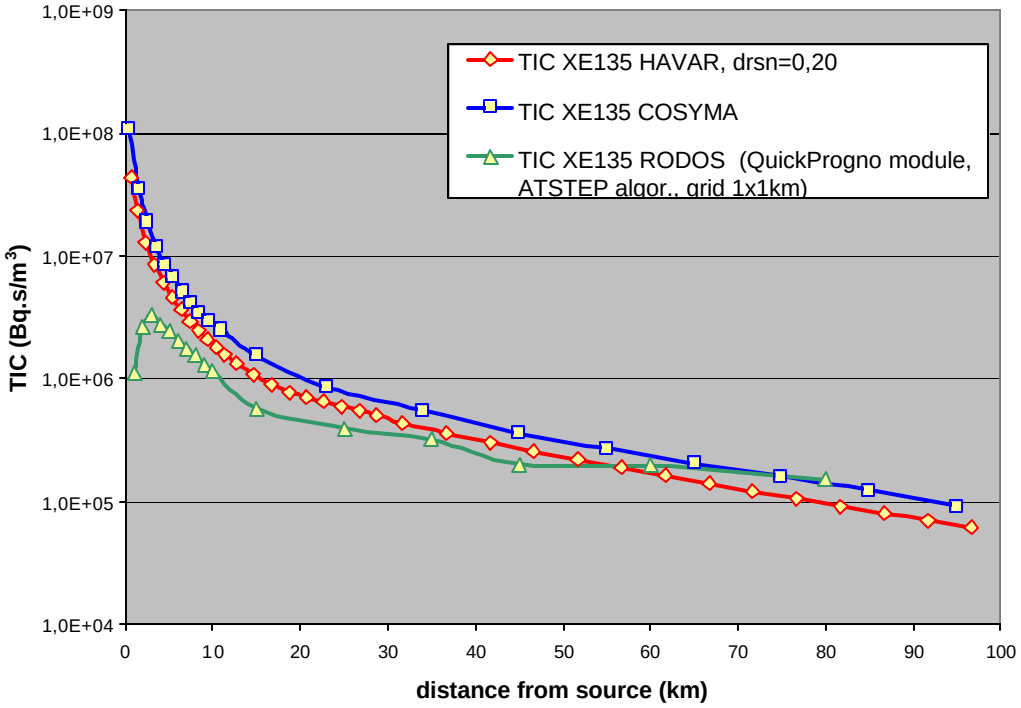


Fig.3: Comparison of COSYMA - HAVAR - RODOS results

Large break-LOCA scenario for 1 puff; release of 1.07E+12 Bq of Xe135

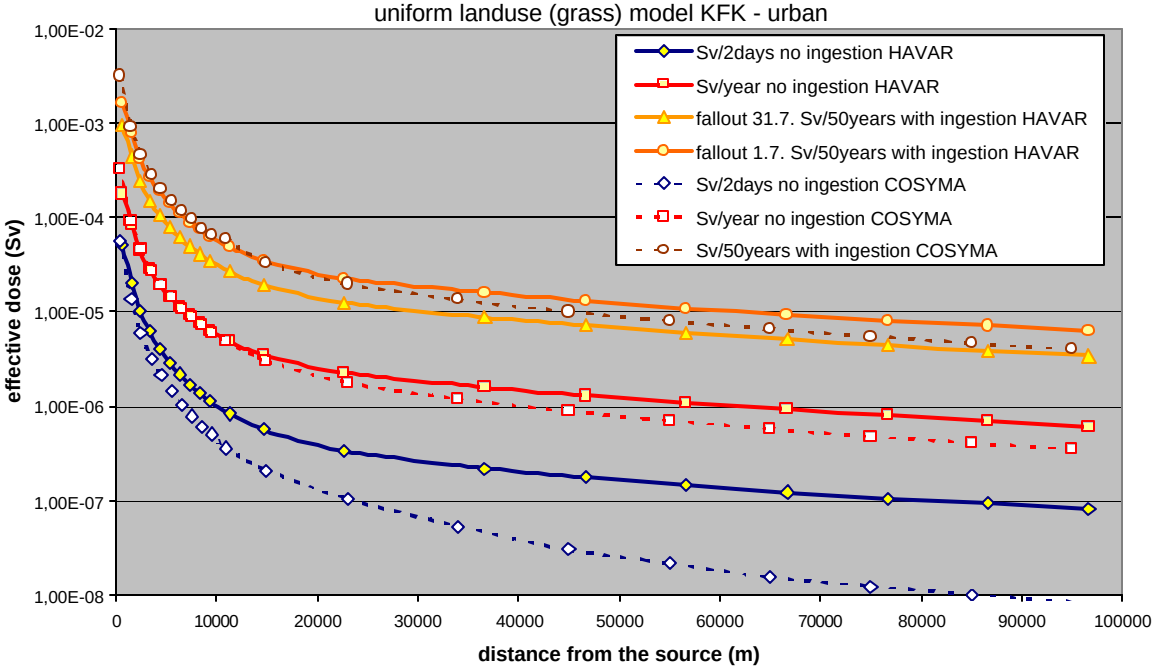
Near-ground time integrated activity concentration of Xe135 in air [Bq.s.m-3]



All values related to the COSYMA and HAVAR results mean the *maximum values under the plume axis* whilst the RODOS results are *averaged ones on the calculated tiles* (1km x 1 km on Fig. 3, 0.75 km x 0.75 km on Fig. 5). More detailed analysis (including the RODOS results) for noble gas (Xe135) is illustrated on Fig. 3. Let us remark that the RODOS results were taken away directly from the graphical output on display. Furthermore, its low values in vicinity of the source are caused by our rather rough selection of calculation grid 1 km by 1 km on which the ATSTEP RODOS algorithm generates the mean resulting values. To avoid misunderstanding we have to declare, that the advanced RODOS system provides much more precise and sophisticated options than are our simplifications mentioned on Fig. 3 and later on Fig. 5. The reader of this report is kindly asked to treat the RODOS results as a draft illustration. On the other hand the RODOS customisation in its complex UNIX environment requires enormous effort which for this case don't correspond to our illustrative aims.

Roughly speaking, the doses are calculated in common such a product of the respective main driving variable and the corresponding dose conversion factor. As the main driving variables are found out in good consent (Fig. 1, 2, 2a, 3), we should have to expect good consent of the doses. But more detailed analysis has revealed a certain discrepancy for doses from inhalation pathway. 2-days doses (from Fig.4) or 7-days doses (Fig.5) of the HAVAR results are evidently higher then the COSYMA results.

Fig.4: Effective dose for adults (Sv) - comparison HAVAR and COSYMA
near-standing building 50 x 44 m, direction 1, flat terrain,
uniform landuse (grass) model KFK - urban



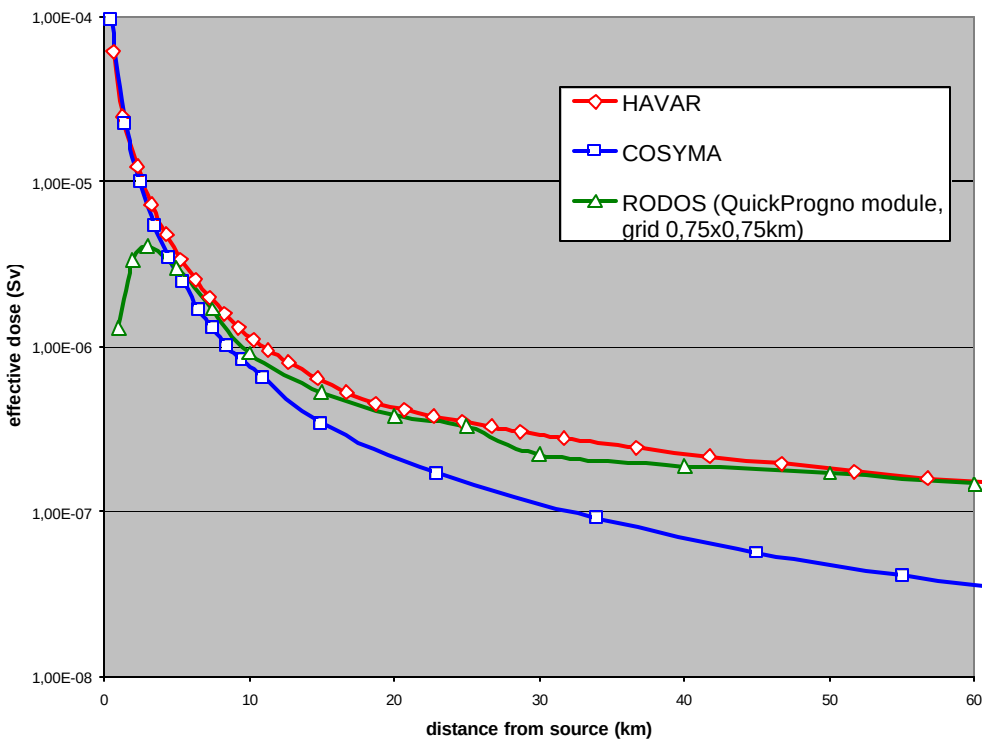
Explanation of this fact follows from the intentionally conservative way how the inhalation dose conversion factor for inhalation (Sv/Bq) are extracted. For HAVAR calculations the most conservative values are taken from the regulation no. 184 of State Office for Nuclear Safety. The dose conversion coefficients published here are based on metabolic models of the latest ICRP publications and different values are given for particular pathways of absorption of substances containing radionuclides in the body liquids. For I131, Cs134 and Cs137 the variability is shown in Table 2, where the conservative (maximum) HAVAR options are marked blue and bold:

Table 2:

nuclide	*Dose conversion coefficient for inhalation (Sv/Bq)*		
	fast	*medium*	*slow*
I131	**7.4E-9**	2.4E-9	1.6E-9
Cs134	6.6E-9	9.1E-9	**2.0E-8**
Cs137	4.6E-9	9.7E-9	**3.8E-8**

The corresponding COSYMA values for dose conversion factors for inhalation lie lower which was proved in [3] from the COSYMA output file *loca.lpt*. As expected (and illustrated on the Fig. 4 and 5), the differences are substantially mitigated for long term doses in the later stages of the accident. As for RODOS results from Fig. 5, the discrimination of the dose factor for inhalation according to lung clearance class (fast, medium, slow) is referred to be based on the NRPB database (1997) and could be extracted from *.../dfinh_nuclide* files in the RODOS DB.

Fig.5 : 7-days effective dose for adults (Sv)
no ingestion, uniform terrain, dispersion: rough terrain



Comment on ingestion calculations:

An audit of the ingestion model implemented into the HAVAR product in comparison with the FARMLAD ingestion model of COSYMA was done in [2]. It enables calculation of doses from internal intake of activity by ingestion for arbitrary Julian day of fallout of radionuclides in relation to vegetation periods, transport of nuclides through animals, processing and consumption delays and others. The basic scheme of the model is presented in [3]. It should be point out that:

- so far only local production and local consumption approach is adopted
- for consumption baskets which have to be split into age categories the averaged consumption data for England was used for COSYMA calculations and average Czech consumption rates was used for HAVAR; no profound consumption basket adjustment was done

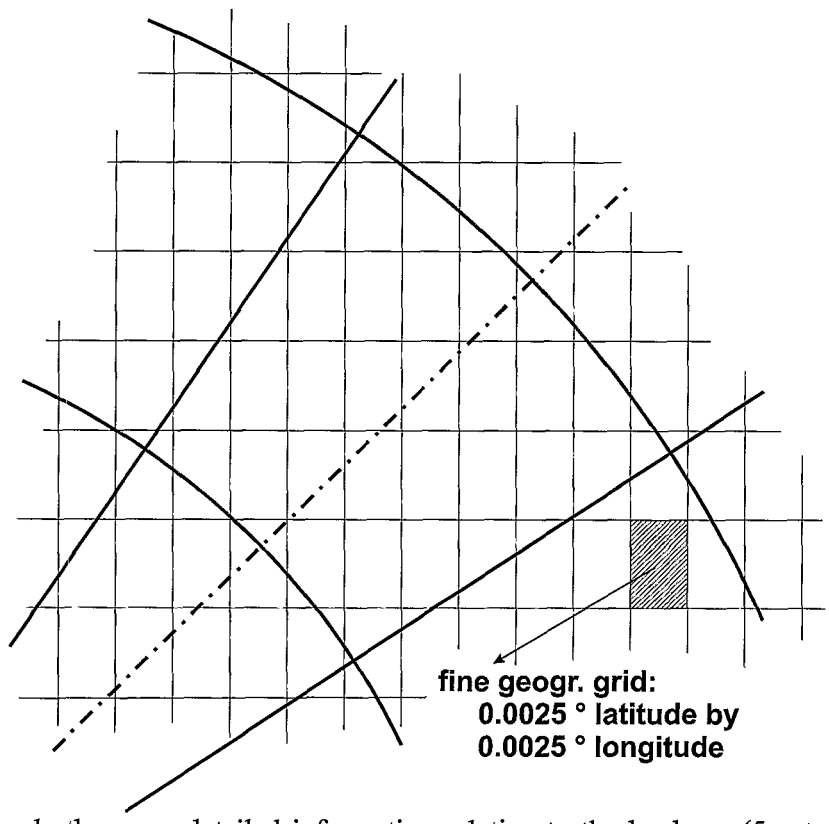
- with regard to the day of fallout, only winter/summer options are offered in COSYMA panels; HAVAR enables calculations with more detailed discrimination (see Fig. 4 - two curves for the day off fallout July 31 and alternatively July 1)

In common, we have to say that it is nearly impossible to adjust all input data to be the same for both codes. The possible differences are analysed during successive sensitivity studies. Nevertheless, the consent of results seems to be good also for doses in late stage of the accident as is evident from Fig. 4. We do not expect great differences caused by relatively small input data inconsistency.

Comment on common local geographical data platform

The further development has established close connection of inputs with the existing external fine gridded geographical data. The newly developed data preprocessor transforms for each NPP site the detailed available spatial information related to elevation, roughness and land use type (in absolute geographical coordinates) to the required calculating polar grid with variable cells (relatively to the NPP position). This way a common data platform is established among codes using geographic grid (e.g. RODOS) and other ones based on calculations in the polar coordinates (COSYMA, HAVAR, MACCS, ...). The process of mutual transformation is illustrated on the Fig. 6. Fine gridded data with high degree of resolution (0.0025 by 0.0025 of geographical degree for longitude by latitude – roughly 180m by 250 m) has been acquired and transformed to the polar coordinates relatively to the site position.

FIG.6: Reconstruction of data in polar coordinates from fine gridded data in geographic coordinates.



For example the more detailed information relating to the land use (5 categories: water, grass, agricultural, forest, urban) enables to generate the occurrence (%) of each category and then to calculate the mean weighted value of the dry deposition velocity v_g on each polar cell (on the basis of the maximum v_g for fully developed plant canopy recommended in the RODOS product

for each physical-chemical form of the radionuclide – aerosol, elemental, organic, noble gas). The value of the dry deposition coefficient is expressed as sum of partial dry depositions velocities weighted by their percentage occurrence on the respective polar section. Alternative way is to derive the prevalent type of the quantity within a polar sector.

Similar transformations were accomplished for spatial distribution of demographic information according to six age categories of population and a specific algorithm for agricultural production gridded data generation from mean district annual values is in progress.

Where we are?

Is our development redundant? Is usefulness of the local national code development questionable? We shall argue against from the point of view of:

- importance to respect the real spatial distribution of specific terrain characteristics around each individual nuclear facility
- ability to take into account local effects (abnormal local precipitation washout, effect of extensive forest areas)
- compliance with national practice: obligatory governmental regulations, agricultural production scheme, consumption chain, practice of national meteorological service
- flexibility in implementation of the latest knowledge (e.g. the forthcoming plans in substitution of obsolete models by latest recommendations [5]) : "alive" code with regular administration and maintenance enabling prompt implementation of improvements and changes
- ability to fulfil a special requirements of external users having character of "worst case" analysis or uncertainty and sensitivity studies (extensive computations of many variants by means of utilisation of rich interactive support of the code), sometimes requiring also source code temporary changes
- symbiosis in connection with other similar codes, which makes developer to be fully-fledged discussion partner, puts the technical people and their knowledge together and brings benefit to the research in the field

References

[1] Pechova E., Pecha P., Nedoma P. : *Application of PC COSYMA code such a verification tool used in stage of NPP design,* Proceedings of the 4-th COSYMA Users Group Meeting, Prague, Sept. 1997 (in English).

[2] Pecha P., Pechova E.: *HAVAR : An interactive code for estimation of radiological consequences of incidental releases of radionuclides to atmosphere from nuclear facility, Part I : Methodology, Part II: Users guide, Part III : Comparison of results with deterministic runs of COSYMA code and partial results of atmospheric module ATSTEP of RODOS system.* Tech. report of Energoprojekt Praha, EGP 4104-6-990010, final version Oct. 1999 (in Czech).

[3] Pecha P., Pechová E.: *Comparative benchmark of radiological consequences estimation for postulated LOCA Accident.* Research report IITA No. 2015, June 2001 (in English)

[4] Pecha P. Nedoma P. Karny M., Kuca P : *Status report on RODOS accreditation for its use in Czech Republic – Local Quality Assurance Process,* RODOS(WG1)-TN(98)-29, Dec. 1998, final version July, 99.(in English)

[5] ADM Liaison Committee: *Review of Deposition Velocity and Washout Coefficient & Flow and Dispersion in the Vicinity of Groups Buildings.* NRPB -R322, publ. on June 2001

Future Plans for MACCS2*

N. E. Bixler, K. L. McFadden, and J. E. Cash
Sandia National Laboratories, PO Box 5800
Albuquerque, NM 87185-0739
Email: nbixler@sandia.gov

Abstract

MACCS2 is a probabilistic, accident-consequence code that estimates the risks from offsite releases during the operation of nuclear installations based on the postulated frequencies and severities of potential accidents. Risks are quantified in terms of doses, predicted early fatalities and health effects, longer-term heath effects from cancer, and economic consequences. Sandia National Laboratories (SNL) are currently developing MACCS2 and a Windows-based front end, WinMACCS, for the United States Nuclear Regulatory Commission (USNRC).

Development of MACCS2 is focused on fixing known bugs, extending limits to allow more flexibility, use of more modern memory management capabilities, and enhancing compatibility with the front end. In addition, a pool-fire model has been added to treat dispersion from a bouyant, finite source. The possibility of extending and improving the economic-consequence model has been explored but not initiated at this time. Other areas for future development are being considered as well.

The objectives for the development of WinMACCS are as follows:

- Develop a Windows-based tool (as opposed to the current DOS-based tool),
- Facilitate and reduce errors in the process of creating or modifying an input file,
- Integrate the capability to assign and evaluate uncertain input values (in addition to the existing MACCS2 capability for uncertain meteorological data), and
- Provide tools for quickly evaluating results.

These objectives are being realized because of the following features of WinMACCS:

- Intuitive model selection using windows forms;
- Logical organization of the MACCS2 input variables;
- Color coding of input-variable categories to show which are needed and which have already been assigned;
- On-line description of the meaning, type, units, and limits of an input variable;
- Automated capability to assign a probability distribution to relevant variables;
- Automated capability to correlate uncertain input variables;
- Capability to export or plot data to facilitate evaluation of the results; and
- Automated regression analysis to determine which input variables most influence uncertainties in the predicted consequences.

* This work was supported by the US Nuclear Regulatory Commission and was performed at Sandia National Laboratories, which is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the US Department of Energy under Contract DE-AC04-94AL85000.

1. Introduction

The US Nuclear Regulatory Commission (NRC) has embarked upon a path on which prescriptive regulations are gradually being replaced by risk-informed regulations. To the present, risk has been quantified largely in terms of core damage frequency (CDF), as determined by Level-1 probabilistic risk assessments (PRAs) and, to a lesser extent, large-early-release frequency (LERF), as inferred from Level-2 PRAs. However, the expectation is that risk to the public, as defined in more normal terms of probability times consequences, will ultimately supplant core-damage frequency and the inferred LERF as the relevant metric. For this to happen, credible tools must be used to perform the Level-2 and Level-3 PRAs. The NRC has supported and continues to support MELCOR (Gauntt et al., 2000) and MACCS2 (Chanin and Young, 1998) as the tools of choice for supporting Level-2 and -3 PRAs, respectively.

MACCS2 estimates the human-health and economic risks from operating nuclear installations, based on the frequencies and severities of potential accidents. Those risk estimates provide one of many inputs for judgements on risk acceptability and on ways to reduce excess conservatism in the regulatory framework. They also enable comparisons to be made with quantitative safety objectives. Determination of the uncertainty associated with these risk estimates plays an important role in effective prioritization and allocation of research and development efforts toward reduction of risk and the appropriate use of the results of risk assessments in regulatory activities.

MACCS2 was originally developed as an input-file-driven, batch tool. While this type of tool has some advantages, the development of input files is often cumbersome and error prone. For these reasons and others, a Windows interface, called WinMACCS, is being developed. WinMACCS is intended to guide the user through the process of developing or modifying an input file, reduce the frequency of input errors, and more efficiently evaluate results. Perhaps more importantly, WinMACCS is intended to streamline the process of running multiple realizations with MACCS2 to evaluate the effect of uncertain input data.

2. MACCS2 Development

Current development of MACCS2 fits into four categories. The first is fixing known bugs and deficiencies in the most recently released version, 1.12. The second is extending limits and enhancing flexibility. The third is enhancing compatibility with the Windows interface, WinMACCS. Finally, a number of modeling improvements is also being considered. Each of these improvement categories is discussed in this section.

There are two known bugs in MACCS2 version 1.12, both of which have been reported to the MACCS2 user community in the form of defect notifications. These defects are connected with the following usages:

1. Invoking multiple source terms via change cards in the ATMOS input and
2. Invoking multiple emergency response scenarios in the EARLY input.

These defects have now been fixed in the current working version of MACCS2. Until a new version is released, MACCS2 users are advised to use workarounds to avoid these bugs.

Another issue that was more recently discovered is that MACCS2 version 1.12 allows cross-wind dispersion parameters to be specified that result in a plume that wraps

around the polar coordinate system multiple times. This problem arises from the power-law representation for crosswind dispersion:

$$\sigma_y = ax^b \qquad \text{Eq. 1}$$

when $b > 1$ and a and x are sufficiently large. The notation used in the MACCS2 user's manual is misleading in that it implies a rectangular coordinate system; whereas, a cylindrical coordinate system is actually used in the code to represent the dispersion process. Thus, Eq. 1 is more correctly expressed as

$$\sigma_\theta = ar^b \qquad \text{Eq. 2}$$

It is unusual for σ_θ to extend around the compass, although it is possible at very low wind speeds under very stable conditions. At such low wind speeds, the Gaussian plume assumptions are dubious. In fact, wind speeds less than 0.5 m/s are not allowed by MACCS2. It is obviously unphysical for σ_θ to extend around the compass multiple times. The current working version of MACCS2 checks the maximum σ_θ that can occur: if the width of the plume extends beyond about 180°, the calculation is terminated with an error message.

Several previous limits have been extended in the current working version. For example, the maximum allowable number of plume segments has been increased from 4 to 10. Also the maximum number of chemical groups has been increased from 10 to 20. Other limits may be extended if there is a need. This issue is discussed further in the context of dynamic memory allocation in a subsequent paragraph.

A significant set of modifications to the MACCS2 code has been made to create a tighter interface between MACCS2 and WinMACCS, which is discussed in the subsequent section. Most of these modifications to MACCS2 are transparent to the user. Information flow between WinMACCS and MACCS2 is primarily through files. WinMACCS creates one or more MACCS2 input files that are compatible with version 1.12 except that they contain tabs as delimiters (instead of spaces). The current working version of MACCS2 has been modified to accept tab characters as field delimiters. MACCS2 now creates a binary output file that is used to communicate results back to WinMACCS. This set of modifications to MACCS2 has no effect on predictions.

A number of improvements to the models in MACCS2 are being considered; one has actually been implemented, a pool fire model (Mills, 1987). This model is implemented as a correction to the plume rise model to account for a finite source (as opposed to a point source). The current implementation assumes that the ultimate rise height is independent of source diameter.

Verification and validation of the pool-fire model has lead to the discovery that the currently implemented model for ultimate plume rise height (Briggs, 1975) appears to overpredict the rise for stability classes A through D. Since ultimate rise height is assumed to be independent of the source diameter (i.e., it is identical for a point source and for a finite-diameter pool fire), this observation extends to calculations that were done before the Mills model was implemented. A modified model for ultimate rise height

based on earlier work by Briggs (1971, 1972) may be implemented as the default model in a future version. The present model will be retained as an option to allow the user to make runs that are comparable to previous work.

Other improvements to the MACCS2 models that are being considered as user options, but not yet implemented, are the following:

1. a change in the treatment of atmospheric dispersion beyond 30 km,
2. a model to account for administration of potassium iodide to diminish health consequences from inhalation of radioactive iodine, and
3. a model to accommodate changes in mixing height during plume transport.

Finally, comparing MACCS2 with a more sophisticated atmospheric transport code to determine whether the simple Gaussian plume model is adequate for Level 3 PRA studies is being considered.

An additional possibility is to restructure MACCS2 to use dynamic memory allocation, a feature that is supported by the newer FORTRAN compilers. Dynamic memory allocation would allow complete flexibility in selecting numbers of input parameters or requests for results. For example, the current extension of the maximum number of plume segments to 10 would become obsolete. The user could choose any number of plume segments, and the code would automatically be dimensioned to that number.

3. WinMACCS Development

MACCS2 version 1.12 was originally created to be a batch-mode tool. As a result, MACCS2 has a number of deficiencies that make it less user-friendly and more cumbersome to use than many of the modern programs that were designed for the Windows environment. The current development of WinMACCS is intended to improve the usability of MACCS2 by creating a user interface between the user and the MACCS2 code. In particular, the interface allows the user to create or modify MACCS2 input files in a more intuitive and efficient manner, which should lead to fewer input errors. It also allows the user to evaluate results more efficiently. Finally, the user interface integrates a stratified Monte-Carlo sampling technique, know as Latin Hypercube Sampling (LHS) (Wyss and Jorgensen, 1998), with MACCS2 so that uncertain input parameters can be sampled and sensitivities evaluated.

The user begins by creating a new project or opening an existing one. The user must then make major decisions that affect the overall calculation, such as which modules to run (ATMOS, EARLY, and CHRONC), choices affecting meteorological sampling, etc. This is illustrated in Fig. 1. These major choices are contained on a *Properties* form. Once these major choices are made, the user is guided to make other required property assignments through a color-coding scheme, as illustrated in Fig. 2. Categories of input parameters are displayed as red hexagons, green checks, or gray hexagons or checks to show that they are needed but not yet assigned, needed and already assigned, or not needed in the current calculation, respectively. The gray checks and hexagons are used to indicate whether an unneeded variable has been set or not, respectively.

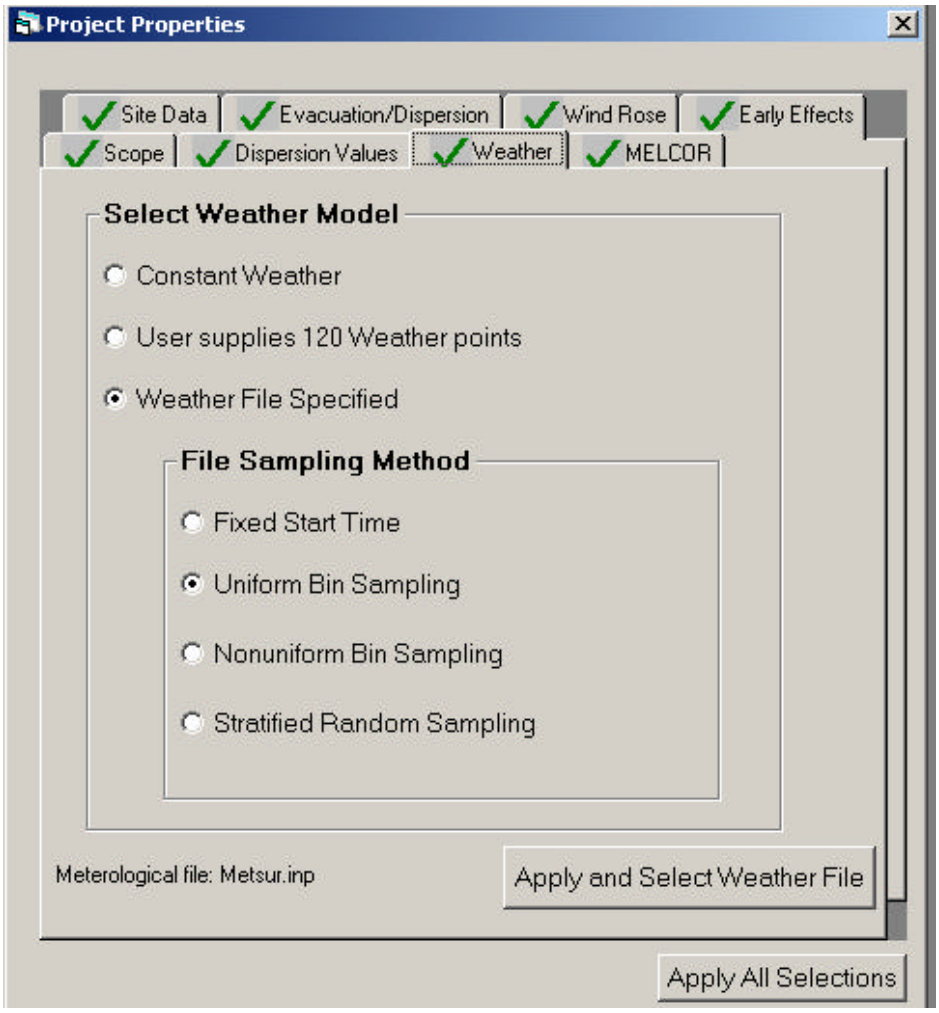


Figure 1. Properties screen used to make major choices affecting a calculation. Selection of weather model is illustrated.

Finally, orange hexagons are used to indicate optional parameters, such as flags to activate optional output.

The user interface contains a description for each input variable, as illustrated in Fig. 3. The description tells the user the type of variable (integer, real, logical, or list), the valid range, the units, and a definition of the variable. In the case illustrated, the user is further instructed that the distance variable must be monotonically increasing. This information should guide the user in assigning values to the input variables.

Many of the real-valued parameters can be assigned an uncertain distribution, as illustrated in Fig. 4. This allows sampling to be invoked via LHS. The user can specify which parameters should be uncertain and the number of realizations. The user can also specify that uncertain input variables are correlated, as illustrated in Fig. 5. The software automates the creation of the input files and allows the user to initiate the MACCS2 runs for the set of realizations that were created.

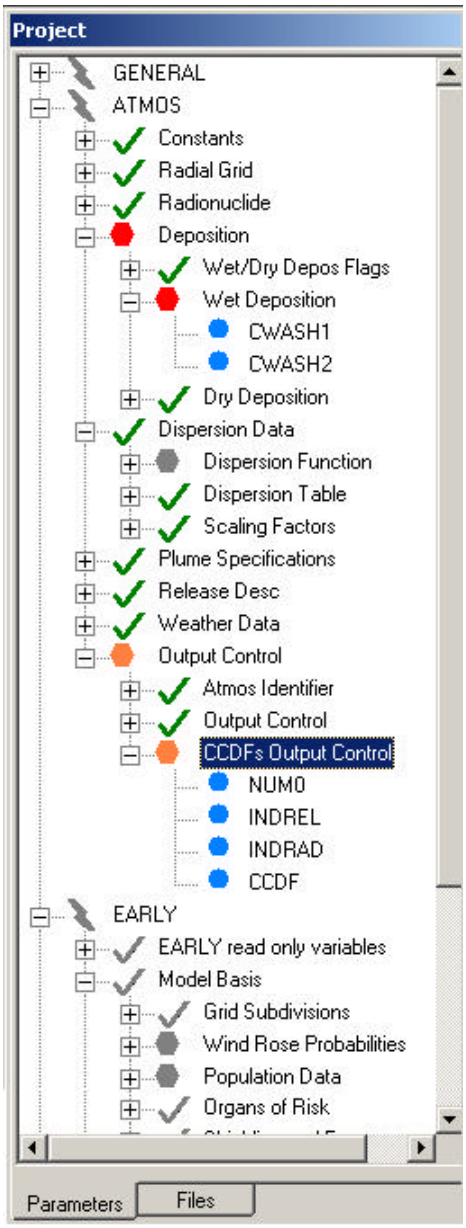
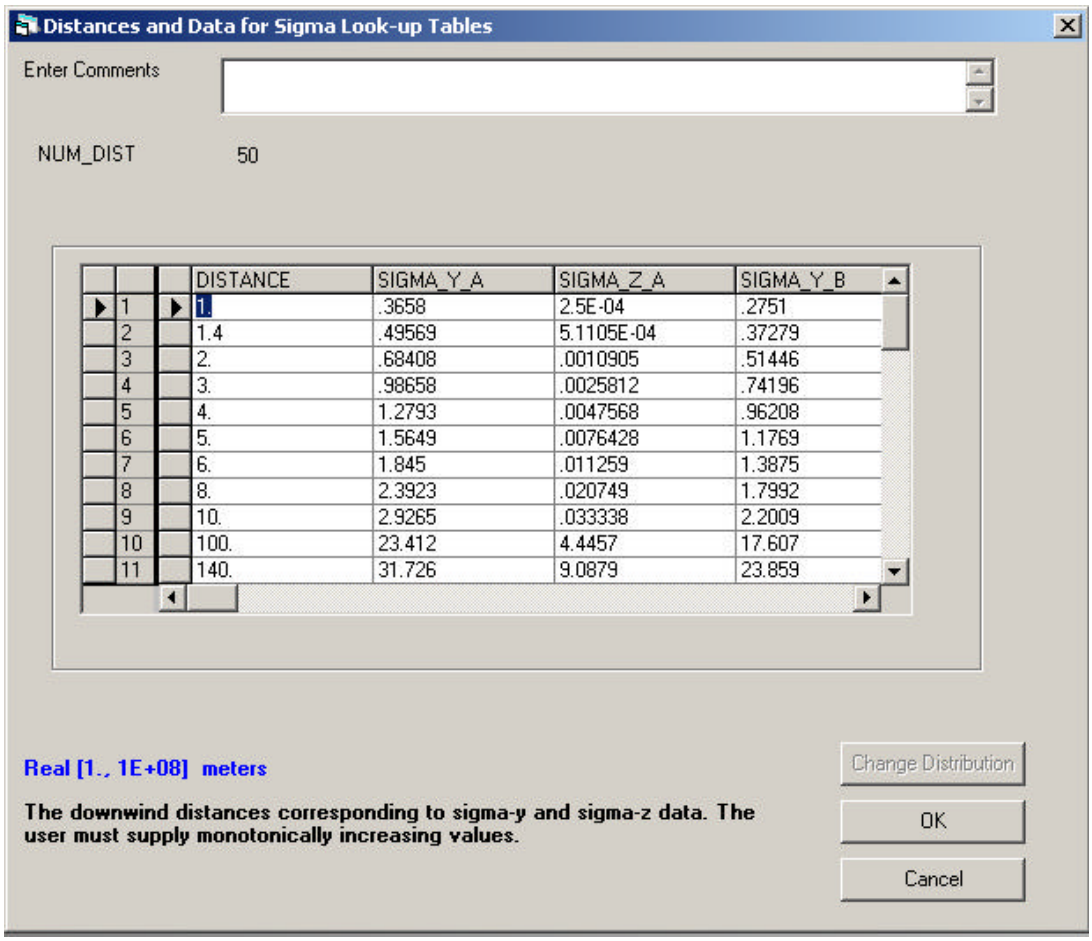


Figure 2. Screen displaying the categories of input data. The color-coding scheme is illustrated.

Whether a single deterministic run or a set of LHS runs is performed, the user can evaluate results quickly by plotting complementary-cumulative-distribution functions (CCDFs). Even individual executions of the MACCS2 code usually involve sampling of weather data (weather trials). CCDFs can be plotted for a deterministic run involving a set of weather trials. CCDFs can also be plotted for a weather quantile or mean for a set of LHS realizations. Finally, a plot containing a number of CCDFs, each representing the weather trials for a LHS realization, can be plotted for the set of realizations, as illustrated in Fig. 6.



	DISTANCE	SIGMA_Y_A	SIGMA_Z_A	SIGMA_Y_B
1	1.	.3658	2.5E-04	.2751
2	1.4	.49569	5.1105E-04	.37279
3	2.	.68408	.0010905	.51446
4	3.	.98658	.0025812	.74196
5	4.	1.2793	.0047568	.96208
6	5.	1.5649	.0076428	1.1769
7	6.	1.845	.011259	1.3875
8	8.	2.3923	.020749	1.7992
9	10.	2.9265	.033338	2.2009
10	100.	23.412	4.4457	17.607
11	140.	31.726	9.0879	23.859

Figure 3. Screen to input tables of atmospheric dispersion parameters. Illustrated is a description of the distance variable used in this table.

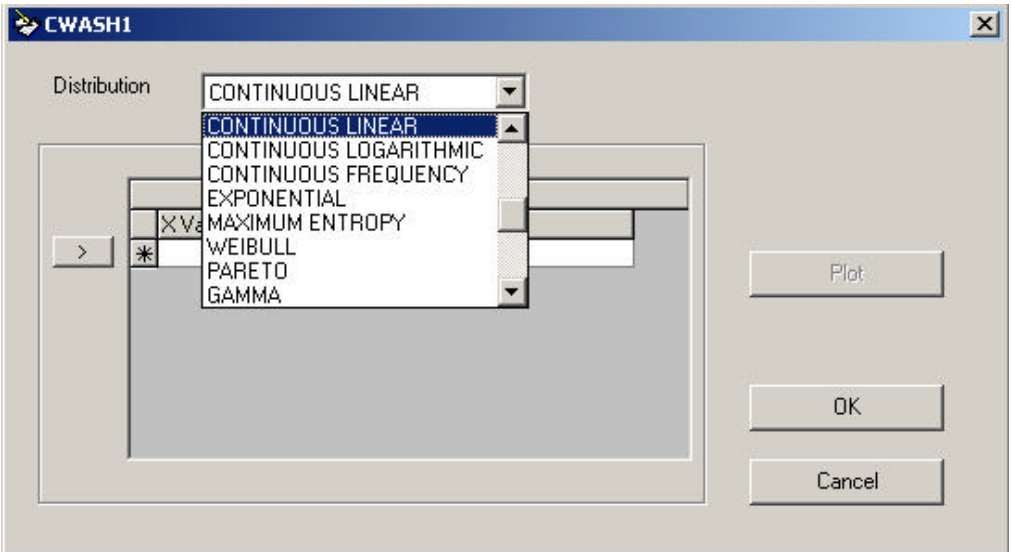


Figure 4. Screen used to assign an uncertain distribution for CWASH1. In the illustration, a continuous linear distribution is chosen from a large list of possible distributions.

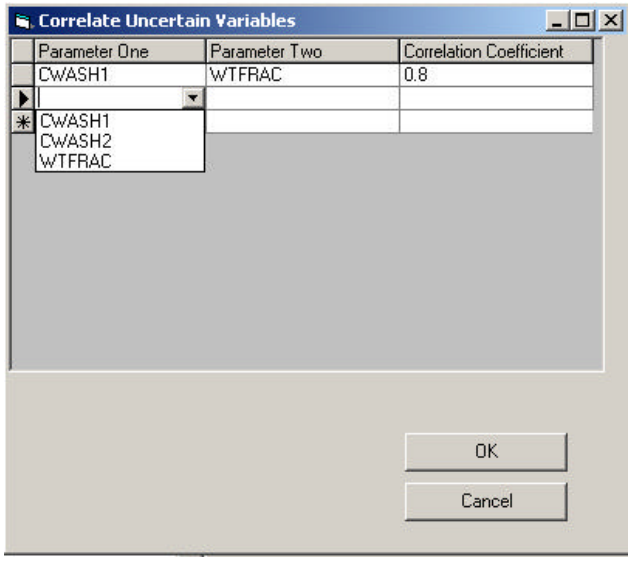


Figure 5. Screen used to assign correlation coefficients between two uncertain parameters. In the illustration, CWASH1 is correlated to WTFRAC with a correlation coefficient of 0.8.

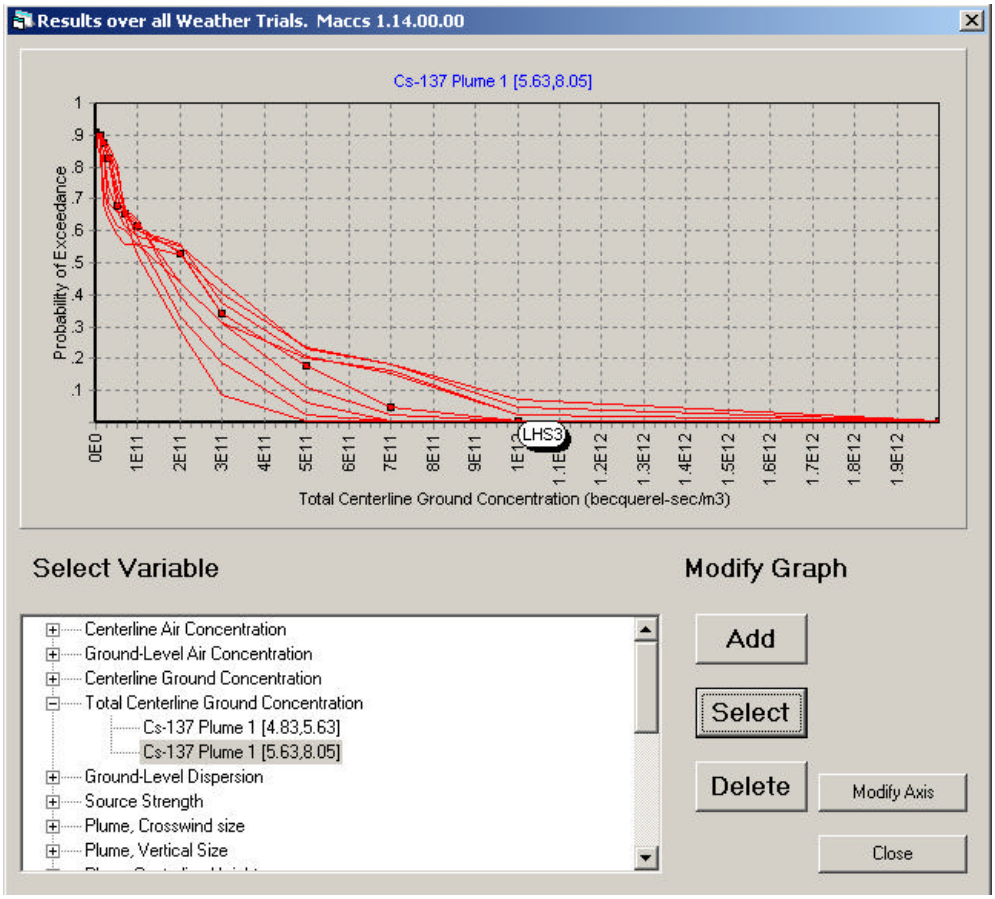


Figure 6. Screen showing CCDFs for total centerline ground concentration of Cs-137 for ten realizations.

4. Current Status and Schedule

The development of WinMACCS is partitioned into three phases. The first phase was completed in September 2001. At that point, WinMACCS supported only the ATMOS input to the MACCS2 code. By the end of September 2002, a Phase 2 version will be completed that supports ATMOS and EARLY, that allows single (deterministic input) runs or multiple-realization runs using LHS, and that includes preliminary plotting capabilities. The current expectation is that the third phase of WinMACCS development will be completed by September 2003. The final version will support the CHRONC module and the COMIDA2 preprocessor as well as ATMOS and EARLY, will improve plotting capabilities, will integrate regression analyses to determine the sensitivities of results to uncertain input values, and will allow current MACCS2 input files to be imported.

New documents describing MACCS2 and WinMACCS will be release concurrently with the deployment of WinMACCS. This should take place during 2004.

5. Summary

A new user interface, called WinMACCS is under development. It significantly enhances the usability of MACCS2 and greatly streamlines the effort needed to conduct Monte-Carlo analyses where one or more (some US Nuclear Regulatory Commission calculations involve more than 200) input variables are uncertain. When completed, WinMACCS will simplify the process of creating or modifying a MACCS2 input file and will automate the processes of performing Monte-Carlo analyses and evaluating the results. The current schedule is to release WinMACCS and the accompanying documentation in 2004.

A new version of MACCS2, which will continue to perform the atmospheric dispersion calculations, will also be released with WinMACCS. The new version of MACCS2 will have all known bugs fixed, will be more flexible, and will likely have several upgraded and new models.

6. References

Briggs, G. A. (1971). *Some Recent Analyses of Plume Rise Observation*, **Proc. Second International Clean Air Congress**, Academic Press, New York.

Briggs, G. A. (1972). *Discussion: Chimney Plumes in Neutral and Stable Surroundings*, **Atmospheric Environment**, 6:507-510.

Briggs, G. A. (1975). Lecture given to the American Meteorological Society, Boston, MA, USA.

Chanin, D. and Young, M. L. (1998). *Code Manual for MACCS2,* NUREG/CR-6613 Vol. 1, SAND97-0594, Sandia National Laboratories, Albuquerque, NM, USA.

Gauntt, R. O., Cole, R. K., Erickson, C. M., Gido, R. G., Gasser, R. D., Rodriguez, S. B., and Young, M. F. (2000). *MELCOR Computer Code Manuals, Vol. 1: Primer and User's Guide*, NUREG/CR-6119, Vol. 1, Rev. 2, SAND2000-2417/1, Sandia National Laboratories, Albuquerque, NM, USA.

Mills, M. T. (1987). *Modeling the Release and Dispersion of Toxic Combustion Products from Chemical Fires*, International Conf. On Vapor Cloud Modeling, Cambridge, MA, USA.

Wyss, G. D. and Jorgensen, K. H. (1998). *A User's Guide to LHS: Sandia's Latin Hypercube Sampling Software*, SAND98-0210, Sandia National Laboratories, Albuquerque, NM, USA.

Linking Radionuclides Release and Source Terms to MACCS2 Code, Coupled with the Results of PSA Levels 1, 2 & 3, and Interfaced with a New Algorithm of Nuclear Consequence Measure (NCM)

By

F. L. (Bill) Cho
Illinois Department of Nuclear Safety
Springfield, IL U.S.A.
Phone: (217) 785-9901; Fax: (217) 782-9762
E-mail: cho@idns.state.IL.US

ABSTRACT

THIS PAPER INTRODUCES A NEWLY DEVELOPED ALGORITHM OF "NUCLEAR CONSEQUENCE MEASURE (NCM)" TO SERVE AS AN ENHANCEMENT FOR THE MACCS2 CODE, SO THAT IT WILL ASCERTAIN THE INPUT OF RADIONUCLIDES FRACTION AND SOURCE TERMS RELEASE INFORMATION BEING USED ACCURATELY AND TRACEABLE. I BELIEVE THAT THIS PAPER WILL ENHANCE A VALUE OF THE MACCS2 CODIFICATION. THUS THE MAIN OBJECTIVE IS TO STUDY THE PHYSICAL FAILURE PHENOMENA OF CORE MELT DAMAGE (CMD) AND ITS CONSEQUENCES THROUGH A DEEP UNDERSTANDING OF ITS PHEMENOLOGOCAL PROCESS, THAT COULD BE CAUSED BY SPECIFIC COMBINATION OF PLANT DAMAGE STATES (PDS) OR SEVERE ACCIDENT SEQUENCES, WHICH EASILY PROMPT INTO A CONTINUED PROGRESSION OF REACTOR PRESSURE VESSEL (RPV) AND CONTAINMENT FAILURES, AND THEN RELEASE RADIONUCLIDES FRACTION AND SOURCE TERMS TO THE ENVIRONMENT, IF THE RPV WERE NOT CONTROLLED AND MANAGED.

INTRODUCTION

This paper reveals a paradigm of analyzing the consequential effects of severe nuclear reactor accident, radionuclides fraction and source terms release, that will influence the MACCS2 codification [1], by coupling with the results of SAPHIA-

PSA Levels 1& 2 quantification process [2], MELCORE [3], STCP [4], PST [5], and XSOR [6]. Those codes are mutually exclusive and useful. However, it lacks of the closed interface and linkage for addressing Plant Damage States (PDS), Severe Accident Sequences, and Risk Consequence. Thus, it is imperative to formulate the consistent baseline information for MACCS2, PSA Levels 1, 2 and 3, and then linking to a new algorithm of NCM.

The principal task is to generate the input of radionuclides fraction and source terms release for MACCS2 by using of MELCOR or other thermal hydraulic codes, which are capable of computing the transport of fission products release for specific plant damage states and/or sever accident event sequences. Through above working processes, we can identify or ascertain the root causes of all probable risk outcomes, which could likely be initiated and/or caused by single or multiple fatal failures and damages of some safety-related Structure, System and Component (SSC), Equipment, and/or Man-made Error/Fault. Subsequently, those affected PDS or Accident Event Sequences could induce the degradation of RPV and Containment failures, if the whole process not been controlled and managed.

A newly developed algorithm, NCM, as intended, will facilitate the necessary links for risk quantification process for the above inter-related codes. Hence, the working procedure of NCM will associate with those degraded PDS and Accident Event Sequences by computing the λ-Indexes (Cho's Weighted Factors), represented by λ_a, λ_b, λ_c....λ_g, for the seven blocks of Normalized and Combined State Variable Terms (NCSVT), which designate the entire nuclear reactor accident phenomenological process. That definition of those NCSVT blocks and the λ-Indexes computation will be illustrated at a later section of this paper.

UNDERSTANDING THE NATURES OF SEVERE NUCLEAR ACCIDENTS

The performance of a Probabilistic Risk Assessment (PRA), Level 2 Analysis for Large Early Release Frequency (LERF), as stated in the ASME Standard for PRA for Nuclear Power Plant Application [7] and NUREG/CR-6595 [8], is a formidable task, if we fail to understand the characteristics of severe accident phenomena. The principal concerns are that we have to deal with multi-disciplinary tasks and to understand various kinds of complex phenomenological scenarios involved. Prominent among these phenomenon are as follows:

- Loss of Containment Isolation Function

- Containment Bypass (ISLOCA, SGTR, Induced SGTR)
- In-vessel and Ex-vessel Failures
- High Pressure Melt Injection (HPME, Corium Impingement)
- Induced RPV Failure (DCH, Containment (venting)
- Hydrogen Combustion
- ATWS-Induced Failure
- Steam Explosion
- Shell Melt-through for BWR Marks I, II and III
- Vacuum Breaker Failure
- Suppression Pool Bypass
- Isolation Condenser Tube Rupture

Starting with the PRA Level 1 risk assessment, we must consider that the threshold of Core Damage Frequency (CDF) from the analysis must be less than the value of 1.0E-06 to assure safe reactor operation. By using this CDF value as the standard, the likelihood of an accident leading to a core melt damage sequence (CMD) will be minimized. It is universally accepted that each nuclear plant must meet minimum plant operability standards and maintain procedures for systems, equipment and components. Additionally, each nuclear plant must maintain human performance requirements and operate under the constraints of the Reliability Centered Maintenance (RCM) Program. These operational considerations also act to minimize the probability of a CMD event.

Historically the consideration of core melt effects and its consequences has prompted the development of Severe Accident Technology by the various NRC research groups, National Laboratories, working with the Industry Core Rulemaking (IDCOR) committees, have addressed many issues resulting from severe core damage scenarios, core melt progression phenomenology, and in-vessel and ex-vessel accident sequences.

Phenomenological issues dealing with core melt include:

1. hydrogen generation from steam oxidation of the Zircaloy cladding which starts at about 1,500 K

2. melting of the stainless steel control rod cladding at 1,700 K

3. melting of molten metallic zirconium, ZrO2 at 2,200 to 2,500 K

4. melting of UO2 at 3,100 K.

Thermal-hydraulic analyses for studying the effects of the reactor core melt phenomena and its severe accident consequences (source term and offsite release consequences), have been extensively undertaken by the various analysis codes such as MELPROG/TRAC, CORMLT, MAAP(IDCOR), SCDAP/RELAP5, MELCOR and MARCH (BMI).

At present, there still exist many technical uncertainties in these studies. A discussion of those uncertainty and sensitivity issues can be found in references [8, 9, 10, and 11]. Some of the areas of uncertainty remaining are:

1. natural circulation in reactor coolant system

2. core melt progression and hydrogen generation

3. steam explosions

4. high-pressure melt ejection

5. core-concrete interaction

6. hydrogen combustion

7. radionuclides chemical forms

8. fission product revaporization

THE ROLE AND FUNCTION OF PSA LEVELS 1, 2 AND 3 EVALUATION, ASSOCIATED WITH RADIONUCLIDES FRACTION AND SOURCE TERMS RELEASE, WHICH SHALL BE DERIEVED FROM A SEVERE NUCLEAR REACTOR ACCIDENT EVALUATION

The nature of severe accident phenomena studies contains deterministic and knowledge uncertainties in analyzing damage mechanisms, and other physical phenomenon. It has been a continuing United States national effort to resolve such issues. However, it remains difficult to reach any comprehensive and conclusive results. With the development of the ASME PRA methodology [7], we have begun to learn a new procedure for PRA Levels 1 and 2 analyses and quantification. However, we still must understand the mechanics of linking Plant Damage States (PDS) and its associated severe accident sequences.

Our current knowledge of containment response can be credited to the U.S. NRC leadership and management, resulting in the characterization of the phenomena of containment failure modes for BWR and PWR nuclear power plants. We are now in a position to adopt the PRA/PSA methodology to link the radionuclides fraction and source terms release from thermal-hydraulic analyses for PRA/PSA Levels 1, 2 and 3, and then linking to MACCS code. The overall strategy of implementing risk assessment process is listed below:

PSA Level-1 Results: Core Damage Frequency or Plant Damage State Frequencies; (SAPHIRE)

PSA Level-2 Results: Containment Failure Frequency, Conditional Containment Failure Probability, Large Early Release Frequency (LERF-L2 Quantification); (SAPHIRE), Parametric Source Term (PST), (Licensee IPE)

PSA Level-3 Results: Identifying and Determining the Safety of Radionuclides Release Level for Assessing Its Health Effects, Environmental Impact, and Financial Liability for the Public Safety and Concerns; (MACCS2)

In order to understand how the results of PSA Levels 1, 2 and 3 evaluations could be affected by the radionuclides fraction and source terms release, we must resolve the following reliability and safety issues, which are encountered with each nuclear power plant, as stated below:

1. What are PRA/PSA Level 1 attributes that contribute or impact Levels 2 and 3 analyses?

2. What are the interactions between frontline systems and its support systems for supporting nuclear reactor safety operations?

3. What are the risk mitigation strategies and measures of ECCS safety functions, so that it can prevent a core melt damage?

THE DETERMINATIONS OF THERMAL HYDRAULIC ANALYSES AND THE UNDERSTANDING OF PHYSICALLY CHAOTIC PHENOMENA, PROMPTING RADIONUCLIDES FRACTION AND SOURCE TERMS RELEASE TO THE ENVIRONMENT

The Sandia National Laboratories studies on Source Terms Analysis [9] had identified the significant State Variables, which can define the characteristics of radionuclides release process and transport phenomena during the Reactor Pressure Vessel (RPV) and Containment failures.

With an understanding of the physical conditions (see Figures 1 & 2) in a severe accident phenomena, we can select the most important representative elements of the State Variables and their constitutive relations to reflect the whole severe accident phenomena, expressed by five explicit equations (Equations 1 through Equation 5 below). These equations are primarily from Sandia's source term studies published in the NRC topical reports. We believe that beginning with these five governing equations a methodology can be developed to simplify the assessment of the consequences of radionuclides release to the environment, resulted from RPV and Containment failures. The goal is to derive a meaningful synthesis of the five fundamental equations into one integral expression (Equation 6) to measure the total likelihood of release consequence to the environment.

These fundamental equations are as follows:

Equation #1: for early per RCS iodine release:

(ST1) = [FCOR * FVES * FCONV/DFE] + DST

For the late release or CCI iodine release:

(ST2) = [(1 – FCOR) * FPART * FCCI * FCONC/DFL] + FLATE + LATE1

Where:

STi = fraction of the core iodine in the Reactor Coolant System (RCS) release to the environment; for accident regime i; where, i = 1 for early release state, i = 2 for late release state;

FCOR = fraction of the iodine in the core released to the vessel before Vessel Breach (VB);

FCONV = fraction of the iodine in the containment from the RCS release that is released by the containment in the absence of any mitigating effects;

FVES = fraction of the iodine released to the vessel that is subsequently released to the containment;

DFE = decontamination factor for RCS release (sprays, etc.);

DST = fraction of core iodine released to the environment due to direct containment heating at vessel breach;

FPART = fraction of the core that participates in the Core Concrete Interaction (CCI);

FCCI = fraction of the iodine from CCI released to the containment;

FCONC = fraction of the iodine in the containment from the CCI release that is released from the containment in the absence of any mitigation effects;

DFL = decontamination factor for late release (spray, etc.);

FLATE = fraction of core iodine in the RCS that is revolatilized and released late in the accident;

LATE1 = fraction of core iodine remaining in the containment that is converted to volatile forms and released late in the accident.

Some of the above State Variables will be expressed later by Equation 6 by a ratio of the specific accident scenario value to its baseline value.

Equation #2 - The fraction of radionuclides released:

FCONV = mVout/mVin

Where:

mVin = mass of radionuclide (or class) released from the vessel to containment atmosphere at or before VB;

mVout = mass of a radionuclide (or class) released from the vessel to containment atmosphere at or before VB that is subsequently released from containment.

Equation #3 - The calculation of the late release of iodine in volatile form:

LATE1 = XLATE* [{FCOR*FVES + (1-FCOR)*FPART*FCCI} - ST - STL + FLATE]

Where:

XLATE = fraction of iodine in the containment late in the accident that assumes a volatile form and is released to the environment;

FCOR = fraction of iodine in the core released to the vessel before VB;

FVES = fraction of the iodine released to the vessel that is subsequently released to the containment;

FPART = fraction of the core that participates in the CCI;

FCCI = fraction of the iodine from CCI released to the containment;

ST = fraction of the core iodine in the RCS released to the environment;

STL = fraction of core iodine released to environment.

Equation #4 - The conversion of each isotope into an equivalent amount of I-131:

$EQN_k = CF_k * I_k * STN_k * \exp[-\lambda_k (TN + DTN/2)]$ $EQ = \Sigma_k EQ1_k + \Sigma_k EQ2_k$

Where:

N = 1 for early release and N = 2 for the late release;

EQN_k = the equivalent amount of I-131 for isotope $_k$ for release N;

CF_k = the isotope conversion factor for isotope $_k$;

I_k = the inventory of isotope $_k$;

STN_k = the release fraction of isotope $_k$ for release N;

λ_k the decay constant for isotope $_k$;

TN = the time of the start of release N, and;

DTN = the duration of release N.

Equation #5 - the chronic fatality weight for each isotope:

$CFW_k = I_k (ST1_k + ST2_k) * (ELCF_k + CLCF_k)/R_k$

Where:

I_k - the inventory of isotope $_k$;

N = early release (1) or late release (2);

STN_k - the release fraction of isotope $_k$ for release N;

CFW_k = the chronic fatality weight, in latent cancer fatalities, for a release of an amount $I_k (ST1_k + ST2_k)$ of isotope $_k$;

$ELCF_k$ = the number of latent cancer fatalities due to the early exposure from a release of an amount R_k of isotope $_k$;

$CLCF_k$ = the number of latent cancer fatalities due to the late exposure from a release of an amount R_k of isotope $_k$;

R_k = the amount of isotope $_k$ released in the MACCS calculation per MACCS2 Code [1], used to determine $ELCF_k$ and $CLCF_k$.

AN EMERGING PARADIGM, EXPRESSED BY THE EQUATION #6 FOR ASSESSING "NUCLEAR CONSEQUENCE MEASURE (NCM)"

An attempt is made to establish a quantifiable methodology to assess the overall risk consequence of radionuclides released to the environment for PSA Levels 2 and 3 considerations. By considering key components of the above five fundamental equations, a simplified equation is proposed to formulate a Nuclear Consequence Measure (NCM) for the consequence determination for any probable accident sequence. This NCM will reflect the severity of any in-site and/or offsite consequences following a radionuclides release from the existing nuclear power plant. A larger NCM indicates a more severe offsite consequence than a lower number.

The proposed Equation #6 is expressed as follows:

$$NCM = \{(st1/ST1)\lambda_a + (st2/ST2)\lambda_b + (fconv/FCONV)\lambda_c + (late1/LATE1)\lambda_d + [\Sigma_k(eq1_k/EQ1_k)]\lambda_e + [\Sigma_k(eq2_k/EQ2_k)]\lambda_f + [\Sigma_k(cfw_k/CFW_k)\lambda_g]\}$$

Where:

NCM = for assessing the in-site and offsite consequences measure by seven Normalized and Combined State Variables Terms (NCSVT) blocks.

Lower Case Terms = measured, calculated or estimated values;

Upper case Terms = baseline value.

λ_i = Seven λ-Indexes (Cho's Weighted Factors) for quantifying the Normalized and Combined State Variable Terms (NCSVT) blocks in Equation #6. The characteristics of those $\lambda_{i=a,b,c...g}$ shall be discussed at the subsequent section for those seven NCSVT blocks, activated by any accident phenomena process at the times of: "Early or Late Release before Vessel Breach [8]" states.

Each block of NCSVT in Equation #6 consists of a scenario-specified calculated, measured or observed value (lower case variable names), divided by its respective baseline value (capitalized variable names). In addition, each term is further multiplied by the "$\lambda_{i=a,b,c,d..g}$ Indexes" respectively for calculating a mixed representation and consideration of reliability and safety for the entire nuclear plant facility.

Each the λ-Index can express the plant degradation process in all Plant Damage State (PDS) or Accident Event Sequences, associated with the physical damage states of Structure, System, Component (SSC), Equipment, and/or Man-made Fault/Error, which shall be expressed by an algorithm of Successive Failure Probability Ratio Multiplication (SFPRM) between the current and previous values.

THE CHARACTERIZATION OF λ-INDEXES (Cho's Weighted Factors) FOR THE NORMALIZED AND COMBINED STATE VARIABLES TERM (NCSVT) BLOCKS IN EQUATION #6

The characteristic of λ-Indexes (Cho's Weighted Factors) has an implicit time-dependent representation, which can express a continued series of the "Successive Failure Probability Ratio Multiplication (SFPRM)" computation, by tracing any change of the reliability value change between the new (current) and old (previous) from those affected PDS and failed Accident Event Sequences in SSC, Equipment, and Man-made

Errors/Faults. Therefore, it is applicable to monitor the change of the plant time history management by tracking any successive reliability changes for the daily, weekly, monthly, semi-annual or annually operation, and concurrently for following up the surveillance or inspection activities to determine any degree of the degraded conditions of malfunction , damage, or loss of system and component safety function, etc.

The general form is expressed as follows:

$$\lambda_{PDS=1,2...N} \text{ or } \lambda_{AES=1,2..m} = \int_{\theta=1...\theta} [FP_\theta(new)/FP_\theta(old)\bullet\bullet\bullet]\bullet(RF_\theta)\bullet(CDF_\theta)d\theta$$

Where:

" $_{\theta=1,....\theta}$": It denotes the certain elements # ($_\theta$) within its failed PDS, or Sequence Cut Sets, which contain the base event failure probability being changed for the participated Accident Event Sequence, resulting from the degraded or damaged cause in Structure, System, Component (SSC), Equipment and Man-made Fault/Error, etc;

RF_θ : Reduction Factor for the individual λ-Index will be required by multiplying a decimal percentage factor of its element CDF contribution, whenever the failed element (PDS) using a value of "1.0" as a new Failure Probability for replacing the previous (old) value.

CDF_θ: a CDF for PDS, contained the "$_\theta$" element, or a CCDP for its Sequence Cut Set Events group, contained the failed basic event for the Accident Event Sequence.

The necessary computational features for RF_θ and CDF_θ will be demonstrated in the subsequent section of An Illustrated Example.....

Since the objective of risk assessment is to evaluate whole phenomenological process of a severe nuclear accident, we have to compute a total summation of the $\lambda_{NCSVT=a,b,c,d....g}$ (Cho's Weighted Factors), that can be expressed as shown below, by treating each block of Equation #6 respectively, per the extent of each PSA Levels 1,2 and 3 evaluation:

For PDSs:

$$\lambda_{NCSVT=a,b,c...g} = \sum_{NCSVT=a,b..g} \{\sum_{pdsn=1..n}\{\int_{\theta=1...\theta} [FP_\theta(new)/FP_\theta(old)\bullet\bullet\bullet](RF_\theta)\bullet(CDF_\theta)d\theta\}_{NCSVT=a,b...g}\}$$

• (1.0/The Limit Value, Based on the Plant CDF);

For Accident Event Sequences:

$$\lambda_{NCSVT=a,\,b,\,c\ldots g} = \sum_{NCSVT=a,b..g} \{\sum_{AES=1\ldots.m}\{\leftarrow\!\!\lrcorner_{\theta=1\ldots..\theta} [FP_\theta(new)/FP_\theta(old)\bullet\bullet\bullet]\bullet(RF_\theta)\bullet(CDF_\theta)d\theta\}_{NCSVT=a,b\ldots..g}\} \bullet (1.0/\text{The Limit Value, Based on the Plant CDF};$$

Where:

Each λ-Index (λ_a, $\lambda_{b\ldots}$ and λ_g) represents the individual sum of all Successive Failure Probability Ratio Multiplication (SFPMR) for the respective block of Normalized and Combined State Variables Terms (NCSVT), so that we can assess the PSA Levels 1, 2 and 3 evaluation. Then, the Limit Value of Plant Base CDF shall be referred to the latest Plant Base Modified CDF, being reported in the Licensee's IPE Reports, which had submitted to the NRC.

The algorithm of Successive Failure Probability Ratio Multiplication (SFPRM) can fully express the process of physical changing state of the failed PDS or Accident Sequence through the means of tracking down the path of RPV and Containment failure progression, occurred at the times of "Before, At and Late" Vessel Breach, and then progressed to a Core Melt Damage (CMD) and probably might reach the total damage of RPV/Containment.

Since the determination of λ_a (the first block of NCSVT in Equation #6) basically covers the scope of PSA Levels 1 and 2 quantification, thus we must first comprehend about the underlying root causes of CMD phenomenological occurrence, and then solve the following open tasks, as been discussed in the preceding sections:

1. What are PRA/PSA Level 1 attributes that contribute or impact Level 2 analyses?
2. What are the interactions between frontline systems and its support systems for supporting nuclear reactor safety operations?
3. What are the risk mitigation strategies and measures of ECCS safety functions to prevent a core melt damage?

Now, it becomes apparent that by having well thought processes for above tasks, it will help us to formulate a strategy of linking PSA Levels 1 and 2 results into MACCS2 code. Thus, the most important concern is that whether or not both PRA/PSA and non-PRA/PSA practitioners be able to trace any latest information of radionuclides fraction and source terms release from MELCOR or other thermal-hydraulic codes for

performing PSA Level 2 Analysis from the Licensee's latest Individual Plant Examination (IPE) Reports, which had submitted for the USNRC?

AN ILLUSTRATED EXAMPLE FOR QUANTIFYING AND QUALIFYING THE λ_a INDEX AND NCM (NUCLEAR CONSEQUENCE MEASURE), BASED ON THE LICENCEE'S IPE-RELATED PSA LEVELS 1, 2 AND 3 RESULTS

Part 1: For PSA Level 1Consideration:

Quantification Process: λ_a (Cho's Weighted Factor) for the block #1 of NCSVT in Equation #6:

Step #1 - Computing the Changes of Failure Probability for All Affected PDS: for the following Cases 1 and 2, which had been reported and occurred at the same time instant:

Case 1: Assumed that a "PDS #23 (B14S)" had a 0.54% (1.67E-07) contribution to the plant CDF:

PDS	A Series of Events
#23- (B14S):	LOOP; AC141; AC142; SX; SBO; ***XCC***

A failure of ***XCC***, indicated "core uncover prior to 24 hrs w/RCS cool down being designated as "SEALLOCA"; which was excerpted from a plant IPE;

A conditional failure frequency of "SEALLOCA" had been reported as "3.22E-02", resulting from ***XCC*** failure;

Thus, computing the failure probability for the "new" and "old" damage states as follows:

Old failure probability of XCC: **$FP_{\#23}$ (old)= 3.22E-002;**
New failure Probability of XCC: **$FP_{\#23}$ (new)= 1.00E-000; assumed as a worst case**

Case 2: Assumed that there were another set of the affected PDS, which were: PDS #55 (SE6K), PDS #74 (SL6E), and PDS #85 (L14E), that being

reported, and had contributed a sum of 0.3.% (1.24E-07) to the plant CDF jointly by three PDS in the previous state, and their present state been affected by the damages of ***FC and CI*** as shown below:

PDS	A Series of Events
#55-(SE6K):	SLOCA; ESF-A; ESF-B; CCP; SIP; AFW; ORF; ***FC***
#74-(SL6E):	SLOCA; ESF-A; ESF-B; CCP; SIP; AFW; LPI; FC; ***CI***
#85-(L14E):	LOOP; AC141; AFW; CCP; ***CI***

A series of the multiple damage states were reported as follows:

(1) - PDS #55: ***FC*** (All FAN COOLERS - TRAIN A & B OR SX FAILED) with a Failure Probability (FP) equals "1.0"
(2) - PDS #74***: CI*** (CTMT ISO FAILS - ALL POWER AVIL - SIS FAILED, CS ON) with a Failure Probability (FP) value of "2.31E-01";
(3) - PDS #85: ***CI*** (CTMY ISO FAILS - ALL POWER AVAIL, SIS SUCCESS, CS ACTIVATED) with a Failure Probability (FP) value of "8.01E-03.

Step #2 - Computing the Increase of Failure Probability Ratio for the affected PDS by its respective Reduction Factor, since new Failure Probability being assumed to the value of "1.0":

- - $\lambda_{\#23}$ for PDS #23 = (1.0/3.22E-02)•0.0054 = 0.168
- - $\lambda_{\#55}$ for PDS #55 = (1.0/1.0) = 1.0
- - $\lambda_{\#74}$ for PDS #74 = (1.0/2.31E-01•0.003 = 0.0129
- - $\lambda_{\#85}$ for PDS #85 = (1.0/8.01E-03)•0.003 = 0.372

Step #3 - Computing the Increase of Plant CDF by the individually damaged PDS:

	Old CDF_0		**Per Step #2**		**Step #3**
• - by ($\lambda_{\#23}$ *	1.67E-07)	=	(0.1680 *	1.67E-07) =	2.67E-08
• - by ($\lambda_{\#55}$ *	5.76E-08)	=	(1.0000 *	5.76E-08) =	5.76E-08
• - by ($\lambda_{\#74}$ *	3.84E-08)	=	(0.0129 *	3.84E-08) =	4.95E-10
• - by ($\lambda_{\#85}$ *	2.75E-08)	=	(0.3720 *	2.75E-08) =	1.01E-08

A Sum of the Plant CDF Increased = 9.49E-08,

Step #4 - Calculating the value of λ_a, affected by all above degraded PDS for NCM:

: Quantifying the Index of λ_a based on the Plant's System/Component Reliability, not yet accounting the possible effects by the releases of Radionuclide Fraction and Source Terms Release, since there was no report of "Vessel Breach (VB)"being occurred. However, we like to see how the Licensee's IPE results had assessed their concerns on the possible release issue: this can be assessed by λ_a for the block #1 of NCSVT in Equation #6:

λ_a = **9.49E-08 /3.09E-05 = 0.00307 < 1.0 ; an insignificant case , which had no radionuclides released**

Where: **"3.09E-05"** which was available data for the Plant Base CDF from an existing nuclear power plant, considering this as the basis of the Limit Value.

Since there will likely be no CMD Phenomena, so that the In-vessel and Ex-vessel Analyses are not needed. Thus, there will be no any further off-site consequence by Radionuclids Fraction and Source Terms Release to the environment. But we still check the Licensee's IPE assessment.

Step #5 - Tracing other available source of Radionuclides Fraction and Source Terms Release, which had been determined by the Licensee in their IPE Report to the NRC.

Based the above evaluation of λa, which is one of seven NCSVT blocks in Equation #6, we may look into any possibility of Radionuclides Fraction Release in the Reactor Coolant System. As being studied, the Licensee's IPE had verified the Reactor Coolant System. At present, there were no baseline information available for NCSVT block #1 for comparing with the Licensee's findings as listed below:

PDS	I	CS	TE	SR
#55-(SE6K)	9.98E-01	9.6E-04	9.6E-04	8.0E-010
#74-(SL6E):	7.43E-03	3.0E-06	3.0E-06	6.0E-08
#85-(L14E):	1.4E-03	5.0E-06	5.0E-06	9.0E-09

; no fraction of radionuclides release for RU, LA, CE,and BA.

ADDITIONAL CONSIDERATIONS TO BE REMEMBERED FOR ENHANCING THE PART 1 OF ABOVE PSA LEVEL 1 QUANTIFICATION PROCESS

Important Note #1:

As intended, the λ-Indexes shall be used for weighing some indefinite failure probability input data and their baseline information from the existing nuclear power plant, thus requiring to use the algorithm of Successive Failure Probability Ratio Multiplication (SFPRM) for tracking down between the present (new) and previous (old) values of failure probability of the elements for the failed or damaged PDS or Accident Event Sequence, and then to be divided by the baseline data. It is worthy to note that the end result of λ_a has an inverse relation between the nuclear power plant reliability (failure probability) and the safety level of Radionuclides Fractions and Source Terms Release.

Important Note #2

The essence of above proposed λ_a-Index computation is cited on the observation or surveillance of the factual degradation process and the failure-damage occurrences in SSC, Equipment and Man-made Fault/Error during the operation of nuclear power plant facilities.

Part 2: For PSA Levels 2 and 3 Considerations

We will still adopt the same quantification and qualification processes, shown in Part 1 above. However, **we** must focus on the failures of specific SSC, equipment, human performance reliability (errors and faults) from those affected PDS and Accident Sequence by associating with the failure scenarios and damage information, being

considered and used in ASP/PST and XSOR [2, 3, 4, 5 and 6] in order to establish the necessary links within NCM and PSA Levels 1, 2 and 3 Results consistently per the following concerned areas:

1. Containment Failure Mode
2. Containment Heat Removal System Available
3. Core Concrete Interaction
4. Mode of Vessel Breach
5. SGTR
6. Amount of Oxidation
7. Containment Failure Size
8. Time of Core Damage
9. Others not yet being specified

Basically, PST [5] had used above damage domains to establish the characteristics of Source Term Vector (STV), Source Terms Group (STG), Source Term Event Tree (STET), and Risk Event Tree (RET) in conjunction with the NRC Accident Sequence Precursor (ASP). All of those source information can be integrated and interfaced with NCM, while adopting the same methods of qualification and quantification, as shown in Part 1. Thus, all risk-based information are accountable and consistent between the baseline and quantified values, regardless of the analyst's preference of using their adopted computational code and method.

SOME APPLICABLE BASELINE INFORMATION FOR DETERMINING THE RADIONUCLIDE RELEASE AND SOURCE TERMS, AVAILABLE FOR MACCS2 , PSA LEVELS 2 AND NCM CODIFICATIONS

Recently, the USNRC had issued a draft regulatory Guide DG-113, "METHODS AND ASSUMPTIONS FOR EVALUATING RADIOLOGICAL; CONSEQUENCES OF DESIGN BASES ACCIDENTD AT LIGHT-WATER NUCLEAR POWER REACTORS", in which defined the transport phenomena and the governing equations for radionuclides fraction release and source term determination, and also stated the Release Fractions for DBA LOCA per Table 1 and for Non-LOCA Fraction of Fission Product Inventory in Gap per Table 2.

Furthermore, the USNRC Regulatory Guides [11, 12, 13, 14] had addressed nine (9) DBA for evaluating radiological consequences for PWR and PWR nuclear plants for:

- - LWR Loss of Coolant Accident
- - Fuel Handling Accident
- - BWR Rod Drop Accident

- BWR Main Steam Line Break Accident
- PWR Steam Generator Tube Rupture Accident
- PWR Main Steam Line Break Accident
- PWR Locked Rotor Accident
- PWR Rod Ejection Accident

As can be seen from above list, those DBA were not discussed about whether or nor it had been occurred on or before or within the Core-Melt-Damage (CMD) progressions of Reactor Vessel and containment failures. Besides, the NRC had not discussed about whether or not those accidents could associate with any failed PDS Accident Event Sequences, any of ECCS safety functions, that could affect the increase of CDF beyond the Plant Limit, thus prompting a CMD, and then releasing the fractions of Radionuclides and Source Terms to the environment.

As I see now, the next level of the PSA Level 3 consideration, we have to rely on MASS2 in conjunction with the codes of Parametric Source Term, PST [5] and XSOR [6], which are capable of modeling the transport of radionuclides release and source terms from the core, through the reactor vessel, through the containment, and to the environment. At present, PST does provide a proper computational method for source term determination, but requiring extensive In-Vessel and Ex-Vessel modeling with the time-dependent progression. The merit and value of PST[5] can be coupled with SAPHIRE [2] /ASP Source Term Vector (STV) information by linking with the PSA Levels 1 and 2 Accident Event Sequences, so that it enables us to determine the fraction of radionuclides release and source term determination. The whole linking process for making a viable "Bridge Event Tree (BET)" for the PRA Levels 2 and 3 quantification is still an extremely rigorous, complex, and tedious effort.

Therefore the creation of Nuclear Consequence Measure (NCM) algorithm is necessary, so that it can integrate with all viable information of the joint Reliability and Safety considerations. The fundamental of NCM paradigm is represented by an integral synthesis expression of the Equation #6, which associates with seven (7) Normalized and Combined State Variables Terms (NCSVT) blocks by multiplying λ-Indexes (Cho's Weighted Factors) through the means of Successive Failure Probability Ratio Multiplication algorithm. Thus, this newly developed paradigm can compile and evaluate all risk-based information from the sources of the existing nuclear plant operations and management activities, particularly enabling us to measure the degradation process, occurred in any kind of the time frame by a continued and concerted examination for those affected PDS or degraded Accident Event Sequences, if required.

CONCLUDING REMARK

So far, this paper presents a viable means for risk quantification and qualification by using the above discussed paradigm and algorithm of computing the λ-Indexes and NCM, interfaced with the PSA Levels 1, 2 and 3 [2 &3] results, and with all applicable thermal-hydraulic codes [3, 4, 5, & 6], whatever being used.

Thus I have demonstrated that the importance of MACC2 Input data have to be accountable and to have a complete transparent information throughout all quantification and qualification processes, so that we will be enabling to treat all pertinent risk-based information consistently, but not to be separated or de-coupled. It is important that we ought to link and interface with all applicable data, so that we are qualified to perform a truly integrated risk evaluation. By virtue of the NCM algorithm, we will be able to link all data sources, while we are dealing with both reliability and safety issues concurrently.

The proposed methodology could reduce a great deal of times and efforts needed to quantify the results of source terms and radionuclides fraction release. It is my hope that the "***NCM, Coupled with λ-Indexes Computation***" will soon become a most practical process for risk quantification and qualification for assessing the reliability and safety issues for the nuclear energy generation communities to be a worldwide application.

REFERENCES

[1] - U.S. Nuclear Regulatory Commission, *Code Manual for MACCS2, NUREG/CR-6613, Volumes 1,and 2, April 1998.*

[2] - Idaho National Engineering and Environmental Laboratory, *System Analysis Programs for Hands-on Integrated Reliability Evaluation (SAPHIRE), Version 6.7, July 2000*

[3] - S. E. Dingman, C.J. Shaffer, A.X. Payne, and M.K. Carmel of Sandia National Laboratories, *MELCOR ANALYSIS FOR ACCIDENT PROGRESSION ISSUES/NUREG/CR-5331, January 1991*

[4] - J. A. Gieseke, P. Cybulskis, H. Jordan, K.W. Lee, etc. of Battelle's Columbus Laboratory, *SOURCE TERM CODE PACKAGE- USER'S GUIDE-STCP, MODE 1/ NUREG/CR-4587, July 1986*

[5] - Idaho National Engineering Laboratory (INEL), *The Parametric Source Term (PST), INEL-96/0308, October 1996.*

[6] - H. N. Jow, W. B. Murfin, and J. D. Johnson, *XSOR Code Users Manual, NUREG/CR-5360, November 1993*

[7] - American Society of Mechanical Engineering, *"American National Standard for Probabilistic Risk Assessment for Nuclear Power Plant" Application, Rev 14A, May 11 2001.*

[8] - U.S. Nuclear Regulatory Commission, *An Approach of Estimating of Various Containment Failure Modes and Bypass Events, NUREG/CR-6595, January 1999.*

[9] - U.S. Nuclear Regulatory Commission, *Reactor Risk Reference Documents, , NUREG-1150, Volumes 1, Volume 2, and Volume 3 February 1987.*

[10] - U.S. Nuclear Regulatory Commission, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, Regulatory Guide 1.183, July 2000.*

[11] - R.L. Iman, J.C. Helton, and J.D. Johnson, *PARTITION: A Program for Defining the Source Term/Consequence Analysis Interface in the NUREG-1150 Probabilistic Risk Assessments, NUREG/CR-5253, SAND86-2940, November 1989.*

[12] - USNRC, *Assumption Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors, Regulatory Guide 1.3, Revision 2, June 1974*

[13] - USNRC, *Assumption Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, Regulatory Guide 1.4, Revision 2, June 1974*

[14] - USNRC, *Assumption Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors, Regulatory Guide 1.5, March 1971.*

Uncertainty Analysis System Using MACCS2

Susumu Ryufuku*, Mikihiro Oheda*, Masato Terakado*, Naohiro Kurosawa*, and Susumu Mitake+

*Visible Information Center, Inc.
440 Muramatsu Tokai-mura Ibaraki-ken, 319-1112 Japan
call@vic.co.jp
+Institute of Nuclear Safety, NUPEC,
17-1 Toranomon 3-Chome Minato-ku Tokyo, 105-0001 Japan
mitake@nupec.or.jp

1. Abstract

An uncertainty analysis system using MACCS2 was developed to evaluate environmental parameters related to probabilistic safety analysis for Japan's nuclear fuel recycle facility. The system was tested using site-specific data for Japan's nuclear fuel recycle facility and reasonable results were obtained. Further validation through comparison with other studies will be completed for both the system and the input parameters.

Features of this system are:

(1) The calculation system consists of LHS(1), MACCS2, COMIDA2, and SPOP(2) codes. The interface program was developed to; transfer data files between codes, run the required codes for the number of uncertainty data sets generated, and retrieve necessary data for statistical processing from output files.

(2) The Windows® interface simplifies the setting of uncertain input parameters (upper / lower / distribution) and the generation of resultant graphs and tables.

(3) Uncertain input parameters and their data are determined by referring to statistics published in Japan, NUREG/CR-4551, IAEA No. 364, et al. (3,4,5,6,7,8) Meteorological data were originally provided by the nuclear fuel recycle facility.

(4) Resulting doses are evaluated using uncertainty factor (UF), mean uncertainty coefficient (MUC)(9), and CCDF for uncertainty analysis, and Pearson's correlation coefficient (PAER), Spearman' s correlation coefficient (SPEA), partial correlation coefficient (PCC), partial rank correlation coefficient (PRCC), and scatter diagram for sensitivity analysis.

2. The System Structure

Fig. 1 shows the flow chart of the uncertainty analysis system. To prevent misleading results from occurring during the combination process, this system accepts the following input options:

- IPLUME = 1: Straight-line dispersion model[10]
- NUMREL = 1: One plume segment
- EVATYP = 'RADIAL': Evacuation along radial direction.
- One source term scenario and one emergency scenario.

First, the user must complete the base calculation by using the MACCS2 code so the input files can be used as bases to insert fluctuant data generated by the LHS code. The interface program repeats and makes input files which include fluctuant data, and then runs the COMIDA2 and MACCS2 codes *N* (number of data sets) times. The interface program assembles the following output data and accumulates same in files for statistical processing:

- Centerline dose of cloudshine, groundshine, inhalation and resuspension inhalation by EARLY module (Highest result along the radial distance is selected on each pathway)
- Population dose of total pathways by EARLY + CHRONC, groundshine, resuspension inhalation, ingestion, and total pathways by CHRONC module

'MEAN' of above items and fluctuant input data are accumulated in the SPOP input file for SPOP solution. 'CCDF' data and 50th, 90th, 95th, 99th, and 99.5th 'QUANTILES' data of population dose (total pathways by EARLY + CHRONC) are accumulated in the collected CCDF data file to derive UF (= 95% / 5% of CCDFs distribution at designated probability (Y-axis)) and MUC (= 95% / mean of CCDFs distribution at designated probability (Y-axis)). Each CCDF curve discussed here is due to meteorological sampling distribution of each MACCS2 run. CCDF curve generated by SPOP is due to the fluctuant data generated by LHS.

Correlation coefficients such as PEAR, SPEA, PCC, and PRCC derived by the SPOP code are organized as tables. The user can choose how many ranks to show in the table, for example, input parameters that induced the top three correlation coefficient values. Regression between input and output parameters can be visually confirmed by generating the scattering graph.

When determining UF and MUC, the user chooses the probability (Y-axis) first. The system automatically draws a graph of aggregated CCDF curves and a graph of 95%, mean, and 5% curves.

Fluctuation of population distribution and dose conversion factors are not considered in this system. Since input and output files are overwritten during repeating calculations, memory space is saved.

3. Performance tests

Criticality events and fire accidents mentioned in the licensing application of the nuclear fuel recycle facility[11] are referred to for examples in order to examine the system and the parameters. The number of data sets LHS generated was $N = 1000$.

3.1 Input parameters

(1) Fixed Parameters

Source term for criticality events and fire accidents were taken from the licensing application of the nuclear fuel recycle facility. Census data in 1995[12,13] were used to derive concentric population data for the site of interest. The dose conversion factor data file used was FGRDCF[8] based on EPA Federal Guidance Report No. 11 (1988), and No.12 (1993). Meteorological data observed in the nuclear fuel recycle facility were converted in to MACCS2's meteorological data.

(2) Uncertainty Parameters

MACCS2: data (e.g. deposition velocity, shielding factor, and resuspension data) are mainly based on Helton et al. (Reliability Engineering and System Safety 48 (1995)) and NUREG/CR-4551 (1990). Reasonable values of delay time were defined for sheltering and evacuation.

COMIDA2: food consumption data are based on Japan's statistics. Agricultural time parameters (e.g. tillage and harvest) were determined using statistics for the area where the facility is located. IAEA No. 364 (1994) and NUREG/CR-6613 Vol.2 (1998) were used to determine other parameters (e.g. animal product transfer coefficients).

3.2 Results

Partial resultant data are shown in **Table 1** and **Fig. 2**. **Table 1** shows the results of sensitivity analysis for population dose (total pathways, EARLY+CHRONC). The top graph in **Fig. 2** shows the distribution of CCDF due to fluctuant data generated by the LHS code.

The 95% and 5% envelope and the mean were derived. The UF and MUC were determined as shown in the middle graph. A sample of scattering graph, the relationship between σ_y-scaling factor and the inhalation dose is shown in the bottom of **Fig.2**. This combination exhibited one of the highest correlations.

4. Discussion

As shown in **Table 1**, 'date of harvest' presented the highest correlation in criticality event. Ingestion is the major pathway due to iodine nuclides. The fluctuation of the accident date is determined by meteorological bin sampling reflecting that an earlier harvest gives less chance of nuclide deposition onto crops and vice versa. In fire accident, input parameters related to inhalation showed high correlation. Because the released material consists of plutonium nuclides, parameters related to inhalation show higher ranks.

UF and MUC on the 95th quantile (5% in Y-axis) in criticality events case are 10.0 and 2.61, respectively. UF and MUC in fire accidents reflected similar values.

Besides the results mentioned above, other results confirmed that this system functioned correctly. However, to validate the system and the parameters, further examination is necessary by comparison with other studies.

5. Conclusions

- An Uncertainty analysis system using MACCS2 was developed. An interface using a Windows® interface simplifies the operations of establishing input data, running codes, processing results, and making resultant graphs and tables.
- Site-specific data for Japan's nuclear fuel recycle facility were prepared.
- The uncertainty analysis system was examined using site-specific data for the nuclear fuel recycle facility and reasonable results were obtained.
- Further comparison with other studies is necessary to substantiate validation of the system and the parameters.

Table 1 Sensitivity Analysis Result Sample

Population Dose (Total pathways, EARLY+CHRONC)

(a) Criticality event

Rank	PEAR (*1)	SPEA	PCC	PRCC
1	Date of harvest 4.08E-01	Date of harvest 4.82E-01	Date of harvest 5.97E-01	Date of harvest 7.42E-01
2	Dry deposition velocity 3.85E-01	σ_z-scaling factor -3.68E-01	Dry deposition velocity 5.55E-01	Dry deposition velocity 6.43E-01
3	σ_z-scaling factor -3.22E-01	Dry deposition velocity 3.61E-01	σ_z-scaling factor -4.73E-01	σ_z-scaling factor -5.93E-01

Ingestion is the major pathway due to iodine nuclides. The fluctuation of the accident date is determined by meteorological bin sampling so that an earlier harvest gives less chance of nuclide deposition onto crops and vice versa.

(b) Fire accident

Rank	PEAR	SPEA	PCC	PRCC
1	Resuspension coefficient (m=1) (*2) 5.32E-01	Resuspension coefficient (m=1) 6.55E-01	Inhalation protection factor (normal activity) 5.49E-01	Inhalation protection factor (normal activity) 8.46E-01
2	Resuspension half-life (m=3) -4.53E-01	Resuspension half-life (m=3) -5.92E-01	Breathing rate (normal activity) 4.49E-01	Breathing rate (normal activity) 7.79E-01
3	Resuspension coefficient (m=2) 4.41E-01	Resuspension coefficient (m=2) 5.73E-01	Resuspension half-life (m=3) -3.58E-01	σ_z-scaling factor -6.47E-01

Since released material consists of plutonium nuclides, parameters related to inhalation shows higher ranks.

Note

*1 PEAR: Pearson's correlation coefficient
SPEA: Spearman's correlation coefficient
PCC: Partial correlation coefficient
PRCC: Partial rank correlation coefficient

*2 Terms regarding resuspension(14) are shown below:

$$Rw(t) = \sum_{m=1}^{3} \left(RC_m \exp(-\lambda_m t) \right)$$

where, Rw(t): Resupsension weathering function, RC_m: Resuspension coefficient [m^{-1}], λ_m: Resuspension half-life [s^{-1}], t: time

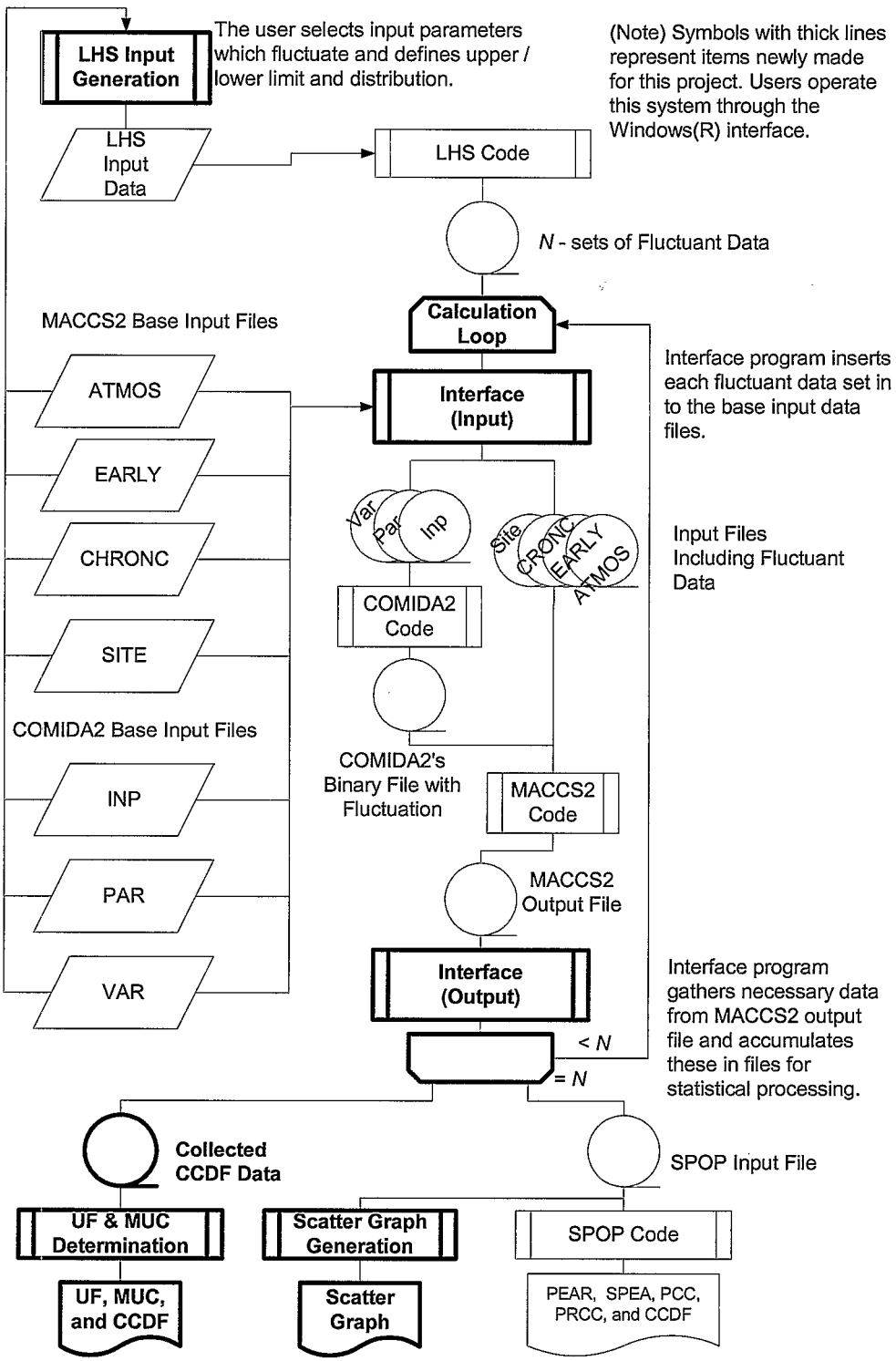


Fig.1 Flow of uncertainty analysis.

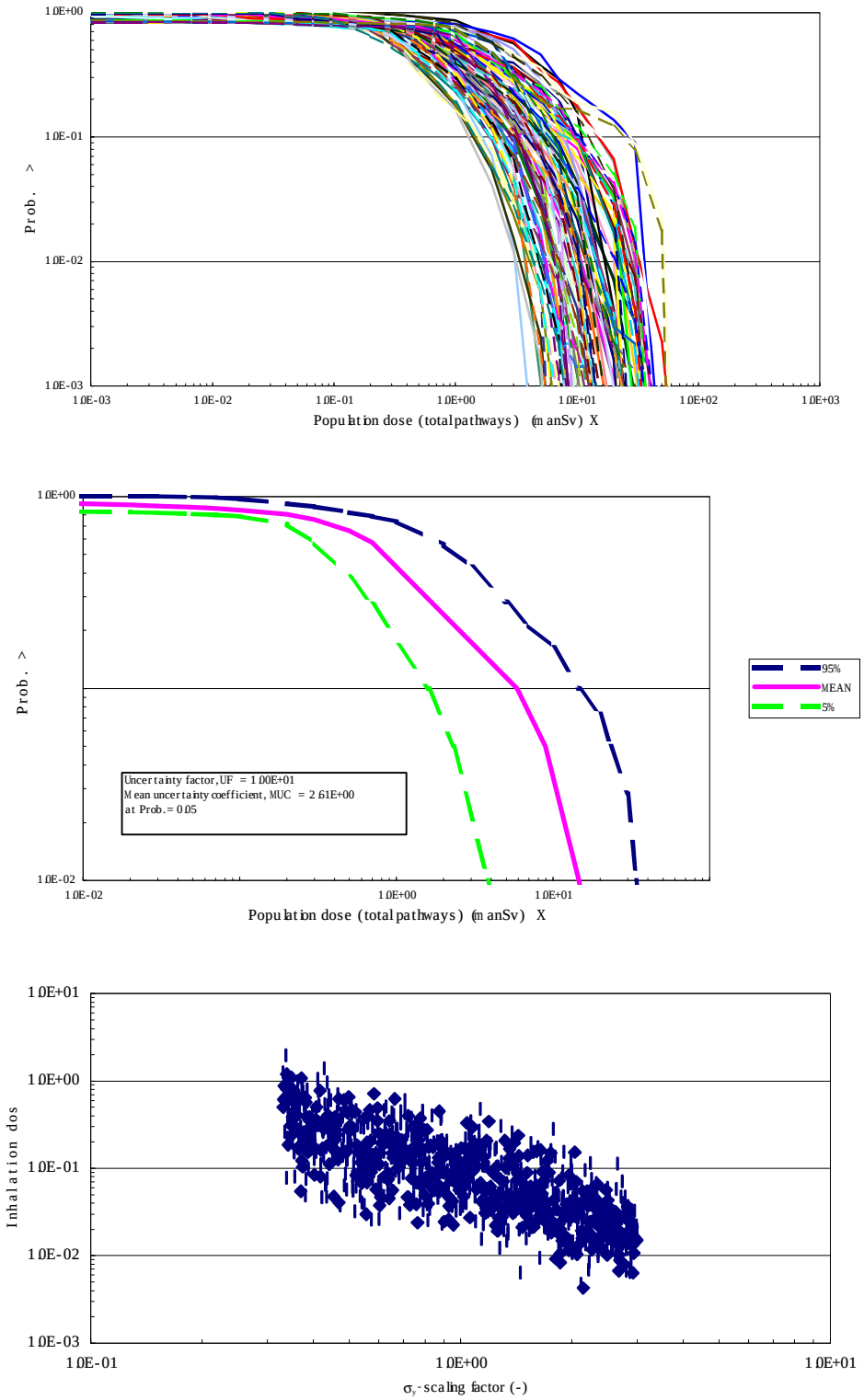


Fig. 2 Result samples.
Criticality event, 1000 sample sets

Acknowledgement

The authors are grateful to Dr. Sigeru Kumazawa from the Institute of Nuclear Safety, NUPEC. His important advice during the implementation of this project has been invaluable.

Reference

(1) Iman et al.: "A FORTRAN 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use with Computer Models", NUREG/CR-3624 (1984)

(2) Saltelli and Homma: "LISA Package User Guide Part II SPOP (Statistical Post Processor), Uncertainty and Sensitivity Analysis for Model Output. Program Description and User's Guide", EUR1392EN (1992)

(3) Statistics Bureau, Ministry of Public Management, Home Affairs, Posts and Telecommunications: "Nihon no Tokei 2000" (2000) (in Japanese)

(4) Statistics and Information Department, Ministry of Agriculture, Forestry and Fishery: "Nousakumotsu Sakugatabetsu Seiiku Stage Souran" (1992) (in Japanese)

(5) Sprung et al.: "Evaluation of Severe Accident Risks: Quatification of Major Input Parameters, MACCS Input", NUREG/CR-4551 Vol.2 Rev.1 Pt.7 (1990)

(6) IAEA: "Handbook of Parameter Values for the Prediction of Radionuclide Transfer in Temperate Environments", Technical Report Series No. 364 (1994)

(7) Helton et al.: Reliability Engineering and System Safety 48, pp.91-127 (1995)

(8) Chain and Young: "Code Manual for MACCS2 Preprocessor Codes COMIDA2, FGRDCF, IDCF2", NUREG/CR-6613, SAND97-0594 Vol.2 (1998)

(9) Proceedings Workshop on Expert Judgment and Accident Consequence Uncertainty Analysis (COSYMA) (1999)

(10) Chain and Young: "Code Manual for MACCS2: User's Guide", NUREG/CR-6613, SAND97-0594 Vol.1 (1998)

(11) Japan Nuclear Fuel Limited: "Saishori Jigyosho Saishori Jigyo Shitei Sinseisho, Tempu Shorui 8, pp. 8-4-19 to 20" (1996) (in Japanese)

(12) Statistics Bureau, Ministry of Public Management, Home Affairs, Posts and Telecommunications: "Kokusei Chosa Chiiki Mesh Tokei (02 Amomori-ken)" (1995) (in Japanese)

(13) Geographical Survey Institute, Ministry of Land, Infrastracture and Transport: "Suchi Chizu Joho" (1989) (in Japanese)

(14) Jow et al.: "MELCOR Accident Consequence Code System (MACCS) Model Description", NUREG/CR-4691, SAND86-1562 Vol.2 (1990)

OSCAAR Development and Applications

Toshimitsu HOMMA

Japan Atomic Energy Research Institute
Tokai Research Establishment
Tokai, Ibaraki 319-1195, Japan
e-mail: homma@popsvr.tokai.jaeri.go.jp

Abstract

OSCAAR is an accident consequence assessment code designed for use in probabilistic safety assessment (PSA) of nuclear reactors in Japan. As part of the verification and validation study, uncertainty and sensitivity analysis with OSCAAR is being performed to provide information on the range of the uncertainty associated with the PSA level 3 outputs, and identify the important variables that contribute most to the uncertainty of the predicted outputs. Within two types of uncertainty involved in accident consequence assessments, the study on the stochastic uncertainty has been first undertaken. Then, the preliminary study on the subjective uncertainty has been also performed to address the uncertainties in the predicted consequences caused by the uncertainties in the values of the input parameters.

1. Introduction

An accident consequence assessment code, OSCAAR, has been developed within the research activities on PSA at the Japan Atomic Energy Research Institute. OSCAAR is primarily designed for use in PSA of nuclear reactors in Japan. OSCAAR calculations, however, can be used for a wide variety of applications including siting, emergency planning, and development of design criteria, and in the comparative risk studies of different energy systems.

After the international exercise organized by NEA and CEC, our efforts are mainly made upon the verification and validation of the individual modules and the whole OSCAAR system. OSCAAR was applied to the Dose Reconstruction scenarios of the BIOMASS Theme 2 organized by IAEA for the validation of the atmospheric dispersion and deposition, and food chain transport modules. As part of the verification study, uncertainty and sensitivity analysis with OSCAAR is being performed to provide information on the range of the uncertainty associated with the PSA level 3 outputs, and identify the important variables that contribute most to the uncertainty of the predicted outputs. Within two types of uncertainty involved in accident consequence assessments,

the study on the stochastic uncertainty has been first undertaken. The meteorological sampling scheme appropriate for the trajectory dispersion model in OSCAAR has been developed for use in accident consequence assessments. Then, the stochastic uncertainties associated with the meteorological sampling scheme have been quantified. The preliminary study on the subjective uncertainty has been also performed to address the uncertainties in the predicted consequences caused by the uncertainties in the values of the input parameters.

2. Development of OSCAAR

As shown in Figure 1, OSCAAR consists of a series of interlinked modules and data files that are used to calculate the atmospheric dispersion and deposition of selected radionuclides for all sampled weather conditions, the subsequent dose distributions and health effects in the exposed population. Several stand-alone computer codes and databases can also be used to prepare, in advance, necessary input data files for OSCAAR.

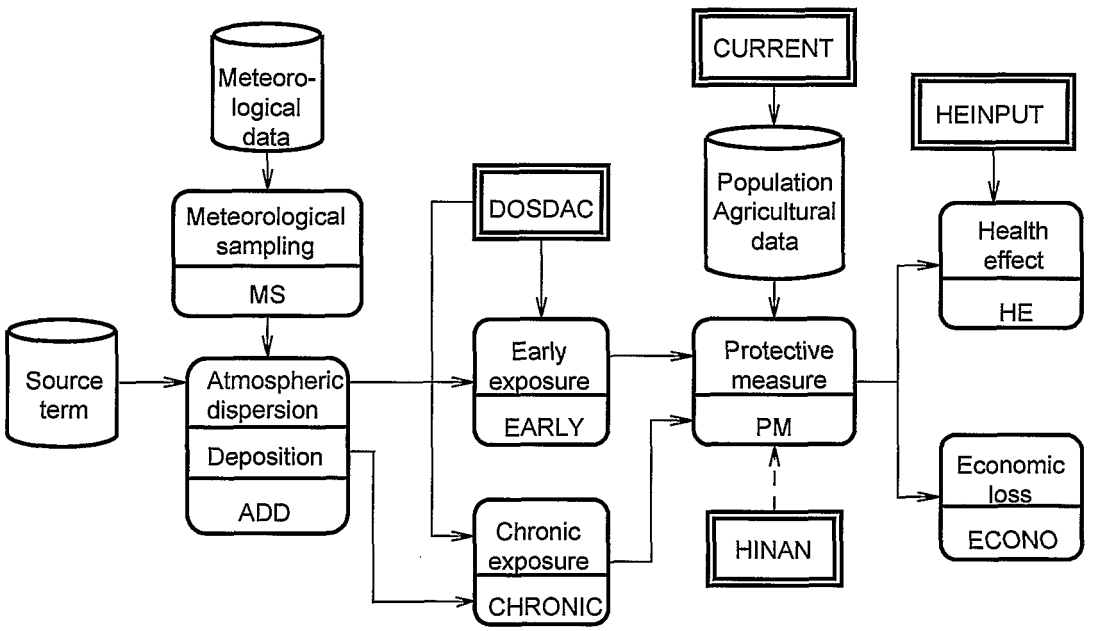


Figure. 1 Schematic representation of the OSCAAR code system

Recent development of OSCAAR has been devoted to the revision of these databases and input files. The dose conversion factors to translate the activity in the environment to an equivalent dose to humans are provided in OSCAAR by the pre-processor code, DOSDAC code system. After the publication of ICRP 30, age-specific biokinetic models were given in ICRP publication 56, 67 and 69. ICRP Publication 66 also gave a revised kinetic and dosimetric model of the human respiratory tract. Those

recent models are included in DOSDAC to produce the most recent set of dose conversion factors used in acute and chronic conditions.

Another recent development of OSCAAR was automated capability to implement the uncertainty and sensitivity analysis (UA/SA) using OSCAAR. OSCAAR has been coupled with the UA/SA techniques to quantify uncertainty associated with accident consequence assessment and to identify uncertain processes and important parameters contributed to consequences. The software package PREP/SPOP is used to allow for an automatic performance of all necessary steps in uncertainty and sensitivity analysis. The different sampling schemes, such as pure random, Latin hypercube (LHS) and quasi-random sampling, are allowed to be used in PREP (Homma and Saltelli, 1992). SPOP performs the uncertainty and sensitivity analyses on the output of the model. The objective of the uncertainty analysis is to quantify the model output statistics: output mean, confidence bounds and percentiles. The sensitivity of the output to the input parameters is investigated by examining the relationships between input and output variables. SPOP includes several parametric and non-parametric techniques, based on regression-correlation measures, as well as some "two-sample" tests; variance-based methods (Saltelli and Homma, 1992). Variance-based methods are also available for ascertaining if a subset of input may account for the output variance without any linearity assumptions (Homma and Saltelli, 1996).

3. Model Validation

For the validation of OSCAAR, the OSCAAR-CHRONIC module has been applied to the Chernobyl scenario (Scenario A4) of BIOMOVS (BIOspheric Model Validation Study) Phase I, which starts with daily concentrations of I-131 in air and requests the prediction of concentrations of I-131 in vegetation and milk for several locations in the northern hemisphere (Peterson et al., 1996). The performance of other OSCAAR modules such as ADD, EARLY, and CHRONIC was examined in the Hanford and Iput test scenarios of BIOMASS by implementing atmospheric dispersion and deposition calculations, food chain transport analysis, and dose calculations (Homma et. al., 2000a; Homma and Matsumaga, 2001).

The Hanford test scenario was an acute, inadvertent release of I-131 (2.66 10^{12} Bq) from the 60 m stack of the Hanford Purex Chemical Separations Plant that occurred on Sept 2-5, 1963. Monitoring data were collected in nine counties in the northwestern United States over the two-month period following the release. The assessment tasks of this model test were to compare the release dispersion patterns in the Hanford region that were predicted by the model, to compare predictions of I-131 body content and concentrations in environmental materials with observed values for several locations, and to analyze the radiation doses from I-131 that are predicted by the model.

The expected uncertainties in the predicted results were also estimated using parameter uncertainty analysis with PREP/SPOP. The analysis included 14 uncertain input parameters, covering the atmospheric dispersion and deposition model, and the food chain model. In this analysis, 100 sets of input parameter values were generated by PREP, and 100 runs of the OSCAAR code were performed to provide 100 estimates of total deposition, time-integrated concentrations of I-131 in pasture grass and milk, and dose from several exposure pathways at six locations. Comparison of model predictions with observations was possible for only a few endpoints including air sample measurements, concentrations in vegetation and milk and thyroid burden in two specified individuals.

The predicted I-131 air concentrations by OSCAAD-ADD were compared with air measurements for twenty-one locations provided in the Hanford scenario. Since the spatial and temporal variations of air concentrations of I-131 showed a complicated pattern, the ADD transport and dispersion calculations made using wind data at the release height and wind fields by simple interpolation of the surrounding surface wind data indicated limited capabilities. While ADD predicted well along a line approximately northwest to southeast, the high overpredictions were found to the east of the release point.

The predictions of I-131 time-integrated concentrations in milk at six locations are given in Figure 2 in the form of boxplots in which those measurements for four locations are also included. The boxplots show the 1st, 5th, 25th, 50th (median), 75th, 95th, and 99th percentile of the predicted values. Additionally, the mean of the predictions is shown by the square symbol as well as the minimum and maximum predictions. The 95 percent subjective upper confidence level on the milk is 18 (at Pasco) or more times larger than the 5 percent subjective lower confidence level. The predictions of I-131 concentration in milk seem to be in better agreement with measurements at Farm A, Farm B and Pasco (the test sites southeast of the release) except at Ringold (east of the release). This may be mainly due to the overpediction of total deposition of I-131 in the case of

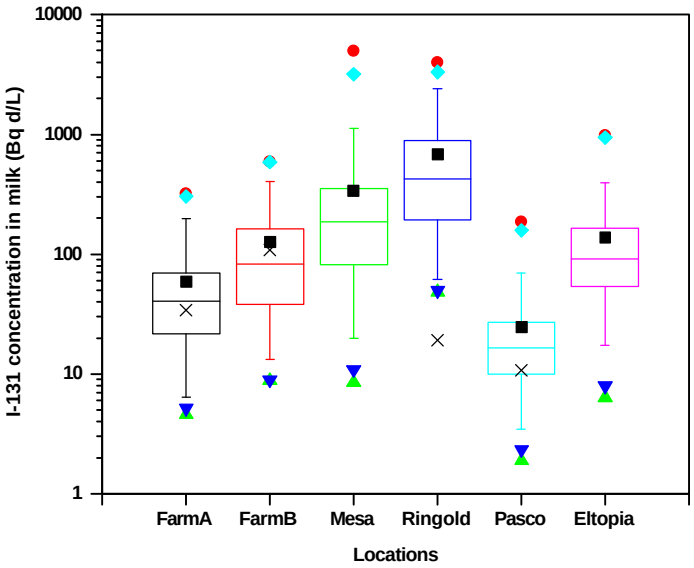


Figure 2. Boxplots of I-131 concentrations in milk at several locations

Ringold.

The Hanford test scenario provided a good opportunity to evaluate the performance of OSCAAR. Although it is difficult to perform a model validation over the entire set of conditions to which accident consequence assessment codes like OSCAAR may be applied, the Hanford test scenario is valuable because we can start with source terms and examine atmospheric dispersion and deposition calculations, food chain transport analysis, and dose calculations. The OSCAAR food chain model performs relatively well when the predictions of deposition are well.

4. Meteorological sampling

One of the important parameters, which influence the magnitude of the consequences following an accidental release of radionuclides to the air, is the sequence of weather conditions that is encountered by the dispersing plume. A number of meteorological sampling methods, such as purely random, cyclic and stratified sampling methods are widely utilized in accident consequence codes, but they were mainly related to codes using the straight-line dispersion model. The purposes of this study were to design an appropriate meteorological sampling scheme for use with the puff trajectory dispersion model used in OSCAAR and to find the statistical variability in the probability distribution of the consequences, resulting from the adoption of different meteorological sampling schemes (Homma et. al., 2000c).

In developing the new sampling scheme, all possible sequences of weather conditions from one year of hourly meteorological data, in other words, 8760 starting times for the sequences were selected to produce the possible probability distribution of the consequences. Since the details of the meteorological sampling scheme are more important when large releases and effects involving dose thresholds are considered, the source term including considerable large releases was used for the study.

Table 1. Meteorological sampling scheme used in OSCAAR

Initial wind direction	9 groups(NNE, NE, ENE-E, ESE-SE, SSE-S, SSW-W, WNW, NW, NNW-N)
Wet sequences: Rainfall within 10 km	 2 groups (0-5, >5 mm)
Dry sequences: Mean stability categories to 10 km Travel time to 20 km	 3 groups (A-C, D, E-G) 3 groups (< 2.5, 2.5 - 5, > 5 hours)

From the results of the sensitivity analysis, the new stratified sampling scheme was

designed to contain a set of options for identifying "wet" and "dry" sequences. The scheme designed included the nine groups of initial wind direction with two groups for wet conditions and nine for dry. The wet sequences were specified in terms of the total amount of rain within the distance of interest. The dry sequences were described in terms of the mean stability category and the travel time to a selected distance. The scheme used in this analysis is summarized in Table 1. From a total of ninety-nine groups, the representative sample of 144 sequences was selected for this analysis by choosing at least one sequence per 100 sequences at random from each group.

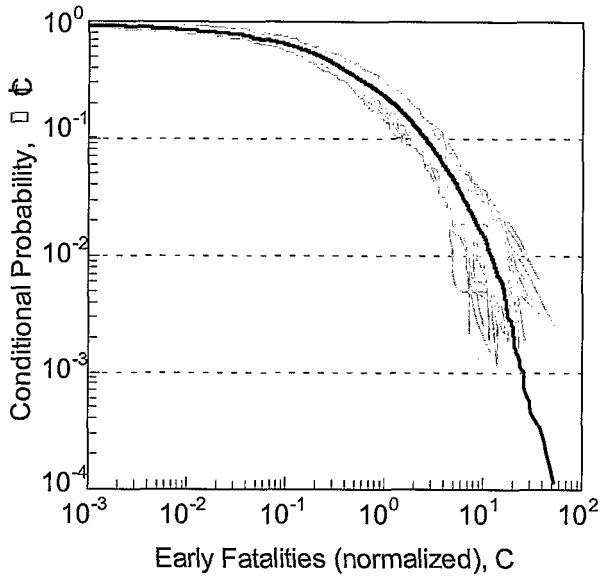


Figure 3. Performance of new stratified sampling scheme

In order to verify the performance of the new stratified sampling scheme, 1000 different sets of complementary cumulative distribution function (CCDF) of early health effects with 144 weather sequences were produced by selecting different sets of weather sequences from the stratified sampling groups. Figure 3 shows the comparison of the results from these sampling schemes with a "real" CCDF calculated from a set of whole 8760 sequences. The statistical variability defined by a ratio of 95% bound to the mean value at the 99 percentile was calculated by a factor of about 1.6 from these CCDF.

5. Level 3 PSA for Internal Events of the Model Plant

The objectives of the level 3 PSA of the model plant at JAERI are to find needs for further improvement of risk assessment methodology and to obtain a better understanding of the controlling factors of public risk. Other recent important objective is to provide inputs for discussions by the Nuclear Safety Commission on safety goal, effectiveness of emergency planning, siting criteria etc. In order to obtain an estimated profile of consequences for various accident conditions, the consequence assessments have been made of postulated core damage accidents with source terms derived from a generic level 2 PSA of the model plant (Homma et. al., 2000b). The model plant selected is a BWR5 with Mark-II type containment, which is a typical nuclear power plant in Japan. It should be noted, however, that the analysis does not fully consider accident management measures currently under preparation in Japan.

Based on the results from accident progression analyses, the core damage sequences were categorized into five groups by the similarity in timing of containment overpressure failure after core melts initiation. The source terms used in the analysis have been calculated by a severe accident analysis code, THALES-2 for each represented sequence from these five core damage accident sequences (Ishikawa et. al., 2002). Each sequence was considered to have four different release modes that included the early containment failure due to energetic events, overpressure failure mode, the release control mode by containment venting, and the accident termination mode by containment spray. The off-site consequences have been assessed for 144 different weather sequences taking into account the countermeasures by an accident consequence assessment code, OSCAAR. The countermeasures considered for the populations were sheltering in their home, sheltering in the specified concrete building followed by evacuation, and relocation.

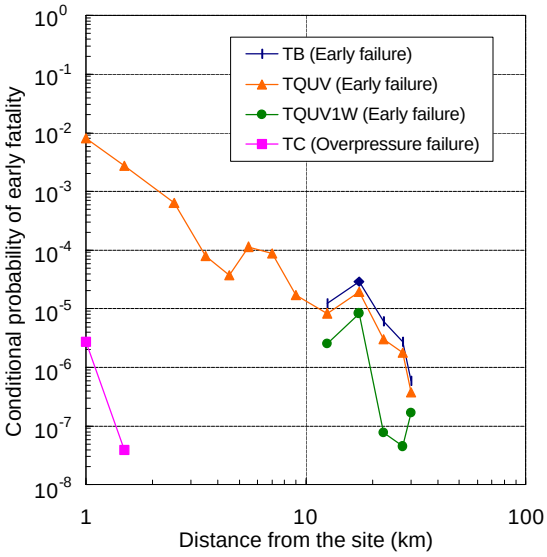


Figure 4. Conditional probability of early fatality

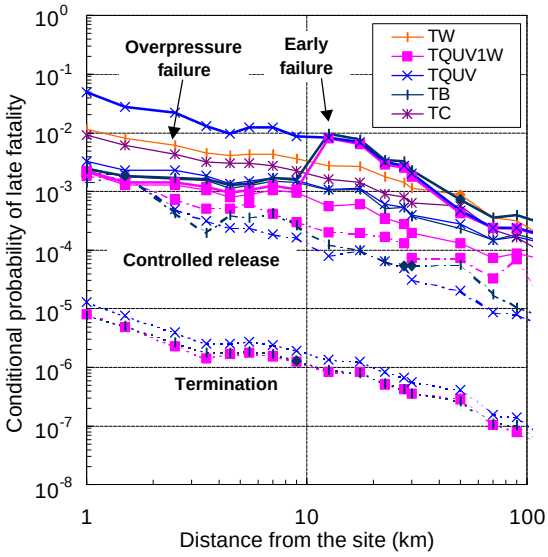


Figure 5. Conditional probability of latent cancer fatality

Figures 4 and 5 show the conditional probabilities of both early fatality and late cancer fatality as a function of distance from the release point for various core damage sequences. Early fatalities appear only for an early containment failure sequence. Because of its threshold, the curve of early fatality is strongly dependent on the distance from the release point. The curves of late fatality depend strongly on contaminated failure mode and decrease with a less pronounced slope. Considering that the level 2 PSA for the reference plant provided a core damage frequencies of 5×10^{-7} per reactor year and containment failure frequency of 1.7×10^{-7} per reactor year, the risks of both early fatality and late cancer fatality to the average individuals in the vicinity of the plant would be much less than the quantitative health objectives of the USNRC. The results also indicated

that collective dose was dominated by the long-term exposure from the ground and food contamination and their magnitude was strongly dependent on releases of iodine and cesium. On the other hand, early doses were governed by both external irradiation and inhalation from the radioactive cloud.

6. Uncertainty and sensitivity analysis using OSCAAR

In order to verify the systematic methodology of implementing the uncertainty and sensitivity analysis using OSCAAR, a preliminary analysis for the uncertainties in the predicted consequences caused by the uncertainties in the values of the input parameters were carried out (Homma et. al., 2002). Attention focused on the risk of early health effects in the analysis because a number of previous analyses indicated that the predictions of early health effects were much more sensitive to differences in modeling assumptions and parameter values than the predictions of other endpoints such as population dose and number of late health effects

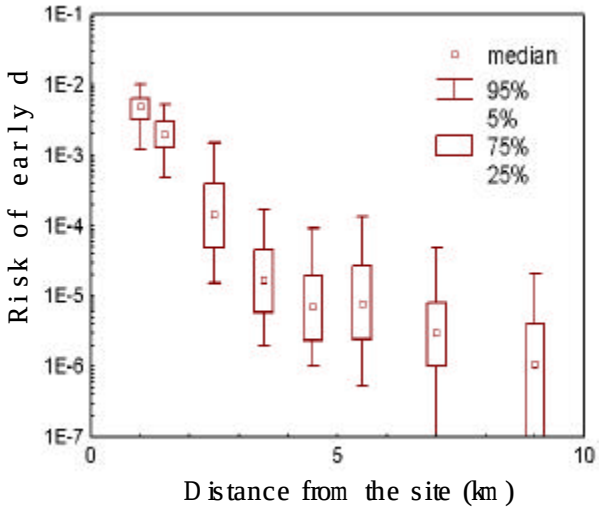


Figure 6. Distribution of risk of early death

The release considered was due to an early containment failure, which has been found to dominate the predicted risk of early health effects from all accident sequences considered in our consequence analysis. We consider a total of 23 uncertain input parameters describing the dispersion and deposition of radionuclides in the atmosphere, dose received to the public and resulting early health effects. Because of the preliminary nature of this analysis, the ranges of parameter values adopted in the analysis were based on the literature review, only both uniform and log-uniform probability distributions are used, and no correlation between parameters is considered.

A total of 128 runs of the OSCAAR code was performed to assess the effect of uncertainties in the 23 input parameters on the endpoints. Each of 128 runs in the uncertainty analysis used the same 144 weather conditions. The quantities considered in the uncertainty and sensitivity analysis were the expectation values and some of the percentiles of the CCDFs for the endpoints. The endpoint of interest here was the population weighted individual risk of early death at the different distances from the release point.

Figure 6 shows the resulted distributions of this endpoint in the form of box plots. The expectation value of the individual risk of early death sharply decreases with increasing the distance from the site. The uncertainty factor defined by the ratio of the 95th to the 5th percentiles on that risk was about a factor of eight close to the site. It increased with increasing distance from the site. This reflected the larger uncertainty where doses were just above or below the threshold for early death.

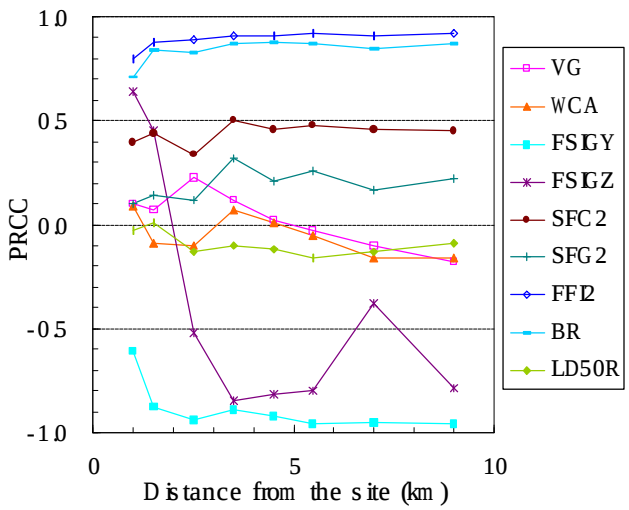


Figure 7. PRCC for each parameter

Among a number of the sensitivity indicators in SPOP, the partial rank correlation coefficients (PRCCs) were calculated to provide a measure of the relationship between the input and output values. The coefficient of determination R^2, which indicates how much of the variation of the output values can be explained by a linear (monotonic) relationship between the output and input values, was always above 0.95 at all distances in this analysis. As shown in Figure 7 the parameters whose uncertainties made important contributions to the uncertainties in the individual risk of early death were the factors for both horizontal and vertical dispersion, the filtering factor for inhalation in concrete building, and the breathing rate. The analysis for the 99th percentile of the CCDFs for this endpoint showed the similar results. Considering the inhalation dose in the passing plume was dominant to the risk of early death for this source term, it seemed reasonable to identify the filtering factor for inhalation and the breathing rate as important sources of uncertainty. The dispersion parameters also made a major contribution to the uncertainty in the values of individual risk, but the parameters connected with the dose-response relationship for early health effects did not make any contribution to the uncertainty.

Reference

Homma, T., Inoue, Y., and Tomita, K. (2000a). OSCAAR calculation for the Hanford dose reconstruction scenario of BIOMASS Theme 2, JAERI-Research 2000-049.

Homma, T., Ishikawa, J., Tomita, K. and Muramatsu, K. (2000b). Radiological consequence assessments of degraded core accident scenarios derived from a generic Level 2 PSA of a BWR, JAERI-Research 2000-060, in Japanese.

Homma, T., Liu, X. and Tomita, K. (2000c). Uncertainty and sensitivity of accident

consequence assessments on meteorological sampling schemes. *Proceedings of the 5th International Conference on Probabilistic Safety Assessment and Management (PSAM5)*, 2753-2758, Nov 27- Dec 1, 2000, Osaka,

Homma, T. and Matsunaga, T. (2001). OSCAAR calculations for the Iput dose reconstruction scenario of BIOMASS Theme 2, JAERI-Research 2000-059.

Homma, T. and Saltelli, A. (1992). LISA package user guide Part I, PREP Preparation of input sample for Monte Carlo simulations, EUR-13922, CEC, Brussels.

Homma, T. and Saltelli, A. (1996). Importance measures in global sensitivity analysis of nonlinear models. *Reliab. Engng. Syst. Safety* **52**, 1-17.

Homma, T., Tomita, K. and Inoue, Y. (2002). Uncertainty and sensitivity analysis using an accident consequence assessment code OSCAAR. *Proceedings of the 6th International Conference on Probabilistic Safety Assessment and Management (PSAM6)*, June 23-28, 2002, Puerto Rico.

Ishikawa, J., Muramatsu, K. and Sakamoto, T. (2002). Systematic source term analyses for Level 3 PSA of a BWR with Mark-II type containment with THALES-2 code, *Proceedings of the 10th International Conference on Nuclear Engineering (ICONE10)*, Arlington, April 14-18, 2002.

Saltelli, A. and Homma, T. (1992). LISA package user guide Part III, SPOP Uncertainty and sensitivity analysis for model output, EUR-13924, CEC, Brussels.

Research Reactor Accidents: Analysis and Impacts

Ioannis G. Kollas
NCSR "Demokritos"
Institute of Nuclear Technology and Radiation Protection
153 10 Aghia Paraskevi, Attiki, Greece

Abstract

An assessment of the environmental and health impacts that would result from the accident spectrum of a research reactor is performed. As a reference case the Greek research reactor, a typical 5 MW open-pool type, light water moderated and cooled research reactor is employed. The reactor is currently operating with a mixed core of low enrichment uranium fuel containing 20% of U^{235}, and high enrichment uranium fuel containing ~93% of U^{235}. The reactor is located within the limits of Athens the capital of Greece, a large population center with almost three and a half million inhabitants. The evaluation of the safety of the reactor is based on the analysis of the response of the reactor to postulated disturbances, malfunctions, failures of equipment, or operator errors. The analysis focuses on the risks stemming from the design basis accident, a loss of coolant accident that is considered the most serious accident for the Greek research reactor. The estimation of accident consequences is performed with the MACCS2 code. Collective exposure and latent health effects, individual doses and individual cancer risks and environmental impacts are estimated to a distance of 32 km from the reactor site. The results of the analysis indicate that the design basis accident would induce consequences of a non-trivial but rather limited magnitude with the current operating schedule, and with either the high or low enrichment uranium fuel core, while the other less serious accidents would induce consequences that would be of a rather trivial nature. No early fatalities or injuries would be expected from any of the accidents.

1. Introduction

The Greek Research Reactor (GRR) is a typical 5 MW open-pool type, light water moderated and cooled research reactor with MTR type fuel elements. GRR is located within the limits of Athens city the capital of Greece, at the foot of Imittos Mountain in the Athens suburb of Aghia Paraskevi, Fig. 1, about 8 km from the city center. The reactor is currently operating with a mixed core of low enrichment uranium (LEU) fuel containing 20% of U^{235}, and high enrichment uranium (HEU) fuel containing ~93% of U^{235}.

In the past an accident consequence analysis of GRR, which was fueled at that time entirely by HEU fuel, was performed using the CRAC2 code, based on limited accident sequences, an old demographic record, and on meteorological data from the National Observatory of Athens located at a distance of about 10 km from the reactor site(1).

The present analysis is based on the Safety Analysis Report of GRR(2). The yearly meteorological record that is employed in the calculations is a GRR site meteorological record that has been set up by appropriate processing of the data produced by the instrumentation of the meteorological mast installed on the "Demokritos" premises. The demographic data that were utilized, were drawn from the 1991 national census, and the area that is covered in the consequence analysis, i.e. the greater Athens region, includes over 3,400,000 inhabitants, and

extents to a distance of 32 km from the reactor site. The estimation of accident consequences was performed by the MACCS2 code[3,4].

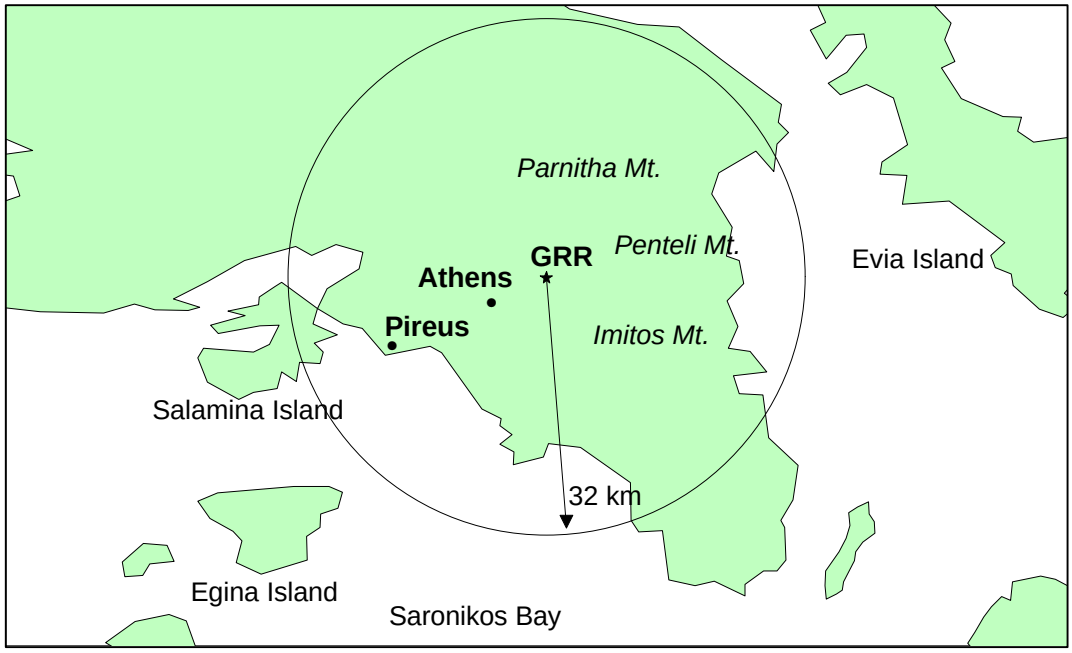


Fig. 1 Athens Area with the GRR Site and Range of Analysis

2. Accident Analysis

The evaluation of the safety of GRR is based on the analysis of the response of the reactor to postulated disturbances, and to postulated malfunctions, failures of equipment, or operator errors. The accidents studied are classified into groups that ensure the consideration of a large spectrum of appropriate accidents[2]. The following four categories of GRR accidents were analyzed:

- Decrease in heat removal accidents (loss of flow accidents),
- Reactivity accidents,
- Loss of coolant accidents, and
- Minor accidents.

In the first category the following accidents are included: (a) Primary Pump Failure, (b) Safety Flapper Failure, (c)Valve Failure, (d) Partial Flow Blockage to Fuel Assemblies, (e) Loss of Flow in the Secondary Cooling, (f) Loss of Secondary Coolant, and (g) Cooling Tower Air Blowers Failure.

Among these accidents the Partial Flow Blockage to Fuel Assemblies accident has a relatively high probability of occurrence, and has already occurred in at least three reactors of the same type[5]. This accident is considered the most serious credible accident not only in this group, but also among all credible accidents. This kind of accident could occur if a pliable object is accidentally introduced into the reactor pool and by being deposited on top of a fuel element, blocks coolant flow through it. When such a coolant flow blockage accident (CFBA) occurs, only a part of the elements involved melt in practice, before the negative reactivity inserted into the core decreases the reactor power drastically[2].

The estimation of the source term of this accident is based on realistic assumptions about first the potential initiation scenarios, and second radioactive releases into the environment. The following assumptions are implemented:

- Due to melting 50% of fuel failure occurs in 6 of the total 32 fuel elements of the core, that contain 18.75% of the fission product core inventory.
- This failure corresponds to a net 9.375% of the fission product core inventory, from which 100% of noble gases, 50% of iodines and 1% of other fission products are released into the pool water.
- 100% of noble gases and 10% of iodines and other fission products are released from the pool water into the reactor building.
- Fission products are released into the atmosphere within 16 hours through a 50m stack, after passing a system of absolute and activated charcoal filters with a retaining efficiency of 90% for iodines, 95% for particulates, and 0% for noble gases.

The eventual release fractions of the total GRR inventory for the CFBA are as follows:

Noble gases:	$9.375 \cdot 10^{-2}$
Iodines:	$4.6875 \cdot 10^{-4}$
Other fission products:	$4.6875 \cdot 10^{-6}$

In the second group of reactivity accidents the following are included: (a) Maximum Start-up Accident, (b) Maximum Reactivity Insertion Accident, (c) Loading Accident, (d) Beam Tube Flooding, (e) Removal of In-core Experiments, (f) Failure of In-core Experiments, (g) Control Rod Failures, (h) Control System Runaway, (i) Movement of Core against Thermal Column, and (j) Excursion Following Decay of Xe-135 Transient.

Among these accidents the Maximum Reactivity Insertion Accident is the most serious leading at maximum to the melting of less then 2% of the fuel plates[2], a figure quite smaller then the corresponding figure of the CFBA. Thus CFBA impacts would be quite larger then impacts induced by this reactivity accident.

The third group of loss of coolant accidents consists of the following: (a) Rupture of the Reactor Pool Wall, (b) Rupture of a Beam Tube or of the Thermal Column, (c) Damage in the Cooling System, and (d) the Design Basis Accident, a typical loss of coolant accident (LOCA).

Among the accidents considered in the safety analysis report this LOCA that would lead to a guillotine break of the largest pipe connected to the bottom of the reactor pool, and would result in the melting of 19.75% of the core is considered to be the most severe accident in respect to potential consequences for the population residing in and around Athens. This accident is taken as the Design Basis Accident (DBA) of GRR, and it delineates the radiation risks of all other accidents considered. The accident is postulated to occur when the reactor is in full power and the pool water at its minimum level of 8.5 m. At the time of the rupture it is assumed that the reactor is tripped and the only power generation is due to decay heat. The tank water level starts to decrease due to water leak from the ruptured pipe. The minimum time for total core uncovering is estimated to be of the order of 1400 s[6]. Around this time the core temperature starts to increase substantially since the core is practically cooled only by air. The air-cooled reactor core exchanges heat through its outer surfaces with the surrounding air and structures by natural convection and radiation mechanisms respectively. The closely packed fuel elements are exchanging heat by air conduction and heat radiation, whereas a natural air stream is developed through the hot fuel channels due to buoyancy. The time of

core melt occurrence is conservatively estimated to be about 4000 s[7], and calculations suggest an upper core melt limit of 19.75%. Additional calculations show that some efficient core cooling is needed for the first four hours after the initiation of the accident to avoid melting of the core. The evaluation of the source term for this accident is based on conservative estimations of fission product releases to the reactor-operating floor[8], and further on conservative assumptions on the effectiveness of filter mitigation and release into the environment. Specifically radionuclides from the 19.75% melted part of the core are first released into the reactor building in the following fractions: Noble gases: 100%, halogens: 50%, and all other fission products: 1%. Furthermore filter mitigation is ignored and all radionuclides released into the reactor building, escape into the environment through leaks of the reactor building within four hours.

The eventual release fractions of the total GRR inventory for this accident are as follows:

- Noble gases: $1.975 \cdot 10^{-1}$
- Halogens: $9.875 \cdot 10^{-2}$
- Other fission products: $1.975 \cdot 10^{-3}$

The above release fractions can be considered as an upper limit of releases.

Finally among the minor accidents group the following are included: (a) Fuel Element Cladding Failure, (b) Loss of Ventilation in Beam Tubes and Thermal Column, (c) Loss of Primary Electric Power, (d) Criticality in Fuel Storage, and (e) Deterioration of Primary Coolant Quality.

None of these minor accidents would produce non-trivial releases.

To assess the source term of GRR accidents, thirty-five isotopes for the LEU core, and thirty-one isotopes for the HEU core, the most important from a radiological point of view, are taken into account in the estimation of the reactor inventory[8]. The operating schedule of GRR amounts to 8 hours/day for 5 days/week. The release characteristics and release fractions of the two accidents considered in detail in the consequence analysis are presented in Table 1.

3. Consequence Analysis

In the estimation of GRR accident consequences some additional assumptions are made. First an exclusion area of 400 m around the reactor is established, excluding thus the personnel of the reactor and of the "Demokritos" research center where the reactor is sited, from the calculations. Second no emergency measures, that would mitigate the consequences of the accidents during their progress, are taken into account.

Consequence estimations are based on both early and chronic exposure. Health and environmental impacts are estimated to a distance of 32 km from the reactor site, covering thus the whole extent of Athens and its surroundings.

Table 1. Accident Characteristics and Release Fractions

Accident Identification	LOCA	CFBA
Release characteristics		
Frequency of release (yr^{-1})	$<10^{-8}$	$4.88 \cdot 10^{-4}$
Number of segments released	1	1
Time of release (hr)	1.0	2.0
Release duration (hr)	4.0	16.0
Release height (m)	0.0	50.0
Release energy (MW)	0.0	0.0
Release fractions		
Xe/Kr group	$1.975 \cdot 10^{-1}$	$9.375 \cdot 10^{-2}$
I group	$9.875 \cdot 10^{-2}$	$4.688 \cdot 10^{-4}$
Cs group	$1.975 \cdot 10^{-3}$	$4.688 \cdot 10^{-6}$
Te group	$1.975 \cdot 10^{-3}$	$4.688 \cdot 10^{-6}$
Sr group	$1.975 \cdot 10^{-3}$	$4.688 \cdot 10^{-6}$
Ru group	$1.975 \cdot 10^{-3}$	$4.688 \cdot 10^{-6}$
La group	$1.975 \cdot 10^{-3}$	$4.688 \cdot 10^{-6}$
Ce group	$1.975 \cdot 10^{-3}$	$4.688 \cdot 10^{-6}$
Ba group	$1.975 \cdot 10^{-3}$	$4.688 \cdot 10^{-6}$

The analyses have shown that in most cases the LEU core accident consequences are very close to the corresponding consequences for the HEU core, and therefore in the following figures only the LEU core results are presented. Exceptions to this are impacts concerning mainly bone effects (and to a lesser extent lung effects), which are significantly affected by the Pu isotopes buildup in the LEU core, and their release into the environment. The accident releases for the HEU core do not practically include any substantial amounts of Pu isotopes, due to the high concentration of U^{235} (~93%) in the reactor core, and the absence therefore of significant amounts of U^{238} that would eventually produce Pu isotopes.

Figs. 2a and 2b present the Cs^{137}, Sr^{90}, and I^{131} centerline air concentrations for the LOCA and CFBA accident scenarios respectively versus distance from GRR. Figs. 3a and 3b present the Cs^{137}, Sr^{90}, and I^{131} centerline ground concentrations for the LOCA and CFBA accident scenarios respectively versus distance from GRR, including also the total centerline ground concentration. These concentrations are due to the passage of the radioactive cloud, and they refer to the early phase of the accident.

Health impacts include early and latent cancer fatalities and cancer injuries, risks of latent cancer death and cancer injury, whole body and organ doses of an individual at different distances from the reactor site, collective population doses, etc. The analyses performed have shown that no early fatalities or early injuries would be expected among the Athens region population from any of the accidents studied.

Fig. 4 presents the centerline whole body and thyroid lifetime doses versus distance from GRR, while Fig. 5 the centerline cancer fatality risk versus distance from GRR, for both accidents. Finally in Table 2 and for the LEU core, some early, late, and total health

consequences and collective doses, are presented, for the 3,414,000 inhabitants of the greater Athens region.

In Table 3 some consequences for the LOCA are presented for both the LEU and HEU cores, were it is clear that with the exception of bone consequences, which are significantly affected by the presence of Pu isotopes and differ by more than 20%, most values of the other consequences are very close to each other and differ at most by 1-2% percent or concerning lung effects by about 3.5%.

Concerning environmental consequences that would be induced by the DBA and the CFBA it is apparent from Figs. 2a, and 2b that air concentrations stemming from the DBA-LOCA are almost three orders of magnitude larger then the concentrations induced by the CFBA. Ground concentrations, Figs. 3a, and 3b, expose a larger difference at close range, where the differences are almost four orders of magnitude. Taken that all meteorological sequences employed in the consequence analysis are exactly the same for both DBA-LOCA and CFBA, and that the difference in the source terms of the two accidents are about two and a half orders of magnitude (see Table 1), the additional difference in air and ground concentration near the reactor is due to the different release characteristics of the two accidents, and mainly to the ground release for the DBA-LOCA, in respect to the 50 m stack release for the CFBA. Indeed since all environmental releases are scaled by the same ratio for each radionuclide released, radionuclide concentrations and deposition would follow the same pattern if the release characteristics were the same. The difference is more profound in the case of ground concentrations, were the ground release of the DBA-LOCA produces higher ground concentrations near the reactor. This is also the reason that individual life-time doses and cancer fatality risk are differing by more then three orders of magnitude near the reactor, and by slightly more then two orders of magnitude at longer distances.

Concerning now the overall results that apply to the whole range, i.e. population health consequences, collective dose, and population-weighted risk, Table 2, the difference between the DBA-LOCA and the CFBA results are about two and a half orders of magnitude, reflecting the difference in the source terms as expected, since when doses are relatively small and only latent health effects are expected, and in the absence of any countermeasures, the pattern of proportionality is also true for health effects. In such case health effects are directly proportional to dose, and doses in turn to radionuclide concentrations.

Fig. 2a Cs^{137}, I^{131} and Sr^{90} Air Concentrations, LOCA (LEU)

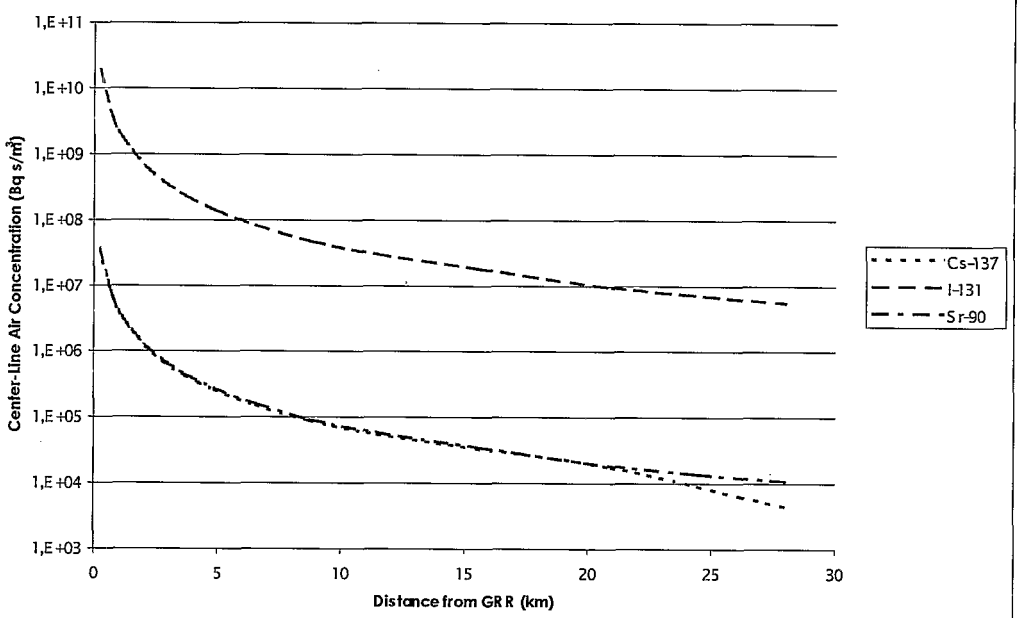


Fig. 2b Cs^{137}, I^{131} and Sr^{90} Air Concentrations, CFBA (LEU)

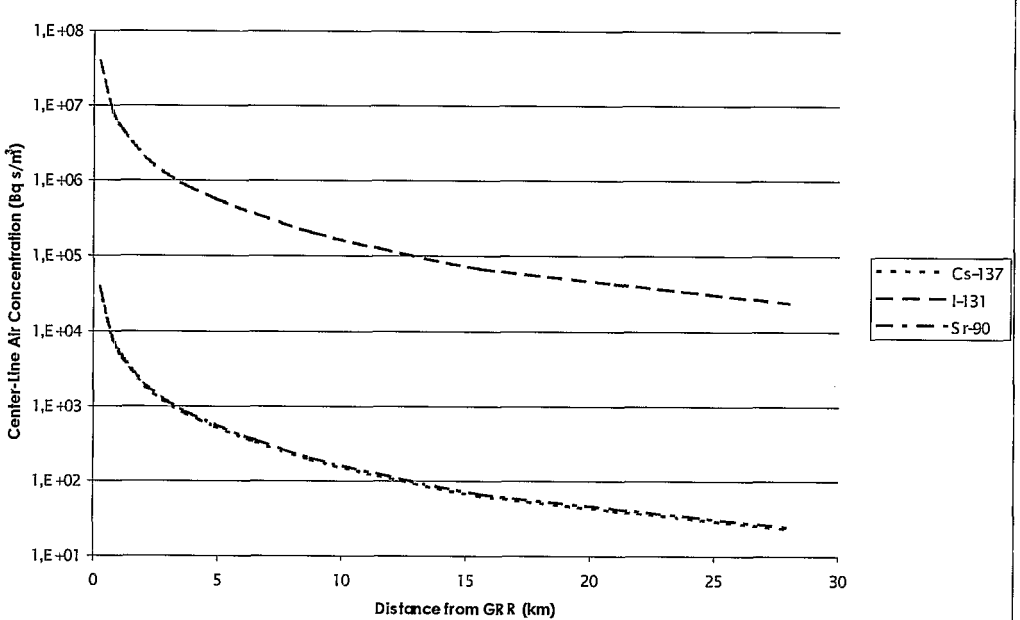


Fig. 3a Cs^{137}, I^{131}, Sr^{90} and Total Ground Concentrations, LOCA (LEU)

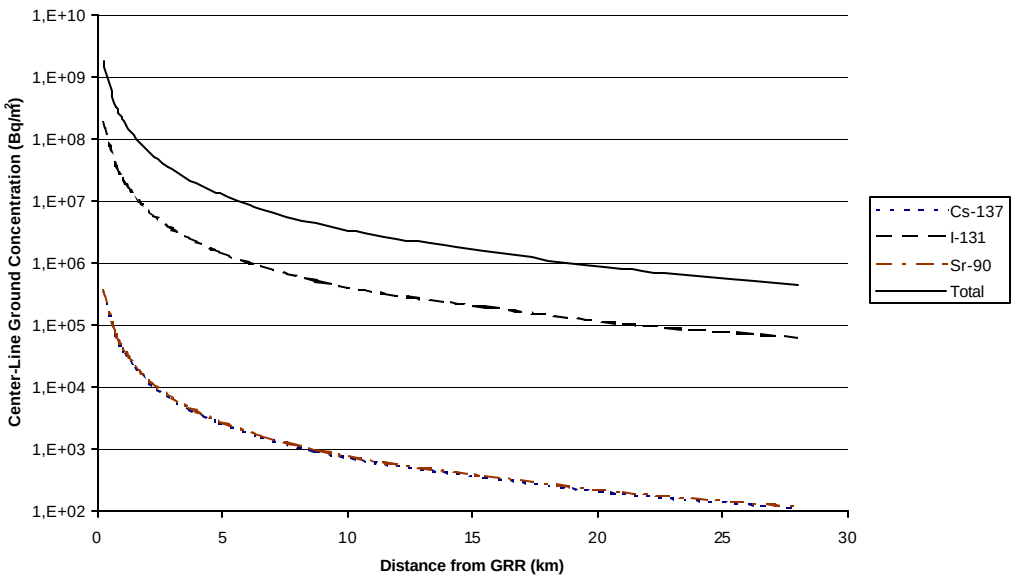


Fig. 3b Cs^{137}, I^{131}, Sr^{90} and Total Ground Concentrations, CFBA (LEU)

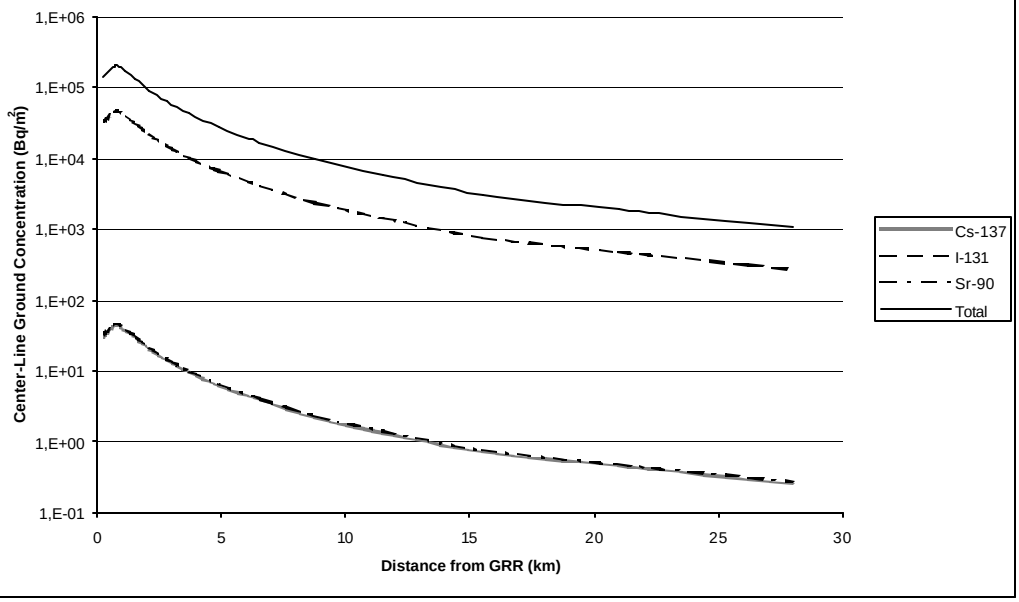


Fig. 4 Whole Body and Thyroid Life-Time Doses (LEU)

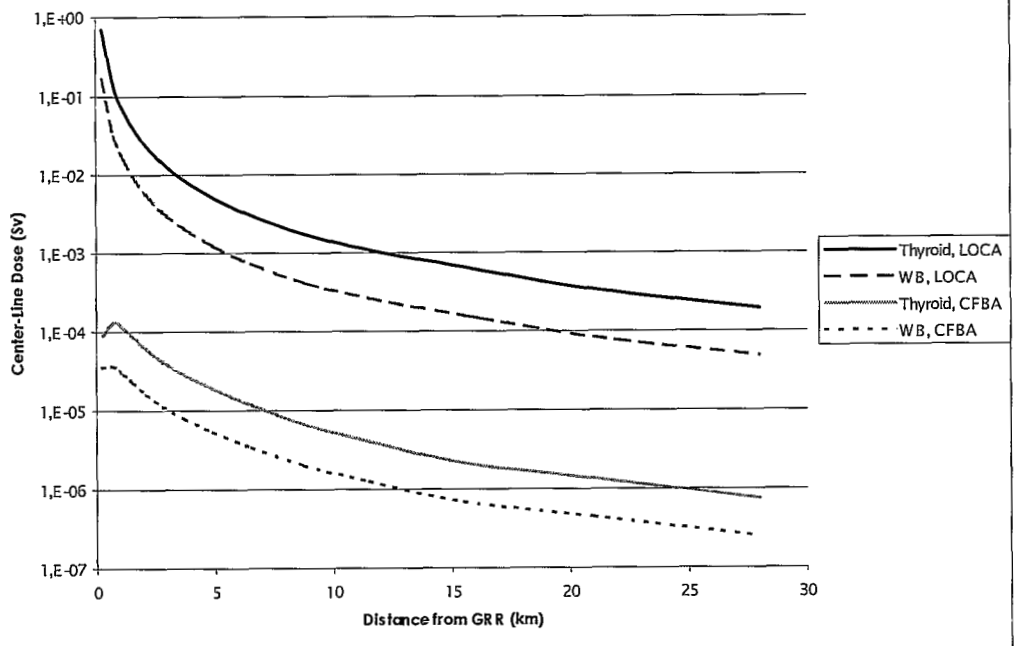


Fig. 5 Center-Line Cancer Fatality Risk (LEU)

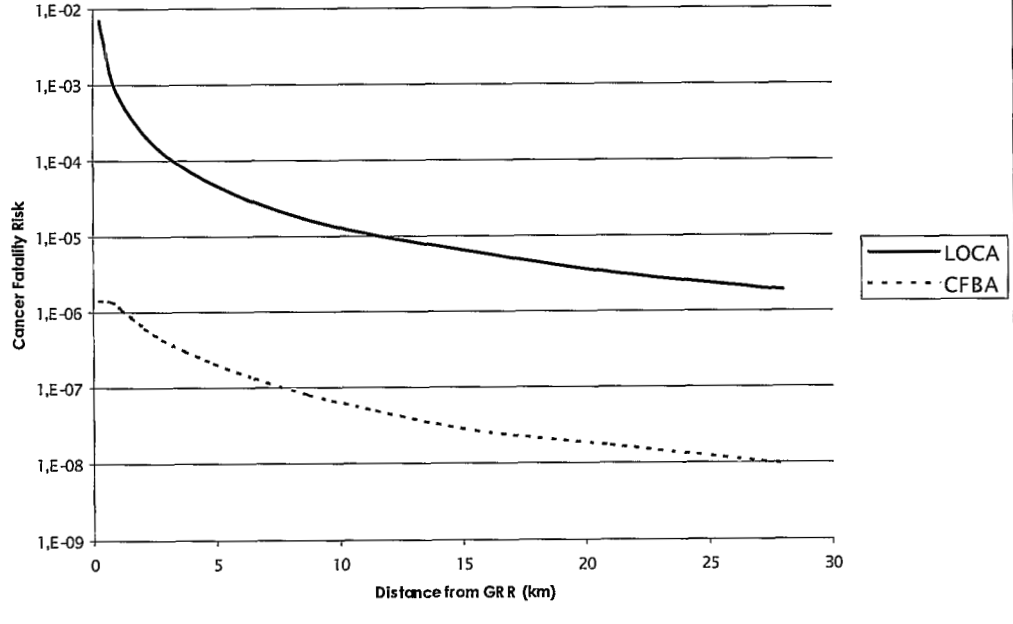


Table 2. GRR Accident Analysis Consequences (LEU Core, Mean Values)

Consequence	Value	
	LOCA	CFBA
Health		
Cancer Fatalities (cases)		
Total	$6.41 \cdot 10^{0}$	$1.95 \cdot 10^{-2}$
Lung	$1.88 \cdot 10^{0}$	$4.93 \cdot 10^{-3}$
Thyroid	$4.65 \cdot 10^{-1}$	$1.45 \cdot 10^{-3}$
Breast	$6.22 \cdot 10^{-1}$	$2.14 \cdot 10^{-3}$
GI tract	$1.63 \cdot 10^{0}$	$5.09 \cdot 10^{-3}$
Leukemia	$5.22 \cdot 10^{-1}$	$1.63 \cdot 10^{-3}$
Bone	$7.35 \cdot 10^{-2}$	$2.20 \cdot 10^{-4}$
Other	$1.21 \cdot 10^{0}$	$3.99 \cdot 10^{-3}$
Cancer Injuries (cases)		
Total	$1.44 \cdot 10^{+1}$	$4.49 \cdot 10^{-2}$
Population Dose (person·Sv)		
Total - Whole Body	$1.66 \cdot 10^{+2}$	$5.22 \cdot 10^{-1}$
Total - Lungs	$2.42 \cdot 10^{+2}$	$6.37 \cdot 10^{-1}$
Total - Red Marrow	$1.08 \cdot 10^{+2}$	$3.36 \cdot 10^{-1}$
Total - Lower LI	$9.71 \cdot 10^{+1}$	$3.03 \cdot 10^{-1}$
Total - Thyroid	$6.46 \cdot 10^{+2}$	$2.02 \cdot 10^{0}$
Total - Bone Surface	$1.63 \cdot 10^{+2}$	$4.90 \cdot 10^{-1}$
Total - Breast	$1.15 \cdot 10^{+2}$	$3.97 \cdot 10^{-1}$
Total - Bladder Wall	$8.80 \cdot 10^{+1}$	$2.89 \cdot 10^{-1}$
Acute - Skin	$6.69 \cdot 10^{+2}$	$1.15 \cdot 10^{0}$
Acute - Stomach	$3.45 \cdot 10^{+1}$	$1.53 \cdot 10^{-1}$
Total Long Term Dose	$8.92 \cdot 10^{+1}$	$2.20 \cdot 10^{-1}$
Ingestion	$5.62 \cdot 10^{0}$	$1.39 \cdot 10^{-2}$
Population Weighted Cancer Fatality Risk (1)		
0-32 km Average	$1.84 \cdot 10^{-6}$	$5.63 \cdot 10^{-9}$
0-1 km Average	$1.04 \cdot 10^{-4}$	$1.18 \cdot 10^{-7}$
1-10 km Average	$2.84 \cdot 10^{-6}$	$9.21 \cdot 10^{-9}$
10-32 km Average	$4.49 \cdot 10^{-7}$	$1.72 \cdot 10^{-9}$

(1) Only direct exposure pathways – Ingestion not included

Table 3. GRR Accident Analysis Consequences (LOCA, Mean Values)

Consequence	Value	
	LEU Core	HEU Core
Health		
Cancer Fatalities (cases)		
Total	$6.41 \cdot 10^{0}$	$6.38 \cdot 10^{0}$
Lung	$1.88 \cdot 10^{0}$	$1.82 \cdot 10^{0}$
Thyroid	$4.65 \cdot 10^{-1}$	$4.74 \cdot 10^{-1}$
Breast	$6.22 \cdot 10^{-1}$	$6.31 \cdot 10^{-1}$
GI tract	$1.63 \cdot 10^{0}$	$1.65 \cdot 10^{0}$
Leukemia	$5.22 \cdot 10^{-1}$	$5.15 \cdot 10^{-1}$
Bone	*$7.35 \cdot 10^{-2}$*	*$5.91 \cdot 10^{-2}$*
Other	$1.21 \cdot 10^{0}$	$1.23 \cdot 10^{0}$
Cancer Injuries (cases)		
Total	$1.44 \cdot 10^{+1}$	$1.45 \cdot 10^{+1}$
Population Dose (person·Sv)		
Total - Whole Body	$1.66 \cdot 10^{+2}$	$1.65 \cdot 10^{+2}$
Total - Lungs	$2.42 \cdot 10^{+2}$	$2.34 \cdot 10^{+2}$
Total - Red Marrow	$1.08 \cdot 10^{+2}$	$1.06 \cdot 10^{+2}$
Total - Lower LI	$9.71 \cdot 10^{+1}$	$9.83 \cdot 10^{+1}$
Total - Thyroid	$6.46 \cdot 10^{+2}$	$6.58 \cdot 10^{+2}$
Total - Bone Surface	*$1.63 \cdot 10^{+2}$*	*$1.31 \cdot 10^{+2}$*
Total - Breast	$1.15 \cdot 10^{+2}$	$1.17 \cdot 10^{+2}$
Total - Bladder Wall	$8.80 \cdot 10^{+1}$	$8.92 \cdot 10^{+1}$
Acute - Skin	$6.69 \cdot 10^{+2}$	$6.79 \cdot 10^{+2}$
Acute - Stomach	$3.45 \cdot 10^{+1}$	$3.50 \cdot 10^{+1}$
Total Long Term Dose	$8.92 \cdot 10^{+1}$	$8.81 \cdot 10^{+1}$
Ingestion	$5.62 \cdot 10^{0}$	$5.72 \cdot 10^{0}$
Population Weighted Cancer Fatality Risk (1)		
0-32 km Average	$1.84 \cdot 10^{-6}$	$1.83 \cdot 10^{-6}$
0-1 km Average	$1.04 \cdot 10^{-4}$	$1.04 \cdot 10^{-4}$
1-10 km Average	$2.84 \cdot 10^{-6}$	$2.84 \cdot 10^{-6}$
10-32 km Average	$4.49 \cdot 10^{-7}$	$4.47 \cdot 10^{-7}$

(1) Only direct exposure pathways – Ingestion not included

4. Conclusions

The results of the analyses indicate that the design basis LOCA would induce consequences of a non-trivial but rather limited magnitude with the current 8 hours/day, 5 days/week operating schedule of GRR. Nevertheless these consequences were considered unacceptable, and appropriate engineered safeguards were introduced into the reactor system(9) in order to avoid the emergence of such an accident. In short a 250 m^3 water storage pool has been constructed uphill of the reactor at an elevation ~30 m higher than the reactor pool surface, and a 5 cm diameter pipeline from this pool penetrates the wall of the reactor building, providing thus through nozzles a spray capability of the reactor core when needed. This storage pool is connected to the main water system of NCSR "Demokritos". In case of a severe leakage of demineralized water from the primary cooling system, water flow can be provided through a flexible plastic pipe connected manually to the pipeline to reduce the leakage rate.

The impacts of the most serious credible GRR accident, the CFBA, and of all other accidents considered in the safety analysis report of GRR, are trivial.

References

1. J.G. Kollas, Individual and Social Risk of a Research Reactor in a Large Population Center, Nucl. Safety, v.25(5), p.633-34, 1984.
2. C. Papastergiou, Editor, Safety Analysis Report of the Greek Research Reactor –1 (GRR-1), Vol. II, 1985.
3. D.I. Chanin, and M.L. Young, Code Manual for MACCS2: Volume 1, User's Guide, SAND97-0594 / UC-610, Sandia National Laboratories, 1997.
4. D.I. Chanin, et. al., MELCOR Accident Consequence Code System (MACCS), NUREG/CR-4691, SAND86-1562, vols. 2-3, Sandia National Laboratories, 1990.
5. Baggentos, M. et. al., Sicherheitsbericht fur 20 MW des Forschungdreakrors "SAPHIR", TM-533-80-3, Wurenlingen, 1980.
6. Bartzis J.G., Temperature Estimations in Air Cooled Fuel Elements of an Open Pool Research Reactors Core after a Loss of Coolant Accident, DEMO Report 83/17, 1983.
7. J. Bartzis, Whole Core Analysis of the Open Research Reactor Under the Most Severe Loss of Coolant Accident Conditions, DEMO Report 83/3, 1984.
8. Anoussis J., and Armyriotis J., Fission Product Release from a Partial or Total Core Melt after a GRR Loss of Coolant Accident (in Greek), Reactor Safety Committee Internal Report, 1983.
9. C. Papastergiou, and C. Mitsonias, Emergency Core Cooling System for the Greek Research Reactor-1 "Demokritos", DEMO Report 84/1G (in Greek), 1984.

FOURTH INTERNATIONAL MACCS USERS GROUP MEETING

Principality of Monaco
September 6, 2002

LIST OF PARTICIPANTS

Name	Affiliation/Address/	Email
Nathan Bixler	Sandia National Laboratories Modeling and Analysis Dept. 6415 MS 0739, P.O. Box 5800 Albuquerque, NM 87185-0739 USA	nbixler@sandia.gov
Toshimitsu Homma	Japan Atomic Energy Research Dept. of Reactor Safety Research Institute Tokai Research Establishment 2-4 Shirakata Shirane, Tokai-mura, Naka-gun 319-1195 Ibaraki-ken JAPAN	homma@popsvr.tokai.jaeri.go.jp
Ioannis Kollas	NCSR "Demokritos" P.O. Box 60228 15310 Aghia Paraskevi Aghia Paraskevi, Attiki 1153 10 GREECE	kollas@ipta.demokritos.gr
Naohiro Kurosawa	Visible Information Center, Inc. 440 Muramatsu Tokai-mura, Ibaraki-ken 319-1112 JAPAN	kurosawa@vic.co.jp
Jocelyn Mitchell	USNRC RES/DRAA/RPER WMB TWFN MS 9E31 Washington, DC 20555-0001 USA	jam@nrc.gov
Vinod Mubayi	Brookhaven National Laboratory Dept. of Energy Sciences & Technology Bldg, 475C, 12 S. Upton P.O. Box 5000 Upton , NY 11973-5000 USA	mubayi@bnl.gov
Steve Ryufuku	Visible Information Center, Inc. 440 Muramatsu Tokai-Mura, Ibaraki-ken 319-1112 JAPAN	ryufuku@vic.co.jp

Name	Affiliation/Address/	Email
Laszlo Sagi	KFKI Atomic Research Institute P.O. Box 49 Budapest H-1525 HUNGARY	sagi@sunserv.kfki.hu
Ulf Tveten	Institute for Energy Technology P.O. Box 40 N-2027 Kjeller NORWAY	ulf.tveten@ife.no